IOWA ELECTRIC LIGHT & POWER, CO.

Duane Arnold Energy Center

INDIVIDUAL PLANT EXAMINATION (IPE)

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PRINCIPAL CONTRIBUTORS

Iowa Electric Light & Power Company

Safety Analysis Group

Bradley G. Hopkins
Terry L. Lanc
William J. Miller
Michael A. Stewart
Richard M. Wachowiak*

Systems Engineering

Randy Best
Curt Bock
Tim Erger
Bhoje Gowda
Paul Hansen
Gary Hawkins
Brian Leimkuhler
Steve Tait
Greg Whittier

ERIN Engineering & Research, Inc.

V. M. Andersen
E. T. Burns
William Dagan
J. A. Hall
Alan Horn
E. A. Hughes
Larry Lee
T. P. Mairs
Kazem Mohammadi
W. Sanford
Kaing Zee

Gabor, Kenton, & Associates

J. R. Gabor
M. B. Murray
D. E. Vanover

* Main principal contributor and project manager.

Table of Contents

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EXTERNAL INDEPENDENT REVIEWERS

James H. Moody, Moody Consulting
G. W. Parry, PhD., Halliburton NUS

IN-HOUSE REVIEW TEAM

Jeff Axline, Technical Support (Electrical)
Russell Becker, Training Supervisor Administrator, Emergency Planning
Dan Berchenbriter, Simulator Training/Safety Analysis
Randy Best, Systems Engineering
Doug Blair, Quality Assurance Internal Audit
Curt Bock, Systems Engineering
Matt Brandt, Plant Performance, Nuclear Engineer
Mark Clark, Plant Performance Engineering
Tim Erger, Systems Engineering
Bhoje Gowda, Systems Engineering
Paul Hansen, Systems Engineering
Gary Hawkins, Systems Engineering
Brad Hopkins, Safety Analysis
Bruce Klotz, Quality Assurance, Technical
Brian Leimkuehler, Systems Engineering
Ron McGee, Technical Support
Dave Mienke, Plant Performance, Reactor Engineer
Bill Miller, Safety Analysis*
Dick Peterson, Systems Engineering
Lenny Sueper, Technical Support
Steve Tait, Systems Engineer
Donald Vest, Simulator Training Instructor
Lee Votroubek, Systems Engineering
Rick Wachowiak, Safety Analysis
Greg Whittier, Systems Engineering

* Currently on assignment to INPO

Executive Summary

The Duane Arnold Energy Center Individual Plant Examination (IPE) report presented is a summary of an extensive and comprehensive study performed to meet the requirements of NRC Generic Letter 88-20. This generic letter requires nuclear utilities to identify and address important contributors to risk, and to implement improvements that the utility believes are appropriate for their plant. The IPE is one of four efforts required for closure of the severe accident issues:

Individual Plant Examination (IPE)

This includes all internal events and internal flooding. External event analysis is the subject of a separate analysis.

Containment Performance Improvements (CPI)

This is the development of generic containment performance improvements with respect to severe accidents. This effort has been concluded with the request to install 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 primary external events to be investigated are seismic and internal fires. External floods, high winds and tornadoes, and transportation accidents are also evaluated.

Accident Management (AM)

This involves the development of a program to use the IPE and IPEEE to enhance the accident management capabilities at DAEC. This program is still under development.

The DAEC IPE is a 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 presents the containment performance response to severe accidents.

The Level 1 analysis resulted in a total core damage frequency (CDF) of 7.84E-6 per reactor-year. Internal flooding is estimated to be a negligible contributor to the CDF. Figures 1.4-1, 1.4-2, and 1.4-3 show the core damage contribution by initiators and initiator type. The results of the Level 2 analysis are summarized by Figure 1.4-4, which shows the containment performance by magnitude and timing of release.

It is concluded that DAEC is a plant with a low risk of core damage and fission product release. It has only one sequence that meets the 1E-6 screening criteria and even this sequence is just at 1E-6. There are, therefore, no sequences or phenomena that are identified in this study that would make DAEC an outlier plant. As a result, no further changes would appear to be necessary.

List of Tables

LIST OF FIGURES

ACRONYMS AND ABBREVIATIONS

AC	Alternating Current
ADS	Automatic Depressurization System
AOP	Abnormal Operating Procedure
AOV	Air Operated Valve
APRM	Average-Power-Range Monitor
ARI	Alternate Rod Insertion
ASEP	Accident Sequence Evaluation Program
ATTS	Analog Transmitter Trip System
ATWS	Anticipated Transient Without Scram
BCB	Battery Control Board
BCC	Breaker Control Circuit
BHEP	Basic Human Event Probability
BIIT	Boron Injection Initiation Temperature
BOP	Balance of Plant
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owner's Group
CAFTA	Computer Assisted Fault Tree Analysis
CCF	Common Cause Failure
CDF	Core Damage Frequency
CDWR	Catastrophic Drywell Rupture
CET	Containment Event Tree
CMAR	Corrective Maintenance Action Request
CND	Condensate
CRD	Control Rod Drive
CRDH	Control Rod Drive Hydraulic
CS	Core Spray
CSC	Containment Spray Cooling
CSF	Critical Safety Function
CST	Condensate Storage Tank
CV	Control Valve
CWS	Circulating Water System
CWWR	Catastrophic Wetwell Rupture
DAEC	Duane Arnold Energy Center Nuclear Power Generating Station

DBD	Design Basis Document
DC	Direct Current
DG	Diesel Generator
DHR	Decay Heat Removal
DR	Deviation Report
DSR	Daily Status Report
DWHL	Drywell Head Leakage
DWHR	Drywell Head Rupture
DWL	Drywell Leakage
DWR	Drywell Rupture
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFT	Emergency Filtration Train
EHC	Electro-Hydraulic Control
EOP	Emergency Operating Procedure
EPIC	Emergency Plant Information Computer
EPRI	Electric Power Research Institute
ESW	Emergency Service Water
FMEA	Failure Modes and Effects Analysis
FSAR	Final Safety Analysis Report
FW	Feedwater
FWS	Feedwater System
GSI	Generic Safety Issue
GSW	General Service Water System
HCTL	Heat Capacity Temperature Limit
HCU	Hydraulic Control Unit
HELB	High Energy Line Break
HEP	Human Error Probability
HPCI	High Pressure Coolant Injection (System)
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation, and Air Conditioning
IAS	Instrument Air System
IDCOR	Industry Degraded Core Rulemaking Program
IE	Iowa Electric Light and Power Co. (also Iowa Electric)
IEEE	Institute of Electrical and Electronic Engineers

ILRT	Integrated Leak Rate Test
INPO	Institute of Nuclear Power Operations
IORV	Inadvertent Opening of Safety Relief Valve
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination for External Events
ISI	In-Service Inspection
ISLOCA	Interfacing System LOCA
IST	In-Service Testing
LCO	Limiting Condition of Operation
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection (a mode of RHR)
LPCS	Low Pressure Core Spray
MAAP	Modular Accident Analysis Program
MAR	Maintenance Action Request
MCB	Main Control Board
MCC	Motor Control Center
MG	Motor Generator
MGL	Multiple Greek Letter
MOSR	Monthly Operating Status Report
MOV	Motor Operated Valve
MS	Main Steam Supply System, Main Steam
MSIV	Main Steam Isolation Valves
MWO	Maintenance Work Order
NCR	Non-Conformance Report
NHEP	Nominal Human Error Probability
NPRDS	Nuclear Power Reliability Data System
NPSH	Net Positive Suction Head
NRC	U.S. Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OP	(System) Operating Procedure
OR	Occurrence Report
OSP	Offsite Power
P&ID	Piping and Instrumentation Drawing

PCIS	Primary Containment Isolation System
PCP	Primary Containment and Purge
PCS	Power Conversion System
PCV	Pressure Control Valve
PDS	Plant Damage State
PMAR	Preventive Maintenance Action Request
PRA	Probabilistic Risk Assessment
PSAR	Preliminary Safety Analysis Report
QA	Quality Assurance
RBCCW	Reactor Building Closed Cooling Water System
RBCLCS	Reactor Building Closed Loop Cooling System
RCIC	Reactor Core Isolation Cooling (System)
RF	Recovery Factor
RHR	Residual Heat Removal
RHRSW	Residual Heat Removal Service Water
RMIEP	Risk Methods Integration Evaluation Program
RO	Reactor Operator
RPS	Reactor Protection System
RPT	Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Clean-up
RWS	River Water Supply System
SBO	Station Blackout
SCET	Streamlined Containment Event Tree
SDC	Shutdown Cooling
SDIV	Scram Discharge Instrument Volume
SDV	Scram Discharge Volume
SHARP	Systematic Human Action Reliability Procedure
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control
SNB	System Notebook
SORV	Stuck Open Relief Valve
SPC	Suppression Pool Cooling
SPDS	Safety Parameter Display System

SPS	Suppression Pool Spray
SRI	Safety Review Item
SRO	Senior Reactor Operator
SRV	Safety Relief Valve
SS	Shift Supervisor
ST	Surveillance Test
STD	(System) State/Transition Diagram
SV	Solenoid Valve
SW	Service Water
SWS	(Normal) Service Water System
TAF	Top of Active Fuel
TBCLCS	Turbine Building Closed Loop Cooling System
TBV	Turbine Bypass Valve System
UFSAR	Updated Final Safety Analysis Report
USI	Unresolved Safety Issue
VSS	Vapor Suppression System
WREDF	Work Request Event Deficiency Form
WTF	Work Tracking Form

1. SUMMARY

1.1 BACKGROUND AND OBJECTIVES

In 1988, the NRC issued Generic Letter 88-20 (GL 88-20) requiring each utility to perform an Individual Plant Examination (IPE) for severe accident vulnerabilities.

In order to satisfy the requirements of GL 88-20, Iowa Electric Light and Power Company (IEL&P) elected to perform an IPE for the Duane Arnold Energy Center (DAEC) by utilizing a Probabilistic Risk Assessment (PRA) approach. The DAEC IPE effort consists of both a Level 1 PRA and a Level 2 PRA. The DAEC Level 1 PRA is an integrated analysis of plant and system responses to a wide spectrum of internal events such as reactor scrams, loss of off-site power, loss-of-coolant accidents, and other special initiators. The Level 2 PRA considers core damage timing and subsequent containment challenges to quantitatively assess the potential for significant release of radioactivity. In summary, emphasis of the DAEC IPE is on the quantification of plant core damage frequency and the evaluation of containment performance.

The events analyzed in the IPE are, in many cases, beyond the original design basis of the plant, are extremely unlikely, and are not expected to occur within the life of the plant. However, the performance of such an analysis provides insight into system and plant capability and provides a tool for the quantitative evaluation of potential plant improvements. As stated in greater detail below, that is the fundamental goal of the DAEC IPE effort.

GL 88-20 stated several objectives the NRC expected to be accomplished by the performance of an IPE. The DAEC IPE has been completed to meet the following NRC objectives:

- To develop an appreciation for severe accident behavior at DAEC.

- To understand the most likely severe accident sequences that could occur at DAEC.

- To gain more quantitative understanding of the overall probabilities of core damage and fission product releases.

- If necessary, to reduce the overall probabilities of core damage and fission product releases at DAEC by modifying, where appropriate, hardware, procedures, or training that would help prevent or mitigate severe accidents.

The process for performance of the IPE was also designed to meet Iowa Electric objectives. Paramount among these objectives was that the system knowledge gained by performing fault tree analysis be retained by in-house engineering personnel. For this reason, ownership and responsibility for system fault trees was established with the Iowa Electric System Engineers. Secondly, aspects of the IPE that were supported by external consultants were performed in a manner to maximize technology transfer to Iowa Electric staff.

1.2 PLANT FAMILIARIZATION

This IPE model represents the plant (hardware, procedures, etc.) as of startup from the 1992 refueling outage (April 1992), with the exception of inclusion of the hard pipe vent from the suppression pool which is included in the models. Installation of the vent is scheduled for completion later in 1992.

The DAEC site consists of approximately 500 acres located in Linn County in eastern Iowa adjacent to the Cedar River. The distance from the reactor site to the nearest site boundary on land is approximately 2000 ft. Major structures on the site include the reactor building, turbine building, control building, radwaste building, administration building, machine shop, security building, technical support center, data acquisition center, air compressor building, offgas retention building, intake structure, pump house, cooling towers, training center, switchyard, low-level radwaste processing and storage facility, and offgas stack.

The nuclear steam supply system (NSSS) is a General Electric (GE) boiling water reactor (BWR) type with a design thermal output rating of 1658 MWt. The DAEC generation limit is 589 MW gross at a power factor of 0.9. Heat balance and safety analyses were performed at 102% of rated power, which corresponds to 1691 MWt.

The NSSS is similar to other GE nuclear plants of this vintage. It uses movable, bottom-entry control rods for reactivity control, and generates dry steam in the reactor pressure vessel through a moisture separator and steam dryer. The NSSS is located within the reinforced concrete reactor building.

The Duane Arnold Energy Center received its operating license on February 22, 1974 and began commercial operation on February 1, 1975.

1.2.1 General Plant Characteristics

Figure 1.2-1 is a simplified schematic of the DAEC nuclear plant. Feedwater is injected into the reactor pressure vessel (RPV) where its flow is accelerated by the Reactor Recirculation System prior to passing through the core. The core is cooled by this flow, and the coolant boils off as a steam-water mixture. This mixture is converted to high

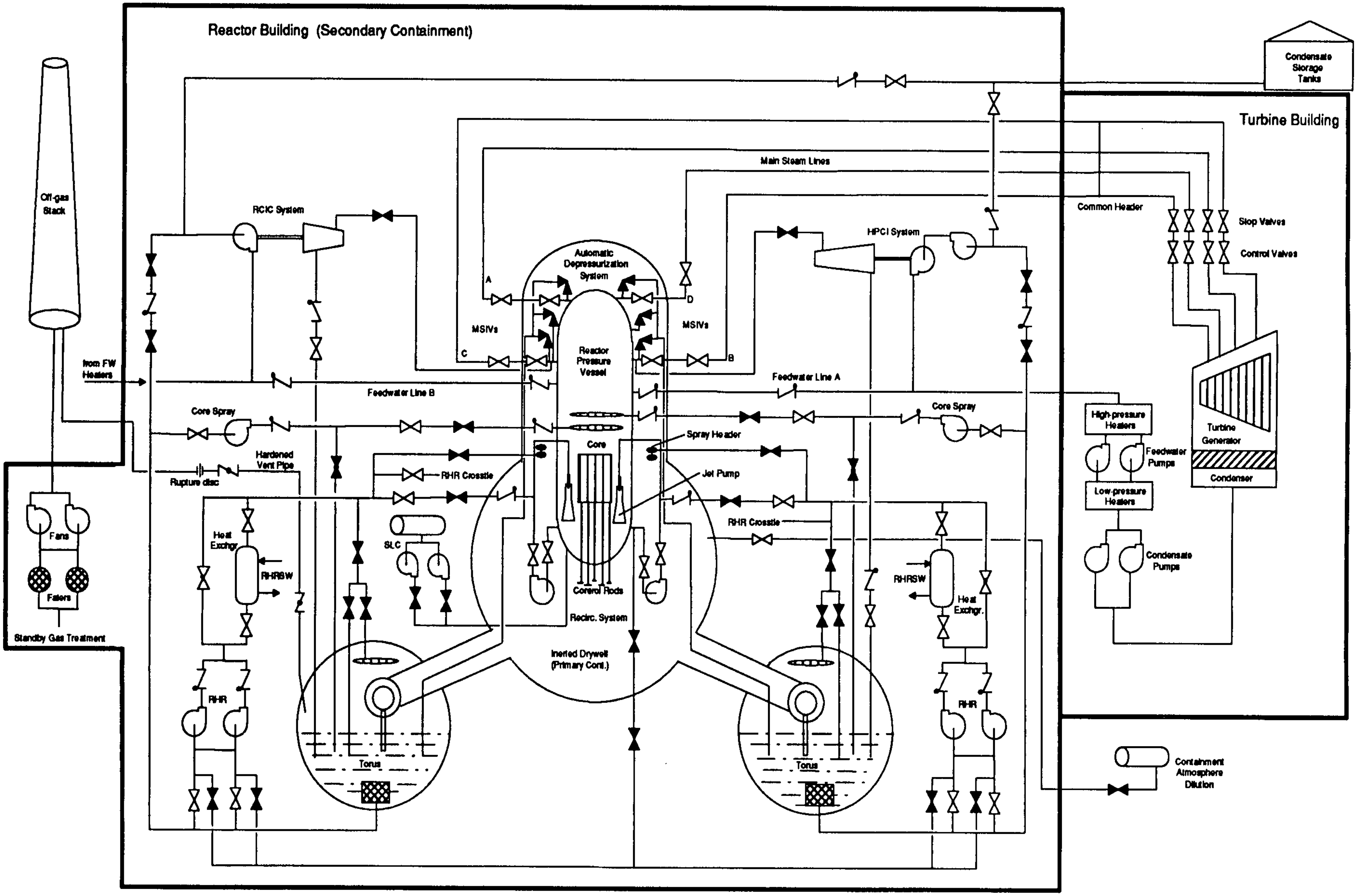


Figure 1.2-1. Simplified Schematic of the DAEC Nuclear Power Plant

quality steam by the moisture separator and dryer assemblies prior to exiting the RPV. The steam exits the primary containment via four steam lines to the turbine and main condenser.

The reactor core is composed of slightly enriched uranium dioxide pellets contained in sealed Zircaloy-2 tubes. These fuel rods are organized side-by-side within a Zircaloy-4 channel. The fuel assemblies are supported axially by the fuel support piece located on the core plate, and laterally by the fuel support piece and the upper tie plate. The fuel assemblies are further organized into fuel cells consisting of four assemblies surrounding a single control rod. The control rods consist of a sheathed cruciform array of stainless steel tubes filled with boron carbide (B_4C) powder. New fuel is introduced during refueling outages to replace some of the previously burned fuel in accordance with fuel management schemes.

The reactor pressure vessel and the RPV internals contain and support the fuel and control rods. The RPV is cylindrical with hemispherical heads. It is fabricated of carbon steel and clad internally with stainless steel except for the top head. The moisture separator and steam dryer assemblies produce 99.9% quality steam.

The Reactor Recirculation System pumps reactor coolant through the core to remove the energy generated in the fuel. This is accomplished by two recirculation loops external to the reactor vessel. Each loop has one motor-driven recirculation pump. Recirculation pump speed can be varied to allow some control of the reactor power level through the effects of coolant flow rate on moderator void content.

Auxiliary systems are provided to control RPV level and pressure, purify reactor coolant water, cool system components, remove residual heat when the reactor is shutdown, cool the spent fuel storage pool, sample reactor coolant water, provide for emergency high and low pressure core cooling, and vent and drain the RPV.

The safety features designed into DAEC have sufficient redundancy of component and power sources that under conditions of the design basis loss-of-coolant accident, the systems can maintain primary containment integrity and limit public exposure to below the limits set forth in 10CFR100. These safety features are:

(1) The Primary Containment System which uses a pressure suppression containment to house the reactor vessel, the reactor coolant recirculation loops, and other branch connections of the reactor primary system. The pressure suppression system consists of a drywell and a pressure suppression chamber that are interconnected through a series of vent pipes, isolation valves, cooling systems, and other equipment.

(2) The primary containment and reactor vessel isolation control system which automatically initiates the closure of isolation valves to close off all potential leakage paths for radioactive material to the environment. This action is taken on the indication of a potential breach in the nuclear system process barrier.

(3) The secondary containment consisting of the reactor building which completely surrounds the primary containment. The building will provide secondary containment when the primary containment is closed and in service and primary containment during periods when the primary containment is open, such as during refueling periods. The reactor building houses refueling and reactor servicing

equipment, new and spent fuel storage facilities, and other reactor safety and auxiliary systems.

(4) Eight main steam line isolation valves, two of which are located in series on each of the four main steam lines limit the loss of reactor coolant from the RPV stemming from either a major leak from the steam piping outside the primary containment, or a malfunction of the pressure control system resulting in excessive steam flow from the reactor vessel. They also limit the release of radioactive materials by closing the nuclear system process barrier in case of a gross failure of fuel cladding integrity. Additionally, they limit the release of radioactive materials by closing the primary containment barrier in case of major leak from the nuclear system inside the primary containment.

(5) A venturi-type main steam line flow restrictor is installed in each steam line close to the reactor vessel. These devices limit the loss of water from the reactor vessel before the main steam line isolation valves are closed in case of a main steam line break outside the primary containment and prevent uncovering of the core.

(6) The High Pressure Coolant Injection (HPCI) system provides and maintains an adequate coolant inventory inside the reactor vessel to prevent excessive fuel clad temperatures as a result of small and intermediate breaks in the nuclear system process barrier. A high pressure system is needed for small breaks because the reactor vessel depressurizes slowly, preventing low pressure systems from injecting coolant.

(7) The Automatic Depressurization System (ADS) acts to rapidly reduce reactor vessel pressure in a LOCA situation in which the HPCI system fails to provide adequate cooling water flow. The depressurization provided by the system enables the low pressure emergency core cooling systems to deliver cooling water to the reactor vessel.

(8) A portion of the Residual Heat Removal (RHR) System is provided to spray water into the primary containment as an augmented means of removing energy from the containment following a LOCA.

(9) The Standby Liquid Control (SLC) System provides a redundant, independent, and functionally diverse method of bringing the nuclear fission reaction to subcriticality and maintaining subcriticality as the reactor cools, in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner.

(10) A control rod velocity limiter is attached to each control rod to limit the velocity at which a control rod can fall out of the core should it become detached from its control rod drive.

(11) Control rod drive housing supports are located underneath the reactor vessel near the control rod housings. The supports limit the travel of a control rod in the event that a control rod housing is ruptured.

(12) The Standby Gas Treatment System (SGTS) consists of two identical processing streams, either of which is capable of exchanging the reactor building volume once in a 24-hour period. SGTS maintains a slight negative internal building pressure and will process all gaseous effluents before discharge from the elevated release point.

(13) Two automatic starting, full capacity diesel generators provide standby ac power. Each diesel generator is capable of supplying the power required to shut down and maintain the plant in a safe condition in the event of total loss of normal power sources.

(14) The plant dc power supply system consists of two 125V batteries and one 250V battery, each with its own battery charger, circuit breakers, and buses. In addition, two independent 24V buses are provided, each supplied by a center grounded 48V battery and a charger.

(15) The RHR Service Water and Emergency Service Water Systems provide cooling to various safety-related loads. The RHR service water pumps in each loop are connected electrically to the same bus as the diesel generator in their emergency loop to ensure the emergency equipment service water supply in the event the offsite ac power supply is lost.

(16) The main steam line radiation monitoring system consists of four gamma radiation monitors located external to the main steam lines

just outside of the primary containment, and are designed to detect a gross release of fission products from the fuel.

(17) The Reactor Protection System (RPS) consists of the various monitors and detectors used as inputs to circuitry designed to initiate a rapid, automatic shutdown (scram) of the reactor to prevent fuel cladding damage and reactor cooling system pressure boundary damage due to abnormal operational transients. It uses a one out of two taken twice logic. The RPS is a very reliable system that uses various redundant means to initiate control rod insertion, such as backup scram pilot valves and a separate Alternate Rod Insertion (ARI) mechanism, as well as providing the means for operator manual scram.

The main generator is an 1800-rpm, three phase, 60 cycle generator which produces AC power at 22 kV which is stepped up to 161 kV by the main transformer bank for delivery to the high voltage offsite transmission lines. There is a stepdown auxiliary power transformer to provide power to all of the station auxiliaries during normal operating conditions.

The safety-related loads are normally supplied by offsite power via the startup transformer. During maintenance of the startup transformer, safety-related loads are supplied from the 161 kV or 345 kV transmission system through the standby transformer.

The turbine is an 1800 rpm condensing turbine consisting of a single-flow high pressure shell and two double-flow low pressure shells. A turbine bypass system is provided that passes steam directly to the main condenser when the reactor steaming rate exceeds the load demand of the turbine generator.

The reactor is refueled with equipment designed to handle spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site. Underwater transfer of spent fuel provides an optically transparent radiation shield, as well as a reliable source of coolant for removal of decay heat. This system also provides capability for receiving, handling and storing of new fuel.

The radioactive waste treatment systems provide all equipment necessary to collect, process, monitor, and discharge radioactive liquid, gaseous and solid wastes that are produced during reactor operation.

The Liquid Radwaste System collects, treats, stores, and disposes of all radioactive liquid wastes. These wastes are collected in sumps and drain tanks at various locations throughout the plant and then transferred to the appropriate collection tanks in the radwaste building for treatment, storage, and disposal. Processed liquid wastes are normally returned to the Condensate System.

The Gaseous Radwaste System collects gaseous discharges from the main condenser air ejectors and gland seal condenser. In addition, this system processes and delivers the gases to the main stack for elevated release to the atmosphere. The gaseous radwaste system is continuously monitored by the main stack radiation monitor and the air ejector offgas radiation monitor.

1.2.2 Description of Key Plant Features and Systems

This section of the report is provided to outline the systems of the plant which present an unusual or unique design when compared to other General Electric plants of similar design and vintage. Only those systems having outstanding characteristics shall be included. For these systems, a simplified drawing which details the components as modeled in the fault trees is presented. Table 1.2-1 presents a summary of a comparison between DAEC Nuclear Plant and other NUREG-1150 plants. This table includes several systems which shows the similarities as well as the unique features of the DAEC plant.

1.3 OVERALL METHODOLOGY

The DAEC IPE consists of the following four (4) elements to meet GL 88-20:

1) Front-End Analysis

2) Back-End Analysis

3) Consideration of Safety Features and Plant Improvements

4) IPE Utility Team and Internal Review

The methodology of each of these elements is briefly introduced below.

Table 1.2-1
PLANT COMPARISON TABLE - DAEC VS. NUREG-1150 PLANTS

System/ Feature	DAEC	Peach Bottom	Grand Gulf
Coolant Injection Systems	· High-pressure coolant injection system provides coolant to the reactor vessel during accidents in which system pressure remains high, with one train and one turbine-driven pump. · Reactor core isolation cooling system provides coolant to the reactor vessel during accidents in which system pressure remains high, with one train and one turbine-driven pump. · Low-pressure core spray system provides coolant to the reactor vessel during accidents in which vessel pressure is low, with two trains each with one motor-driven pump. · Low-pressure coolant injection system provides coolant to the reactor vessel during accidents in which vessel pressure is low, with two trains and four motor-driven pumps. · RHR service water system crosstie provides coolant makeup source to the reactor vessel during accidents in which normal sources of emergency injection have failed (low RPV pressure), with two trains and four pumps for crosstie. · In addition to RHRSW, GSW, ESW, well water and the fire water system can be used as a last resort source of low-pressure coolant injection to the reactor vessel. · Control rod drive system provides backup source of high-pressure injection, with two pumps. · Automatic depressurization system for depressurizing the reactor vessel to a pressure at which the low-pressure injection systems can inject coolant to the reactor vessel: 4 ADS relief valves/capacity 880,000 lb/hr. In addition, there are four non-ADS relief valves.	· High-pressure coolant injection system provides coolant to the reactor vessel during accidents in which system pressure remains high, with one train and one turbine-driven pump. · Reactor core isolation cooling system provides coolant to the reactor vessel during accidents in which system pressure remains high, with one train and one turbine-driven pump. · Low-pressure core spray system provides coolant to the reactor vessel during accidents in which vessel pressure is low, with two trains and four motor-driven pumps. · Low-pressure coolant injection system provides coolant to the reactor vessel during accidents in which vessel pressure is low, with two trains and four motor-driven pumps. · High-pressure service water crosstie system provides coolant makeup source to the reactor vessel during accidents in which normal sources of emergency injection have failed (low RPV pressure), with one train and four pumps for crosstie. · Control rod drive system provides backup source of high-pressure injection, with two pumps/210 gpm (total)/1,100 psia. · Automatic depressurization system for depressurizing the reactor vessel to a pressure at which the low-pressure injection systems can inject coolant to the reactor vessel: 5 ADS relief valves/capacity 820,000 lb/hr. In addition, there are six non-ADS relief valves.	· High-pressure core spray (HPCS) system provides coolant to reactor vessel during accidents in which system pressure remains high or low, with one train and one motor-driven pump. · Reactor core isolation cooling system provides coolant to the reactor vessel during accidents in which system pressure remains high, with one train and one turbine-driven pump. · Low-pressure core spray system provides coolant to the reactor vessel during accidents in which vessel pressure is low, with one train and one motor-driven pump. · Low-pressure coolant injection system provides coolant to the reactor vessel during accidents in which vessel pressure is low, with three trains and three pumps. · Standby service water crosstie system provides coolant makeup source to the reactor vessel during accidents in which normal sources of emergency injection have failed, with one train and one pump (for crosstie). · Firewater system is used as a last resort source of low-pressure coolant injection to the reactor vessel, with three trains, one motor-driven pump and two diesel-driven pumps. · Control rod drive system provides backup source of high-pressure injection, with two pumps/238 gpm (total)/1103 psia. · Automatic depressurization system (ADS) depressurizes the reactor vessel to a pressure at which the low-pressure injection systems can inject coolant to the reactor vessel, with eight relief valves/ capacity of 900,000 lb/hr. In addition, there are 12 non-ADS relief valves. · Condensate system used as a backup injection sourco.

System/ Feature	DAEC	Peach Bottom	Grand Gulf
Heat Removal Systems	· Residual heat removal/suppression pool cooling system to remove heat from the suppression pool during accidents, with two trains and four pumps and 2 heat exchangers. · Residual heat removal/shutdown cooling system to remove decay heat during accidents in which reactor vessel integrity is maintained and reactor at low pressure, with two trains and four pumps. · Residual heat removal/containment spray system to suppress pressure and remove decay heat in the containment during accidents, with two trains and fcur pumps.	· Residual heat removal/suppression pool cooling system to remove heat from the suppression pool during accidents, with two trains and four pumps and 4 heat exchangers. · Residual heat removal/shutdown cooling system to remove decay heat during accidents in which reactor vessel integrity is maintained and reactor at low pressure, with two trains and four pumps. · Residual heat removal/containment spray system to suppress pressure and remove decay heat in the ccntainment during accidents, with two trains and four pumps.	· Residual heat removal/suppression pool cooling system removes decay heat from the suppression pool during accidents, with two trains and two pumps. · Residual heat removal/shutdown cooling system removes decay heat during accidents in which reactor vessel integrity is maintained and reactor is at low pressure, with two trains and two pumps. · Residual heat removal/containment spray system suppresses pressure in the oontainment during accidents, with two trains and two pumps.
Reactivity Control Systems	· Control rods. · Standby liquid oontrol system, with two parallel positive displacement pumps rated at 28 gpm por pump. . Both supply equivalent 86 gpm due to concentrated beron.	· Control rods. · Standby liquid control system, with two parallel positive displacement pumps rated at 43 gpm por pump, but each with 86 gpm equivalent because of the use of enriched boron.	· Control rods. · Standby liquid control system, with two parallel positive displacement pumps rated at 43 gpm por pump.
Key Support Systems	· DC power with up to 12-hour station batteries under certain circumstances. · Emergency AC power from two diesel generators. · Emergency service water provides cooling water to various safety systems and components.	· DC power with up to approximately 12-hour station batteries. · Emergency AC power from four diesel generators shared botween two units. · Emergency servico water provides cooling water to safety systems and components shared by two units.	· DC power with 12-hour station batteries. · Emergency AC power, with two diesel generators and third diesel generator dedicated to HPCS but with crossties. · Suppression pool makeup system provides water from the upper containment pool to the suppression pool following a LOCA. · Standby service water provides cooling water to safety systems and components.
Containment Systems	· Containment venting used when suppression pool cooling and containment sprays have failed to reduce primary containment pressure. A hard-piped vent is currently being installed.	· Containment venting used when suppression pool cooling and containment spray shave failed to reduce primary containment pressure. A hard-piped vent is already installed.	· Containment venting is used when suppression pool cooling and containment sprays have failed to reduce primary ccntainment pressure. · Hydrogen igniter system prevents the buildup of large quantities of hydrogen inside the containment during accident conditions.
Containment Structure	· BWR Mark I. · 0.31 million cubic feet. · 56 psig design pressure.	· BWR Mark I. · 0.32 million cubic feet. · 56 psig design pressure.	· BWR Mark III. · 1.67 million cubic feet. · 15 psig design pressure.

1.3.1 Front-End Analysis - Level 1

The DAEC IPE uses a small event tree-large fault tree approach to quantify the core damage frequency (CDF) from a representative set of initiating event groups. The DAEC IPE initiating event groups were developed from a review of the DAEC FSAR, EPRI reports, NUREG reports of other PRAs (including NUREG-1150), and plant-specific experience. The initiating event groups considered are:

- Turbine Trip with Bypass
- Loss of Feedwater
- MSIV Closure
- Loss of Condenser Vacuum
- Loss of River Water Supply
- Inadvertent Open Relief Valve
- Loss of Instrument Air
- Loss of Division II DC
- Large Break LOCA
- Medium Break LOCA
- Small Break LOCA
- Loss of Offsite Power
- Anticipated Transient Without SCRAM

The quantification of these initiating event frequencies are based on DAEC specific data where sufficient operating experience data has been accumulated. The safety challenges to DAEC following each of these initiating events requires that plant systems respond in a pre-determined manner. The responses are necessary to ensure five (5) critical safety functions:

- Subcriticality,

- Core cooling,
- Heat sink,
- Containment, and
- Inventory.

These critical safety functions are directed at the maintenance of the three (3) primary barriers of radioactivity release:

1) Fuel cladding,
2) RCS boundary, and
3) Containment.

A review of DAEC design concluded that 18 systems should be analyzed in fault trees to model the plant response to the initiators. The 18 systems are:

1) Automatic Depressurization and Safety Relief System
2) Condensate and Main Condenser
3) Control Rod Drive
4) Low Pressure Core Spray
5) Emergency Service Water
6) Feedwater
7) General Service Water
8) High Pressure Coolant Injection
9) Instrumentation
10) AC/DC Electric Power
11) Reactor Core Isolation Cooling
12) Residual Heat Removal
13) Residual Heat Removal Service Water

14) Standby Liquid Control
15) Torus Vent/Vapor Suppression
16) Well Water
17) River Water
18) Recirculation Pump Trip

Each of these systems was modeled in a logical, linked fault tree such that the front-line systems were dependent on the appropriate support systems. The component failure data was generic data taken from other PRAs or data sources and supplemented with plant specific data collected for the DAEC IPE project. The system fault trees included basic events for:

- failure to start, run, operate, etc.
- system maintenance unavailability
- pre-initiator mis-alignment conditions
- pre-initiator mis-calibration
- post-initiator operator action and recovery actions

The last three categories of basic events are human reliability related considerations.

Human Reliability Analysis (HRA) is the method used to describe qualitatively and quantitatively the occurrence of human errors in nuclear power plant operation that affect system and plant availability and reliability. Because human interactions in complex human-machine systems have caused and propagated accidents, quantifying the probabilistic nature of human error is factored into the assessment of effects of human performance during accident situations in the Duane Arnold IPE. Human Reliability Analysis is the overall methodology used to predict Human Error Probabilities (HEPs) of events that can lead to or be the cause of nuclear power plant accidents. Section 3.3.3

of this report identifies specific methodological techniques used in the quantitative evaluation of the HEPs.

The common cause data for the front-end analysis uses data from NUREG/CR-4550 (Volume 1), NUREG/CR-4780, EPRI NP-3967, and the EPRI Common Cause Database. From these sources (and others where noted), appropriate beta factors were chosen and, consistent with NUREG/CR-4550, calculated using the multiple greek letter (MGL) method. Based on previous common cause effects for the following components:

- Air-operated valves
- Batteries
- Breakers
- Check valves
- Diesel generators
- Motor-operated valves
- Motor-driven pumps
- Relief valves
- Solenoid valves
- Fans/containment air coolers;

the Front-End transient event trees were analyzed and the resulting core damage sequences placed into accident sequence classifications. These classifications are shown in Table 1.3-1.

The accident classifications above enable transition of the results into the containment event trees (CETs) utilized in the Back-End analyses.

Table 1.3-1

ACCIDENT SEQUENCE CLASSIFICATIONS

Accident Class Designator	Subclass	Definition	WASH-1400 Designator Example
Class I	A	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high.	TQUX
	B	Accident sequences involving a station blackout and loss of coolant inventory makeup.	T_EQUV
	C	Accident sequences involving a loss of coolant inventory induced by an ATWS sequence with containment intact.	T_TC_MQU
	D	Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced to 200 psi.; i.e., accident sequences initiated by common mode failures disabling multiple systems (ECCS) leading to loss of coolant inventory makeup.	TQUV
	E	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high and DC power is unavailable.	---
Class II	A	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage induced post containment failure	TW
	L	Accident sequences involving a loss of containment heat removal with the RPV breached but no initial core damage; core damage after containment failure.	AW
	T	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage induced post high containment pressure	N/A
	V	Class IIA or IIL except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	TW

Table 1.3-1

ACCIDENT SEQUENCE CLASSIFICATIONS

Accident Class Designator	Subclass	Definition	WASH-1400 Designator Example
Class III (LOCA)	A	Accident sequences leading to core damage conditions initiated by vessel rupture where the containment integrity is not breached in the initial time phase of the accident.	R
	B	Accident sequences initiated or resulting in small or medium LOCAs for which the reactor cannot be depressurized prior to core damage occurring.	S_1QUX
	C	Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is a low pressure and no effective injection is available.	AV
	D	Accident sequences which are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate, challenging the containment integrity with subsequent failure of makeup systems.	AD
Class IV (ATWS)	A	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post containment failure.	$T_TC_MC_2$
	L	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially breached (e.g., LOCA or SORV); core damage induced post containment failure.	N/A
	T	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post high containment pressure.	N/A
	V	Class IV A or L except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	N/A
Class V	--	Unisolated LQCA outside containment	N/A

The results of the Front-End analysis can be used in the screening and uncertainty analyses necessary to meet GL 88-20 requirements. They are presented in a fashion appropriate to the GL 88-20 reporting requirements.

1.3.2 Back-End Analysis - Level 2

The Back-End Analysis fundamentally consists of an assessment of the DAEC containment design and features to mitigate the challenges revealed from the Level 1 results. This is done by identifying a list of potential containment challenges from NRC and industry containment studies including NUREG-1150, IDCOR reports and Severe Accident Research Program reports. Associated with each potential containment challenge are design features which provide protection against such challenges.

Using knowledge of the plant design and the potential containment challenges, containment event trees (CETs) were developed. The DAEC model directly links the Front-End results to the CETs through linked event trees and fault trees. Two different basic containment event tree structures are used to describe containment response:

- Class I and III CETs: Containment initially intact. These sequences are characterized by an initial loss of coolant makeup to the reactor vessel that leads to core damage.

- Class II and IV CETs: Containment initially failed before core melt. For these classes of accidents, the primary containment boundary would fail before the molten core penetrates the reactor vessel. In Class II accident sequences, the inability to remove heat from the containment results in heat up of the suppression pool and a gradual containment pressurization. A more rapid pressurization is expected for Class IV accidents. Reactor power remains above decay heat levels so that the amount of energy transferred to the suppression pool exceeds its heat removal capacity.

The containment event tree is a tool for identifying and analyzing the spectrum of accident scenarios which may evolve following postulated core damage accidents. Each sequence in the CET is characterized in terms of the release of radionuclides to the environment (note that not all sequences result in a release).

The continuous spectrum of possible radionuclide release scenarios is represented by a discrete set of release categories or bins. The end states of the containment and phenomenological event sequences may be characterized according to certain key quantitative attributes that affect offsite consequences. These attributes include two important factors:

- Timing; and
- Total quantity of fission products released.

The description of the source term, the release timing, and the implications of each are determined using the results of MAAP calculations and past PRA evaluations. The information developed in previous studies has been used in making subjective assessments for these source term characterizations. The event sequences contributing to a radionuclide release are ranked on the basis of the product of the relative consequences (based on estimated radionuclide release fractions of noble gases, CsI, and Te) and their respective conditional probabilities, so that potentially risk-dominant scenarios are identified and adequately represented. Those scenarios that exhibit similar release characteristics in timing and radionuclide fractions are sorted and combined into groups of release categories to reduce the number of sequences required to calculate the risk profile.

It is necessary to develop a plan for addressing phenomenological uncertainties, in general, and NRC concerns specifically. NRC concerns are, for the most part, documented in a pair of letters to IDCOR. In these letters, a total of 18 phenomenological issues are addressed (T. Speis, USNRC, letters to A. Buhl, International Technology, dated September 22, 1986 and November 26, 1986). For most of these, the issues were either resolved or there were identified specific steps (generally uncertainty analyses) which should be taken to address the issue. For a few issues, no path for resolving the differences was identified.

Based on these letters and other documents, the NRC included a list of parameters in NUREG-1335 for which sensitivity cases may be performed. The following are some of the parameters listed in NUREG-1335:

- In-vessel hydrogen generation
- Core relocation characteristics
- Mode of RPV melt-through
- Containment performance
- Revaporization of deposited fission products.

These phenomena along with others are evaluated in either of two ways:

- Probabilistic sensitivity assessment requires the analyst to use a range of point estimate values to describe the frequency of occurrence for system performance and operator recovery, and phenomena considered in the model. The resulting impact on the model then reflects its sensitivity (i.e., in terms of changes in frequency), to these issues.

- Deterministic sensitivity assessment considers the extremes of the physical model (e.g., MAAP) used to represent the accident phenomena. The results of these deterministic calculations indicate the influence on the physical plant response associated with variations in the phenomenological modeling.

1.3.3 Specific Safety Features and Potential Plant Improvements

The DAEC IPE process provides for an assessment, by IEL&P, of the results in an effort to determine what modifications or improvements, if any, should be completed. This review is assisted by the performance and review of sensitivity analyses and importance results by the IPE project team.

1.3.4 Internal Review

The NRC has stated that the quality and comprehensiveness of the results derived from an IPE will depend on the vigor with which the utility applies the method of examination and on the utility's commitment to the intent of the IPE. Furthermore, the NRC has stated that the maximum benefit from the IPE would be realized if the licensee's staff were involved in all aspects of the examination to the degree that the knowledge gained from the examination becomes an integral part of plant procedures and training programs. Therefore, the NRC has requested each licensee to use its staff to the maximum extent possible in conducting the IPE by:

- Having utility engineers, involved in the analysis as well as in the technical review. The basis for the request in the GL 88-20 for involvement of utility staff in the IPE review is the belief that the maximum benefit from the performance of an IPE would be realized if the utility's staff were involved in all aspects of the examination and that involvement would facilitate integration of the knowledge gained from the examination into emergency operating procedures and training programs.

- Formally including an independent in-house review to validate both the IPE process and its results.

IEL&P has met these two requirements in the performance of the DAEC IPE.

1.4 SUMMARY OF RESULTS AND MAJOR FINDINGS

It is concluded that DAEC is a plant with a low risk of core damage and fission product release. It has only one sequence that meets the 1E-6 screening criteria and even this sequence is just at 1E-6. There are, therefore, no sequences or phenomena that are identified in this study that would make DAEC an outlier plant. As a result, no further changes would appear to be necessary.

The Level I analysis resulted in a Core Damage Frequency (CDF) of 7.84E-6 per reactor-year. Internal flood initiators are estimated to have an insignificant contribution to CDF. Figures 1.4-1 and 1.4-2 show the CDF by initiator. Figure 1.4-3 shows the CDF by initiator type. Figure 1.4-4 shows containment performance by magnitude and timing of release. Figure 1.4-5 shows radionuclide release by containment failure mode.

1.5 REPORT ORGANIZATION

NUREG-1335, "Individual Plant Examination: Submittal Guidance", provides a standard Table of Contents for the response to Generic Letter 88-20. This report adheres to the standard format provided. Basically, this consists of the following sections:

1. Executive Summary
2. Examination Description
3. Front-End Analysis
4. Back-End Analysis
5. Utility Participation and Internal Review Team
6. Plant Improvements and Unique Safety Features
7. Summary Conclusions

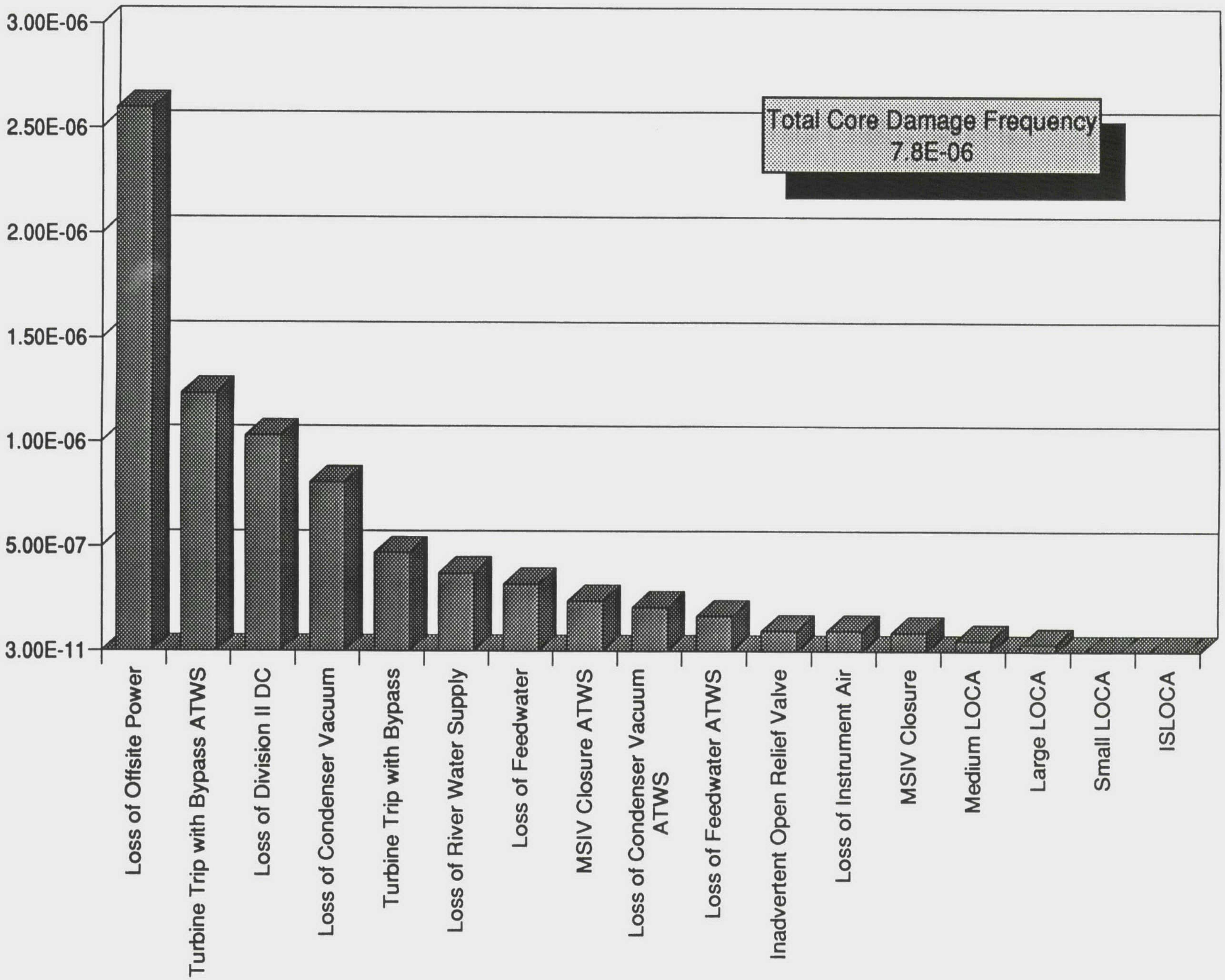


Figure 1.4-1
Core Damage by Initiator

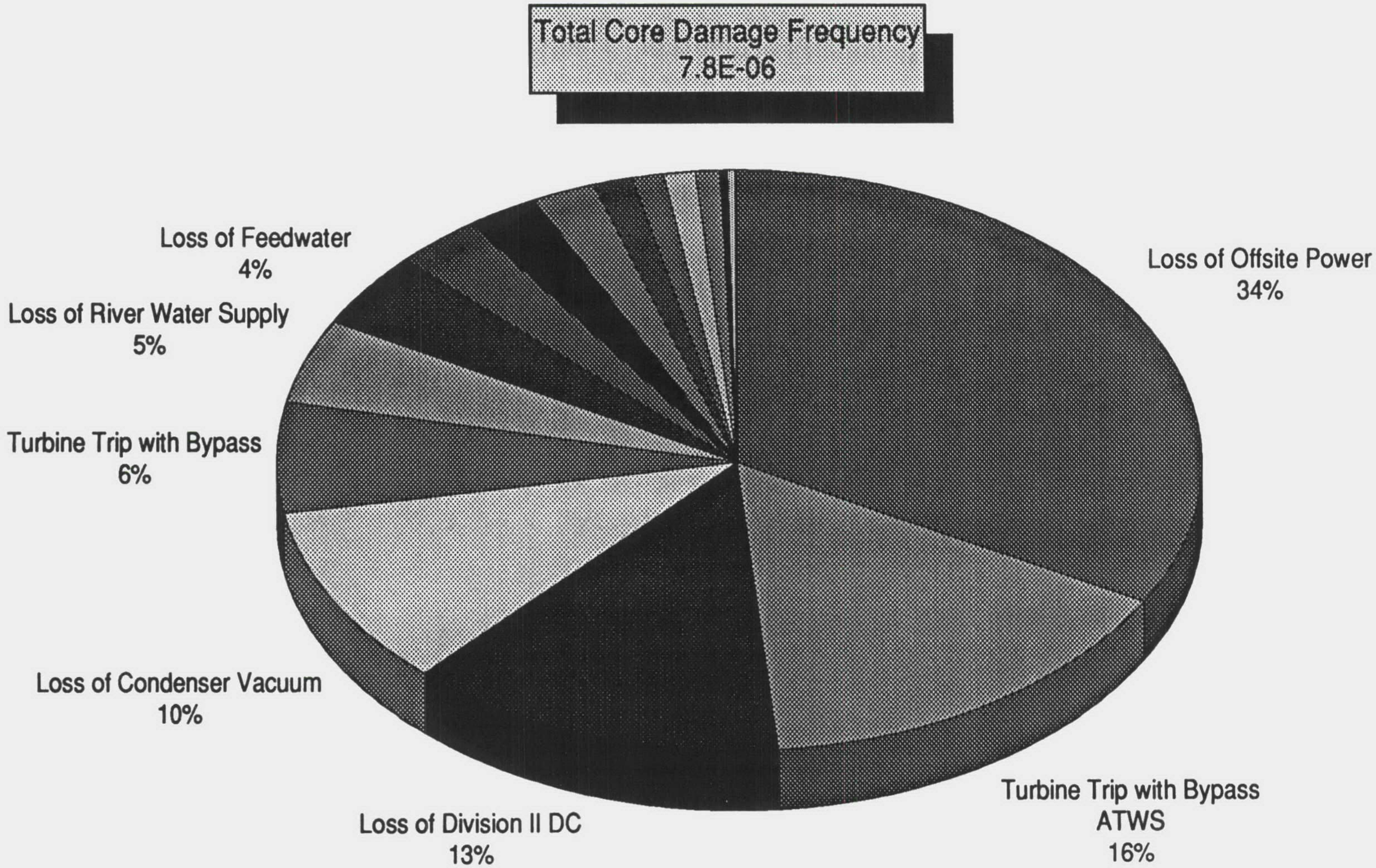


Figure 1.4-2
Contribution to Core Damage by Initiator

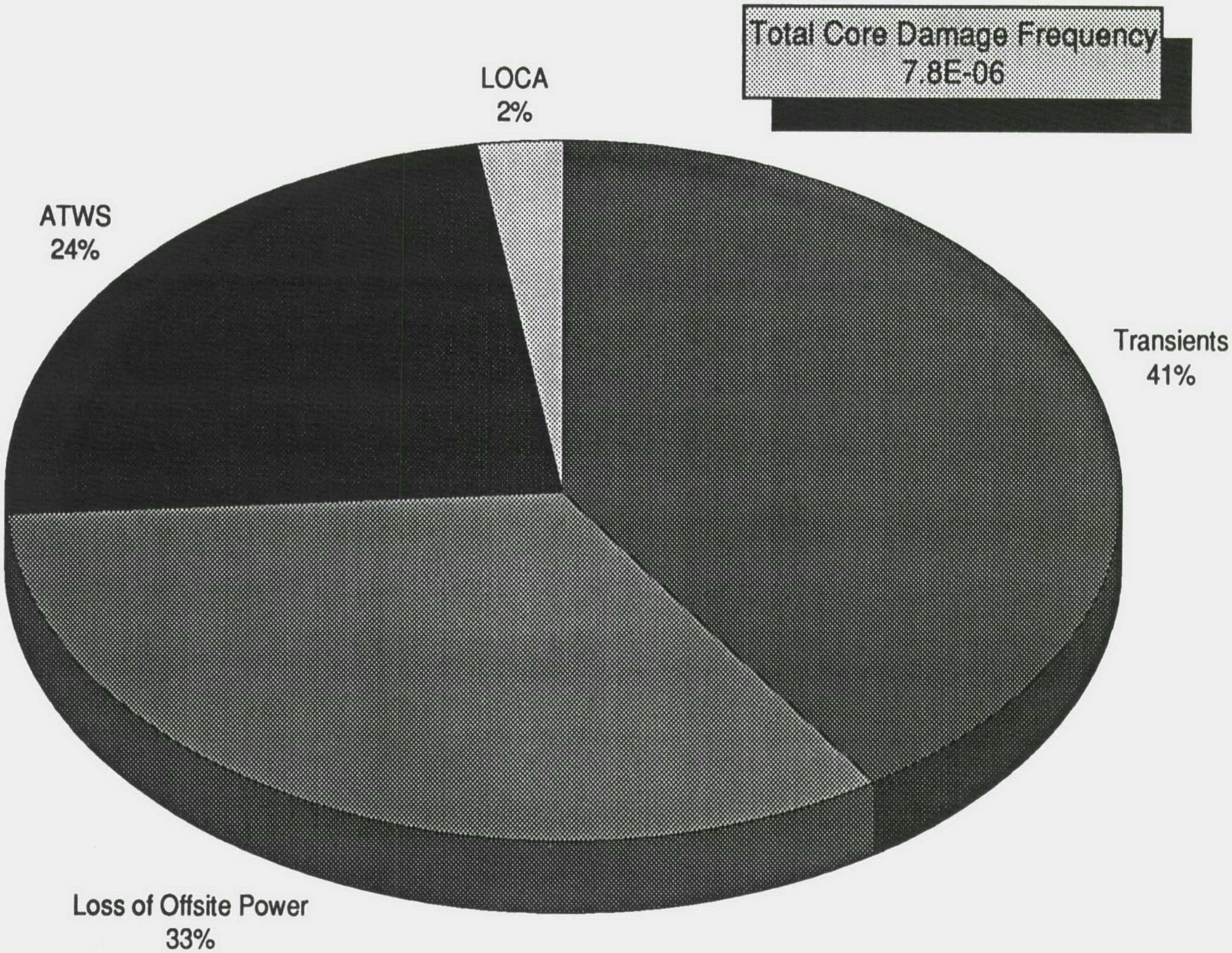


Figure 1.4-3
Contribution to Core Damage

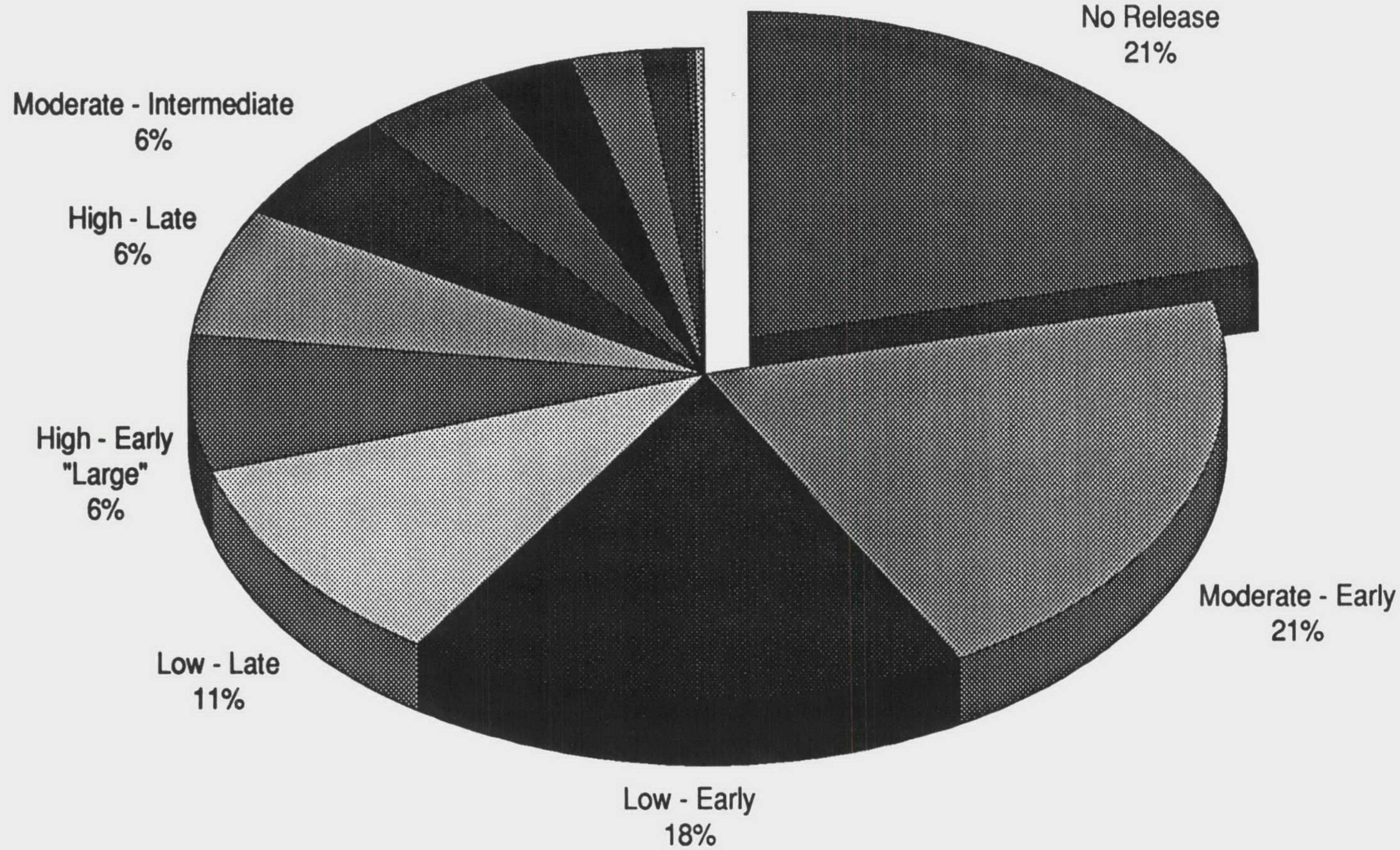


Figure 1.4-4
Containment Release Magnitude and Timing

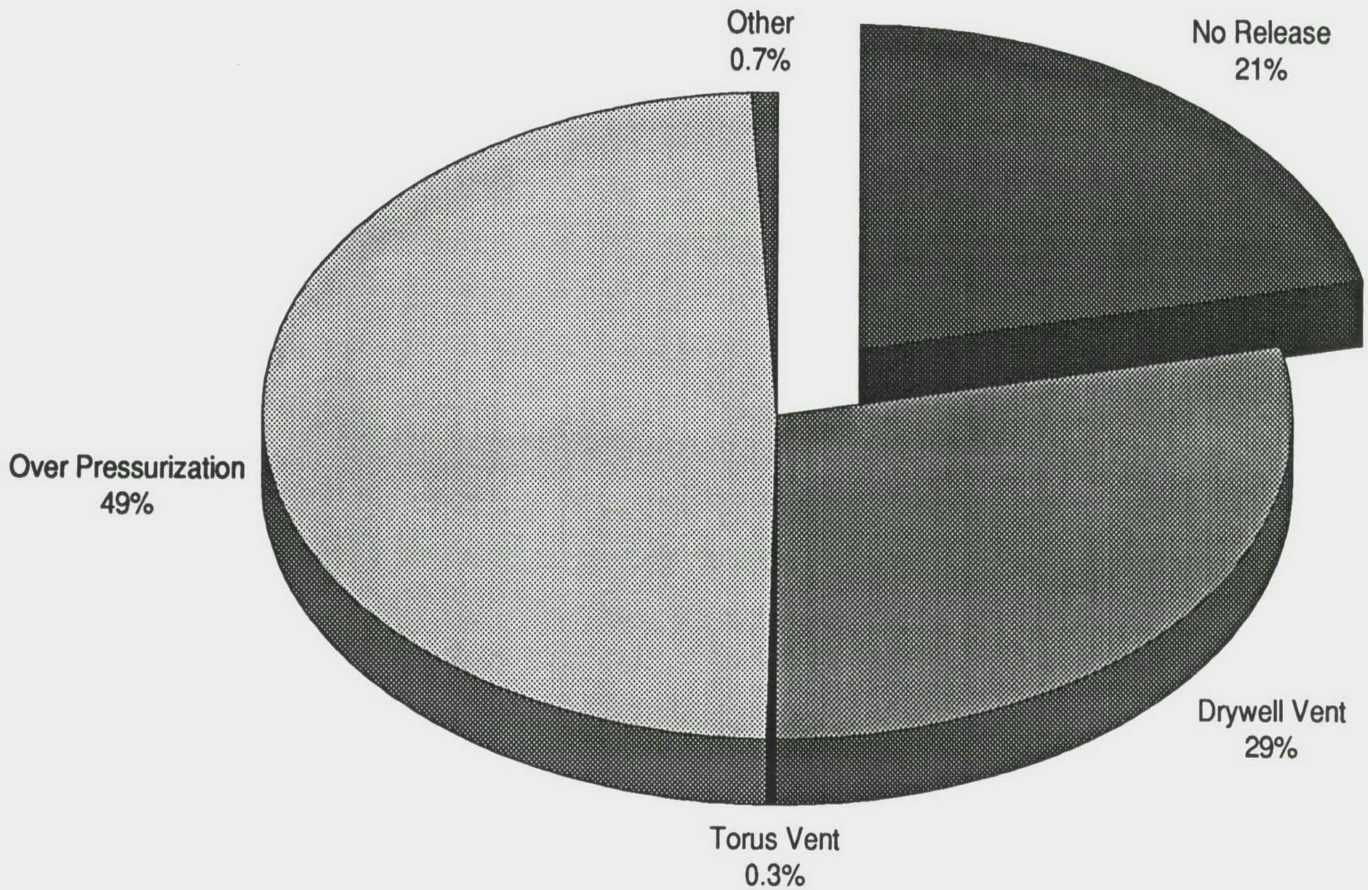


Figure 1.4-5
Containment Failure Modes

2. EXAMINATION DESCRIPTION

2.1 INTRODUCTION

This section describes how the IPE ensures that the primary objectives of NRC 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 Level 2 PRA.

2.2 CONFORMANCE WITH GENERIC LETTER AND SUPPORTING MATERIAL

The DAEC engineering and plant staff have been involved with the IPE process since its inception. They directed all aspects of the analysis with consulting services provided by ERIN Engineering, Gabor, Kenton & Associates, and Chicago Bridge & Iron. This was done to insure the knowledge gained from the examination would become an integral part of plant procedures and training programs and allow any possible future activities to be performed with limited involvement by consultants. Further details of the organization are provided in Section 5.

Several comprehensive reviews of the IPE work were performed by I.E. personnel in addition to the standard practice of calculation verification. A review team composed of a multidisciplinary group of plant and corporate staff members reviewed project information during development and prior to publication of this report. Plant system engineers prepared the system notebooks which formed the foundation of the level 1 analysis.

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 3.3.6. 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. These functions are listed as follows:

- Reactivity Control
- Primary Pressure Control
- Containment Pressure Control
- Reactor Coolant Inventory Control
- Containment Environment Control
- Reactor Coolant Inventory Control (injection post containment challenge)

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 I analysis was used as a direct input to the Level II sequence quantifications. The Level II was a detailed set of containment event trees (CETs). The DAEC Level II PRA uses an approach that was developed as part of the Nuclear Management and Resources Council (NUMARC) evaluation of containment performance. This evaluated all systems, phenomena, and operator actions important to containment performance during severe accidents. This approach allows considerable detail to be reflected in the overall containment performance without losing the ability to depict the response in a containment event tree format.

The CETs include the time sequence response with potential failure paths identified. Each failure path can then be characterized in terms of the timing and severity of the failure path itself. Similar failure paths occurring in similar time frames can be grouped together or binned to evaluate potential off-site health effects. This evaluation requires an understanding of containment ultimate capability as well as scenario-specific progression. Each event tree scenario is characterized and similar risk-contributing scenarios are binned together. Each can then be characterized by a type of source term or release fraction which can be associated with potential off-site health effects.

Like the Front-End analysis, the containment event tree is analyzed through the linking of fault trees. Whenever possible, fault trees used in the Front-End analysis are called into the CETs to avoid duplication of effort and to propagate dependencies.

2.3.2 System Analysis

2.3.2.1 Systems List for Fault Tree Development

A review of DAEC design concluded that 18 systems should be analyzed in fault trees to model the plant response to the initiators. The 18 systems are:

1) Automatic Depressurization and Safety Relief System
2) Condensate and Main Condenser
3) Control Rod Drive Pumps
4) Core Spray
5) Electric Power (AC/DC)
6) Emergency Service Water
7) Feedwater
8) General Service Water
9) HPCI
10) Instrumentation
11) RCIC
12) Recirculation Pump Trip
13) Residual Heat Removal
14) Residual Heat Removal Service Water
15) River Water
16) Standby Liquid Control
17) Torus Vent/Vapor Suppression
18) Well Water

Plant-specific fault trees were developed for each of these systems. The plant-specific fault trees were used to determine system availabilities for each system. The process of developing the plant-specific fault tress involve first documenting the system description in a system analysis. The IPE model represents the plant (hardware, procedures, etc.) as of startup from the 1992 refueling outage (April 1992), with the exception of inclusion of the hard pipe vent from the suppression pool which is included in the models. Installation of the vent is scheduled for completion later in 1992. The system analysis includes a simplified piping and instrumentation diagram, a dependency matrix, and a set of identified operating characteristics to enable the development of a fault tree logic model.

The logic model was checked for consistency and analyzed using the CAFTA computer code package. Dominant cutsets were identified as a result of the computer code application for overall system unavailability determination. Cutsets are ranked in terms of their contribution to the overall failure probability. Ranked cutsets were reviewed for accuracy and highlighted for evaluation by plant operations personnel.

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, UFSAR and operations manual descriptions.

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
System Description

Fault Tree Structure
Support Systems
Instrumentation and Controls
Technical Specifications
Operating Procedures
Test & Maintenance Procedures
Normal Operation Description
Component Trips
System Performance During Transients & Emergency Operations
Success Criteria
System Initiations
Impact of Failure of Systems on Other Systems
System Performance
Operating Experience
Assumptions
Support Drawings
System Walkdown Information

Multiple top events were defined for fault trees that served multiple functions. 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 river water system model contains only one general model for loss of flow to the Stilling Basin. This model would be the same for each of the specific service water systems that the river water system supplies; 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 elements were considered to be very important in developing the Duane Arnold 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 Duane Arnold fault trees are listed below:

All major active components - e.g. motors, pumps, diesel generators.

All components required to change position to fulfill function (including check valves).

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.

Generally pipe failures are not considered. However, pipe failures are considered for LOCA, ISLOCA, and Feed/Steam Line Breaks. These are high-energy line breaks or low-energy pipes that break due to pressurization from high-energy pipes. The only low-energy line break considered was SLC suction.

2.3.2.4 Dependency Treatment

The information that has been compiled by the individual system dependency matrices is combined into a master matrix shown in Section 3.2.3.

Dependency matrices were also developed as part of system fault tree modeling. These matrices are presented in Section 3.2.1 of this report. The dependency matrices, along with supporting text, were developed to document the following:

- Initiator effect on frontline and support systems,
- Support system effect on frontline and other support systems, and
- 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 (SAIC/EPRI) was used for managing and solving fault trees. This was run on a 80 486-based personal computer.

DAEC 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 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 (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 example, the CSF called Containment Environment Control is defined by re-establishing PCS (power conversion system), torus cooling, containment venting. In this example the PCS is further defined by other systems.

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 were accounted for by removing the cutsets from failed functions that would be subsummed by the cutsets that were in successful functions. 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 frequency for Duane Arnold.

The frequency 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 deleting the successful CSF cutsets.

Throughout these analyses, a truncation limit of 3E-11/yr 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

Duane Arnold 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:

- o motor driven feedpumps which are independent of main steam availability (+)
- o reliable switchyard configuration (+)
- o RHRSW, ESW, and GSW Systems capable of injection through RHR (+)

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

- o Large relative relief capacity with large accumulators on the air supply to SRVs (+)
- o Long term SRV activation for depressurization is independent of the availability of AC power (+)
- o 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. (-, +)
- o Unpiped safety valves. (-)

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

- o To meet the NRC ATWS Rule DAEC achieves equivalence by having two pumps delivering a minimum combined flow of 45 gpm of sodium pentaborate at 13 wt% concentration of natural boron-10. Each SLC pump is capable of a 28 gpm flow rate (-, +).

- o The bypass capacity of the turbine bypass valves is 25%. (+)

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

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

- o The emergency batteries have a minimum of four hours of capacity. 1D1 can last up to 6 hours without operator action. 1D2 can last 12 hours. D4 can last 30 hours. The batteries were upgraded several years ago and substantially upsized. (+)

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

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

- o Containment venting is essentally independent of AC power. A large accumulator on the air lines provides some independence, and the ability to repressurize the accumulator with a bottled gas source allows long term independence in station blackout scenarios. (+)

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. A potentially negative implication with respect to debris cooling is that the sump depth is 3 feet.

2.4.2 PRA Used for Comparison

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

1. The diversity of high and low pressure injection systems made the probability of a loss of coolant injection very low. Duane Arnold 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 Duane Arnold models. MAAP analysis of each of the functional sequence types was performed to determine the effects of plant conditions on system response.

3. Common cause failures contribute 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 Duane Arnold plant systems using the beta factor approach, and common cause factors were included explicitly in the fault trees.

4. Containment venting for DHR was considered.

5. 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.

7. Containment failure pressure was assumed to be about 130 psig. The drywell head is assumed to be the most likely place the containment fails.

See Section 6 of this report for specific insights on the DAEC PRA.

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 Final Safety Analysis Report (UFSAR)
    - Initiating event
    - System success criteria
    - System descriptions

2. Plant Operating Instructions
    - Operating procedures

3. Emergency Operating Procedures:
    - System operations during an emergency
    - Operator actions during an emergency

4. Duane Arnold Drawings
    - System components
    - System layout

- System interconnections

5. Scram Reports, Shutdown Reports, License Event Reports
   - Failure data
   - Plant response

6. Plant Surveillance Procedures
   - Test frequencies

7. NPRDS
   - Generic failure data applicability

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

9. 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 prepare the 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 areas outside containment including the reactor building, the torus room, the turbine building, the pumphouse and intake structure, and the simulator. This walkdown included members of the DAEC PRA group, and consultants. The human error analysis walkdown included a DAEC analyst responsible for the 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 and a consulting engineer. One of the PRA analysts was NRC SRO Licensed on DAEC. They looked at flood sources, components, supplies and drains in each area, and the interconnections to adjacent areas.

3.1.1 Initiating Events

3.1.1.1 Plant Specific and Generic Initiating Events

Events which require a manual shutdown are called initiating events. There are many potential types of initiating events. They include internal events, (e.g., loss of feedwater, turbine trip, MSIV closure, LOCA, etc.), as well as external events (e.g., earthquakes, fires, tornadoes, etc.). This report focuses on internal events in accordance with Generic Letter 88-20. Evaluation of initiators caused by external events will be addressed as a part of DAEC's response to the NRC's IPEEE requirement. Table 3.1-1 summarizes the initiating events evaluated in the DAEC IPE and provides the frequency for each initiating event.

3.1.1.2 Initiating Event Frequencies

Transient occurrence data from the period 03/03/74 through 09/18/90 were used to derive the plant specific initiating event frequency estimates. Descriptions of the occurrences from scram reports, LERs, shutdown reports, transient occurrence reports, NSAC reports, and STA incident 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 Table 3.1-1.

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
    - ISLOCA

2. Anticipated transients and special initiators.
    - Turbine trip
    - Loss of feedwater
    - MSIV closure
    - Loss of condenser vacuum
    - Loss of river water
    - Inadvertent open relief valve
    - Loss of air
    - Loss of DC bus

3. Loss of offsite power.

4. Anticipated transients without scram (ATWS).
    - Turbine trip with bypass
    - Loss of feedwater
    - MSIV closure
    - Loss of condenser vacuum

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. The DAEC risk analysis 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 DAEC 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) system due to low vessel pressure, and

   (c) results 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) HPCI or RCIC operation are sufficient to prevent uncovering the core, and

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

4. LOCAs Outside Containment (including 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 pressure relief 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

EPRI NP-220, previous risk analyses, NUREG/CR-2300, DAEC UFSAR, LER data base, NUREG/CR-3862, BWROG Scram Reduction effort, as well as detailed plant specific reviews and analysis develop a preliminary list of transient and special initiator events appropriate for consideration in the DAEC PRA. Initiating events which were found to have a low initiating frequency or minimal impact on plant shutdown include loss of instrument nitrogen (because of accumulators and air system redundancy), loss of reactor building closed cooling water (because it affects only the CRD pumps), 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 DAEC Level 1 results because modifications to reactor level instrumentation, during the 1992 refueling outage, would preclude loss of drywell cooling effects as an initiating transient.

<u>Loss of Offsite Power</u> - The loss of offsite power initiating event was modeled separately from the other anticipated transients. The primary factors which required special treatment were consideration of recovery of offsite power and repair of diesel generators.

<u>ATWS</u> - This category included the most frequent anticipated transients, 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 DAEC 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 will 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.

Table 3.1-1

DAEC INITIATING EVENT FREQUENCY ESTIMATES

TYPE OF EVENT TREE	INITIATING EVENT	DESIGNATOR	FREQUENCY (per year)	SOURCE
TRANSIENT	Turbine Trip with bypass Loss of Feedwater MSIV Closure Loss of Condenser Vacuum IORV Loss of Offsite Power	TT TF TM TC TI TE	3.6 .12 .64 .12 .064 .117	Plant Data Plant Data Plant Data Plant Data Plant Data Plant Data(1)
LOCA	Large LOCA Medium LOCA Small LOCA ISLOCA	A S1 S2 V	3E-4 3E-3 1E-2 1.8E-4	Brunswick PRA WASH-1400 WASH-1400 (2)
SPECIAL INITIATORS	Loss of River Water Loss of Instrument Air Loss of Single 125V DC Bus	TR TA TDC	7.5E-4 8E-3 1E-3	Generic (3) Generic (6) Generic (5)
ATWS	ATWS	-----	(4)	NUREG-0460

(1) Using Plant data combined with site specific information and NUMARC 87-00 methodology.

(2) Using NSAC/154 as guidance, plant specific models were created and generic failure data for basic events was utilized. A specific initiator was generated for each low pressure line that was considered in the analysis. The number in the above table is the sum of these individual initiators.

(3) No sudden complete loss of service water observed in 1340 reactor years. For conservatism assume 1 failure in that time and get 1/1340 = 7.5E-4.

(4) ATWS sequences are created by logically combining mechanical and electrical failure to scram with the failure of alternate rod insertion (ARI). This gives an ATWS initiator number of 1.02E-5 which is then multiplied by the transient initiators to give the ATWS frequency for a specific transient ATWS event.

(5) Generic data with events that are not applicable to DAEC removed.

(6) Since DAEC has modified the Feedwater/Condensate system to cope with a short term loss of air, this is based on industry events lasting longer than 1 hour.

3.1.2 Event Sequence Analysis

3.1.2.0 General Methodology

The purpose of this section is to provide a description of the Event Sequence Analysis for DAEC. In the explanation, references will be made to the functional level fault trees that define the assumptions used in the quantification. The functional events in the trees are generally modeled by a single fault tree, but some may require a special version of the fault tree, depending on the accident initiator and/or accident sequence.

This section is organized such that each event tree is discussed in a separate section. Function level fault trees, or top logic, corresponding to the event trees headings are presented in Section 3.2.2. Table 3.1-2 lists the event sequences that are analyzed.

The BWROG Emergency Procedure Guidelines identify three basic safety functions:

- Reactor Pressure Vessel Control
- Primary Containment Control
- Secondary Containment Control

Additional detail is provided through the Contingency Procedures, which are entered if the operator is unable to successfully accomplish and confirm the stabilization of the vessel or containment parameters due to system or component failures. These address:

- Adequate Reactivity Control
- Reactor Pressure Control

Table 3.1-2

Summary of the Event Tree Models

INITIATOR	SECTION
Transient	
Turbine Trip	3.1.2.1
Loss of Feedwater	3.1.2.2
MSIV Closure	3.1.2.3
Loss of Condenser Vacuum	3.1.2.4
Stuck Open Relief Valve	3.1.2.5
Loss of Offsite Power	3.1.2.6
LOCA	
Large Break	3.1.2.7
Medium Break	3.1.2.8
Small Break	3.1.2.9
Special Initiator	
Loss of Instrument Air	3.1.2.10
Loss of River Water	3.1.2.11
Loss of a DC Bus	3.1.2.12
ATWS	
Turbine Trip	3.1.2.13
Loss of Feedwater	3.1.2.14
MSIV Closure	3.1.2.15
Loss of Condenser Vacuum	3.1.2.16
Other	3.1.2.17
Containment Bypass	
ISLOCA	3.1.2.18

- Maintenance of Vessel Water Level
- Limitation of Drywell Pressure
- Limitation of Suppression Pool Temperature
- Maintenance of Suppression Pool Water Level
- Limitation of Drywell Temperature
- Limitation of Containment temperature.

The failure to maintain any of these functions could result in a threat to either the ability to maintain core cooling or to the ability to maintain containment integrity. Table 3.1-3 correlates the safety functions identified in the EPG's with the safety functions modeled in the DAEC IPE.

The following are brief descriptions of each of the safety functions applicable to the DAEC analysis:

Adequate Reactivity Control

This safety function refers to the control of the fission process in the core such that the reactor power is maintained at an appropriate level. Failure to do so could result in:

1. Fuel damage if cooling is not maintained commensurate with the power level; and

2. Containment pressurization if containment pressure relief does not match the pressurization rate.

Table 3.1-3

Comparison of the Critical Safety Functions Derived from the BWROG EPGs, compared with the Accident Sequence Functional Events

BWROG EPGs Critical Safety Functions		Accident Sequence Functional Events
General	Specific	
Reactor Pressure Vessel Control Guideline	RPV Reactivity Control	RPV Reactivity Control
	RPV Pressure Control	RPV Pressure Control
	RPV Water Level	RPV Inventory Control (High Pressure)
		RPV Inventory Control (Low Pressure)
	RPV Depressurization	RPV Depressurization
Containment Control Guideline	Containment Pressure Control	Limitation of Drywell Pressure
	Suppression Pool Temperature Control	Limitation of Suppression Pool Temperature
	Suppression Pool Level Control	Maintenance of Suppression Pool Level
	Containment Temperature Control	Containment Temperature Control (addressed qualitatively)
	Combustible Gas Control	Addressed in Level II
Secondary Containment Control	Secondary Containment Control Radiation Water Level Temperature	Addressed in Level II

Reactor Pressure Control

Failures include not only the failure of SRVs to open, but failures of the SRVs to reclose as well. Possible consequences of a failure to control RPV pressure include vessel overpressurization and/or rupture, and subsequent vessel blowdown resulting in loss of coolant inventory. This can also challenge the containment due to steam continuously flowing to the suppression pool through the SRVs or directly into the drywell through the code safeties.

Maintenance of Vessel Water Level

In general, this refers to maintaining an adequate level of water in the reactor pressure vessel. A failure of this critical safety function could result in fuel damage if the fuel becomes uncovered and remains uncovered for a sufficient period of time such that clad temperatures rise to the point of significant metal/water reaction. A failure is defined as 1/3 core height and decreasing.

Limitation of Drywell Pressure

Failure of this CSF could also result in a rising primary containment pressure, with eventual primary containment failure. It is possible that primary containment failure can cause a saturated suppression pool to flash to steam, causing a loss of NPSH to all makeup systems using the pool as a suction source. It is also possible that a catastrophic containment failure can render any other injection source unavailable due to the uncertainty of the integrity of instrumentation, piping, or structural components following the failure.

Limitation of Suppression Pool Temperature

This refers to maintenance of the suppression pool temperature below the minimum required for NPSH of ECCS pumps that use the suppression pool as a source. Consideration is also given to the most limiting suppression pool temperature LCO, and other limits that affect operations. Some of these are Boron Injection Initiation Temperature, Drywell Spray Initiation Limit, or Heat Capacity Temperature Limit. Failure to maintain the suppression pool temperature within these limits could result in:

1. Adverse effects on containment;
2. Adverse Impacts on ECCS equipment; or
3. Failure to provide sufficient steam suppression, which in turn could result in unacceptable containment pressures and temperatures.

Maintenance of Suppression Pool Water Level

This is defined as maintenance of the suppression pool water level below the Suppression Pool Load Limit and above the elevation of the safety relief valve discharge "Tee" quenchers. Suppression pool water level has a potentially strong impact on containment integrity. Failure to suppress steam would result in a rising primary containment pressure, potentially leading to containment failure and subsequent core uncovery. This is not a dominant concern at DAEC because any water that is taken from the suppression pool and injected into the RPV will eventually return to the pool via various paths.

Limitation of Drywell Temperature

This is defined as maintenance of the temperature below the maximum temperature for which the drywell is designed. Failure to do so could compromise important ECCS equipment or the integrity of primary containment. Environmental Qualification considerations are given to the equipment in the drywell.

These functions, and the top event headings that comprise them, are described for anticipated transients in the following discussions. These top events will vary slightly in response and quantification in each event tree, depending on the initiator and/or sequences.

Each event tree has specific system functions as headings. Only the front line systems and the systems needed to define the end states are identified as top events. This methodology is based on the "small event tree/large fault tree" approach. All support systems' contributions to the core damage frequency are modeled by including support system dependencies in the fault trees of the front line systems. Table 3.1-4 lists the front line systems that were used to model the DAEC Critical Safety Functions.

Reactivity Control

Reactor Subcritical

C(SCRAM)

Failure to bring the reactor subcritical results in an accident sequence leading to a transfer to the ATWS event trees. This includes both the electrical and mechanical failures of the Reactor Protective System. The sequences quantified in the general transient trees are those in which sufficient control rods are successfully inserted by the Reactor Protective System.

The ATWS events have a more elaborate quantification of reactivity control that includes the operation of the Standby Liquid Control system. It is explained in more detail in section 3.1.2.13 - Turbine Trip with Bypass ATWS.

Table 3.1-4

Translation of Accident Sequence Functional Events to System Designator

Accident Sequence Functional Events	Systems
Reactivity Control	RPS SLC
RPV Pressure Control	SRVs TBVs HPCI/RCIC Steam Lines*
RPV Inventory Control (High Pressure)	Feedwater HPCI RCIC CRD
RPV Depressurization	SRVs TBVs* HPCI/RCIC Steam Lines*
RPV Inventory Control (Low Pressure)	LPCI Core Spray Condensate RHR Service Water ESW GSW
Containment Pressure Control	Main Condenser Torus Cooling Containment Spray Containment Venting
Containment Temperature Control**	Drywell Coolers Containment Spray
Vapor Suppression	Vacuum Breakers Drywell Sprays

* No credit taken in DAEC Analysis

** Not explicitly modeled

Pressure Control

Safety Relief Valves Open

M(SRVSOPEN)

This event represents the opening of the safety relief valves and the code safety valves to limit rector coolant pressure below the design pressure of the RPV. Failure of a sufficient number of valves to open may lead to excessive pressure and the potential for an induced LOCA. For many of the transients two of the valves are required to open to be successful. However, for events with the MSIVs open and the bypass valves operable, the TBVs provide adequate pressure control.

Safety Relief Valves Reclose

P(SRVSCLOSE)

The relief valves that open as a result of a transient must reclose to prevent discharge of excessive reactor coolant, or discharge of excessive heat to the suppression pool. The impact on plant safety arises from the additional heat load on the RHR system due to the stuck open relief valve(s).

WASH-1400 assumes that the occurrence of a single stuck open relief valve does not affect the capability of the PCS to remove an adequate amount of heat to preserve containment integrity. The SORVs do not affect the PCS, but impact the ability to preserve the steam flow to the main condenser due to reduced RPV pressure. Therefore, in the event tree quantification, the PCS reliability is the same as in the non-SORV cases, while the turbine bypass valves are assigned comparatively higher unavailabilities.

It should be noted that industry operating experience data demonstrate that 85% of SORV incidents have reclosed on their own as the primary system pressure drops below relief valve spring pressure.

Vapor Suppression

D(VAPOR:SUPP)

Vapor Suppression

During LOCAs, the high temperature primary steam is released directly into the drywell. This can also occur if a SRV tailpipe breaks during relief to the torus. Downcomer pipes transfer the steam and other gases from the drywell to the suppression pool. The steam is condensed in the suppression pool; thereby, limiting the pressure rise in the drywell. Containment overpressurization may occur if the suppression pool is bypassed, possibly due to stuck open vacuum breakers. Subsequent containment overpressurization failure could have an adverse impact on the availability of low pressure injection systems.

Drywell Sprays

In the initial phases of small and medium LOCAs, the drywell spray mode of RHR can limit containment pressure in the case of a vapor suppression failure.

Success for this function requires one RHR pump, one drywell spray valve, and successful operator action to manually initiate drywell spray.

Failure of the vapor suppression function leads to an overpressurization and subsequent failure of the primary containment.

High Pressure Coolant Injection

Main Condenser Available

Q(MC:AVAIL), Q(MC:RECOV)

This node identifies the ability of the main condenser to provide a makeup source and an external heat sink. For success at this node the MSIVs in one of the four main steam lines must remain open (or be reopened following the initiating event) and at least one of the circulating pumps must be operating to deliver cooling water to the main condenser.

Success at this node indicates that the main condenser is available as a heat sink and may be used to complete the steam conversion process. Failure at this node indicates that the main condenser is not available as a heat sink, but the feedwater/condensate system may still be considered for coolant injection if there is additional makeup to the hotwell.

Feedwater/Condensate Available

Q(FW:CND), Q(FW:CND:RECOV)

The feedwater and condensate system is used as the normal method of maintaining an adequate coolant inventory in the reactor vessel. To be successful, the feedwater and condensate system must inject water into the reactor vessel within 30 - 60 minutes of the initiating event, depending on the RPV depressurization status. Since the feedwater pumps are motor driven, main steam is not required for the operation of this system.

The initial reactor water level swell following the turbine trip may lead to a feedwater trip on high reactor water level if the operators do not take manual control of feedwater level controller in time. This is one of the immediate operator actions following a reactor trip. If feedwater trips, the operator must take action to recover it or automatic safety system actuation will take place after the water level in the reactor vessel decreases below 119 1/2 inches above TAF.

Following a feedwater trip at 211 inches, the reactor vessel water level drops quite rapidly. The conditional probability of the operator regaining feedwater prior to 119 1/2" is included in the feedwater system fault tree. Because feedwater is a normally operating system with which the operators are extremely familiar, this system represents an important method of maintaining reactor coolant inventory.

Success at this node indicates that the feedwater/condensate system is being used as a high pressure injection source. If the main condenser is also available, then containment heat is being rejected, and the sequence transfers to a safe end state. Failure at this node indicates that the feedwater/condensate system is not supplying high pressure injection, requiring the use of the HPCI or RCIC systems.

HPCI/RCIC Available

U(H:R), U(H), U(R)

HPCI and RCIC are steam driven high pressure injection systems. Their steam supply is provided from the main steam lines upstream of the MSIVs, and they discharge into the feedwater lines. Both systems are initially aligned to take a suction from the CSTs. On either a high Torus level or a low CST level indication, HPCI will automatically transfer to take a suction from the Torus. RCIC transfers only on low CST level.

The Reactor Core Isolation Cooling system is designed to start automatically upon receipt of a low-low reactor water level signal. The operating pressure range is from 1100 psig down to 100 psig. The flow capacity of the RCIC system is 400 gpm.

The High Pressure Coolant Injection system is also designed to start automatically upon receipt of a low-low reactor water level signal. It is also started on a high drywell pressure signal. Injection begins within 30 seconds. Its operating pressure range is from 1100 psig down to 150 psig. The flow capacity of the HPCI system 3000 gpm.

Success at this node indicates that either RCIC or HPCI is supplying high pressure injection to the RPV. Failure at this node indicates that all high pressure injection is unavailable, requiring RPV depressurization.

These trees are quantified with two different sets of success criteria for each event tree. One, which the systems can be used for injection throughout the sequence, requires the suction swap to the torus to be successful and that AC power to the chargers is necessary for the long term survival of the batteries. The other set, which is used in sequences where HPCI/RCIC provide core injection while the operators align alternate low pressure injection to be successful, does not require either the torus or AC power. The second set criteria are used in sequences that involve a stuck open relief valve or an equivalent LOCA.

Timely Depressurization

Depressurization

X(TIME:RX:DEP)

In the event that the high pressure systems are unavailable to maintain adequate coolant inventory, the RPV can be depressurized to allow the use of the core spray, RHR, or other low pressure injection. The principal method of depressurization is by manual operation of the SRVs.

ADS initiation will automatically occur if a low-low-low reactor water level occurs coincident with a confirmatory low reactor water level and a signal indicating that one of the core spray or RHR pumps is running. This will occur only if ADS has not been locked out by the operators, per procedure.

The preferred method of depressurization is through the Turbine bypass valves to the main condenser. The main steam isolation valves must be maintained open for this to be successful. Steaming through HPCI and RCIC will also depressurize the RPV, however the operators need to take manual control of the flow split between the RPV and the CST prior to a high reactor water level trip of these systems. Only depressurization through the SRVs is considered in this node.

The timing of the reactor depressurization during the accident sequence can affect core integrity. In order to minimize core uncovery, it would be preferable if the ADS initiation occurred soon (approximately 10 minutes) after the reactor trip.

Success at this node indicates that the RPV has been depressurized in time to allow injection of low pressure systems to prevent the onset of core damage. Failure at this node indicates that the RPV remains at high pressure and that core damage occurs.

Low Pressure Coolant Injection

Core Spray or LPCI Available

V(CS:LPCI), V(LPCI)

The Core Spray system initiates after a low-low-low water level signal, or a high drywell pressure signal. Though the spray pumps may be running, water is not injected unless RPV pressure is less than 300 psi.

The Low Pressure Coolant Injection system is similar to the Core Spray system, however it does not begin to inject until 250 psig. Both CS and LPCI take suction from the torus.

Success at this node indicates that either Core Spray or LPCI is providing low pressure injection to the RPV. Failure at this node indicates that neither is providing injection, requiring injection from the alternate low pressure systems.

External Water Injection

V(C:R:G:E), V(R:G:E)

If the core spray and LPCI systems fail or are unavailable, other alternate low pressure systems may be manually aligned. The systems considered under this node are Condensate injection through the feedwater lines, and either the RHR Service Water, Emergency Service Water, or General Service Water injection through the RHR-RHRSW cross tie.

Successful use of condensate injection is conditional upon earlier feedwater failure. This is handled within the fault trees that makeup this function.

Success at this node indicates that at least one of the systems listed above is supplying low pressure injection to the RPV. Failure at this node indicates that no coolant injection is being supplied. The RPV is at low pressure and core damage occurs.

Containment Heat Removal

Power Conversion System

Z(PCS:RESTORE)

The use of the PCS as a method of containment heat removal is possible if at least one main steam line path can be maintained, and there is not a large diversion of decay heat directly to the suppression pool. For most cases, the PCS can either remain intact throughout the transient, or be regained with a fairly high degree of confidence. The probability of regaining the PCS, even if the MSIVs close early in the transient, is likely depending on whether the condenser vacuum can be restored using the mechanical vacuum pump.

For the PCS to successfully transfer decay heat to the environment, all of the following are required:

- One complete feedwater/condensate piping system is operable and able to deliver water from the condenser hotwell to the reactor vessel. This requires that the condensate and feedwater pumps be operable, or a condensate pump be operable. The operators must reduce reactor pressure below 540 psi to use condensate.

- The main steam line isolation valves in one of the four main steam lines must remain open, or be reopened if they close as a result of the initiator. The turbine bypass valves must be operable. If condenser vacuum falls below seven inches of Hg, the low vacuum interlocks on the bypass valves must be overridden.

- At least one of the circulating water pumps must be operable and delivering cooling water to the main condenser.

- Steam line drains can pass about 5% steam to the condenser without condenser vacuum. This has not been credited in this analysis.

There are some events in which this capability may be inhibited. These are:

- Isolation of the MSIVs due to low-low-low reactor water level

- Loss of the instrument nitrogen, leading to the inability to keep the MSIVs open

- High temperatures in the steam tunnel

- Loss of condenser vacuum

- High steam flow signals from the steam lines

- Low primary system pressures with the mode switch in "RUN".

Since the availablilty of water in the hotwell for feedwater injection is partly dependent on the PCS, this heading will appear twice in the event trees. Once prior to the Feedwater/Condensate heading. This will cover the non-isolation and early (~30 min) recovery cases. The second will deal with late recovery (after isolation due to depressurization or other means) as part of Containment Heat Removal. If early PCS fails (due to hardware failures), it will be assumed that late PCS recovery by reopening the MSIVs will not be possible.

Given successful coolant injection can be maintained without the core being uncovered, the operators can easily restore the PCS unless an equipment failure has occurred that induces an isolation.

Success at this node indicates that the main condenser is being used as the primary containment heat removal method. Failure at this node indicates that containment heat removal must be provided through cooling of the Torus.

Torus Cooling Mode of RHR

W(TCOOL)

The RHR system must provide a complete flow path from and to the containment through at least one RHR heat exchanger. In addition, the RHR Service Water system must provide cooling water to the corresponding RHR heat exchanger from the Emergency Service Water pits in order for RHR to effectively remove decay heat from containment for transients and LOCAs with successful reactivity control.

Success at this node indicates that the torus cooling mode of RHR is being used to remove heat from the containment. Failure at this node indicates that the normal containment heat rejection methods are unavailable, and containment venting is required.

Containment Vent

W(VENT)

Venting may be initiated after other means of containment heat removal are unavailable and uncontrolled release of drywell steam into the reactor building is imminent. Containment venting directs the release of steam to the containment venting pathways in a controlled manner, and allows the release to be scrubbed by the suppression pool water.

Success at this node indicates that containment heat is being rejected via a Torus vent path. Failure at this node indicates that the containment vent pathway has not been opened, and that containment failure due to overpressurization cannot be prevented.

Injection Post Containment Challenge

QUV(PST:CNT:CHL)

A number of past BWR PRAs have inferred that core melt would follow containment failure. NUREG-1150 assumed that after containment failure core damage could be prevented by adequate injection. The DAEC model is similar except less credit is taken for continuing injection after containment failure than in NUREG-1150 because of EOP interpretation differences.

In the event that containment integrity has been breached, by failure or venting, there could be detrimental effects on the ability to continue core cooling. Such effects include:

- Loss of net positive suction head for pumps that take suction from the suppression pool

- Harsh reactor building environmental conditions

- Steam binding of ECCS pumps

- Failure at penetrations of injection systems due to containment catastrophic failure

Conditional probabilities are included in the fault trees for injection systems to account for these possibilities.

The QUV node is interpreted in two ways depending on whether the failure sequence is a transient or small LOCA, or a LOCA or SORV.

<u>Transient or Small LOCA</u>

Continued injection to the RPV at high containment pressure can either be with water from the suppression pool or external sources. With no containment heat removal, at approximately 18 hours after the initiating event, containment pressure will reach the "Maximum Primary Containment Water Level Limit" (MPCWLL - 95 feet or 53 psia). The EOPs instruct the operators to terminate injection into the primary containment from sources external to the containment. At approximately 24 hours, containment pressure reaches 100 psia, which will close the SRVs due to high nitrogen backpressure on the pilot valve. HPCI and RCIC are incapable of injection from the suppression pool due to high pool temperatures, and Core Spray and RHR cannot inject due to a repressurization of the reactor vessel.

The model assumes that the operators follow procedures and use only injection from the suppression pool. This assumption forces the end state to core damage before containment failure. It is assigned a Class IIT damage state.

<u>LOCA or SORV</u>

In LOCA cases, the RPV will not repressurize due to closure of the SRVs. Therefore there is a chance that Core Spray or RHR can continue to provide injection after containment high pressure. There is a risk that these systems will fail concurrently with containment failure due to the adverse conditions mentioned above.

It is assumed that if the containment breach is due to controlled venting, there will be no impact on the injection systems. If the breach is due to containment failure, a point estimate of 0.66 is assigned to the failure of the remaining low pressure systems.

In the following sections, each event tree will be presented. A general description of the tree will be given, followed by the information necessary to convert the general functional descriptions (above) for use in the tree. Figures 3.1.2-1 through 3.1.2-17 provide a graphical representation of each of the analyzed events.

3.1.2.1 Turbine Trip with Bypass Initiator (TT)

The turbine trip initiator represents malfunctions that result in a manual or automatic trip of the main turbine. At the time of this event, offsite power, the main condenser, and the turbine bypass valves are initially available. See Figure 3.1-1.

3.1.2.1.1 General Description

Examples of events that contribute to the turbine trip initiator include electric load rejection, spurious reactor trips, low feedwater flow, trip of a recirculation pump, or inadvertent turbine control valve closure. This type of initiator challenges, but does not directly disable critical plant safety functions.

The following is a typical plant short term response indicating the various safety functions that are challenged after a turbine trip event.

- Turbine trip initiates rapid closure of the control valves and the main stop valves

- Turbine stop valve closure initiates an RPS signal causing the reactor to scram (Reactivity Control)

- Excess reactor steam initiates opening of the turbine bypass valves (Primary Pressure Control)

- High vessel level can trip main feedwater pumps (Reactor Coolant Inventory) if operators fail to take immediate action to maintain main feed.

- Low vessel level setpoints may initiate HPCI/RCIC or a Group 1 isolation

3.1.2.1.2 Event Tree Node Descriptions

The Turbine Trip with Bypass event tree uses the default quantification of the function headings for all nodes.

3.1.2.2 Loss of Feedwater Initiator (TF)

3.1.2.2.1 General Description

The Loss of Feedwater initiator represents a greater challenge to the reactor coolant makeup function than the Turbine Trip with Bypass initiator. However, it is potentially less severe than isolation events since the main condenser is initially available. With the main condenser available, the turbine bypass valves are initially available for primary pressure control.

For a Loss of Feedwater initiator, normal makeup to the reactor vessel is lost or reduced creating conditions that require either an automatic or manual scram.

For the most part, this event is similar to the Turbine Trip with Bypass except feedwater is initially unavailable. This requires that other sources of reactor coolant makeup must operate if feedwater cannot be restored. See Figure 3.1-2.

3.1.2.2.2 Event Tree Node Descriptions

Most of the function quantifications for the Loss of Feedwater are the same as those for Turbine Trip with Bypass. The following sections list the differences for the affected nodes.

Main Condenser Available, Q(MC:AVAIL)

This node requires that either one train of Feedwater or Condensate be available for long term condenser use as a heat sink. This quantification includes the failures of these systems to restart as potential failure modes.

Feedwater and Condensate Recovered, Q(FW:CND)

The quantification of this node includes the possibility that the initiator was caused by a catastrophic failure of the Feedwater system. In addition, if the Feedwater system is recoverable, the failure of the equipment to start is also included as a potential failure mode.

PCS Reestablished, Z(PCS:RESTORE)

This node requires that either one train of Feedwater or Condensate be available for long term condenser use as a heat sink. This quantification includes the failures of these systems to restart as potential failure modes.

3.1.2.3 MSIV Closure Initiator (TM)

3.1.2.3.1 General Description

The MSIV closure initiator represents malfunctions that result in isolation of all main steam lines due to the closure of at least one MSIV in each line. The cause of the Group 1 isolation, however, is not considered to be non-recoverable. Any transient that would result in a non-recoverable isolation would be evaluated in the Loss of Condenser Vacuum Initiator (Section 3.1.2.4). This transient presents a more significant challenge to the reactor coolant makeup system than the turbine trip transient. See Figure 3.1-3.

The MSIV event may result in a plant response that challenges various safety functions. The following is a possible sequence of events:

- When two MSIVs reach 90% open position, a reactor scram signal is initiated (Reactivity Control)
- SRVs cycle open and closed (Primary Pressure Control)
- Low reactor water level initiates HPCI and RCIC (Reactor Coolant Inventory)

3.1.2.3.2 Event Tree Node Descriptions

This tree uses essentially the same function quantifications as the Turbine Trip with Bypass initiator with the following exceptions:

<u>SRVs Open, M(SRVSOPEN)</u>

Reactor pressure vessel pressure control is required for the MSIV Closure initiating event to limit the resultant pressure transient. Since the main condenser is unavailable due to the isolation of the main steam lines, the safety relief valves or safety valves are needed to accomplish RCS pressure control. Failure of a sufficient number of relief valves to open may lead to excessive reactor pressure, potentially creating a LOCA condition.

The design of the safety relief valves is to provide sufficient pressure relief capacity to prevent the primary system pressure from exceeding 110% of the vessel design pressure during the most severe pressurization transient from operating conditions. The success criteria for the SRVs and code safeties during this event requires only three valves to open. This is based on the flow capacity of the relief valves, engineering judgement, plant operating experience, and a review of other PRAs.

The failure of the SRVs to open following the MSIV closure, is assumed to lead to a large LOCA event. Therefore, the event tree branch with the SRV failure is not explicitly analyzed.

Main Condenser Available, Q(MC:AVAIL)

This function is not credited.

Feedwater and Condensate Available, Q(FW:CND)

The Feedwater and Condensate systems will remain available from the start of the event. For long term injection to be successful, the operators must take manual action to establish makeup from the CST to the hotwell.

External Water Available for Low Pressure Injection, V(C:R:G:E)

In order for Condensate to be used as a low pressure injection source, the operators must take manual action to establish makeup from the CST to the hotwell.

3.1.2.4 Loss of Condenser Vacuum Initiator (TC)

3.1.2.4.1 General Description

The loss of condenser vacuum initiator represents malfunctions that result in the partial or complete loss of vacuum in the main condenser sufficient to render the system incapable of performing its function as a heat sink. The main condenser can be used as a makeup source. This transient presents a significant challenge to the reactor coolant makeup system since it is a loss of the normally operating heat sink. See Figure 3.1-4.

The main condenser is assumed to be unavailable for the duration of the transient.

The loss of condenser vacuum event may result in a plant response that challenges various safety functions. The following is a possible sequence of events:

- When two MSIVs reach 90% open position, a reactor scram signal is initiated (Reactivity Control)

- SRVs cycle open and closed (Primary Pressure Control)

- Low reactor water level initiates HPCI and RCIC (Reactor Coolant Inventory)

3.1.2.4.2 Event Tree Node Descriptions

This tree uses essentially the same function quantifications as the Turbine Trip with Bypass initiator with the following exceptions:

<u>SRVs Open, M(SRVSOPEN)</u>

This function is the same as used in MSIV Closure.

<u>Feedwater and Condensate Available, Q(FW:CND)</u>

The Feedwater and Condensate systems will remain available from the start of the event.

For injection to be successful, the operators must take manual action to establish makeup from the CST to the hotwell. In addition, the condensate and feedwater pumps need to be restarted for success.

External Water Available for Low Pressure Injection, V(C:R:G:E)

In order for Condensate to be used as a low pressure injection source, the operators must take manual action to establish makeup from the CST to the hotwell.

3.1.2.5 Loss of River Water Supply (TR)

The loss of river water supply initiator represents malfunctions that result in the loss of service water supplied by the Cedar River. See Figure 3.1-5.

3.1.2.5.1 General Description

This transient initiator is dominated by the common cause loss of all four River Water Supply pumps. This can be caused by a trip of the running pump with a subsequent failure of the remaining pumps to start. Other initiator causes, such as river diversion due to a seismic event, are not included in this analysis.

This transient causes the loss of circulating water makeup (and subsequent loss of main condenser), loss of Emergency Service Water, and loss of Residual Heat Removal Service Water. In order for decay heat to be removed, either the containment vent must properly operate, or AOPs must be followed to ensure that the mechanical vacuum pump maintains the condenser available while well water is maximized as a makeup to the circulating water system.

A failure to SCRAM in this situation would rapidly lead to an overpressurization of the

RCS, and vessel breach. Core damage is assumed for this situation.

The following is a typical plant short term response indicating the various safety functions that are challenged after a loss of river water supply event.

- On a loss of RWS, operators are directed to scram.

- Maintain cooldown via the main condenser as long as possible.

- It is likely that a loss of condenser vacuum initiates a Group 1 isolation. SRVs must open for pressure control.

- High vessel level trips main feedwater pumps (Reactor Coolant Inventory)

- Low vessel level may initiate main steamline isolation and HPCI/RCIC operation (Reactor Coolant Inventory)

3.1.2.5.2 Event Tree Node Descriptions

The structure of this tree is identical to that of a MSIV closure, however there is no river water supply available. The following sections describe the impact of this condition.

<u>Vapor Suppression, D(VAPOR:SUPP)</u>

Since there is no river water available to make up to the Emergency Service Water system, there is no diesel generator backup AC power available in case of failures of the normal offsite AC power source.

The Drywell Spray mode of RHR does not require RHRSW.

Feedwater and Condensate Available, Q(FW:CND)

Since there is no river water available for make up to the circulating water basin, well water must be available so that the condensing function of the main condenser can succeed. This provides a suction source from the hotwell. The hotwell makeup from the CST is the backup to this source of water.

HPCI or RCIC Available, U(H:R)

Since there is no river water available to make up to the Emergency Service Water system, there is no diesel generator backup AC power available in case of failures of the normal offsite AC power source.

The long term HPCI or RCIC function needs AC power to charge the batteries and a suction swap to the torus to succeed.

LPCI Available, V(LPCI)

LPCI is the only low pressure ECCS system available for injection. Core Spray requires ESW to cool its pumps. ESW fails due to the loss of River Water Supply.

Since there is no river water available to make up to the Emergency Service Water system, there is no diesel generator backup AC power available in case of failures of the normal offsite AC power source.

External Water Injection Available (Low Pressure), V(C:G)

Due to a loss of river water, the injection function of RHRSW and ESW can not be successful. Only the Condensate and GSW, with make up to the circulating water pit by well water, systems are available for alternate injection.

PCS Reestablished, Z(PCS:RESTORE)

In order for the main condenser to be used as an ultimate heat sink, condenser vacuum must be maintained. This can be accomplished using the mechanical vacuum pump. In addition, makeup must be provided to the circulating water system to balance losses to the cooling towers. Maximizing Well Water to the circulating water pit provides a success path for this function.

Torus Cooling, W(TCOOL)

The Suppression Pool Cooling mode of RHR requires RHRSW to be provided to the RHR heat exchangers. RHRSW is not available during this transient.

Containment Vent, W(VENT)

Since there is no river water available to make up to the Emergency Service Water system, there is no diesel generator backup AC power available in case of failures of the normal offsite AC power source.

3.1.2.6 Inadvertent Open Relief Valve (TI)

The inadvertent open relief valve initiator represents equipment malfunctions that result in an open relief valve. See Figure 3.1-6.

3.1.2.6.1 General Description

Examples of events that contribute to the inadvertent open relief valve initiator include mechanical, electrical, and human error. This type of initiator challenges, but does not directly disable critical plant safety functions. In fact, it is possible that this type of transient would not necessarily cause a reactor trip. The suppression pool would heat up quickly and require a reactor shutdown.

3.1.2.6.2 Event Tree Node Descriptions

The plant response for this transient is very similar to that of a Turbine Trip with Bypass. The following sections describe the differences in the nodes from the base quantification.

<u>SRVs Reclosed at Low Pressure, P(SRVSCLOSE)</u>

Industry data has shown that 85% of all inadvertent/stuck open relief valves reclose when the reactor pressure is reduced below 200 psid. This node uses a split fraction with a value of 0.15 to indicate the probability that the open relief valve does not reclose.

3.1.2.7 Loss of Instrument Air Initiator (TA)

3.1.2.7.1 General Description

A 1992 modification installed accumulators on the Feedwater regulation equipment. This allows DAEC to survive a short duration (up to 1 hour) loss of instrument air. This initiator includes only loss of air incidents that are not recovered within one hour.

The Loss of Instrument Air initiator represents a challenge to the reactor coolant makeup function. However, it is potentially less severe than isolation events since the main condenser is available to the extent random failures allow it to be available. With the main condenser available, the turbine bypass valves are initially available for primary pressure control. There is high confidence that DAEC can reach a safe shutdown condition irrespective of the availability of instrument air compressors because most of the air loads are backed up by nitrogen or by accumulators.

In addition, accumulators on the feedwater regulation valves provide up to one hour of operation following the loss of air. Condensate does not trip on a loss of air.

For a Loss of Instrument Air initiator, normal makeup to the reactor vessel may be lost or reduced creating conditions that require either an automatic or manual scram.

For the most part, this event is similar to the Loss of Feedwater since a long term (greater than one hour) loss of air causes a feedwater regulation valve lockup. Feedwater would be initially available in this transient, however it is conservatively assumed that feedwater would not be available for injection. See Figure 3.1-7.

3.1.2.7.2 Event Tree Node Descriptions

The event tree for this transient is similar to a Loss of Feedwater. The following sections describe the differences between the Loss of Instrument Air and Loss of Feedwater node quantifications.

Vapor Suppression, D(VAPOR:SUPP)

The alternate supply lineup for River Water Supply requires air to reposition valves. This affects the supply of ESW to the diesel generators, reducing the reliability of the essential AC power busses.

HPCI or RCIC Available, U(H:R)

On a loss of air, many of the HPCI and RCIC turbine auxiliaries lose their back up gas supply. These valves will then rely on nitrogen alone.

Depressurization, X(TIME:RX:DEP)

On a loss of air, a Group 3 islolation signal is generated. There are nitrogen accumulators on these valves, so the impact is small.

Core Spray or LPCI Available, V(CS:LPCI)

The alternate supply lineup for River Water Supply requires air to reposition valves. This affects the supply of ESW to the diesel generators, reducing the reliability of the essential AC power busses.

External Water Available for Low Pressure Injection, V(R:G:E)

The alternate supply lineup for River Water Supply requires air to reposition valves. This affects the supply of ESW to the diesel generators, reducing the reliability of the essential AC power busses. This also reduces the reliability of ESW and RHRSW for injection.

Condensate is not available for low pressure injection due to the lock up of the feedwater regulating valves.

Torus Cooling, W(TCOOL)

The alternate supply lineup for River Water Supply requires air to reposition valves. This affects the supply of ESW to the diesel generators, reducing the reliability of the essential

AC power busses. This also reduces the reliability of RHRSW for cooling the Residual Heat Removal heat exchangers.

Containment Vent, W(VENT)

The alternate supply lineup for River Water Supply requires air to reposition valves. This affects the supply of ESW to the diesel generators, reducing the reliability of the essential AC power busses.

The air supply to the control valves is unaffected because of the oversized accumulators installed in the system.

3.1.2.8 Loss of Division II DC Power Initiator (TDC)

The Loss of a 125V DC Power to a Single Bus initiator represents malfunctions that result in the loss of DC power to Division II systems. At the time of this initiating event, offsite power, the main condenser, and the turbine bypass valves are initially available. See Figure 3.1-8.

3.1.2.8.1 General Description

Examples of events that contribute to the loss of DC power initiator include battery failures, transformer failures, breaker trips, and bus failures. This type of initiator directly disables critical plant safety functions.

The loss of Division II DC bus initiator proceeds much like a Turbine Trip with Bypass. The loss of the DC bus severely degrades the systems available to recover from this transient. The Division II diesel will be unavailable, as will DC control power to many ECCS components. This division was chosen for analysis because it provides control power for the transfer for non-essential power from the main generator to the startup transformer.

The following is a typical plant short term response indicating the various safety functions that are challenged after a turbine trip event.

- Turbine trip initiates rapid closure of the control valves and the main stop valves
- Turbine stop valve closure initiates an RPS signal causing the reactor to scram (Reactivity Control)
- Excess reactor steam initiates opening of the turbine bypass valves (Primary Pressure Control)
- High vessel level trips main feedwater pumps (Reactor Coolant Inventory)
- Low vessel level may initiate main steamline isolation and HPCI/RCIC operation (Reactor Coolant Inventory)

3.1.2.8.2 Event Tree Node Descriptions

Abnormal Operating Procedures instruct the operators to locally, manually close in any stored energy breakers in the event that DC control power is lost. This allows the transfer of non-essential power to the switchyard and the manual initiation of safety systems that are disabled by the initiator. These actions are modeled in each of the quantifications with a single event. All node quantifications use this assumption, and the Turbine Trip with Bypass nodes are modified accordingly.

The following sections describe differences in the quantification of the event tree nodes other than the loss of Division II DC power.

DC Power, E(BOTH:DIVS)

This node is the common cause failure of the opposite DC bus (Division I). Since there are no specific procedures governing the event of loss of all 125V DC, it is conservatively assumed that these sequences lead directly to core damage.

HPCI or RCIC Available, U(H:R)

The Loss of Division II DC eliminates the availability of HPCI as an injection source.

3.1.2.9 Large Break Loss of Coolant Accident (A)

The Large Break LOCA initiator represents malfunctions that result in a break of a large pipe which allows coolant to spill inside or outside containment. The size of the break is such that immediate depressurization of the RPV occurs, and therefore HPCI is incapable of providing makeup. This initiator may be used to include such incidents as inadvertent ADS which creates similar symptoms to a large LOCA in terms of initial RPV level. Unlike the UFSAR Design Basis Accident, at the time of this initiating event offsite power is available. See Figure 3.1-9.

3.1.2.9.1 General Description

Examples of events that contribute to the Large Break LOCA initiator include Main Steam lines, Main Feedwater lines, HPCI/RCIC Steam lines (both unisolated pipe ruptures and turbine casing failures), Interfacing System LOCA, and RPV ruptures. This type of initiator greatly challenges and may directly disable critical plant safety functions. The analysis assumes that only low pressure injection systems will be available to provide injection.

In addition to the systems response, the containment is also severely challenged in this scenario. Adequate containment pressure suppression must be available immediately following a large LOCA.

3.1.2.9.2 Event Tree Node Descriptions

Following a Large LOCA, containment pressure suppression is required, no high pressure systems are credited for injection, and long term containment heat removal is required for the prevention of core damage. The following sections define the nodes that are used in the quantification of this event tree.

Vapor Suppression, D(VAPOR:SUPP)

Since the containment pressure transient occurs quickly after the LOCA, it is assumed that containment sprays will not actuate soon enough to mitigate the pressure spike. Therefore, only the drywell to torus downcomers are available to limit pressure. A single stuck open drywell to torus vacuum breaker will fail this function.

Core Spray or LPCI Available, V(CS:LPCI)

In the event of a Large LOCA, LPCI Loop Select logic will be actuated and required to select the proper recirculation loop for injection. If the logic fails to select a loop or selects the broken loop, the LPCI function will fail.

External Water Injection Available, V(C:R)

Prompt reflood of the reactor vessel is important following a Large LOCA. The lineup for ESW or GSW for low pressure injection takes several minutes to accomplish. These are therefore inadequate to prevent core damage. Condensate and RHRSW, on the other hand, can be aligned from the control room, and are considered viable means of core reflood.

Torus Cooling, W(TCOOL)

This node is quantified the same as in the base case.

Containment Venting, W(VENT)

This node is quantified the same as in the base case.

Injection Post Containment Challenge, QUV(PST:CNT:CHL)

This node is quantified the same as in the base case.

3.1.2.10 Medium Break Loss of Coolant Accident (S1)

The Medium Break LOCA initiator represents malfunctions that result in a break of a large pipe which allows coolant to spill inside or outside containment. The size of the break is such that it is large enough that RCIC is incapable of providing sufficient makeup, but not so large that the reactor will depressurize quickly. This allows HPCI to be used as an injection source. This initiator may be used to include such incidents as inadvertent ADS which creates similar symptoms to a large LOCA in terms of initial RPV level. At the time of this initiating event, offsite power, the main condenser, and the turbine bypass valves are initially available. See Figure 3.1-10.

3.1.2.10.1 General Description

Examples of events that contribute to the Medium Break LOCA initiator include RCIC

Steam lines (both unisolated pipe ruptures and turbine casing failures), Interfacing System LOCA, and RPV ruptures. This type of initiator greatly challenges and may directly disable critical plant safety functions.

3.1.2.10.2 Event Tree Node Descriptions

The flow of this accident will generally follow that of an MSIV Closure. In this accident, the condenser will be available only for a makeup source to Feedwater and Condensate. It will not be considered a valid method of containment heat removal.

The following sections describe the node quantifications for Medium LOCA that are different from the base case.

<u>Vapor Suppression, D(VAPOR:SUPP)</u>

In this case steam must be directed from the drywell airspace into the suppression pool via the drywell to torus downcomers. If the water level in the torus is too low, or there is an open vacuum breaker, this will fail. In the case of a Medium LOCA, drywell sprays are adequate to provide a backup to the vapor suppression function.

<u>Main Condenser Available (Early), Q(MC:AVAIL)</u>

This node considers only the availability of the hotwell as a water source for feedwater. Since the water level in the reactor will rapidly fall in this type of accident, the MSIVs will isolate, rendering the main condenser incapable of removing decay heat.

<u>Feedwater and Condensate Available, Q(FW:CND)</u>

This node is quantified the same as in MSIV Closure.

HPCI or RCIC Available, U(H:R)

This node is quantified with RCIC unavailable. RCIC cannot provide sufficient flow to replace the inventory lost during a Medium LOCA prior to core damage.

External Low Pressure Injection, V(C:R:G:E)

This node is quantified with the main steam lines isolated. This requires that the operators take manual action to provide makeup to the hotwell from the CST in order for condensate to be a successful source of low pressure makeup.

3.1.2.11 Small Break Loss of Coolant Accident Initiator (S2)

The small break LOCA initiator represents malfunctions that result in a piping break greater than the capacity of the Control Rod Drive pumps. This accident progresses identically to the Turbine Trip with Bypass initiator. It is included separately to be consistent with previous probabilistic risk analyses. See Figure 3.1-11.

3.1.2.12 Loss of Offsite Power (LOOP) Initiator

The loss of offsite power initiator dramatically affects both the ECCS systems and the balance of plant (BOP) systems. In addition, the loss of offsite power event challenges the emergency AC power systems to provide AC power. The LOOP initiator and the failure of these systems would result in what is commonly referred to as a station blackout (SBO).

The loss of offsite power and station blackout probabilistic evaluations concentrate on the description of possible sequences which may occur during the operator's attempt to successfully maintain core cooing and containment heat removal while attempting to restore either offsite or emergency AC power and mitigate the circumstances of this scenario. See Figure 3.1-12 (multiple pages).

3.1.2.12.1 General Description of the LOOP/SBO Event Tree Model

The event tree for the loss of offsite power transient initiator depicts the general accident scenarios:

1) Recovery of offsite power,

2) Partial power available to the station from an emergency source, and

3) station blackout.

Six time phases are developed to model the time varying dependencies of SBO scenarios. These six time phases are listed below along with the major dependency used to determine each phase:

- 0 - 1 hour (HPCI/RCIC failure recoverable.)

- 1 hour - 2 hours (allowable operation of HPCI/RCIC without action to mitigate high temperature trips)

- 2 - 4 hours (action to bypass high room temperature trips of HPCI and RCIC required).

- 4 - 8 hours (action to bypass high room temperature trips of HPCI and RCIC required. Action required to extend RCIC battery life beyond 6 hours)

- 8 - 15 hours (action to extend HPCI battery life beyond 12 hours required. Depressurization on Heat Capacity Temperature Limit (HCTL) and actions required to extend HPCI battery life beyond 8 hours.)

- Beyond 15 hours (All DC systems depleted)

The rationale for selecting these time phases is described below:

- 0 - 1 hour: In the event a SBO occurs, the reactor must be tripped and isolated. The SRV's must periodically operate to provide overpressure protection which results in reactor inventory being transferred to the suppression pool. Inventory makeup is therefore required from a high pressure injection source before the reactor water level decreases below an acceptable level. Approximately one hour is available for initiation of high pressure makeup without significant adverse impact on the core if the RPV remains at high pressure. The credited makeup sources are HPCI, RCIC, and main feedwater/condensate systems. If these systems are not available, manual depressurization and LPCI/CS can be used when offsite or emergency AC power is restored at 1 hour.

- 1 - 2 hours: High area temperature (i.e., room environments) may increase the failure frequency of the HPCI and RCIC systems. The DAEC SBO submittal indicates that the HPCI and RCIC rooms reach peak temperatures in approximately 1 hour, although both temperatures are calculated to be below the 175°F trip point. High temperatures in the HPCI and RCIC rooms and steam tunnel may cause an increase in the failure frequency of essential equipment due to thermal effects. For example, without operator action it is calculated that the HPCI room reaches 150°F. Essential equipment in the HPCI room is EQ rated for 148°F at 100% humidity. Although the temperature is exceeded by only a small margin, the potential exists that thermal failure of this equipment may be induced. Therefore, 2 hours was selected as the maximum SBO time duration for which these potential temperature effects would not result in a high temperature degradation of HPCI and RCIC system components.

- <u>2 - 4 hours:</u> As discussed in the previous time phase, the HPCI/RCIC room temperatures gradually increase to a level that is approximately equal to the maximum allowable (qualified) temperature for the HPCI/RCIC components. Extended operation of the system at this evaluated temperature could result in accelerated aging and therefore, a higher probability of age/heat induced component failure. In addition, there is a possibility of a spurious HPCI/RCIC system isolation signal on high room temperature. The SBO analyses show that a notable temperature decrease occurs if the HPCI/RCIC room doors are opened. Therefore, the 2-4 hour time phase was selected to represent the time in which operator action must occur to bypass the high temperature trips on the HPCI/RCIC systems and to perform other actions to augment room ventilation (open doors). It is conservatively assumed that failure of this operator action results in HPCI/RCIC unavailability at 4 hours.

- <u>4 - 8 hours:</u> The HPCI batteries are capable of supporting system operation for at least 8 hours without operator action to reduce system loads. However, the RCIC batteries require load shedding to extend battery capability beyond 6 hours. If operators fail to reduce loads on the RCIC battery within the first two hours of the SBO event results in RCIC system failure at approximately 6 hours. If all high pressure injection is lost at 6 hours, there is sufficient vessel inventory to allow an additional 2 hours of operation without significant adverse effects. Therefore, if offsite or emergency AC power is restored at 8 hours, manual depressurization and LPCI/CS can be used to recover vessel inventory (level).

- 8 - 15 hours: It is expected that containment conditions would approach the HCTL within this time phase, thereby requiring the operating crew to reduce the potential challenge to containment integrity by depressurizing the RPV. Should the operating crew decide to perform an emergency depressurization of the RPV without regard for the operating ECCS, the turbine driven systems could be rendered inoperable, potentially causing core damage if the operators were subsequently unable to restore coolant inventory using the Diesel Powered Fire Pump (DFP). Successful injection with the DFP is questionable when the containment is at high pressure (The RPV pressure cannot be any lower than 50 psi above containment). Therefore, the DFP is not considered a viable injection source.

  RCIC is assumed to be unavailable since battery capacity is insufficient even with load shedding. HPCI will be available for up to 12 hours provided operator action is performed to reduce HPCI battery loads within the first two hours of the SBO event, to bypass high temperature trips, and to augment room ventilation (open doors). If all high pressure injection is lost at 12 hours, there is sufficient vessel inventory to allow an additional 3 hours of operation without significant adverse effects. Therefore, if offsite or emergency AC power is restored at 15 hours, manual depressurization and LPCI/CS can be used to recover vessel inventory (level).

- Beyond 15 Hours: At this time, HPCI and RCIC are unavailable due to battery depletion. The only possible injection source is the diesel-powered fire pump (DFP). However, when battery power is lost, the SRVs can no longer remain open by manual actuation and the RPV will repressurize and begin losing inventory at high pressure. Repressurization of the RPV would preclude makeup from the DFP. Therefore, SBO beyond 15 hours is assumed to result in core damage.

3.1.2.12.2 Event Tree Nodal Descriptions

The following sections describe the differences between the quantification of the nodes for the LOOP initiator and the base case.

Loss of Offsite Power Initiating Event, LOOP

The LQOP initiating event results in the failure to supply the normal AC distribution system with power from offsite sources. The quantification of the LOOP initiator has been derived based on DAEC plant specific experience, and is documented in the DAEC Initiating Event Notebook. The initiating frequency used at DAEC, as cited in the notebook, is 0.117 per reactor year. The time phase split fractions were generated from the NUREG-1032 Cumulative Offsite AC power recovery methodology and the NUREG/CR-1362 Emergency AC power recovery methodology.

Diesel Generators Available, P(DIESELS)

This node represents the probability that both emergency diesel generators fail to start and load upon the receipt of a LOOP signal. If the diesels are successful, the event tree transfers to the base case (Turbine Trip with Bypass) event evaluation.

RCIC, HPCI Operation During Station Blackout, U (H:R)

These top events model operation of RCIC or HPCI systems during the different time phases represented in the station blackout model.

RCIC and HPCI availability during a station blackout is strongly time dependent. This dependency is principally due to the time varying auxiliary system capabilities. The following considerations affect the availability of these injection systems:

- o DC power availability as batteries deplete,
- o Overheating of turbine lube oil caused by elevated suppression pool water temperature if suction from the suppression pool is being used, and
- o Availability of room cooling and containment heat removal requirements to avoid automatic isolation on high room temperature or high turbine exhaust back pressure.

Although these systems are designed to start and run without AC power and are steam driven, they require DC for control power. The DC batteries can supply the HPCI and RCIC load for at least 8 and 6 hours, respectively, without recharging. Additional time may be available for successful battery operation if load shedding is accomplished by the operator or automatically within the first two hours of the event. However, other effects can degrade the performance of these systems. These effects are related to phenomenological conditions that must be circumvented by disabling protective circuitry or by implementing contingency procedures. Therefore, failure to shed DC loads could result in the unavailability of RCIC to support SBO durations longer than 8 hours and unavailability of HPCI to support a 15 hour SBO duration. Failure to bypass isolation circuitry and establish room ventilation could result in the unavailability of both systems to support SBO durations longer than 4 hours.

HPCI is operated in the automatic mode during the entire time of the station blackout. This means it turns off at high RPV level and reinitiates at low RPV level. (High drywell pressure initiation signal is assumed not to reinitiate HPCI when the high level signal is removed.) This operating philosophy results in the HPCI system cycling on and off. These restarts (13 restarts are calculated using MAAP) would result in a conditional failure probability over each time phase that is accounted for in the quantification process.

Success at this node requires that either HPCI or RCIC has initiated and is injecting into the vessel.

Feedwater and Condensate Recovered, Q (FW:CND:RECOV)

This node represents the recovery of feedwater and condensate as a backup to HPCI and RCIC system failure. This node is credited only if offsite AC power is restored no later than 30 min. After 30 min., recovery of feedwater and condensate is not considered credible. This node is the same as that used in the MSIV Closure event tree.

RPV Depressurization, X(TIME:RX:DEP)

In the event that the high pressure systems are unavailable to maintain adequate coolant inventory, the RPV can be depressurized to allow the use of low pressure coolant injection systems. The principal method of depressurizing the RPV is by manually opening the SRVs.

Success at this node requires that the operator has manually initiated emergency blowdown to depressurize the RPV. Failure at node X means that the RPV remains pressurized even though all high pressure injection systems may be unavailable.

In the cases where offsite or emergency AC power is recovered, the fault tree quantification, is similar to that developed for the base case. The ability of the operator to depressurize the RPV based on water level indication during a station blackout event is considered less reliable, since the ADS and BOP systems are unavailable. Additionally, it is conservatively assumed that the HPCI and RCIC systems are incapable of depressurizing the RPV.

Operator Depressurizes the RPV Upon HCTL, X(HCTL:DEP)

The EOPs allow the operator to maintain RPV pressure less than the HCTL by manually controlling system pressure. However, if the operator fails to maintain RPV pressure within the prescribed limit, emergency blowdown is required.

Success at this node requires that the operator manually control RPV pressure less than HCTL, or perform an emergency blowdown, but maintain sufficient RPV pressure for HPCI/RCIC operation. This action is assumed to cause minimal effect on high pressure ECCS maintaining coolant inventory. Failure at node "X" means that the operator is unable to control RPV depressurization, so as to render HPCI and RCIC inoperable. Conversely, success at this node means that the operator is maintaining the RPV and containment conditions within the HCTL.

Core Spray or LPCI Available, V(CS:LPCI)

In the cases where offsite AC power has been restored, this node is quantified using the base case assumptions. In the cases in which Emergency AC Power is recovered, it is assumed that only one division of power has been recovered. It is further assumed that the recovered diesel will not subsequently fail. The quantification of the cumulative recovery probabilities address the multiple failures of a single diesel generator.

External Water Injection, V(C:R:G:E)

In the cases where offsite AC power has been restored, this node is quantified using the base case assumptions. In the cases in which Emergency AC Power is recovered, it is assumed that only one division of power has been recovered. It is further assumed that the recovered diesel will not subsequently fail. The quantification of the cumulative recovery probabilities address the multiple failures of a single diesel generator.

Operators Bypass High Temperature Trips, TR (OP:BYP:TRIP)

During station blackout conditions, forced room cooling and ventilation is unavailable in HPCI and RCIC system equipment rooms. The loss of room cooling to these areas pose two problems to the operating crew:

1) The automatic trip circuitry that isolates HPCI and RCIC in the case of a steam line break can actuate upon high room temperature; and

2) Equipment damage can result from thermal effects.

Either of these failure modes can disable both HPCI and RCIC.

This event node accounts for the possibility that the operating crew accomplishes two actions to prevent HPCI/RCIC system failure from these effects. The first action requires the operator to defeat the high room temperature trip circuitry for both systems. The follow-on action is to provide natural circulation ventilation to the equipment rooms and maintain sufficient ventilation to these areas to facilitate equipment operation by opening the room doors.

Success at this node requires the operating crew to accomplish both actions within 4 hours from event initiation. Failure to accomplish these actions is assumed to result in the loss of both high pressure systems, HPCI and RCIC, at 4 hours.

Operators or Automatic Systems Load Shed DC Power Supplies Within 2 Hours, SH(OP:LD:SHED)

The DC batteries can provide a reliable source of power to their respective buses during a station blackout event. Engineering evaluations by IELP indicate that the HPCI & RCIC batteries can survive 8 and 6 hours, respectively, without load shedding. However, the HPCI and RCIC batteries can survive at least 12 and 8 hours, respectively, with load shedding. Therefore, the SBO event tree model credits the HPCI batteries for providing sufficient DC power to vital equipment beyond 8 hours if the operators or automatic systems can conserve DC power by shedding non-essential DC loads with the first 2 hours of the event.

Successful implementation of this procedure allows the HPCI and RCIC systems to remain functional for up to 12 and 8 hours, respectively. Failure to conserve DC power is assumed to result in the depletion of the HPCI and RCIC 125V DC station batteries at 8 and 6 hours, respectively.

Main Condenser Available, Z(PCS:RECOV)

This node is only credited in the time phases in which Offsite AC power has been recovered. It is quantified as in the base case.

Torus Cooling, W(TCOOL)

In the cases where offsite AC power has been restored, this node is quantified using the base case assumptions. In the cases in which Emergency AC Power is recovered, it is assumed that only one division of power has been recovered. It is further assumed that the recovered diesel will not subsequently fail. The quantification of the cumulative recovery probabilities address the multiple failures of a single diesel generator.

Containment Venting, W(VENT)

In the cases where offsite AC power has been restored, this node is quantified using the base case assumptions. In the cases in which Emergency AC Power is recovered, it is assumed that only one division of power has been recovered. It is further assumed that the recovered diesel will not subsequently fail. The quantification of the cumulative recovery probabilities address the multiple failures of a single diesel generator.

Injection Post Containment Challenge, QUV(PST:CNT:CHL)

This node is not credited in the Loss of Offsite Power event analysis.

3.1.2.13 Turbine Trip with Bypass ATWS (TTC)

The turbine trip ATWS represents malfunctions that result in manual or automatic trip of the main turbine with failure to achieve scram of the reactor. At the time of this initiating event, offsite power, the main condenser, and the turbine bypass valves are initially available. See Figure 3.1-13 (multiple pages).

3.1.2.13.1 General Description

Examples of events that contribute to the turbine trip initiator include electric load rejection, spurious reactor trips, low feedwater flow, trip of a recirculation pump, or inadvertent turbine control valve closure. This type of initiator challenges, but does not directly disable critical plant safety functions.

3.1.2.13.2 Event Tree Node Descriptions

The following sections describe the differences between the quantification cf the fault trees for the ATWS cases and the base case.

SCRAM, C(SCRAM:MECH), C(SCRAM:ELECT), C(RECIRC:TRIP), C(ARI)

The ATWS trees evaluate a more detailed model of the reactivity control function than do the non-ATWS trees. This function is split into four sections: mechanical failures, electrical failures, recirculation pump trip failures, and ARI failures. The mechanical and electrical SCRAM failures rates are the same as those used in the base case evaluation.

The recirculation pump trip node satisfies the reactivity control safety function if the recirculation flow in the reactor can be reduced sufficiently to increase the void fraction in the core, thereby introducing negative reactivity. It has an automatic actuation, so its response time is acceptable for the SCRAM function.

The Alternate Rod Insertion (ARI) function satisfies the reactivity control safety function if the ARI system can introduce an alternate signal to insert control rods into the reactor core. This is a manual action, so its response time is considered to be slower than the automatic SCRAMs. It will only be considered following a successful recirculation pump trip. Also, the ARI signal will not satisfy the SCRAM function if the failure to insert control rods is due to mechanical failures.

SRVs Open, M(SRVS:OPEN)

Following an ATWS event with a successful recirculation pump trip, reactor power can be as high as 50%. The primary method of RPV pressure control following a reactor trip is through the Turbine Bypass Valves (TBVs) into the main condenser. The TBVs at DAEC have a capacity of 25% steam flow. It is therefore necessary to augment this capacity by steaming to the suppression pool through the SRVs.

Feedwater Runback, Q(FW:RUNBACK)

In ATWS scenarios with the condenser unavailable, the feedwater pumps are required to run-back in order to reduce reactor power to a level below the capacity of the SRVs. In the TTC case, the main condenser is initially available, so this split fraction is assigned 0.0.

Standby Liquid Control Initiation, C(SLC:EARLY), C(SLC:LATE)

This function satisfies the manual reactivity control function by introducing a solution of borated water in sufficient quantity and concentration to the core to bring reactor power to less than 4%.

For early SLC to be successful, the operators must begin injection within 6 minutes of the transient. Success at this node implies that the suppression pool heatup rate will be slow enough to allow venting to be a viable means of containment heat removal.

The late injection of SLC is successful if the operators begin to inject within 40 minutes of the transient if the condenser is available, or 20 minutes if the condenser is not available. The suppression pool heatup rate is sufficient that 2 trains of RHR in torus cooling mode are required for containment heat removal.

Main Condenser Available, Q(MC:AVAIL)

The use of this node is an enhancement to the generalized event tree to demonstrate the DAEC capability to provide high pressure coolant injection (Reactor coolant inventory function) using motor-driven feedwater pumps.

This node identifies the ability of the main condenser to provide a makeup source and an external heat sink in response to a turbine trip initiating event. For success at this node the MSIVs in one of the four main steam lines must remain open (or be reopened following the initiating transient) and at least one of the main condenser circulating water pumps must be operating delivering cooling water to the main condenser.

Success at this node indicates that the main condenser is available as a heat removal source and may be used to complete the steam conversion process if the feedwater/condensate is operable. Failure at this node indicates that the main condenser is not available as a heat removal source, but the feedwater/condensate system may still be considered for coolant injection.

Feedwater/Condensate High Pressure Injection Available, Q(FW:CND)

The main difference between the quantification of this node for ATWS and the base case is that the operators do not have the time to perform manual actions outside the control room. Each of these operator actions are conservatively set to TRUE.

HPCI High Pressure Injection Available, U(HPCI)

In the ATWS cases, RCIC does not have the flow capacity to provide adequate makeup to the reactor vessel. The Loss of Offsite Power event tree quantification provides a node for HPCI alone.

Operator Inhibits ADS, X(ADS:INHIBIT)

The operator normally will inhibit the ADS system from actuating automatically. This function satisfies the inventory control function.

Operator Bypasses HPCI Trips, TR(OP:BYP:TRIP)

As in the Loss of Offsite Power cases, the long term use of HPCI will result in a heatup of the HPCI room and the steam tunnel. There is a likelihood that the automatic high temperature trips for HPCI will operate during the ATWS event. This node models the operators successfully bypassing those trips to ensure long term injection.

CS or LPCI Low Pressure Injection Available, V(CS:LPCI)

The main difference between the quantification of this node for ATWS and the base case is that the operators do not have the time to perform manual actions outside the control room. Each of these operator actions are conservatively set to TRUE.

Alternate External injection is not considered in ATWS cases due to the length of time required to set up the alignment.

RPV Level Not Controlled to High, L(LEVEL:NOT:HI)

When the high capacity low pressure injection systems are used for inventory control, it is necessary to ensure that reactor level is not maintained too high. This can cause insertion of positive reactivity due to the injection of cold water. It can also provide a means of diluting the boron concentration in the reactor coolant. It is conservatively assumed to lead to a re-criticality of the core if this node fails.

This node is only considered in the cases in which Core Spray or LPCI provide makeup to the core. These systems are high capacity and do not have automatic high vessel level trips.

Adequate Level/Power Control, L(CONTROLLED)

This function models the operators following procedures and adequately controlling reactor power with level. Failure at this node implies that the operators have lost control of reactor power. It conservatively leads to a re-criticality of the core.

RPV Level Above One-Third Core Height, L(LEVEL:NOT:LOW)

If level is maintained too low, steam cooling of the upper portions of the core will not be adequate. This node models the operators failure to follow procedures and maintain water level at a reasonable level.

Torus Cooling, W(TCOOL)

The RHR system is credited with providing torus cooling as well as other containment heat removal methods. Success at this node requires a complete path from and to the containment through at least one RHR heat exchanger. In addition, the RHR service water system must provide cooling water to the operating RHR heat exchanger. Two RHRSW pumps are required to adequately remove containment heat.

3.1.2.14 Loss of Feedwater ATWS (TFC)

The loss of feedwater ATWS represents malfunctions that result in loss of one or more trains of the feedwater system with failure to achieve SCRAM of the reactor. At the time of the initiating event offsite power, the main condenser, and the turbine bypass valves are initially available. See Figure 3.1-14 (multiple pages).

3.1.2.14.1 General Description

This type of accident is similar to the Turbine Trip ATWS described above, except that feedwater must be recovered in order to be a viable source of high pressure injection. In this case, the feedwater pumps do not need to "run back" in order to control reactor power.

3.1.2.14.2 Event Tree Node Descriptions

The following sections describe the differences between the quantification of the fault trees for the Loss of Feedwater ATWS and the Turbine Trip with Bypass ATWS cases.

Feedwater Runback, Q(FW:RUNBACK)

The Feedwater Runback node is not necessary in this tree. Since feedwater is initially unavailable, it is unable to inject sufficient cold water to cause an increase in reactivity.

Main Condenser Available, Q(MC:AVAIL)

Main feedwater is initially unavailable. It must be recovered in order to be used to return water from the hotwell to the reactor vessel. This is the same quantification as used in the Loss of Feedwater transient tree.

Feedwater and Condensate Available, Q(FW:RECOV)

Main feedwater is initially unavailable. It must be recovered in order to be used to return water from the hotwell to the reactor vessel. This is the same quantification as used in the Loss of Feedwater transient tree.

3.1.2.15 MSIV Closure ATWS (TMC)

The MSIV ATWS represents malfunctions that result in Group 1 isolation, but not so severe that the main condenser cannot be recovered as a heat removal system, with failure to achieve SCRAM of the reactor. At the time of the initiating event offsite power and feedwater are initially available. See Figure 3.1-15 (multiple pages).

3.1.2.15.1 General Description

This type of accident is similar to the Turbine Trip ATWS described above, except that the main steam lines must be reopened in order to use the condenser as a heat sink.

3.1.2.15.2 Event Tree Node Descriptions

The following sections describe the differences between the quantification of the fault trees for the MSIV Closure ATWS and the Turbine Trip with Bypass ATWS cases.

Feedwater Runback, Q(FW:RUNBACK)

The Feedwater Runback node is very important in cases in which the main steam lines are isolated. If this function fails, the pressure transient along with the continued addition of cold water can cause overpressurization of the reactor coolant system. It is conservatively assumed that a failure of feedwater runback will cause a large RCS breach, and lead directly to core damage.

Main Condenser Available, Q(MC:AVAIL)

In this case, the main condenser needs to be recovered. The quantification of this node is the same a the Z(MC:RECOV) node used in the base case.

3.1.2.16 Loss of Condenser Vacuum ATWS (TCC)

The MSIV ATWS represents malfunctions that result in a loss of the main condenser, with failure to achieve SCRAM of the reactor. These are so severe that the main condenser cannot be recovered as a heat removal system. At the time of the initiating event offsite power and feedwater are initially available. See Figure 3.1-16 (multiple pages).

3.1.2.16.1 General Description

This type of accident is similar to the Turbine Trip ATWS described above, except that the main condenser is not a viable heat sink.

3.1.2.16.2 Event Tree Node Descriptions

The following sections describe the differences between the quantification of the fault trees for the Loss of Condenser Vacuum ATWS and the Turbine Trip with Bypass ATWS cases.

Feedwater Runback, Q(FW:RUNBACK)

The Feedwater Runback node is very important in cases in which the main steam lines are isolated. If this function fails, the pressure transient along with the continued addition of cold water can cause overpressurization of the reactor coolant system. It is conservatively assumed that a failure of feedwater runback will cause a large RCS breach, and lead directly to core damage.

Main Condenser Available, Q(MC:AVAIL)

The main condenser is not available in this accident.

3.1.2.17 Other ATWS Events

The probability of an Anticipated Transient Without Scram event is of the order 1E-5. When this conditional probability is multiplied by the individual initiating events discussed previously in Section 3.1.2, the probabilities of the individual ATWS events was determined to be insignificant for all events except Turbine Trip, Loss of Feedwater, MSIV Closure, Loss of Condenser Vacuum, and Loss of Offsite Power. However, the probability of the Loss of Offsite Power initiator when multiplied by the conditional unavailability of diesel generators yielded a probability so low as to warrant no investigation. This leaves the four ATWS events discussed previously. (See Subsections 13, 14, 15, and 16.)

3.1.2.18 Interfacing System Loca (V)

This type of accident involves a breach of the pressure boundary between the high pressure RCS and the low pressure ECCS systems. The pressurization of these systems can result in a leak or catastrophic failure of the low pressure piping. These are treated as large loss of coolant accidents with containment bypass. See Figure 3.1-17.

3.1.2.18.1 General Description

There are three types of failures that can lead to an ISLOCA. These are overpressurization of the Core Spray Inject lines, the RHR Inject lines, or the Shutdown Cooling Suction lines. The inject line failures have isolation valves that are capable of stopping the LOCA, while the suction line does not.

In these accidents, it is conservatively assumed that the line break occurs in one of the ECCS corner rooms and disables all of the equipment in that room. This is due to environmental considerations. It is also assumed that the effects of a leak in the piping, which is much more likely than a rupture, will have the same effects as the rupture. The train of the broken system is also assumed to be unavailable to perform its intended function.

3.1.2.18.2 Event Tree Node Descriptions

This accident progresses much the same way as a Large LOCA. The containment is bypassed in these accidents. The following sections describe the differences between the ISLOCA node quantification and the Large LOCA case.

Core Spray or LPCI Available, V(CS:LPCI)

In the cases in which the break is in a core spray line, it is assumed that the equipment in the same room as the affected core spray train is failed. In the cases that involve a RHR inject line, the equipment in the associated room is failed, in addition to the opposite train of RHR injection. This is due to the normally open RHR cross-tie line. If the break is in a SDC suction line, it is assumed to break in one of the corner rooms. The equipment in that room is assumed to be disabled.

External Water Available for Alternate Injection, V(C:R:G:E)

The equipment in the room associated with the line break is assumed to be unavailable. This affects one train of alternate injection. The condensate function is not affected.

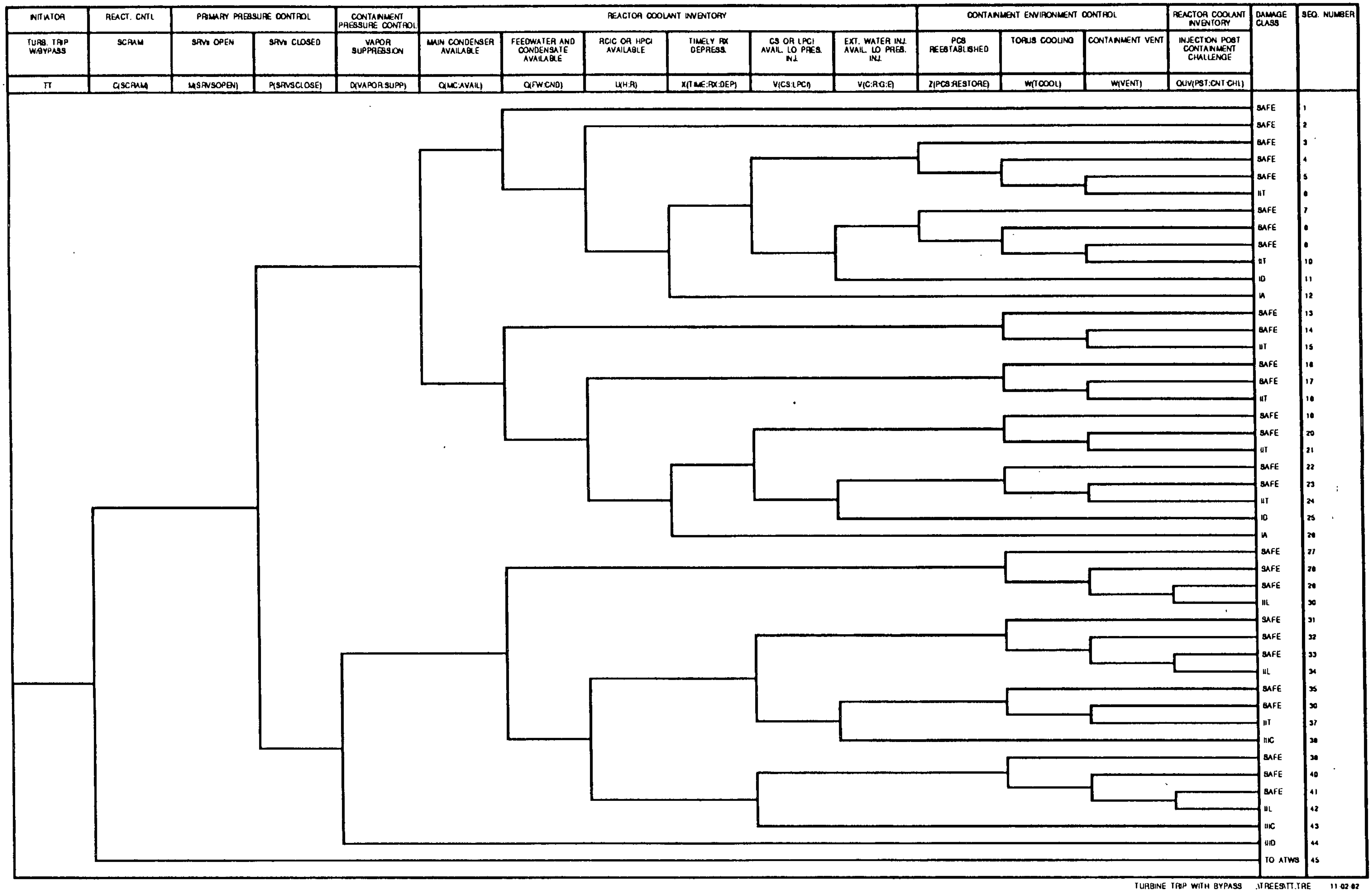


Figure 3.1-1 Turbine Trip with Bypass Event Tree

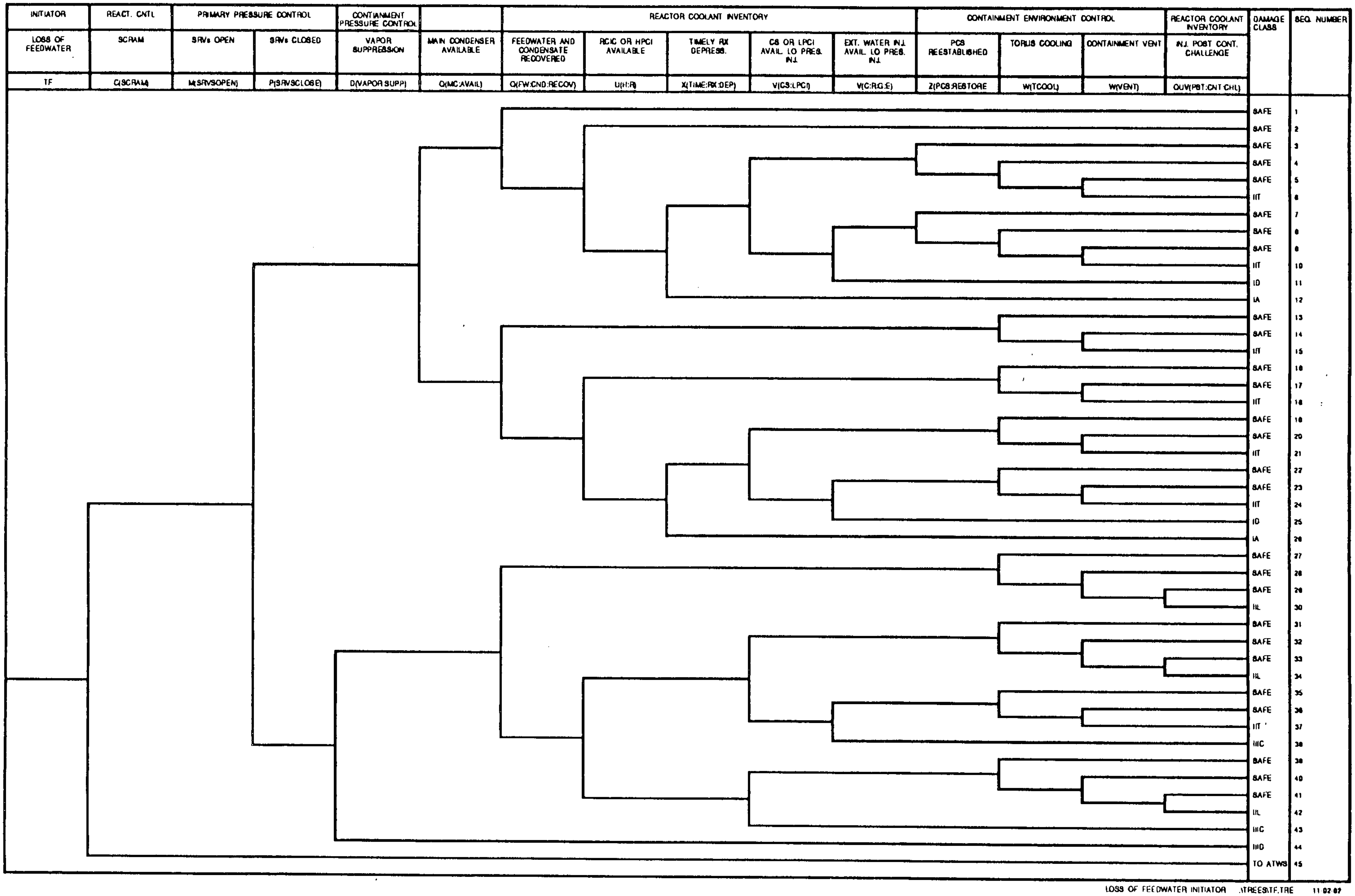


Figure 3.1-2 Loss of Feedwater Event Tree

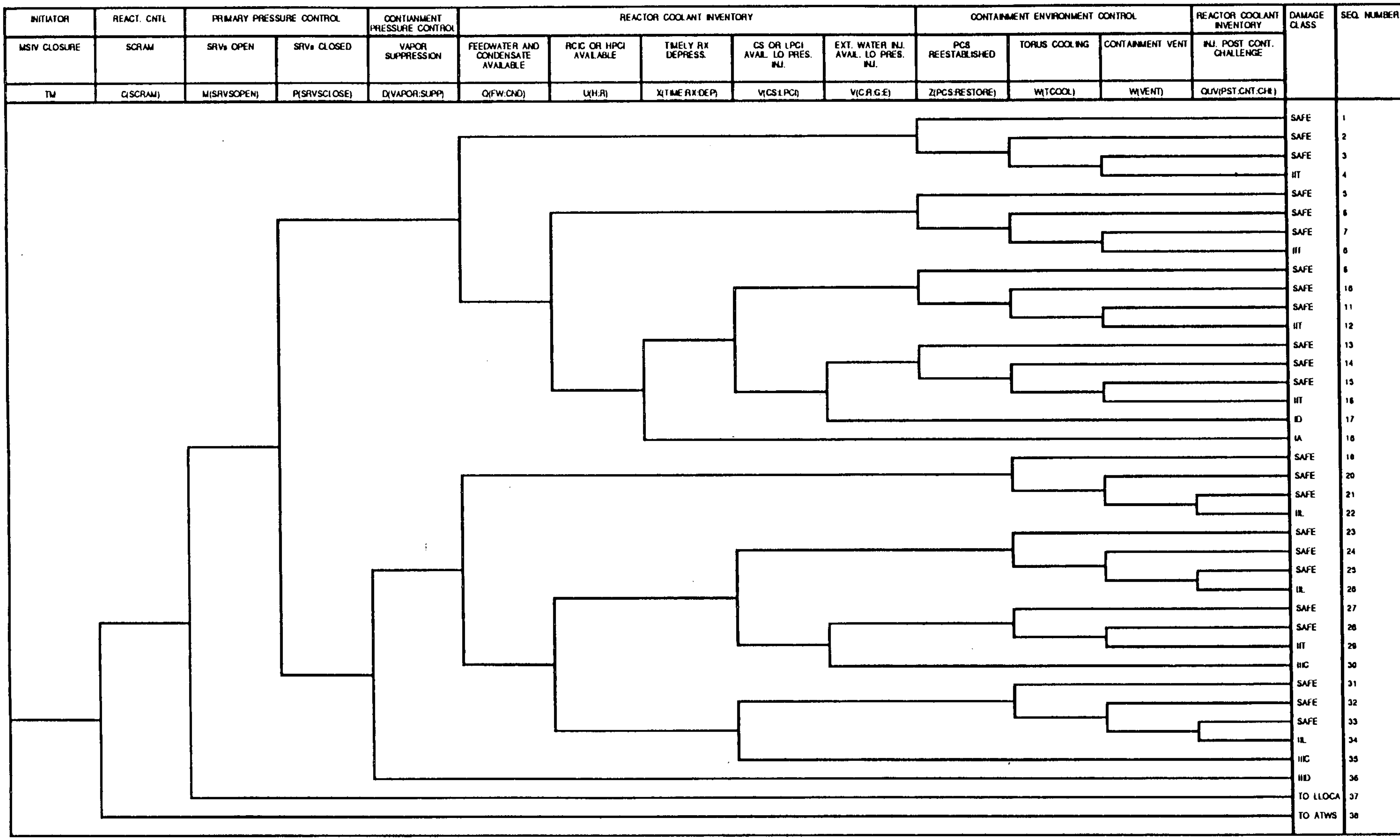


Figure 3.1-3 MSIV Closure Event Tree

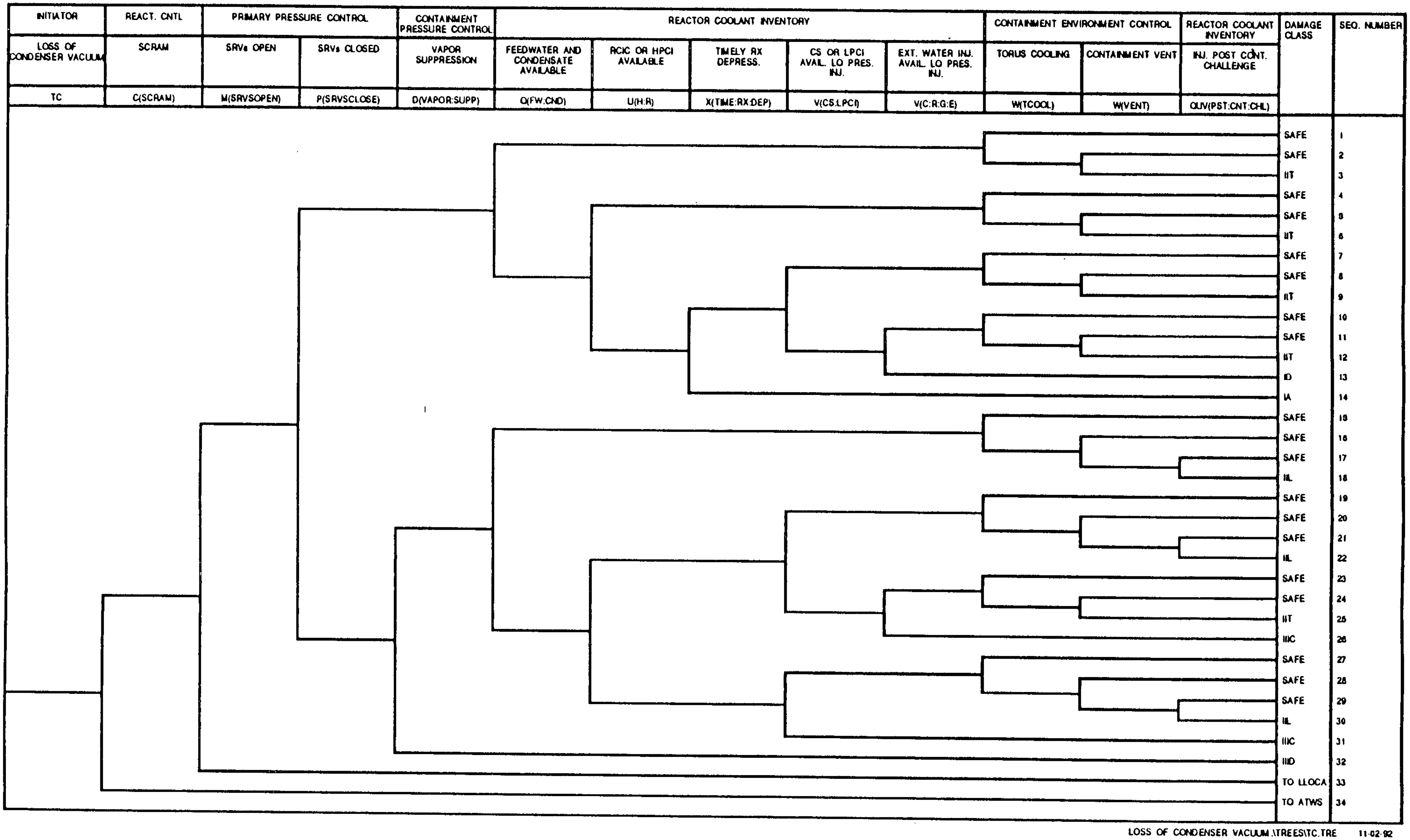


Figure 3.1-4 Loss of Condenser Vacuum Event Tree

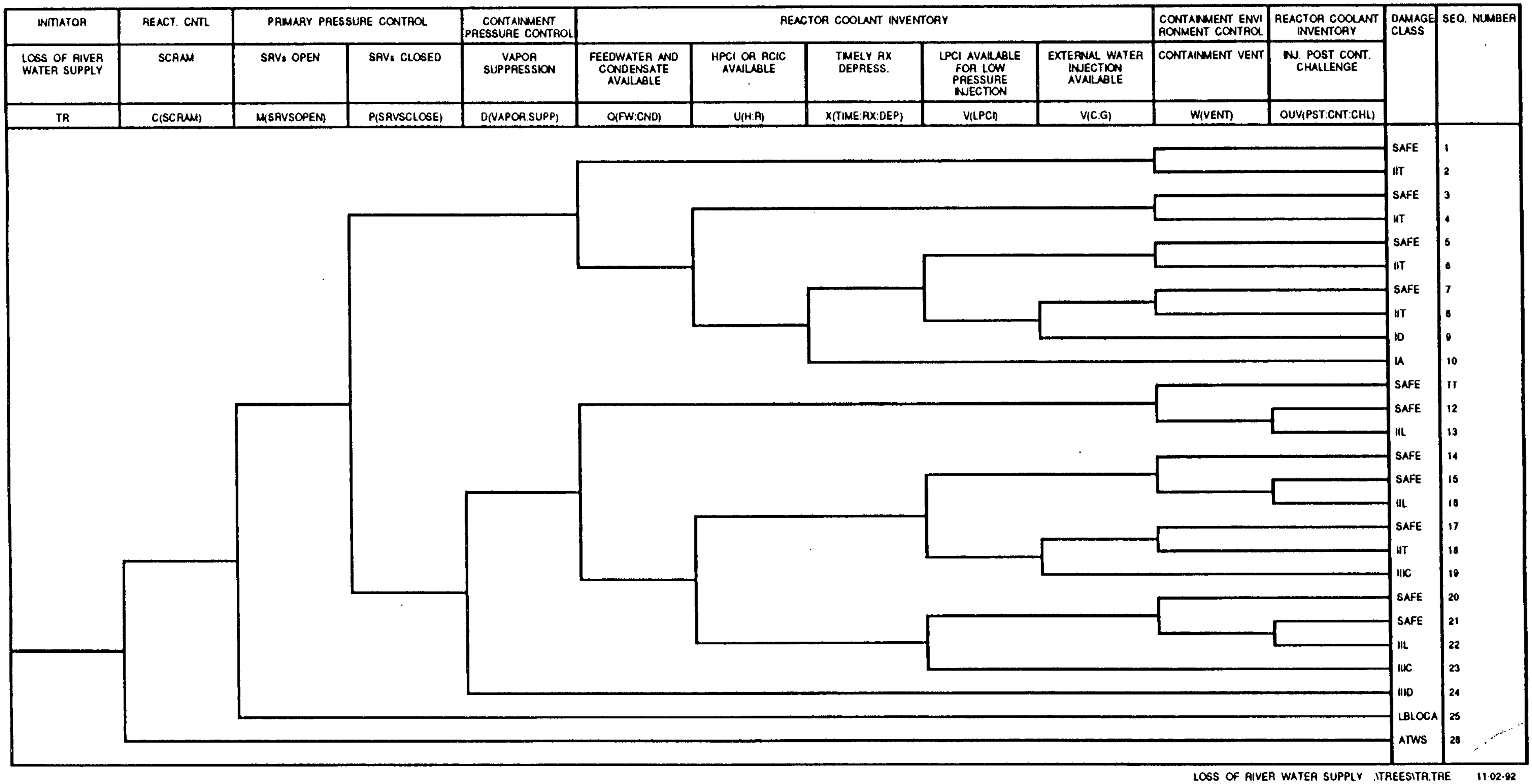


Figure 3.1-5 Loss of River Water Supply Event Tree

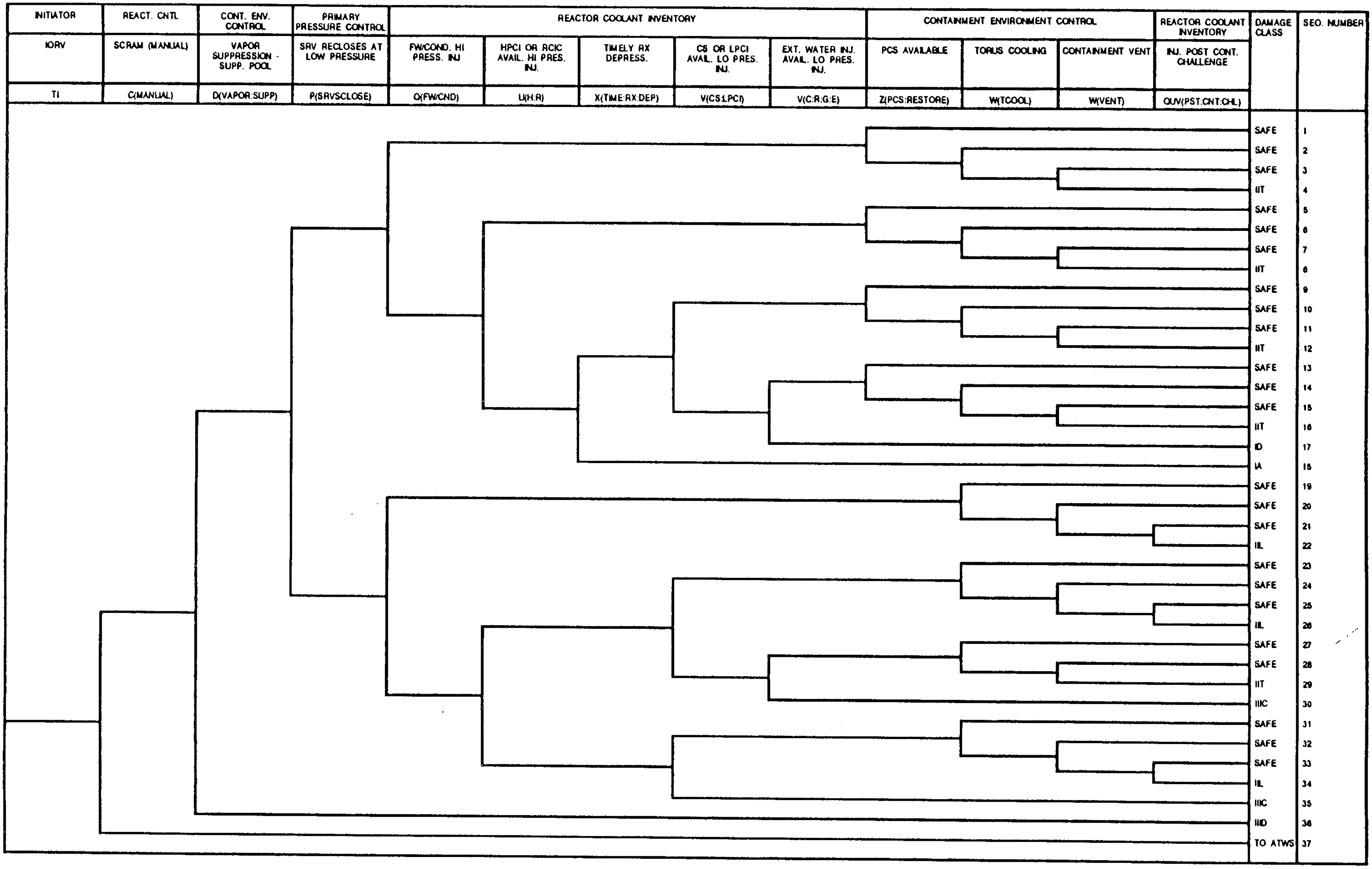


Figure 3.1-6 Inadvertent Open Relief Valve Event Tree

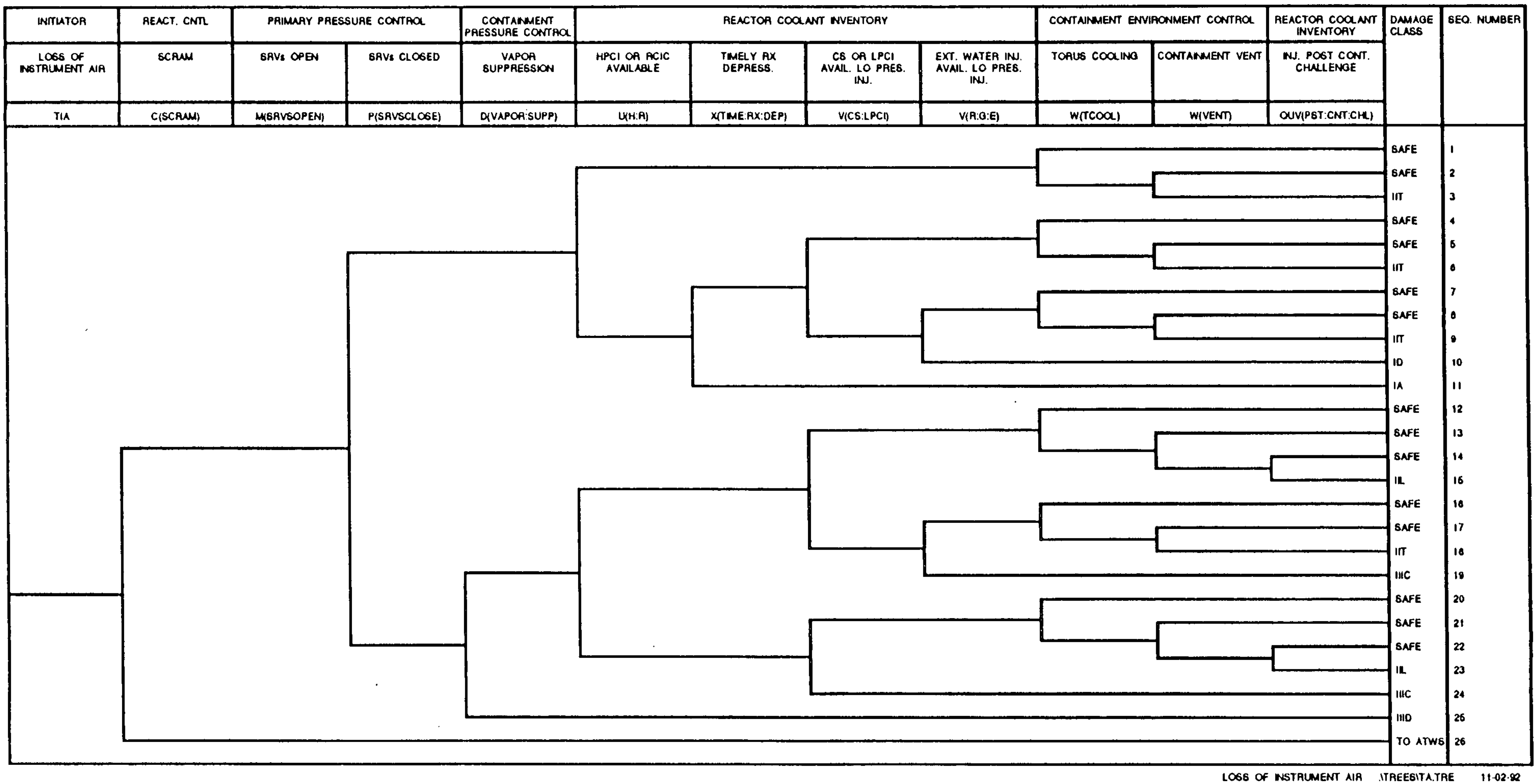


Figure 3.1-7 Loss of Instrument Air Event Tree

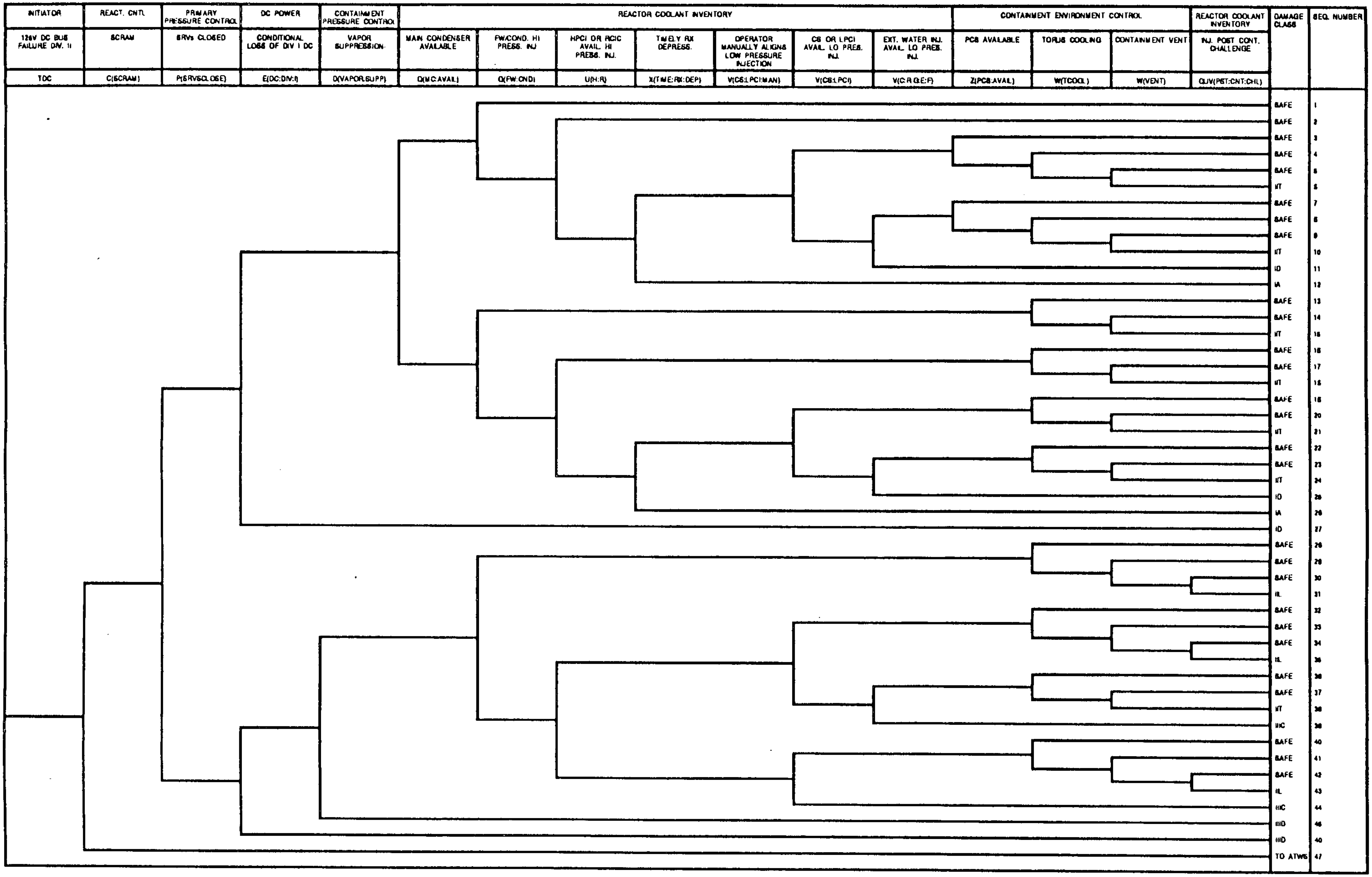
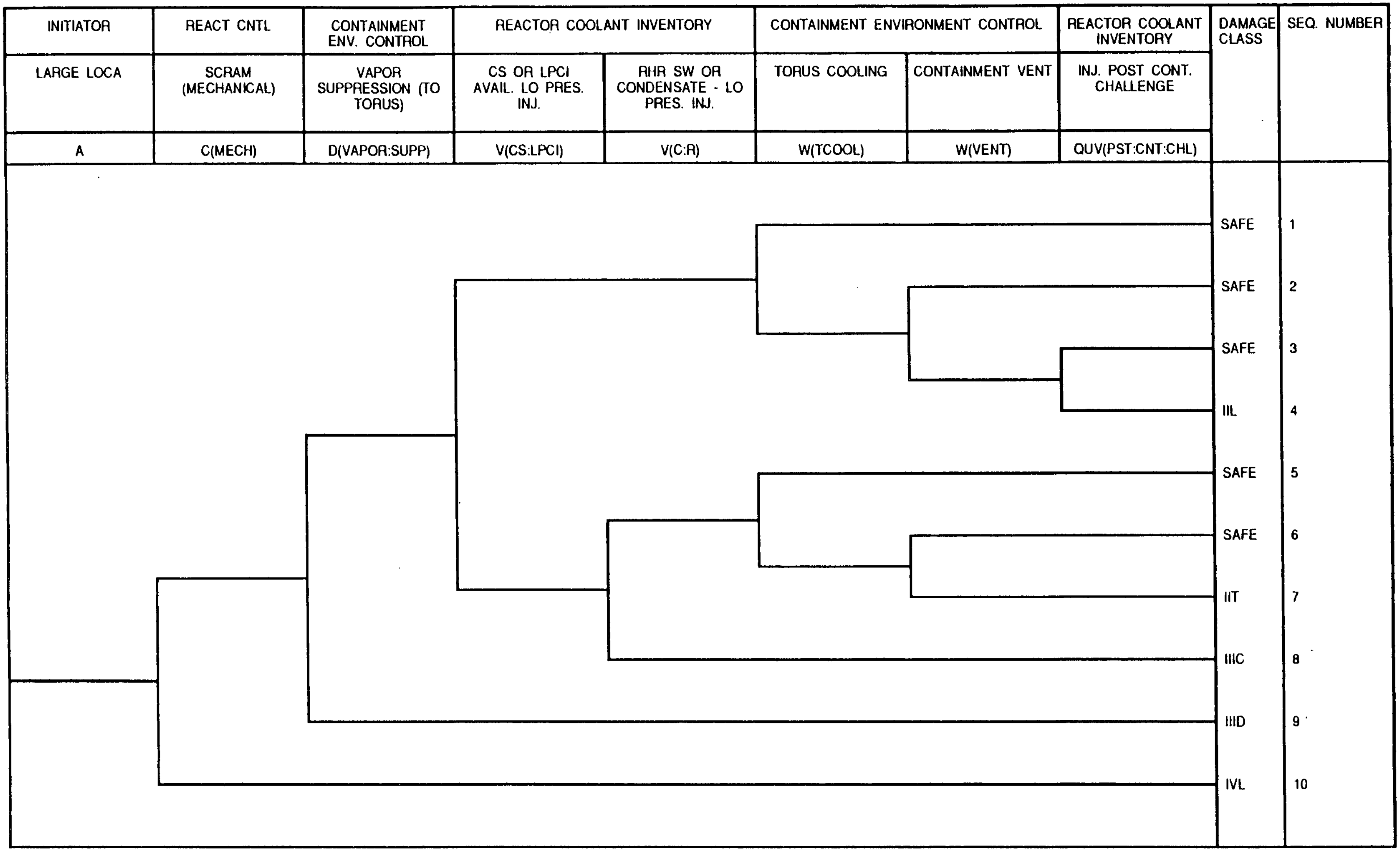
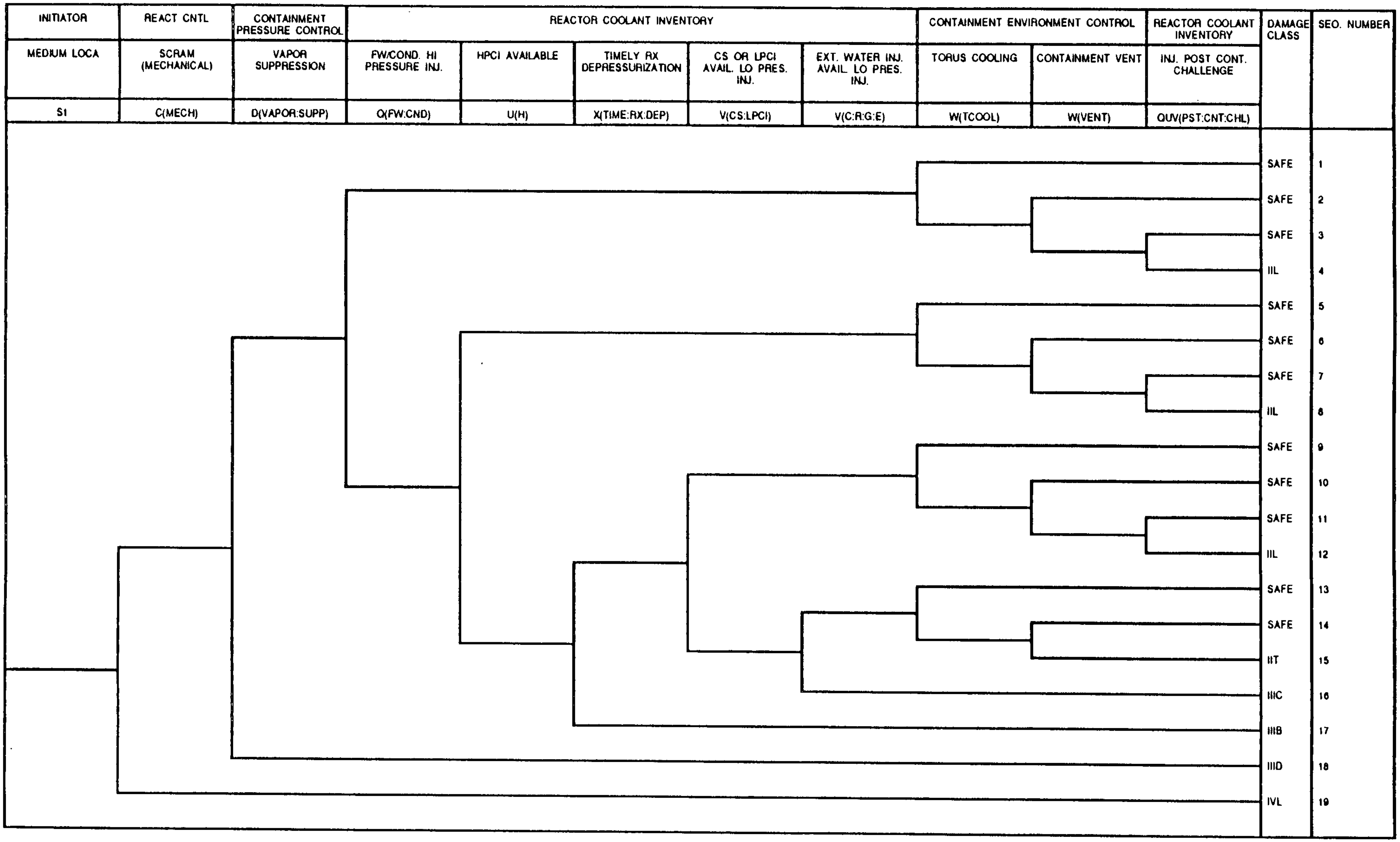


Figure 3.1-8 125 V DC Buss Failure (Division II) Event Tree



LARGE LOCA .\TREES\A.TRE 11-02-92

Figure 3.1-9 Large LOCA Event Tree



MEDIUM LOCA \TREES\S1.TRE 11-02-92

Figure 3.1-10 Medium LOCA Event Tree

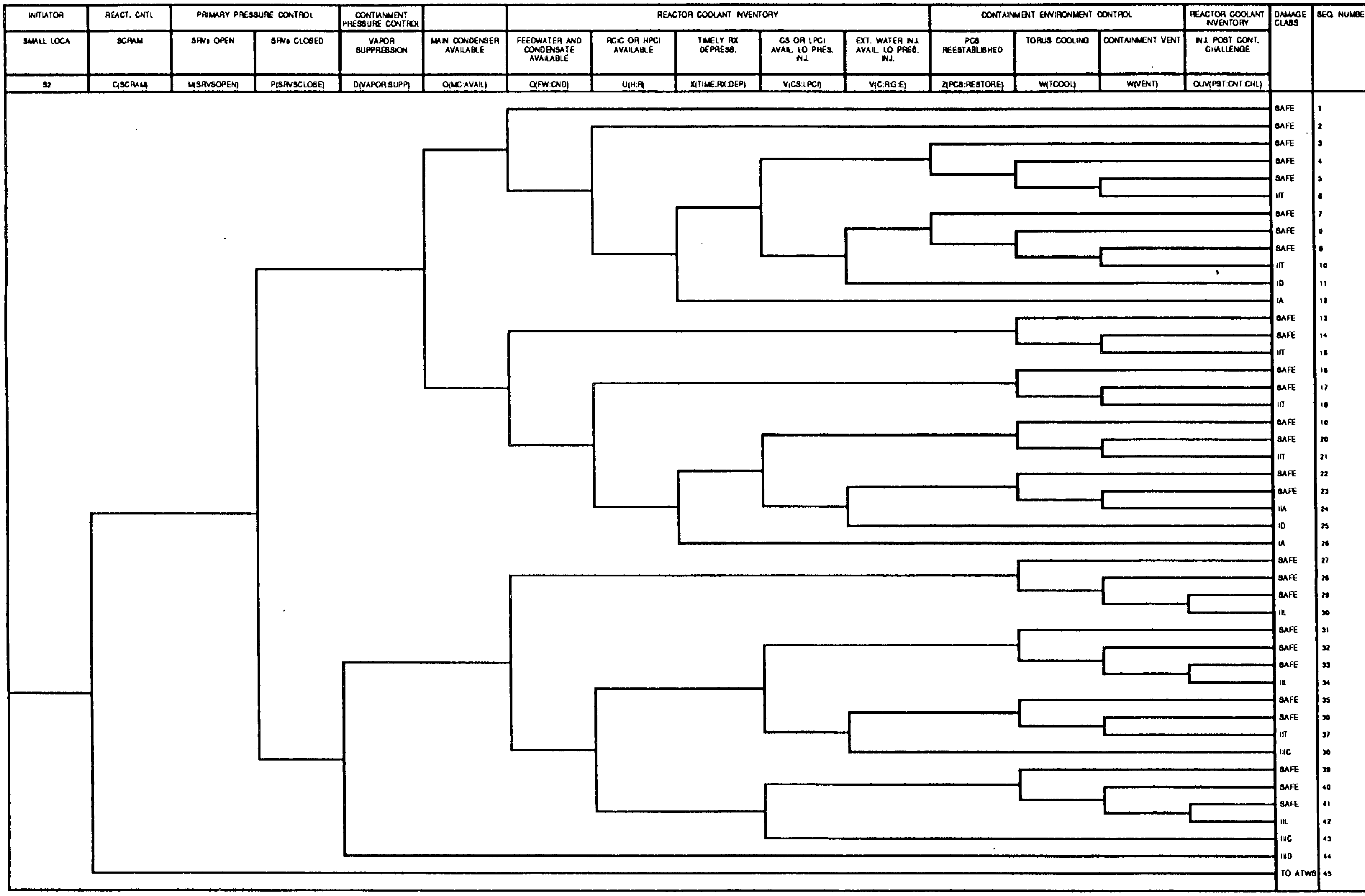
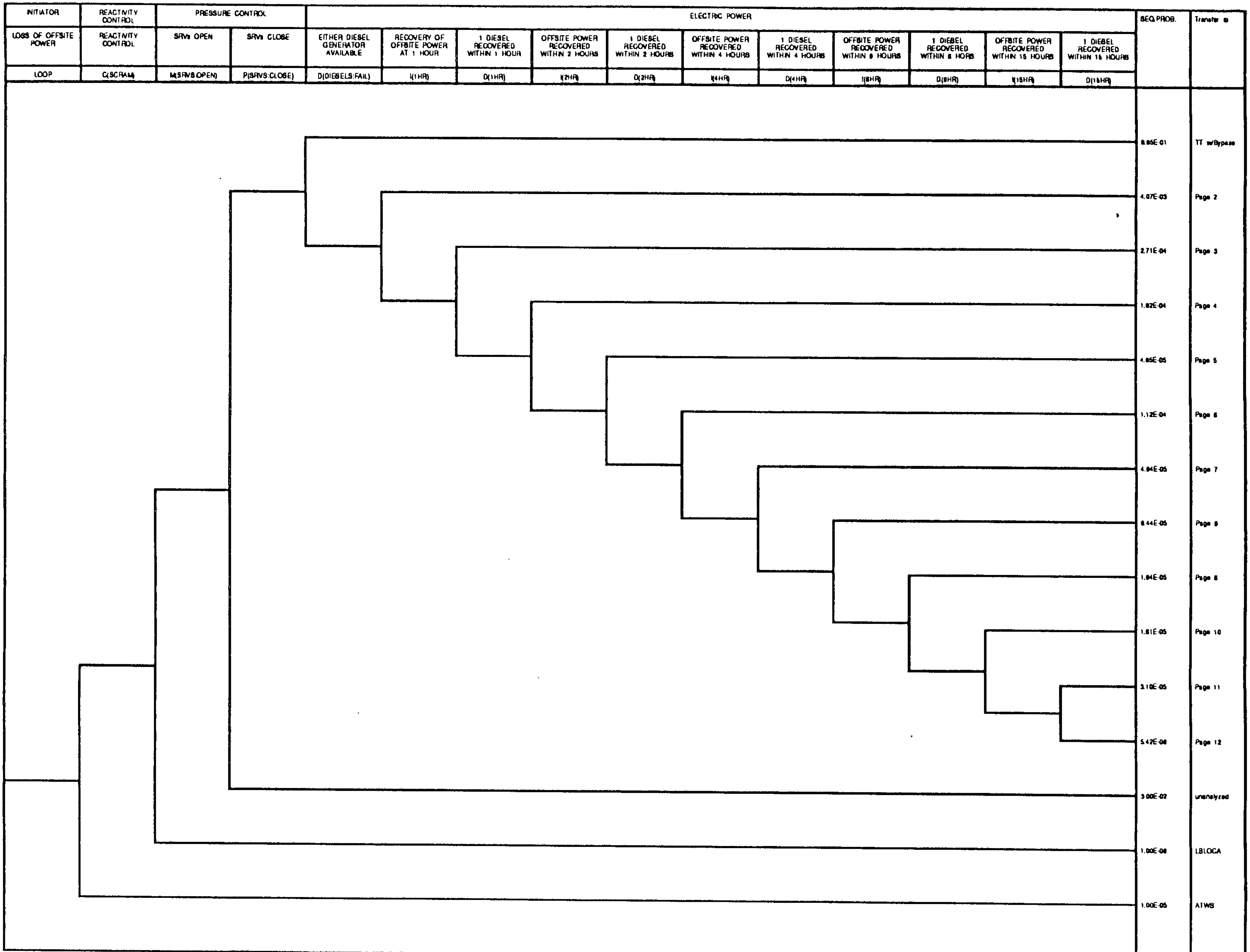


Figure 3.1-11 Small LOCA Event Tree



Loss of Offsite Power Page 1 .\TREES\LOSP1.TRE 11-02-92

Figure 3.1-12 Loss of Offsite Power Event Tree

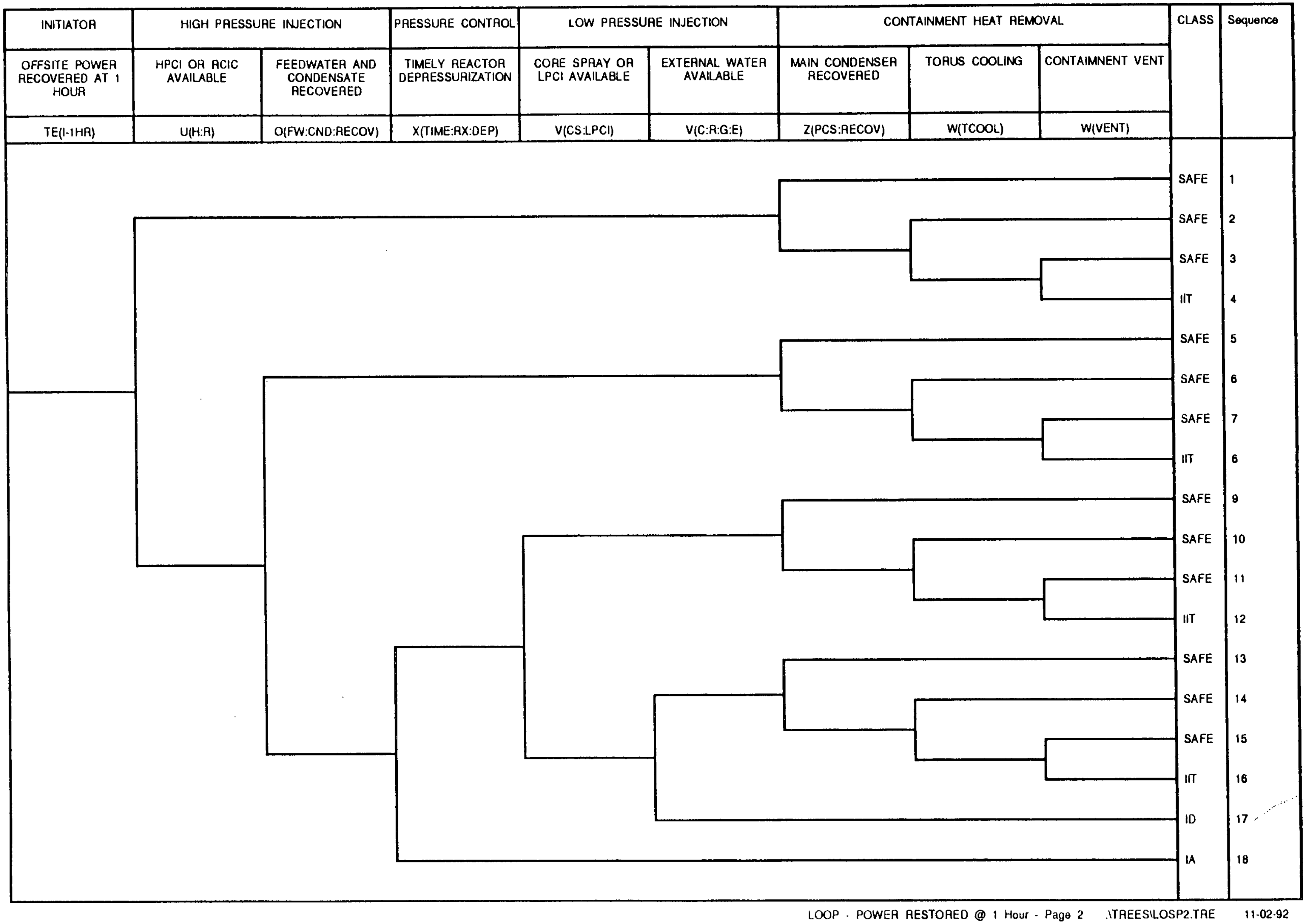


Figure 3.1-12 (Cont.) Loss of Offsite Power Event Tree

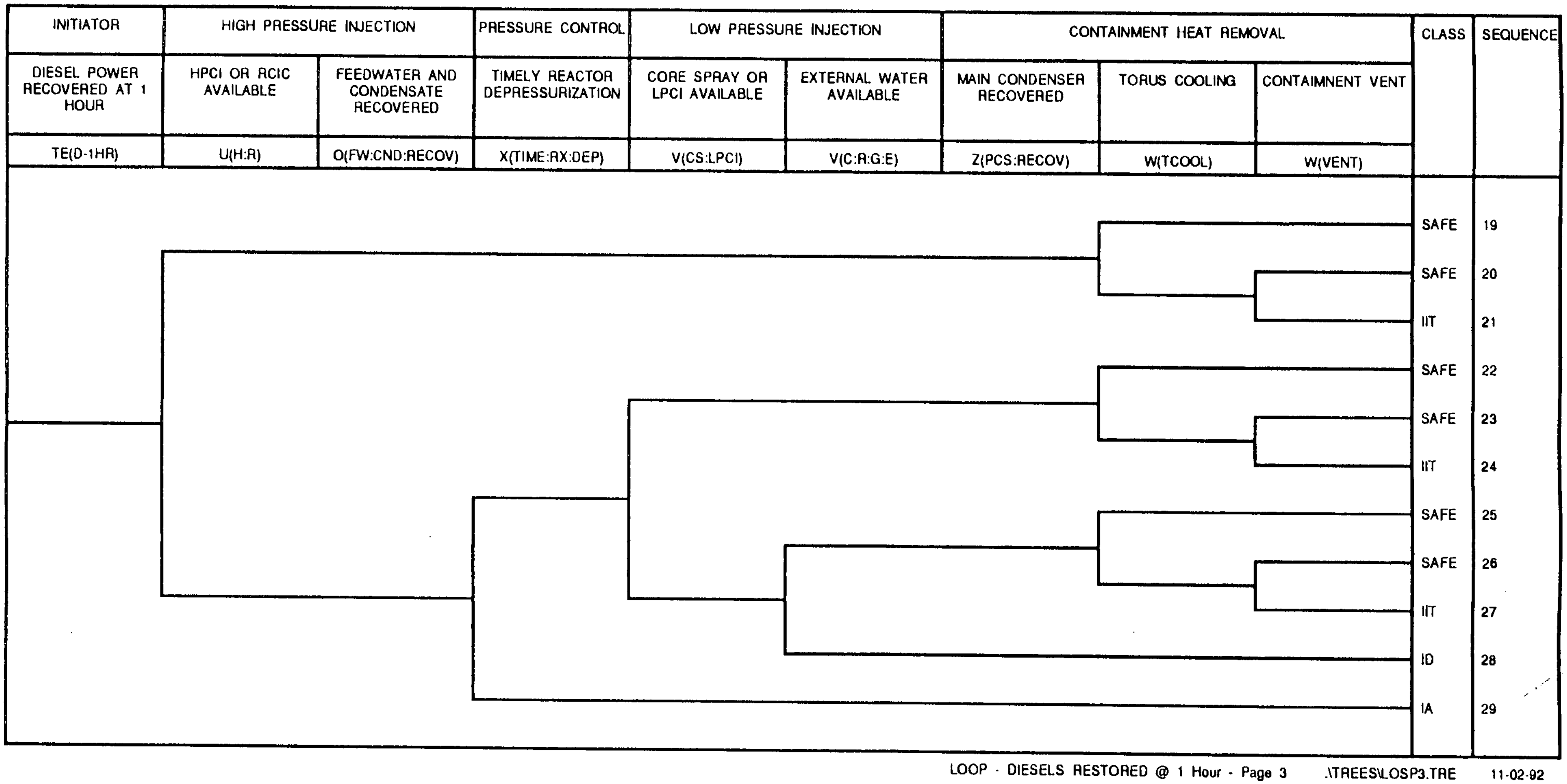
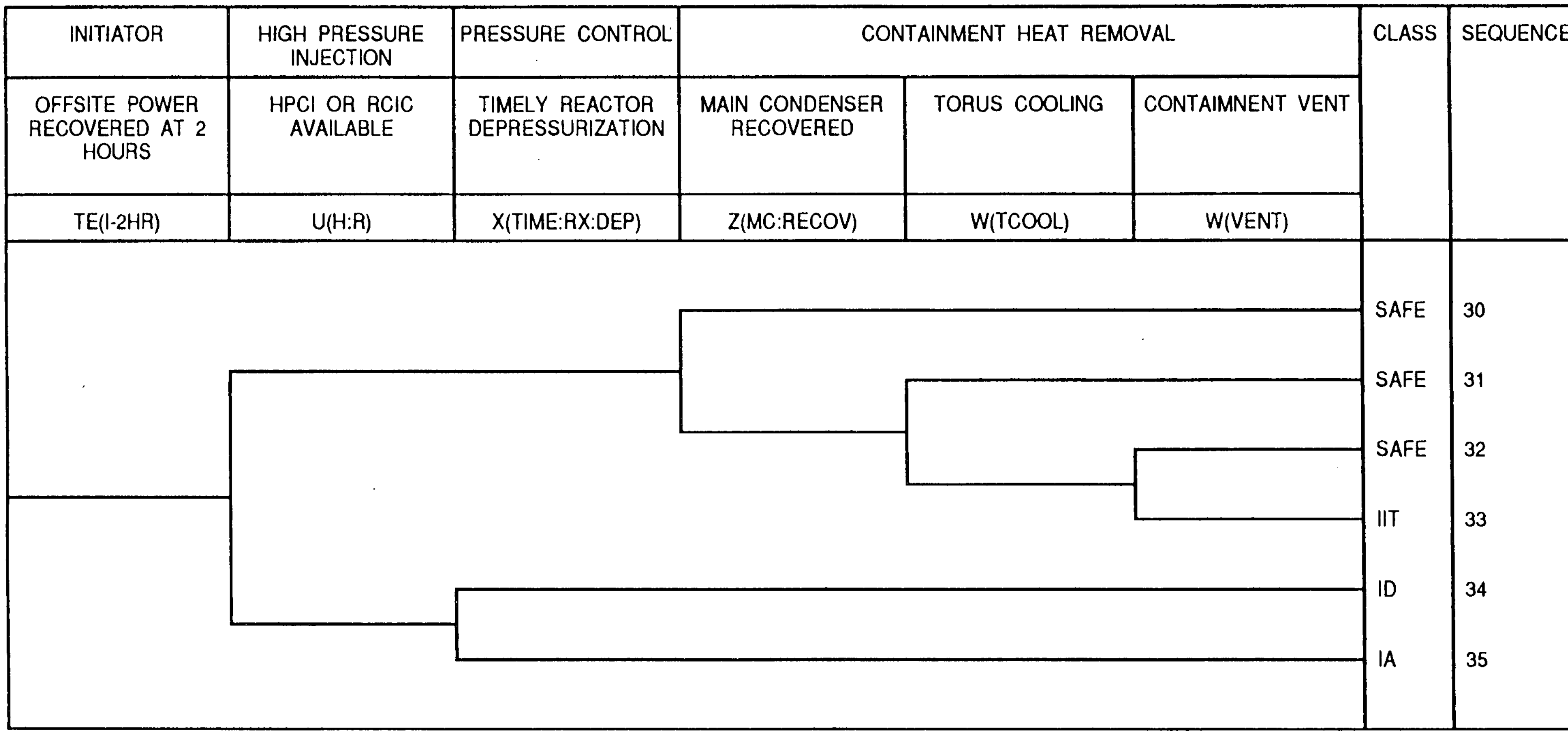
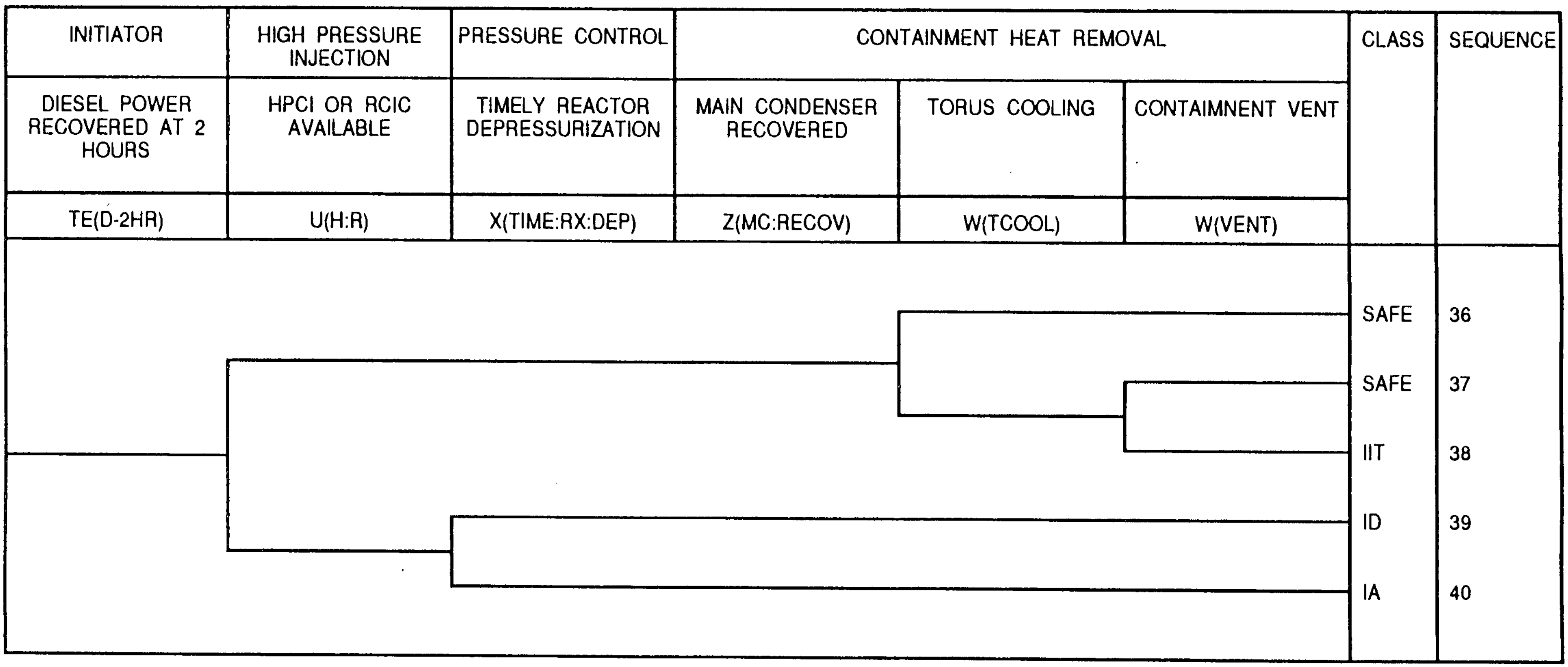


Figure 3.1-12 (Cont.) Loss of Offsite Power Event Tree



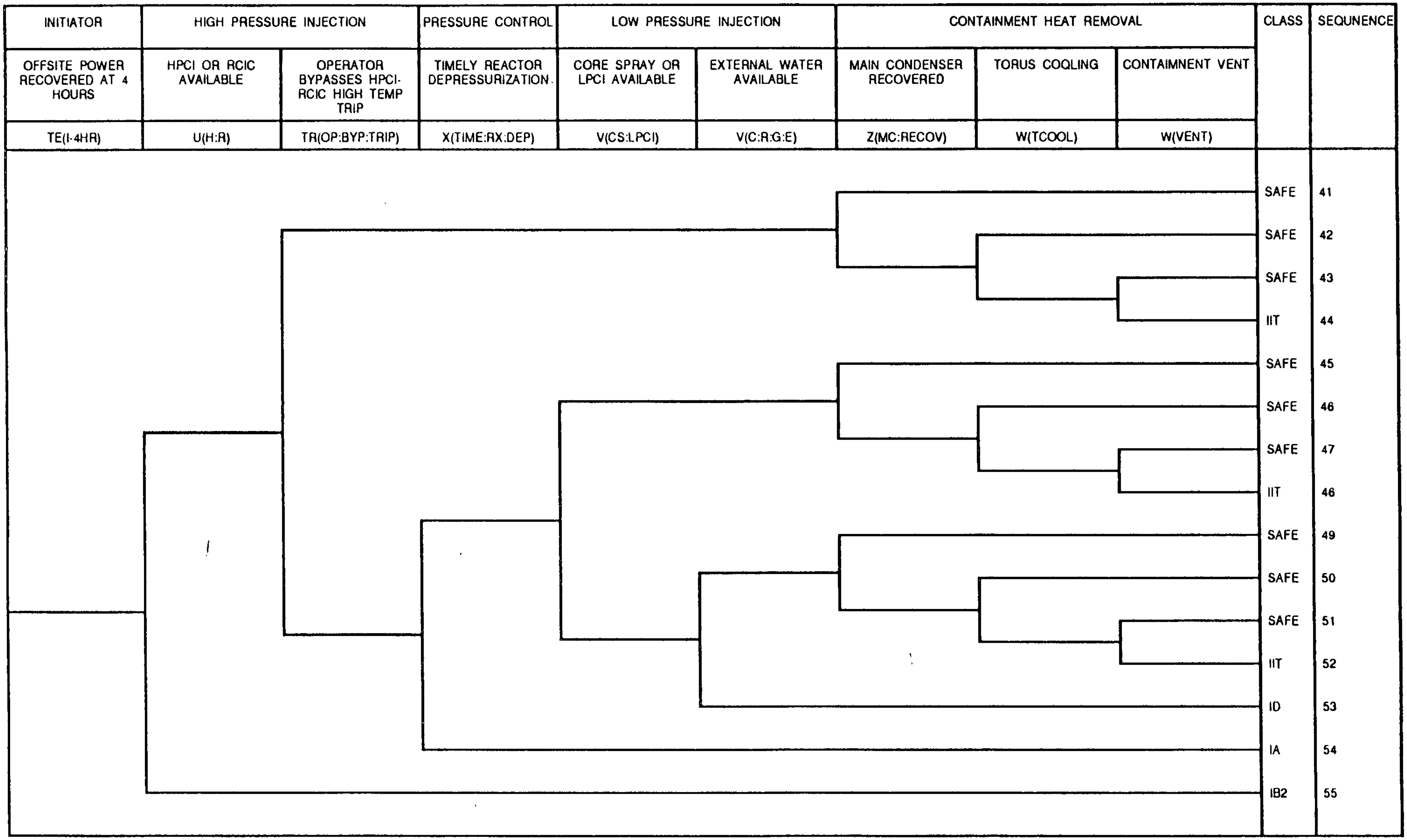
LOOP - POWER RESTORED @ 2 HOURS - Page 4 .\TREES\LOSP4.TRE 11-02-92

Figure 3.1-12 (Cont.) Loss of Offsite Power Event Tree



LOOP - DIESELS RESTORED @ 2 HOURS - Page 5 .\TREES\LOSP5.TRE 11-02-92

Figure 3.1-12 (Cont.) Loss of Offsite Power Event Tree



LOOP - POWER RESTORED @ 4 HOURS - Page 6 .\TREES\LOSP6.TRE 11-02-92

Figure 3.1-12 (Cont.) Loss of Offsite Power Event Tree

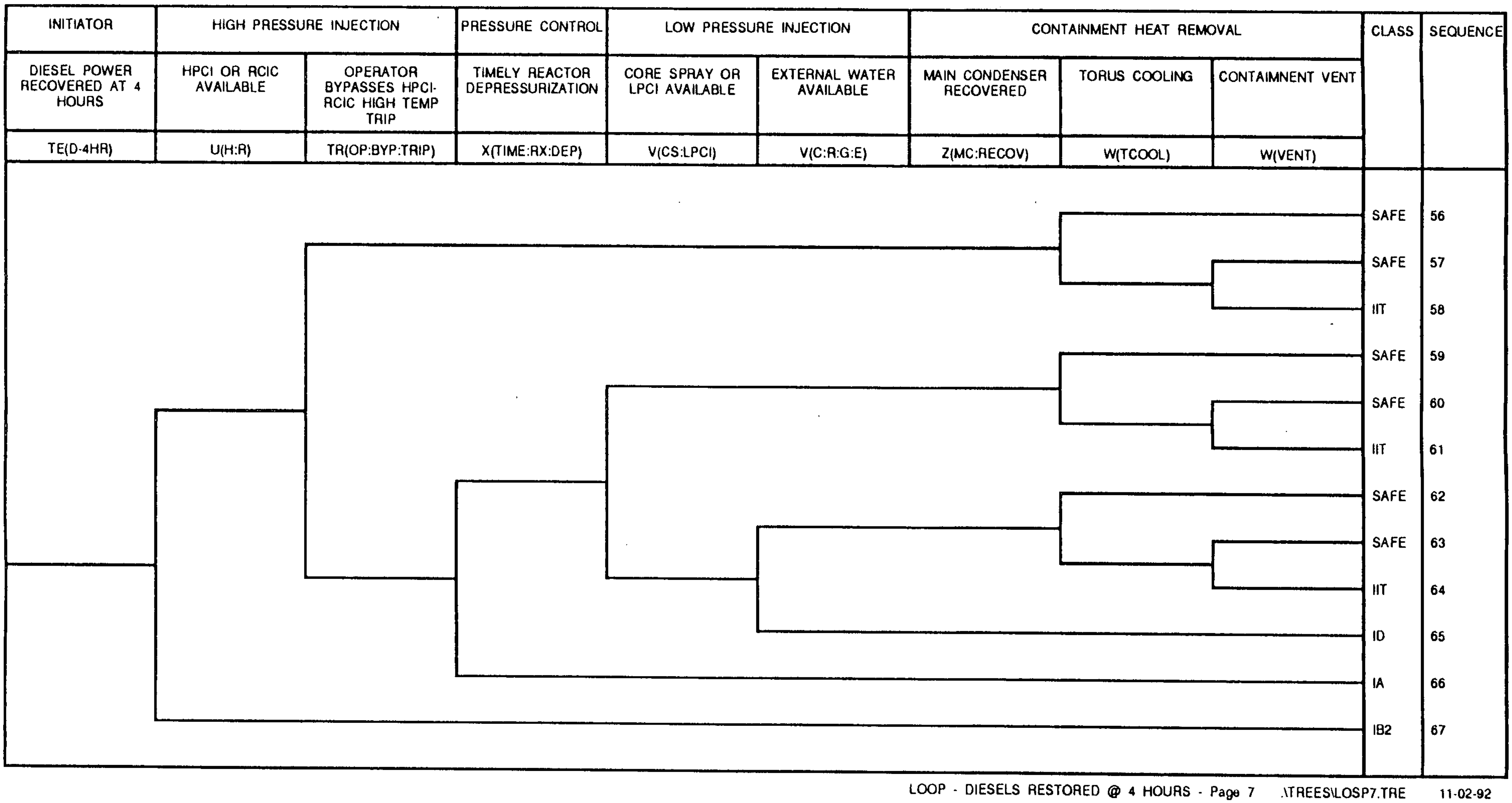
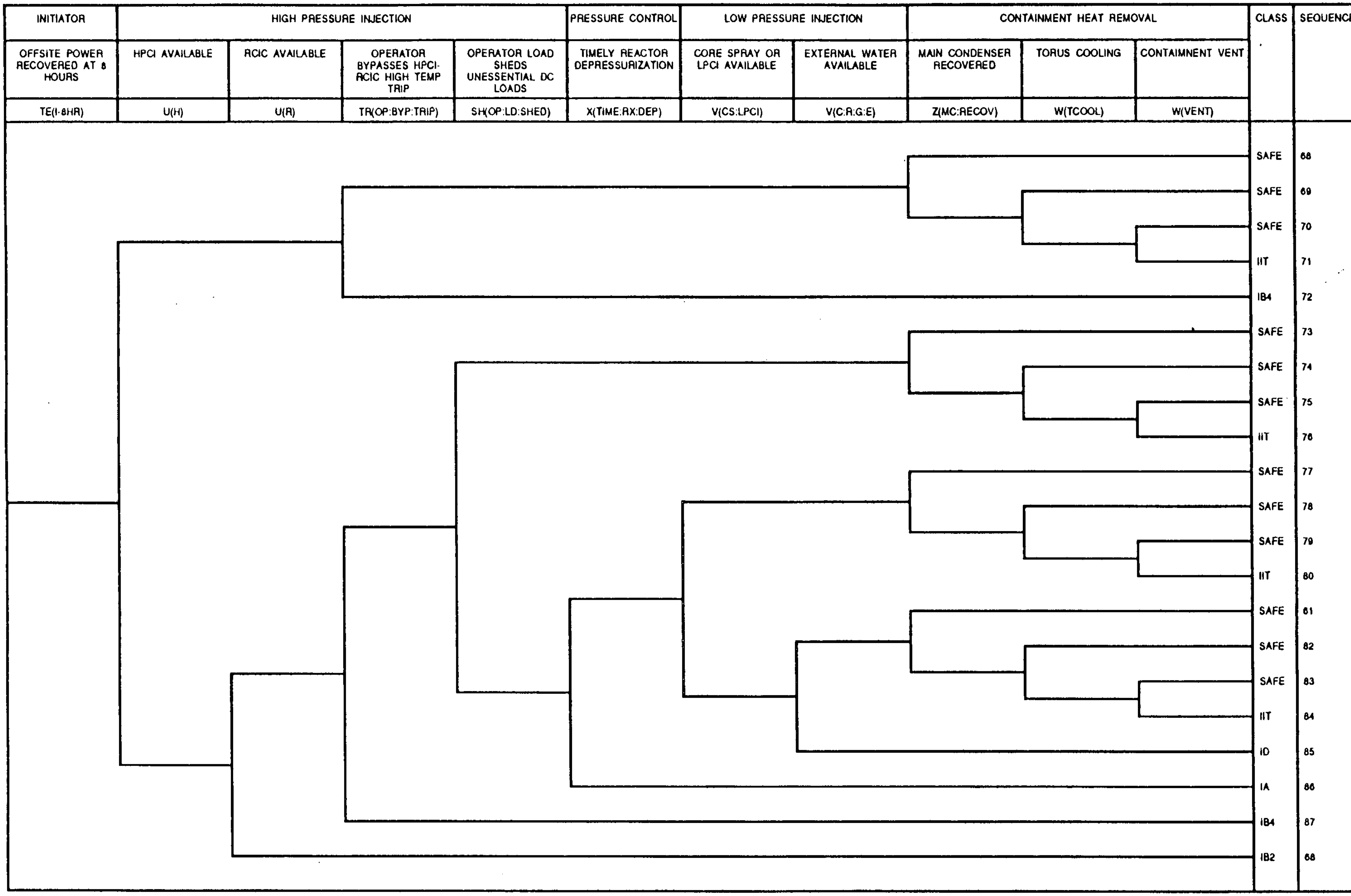


Figure 3.1-12 (Cont.) Loss of Offsite Power Event Tree



LOOP - POWER RESTORED @ 8 HOURS - Page 8 .\TREES\LOSP8.TRE 11-02-92

Figure 3.1-12 (Cont.) Loss of Offsite Power Event Tree

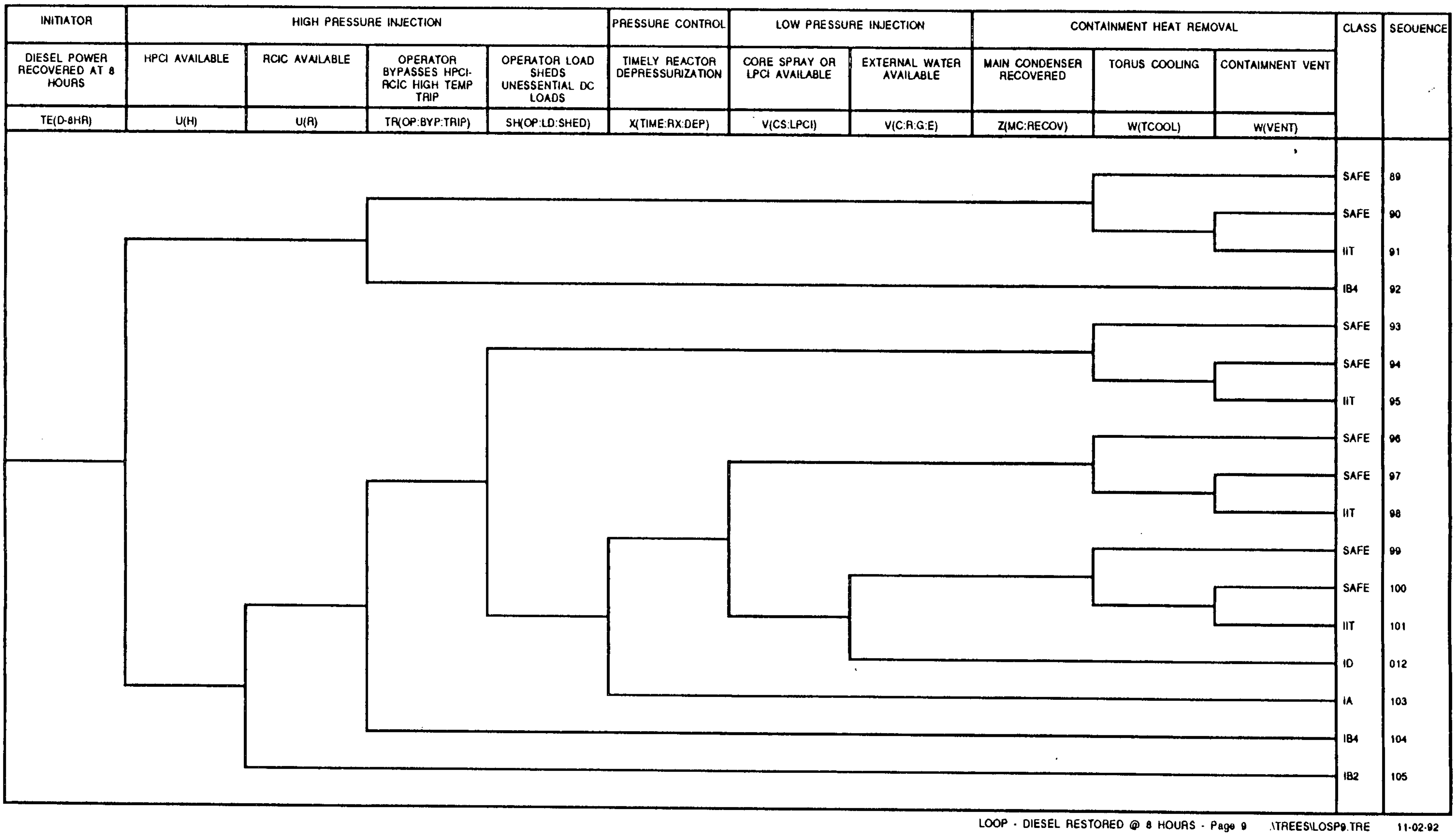


Figure 3.1-12 (Cont.) Loss of Offsite Power Event Tree

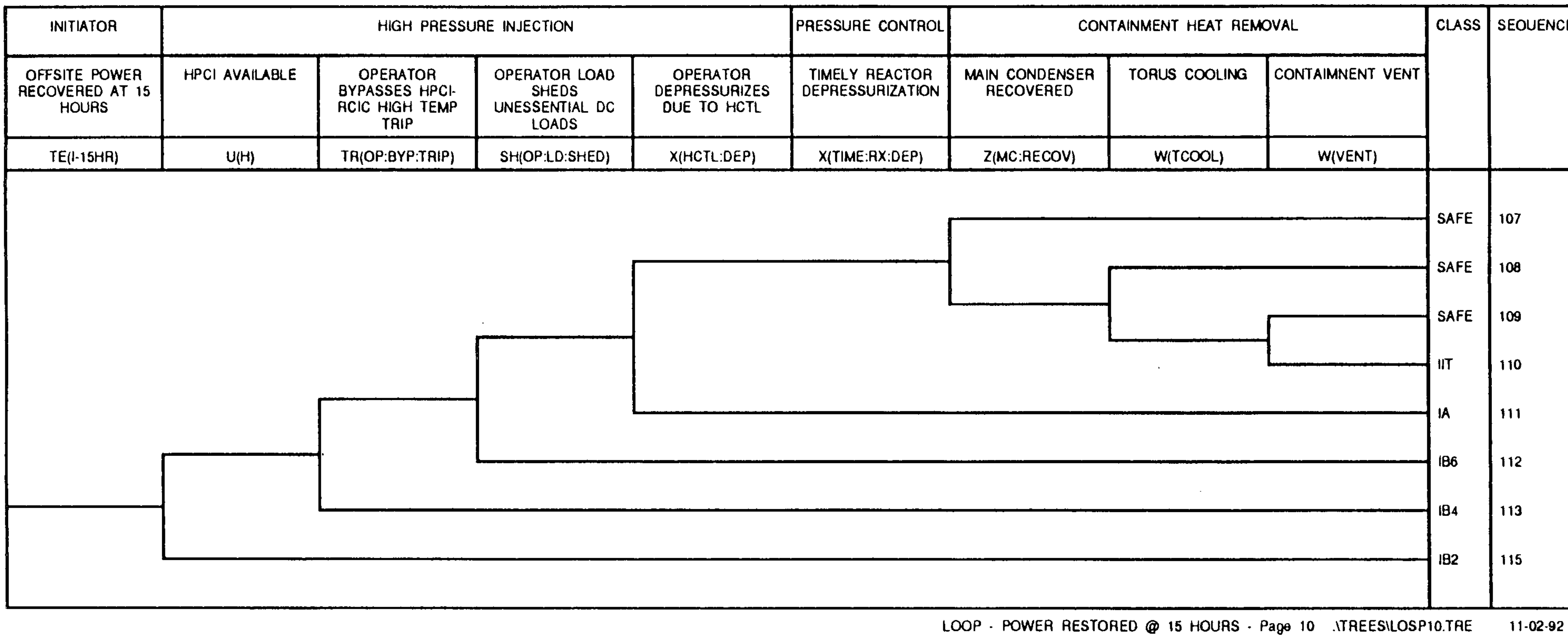


Figure 3.1-12 (Cont.) Loss of Offsite Power Event Tree

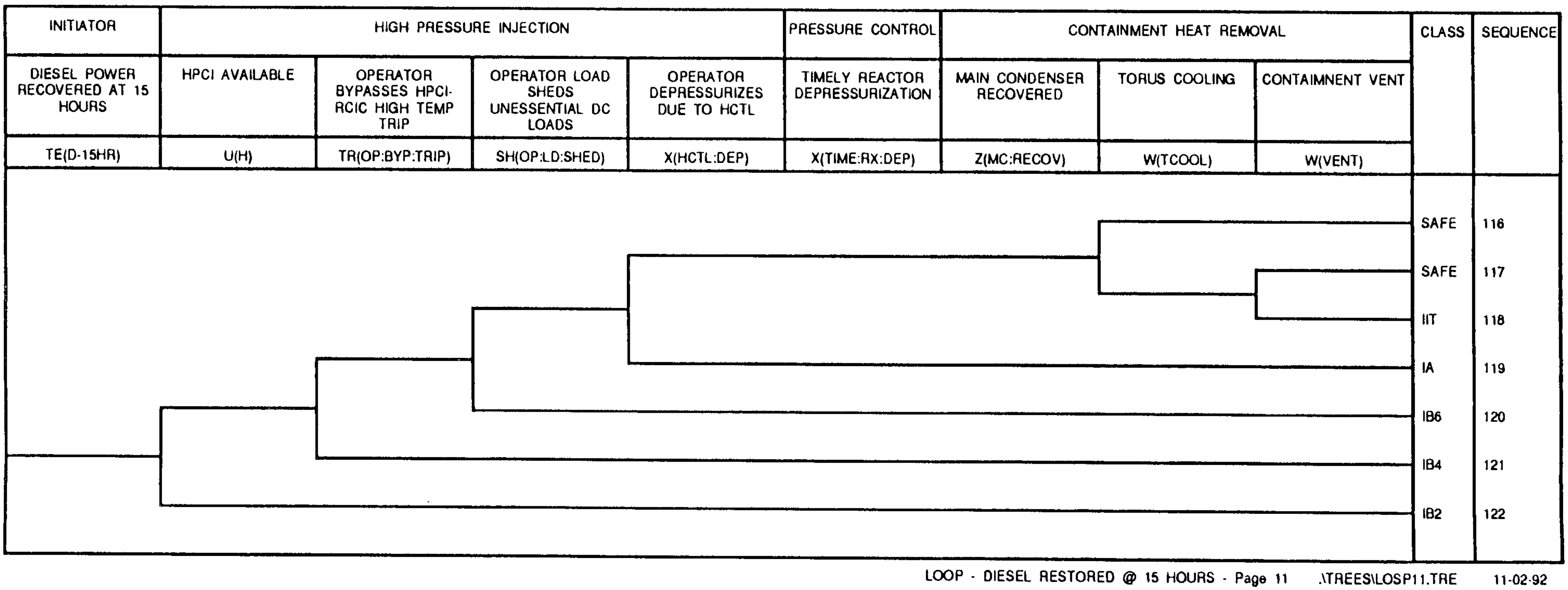
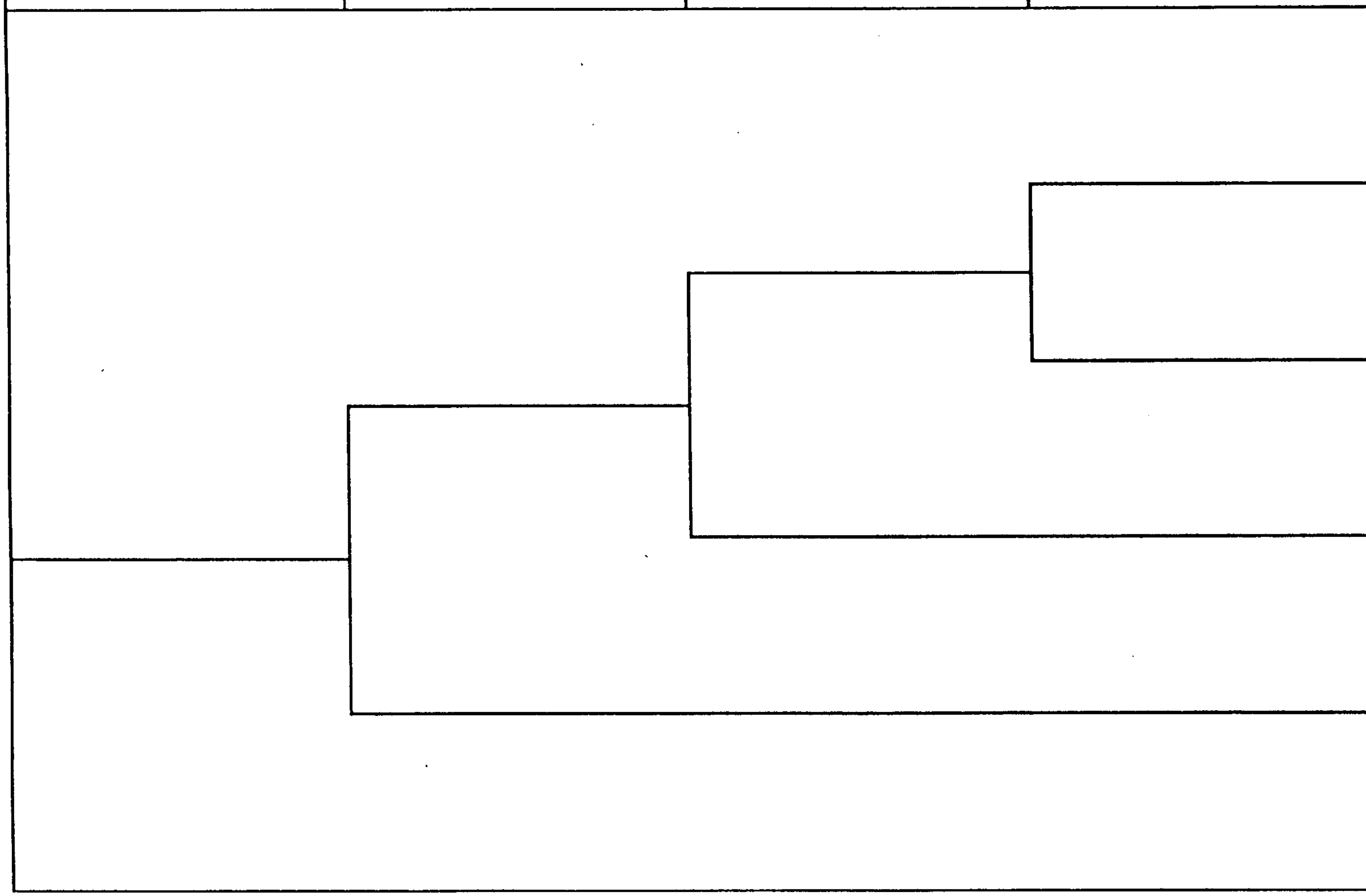


Figure 3.1-12 (Cont.) Loss of Offsite Power Event Tree

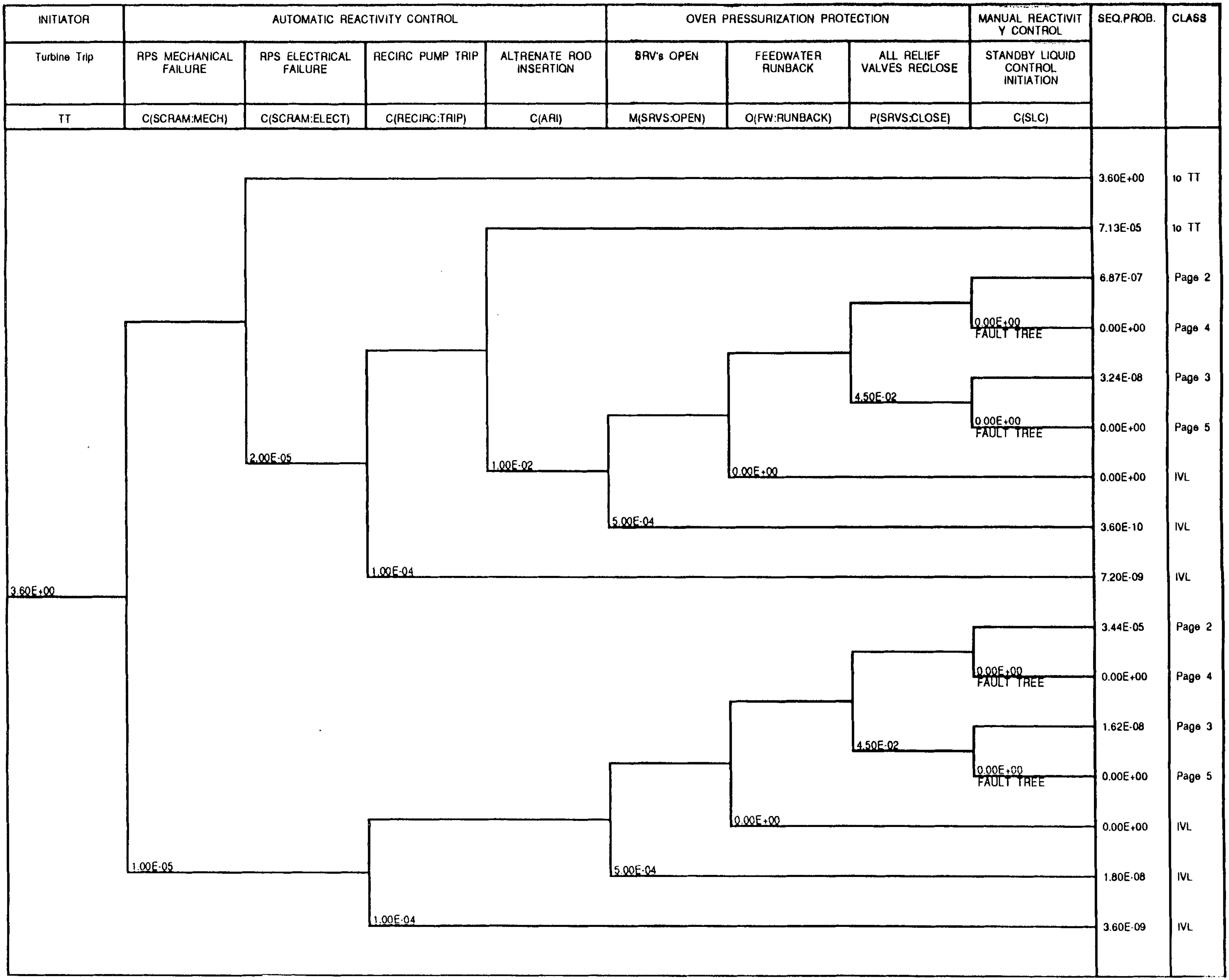
INITIATOR	HIGH PRESSURE INJECTION			CLASS	SEQUENCE
STATION BLACKOUT FOR MORE THAN 15 HOURS	HPCI AVAILABLE	OPERATOR BYPASSES HPCI-RCIC HIGH TEMP TRIP	OPERATOR LOAD SHEDS UNESSENTIAL DC LOADS		
TE(BLACKOUT)	U(H)	TR(OP:BYP:TRIP)	SH(OP:LD:SHED)		



CLASS	SEQUENCE
IBL	123
IB6	124
IB4	125
IB2	126

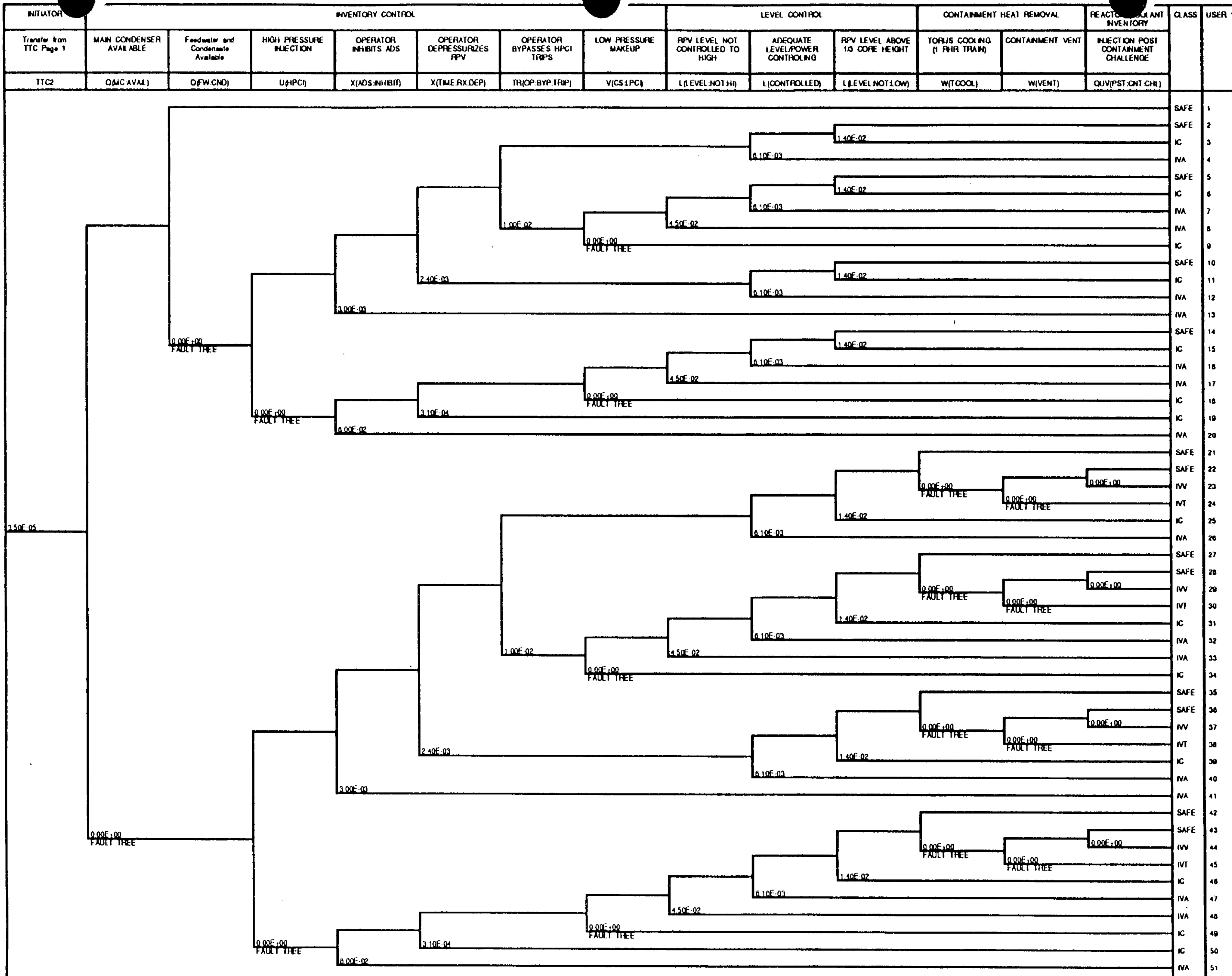
LOOP - BLACKOUT MORE THAN 15 HOUR - Page 12 .\TREES\LOSP12.TRE 11-02-92

Figure 3.1-12 (Cont.) Loss of Offsite Power Event Tree



Turbine Trip ATWS · Page 1 .\TREES\TTC1.TRE 11-02-92

Figure 3.1-13 Turbine Trip with Bypass ATWS Event Tree



Turbine Trip ATWS - Page 2 \TREES\TTC2.TRE 11-02-92

Figure 3.1-13 (Cont.) Turbine Trip with Bypass ATWS Event Tree

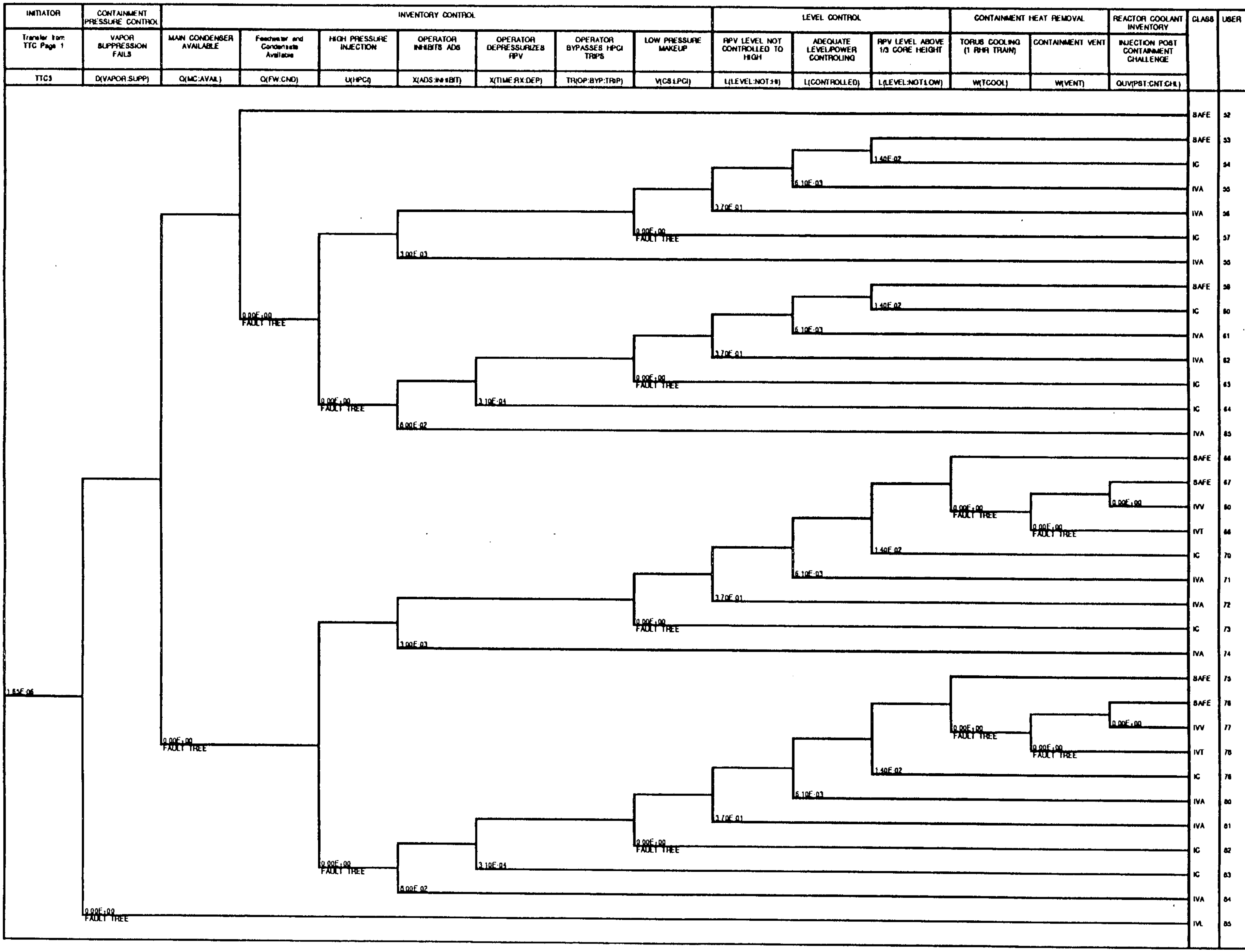


Figure 3.1-13 (Cont.) Turbine Trip with Bypass ATWS Event Tree

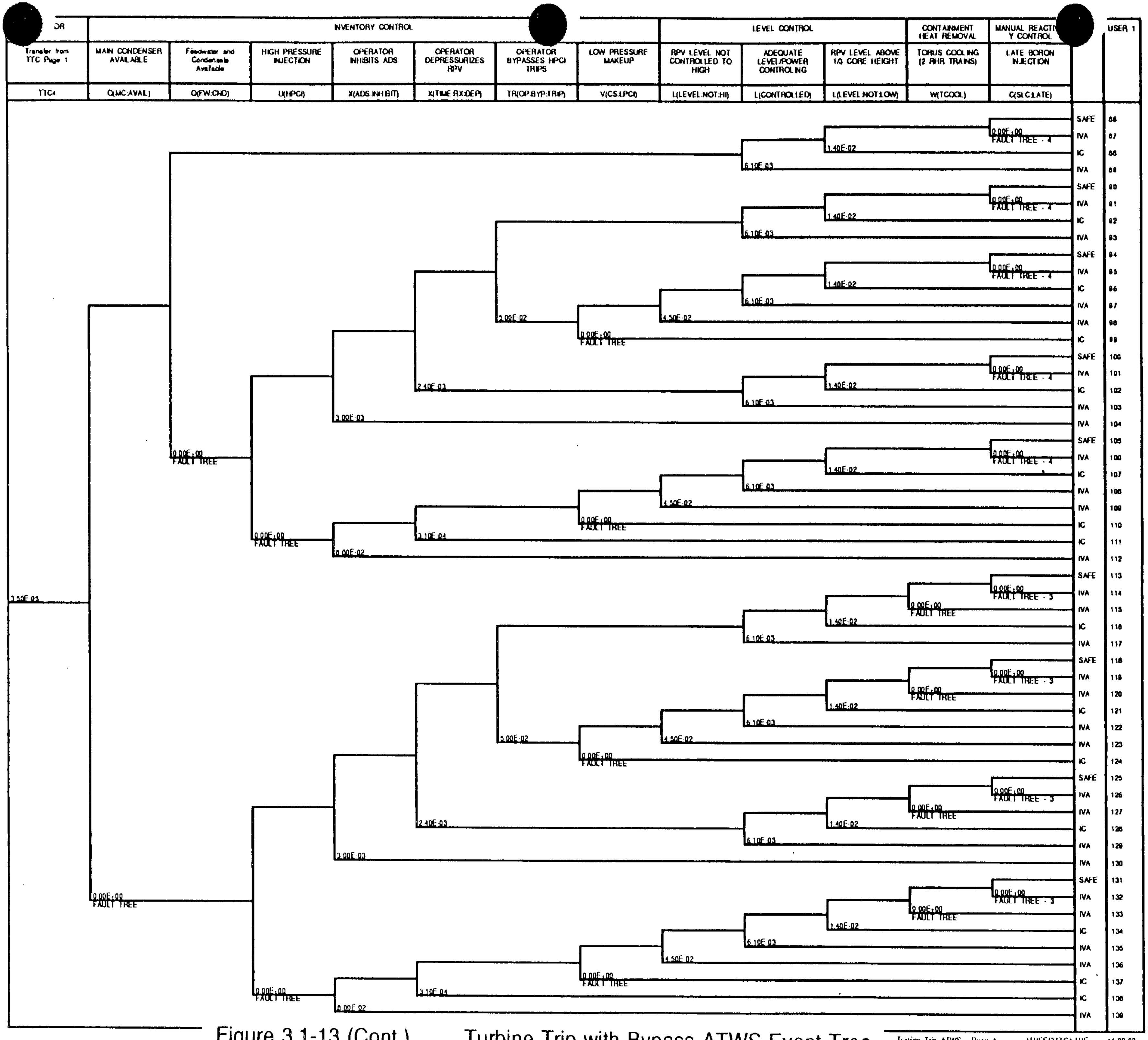


Figure 3.1-13 (Cont.) Turbine Trip with Bypass ATWS Event Tree

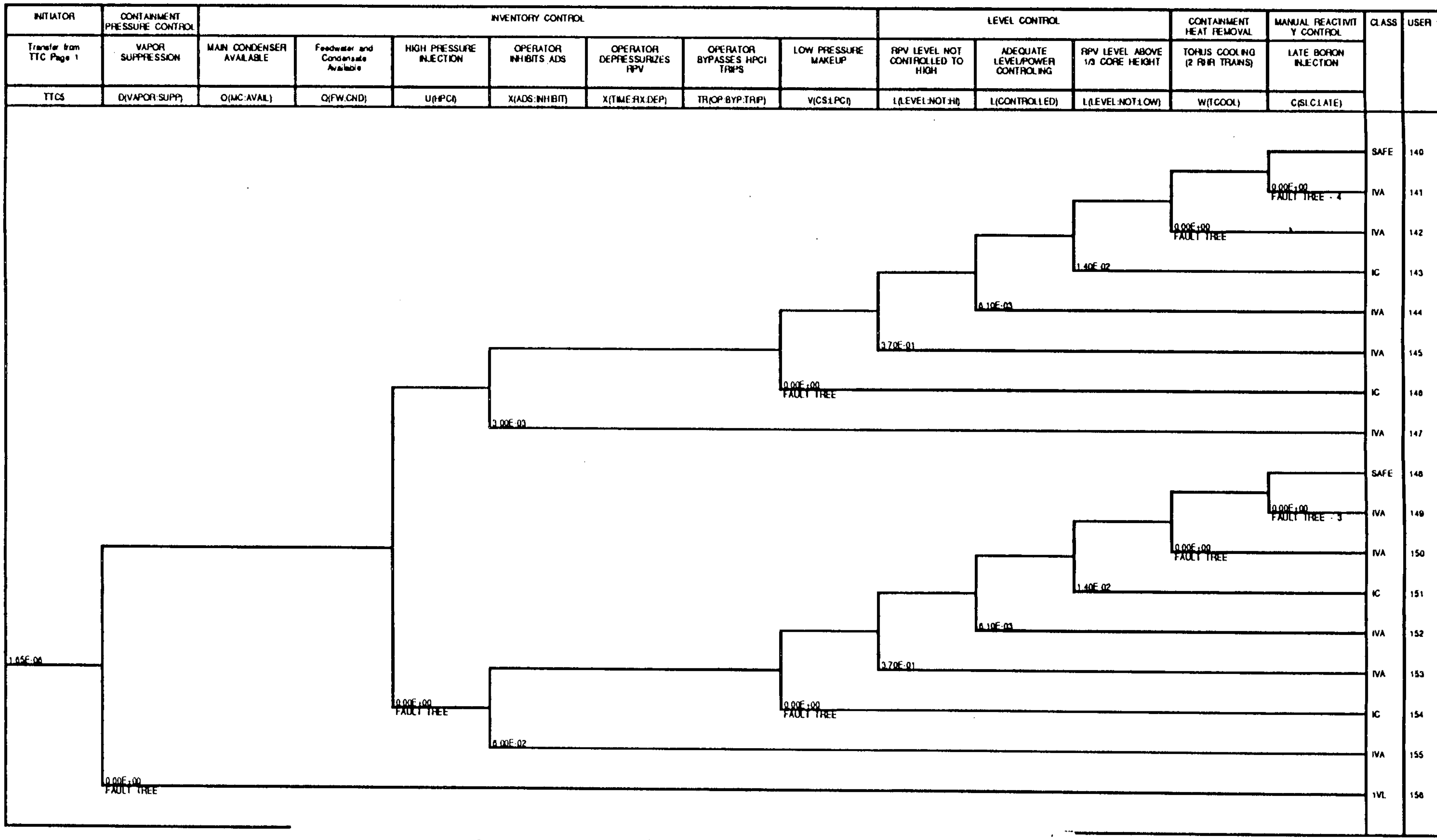


Figure 3.1-13 (Cont.) Turbine Trip with Bypass ATWS Event Tree

Turbine Trip ATWS - Page 5 \TREES\TTC5.TRE 11-02-92

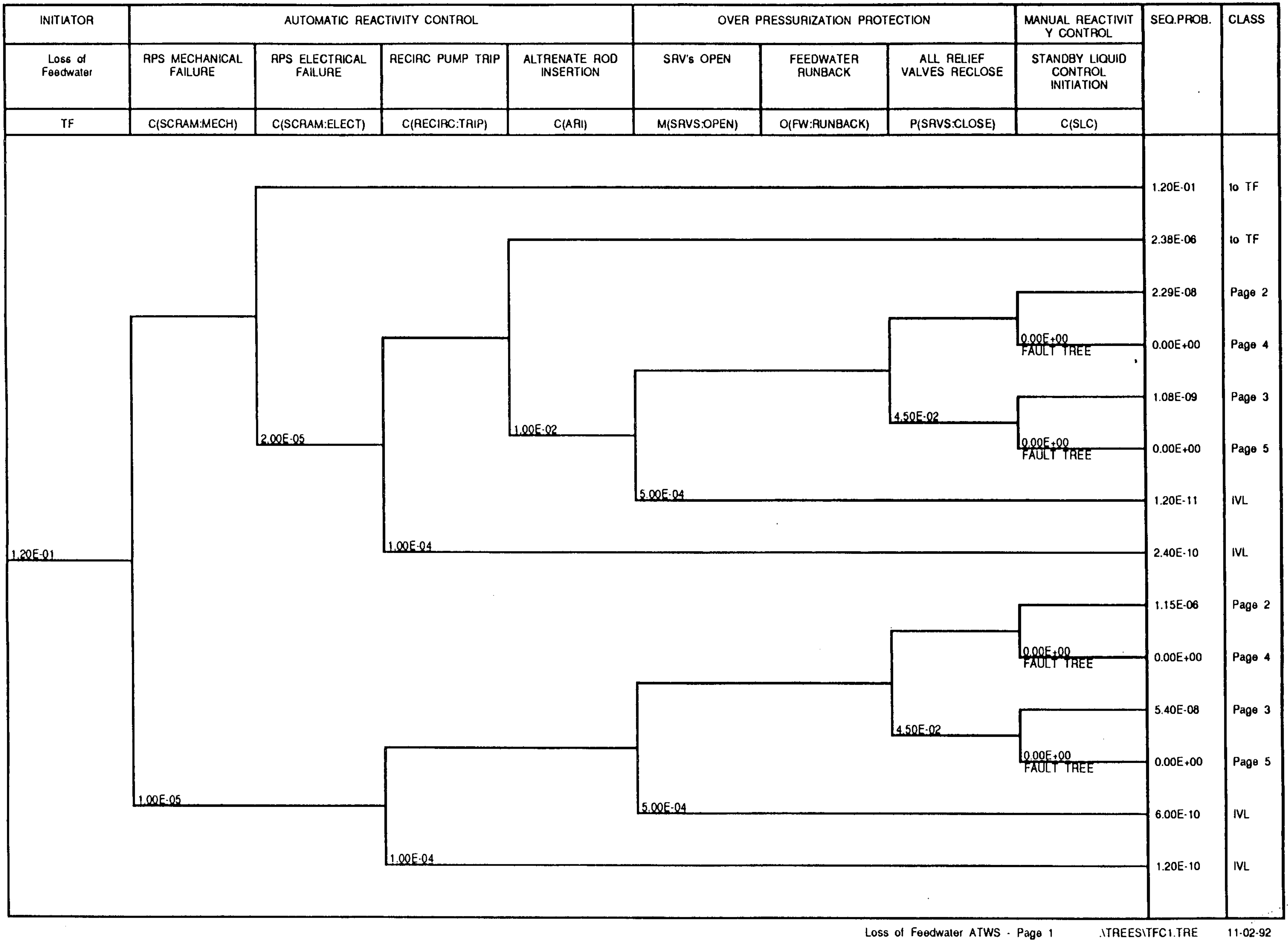


Figure 3.1-14 Loss of Feedwater ATWS Event Tree

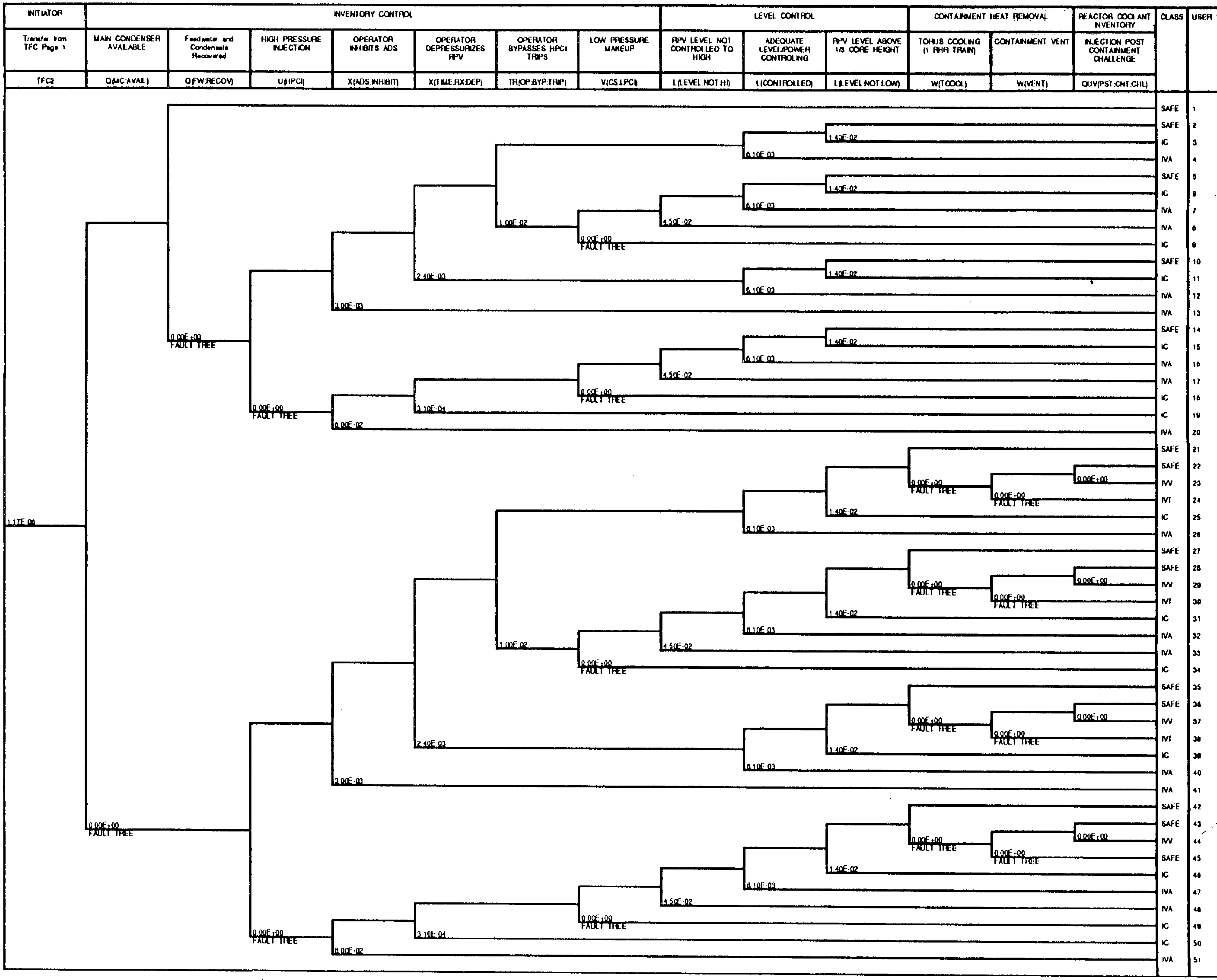
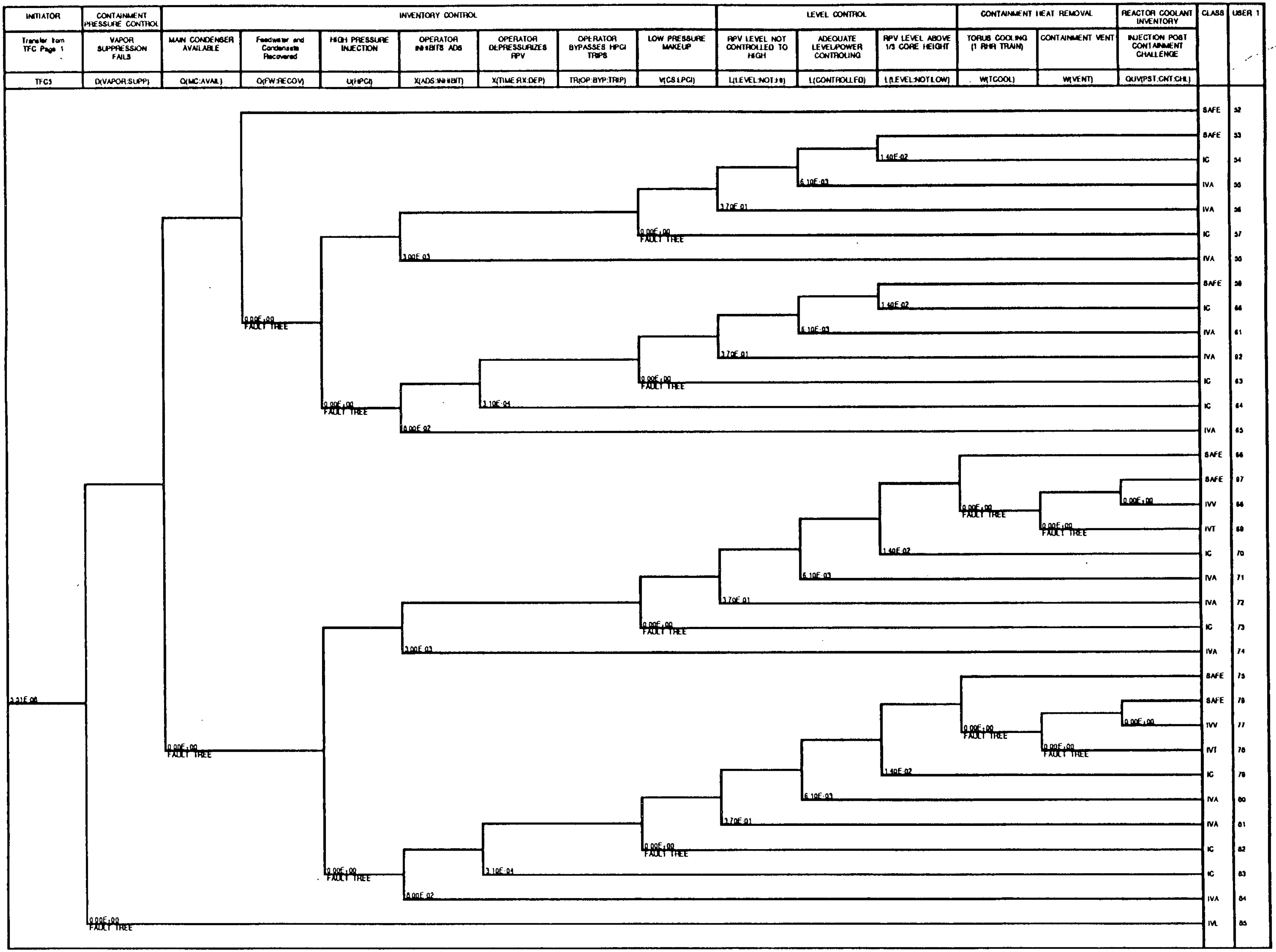


Figure 3.1-14 (Cont.)   Loss of Feedwater ATWS Event Tree



Loss of Feedwater ATWS - Page 3 .\TREES\TFC3.TRE 11-02-02

Figure 3.1-14 (Cont.) Loss of Feedwater ATWS Event Tree

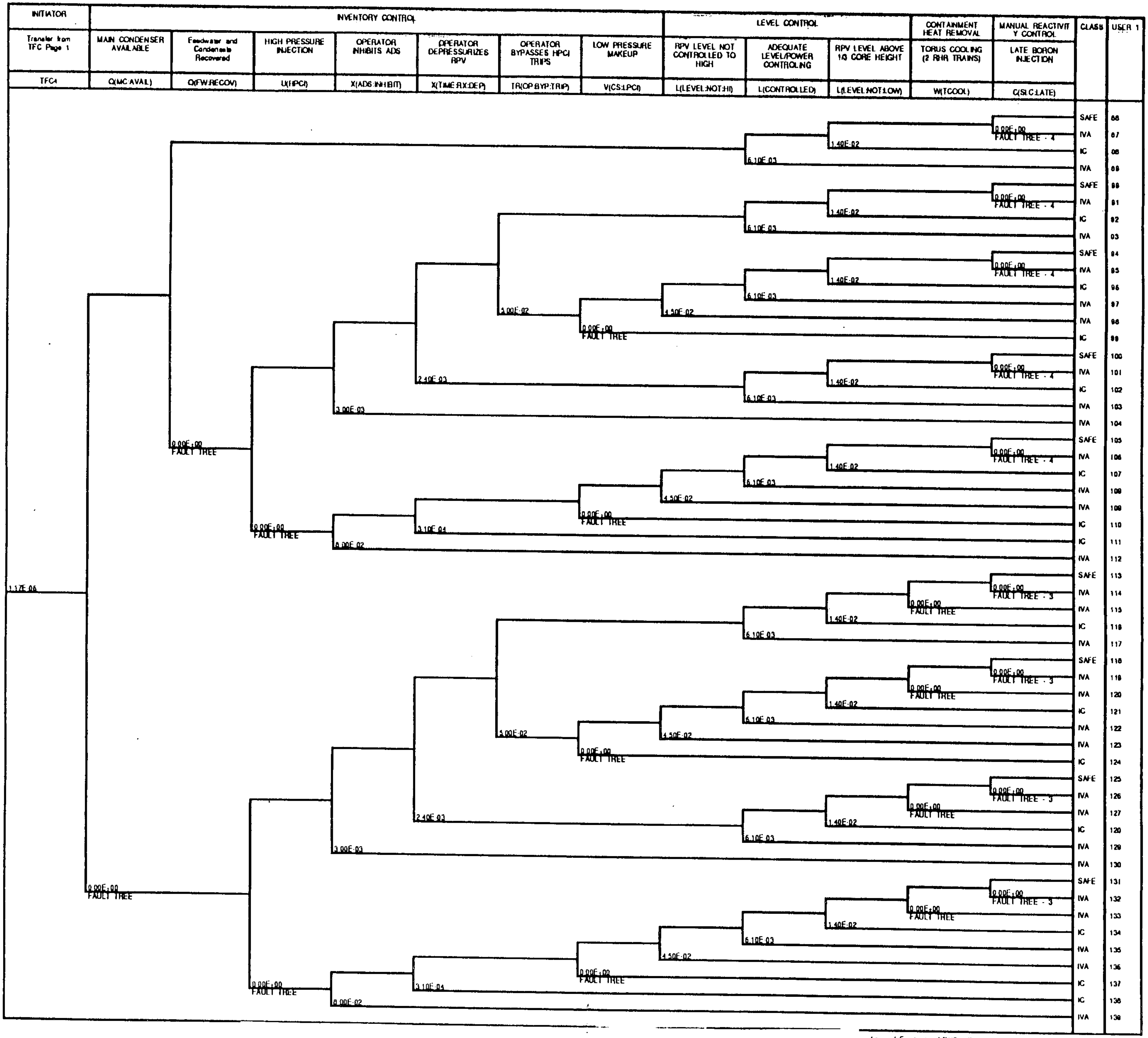


Figure 3.1-14 (Cont.) Loss of Feedwater ATWS Event Tree

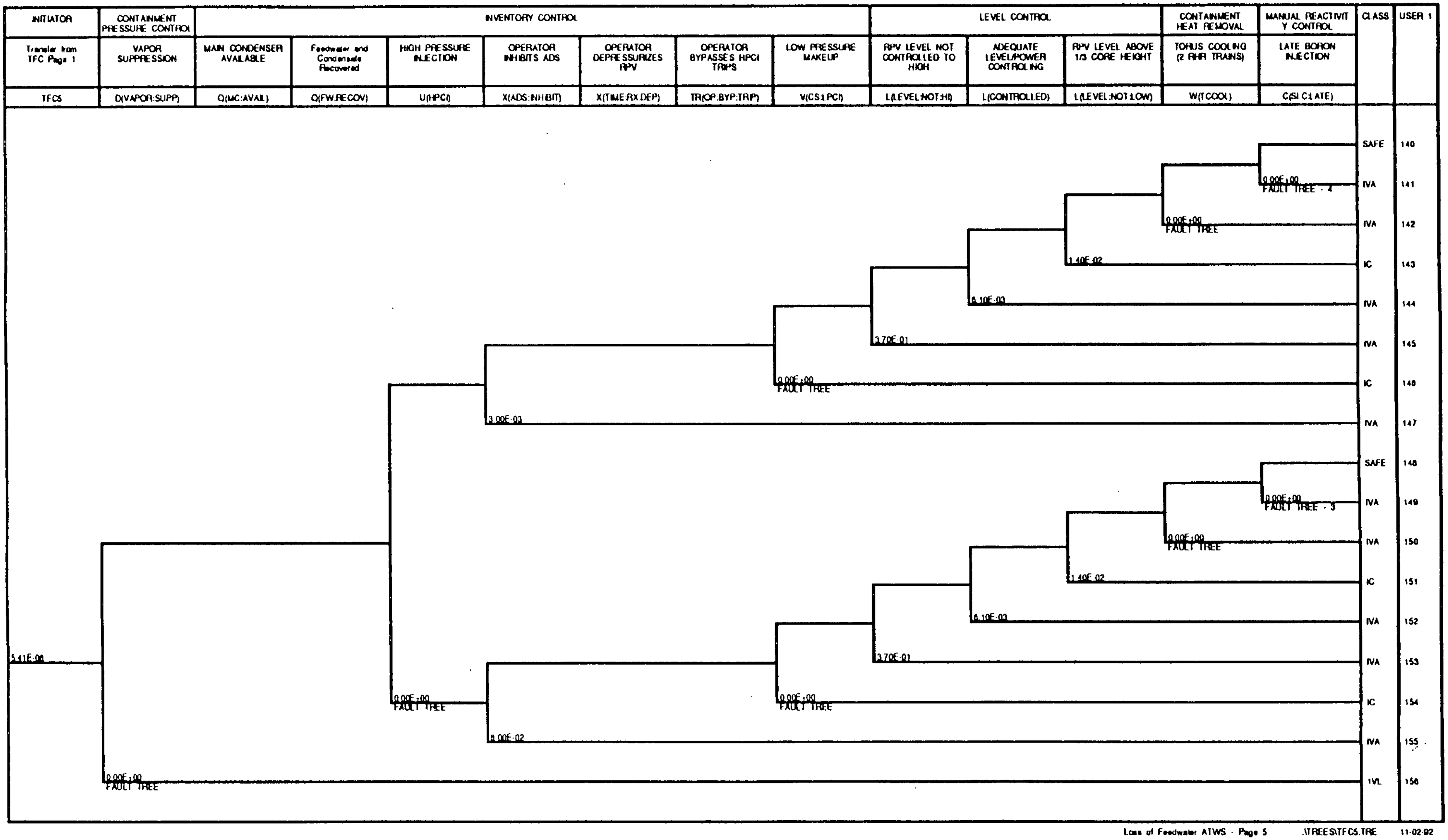


Figure 3.1-14 (Cont.) Loss of Feedwater ATWS Event Tree

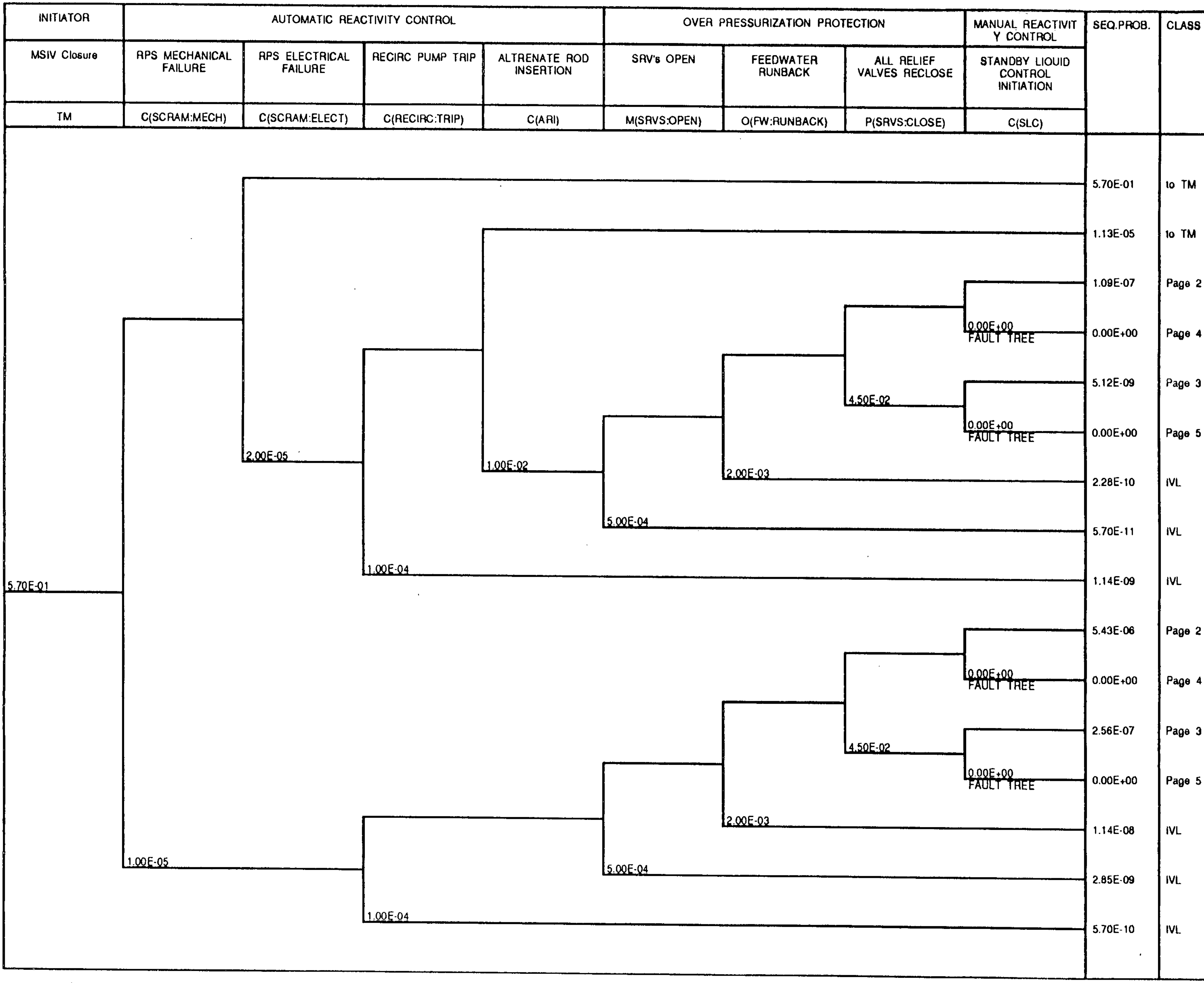


Figure 3.1-15 MSIV Closure ATWS Event Tree

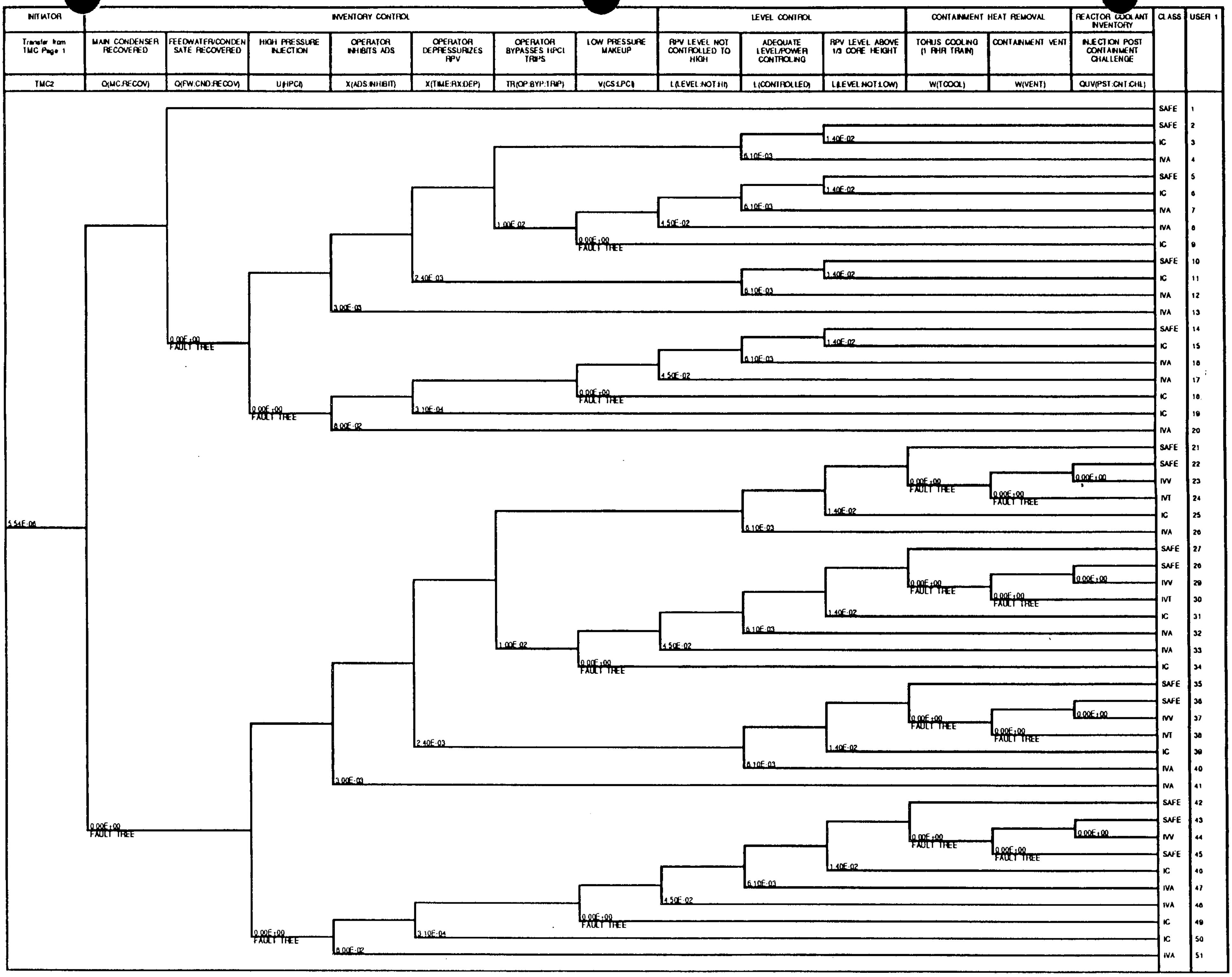


Figure 3.1-15 (Cont.) MSIV Closure ATWS Event Tree

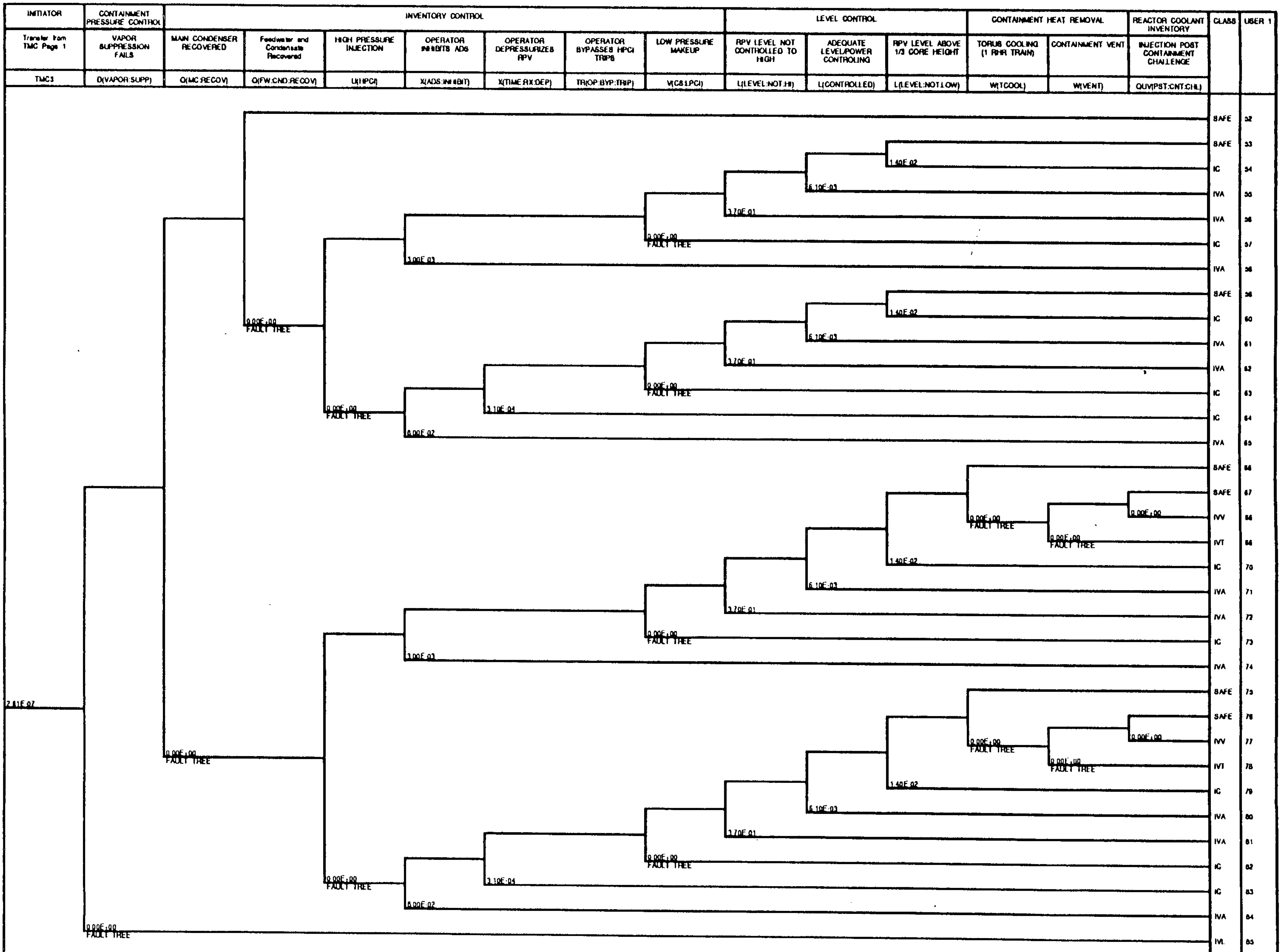


Figure 3.1-15 (Cont.) MSIV Closure ATWS Event Tree

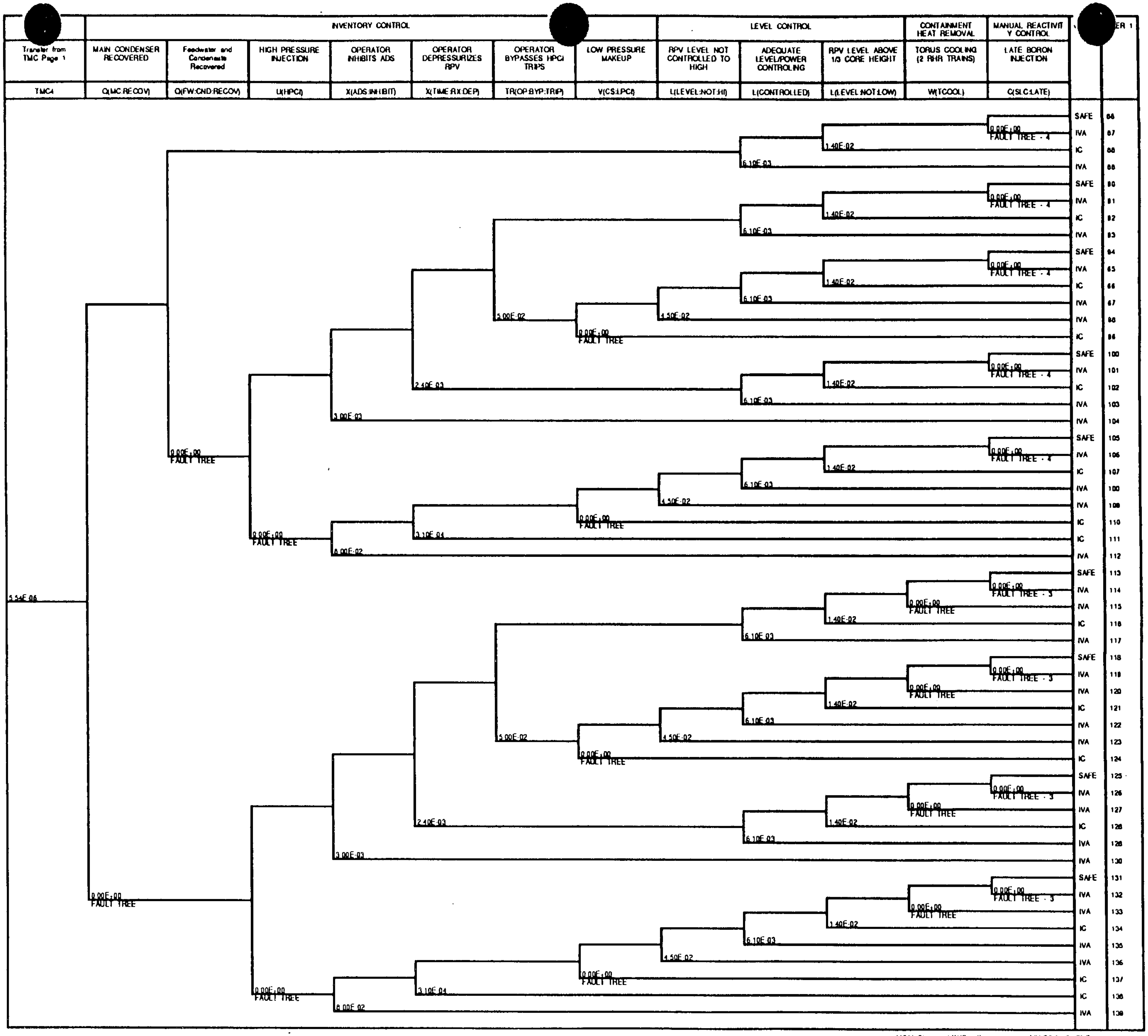


Figure 3.1-15 (Cont.) MSIV Closure ATWS Event Tree

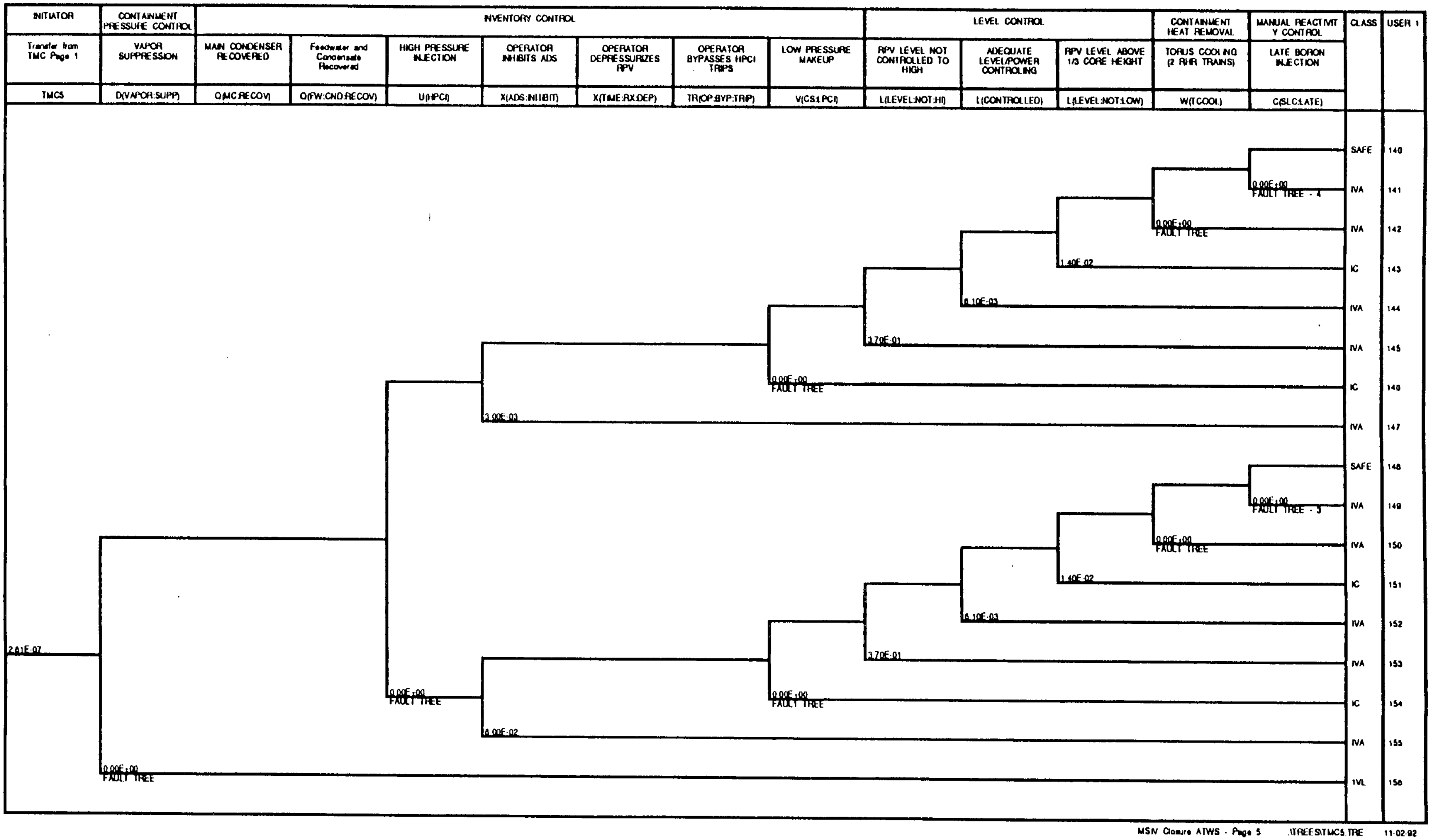


Figure 3.1-15 (Cont.) MSIV Closure ATWS Event Tree

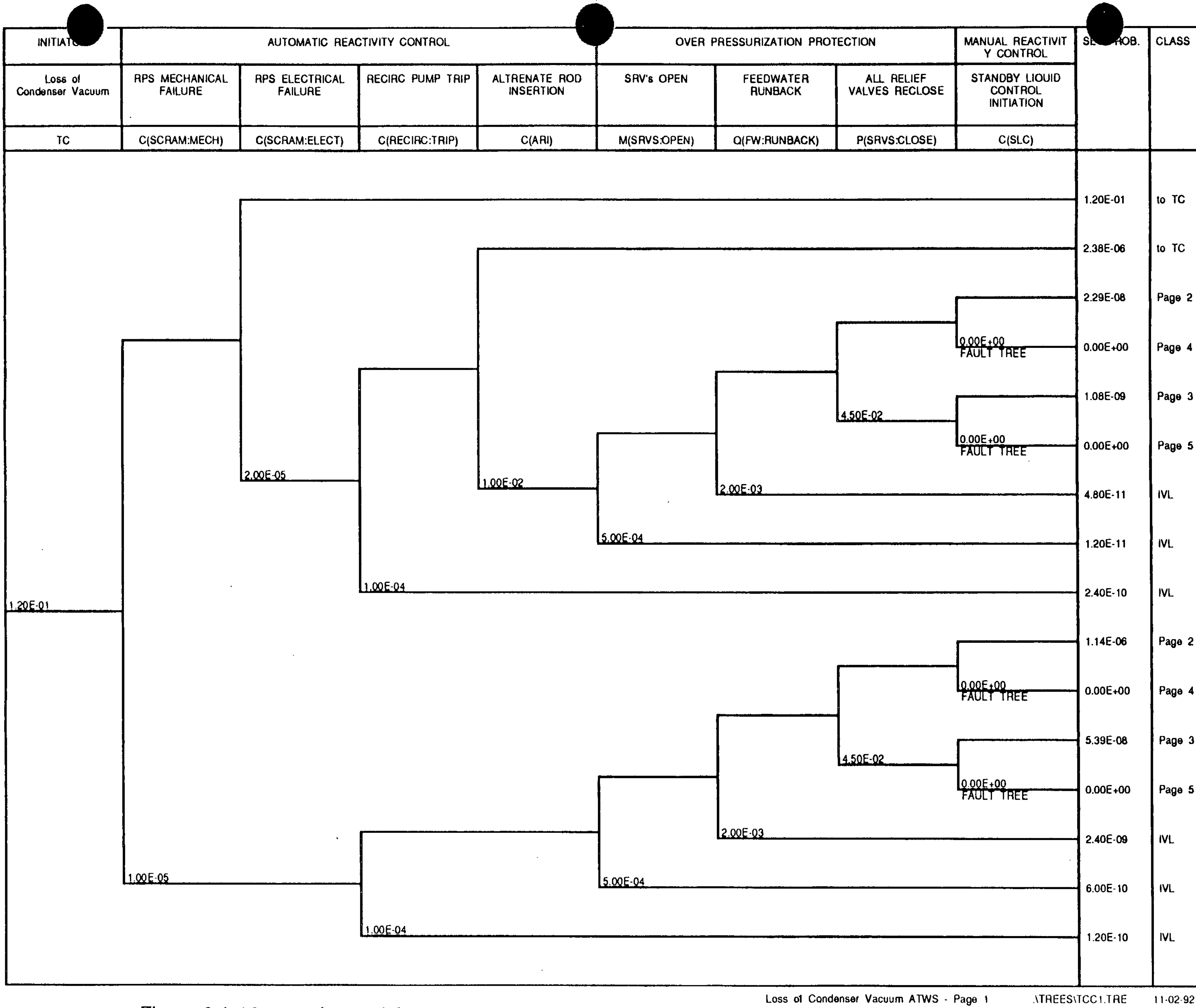


Figure 3.1-16 Loss of Condenser Vacuum ATWS Event Tree

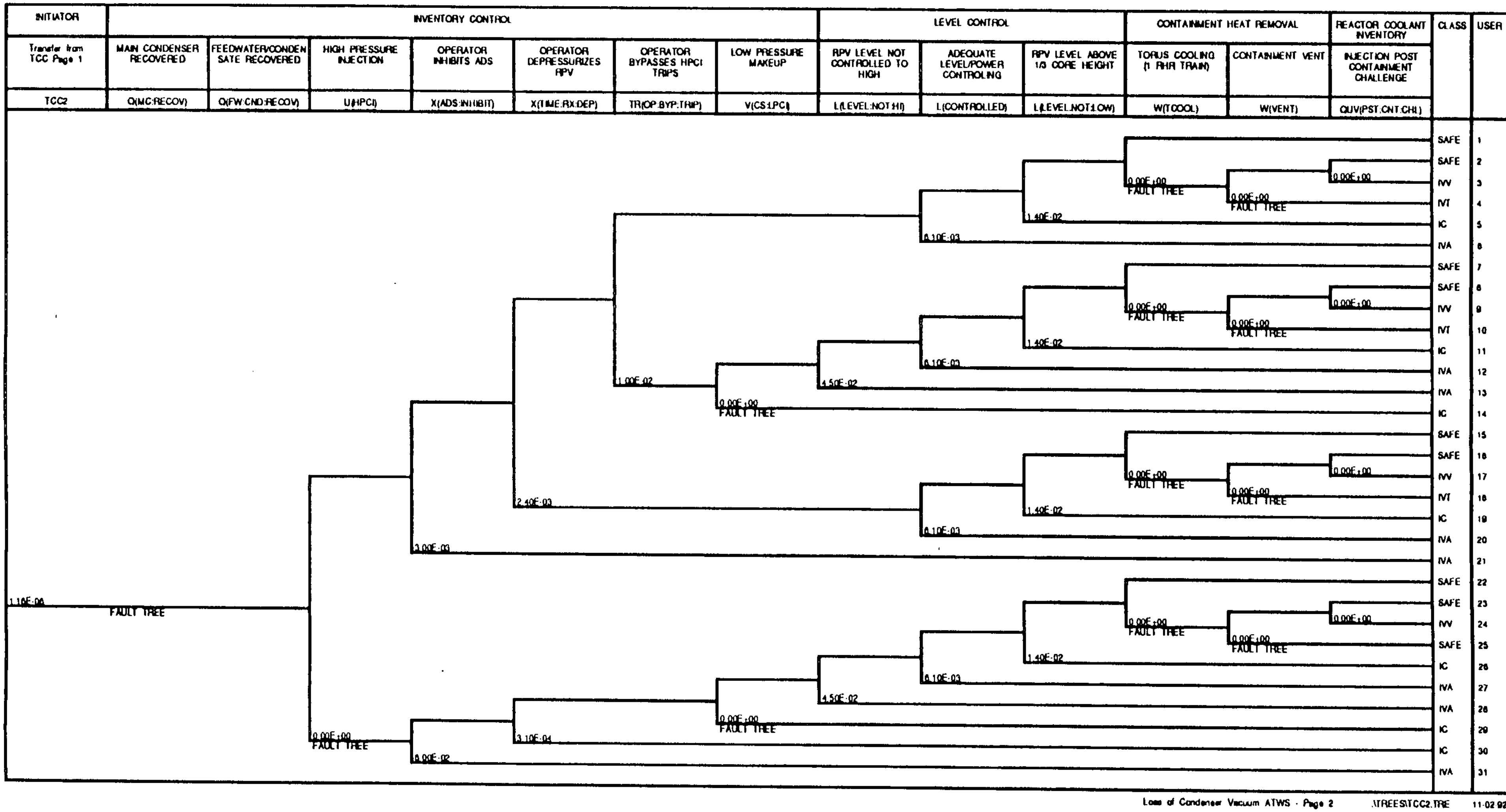


Figure 3.1-16 (Cont.) Loss of Condenser Vacuum ATWS Event Tree

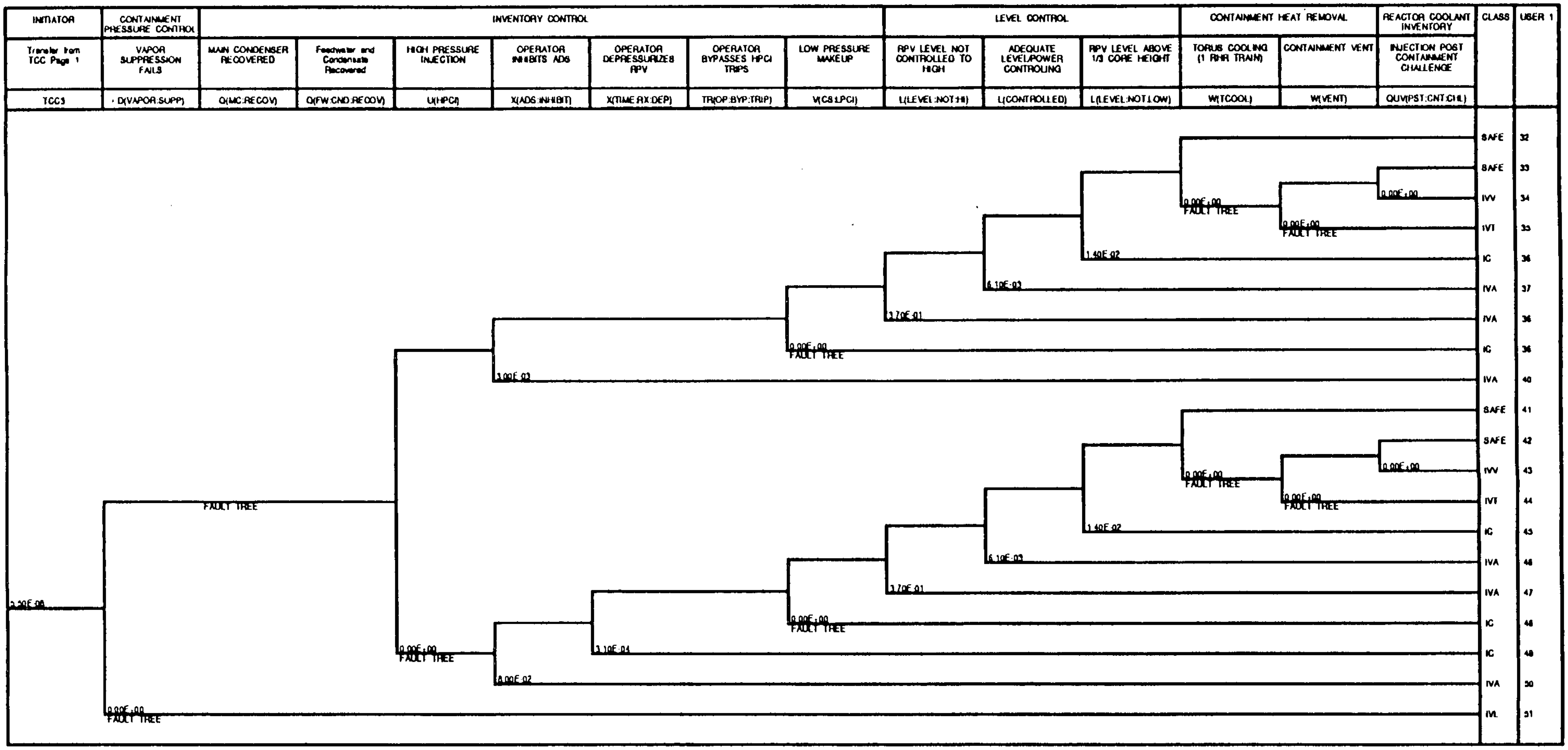


Figure 3.1-16 (Cont.) Loss of Condenser Vacuum ATWS Event Tree

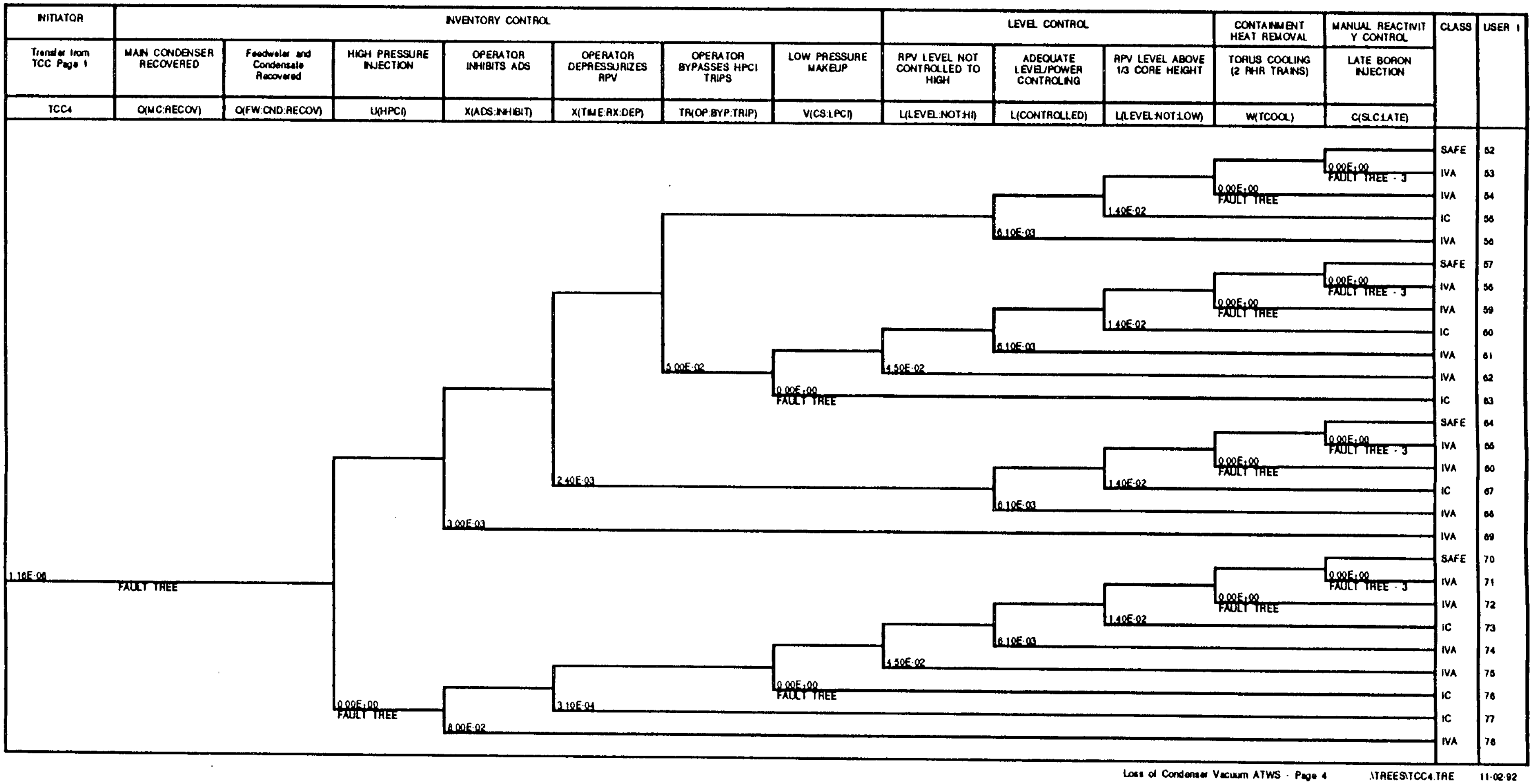


Figure 3.1-16 (Cont.) Loss of Condenser Vacuum ATWS Event Tree

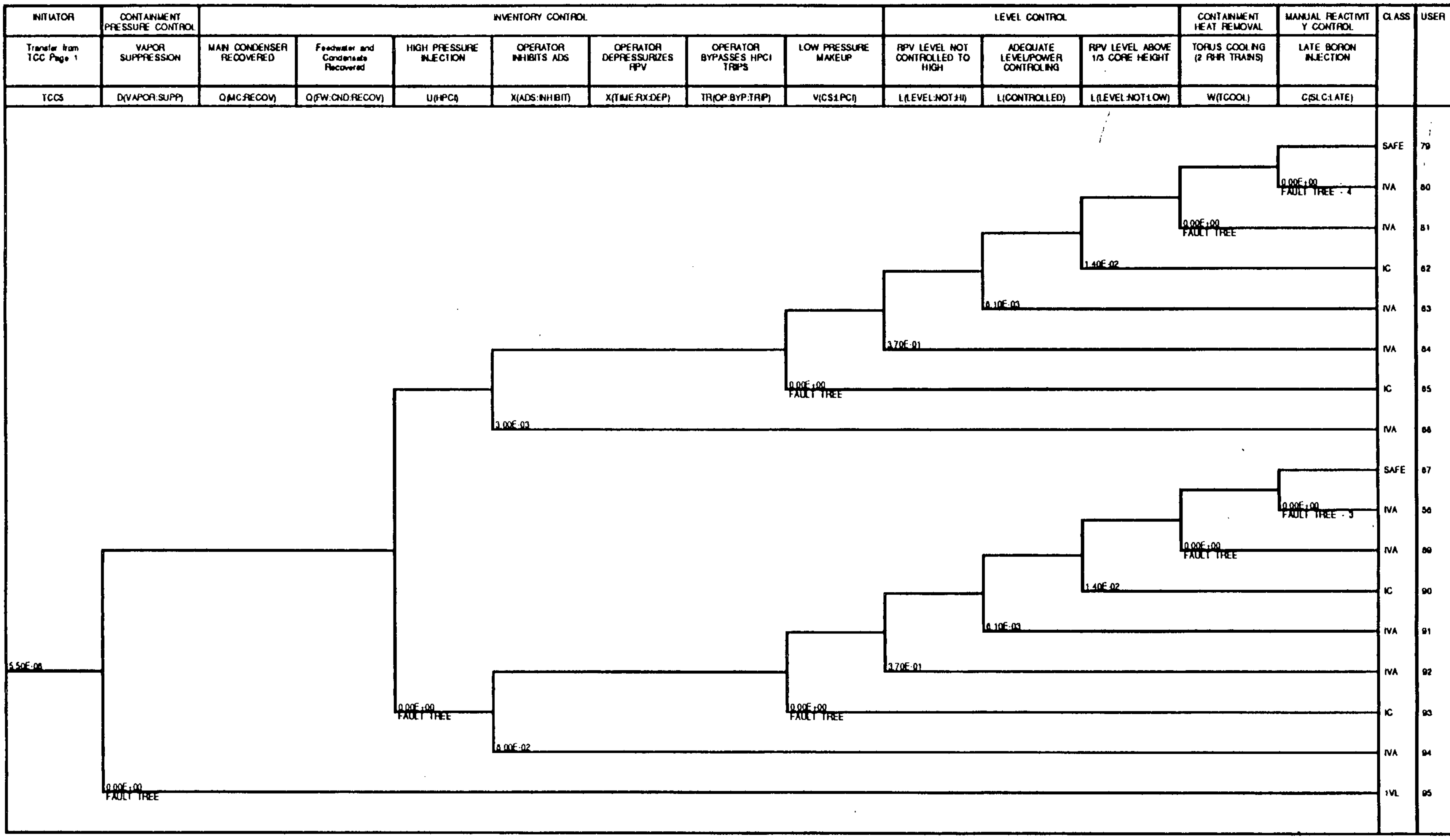


Figure 3.1-16 (Cont.) Loss of Condenser Vacuum ATWS Event Tree

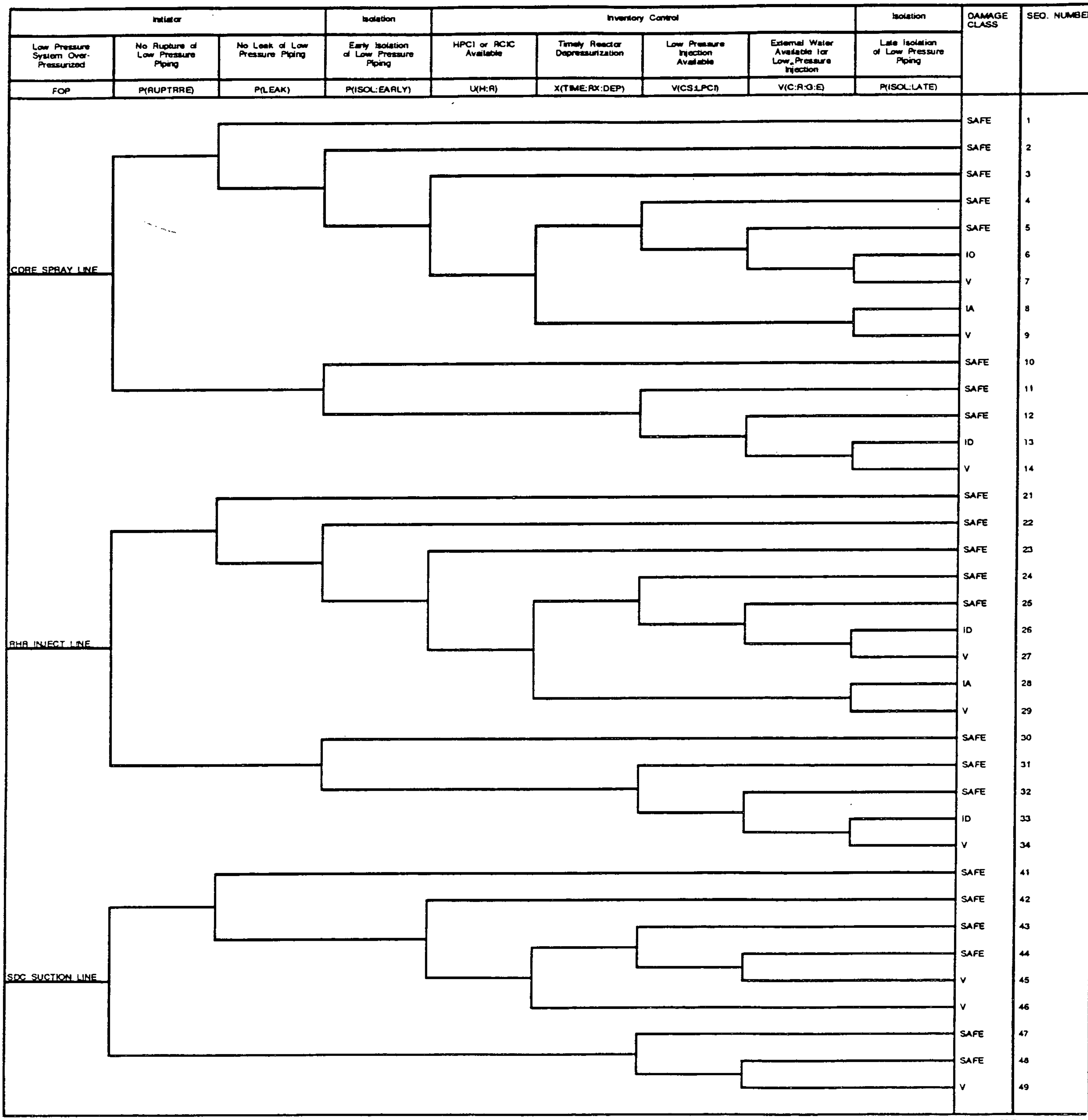


Figure 3.1-17 Interfacing Systems LOCA Event Tree

3.1.3 Accident Sequence Class Description

Due to the similarity of certain accident sequences, with respect to RPV condition and primary containment condition, accident sequences can be grouped into representative classes. For example, transients involving loss of coolant makeup and in which the RPV remains at high pressure are grouped together under a certain accident class; whereas, transients with adequate core cooling but inadequate containment heat removal are grouped under a separate class. In the first case, the RPV and the primary containment are currently intact but core melt is imminent. In the second case, the RPV is initially intact but the primary containment is breached before the onset of core melt.

The grouping of accident sequences into accident classes facilitates both the display of results and the interface between the Level 1 and the Level 2 analyses. The accident classification used in the DAEC IPE is typical of most BWR PRAs and involves the following five general classifications:

1) Transients with loss of coolant makeup
2) Loss of containment heat removal
3) LOCAs
4) ATWS
5) ISLOCA

These five general categories are then further subdivided into more discriminating subclasses. The accident class scheme is summarized in Table 3.1-5 which follows.

Table 3.1-5

ACCIDENT SEQUENCE CLASSIFICATIONS

Accident Class Designator	Subclass	Definition	WASH-1400 Designator Example
Class I	A	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high.	TQUX
	B	Accident sequences involving a station blackout and loss of coolant inventory makeup.	T_EQUV
	C	Accident sequences involving a loss of coolant inventory induced by an ATWS sequence with containment intact.	T_TC_MQU
	D	Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced to 200 psi.; i.e., accident sequences initiated by common mode failures disabling multiple systems (ECCS) leading to loss of coolant inventory makeup.	TQUV
	E	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high and DC power is unavailable.	---
Class II	A	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage induced post containment failure	TW
	L	Accident sequences involving a loss of containment heat removal with the RPV breached but no initial core damage; core damage after containment failure.	AW
	T	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage induced post high containment pressure	N/A
	V	Class IIA or IIL except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	TW

Table 3.1-5

ACCIDENT SEQUENCE CLASSIFICATIONS

Accident Class Designator	Subclass	Definition	WASH-1400 Designator Example
Class III (LOCA)	A	Accident sequences leading to core damage conditions initiated by vessel rupture where the containment integrity is not breached in the initial time phase of the accident.	R
	B	Accident sequences initiated or resulting in small or medium LOCAs for which the reactor cannot be depressurized prior to core damage occurring.	S_1QUX
	C	Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is a low pressure and no effective injection is available.	AV
	D	Accident sequences which are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate, challenging the containment integrity with subsequent failure of makeup systems.	AD
Class IV (ATWS)	A	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post containment failure.	$T_TC_MC_2$
	L	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially breached (e.g., LOCA or SORV); core damage induced post containment failure.	N/A
	T	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post high containment pressure.	N/A
	V	Class IV A or L except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	N/A
Class V	--	Unisolated LOCA outside containment	N/A

3.2.1 System Analysis

This section provides a summary of the 18 systems which were analyzed in the performance of the DAEC IPE effort. The section is broken down into 18 subsections. Each subsection will provide a brief description of the function of the system, the system description, system interfaces and dependencies, system operational constraints and system fault tree models and assumptions used in the analyses. These shall be arranged so that all information pertaining to a system appears in the same subsection.

Data for this section was obtained from the system notebooks. The information contained within them was developed using a process which is summarized in the following paragraphs.

The first step was to gather system data. Typical sources of system data are the Updated Final Safety Analysis Report (UFSAR), Design Basis Documents (DBDs), system descriptions, piping and instrumentation diagrams (P&IDs), system training manuals and the plant operating procedures.

Step two was to perform a preliminary system review. The basic elements of system operation are identified such that the system function, key components and system boundaries are clearly defined and understood. The system parameters for normal operation, abnormal operation and emergency operation are identified. Automatic actuation and alarm setpoints are noted as well. To assist in the information management and ensure a comprehensive review of the system, the system analyst completes a component description and dependency matrix for all components identified as having a dependency on a support system or a requirement for operator action during accident mitigation situations. For the DAEC IPE effort the Systems Engineers were utilized as the systems analysts.

Step three is to perform a familiarization walkdown of each system. It is important to obtain a visual perspective of the system in order to clearly understand its configuration and operation. The scope of the walkdown should include all major active components in the system. Because the DAEC Systems Engineers were used to provide the systems analyst function for the IPE, it can be stated that the systems have been effectively walked down numerous times.

Step four uses Steps 1-3 above to develop a draft system notebook (SNB). The draft system notebook defines the function of the system, success criteria for system operation, description of the system, including P&ID's, system interfaces and dependencies, test and maintenance provisions, etc.

Step five of the system analysis development is to perform a verification walkdown. Again, this walkdown was very efficiently and comprehensively achieved because of the use Systems Engineers for the DAEC IPE. Its purpose is to verify the system configuration, identify sources of common cause failure and evaluate impacts of external events on the system. The system is traced along significant paths rather than being investigated at specific component locations.
The verification walkdown should capture any system specific interactions or special features which may affect the ability of the system to perform in an accident.

Step number six was to provide a complete draft set of SNBs. These were given detailed reviews by the PRA analysts and project personnel in order to achieve an integration and uniformity of information. Although this is listed as step six, a close and continuous communication exists between the PRA Analyst personnel and the Systems Analysts/Engineers during the entire project.

Review of the draft SNBs was step number seven. The draft SNBs received interdisciplinary review from cognizant system engineers as a minimum and plant operations and maintenance personnel as available. The purpose of the review was to verify the overall completeness and accuracy of the information contained in the SNB. To ensure a complete and accurate review, the DAEC IPE project created a review committee made of personnel from a number of plant organizations. The systems analysts/engineers then formally presented their SNBs to the committee.

Step number eight is to issue the final draft SNB. The system analyst/engineer evaluates and dispositions each of the comments received from the draft SNB review process. The system analyst/engineer also makes the necessary changes to the draft SNB following the previous steps as appropriate to close out their disposition of the reviews comments.

The final draft SNB is eligible to receive input from the Plant Model Quantification effort and then be independently checked, reviewed, approved, and issued.

SNB periodic revisions typically may be made as a result of system design changes or through introduction of new documentation.

The following is a list of all systems and the sections where they can be found:

3.2.1.1	Automatic Depressurization and Safety Relief
3.2.1.2	Condensate and Main Condenser
3.2.1.3	Control Rod Drive
3.2.1.4	Core Spray
3.2.1.5	Electric Power (AC/DC)
3.2.1.6	Emergency Service Water
3.2.1.7	Feedwater
3.2.1.8	General Service Water
3.2.1.9	High Pressure Coolant Injection
3.2.1.10	Instrumentation

3.2.1.11 Reactor Core Isolation Cooling
3.2.1.12 Recirculation Pump Trip
3.2.1.13 Residual Heat Removal
3.2.1.14 Residual Heat Removal Service Water
3.2.1.15 River Water
3.2.1.16 Standby Liquid Control
3.2.1.17 Torus/Torus Vent (includes Vapor Suppression)
3.2.1.18 Well Water

3.2.1.0 Generic Modeling Assumptions

The following generic assumptions were used in the development of all the system fault tree models:

1. Valves which are normally open and are not required to change state will be modeled with the failure to remain open for the mission time. Similarly, valves which are normally closed and are not required to change state will be modeled with the failure to remain closed for the mission time. Plugging of valves or valve stem-disk separation causing plugging, or plugging of piping will in general not be modeled. However, if a valve is not flow tested, the interval between flow tests is more than 18 months, the valve is in a "dirty" (seawater or borated water) system, or the plugging event is a single failure of the system, then plugging may have to be modeled.

2. False signals producing erroneous component operations are not modeled. Such failures are assumed to be inherent in the component failure rate data since such failures are likely to have been included. Consequently, events with "spurious actions" will be modeled as part of the component failure.

3. Instrumentation devices (flow, temperature, pressure) which do not provide direct input to system operation are not to be modeled, unless they would block flow of the line if plugged.

4. Passive failures of piping and electric wiring are not modeled, except in special cases (e.g., interfacing systems LOCA).

5. Major flow diversions (flow directed along an incorrect flow path of comparable size to the source flow (4/5 diameter) will be modeled. Minor flow diversion through small instrumentation lines or mini flow lines less than 1/5 diameter of source flow path are not modeled.

6. Exclusive maintenance unavailability cutsets not allowed by the Technical Specifications will be automatically edited out using the post processing edit of the cutsets to eliminate mutually exclusive events.

7. Locked open valves and locked closed valves are included in the fault tree model with a failure rate of zero.

8. Instruments, and their dependencies, for automatic actuations for emergency systems are modeled separately in Section 3.2.1.10.

3.2.1.1 Automatic Depressurization and Safety Relief

3.2.1.1.1 System Function

The Automatic Depressurization System (ADS) operates in conjunction with the low-pressure core cooling systems to ensure adequate core cooling. The ADS provides depressurization of the reactor vessel in the event that the high-pressure core cooling systems cannot maintain adequate reactor vessel water level so that the low pressure core cooling systems may inject into the reactor vessel.

The following design bases are incorporated into the ADS:

a. The ADS provides automatic depressurization of the reactor vessel in the event of small breaks concurrent with failure of high pressure injection systems to allow injection by the low pressure systems.

b. Actuation of the ADS does not require any source of off-site power.

c. The ADS logic permits testing of the system.

d. Safety/Relief Valves (SRVs) and accumulators are located within primary containment in order to satisfy containment isolation requirements.

e. The ADS short term capability is five cycles at drywell design pressure after a period of five hours post-LOCA (original design basis).

f. The ADS long term capability is five cycles at 100 days following a design basis LOCA (extended licensing basis, added by NUREG-0737).

g. The ADS is capable of performing its function taking no credit for non-safety systems.

The ADS utilizes four of six safety/relief valves (SRVs). These four valves receive an open signal after a two-minute time delay which starts when reactor vessel level drops to Level 1 (there is a confirmatory Level 3 signal also used in the logic) and discharge pressure is sensed from any low-pressure ECCS pump.

The four Automatic Depressurization System Safety/Relief Valves and their flow paths are shown on Figure 3.2-1. The flow paths and main steamline connections for the two non-ADS Safety/Relief Valves and the two Code Safety Valves are also shown.

Each ADS SRV is connected to a separate main steamline between the reactor vessel and the Main Steam Isolation Valves. The discharge from each relief valve is piped to the suppression pool through T-Quenchers. The T-Quenchers dissipate the steam below the water level of the suppression pool.

After an SRV reseat, the steam remaining in the discharge line will condense, forming a vacuum which could draw water from the suppression pool into the discharge piping. If the SRV opened again under these conditions, excessive back pressure would be exerted on the valve discharge. To prevent this occurrence, a vacuum breaker relief valve is installed in each discharge line which opens to atmosphere when a vacuum is sensed.

Automatic actuation (opening) of the four ADS SRVs requires both an actuation signal and a nitrogen supply. The nitrogen is supplied by the Containment Atmospheric Control

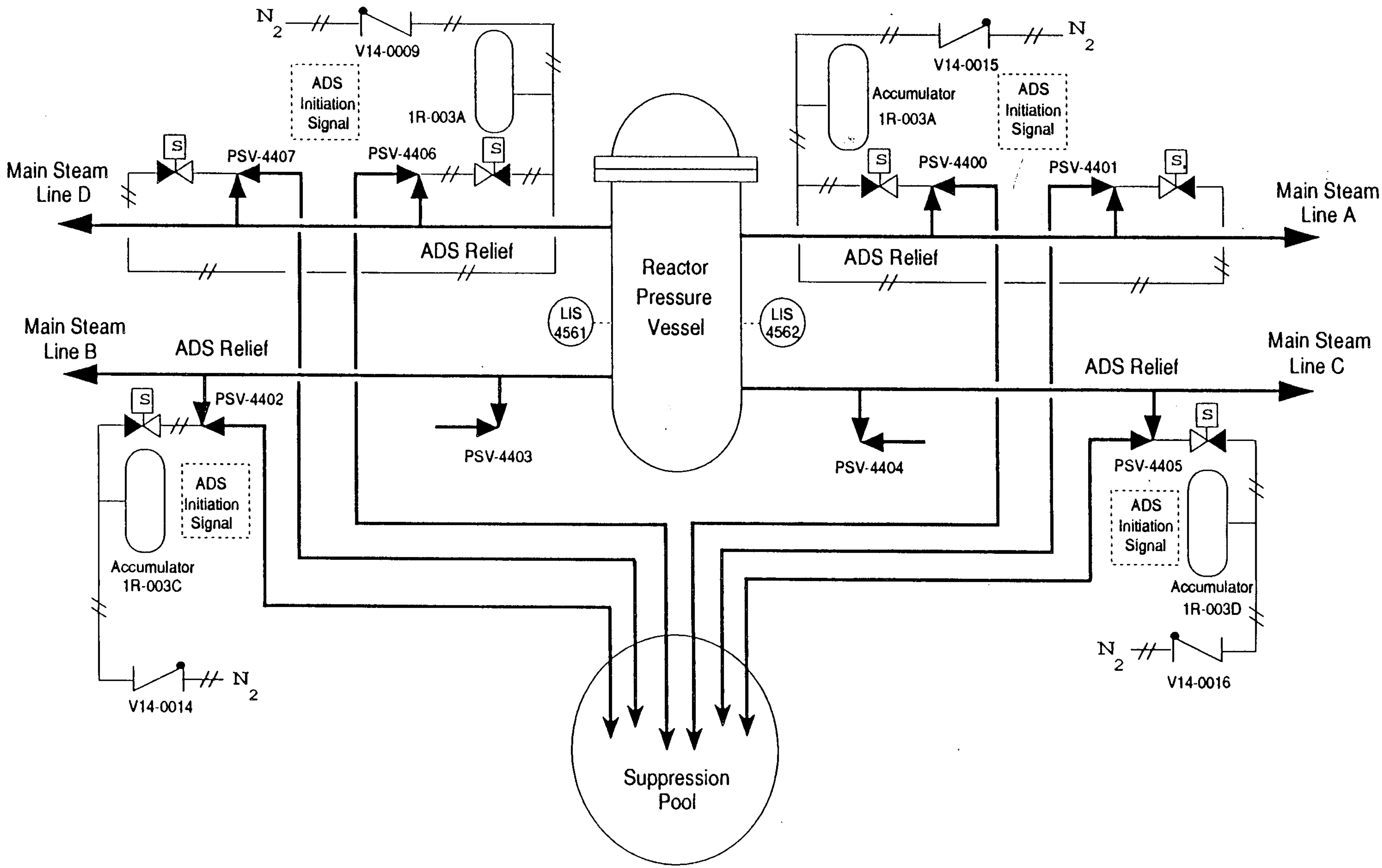


Figure 3.2-1
Automatic Depressurization System

System. A solenoid operated valve, which is energized by either an ADS initiation signal or by a handswitch on Panel 1C03, controls nitrogen flow to its respective ADS SRV.Nitrogen accumulators tap into the nitrogen supply line upstream of each solenoid valve. The accumulators provide a standby pressure source for SRV operation in the event that the normal nitrogen supply is lost. A check valve installed in each nitrogen supply line ensures that the accumulator will not become depressurized by a piping break upstream of the check valve. ADS SRV PSV-4402 and PSV-4405 have dedicated accumulators, while ADS valve PSV-4400 shares an accumulator with non-ADS SRV PSV-4401, and ADS valve PSV-4406 shares an accumulator with non-ADS SRV PSV-4407. These accumulators are greatly oversized and will supply both ADS and non-ADS SRV pneumatic requirements with significant safety margin.

The four ADS SRVs and two non-ADS SRVs can also be remotely opened manually via a handswitch in the control room. Actuation in this manner also requires a nitrogen supply.

The overpressure protection of the nuclear boiler system is accomplished by the NSSS Pressure Relief System which consists of:

- Six Target Rock, two-stage, Main Steam Safety/Relief Valves (SRVs).

- Two Dresser Code Safety Valves.

The six SRVs have a combined capacity of 68.4% (the minimum design capacity is 61.9%).

The two safety valves have a combined capacity of 18.7% (the minimum design capacity is 10.0%).

The safety design basis of the NSSS Pressure Relief System is to:

1. Prevent overpressurization of the nuclear system.

The power generation design basis is to:

1. Prevent the opening of the safety valves during normal plant isolations and load rejections.

2. Not discharge directly into the drywell.

3. Reclose after they function so that normal operations can be resumed.

3.2.1.1.2 System Interfaces and Dependencies

The Automatic Depressurization System (ADS) Fault Tree Model includes support systems required for the ADS system to function in postulated accident scenarios. The systems which support specific ADS components and their effects on ADS operation are identified in the Dependency Matrix shown in Table 3.2-1.

In addition to the systems identified on a component level in the Dependency Matrix, the following systems provide support functions to the ADS system.

Table 3.2-1

AUTOMATIC DEPRESSURIZATION SYSTEM DEPENDENCY MATRIX

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
PSV-4400	Pressure Safety Relief Valve 4400	Drywell	Closed	-	-	Normal: 1D13 Alternate: 1D23	Valve fails to open	N2	None; accumulator available			ADS Auto Signal Start
PSV-4401	Pressure Safety Relief Valve 4401	Drywell	Closed	-	-	Normal: 1D13 Alternate: 1D23	Valve fails to open	N2	None; accumulator available			LLS Auto Initiated
PSV-4402	Pressure Safety Relief Valve 4402	Drywell	Closed	-	-	Normal: 1D13 Alternate: 1D23	Valve fails to open	N2	None; accumulator available			ADS Auto Signal Start
PSV-4405	Pressure Safety Relief Valve 4405	Drywell	Closed	-	-	Normal: 1D13 Alternate: 1D23	Valve fails to open	N2	None; accumulator available			ADS Auto Signal Start
PSV-4406	Pressure Safety Relief Valve 4406	Drywell	Closed	-	-	Normal: 1D13 Alternate: 1D23	Valve fails to open	N2	None; accumulator available			ADS Auto Signal Start
PSV-4407	Pressure Safety Relief Valve 4407	Drywell	Closed	-	-	Normal: 1D23 Alternate: 1D13	Valve fails to open	N2	None; accumulator available			LLS Auto Initiated
ADS Logic "A"	ADS Initiation Signal Logic Train "A"	Drywell	Standing by for initiating signal	-	-	Normal: 1D13	Loss of auto start signal					Reactor Vessel Low Water Level L3 confirmatory & L1 LPCI or CS discharge pressure 2 minute timer

Table 3.2-1

AUTOMATIC DEPRESSURIZATION SYSTEM DEPENDENCY MATRIX
(continued)

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
ADS Logic "B"	ADS Initiation Signal Logic Train "B"	Drywell	Standing by for initiating signal	-	-	Normal: 1D23 Alternate: 1D13	Loss of auto start signal					Reactor Vessel Low Water Level L3 confirmatory & L1 LPCI or CS discharge pressure 2 minute timer
1K14	Nitrogen Compressor		Running	1B43	Compressor fails to run							
CV-4378A	Nitrogen Control Valves 4378A		Open	-	-	-	-	ISA	Valve fails closed	-	-	Closes on Gp. 3 isolation
CV-4378B	Nitrogen Control Valves 4378B		Open	-	-	-	-	ISA	Valve fails closed	-	-	Closes on Gp. 3 isolation
SV-4378A	Pilot valve for CV-4378A	Reactor Building	Closed	1Y11	Valve vents closing CV-4378A							Vents on Gp. 3 isolation
SV-4378B	Pilot valve for CV-4378B	Reactor Building	Closed	1Y21	Valve vents closing CV-4378B							Vents on Gp. 3 isolation

Main Steam System

The four ADS Safety/Relief Valves, as well as two non-ADS Safety/Relief Valves and two Safety Valves, are mounted on the main steamlines between the reactor vessel and the Main Steam Isolation Valves. There is one ADS Safety/Relief Valve connected to each of the four main steamlines within the Primary Containment.

Primary Containment System

ADS and non-ADS Safety/Relief Valves discharge to the suppression pool, where the relatively cool suppression pool water condenses the discharge steam. The two relief valves discharge into the drywell.

Containment Atmospheric Control System

Nitrogen supplied from the Containment Atmospheric Control System is provided for manual actuation of the ADS and non-ADS Safety/Relief Valves. Nitrogen is used as the actuating medium for the ADS valves during the automatic depressurization mode of operation, and for the non-ADS Safety/Relief Valves during pressure relief operation.

Core Spray System

A Core Spray pump discharge pressure signal provides a permissive for ADS initiation.

Residual Heat Removal System

A LPCI pump discharge pressure signal provides a permissive for ADS initiation.

3.2.1.1.3 System Fault Tree Model Assumptions

This section describes any assumptions specific to the system fault tree.

The following assumptions were used in the development of the ADS System fault tree model:

1. Two of the six safety/relief valves are assumed adequate to satisfy the depressurization functions. This tree models both automatic and manual depressurization.

2. Containment Nitrogen Air from either the accumulator or the supply system is necessary to keep the SRV valves open for the depressurization function.

3. This tree does not model the ECCS initiating logic. However, the low pressure pump permissives are explicitly modeled.

3.2.1.1.4 Success Criteria

The Automatic Depressurization System (ADS) functions to provide depressurization of the reactor. The depressurization function may be required for those events that do not directly result in immediate depressurization of the reactor. For these events, the reactor remains pressurized and requires the operation of high pressure injection systems to maintain vessel inventory. If all sources of high pressure injection are unavailable, manual reactor depressurization using the 4 ADS safety relief valves (SRVs) and the two non-ADS SRVs is required to allow operation of the low pressure injection systems. Manual de-pressurization is required since the EOPs direct the operators to "block" automatic ADS actuation.

Success of the SRVs depressurization function is defined as opening of a sufficient number of SRVs to reduce the reactor pressure so that the low pressure ECCS systems (Core Spray and LPCI) can operate to provide core cooling. ADS operation during large LOCAs is not required. Operation of two of six SRVs adequately depressurizes the reactor for all Non-ATWS events.

For ATWS events, the SRVs function to provide overpressure protection in conjunction with the two safety valves. Overpressure protection requires operation of 7 of the 8 SRVs and SVs. Overpressure protection may also be required during transient events and events involving inventory losses that are bounded by a small LOCA if the main condenser is not available. In this case, operation of 3 SRVs is adequate.

DC control power is required for all SRV operations. Normal and backup 125 VDC power for the ADS logic circuits and operation of the Safety/Relief Valves is provided form the two plant 125 VDC battery systems. 125 VDC battery 1D1 normally supplies power for all SRVs.

The operation of the four ADS SRVs, can be automatic or manual. The remaining two SRVs are manual. Automatic operation requires actuation and permissive signals:

- low reactor water level (Level 3 - confirmatory signal)
- low reactor water level (Level 1)
- discharge pressure in any Core Spray or LPCI injection flow path
- two minute timer time-out

3.2.1.2 Main Condenser System

3.2.1.2.1 System Function

Figure 3.2-2 shows a simplified one line diagram of the Main Condenser System. Components drawn solid are components actually modelled; however, those drawn with dashed lines are provided on the diagram for information only. The Main Condenser functions to (1) serve as a heat sink for the low pressure turbine exhaust steam, (2) serve as a heat sink for the high pressure turbine bypass steam, (3) serve as a collecting point for system drains such as feedwater heater condensate, and (4) remove non-condensible gases from the condensate. The Main Condenser is designed to operate continuously at full power.

The combined storage capacity of the low pressure hotwell and high pressure hotwell is based on a five minute, full power condensate flow which amounts to 72,500 gallons. Baffling in the hotwell provides a two minute retention time for decay of short-lived radioactivity.

The Main Condenser and Main Condenser Air Removal Systems basically consist of a high pressure and a low pressure condenser shell, their associated hotwells, two steam jet air ejector (SJAE) units, and a vacuum pump and separator unit.

In order to make the steam cycle efficient, the low pressure turbines exhaust to a vacuum. The vacuum is maintained during operation primarily by the volume reduction that occurs as steam is condensed by rejection of heat to the Circulating Water flowing through the condenser tubes. The condensate formed is then returned to the reactor to complete the cycle. Since the condenser is not air tight, and because non-condensible gases are produced in the reactor during operation; air removal equipment is needed to establish and maintain the vacuum in the condenser.

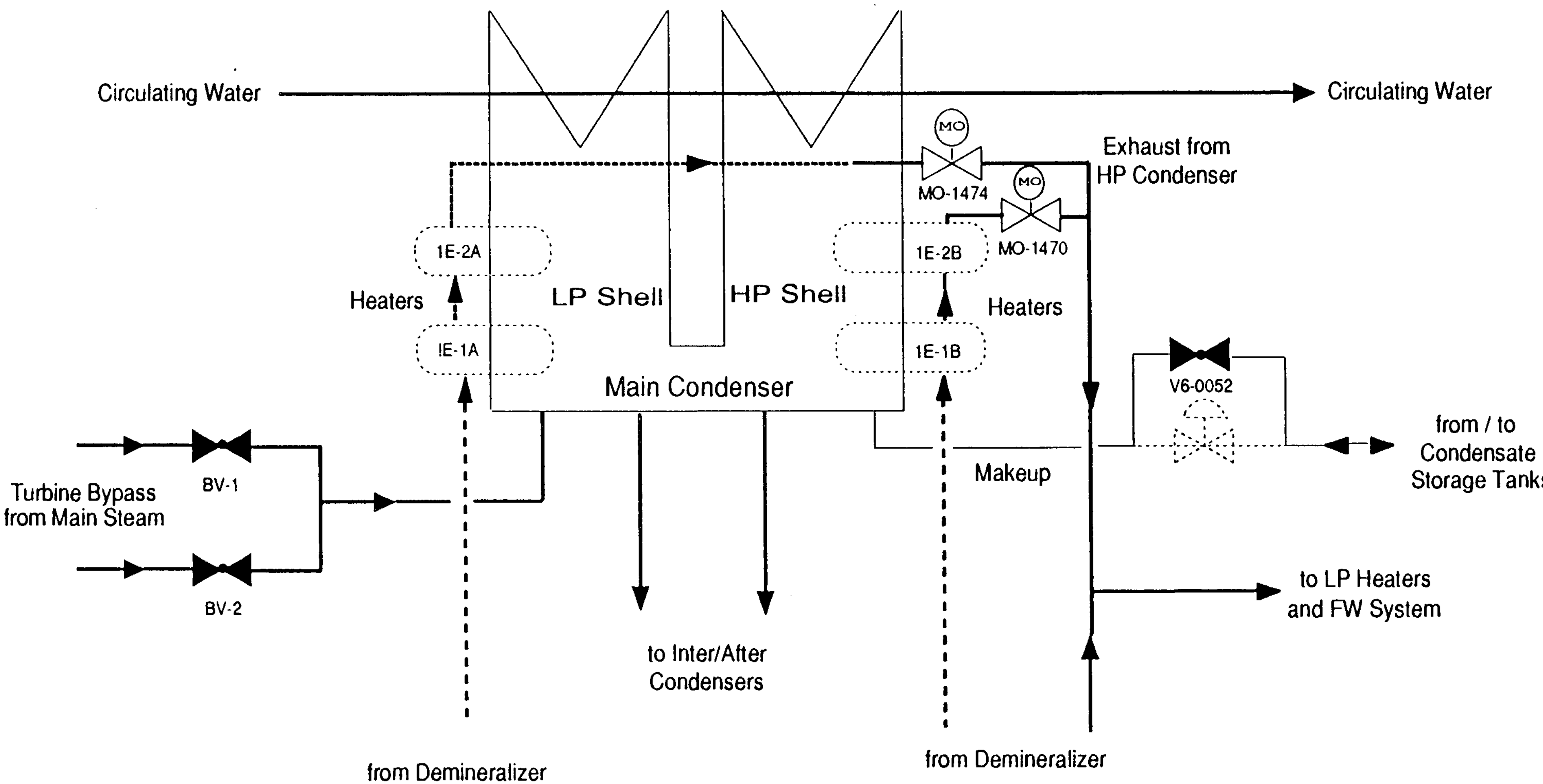


Figure 3.2-2
Main Condenser System

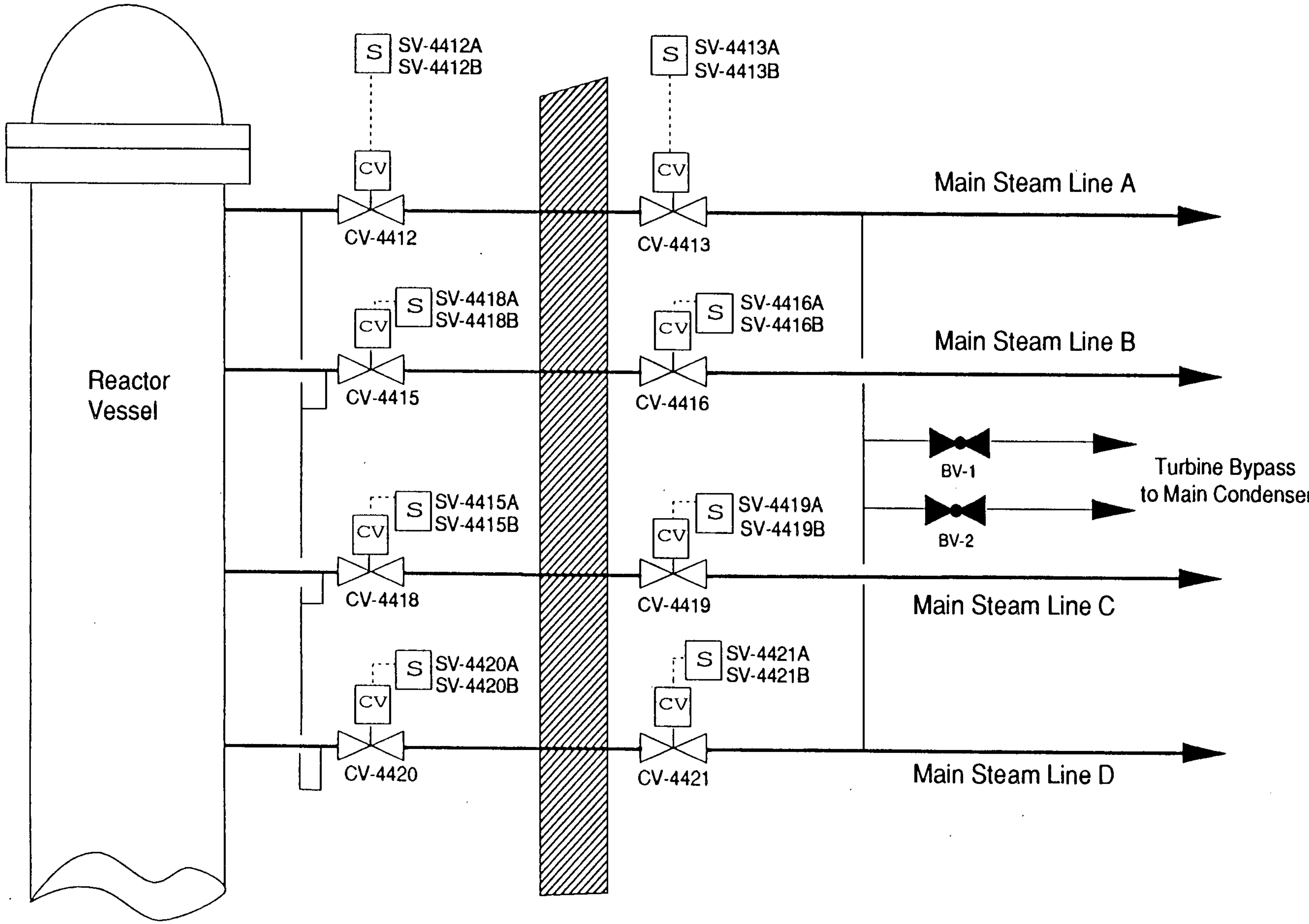


Figure 3.2-3

Main Steam Supply

The major components associated with this system are:

- Main Condenser
- Steam Jet Air Ejectors
- Condenser Vacuum Pump

The SJAEs and Condenser Vacuum Pump are not explicitly modeled.

Main Condenser

The Main Condenser is a horizontal, twin shell, single pass, dual pressure, surface condenser. Each of the two low pressure turbines exhausts to only one of the two condenser shells. Circulating Water is supplied to the condenser shells in a series arrangement which causes water supplied to the second shell to be at a higher temperature than that supplied to the first shell. Steam therefore condenses in the second shell (the high pressure shell) at a higher temperature and pressure than it does in the first shell (the low pressure shell). Although the two shells are physically connected, a water seal between them allows a differential pressure to be maintained.

Each hotwell is provided with baffles to form a labyrinth condensate flow path. This flow path ensures a condensate retention time of at least two minutes to permit decay of short-lived radioactive isotopes.

The turbine exhaust hoods and condenser are protected against excessive steam pressure by atmospheric relief diaphragms. Two relief diaphragms are provided in each low pressure hood that relieve to atmosphere if high exhaust pressures of 5 psig are encountered.

Steam Jet Air Ejectors

Two twin element, two stage, steam jet air ejectors (SJAEs) 1E-8A and 1E-8B are provided. The SJAEs are normally operated with one-half of 1E-8A taking a suction on one of the condenser shells, and one-half of 1E-8B taking a suction on the other condenser shell. Steam supply valves and air inlet valves are provided to isolate the elements not in service.

Condenser Vacuum Pump

A single mechanical vacuum pump is provided to evacuate the turbine and Main Condenser shells during plant startup. One vacuum pump suction line services each Main Condenser shell through a butterfly valve and a manual isolation valve. The vacuum pump discharges through a separator and a delay line to the Offgas Stack.

3.2.1.2.2 System Interfaces and Dependencies

The Main Condenser fault tree model includes support systems required for the Main Condenser to function in postulated accident scenarios. The systems which support specific Main Condenser components and their effects on Main Condenser operation are identified in the Dependency Matrix shown in Table 3.2-2. The condensate pumps are evaluated as part of the Main Feedwater/Condensate System.

3.2.1.2.3 System Fault Tree Model Assumptions

This section describes any assumptions specific to the system fault tree.

The following assumptions were used in the development of the Main Condenser fault tree model:

Table 3.2-2

MAIN CONDENSER DEPENDENCY MATRIX

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
SV-4412A	MSIV 4412 Solenoid Valve A	Cont.	Open	RPS 1200 Ch. A	Solenoid valve returns to vent position							Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.
SV-4412B	MSIV 4412 Solenoid Valve B	Cont.	Open			1D13	Solenoid valve returns to vent position					Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.
SV-4413A	MSIV 4413 Solenoid Valve A	2-G	Open	RPS 120V Ch. B	Solenoid valve returns to vent position							Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.
SV-4413B	MSIV 4413 Solenoid Valve B	2-G	Open			1D23	Solenoid valve returns to vent position					Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.
SV-4415A	MSIV 4415 Solenoid Valve A	Cont.	Open	RPS 120V Ch. A	Solenoid valve returns to vent position							Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.
SV-4415B	MSIV 4415 Solenoid Valve B	Cont.	Open			1D13	Solenoid valve returns to vent position					Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.
SV-4416A	MSIV 4416 Solenoid Valve A	2-G	Open	RPS 120V Ch. B	Solenoid valve returns to vent position							Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.
SV-4416B	MSIV 4416 Solenoid Valve B	2-G	Open			1D23	Solenoid valve returns to vent position					Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.
SV-4418A	MSIV 4418 Solenoid Valve A	Cont.	Open	RPS 120V Ch. A	Solenoid valve returns to vent position							Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.

Table 3.2-2

MAIN CONDENSER DEPENDENCY MATRIX
(continued)

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
SV-4418B	MSIV 4418 Solenoid Valve B	Cont.	Open			1D13	Solenoid valve returns to vent position					Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.
SV-4419A	MSIV 4419 Solenoid Valve A	2-G	Open	RPS 120V Ch. B	Solenoid valve returns to vent position							Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.
SV-4419B	MSIV 4419 Solenoid Valve B	2-G	Open			1D23	Solenoid valve returns to vent position					Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.
SV-4420A	MSIV 4420 Solenoid Valve A	Cont.	Open	RPS 120V Ch. A	Solenoid valve returns to vent position							Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.
SV-4420B	MSIV 4420 Solenoid Valve B	Cont.	Open			1D13	Solenoid valve returns to vent position					Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.
SV-4421A	MSIV 4421 Solenoid Valve A	2-G	Open	RPS 120V Ch. B	Solenoid valve returns to vent position							Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.
SV-4421B	MSIV 4421 Solenoid Valve B	2-G	Open			1D23	Solenoid valve returns to vent position					Low Rx level, low cond. vac., HI steam line tunnel temp., low turb. press.
COND	Condenser									Circ. Water		
CV-4412	Line A INBD MSIV	Cont.	Open					ISA	Valve Closes			

Table 3.2-2

MAIN CONDENSER DEPENDENCY MATRIX
(continued)

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
CV-4413	Line A OUTBD MSIV	2-G	Open					ISA	Valve Closes			
CV-4415	Line B INBD MSIV	Cont.	Open					ISA	Valve Closes			
CV-4416	Line B OUTBD MSIV	2-G	Open					ISA	Valve Closes			
CV-4418	Line C INBD MSIV	Cont.	Open					ISA	Valve Closes			
CV-4419	Line C OUTBD MSIV	2-G	Open					ISA	Valve Closes			
CV-4420	Line D INBD MSIV	Cont.	Open					ISA	Valve Closes			
CV-4421	Line D OUTBD MSIV	2-G	Open					ISA	Valve Closes			
Z-CIRC-WATER	Circulating Water System		Operating	1A1 1A2	Loss of Circ Water	1D11 1D21	Loss of breaker control power	-	-	-	-	Trips on loss of bus voltage

1. The condenser/heat removal tree is used to assess the availability of the condenser as a heat sink for decay heat late in an accident event. Therefore, it is assumed that the condenser is initially unavailable and must be reestablished (at least one MSIV or turbine bypass valve reopened).

2. The condenser is assumed to fail if: 1) no steam supply line is available from the reactor, 2) the circulating water system fails, 3) there is a loss of condenser vacuum, or 4) if the condensate system fails to remove water from the hotwell, thereby filling the hotwell and covering the condenser tubes.

3. Failure of the air removal function of the steam jet air ejectors is assumed to be included in the loss of condenser vacuum basic event.

4. Gravity feed from the condensate tank to the hotwell provides adequate inventory to the condenser to maintain its availability as a reactor makeup source from transient, medium LOCA and small LOCA events.

3.2.1.2.4 Success Criteria

The main condenser can be used as a makeup source and as an external heat sink. The use of the Power Conversion System (PCS) as a method of containment heat removal is possible if at least one main steam line path can be maintained and there is not a large diversion of reactor decay heat directly to the suppression pool. For the PCS to successfully transfer decay heat from the containment to the ultimate heat sink the following equipment are required to be available:

- One complete feedwater condensate piping system is operable and able to deliver water from the condenser hotwell to the reactor vessel. This requires that the condensate and feedwater pumps in the piping system be operable or the condensate pump be operable, and that the operator reduces reactor pressure to below 540 psia by using the relief valves.

- The main steam line isolation valves in one of the four main steam lines must remain open (or be reopened if closed as a result of the initiating transient). The turbine bypass valves must open. If condenser vacuum falls below seven inches of Hg, the low vacuum interlocks on bypass valves must be overridden.

- At least one of the main condenser circulating water pumps must be operable and delivering cooling water to the main condenser to maintain adequate condenser vacuum.

In the event condenser vacuum or steam supply to the condenser is lost, makeup is possible from the condensate storage tanks. This makeup flow path requires operator action to locally open a manual valve. The resulting makeup flow rate is limited to that available using only gravity feed through the 6" line. The condensate storage tanks are located on plant grade elevation while the hotwell is approximately 20 feet below grade. Since the flow rate is limited, this makeup supply is credited only for transient events and events involving inventory losses that are bounded by a medium LOCA.

3.2.1.3 Control Rod Drive System

3.2.1.3.1 System Function

The purpose of the Control Rod Drive System is to position the control rods in the reactor core to control core reactivity and power density. The system responds to a manual control signal and is designed to prevent control rods from withdrawing as a result of a single malfunction. In addition, the system is also designed to quickly shut down the reactor (scram) in emergency situations by rapidly inserting all withdrawn control rods into the core in response to manual or automatic signals. In the event of a transient or small break LOCA, vessel injection via the CRDs could be used as a backup to HPCI/RCIC.

The CRD System consists of the control rod drive mechanisms, the hydraulic control units (HCUs), the scram accumulators, the scram discharge volume, the drive water pumps, and associated flow control valves, filters and piping.

The control rod drive housing supports protect against additional damage to the nuclear system process barrier or damage to the fuel barrier by preventing any significant nuclear transient in the event a drive housing breaks or separates from the bottom of the reactor vessel.

The Control Rod Drive System is designed as a seismic class 1 system and meets the following safety design bases:

a. Design provides for a sufficiently rapid control rod insertion so that no fuel damage results from any abnormal operating transient.

b. Design includes the ability to individually support and position a control rod.

c. Only one drive at a time can be withdrawn.

d. The circuitry provided for the manipulation of control rods is designed so that no single failure can negate the effectiveness of a reactor scram.

e. The design of the system is such that the repair, replacement, or adjustment of any failed or malfunctioning component does not require that any element needed for reactor scram be bypassed unless a bypass of that element is normally allowed.

f. Control rod downward motion is limited, following a postulated CRD housing failure, so that any resulting nuclear transient would not be sufficient to cause fuel damage.

The CRD System hydraulically operates the CRD mechanism using processed Condensate water as hydraulic fluid (refer to Figure 3.2-4). The CRD drive water pumps take suction from the Condensate Storage Tanks or the Condensate reject line and discharge water at a higher pressure than reactor pressure. This water is directed to the flow control station via filters and strainers. The flow control station maintains system flow at a constant rate. This is required to help maintain the drive water and cooling water header pressures constant. A tapoff just upstream of the flow control station is provided to charge the scram accumulators.

The constant flow rate that has been established is directed to the drive water header. The pressure in this line is maintained constant by the drive water pressure control station. The drive water header is connected to all HCUs and this line provides insert and withdrawal water necessary for actual operation of the CRD mechanisms. Drive water

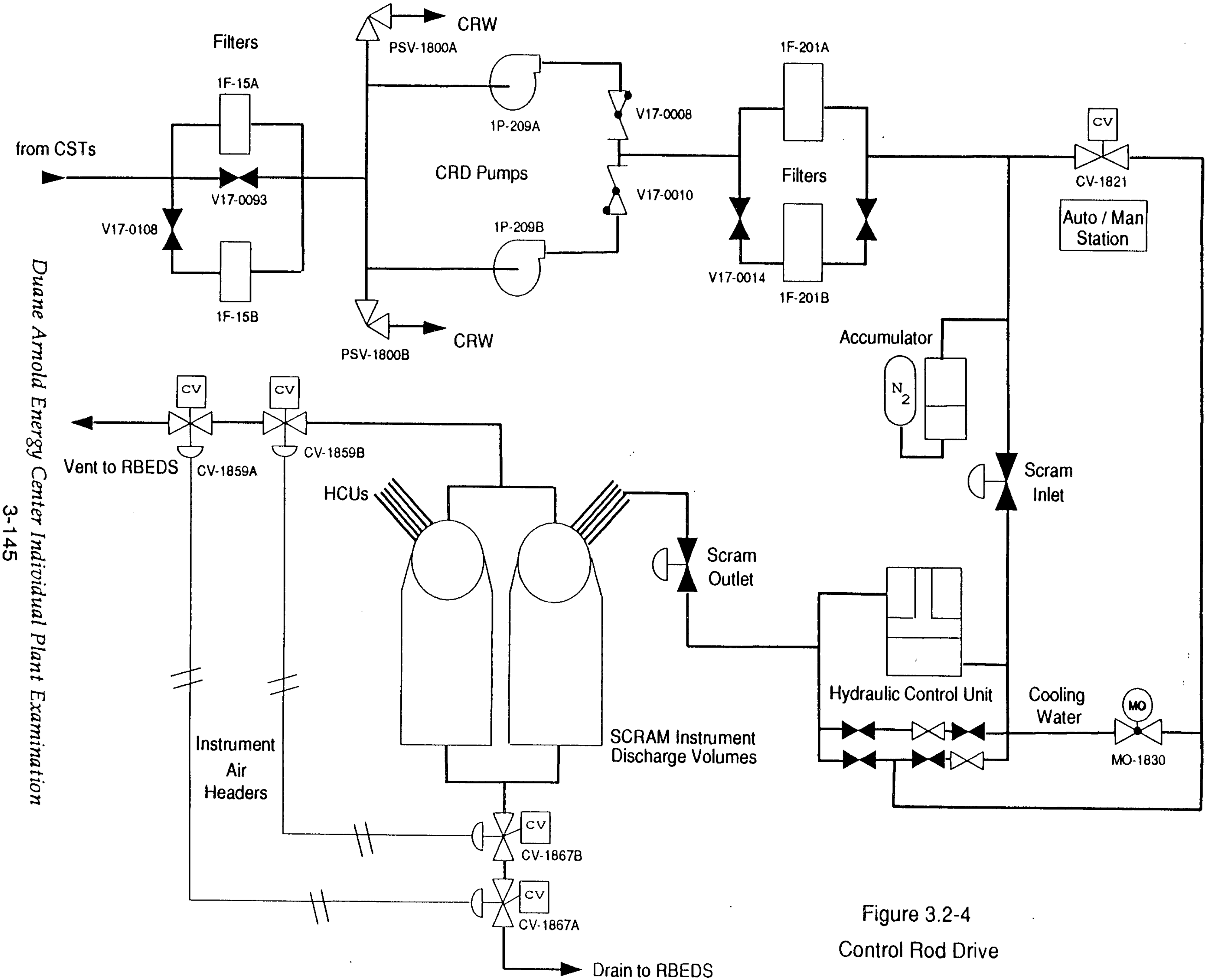


Figure 3.2-4
Control Rod Drive

flow is directed to insert or withdraw the CRD by a four way valve system located on each HCU. These valves direct drive water flow above or below the drive water piston as required.

Cooling water for the HCUs is taken from downstream of the drive water pressure control station and enters the body of the inlet scram valve and flows to the CRD mechanism via the insert line. Cooling water flow enters the insert port and follows various seal leakage paths through the mechanism and eventually makes its way into the reactor vessel.

Since constant flows and pressures are so important to this system, stabilizing valves are installed to help minimize pressure transients during rod motion. These valves have integral flow control devices manually set to correspond with insertion flow rate and withdraw flow rate. During an insert operation, the insert stabilizing valve will shut; during a withdraw operation, the withdraw stabilizing valve will shut. This will maintain overall system flow constant during rod motion.

Also included in the system are scram accumulators and a scram discharge volume. The scram accumulators serve as independent sources of energy to ensure insertion of control rods during a scram. The scram discharge volume provides a low pressure reservoir to contain the water exhausted from all CRDs during a scram. The accumulators and discharge volume are isolated from the CRDs by scram inlet and outlet valves which are solenoid controlled air operated valves.

This system is controlled both automatically and manually. Instrumentation and controls necessary for system startup, testing, and manual adjustments are located in the Control Room.

The major components of the CRD and Hydraulic System are:

- Control Rod Drive Mechanisms
- Hydraulic Control Units
- Hydraulic Supply Subsystem
- Control Rod Drive Housing Supports

A. Control Rod Drive (CRD) Mechanisms - The CRD mechanisms used for positioning the control rods are double acting, mechanically latched, hydraulic cylinders which are individually mounted on the bottom head of the reactor pressure vessel.

B. Hydraulic Control Units (HCUs) - Each hydraulic control unit furnishes pressurized water, on demand, to a drive unit. Each combines all operating valves and components required for the normal or scram positioning of the drive mechanism.

The HCU uses differential hydraulic pressures to insert or withdraw a control rod, and to provide cooling water to the mechanism itself. The solenoid operated valves that control normal movement of the drive are controlled by the Reactor Manual Control System. The solenoid valves associated with scram actions are interconnected with the Reactor Protection System.

C. Hydraulic Supply Subsystem - The hydraulic supply subsystem supplies water at the pressures and flows required by the HCUs. Water is pumped from the Condensate reject line via a network of filters, strainers, flow and pressure regulating devices.

D. Control Rod Drive Housing Supports - Horizontal beams are installed immediately below the bottom head of the reactor vessel between the rows of CRD housings. These beams are bolted to brackets that are welded to the steel form liner inside the reactor vessel support pedestal. Hanger rods are supported from the beams on stacks of disc springs. Support bars are bolted between the bottom ends of the hanger rods. Individual grids rest on the support bars between adjacent beams.

The design ensures that control rod movement following a housing failure is limited to a maximum of approximately three inches in the worst case, substantially less than the normal six inches of movement associated with a notch withdrawal.

3.2.1.3.2 System Interfaces and Dependencies

The Control Rod Drive (CRD) Mechanisms and Hydraulic Supply System fault tree model includes support systems required for this system to function in postulated accident scenarios. The systems which support specific CRD mechanism components and their effects on CRD operation are identified in the Dependency Matrix shown in Table 3.2-3.

In addition to the systems identified on a component level in the dependency matrix, the following systems provide support functions to the CRD system:

Condensate System

The CRD pumps take their suction from the Condensate reject line to the CSTs. During periods when at least one of the condensate pumps is in operation, low conductivity, de-aerated rated water will be supplied for CRD System use. Two flow orifices are provided to reduce the pressure from approximately 420 psig to 20 psig. When neither condensate pump is in operation, an air operated solenoid controlled valve isolates the normal supply

Table 3.2-3

CONTROL ROD DRIVE SYSTEM DEPENDENCY MATRIX

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
1P-209A	CRD Water Pump A	Containment	Running[1]	1A3	Pump fails to start/run	1D13	Loss of Control Power	-	-	RBCCW	-	-
1P-209B	CRD Water Pump B	Containment	Running[1]	1A4	Pump fails to start/run	1D23	Loss of Control Power	-	-	RBCCW	-	-
CV-1821	Control Valve 1821	Containment	Open	-	-	-	-	ISA	Valve fails closed	-	-	-

1. Normal condition for CRD pumps is one running while the other is in standby.

line to prevent draining the reject line and the CRD hydraulic supply is from the CSTs. Under most conditions, the drive pump in service removes approximately 60 gpm from this system; 40 gpm is utilized by the CRD Hydraulic System, 8 gpm is used for seal-purge water in the Reactor Recirculation System and the remainder is returned to the CSTs via the minimum flow bypass.

Reactor Vessel

The CRD mechanisms themselves each represent a reactor vessel penetration in the lower vessel head. A pressure signal is also provided from the reactor vessel to the drive water and cooling water differential pressure instruments.

Reactor Recirculation System

The CRD Hydraulic System supplies seal-purge water to both recirculation pumps. The seal-purge water tapoff is located between the CRD pump discharge filters and the flow nozzle.

Reactor Building Closed Cooling Water Systems

The gear box, and the pump bearings and seals of the CRD drive water pumps are served by heat exchangers supplied from the Reactor Building Closed Cooling Water (RBCCW) System. Solenoid valves in the RBCCW supply lines secure the cooling water when the CRD drive water pump is turned off to prevent condensation.

3.2.1.3.3 System Fault Tree Model Assumptions

This section describes any assumptions specific to the system fault tree.

The following assumptions were used in the development of the CRD system fault tree model:

1. The Reactor Building Closed Cooling Water System (RBCCW) supply to the CRD pumps is not explicitly modeled but is included as a "diamond" (i.e. undeveloped) event. The same is true for the instrument air system.

2. Train A is assumed to be in service with the B train valved out and in standby. The exception being the standby CRD pump which is always valved in when in standby.

3. Scram reset is not considered a system failure (even though it will isolate the charging water header) since the scram cannot be reset unless the reactor level is greater than 170". With a vessel level above 170", the CRD system is not needed for injection.

4. Pump failures due to inadequate NPSH are not included under pump failure conditions. The CSTs are assumed to provide a continuous suction source to the CRD pumps.

5. The Anticipated Transient Without Scram (ATWS) is not modeled in the CRD system fault tree. Since no unique failure modes were identified for DAEC, an industry wide generic probability of failure to scram (by either a hydraulic lock or an electrical failure) is assumed to be acceptable for use due to the similarity in BWR scram systems.

6. A scram signal is required to inject through charging header. SDV vents and drains both valves (either vents or drains fail to close on scram).

3.2.1.3.4 Success Criteria

The Control Rod Drive (CRD) system is used during normal plant operation to maintain hydraulic fluid pressure the control rods. However, following plant trip, the CRD pumps can be used to provide high pressure reactor vessel injection. The system can be used to maintain reactor inventory for transient events, however no credit for CRD flow to prevent core damage has been taken in the Level I evaluation.

Successful injection via the hydraulic control units from the CRD system 4-8 hours after an initiating event requires operation of two CRD pumps. The injection path can be either through the charging water header or the cooling water header. This will depend upon whether or not the scram signal is reset. Successful operation of the system after 8 hours requires operation of only one CRD pump. In both cases, operator action and suction from the condensate storage tank are required. Additionally, RBCCW is required for the CRD pump gear box, bearing and seal heat exchanger cooling.

3.2.1.4 Low Pressure Core Spray System

3.2.1.4.1 System Function

The purpose of the Low Pressure Core Spray (CS) System is to provide and maintain cooling for the core during and following a loss of coolant accident (LOCA).

The following design bases are incorporated into the CS System:

a. The CS System is designed to provide sufficient core cooling for intermediate and large line breaks up to and including the design basis double-ended shear of a 22 inch recirculation suction line, without assistance from any other emergency core cooling system.

b. The CS System provides core protection for small breaks in which the control rod drive water pumps, the RCIC System and the HPCI System are all unable to maintain reactor vessel water level and the Automatic Depressurization System has operated to lower reactor vessel pressure.

c. The system is comprised of two independent loops. Separate AC and DC power sources are supplied for each loop.

d. All components, piping, instrumentation, and switchgear for each CS loop are separated so that any single physical event cannot make both loops inoperable.

e. The injection valves that initiate system flow to the core are located as close as possible to the Primary Containment to minimize the length of system piping exposed to reactor pressure.

f. A test line is provided to allow full flow system testing during normal plant operations.

g. All motor operated valves may be tested from the Control Room during normal plant operations.

A simplified diagram of the CS System is shown in Figure 3.2-5. During the following discussion only loop "A" will be described since both loops are identical.

The CS pumps take a suction on the suppression pool through two motor operated, normally keylocked open suction valves. The pumps are located below the normal level of the suppression pool to ensure proper suction head. Pump discharge is directed to the Core Spray sparger via a pump discharge check valve, two motor operated injection valves, check valve, and a manually operated isolation valve.

When automatically actuated, the CS pumps energize before the inboard injection valve opens. Since pump operation with no flow can lead to pump overheating, a minimum flow bypass line is provided which directs pump flow to the suppression pool. A motor operated isolation valve (MO-2104) in this line receives control signals from a flow switch which monitors sprayline flow rate. MO-2104 is open during system startup allowing pump discharge flow to be diverted to the suppression pool. This valve automatically closes when sprayline flow is sensed after the injection valves open so that all of the cooling water is directed to the reactor vessel.

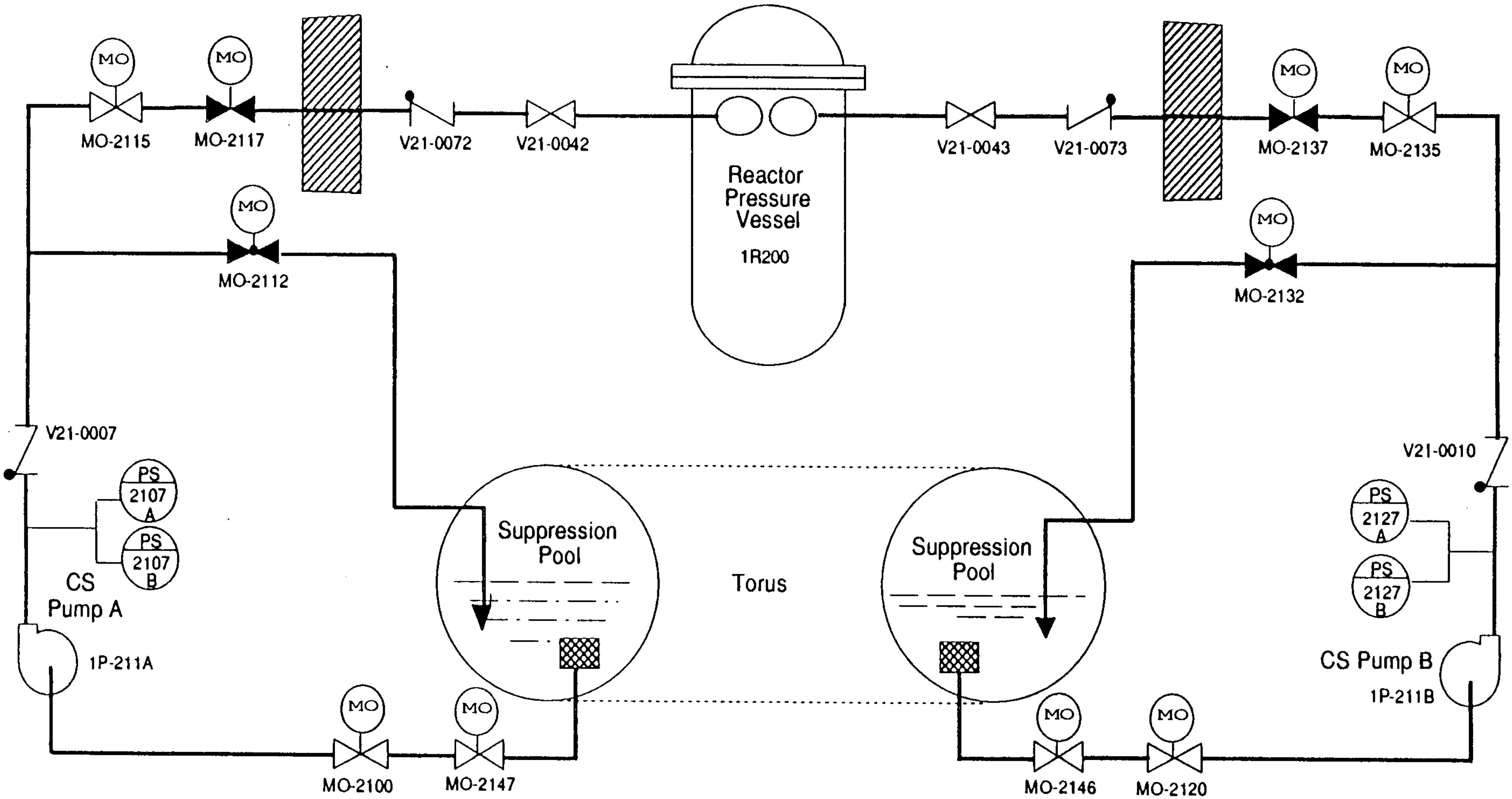


Figure 3.2-5

Core Spray System

Functional testing of the CS System is performed by circulating water from the suppression pool, through the pump, and back to the suppression pool via full flow test line. This testing may be performed during normal plant operations; however, should a CS initiation signal be generated, test line isolation valve MO-2112 will automatically close.

A Residual Heat Removal System line taps into the CS injection header upstream of the outboard motor operated injection valve. This line supplies water from the RHR/Core Spray fill pump to ensure that CS piping up to the inboard motor operated injection valve is kept filled in order to prevent water hammer and minimize any delay in providing cooling water to the core on system initiation.

CS pump suction may also be taken from the Condensate Storage Tanks through a normally closed manual isolation valve. This source of water can be used to flood the vessel and reactor cavity for refueling and as an emergency injection source when the suppression pool is drained.

CS System piping overpressure protection is provided by suction and discharge line relief valves. The suction line relief valve discharges to the Radwaste Sump System, while the suppression pool collects the discharge line relief valve flow.

3.2.1.4.2 System Interfaces and Dependencies

The CS fault tree model includes support systems required for CS to function in postulated accident scenarios. The systems which support specific CS components and their effects on CS operation are identified in the Dependency Matrix shown in Table 3.2-4.

In addition to the systems identified on a component level in the dependency matrix,

Table 3.2-4

CORE SPRAY DEPENDENCY MATRIX

COMP	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	ACTUATION SIGNAL
1P-211A	CORE SPRAY PUMP	1-D	OFF	1A3	PUMP DEENGZD	1D13 1D11	LOSS OF CONTROL POWER	-	-	ESW	V-AC-012	Low rx level or hi drywell press
1P-211B	CORE SPRAY PUMP	1-B	OFF	1A4	PUMP DEENGZD	1D23 1D21	LOSS OF CONTROL POWER	-	-	ESW	V-AC-011	Low rx level or hi drywell press
MO-2112	CORE SPRAY SYSTEM I TEST BYPASS VALVE	1-A South	CLOSED	1B34	NONE DESIRED CLOSED	1D11	NONE - LOSS OF AUTOMATIC SIGNAL	-	-	-	-	-
MO-2132	CORE SPRAY SYSTEM II TEST BYPASS VALVE	1-A North	CLOSED	1B44	NONE DESIRED CLOSED	1D21	NONE - LOSS OF AUTOMATIC SIGNAL	-	-	-	-	-
MO-2115	CORE SPRAY SYSTEM I OUTBOARD VALVE	3-B	OPEN	1B34	NONE DESIRED OPEN	1D11	NONE - LOSS OF AUTOMATIC SIGNAL	-	-	-	-	-
MO-2135	CORE SPRAY SYSTEM II OUTBOARD VALVE	D-A	OPEN	1B44	NONE DESIRED OPEN	1D21	NONE - LOSS OF AUTOMATIC SIGNAL	-	-	-	-	-
MO-2100	CORE SPRAY SYSTEM I SUCTION VALVE	1-D	OPEN	1B34	NONE DESIRED OPEN	-	-	-	-	-	-	-
MO-2120	CORE SPRAY SYSTEM II SUCTION VALVE	1-B	OPEN	1B44	NONE DESIRED OPEN	-	-	-	-	-	-	-
MO-2147	CORE SPRAY SYSTEM I MAIN ISOLATION VALVE	1-A South	OPEN	1B34	NONE DESIRED OPEN	-	-	-	-	-	-	-
MO-2146	CORE SPRAY SYSTEM II MAIN ISOLATION VALVE	1-A North	OPEN	1B44	NONE DESIRED OPEN	-	-	-	-	-	-	-
MO-2117	CORE SPRAY SYSTEM I INBOARD VALVE	3-B	CLOSED	1B34	VALVE FAILS TO OPEN	1D11	LOSS OF AUTOMATIC SIGNAL	-	-	-	-	-

Table 3.2-4

CORE SPRAY DEPENDENCY MATRIX

COMP	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	ACTUATION SIGNAL
MO-2137	CORE SPRAY SYSTEM II INBOARD VALVE	3-1	CLOSED	1B44	VALVE FAILS TO OPEN	1D21	LOSS OF AUTOMATIC SIGNAL	-	-	-	-	-
MO-2104	CORE SPRAY SYSTEM I MINI FLOW BYPASS VALVE	1-A South	OPEN	1B34	NONE MINOR DIVERSION	-	-	-	-	-	-	-
MO-2124	CORE SPRAY SYSTEM II MINI FLOW BYPASS VALVE	1-A North	OPEN	1B44	NONE MINOR DIVERSION	-	-	-	-	-	-	-
1VAC01 1	ROOM COOLER FOR P-211B	1-B	OFF	1B44	NONE - ROOM COOLER NOT REQUIRED	-	-	-	-	ESW	-	-
1VAC01 2	ROOM COOLER FOR P-211A	1-D	OFF	1B34	NONE - ROOM COOLER NOT REQUIRED	-	-	-	-	ESW	-	-

the following systems provide support functions to the CS system:

Condensate and Demineralized Water System

The Condensate Storage Tank can be used by the CS system to perform injection tests to the reactor vessel, flood the vessel and reactor cavity for refueling, or serve as an injection source when the suppression pool is drained. These modes of operation are typically utilized when the plant is shutdown and depressurized. Loss of the CST will not impact CS operation as the normal supply from the torus will remain available.

RHR System

The discharge piping of CS is maintained filled by the RHR fill pump which taps into the Core Spray discharge header. The RHR system "keep fill" line minimizes the chances of water hammer and the time required to provide injection to the vessel. Loss of this portion of the RHR system may impact the efficiency of CS operation, however, it is assumed it's loss does not preclude successful CS operation.

3.2.1.4.3 System Fault Tree Model Assumptions

This section describes any assumptions specific to the system fault tree.

The following assumptions were used in the development of the CS fault tree model.

1. CS may be manually initiated if automatic initiation fails for events other than large break LOCAs.

2. Failure to manually restart after shutoff on high level is modeled.

3. Pipe ruptures due to water hammer are modeled.

4. CS break detection is not modeled.

5. Room cooling is not necessary for successful CS system operation.

6. CS pump cooling from the ESW system is assumed to be required.

7. Minimum flow line operation is not modeled for the CS pumps.

8. CS pump motor breaker failure has been included in the fault tree but is not quantified separately. The generic pump failure data includes pump motor breaker failure.

3.2.1.4.4 Success Criteria

The CS System is sized to provide adequate core cooling for all postulated loss of coolant accidents where the reactor is depressurized. Each CS pump can provide adequate core cooling flow. Operation of the system requires reactor pressure below the shutoff head of the pump.

The success criteria for the CS system is injection from one pump into the reactor. The required flow path consists of suction from the suppression pool, isolation of the full flow test line, and injection into the reactor. The flow path includes a pump minimum flow line. Isolation of this flow path is not required. Successful operation of the CS pumps also requires motor cooling from the ESW system.

Automatic actuation of the CS system requires all of the following:

- high drywell pressure signal or low reactor water level signal
- DC control power
- AC pump and valve power
- low reactor pressure permissive for injection valves to open

For those events which directly result in the immediate depressurization of the reactor (large LOCA), no additional action or ECCS operation is required for CS System injection. CS acts as a backup to the high pressure injection systems (HPCI and RCIC) for those events which do not directly result in the immediate depressurization of the reactor. Operation of the CS System for these events requires the successful operation of ADS.

The CS System is provided with an automatic line fill system which maintains the system discharge piping filled. This is intended to prevent water hammer and other dynamic effects that could occur upon pump start if the line where empty. Failure of this line fill system is a pre-initiator which, if concurrent with operator failure to restore the function before the line empties, may cause system failure due to pipe rupture upon pump start.

3.2.1.5 Electric Power System (AC/DC)

3.2.1.5.1 System Function

Figures 3.2-6 and 3.2-7 show the AC and DC simplified diagrams. The electrical power distribution system supplies power to all site loads either directly or indirectly from the IELP transmission system. The system is designed as a class 1E electrical system and meets the following safety design bases:

a. Provides a high degree of reliability.
b. Maintains the physical independence of the sources of electric power.
c. Provides the means for the detection and isolation of system faults.

Portions of the system that provide power to site loads are designed to meet the following safety design bases:

a. The power sources for the plant are sufficient in number and have adequate electrical and physical independencies to assure that no single failure could interrupt all power at one time.

b. Layout is designed with separation and component redundancy.

c. Redundant loads important to plant operation and safety are split into redundant switchgear sections.

The electrical power system consists of the main, auxiliary, startup, and standby transformers, oil-blast circuit breakers to supply power from the switchyard to the plant, switchgear to distribute power within the plant, and relays for protection of system equipment. The power distribution system consists of two safety related and two non-safety related 4kV switchgear buses and subordinate lower tier distribution

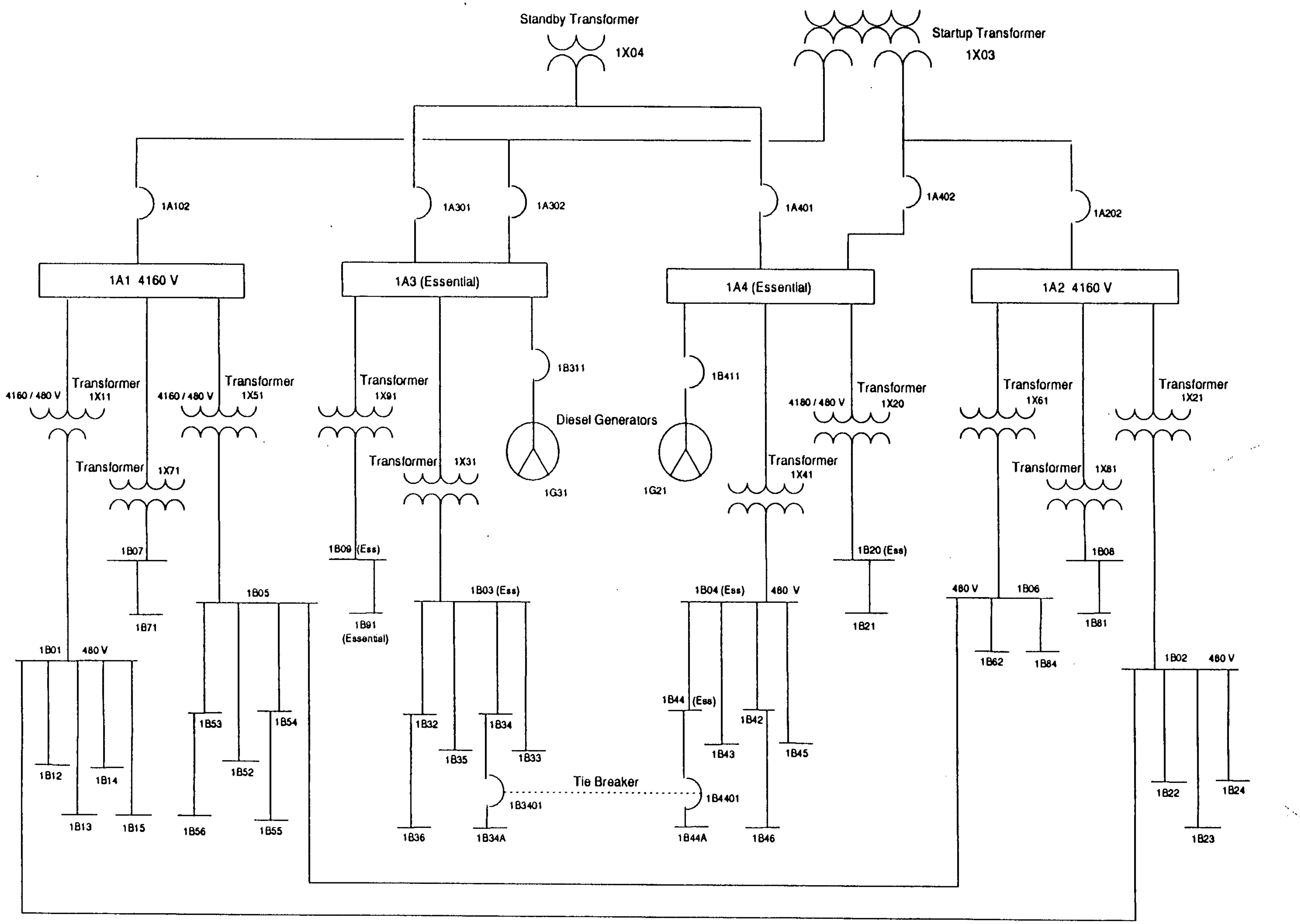


Figure 3.2-6
Electrical Power Distribution System

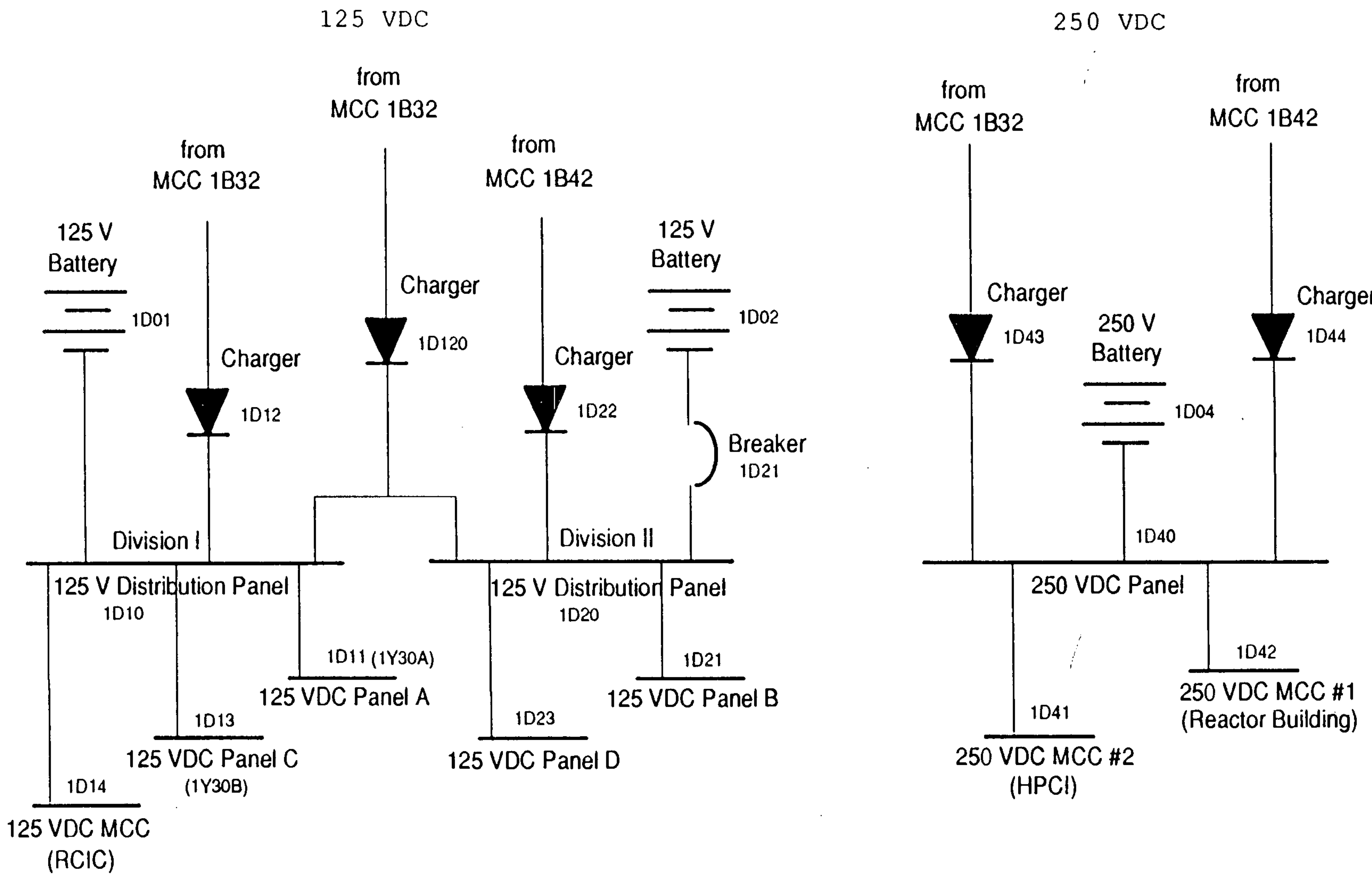


Figure 3.2-7
125 / 250 VDC

buses. Each safety related 4kV bus can be energized by either the startup or standby transformer or a dedicated diesel generator. Each non-safety related 4kV bus can be energized by either the auxiliary or startup transformer.

The Emergency Diesel Generators (EDGs) are required to provide power to the reactor safeguard loads in the event of a loss of offsite power (LOOP). The reactor safeguard loads are those required for safely shutting down the plant and maintaining it in its safe shutdown condition.

The EDGs are designed as a class 1E electrical system and meets the following safety design bases:

1. EDGs are physically and electrically independent of offsite power.
2. Each EDG will pickup its bus automatically on a Loss of Offsite power.
3. A single EDG will be able to supply the loads required to shutdown the plant and maintain it in a safe shutdown condition.

System Initiation - the following is a summary of the EDG start signals:

1. Loss of Offsite Power (LOOP)

   a) Degraded Bus Voltage
   b) Below rated voltage on the Startup and Standby Transformer Secondaries.

2. Loss of Coolant Accident (LOCA)

   a) Low RPV level
   b) High drywell pressure

3. Bus undervoltage ("B" logic only)

Component Trips - the EDG is tripped at:

1. The Governor Shutdown Solenoid (GSS) is energized (the fuel racks move to the no fuel position)

   a. The Stopping Relay energizes:

      1. EDG Lockout
      2. Shutdown Relay Energized (SDR)
      3. Control Switches in Stop (1C08 and 1C93/94)

   b. The Shutdown Relay (SDR) energizes:

      1. Lube Oil Low Pressure
      2. Jacket Coolant Pressure Low
      3. High Crankcase Pressure
      4. Jacket Coolant Temperature High
      5. EDG Start Failure (7 sec)
      6. EDG Overspeed

   c. The Fuel Racks are manually tripped

2. The EDG Output Breaker Trips:

   a. Loss of EDG Excitation

   b. Shutdown Relay Energized

c. EDG Lockout

1. Diff Phase Overcurrent
2. Phase Overcurrent
3. Anti-motoring Signal

d. Bus Lockout

e. Exciter Shutdown Button

All EDG shutdowns are bypassed by the emergency start relays (ESA & ESB) on an emergency start except the EDG lockout, 1C08 Control Switch, and EDG overspeed.

3.2.1.5.2 System Interfaces and Dependencies

The AC Power Fault Tree Model includes support systems required for the AC Power System to function in postulated accident scenarios. The systems which support specific AC Power components and their effects on AC Power distribution are identified in the Dependency Matrix shown in Table 3.2-5.

In addition to the systems identified on a component level in the dependency matrix, the following systems provide support functions to the AC Power System:

125V DC System

The 125V DC system provides control power for automatic breaker trip or closure, diesel engine starting, and field flashing for generator output voltage buildup.

Table 3.2-5
AC POWER SYSTEM DEPENDENCY MATRIX

COMP	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
152-101	4kV SWGR		C			D10	Fails as-is					Unit Trip, UV, Bus Lockout
152-102	4kV SWGR		O			D10	Fails as-is					Unit Trip, UV, Bus Lockout
152-201	4kV SWGR		C			D20	Fails as-is					UV, Bus Lockout
152-202	4kV SWGR		O			D20	Fails as-is					UV, Bus Lockout
152-301	4kV SWGR	10-F	O			D10	Fails as-is					UV, Bus Lockout
152-302	4kV SWGR	10-F	C			D10	Fails as-is					UV, Bus Lockout
152-311	4kV SWGR	10-F	O			D10	Fails as-is					UV, Bus Lockout, LOCA
152-401	4kV SWGR	10-E	O			D20	Fails as-is					UV, Bus Lockout
152-402	4kV SWGR	10-E	C			D20	Fails as-is					UV, Bus Lockout
152-411	4kV SWGR	10-E	O			D20	Fails as-is					UV, Bus lockout, LOCA
G2	Diesel Generator	8-F	Standby			D20	Fails as-is			ESW	Yes	UV, LOCA
G3	Diesel Generator	8-H	Standby			D10	Fails as-is			ESW	Yes	UV, LOCA
1x4	Load Center Xfor		Energized								Yes	
1x21	Load Center Xfor		Energized								Yes	
1x31	Load Center Xfor	10-F	Energized								Yes	
1x41	Load Center Xfor	10-E	Energized								Yes	
1x51	Load Center Xfor		Energized								Yes	
1x61	Load Center Xfor		Energized								Yes	
1x20		10-E	Energized								Yes	
1x91		10-F	Energized								Yes	

Emergency Service Water

The ESW System supplies cooling water to the jacket coolant, lube oil, and scavenging air coolers.

Ventilation System

The respective area ventilation system will prevent excessive area temperatures which would result in long term equipment failure.

3.2.1.5.3 System Fault Tree Model Assumptions

This section describes any assumptions specific to the system fault tree.

The following assumptions were used in the development of the AC Power System fault tree model:

1. Diesel generator fuel transfer system failures are included in the fail to run basic event.

2. Room cooling is required for the essential and non-essential switchgear rooms.

3.2.1.5.4 Success Criteria

The AC Power System is a plant support system with a success criteria which will be defined independent of initiator. The ability of the plant to cope with each initiator is a function of the operation of the required mitigating systems. Those mitigating systems

may require AC power. As such, the success for the AC Power System for all initiators will be described on a per bus or distribution panel basis.
The success criteria for each bus or panel is to have adequate voltage and load carrying capacity to support the operation of the required front-line systems.

Room cooling is required for the essential switchgear rooms. The rooms are successfully cooled by either Division I or Division II control building HVAC. Each division of control building, HVAC requires successful operation of one of two fans. Additionally, there is a conditional probability that a control building chiller will be required on extremely hot days. Successful operation of the chillers requires, either cooling from the well water system or the ESW System.

Room cooling is required for the non-essential switchgear room. The non-essential switchgear room cooling unit requires well water for successful operation.

DC control power is required to support diesel generator start and automatic breaker operations.

3.2.1.6 Emergency Service Water System

3.2.1.6.1 System Function

The Emergency Service Water (ESW) System is modeled for providing a reliable source of cooling water to essential safeguards equipment under a loss of offsite power condition or after a loss of coolant accident.

The following design bases are incorporated into the ESW System:

a. The ESW System uses Cedar River water to remove heat from the essential safeguards systems.

b. Two independent cooling water loops are provided to ensure adequate service water supply for emergency mode operation.

c. The two ESW pumps start automatically in combination with the emergency core cooling systems following a design basis loss of coolant accident, or upon loss of offsite AC power.

d. To ensure that radioactive fluids are not released into the Cedar River or the Cooling Towers and Circulating Water System, the ESW pumps have sufficient head to maintain design flow through emergency coolers with ESW pressure exceeding the component side pressure.

The ESW System provides cooling for all emergency equipment except the Residual Heat Removal (RHR) System heat exchangers. Two independent cooling loops, each supplied by a single full capacity motor driven pump, provide cooling to functionally redundant components. The components cooled by ESW are:

- Emergency Diesel Generators 1G-21 and 1G-31
- RHR and Core Spray Pump Room Cooling Units (2)
- Reactor Core Isolation Cooling (RCIC) Room Cooling Units (2)
- High Pressure Coolant Injection (HPCI) Room Cooling Units (2)
- Control Building Chillers (2)
- RHR Pump Seal Coolers (4)
- Core Spray Pump Motor Coolers (2)
- Essential Air Compressors (SGTS) (2)
- RHR Service Water Pump Motor Coolers (4)

The cooling water flow required by the components in each ESW loop depends on the temperature of the river water and varies from approximately 670 gallons at 80°F to 1130 gallons at 95°F.

The ESW flow path is shown in Figure 3.2-8. Each ESW pump takes a suction from one of the RHR Service Water and ESW wet pits in the Pumphouse. (See also Figure 3.2-9 for the general layout of the water pits.) Water to the wet pits is supplied by the River Water Supply System via the stilling basin. Each pump discharges through a check valve and locked open butterfly valve to the strainer. A removable spool piece allows either pump to supply the ESW loop normally supplied by the other ESW pump.

ESW flow exits the strainer through a locked open valve and is directed to the various system loads. A normally locked closed strainer bypass valve is provided. Its use is minimized to prevent possible fouling of safeguard equipment coolers.

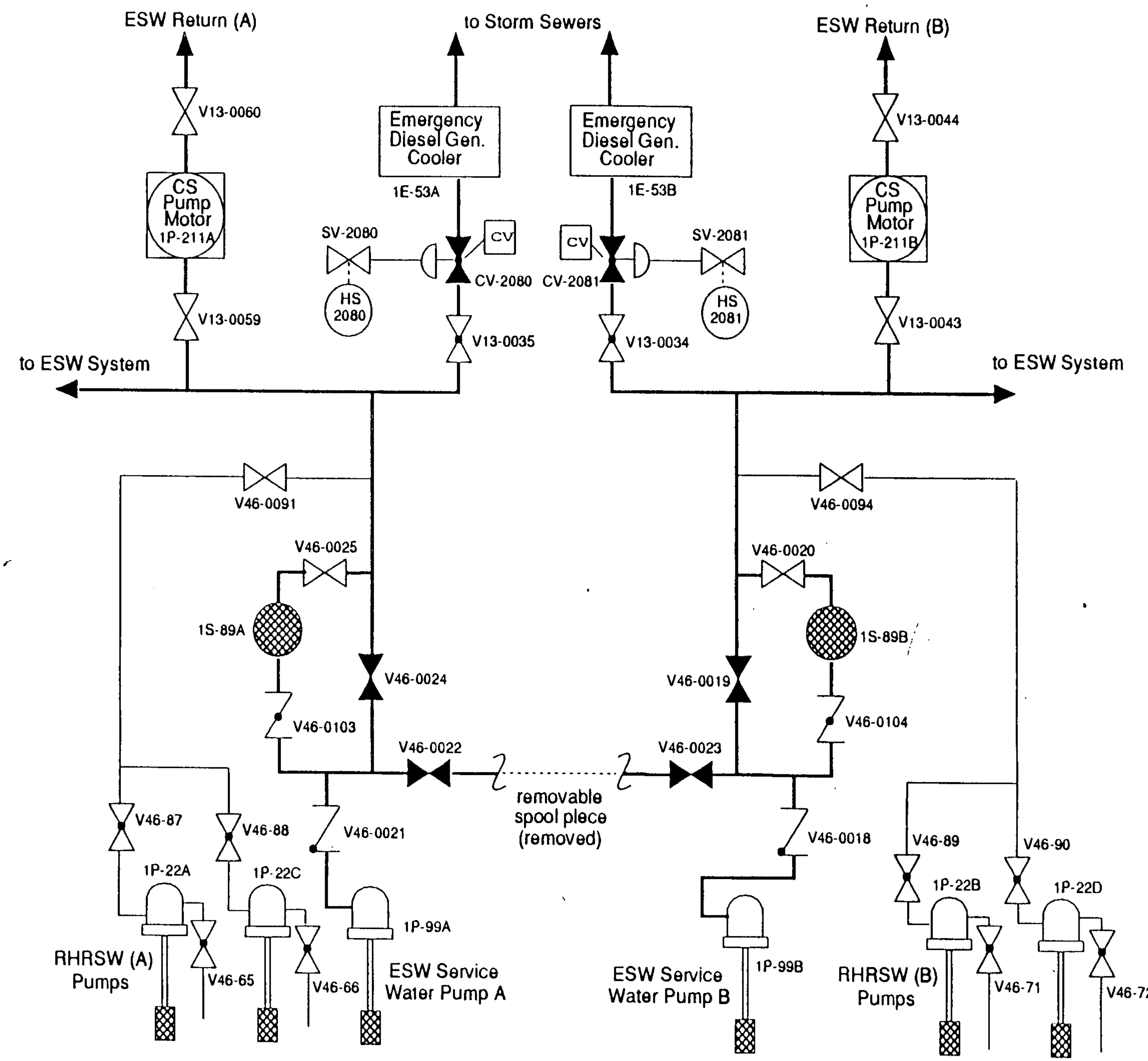


Figure 3.2-8
Emergency Service Water

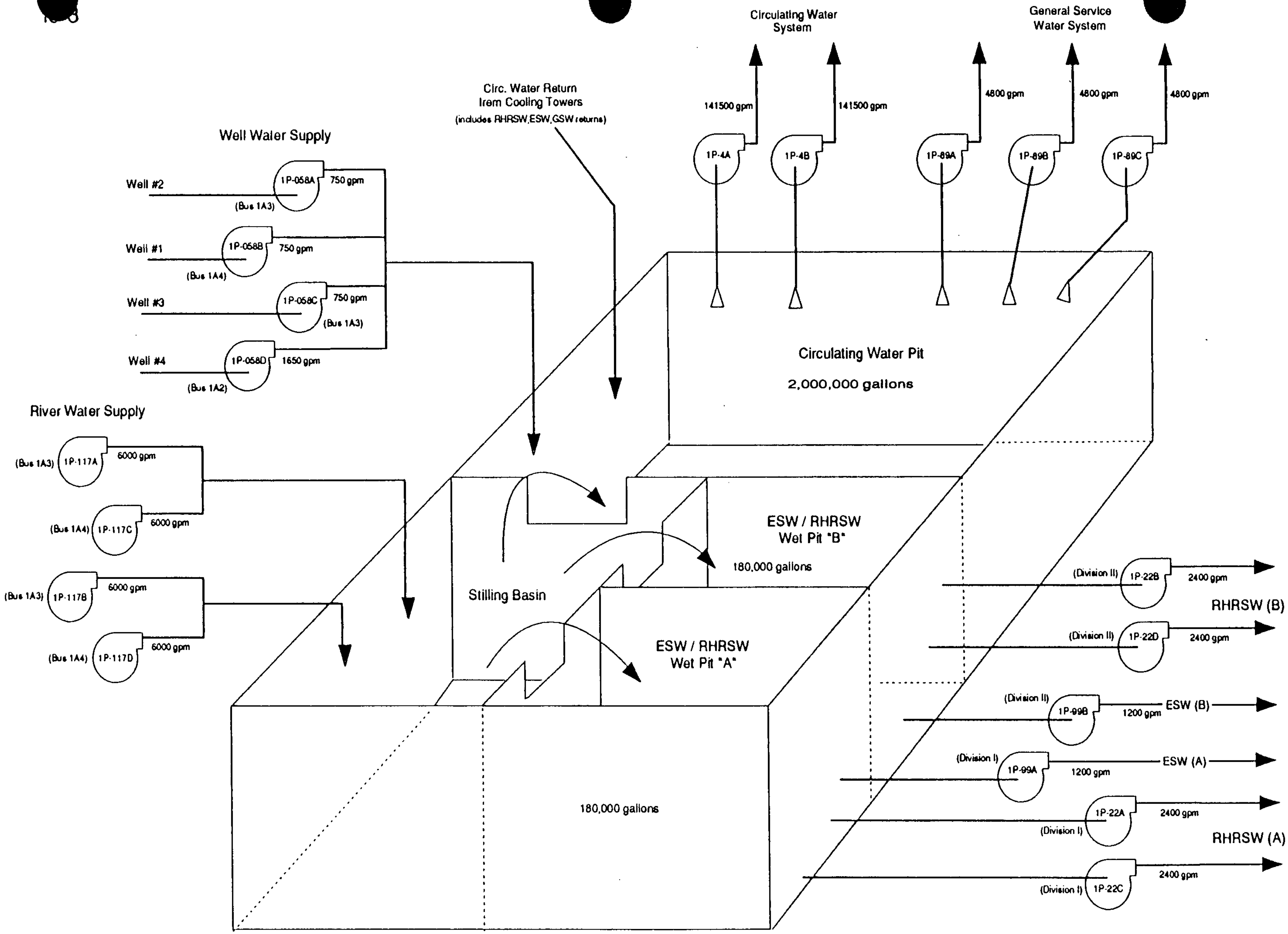


Figure 3.2-9
Water Pits

The first tap off for system loads is for the RHR Service Water Pump Motor Coolers. The amount of flow to each cooler is set by a throttle valve on the discharge line from the cooler. The ESW discharge from the coolers is returned to the wet pits.

The Emergency Diesel Generators (EDGs) receive ESW flow through locked position throttle valves and air operated control valves. The ESW flow for each EDG is through the scavenging air, lubricating oil, and jacket cooling water heat exchangers in series. The ESW discharged from the coolers is directed to the storm sewer.

The Control Building Chillers and Essential Air Compressors are connected to both the Well Water System and the ESW System. When the ESW System is not in operation, the supply of these loads is from the Well Water System, and the return flow from the Control Building Chillers is to the Well Water System while return flow from the H&V Instrument Air Compressors is to the ESW return header. The discharge side of the ESW and Well Water Systems are also cross connected. On ESW System initiation, CV-1956A and B open to allow discharge from the Control Building Chillers to both systems. Motor operated valves can be manually closed to isolate the two systems and allow discharge only to the ESW System header.

The flow path for the remaining system loads is from the supply header through a manual isolation valve, through the load and back to the return header through a manual throttle valve.

The ESW cooling loop discharge headers combine with the RHR Service Water discharge headers. The discharge path for both systems is normally through MO-1998A and B to the Cooling Towers via the Circulating Water System discharge line. Should this path be unavailable due to shutdown of both Cooling Towers, the combined discharge of the RHR Service Water and ESW Systems may be lined up to the dilution structure. Either discharge path is monitored prior to release by a process radiation monitor to detect inleakage of radioactivity.

3.2.1.6.2 System Interfaces and Dependencies

The ESW fault tree model includes support systems required for the ESW to function during postulated accident scenarios. The systems which support specific ESW components and their effects on ESW operation are identified in the Dependency Matrix shown in Table 3.2-6.

In addition to the systems identified on a component level in the dependency matrix, the following systems provide support functions to the ESW system:

River Water Supply and Intake Structure

The RHRSW/ESW wet pits are supplied with water from the Cedar River via the Intake Structure and River Water Supply pumps. In addition, discharge flow from the combined RHRSW/ESW discharge header can be directed to the Intake Structure for de-icing during cold weather operations.

Cooling Towers and Circulating Water System

Normal discharge flow from the RHRSW/ESW combined discharge header is directed to the Cooling Towers via the Circulating Water System piping.

Process Radiation Monitoring

The two possible discharge paths from the ESW System are monitored by process radiation monitors to detect the presence of radioactivity due to inleakage from cooled components.

Table 3.2-6

ESW SYSTEM DEPENDENCY MATRIX

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
CV-2080	Diesel Generator 1G-31 cooling water isolation valve	8-H	Closed			-	-	ISA	Fails open; desired open	-	-	In auto position, valve opens under control of associated diesel generator starting circuit
CV-2081	Diesel Generator 1G-21 cooling water isolation valve	8-H	Closed	-	-	-	-	ISA	Fails open; desired open	-	-	In auto position, valve opens under control of associated diesel generator starting circuit
1P099A	Emergency Service Water Pump A		Off	1B32	Pump fails to run/start	-	-	-	-	-	-	In auto position, pump starts automatically if associated diesel generator starts
1P099B	Emergency Service Water Pump B		Off	1B42	Pump fails to run/start	-	-	-	-	-	-	In auto position, pump starts automatically if associated diesel generator starts
SV-2080	Diesel Generator 1G-31 cooling water isolation valve solenoid	8-H	Energized	1B32	Fails open; desired open	-	-	-	-	-	-	Receives open signal under control of associated diesel generator starting circuit
SV-2081	Diesel Generator 1G-21 cooling water isolation valve solenoid	8-H	Energized	1B42	Fails open; desired open	-	-	-	-	-	-	Receives open signal under control of associated diesel generator starting circuit

3.2.1.6.3 System Fault Tree Model Assumptions

This section describes any assumptions specific to the system fault tree.

Assumptions:

1. Both ESW pumps are initially off and must start and run to meet the required functions.

2. Flow through the various reactor building loads is not modeled in this tree. The loads are piped in parallel paths and so plugging failures and the like do not affect overall loop flow. The individual failures of each load (e.g., heat exchangers plugging) are modeled in the other respective trees.

3. Cooling units are modeled only as receivers of cooling water, fan units are considered part of the failure modes treated under the appropriate system or room cooling units.

3.2.1.6.4 System Criteria

Both ESW pumps must start and run to meet the required functions.

<u>Component Cooling Function</u>

The ESW System functions as a support system to provide cooling water to various heat exchangers and plant equipment. The system transfers heat from front line systems to the plant ultimate heat sink. Since the ESW system is a support system, its success criteria is a function of the operating front line systems requiring heat removal.

For purposes of the IPE, the ESW system can provide cooling to the following equipment:

- CS Pump Motor Coolers
- HPCI Pump Room Coolers
- RCIC Pump Room Coolers
- Essential Switchgear Room Chillers (when necessary)
- RHRSW Pump Motors

The ESW System also provides cooling to the diesel generators. Since there is a delay between the starting of the diesels and the loading of the ESW System pumps, the diesels are designed to operate without ESW flow for a period of time consistent with this delay. It should be noted there are many other components which can be cooled by ESW. These components, however, are either not required in the IPE systems models or have been determined in the individual system analyses as not requiring cooling.

The ESW pumps take suction from the stilling basin (wet pits). Makeup to the stilling basin from one of four River Water pumps is required for ESW operation. This makeup flow must be delivered to the wet pits within 30 minutes (assuming 4008 gpm demand) in order to prevent damage to the ESW pumps due to low NPSH.

Alternate Injection Function

The second function of the ESW is to provide an alternate source of injection to the reactor vessel via the RHRSW flow path. Successful alternate injection requires one of two ESW pumps to be operable, inventory makeup to the stilling basin from one of four river water pumps and manual operator action to initiate alternate low pressure injection.

3.2.1.7 Feedwater System

3.2.1.7.1 System Function

The Feedwater System provides a dependable supply of high quality, preheated feedwater to the reactor vessel at a flow rate which will maintain the desired reactor vessel water level throughout the entire operating range from startup, to full power, to shutdown. The Feedwater System incorporates the following design bases:

a. The feedwater equipment and piping is designed to provide at least 115% of design flow to the reactor at 1100 psi pressure at the reactor vessel feedwater connections.

b. Each reactor feedwater pump can deliver 68% of full flow.

c. A cleanup recirculation line is provided from the feedwater discharge heater to the condenser hotwell in order to minimize corrosion product input to the reactor.

d. Recirculation lines from the discharge of each reactor feedwater pump to the condenser maintain the required minimum flow through the feedwater pumps to prevent overheating.

e. The flow of feedwater is required to be controlled such that reactor level is maintained within a range of several inches in order to achieve optimum steam separator performance. Control should be sufficiently fast to preclude excessive moisture carryover or the initiation of protective systems over the range of normal power operation and during certain anticipated transients.

The Feedwater System receives its supply of water from the Condensate and Condensate Demineralizer Systems. The water delivered to the Reactor Feedwater Pump suction header has been preheated by passing through low pressure heater strings.

Both Reactor Feedwater Pumps take a suction on a common header and discharge through separate feedwater regulating control valves. The feedwater regulating valves are positioned by a control system to maintain reactor water level low enough to minimize carryover of water with the steam and high enough to provide the NPSH required by the reactor recirculating pumps and jet pumps.

The feedwater discharged from the Reactor Feedwater Pumps is further heated by two parallel high pressure heaters. The feedwater, after leaving the HP heater effluent header, enters the containment through two lines.

The major components of the Feedwater System are:

1. Reactor Feedwater Pump and Motor
2. Reactor Feedwater Pump Lube Oil System
3. Reactor Feedwater Pump Sealing System
4. Feedwater Control Valves

As shown in Figure 3.2-10 (see also Figure 3.2-11), the combined header from the second string of low pressure heaters directs flow to the inlet header of the feed pumps. Both feed pumps take a suction on this header via an inlet isolation; valve and discharge to an outlet header via a check valve, a manual isolation and the Feedwater Control Valve. Between the feed pump discharge and the outlet check valve, two tapoffs are provided. The first is a feed pump warm up cross-tie which

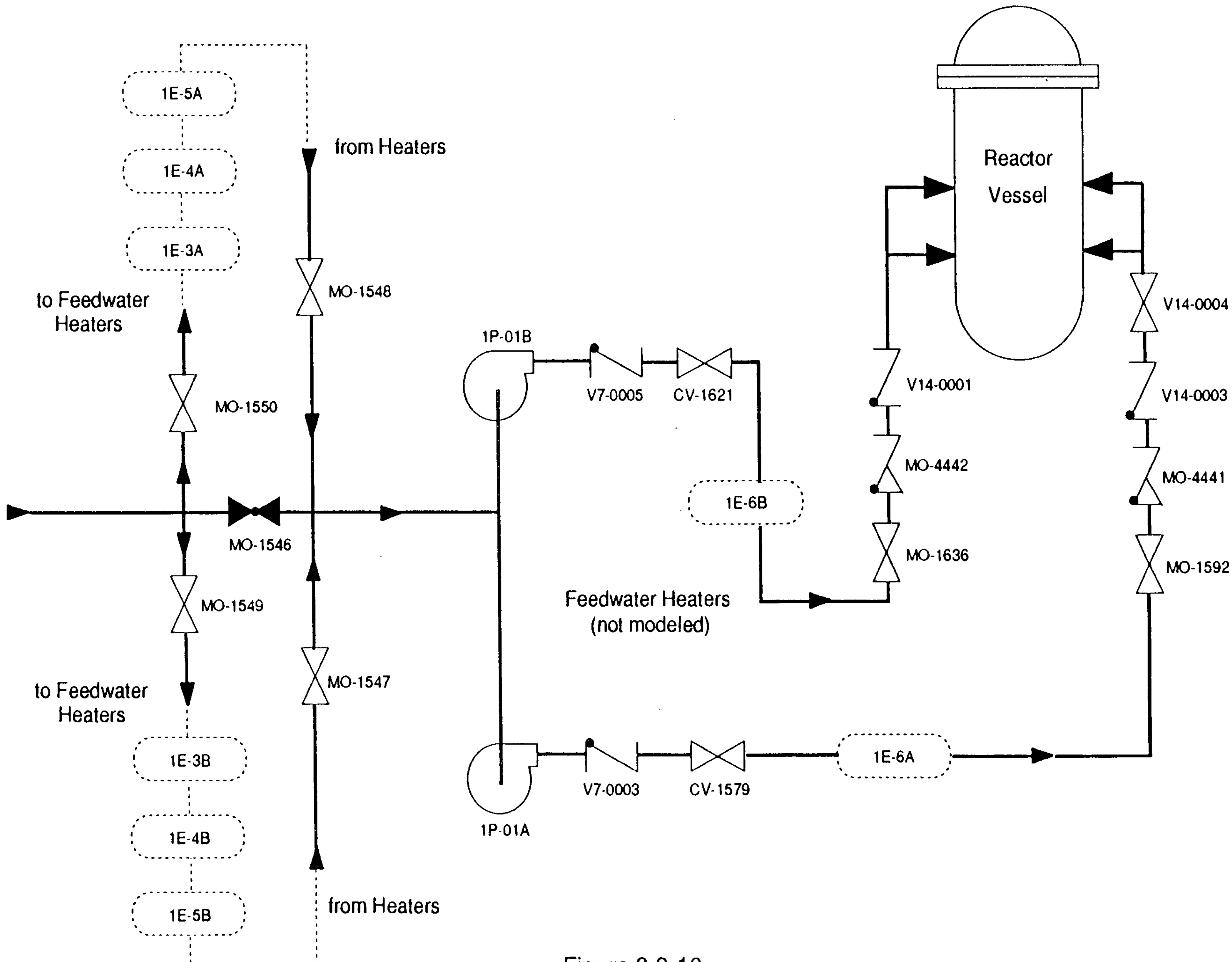


Figure 3.2-10
Feedwater System

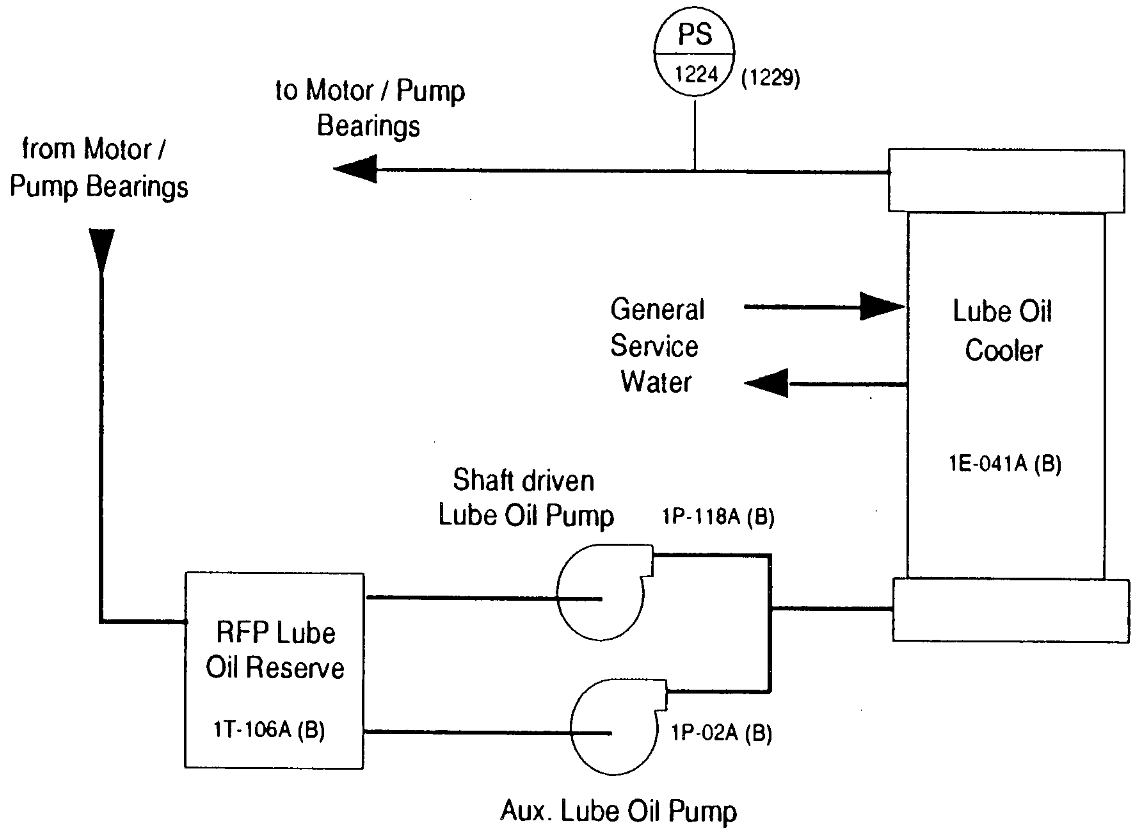


Figure 3.2-11
Feed Pump Lube Oil

allows a small amount of flow through the idle pump to ensure it stays warm. The feedwater pumps must be warmed before starting to prevent impeller contact with the casing and consequent pump damage. The second is a return line to the Main Condenser which ensures adequate feed pump cooling during low power operations.

From the outlet header to the feed pumps, feedwater flow splits and is directed to either of the two high pressure heaters. Feedwater passes through a manual isolation valve and flows through the HP heater. After the heater, feed flow passes through a flow detector, then through the HP heater motor operated outlet valve and into a combined effluent header. Between the outlet valve and the flow detector, a tapoff line to the condenser is provided. This line is used to provide for condensate and feedwater cleanup.

After the effluent header of the HP heaters, feedflow splits and enters the secondary containment. These lines contain a motor operated stop check valve outboard, a check valve and a manual isolation valve inboard. Penetrations between the motor operated stop check and the containment allow for the injection of the HPCI, RCIC and Reactor Water Cleanup System flows. After the manual isolation valve, each feed line splits again for a total of four feed penetrations entering the reactor. Each feed line connects to a feedwater sparger in the reactor which ensures proper distribution.

Condensate and Condensate Demineralizer Systems

The purpose of the Condensate and Condensate Demineralizer Systems is to provide a supply of demineralized water to the Feedwater System from the Main Condenser at the required pressure for Reactor Feed Pump operation. (See Figure 3.2-12.)

The Condensate System incorporates the following design bases:

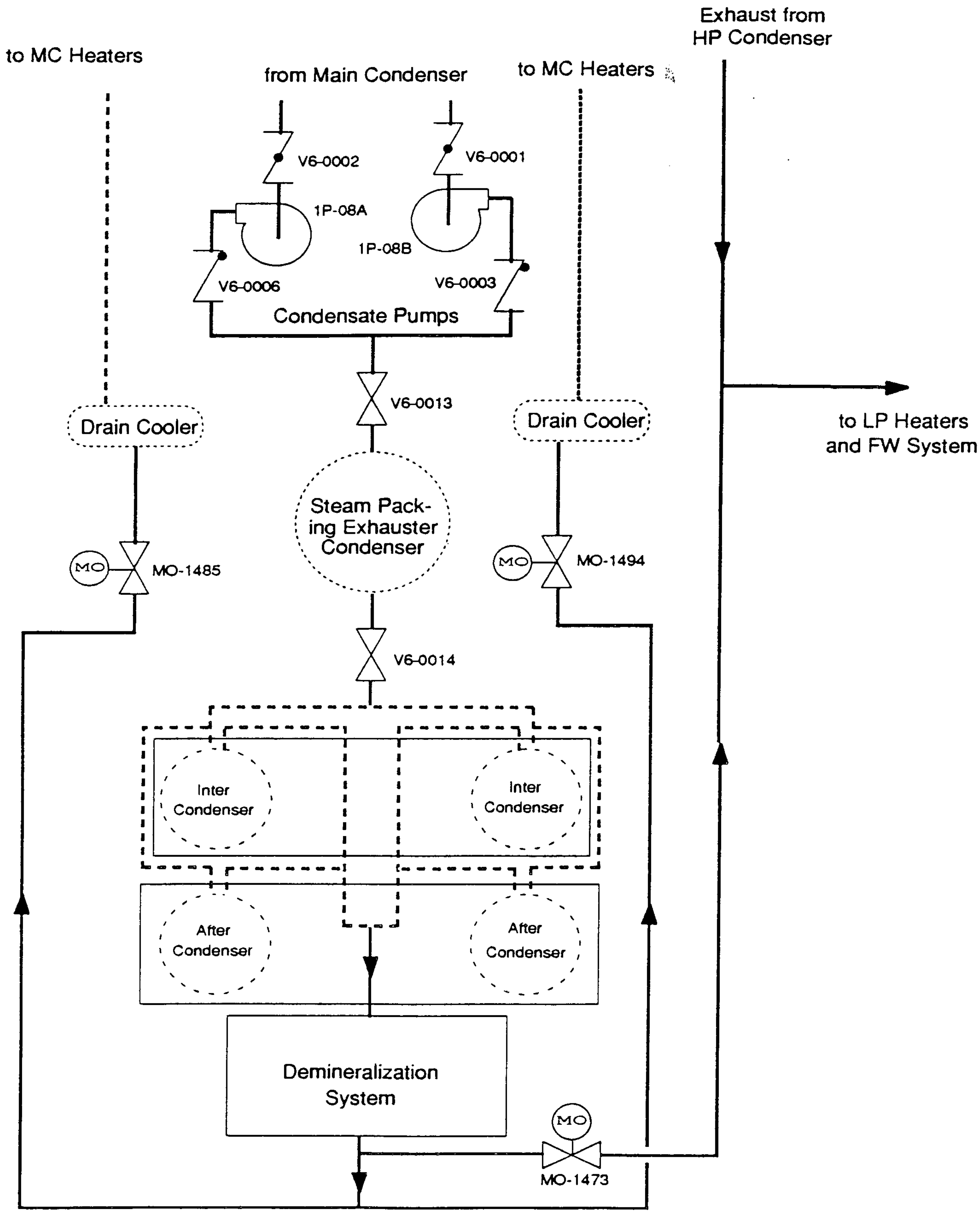


Figure 3.2-12

Condensate System

a. The condensate equipment and piping is capable of providing at least 115% of rated flow at the required pressure for Reactor Feed Pump operation.

b. Each condensate pump can deliver 68% of full flow.

c. Proper Main Condenser hotwell level for both Main Condenser and condensate pump operation is maintained through makeup and rejection between the Condensate Storage Tanks and Main Condenser hotwell. (See Figure 3.2-13.)

d. The Condensate System provides the cooling medium for the Steam Jet Air Ejector condensers and steam packing exhauster. (See Figure 3.2-14.)

e. Process the reactor feedwater design flow of 14,500 gpm, and 15,950 gpm for 10 minute transients.

f. Remove ionic and particulate material from the condensate to maintain the required reactor water quality with minor condenser tube leakage.

g. Provide a final polishing of makeup water entering the cycle.

h. Provide for torus water cleanup during periods when the reactor is in cold shutdown.

Two condensate pumps provide sufficient pressure head to overcome system head losses and deliver the condensate to the feedwater pumps at a pressure sufficient to

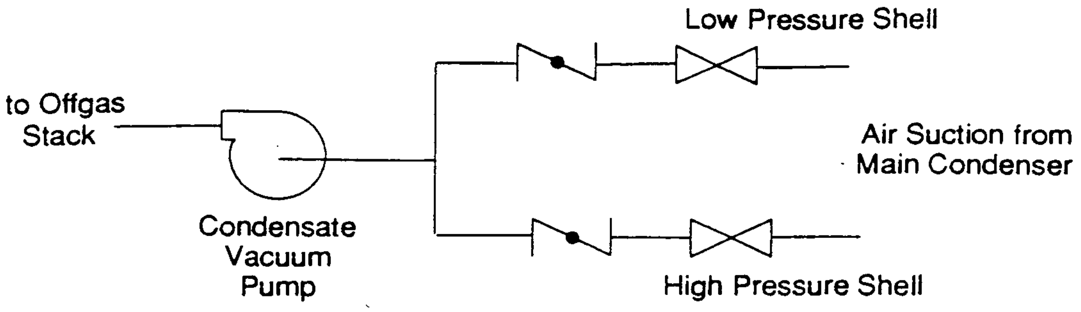


Figure 3.2-13
Main Condenser Vacuum Pump System

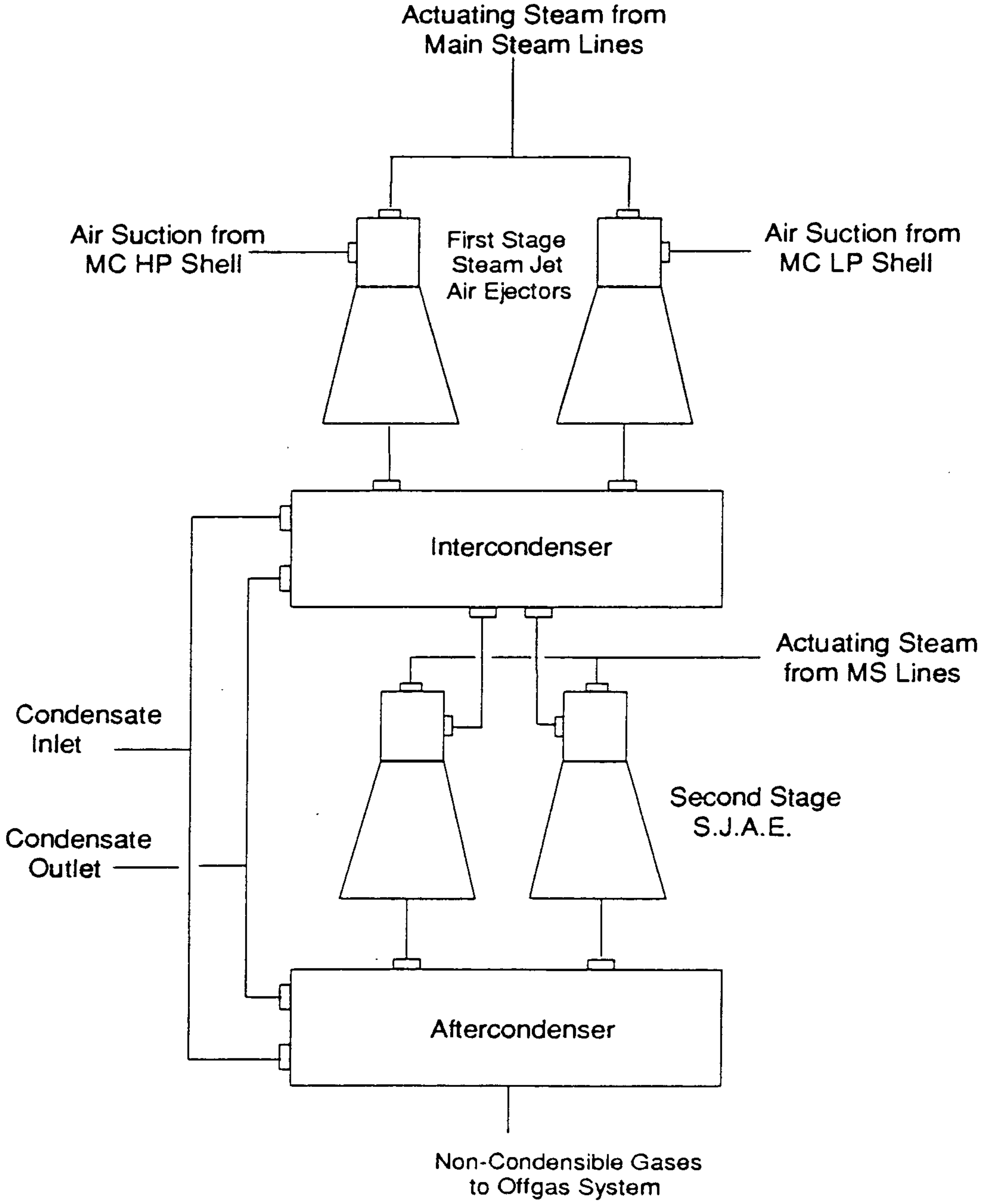


Figure 3.2-14
Steam Jet Air Ejector System

provide adequate net positive suction head. The condensate flow is directed through the steam packing exhauster condenser and steam jet air ejector condensers where it serves as the cooling medium for operation of these components. The condensate enters a common inlet header for the five parallel Condensate Filter Demineralizers, four of which are required for full flow filtration and ion exchange of the condensate water. A recirculation line off the Condensate Demineralizer common inlet line directs sufficient flow back to the condenser hotwell for condensate pump cooling and system startup.

Following demineralization, the condensate is directed to a common outlet header from where it can supply turbine exhaust hood spray, be rejected to the Condensate Storage Tanks, or be directed to two identical heater strings.

In response to the hotwell level control system, a flow control valve allows makeup water to be vacuum dragged from the Condensate Storage Tanks directly to the Main Condenser. A reject line from the common discharge header of the Condensate Demineralizers allows water to be pumped through another valve, controlled by hotwell water level, to the Condensate Storage Tanks. This reject line also serves as the supply source of deaerated, demineralized water for the Control Rod Drive Hydraulic System.

A hotwell transfer pump is provided to allow transfer of water from the Main Condenser hotwell to the Condensate Storage Tanks during plant shutdown periods without requiring operation of the condensate pumps.

The pressure suppression pool (torus) water is cleaned up utilizing the Condensate Demineralizer System. Removable spool pieces are installed during periods of reactor cold shutdown when it is necessary to improve the torus water quality, or drain and store the torus water for reuse.

The major components of the Condensate and Condensate Demineralizer Systems are:

1. Condensate Pumps
2. Steam Packing Exhauster
3. Air Ejector Condensers
4. Condensate Filter Demineralizers
5. Condensate Filter Demineralizer Auxiliaries
6. Hotwell Transfer Pump

The condensate pumps take separate suctions on the condenser hotwell and discharge through check valves to a common header and manual isolation valve. (See Figure 3.2-12.) The inlet header to the condensate pumps is also equipped with a conductivity element and each pump is vented back to the condenser through the seal leakoff line. Leaving the pumps, condensate flow is directed through the steam packing exhauster condenser which is equipped with a bypass line and inlet and outlet isolation valves.

After the outlet isolation valve to the steam packing exhauster condenser, a tapoff is provided for condensate pump sealing water. This line directs a small amount of condensate flow to the seal water reducing station. The reducing station consists of a pressure reducer, backpressure regulator, manual bypass, and a relief valve, all of which act to maintain seal water pressure at 10 psi. The seal water helps retain the condensate and acts as a shaft seal coolant.

Condensate flow is then split and directed to both air ejector condensers. Each air ejector condenser consists of an inter and after surface condenser. At the inter surface condenser, flow is again split and directed simultaneously to the inter and after surface condenser. The discharge of the inter and after surface condensers combines and is directed to the inlet header of the Condensate Filter Demineralizers.

Both air ejector condenser discharges and the condensate pump recirculation line combine at the Condensate Filter demineralizer inlet. In the Condensate Filter Demineralizer inlet header the incoming condensate flow is directed to any of five parallel filter demineralizer tanks. Each tank has an inlet isolation valve, an outlet combination isolation/flow control valve, and a flow measuring orifice. The condensate water from each tank combines into an outlet and is directed to the drain coolers.

The condensate pump recirculation line that taps into the Condensate Filter Demineralizer inlet header provides at least minimum condensate flow to prevent overheating the pumps. This line is connected between the Main Condenser hotwell and the Condensate Filter Demineralizer inlet header. The recirculation line is equipped with tandem-operating control valves and a flow element. The flow of the condensate is measured along with the flow in the recirculation line. These two signals are combined and act on the recirculation line flow control valves to maintain sufficient condensate pump flow.

After flow leaves the Condensate Filter Demineralizer tanks and combines into a common header, it is directed to the drain coolers. It is on this line that a tapoff for the turbine exhaust hood spray is provided. Exhaust hoods in the low pressure turbine direct steam flow to the condensers. During low load operations, steam from the moisture separators and reheaters could cause overheating of these exhaust hoods. To prevent this, a small amount of condensate flow is directed from the filter demineralizer outlet header through a control valve to spray down the exhaust hoods if their temperature exceeds a setpoint.

The condensate header leading to the drain coolers also contains two more tapoffs. The first tapoff is for sealing water to the Reactor Feed Pumps. The second tapoff is provided for rejecting water from the hotwell to the Condensate Storage Tanks to maintain proper hotwell level. Condensate flow is directed to the storage tanks via a flow element and a control valve. After the control valve the line splits. One line is directed to the Condensate Storage Tanks and the other to the Main Condenser via a flow element and a control valve. It is through these makeup and reject lines that hotwell level is controlled.

3.2.1.7.2 System Interfaces and Dependencies

The Feedwater/Condensate System fault tree model includes support systems required for Feedwater/Condensate System to function in postulated accident scenarios. The systems which support specific components and their effects on Feedwater/Condensate System operation are identified in the Dependency Matrix in Table 3.2-7.

In addition to the systems identified on a component level in the dependency matrix, the following systems provide support functions to the Feedwater/Condensate System:

FEEDWATER SYSTEM

Condensate and Condensate Demineralizer Systems

The Condensate and Condensate Demineralizer Systems deliver purified water at sufficient flow and pressure to the Reactor Feedwater Pump suction line. The Condensate System also supplies water to the RFP seals.

Main Condenser

The main condenser hotwell supplies water to the Condensate System which supplies the Feedwater System. The main condenser hotwell can be utilized provided the Main Steam System is available or makeup to hotwell from the CST is provided.

The Main Condenser receives the minimum recirculation flow from the RFP discharge.

General Service Water

The General Service Water System provides cooling for the feedwater pump motors.

Table 3.2-7

FEEDWATER/CONDENSATE SYSTEM DEPENDENCY MATRIX

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
1P-001A	Motor driven feedwater pump 1A	7-C	Run	1A1	Pump fails to run	1D11	Loss of breaker control power	-	-	GSW	-	-
1P-002A	Motor driven auxiliary lube oil pump 2A	7-C	Standby	1B12	Pump fails to start	-	-	-	-	-	-	Pump starts on low lube oil pressure
1P-001B	Motor driven feedwater pump 1B	7-C	Run	1A2	Pump fails to run	1D21	Loss of breaker control power	-	-	GSW	-	-
1P-002B	Motor driven auxiliary lube oil pump 2B	7-C	Standby	1B22	Pump fails to start	-	-	-	-	-	-	Pump starts on low lube oil pressure
1P-008A	Condensate pump 8A	7-E	Run	1A1	Pump fails to run	1D11	Loss of breaker control power	-	-	GSW	-	-
1P-008B	Condensate Pump 8B	7-E	Run	1A2	Pump fails to run	1D21	Loss of breaker control power	-	-	GSW	-	-
MO-1592	Reactor feedwater flow block valve	8-E	Open	1B52	None desired open	-	-	-	-	-	-	-
MO-1636	Reactor feedwater flow block valve	8-E	Open	1B22	None desired open	-	-	-	-	-	-	-
MO-4441	Feedwater stop check valve	2-G	Open	1B32	None desired open	-	-	-	-	-	-	-
MO-4442	Feedwater stop check valve	2-G	Open	1B42	None desired open	-	-	-	-	-	-	-

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
MO-1546	FW heater bypass line isol. valve	7-F	Closed	1B12	Valve fails to open	-	-	-	-	-	-	-
MO-1473	Demin. control valve	8-E	Closed	1B52	Valve fails to open	-	-	-	-	-	-	-
BPV1[1]	Turbine bypass valve 1		Open	1B12	Valve fails closed	-	-	-	-	-	-	
BPV2[1]	Turbine bypass valve 2		Open	1B22	Valve fails closed	-	-	-	-	-	-	

[1] A small number of Main Steam components are included in the Feedwater System model. See Figure 3.2.1.7-3.

Reactor Vessel and Internals

The feedwater flow is discharged into the reactor vessel through four 10" penetrations. Four spargers distributes the feedwater into the downcomer region.

High Pressure Coolant Injection System

The HPCI System delivers water to the reactor vessel via the "A" feedwater line and is distributed into the downcomer region of the reactor. HPCI pump discharge enters the feedwater line inboard of the motor operated feedwater stop check valve which is outside the primary containment.

Reactor Core Isolation Cooling System

The RCIC pump discharge enters the "B" feedwater line inboard of the motor operated stop check valve which is outside the primary containment. RCIC System flow is distributed in the vessel by the two feedwater spargers associated with the "B" feedwater line.

Reactor Water Cleanup System

The RWCU System return to the vessel is via the RCIC pump discharge line and "B" feedwater line.

Extraction Steam and Feedwater Heaters

Feedwater is heated in the high pressure heaters by extraction steam from the high pressure turbine and drains from the reheater drain tanks. RFP seal water and pump leakage is returned to low pressure heater 1E-4 drain line to heater 1E-3.

Containment Atmosphere Control System

Nitrogen pressure for the pneumatic-spring actuators of the inboard feedwater check valves is provided by the Containment Atmosphere Control System.

Reactor Recirculation System

Recirculation pump runbacks to 45% or 20% speed are initiated based on inputs from the Feedwater System.

Rod Worth Minimizer

The Rod Worth Minimizer low power alarm point and low power setpoints are determined by inputs from the Feedwater System.

CONDENSATE SYSTEM

Main Condenser and Condenser Air Removal Systems

The Condensate System transfers water from the main condenser hotwell to the Feedwater System. It also maintains hotwell levels within the prescribed limits.

The Condensate System acts as the cooling medium for the SJAE condensers. Exhaust steam from the air ejectors is cooled and condensed by the inter and after surface condensers, while non-condensible gases are released to the Offgas System.

Feedwater System

The Condensate System supplies water at the proper pressure to the feedwater pumps suction header. Feedwater pump seal injection water is also taken from the Condensate System via the filter demineralizer effluent header.

Turbine Steam Seal and Drain System

The Condensate System also serves as the cooling agent for the steam packing exhauster. The steam packing exhauster receives turbine gland sealing steam and steam leakoff from the turbine bypass, stop, combined intermediate, and control valves.

Condensate and Demineralized Water System

Condensate makeup is supplied from the Condensate Storage Tanks which also receive excess condensate rejected from the condensate cycle. Water for preparation of the precoat and backwashing of the Condensate Demineralizers is supplied by the Condensate and Demineralized Water System.

General Service Water

The General Service Water System provides cooling for the condensate pump motors.

CRD Mechanisms and Hydraulic System

The Condensate System provides a source of deaerated and demineralized water to the CRD System during normal plant operation.

Solid Radwaste System

The Condensate Backwash Receiving Tank receives water and resins backwashed from the Condensate Demineralizers.

3.2.1.7.3 System Fault Tree Model Assumptions

This section describes any assumptions specific to the system fault tree.

The following assumptions were used in the development of the Feedwater/Condensate System fault tree model:

1. The Feedwater and Condensate Systems are initially assumed to be in their normal, 100% power operating configuration with both feedwater and both condensate pumps running, delivering flow to the reactor vessel.

2. The operators are trained to immediately take action to control reactor level on a reactor scram. Failure of the operators to control level following a scram is assumed to cause the feedwater pumps to trip on high reactor level. One feedwater pump must then be restarted successfully for successful feedwater injection.

3. Since the Condensate and Feedwater Systems are required to be very clean water systems, and since they are in continuous operation, plugging of heat exchangers, air ejector condensers, steam packing exhauster, etc. are not considered to be credible faults and are not modeled in the fault tree. Also, it is assumed that tube breaks do not divert enough flow to be considered a failure

mode for feedwater or condensate injection.

4. It is assumed that the auxiliary oil pump must start to supply oil while the feedwater pump is starting. Once the feedwater pump has started it is assumed that only the shaft-driven lube cil pump is required.

5. It is assumed that condensate transfer from the CST to the condenser hotwell is sufficient to provide makeup for successful decay heat removal on loss of the main condenser provided that the CST makeup transfer bypass valve (V06-0052) is opened by the operator. It is also assumed that makeup from the CST is not sufficient for inventory control on a large LOCA.

6. In the high pressure feedwater injection fault tree it is assumed that the condenser is in operation prior to any event. This is a different initial condition than that used for the main condenser and low pressure condensate injection fault trees, where it is assumed that the condenser was previously lost (all steamlines closed). Therefore, spurious Group I isolations are modeled in the feedwater injection fault tree.

7. In the low pressure condensate injection tree, it is assumed that the pumps must be restarted to supply injection flow to the reactor. Injection with low pressure condensate will be a "last resort" method of providing makeup to the reactor. Therefore, it is assumed that the condenser must be re-established and the pumps have to be restarted.

8. The feedwater regulating valves are designed to fail as-is. It is assumed that the plant is operating at full power and that the regulating valves are fully or nearly open at the time of the trip.

9. Since both pumps are required during full power operation, neither loop can be in a testing or maintenance mode.

10. Failure to start feedwater after the trip is included as a single operator action. Failure to restart condensate pumps for low pressure injection is modeled as a single, independent operator action.

11. Valves which are normally open and are not required to change state will be modeled with the failure to remain open for the mission time. Similarly, valves which are normally closed and are not required to change state will be modeled with the failure to remain closed for the mission time. Plugging of valves or valve stem-disk separation causing plugging, or plugging of piping will in general not be modeled. However, if a valve is not flow tested, the interval between flow tests is more than 18 months, or the plugging event is a single failure for the system, then plugging may have to be modeled.

12. The bypass valves modeled in the fault trees, except for the CST to condenser makeup bypass valve, can be operated by the control room operator. It is likely that, if the bypass valve did not open remotely, an operator would be dispatched to open it locally using the manual valve operator. However, these local, manual actions are not modeled in the fault tree at this time.

13. Condensate and feedwater are currently modeled as failing if makeup to the condenser is unavailable.

3.2.1.7.4 Success Criteria

The feedwater and condensate system fault tree is modeled with two top events. The first top event represents successful high pressure injection into the reactor vessel. The second top event represents successful low pressure injection via the Condensate System.

For successful high pressure injection, one condensate pump and one feedwater pump must be available. This is due to the large capacity of each pump (sized to handle 68% of full flow for normal reactor operation). The condensate pump must be available for operation of the feedwater pump.

The Condensate System may also be used for low pressure injection if both feedwater pumps fail. For successful low pressure injection, the reactor must be depressurized to below the condensate pump shutoff head (462 psig) and one condensate pump must be available.

In both cases, an open flow path must be available to supply flow from the condenser hotwell to the pumps and through the injection piping to the reactor. Makeup to the hotwell is required or the condenser must be available. Successful condenser operation requires one main steam line, two of two turbine bypass valves, and the circulating water system. Also, AC power and GSW for condensate and feedwater pump cooling must be available. For Condensate System low pressure injection, DC power must also be available for pump starting.

3.2.1.8 General Service Water System

3.2.1.8.1 System Function

The General Service Water (GSW) System provides cooling water to the Reactor Building Closed Cooling Water (RBCCW) System heat exchangers, and equipment coolers not supplied by the Emergency Service Water (ESW) or Residual Heat Removal Service Water (RHRSW) Systems. The GSW System can also supply water to the Fire Protection System.

The following design bases are incorporated into the GSW System:

a. The General Service Water System is designed to meet plant requirements for startup, normal operation, and shutdown.

b. A spare pump is provided for reliability.

c. Chemical treatment of the cooling water is provided to minimize fouling of equipment.

d. Since the equipment served by the General Service Water System is essential to continued plant operation, two of the three supply pumps are powered from 4160 VAC essential buses.

The GSW System provides water to meet cooling requirements of the RBCCW System and equipment in the Turbine Building. The cooling water used is strained, chemically treated river water which is supplied from the Circulating Water System wet pit and is returned to this system for recycling after being cooled by passage through the Cooling Towers.

The water is treated to prevent the buildup of scale on heat transfer surfaces and to minimize fouling of the heat exchangers by controlling the growth of slime and algae.

The major components of the GSW System are:

General Service Water Pumps

Three half capacity, vertical, two stage, centrifugal pumps each rated at 4800 gpm with a 160 foot head are provided.

Service Water Automatic Strainer

The self-cleaning strainer consists of a steel pressure housing containing a stainless steel straining element in the shape of a vertical cylinder, open on the inlet side over an arc of 120 degrees.

Heat Exchangers

All of the equipment cooled by the GSW System is via heat exchangers. Each of these heat exchangers is fitted with a manual valve in its cooling water discharge line for flow control.

Discharge Line Radiation Detector

A radiation detector is installed in a well in the system discharge line where the GSW System discharge leaves the Reactor Building.

A simplified drawing of the GSW System is provided in Figure 3.2-15. The three GSW pumps take a suction on the circulating water pit (see Figure 3.2-9 for general water

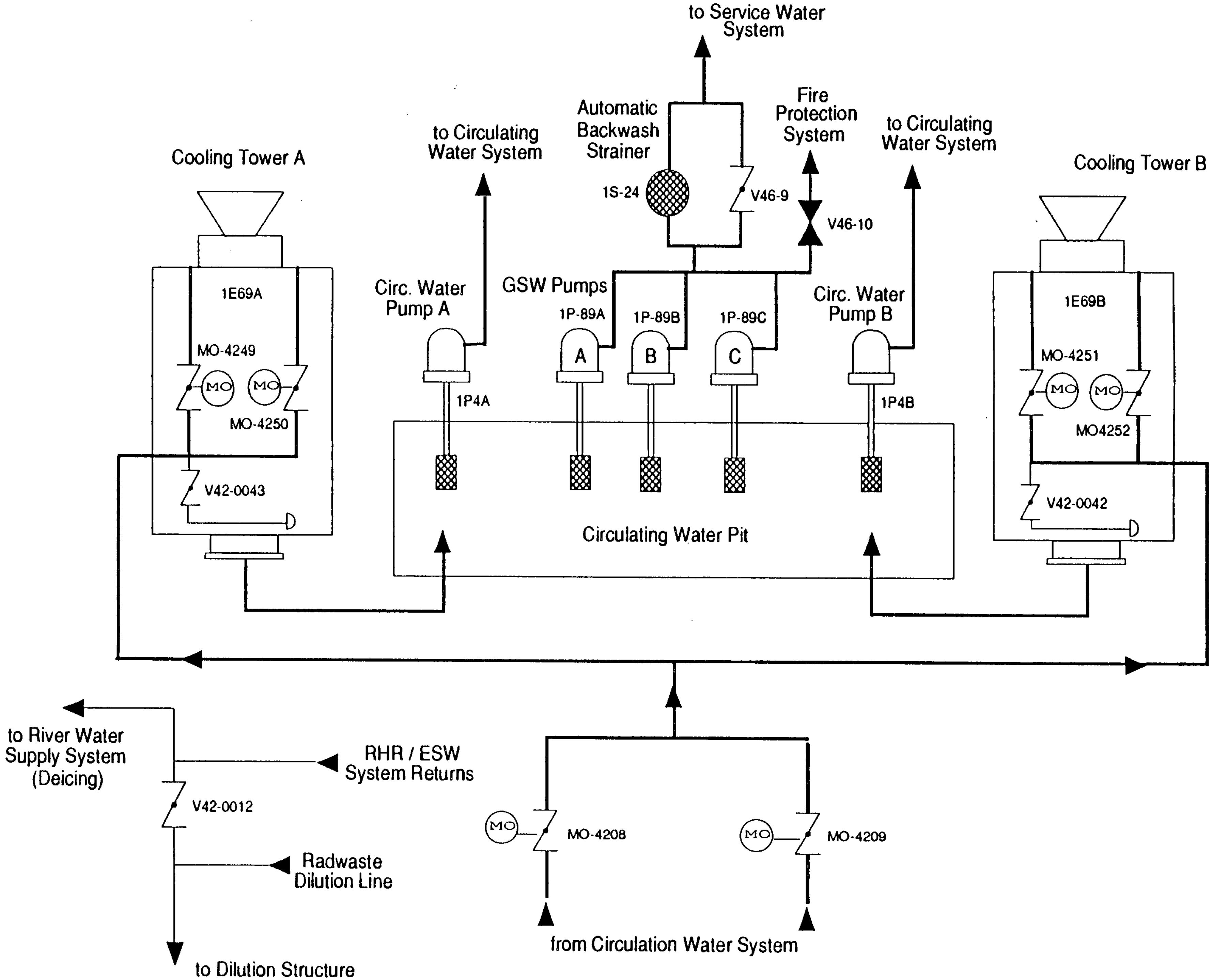


Figure 3.2-15
General Service Water Supply

pit layout) and discharge through a check valve and a butterfly valve into a common header. The GSW header (before the strainer) contains sampling connections and a line which cross connects the GSW System with the Fire Protection System for filling and venting purposes.

The water then flows through an automatic strainer. Periodically the stainer will require backflushing to clean the screen mesh. This is accomplished either automatically through the use of timers and/or a differential pressure switch, or manually by local handswitch operation. When the strainer logic is actuated, the strainer cleaning arm will rotate and system flow will flush the strainer, discharging the flush water through an air operated control valve to a storm drain. A bypass line is provided in the event that the strainer becomes clogged, or maintenance of the strainer assembly is necessary. Use of the bypass line should be minimized to minimize possible fouling of equipment coolers.

A tapoff is provided on the downstream (strained) side of the bypass valve to provide cooling water for the Circulating Water Pump bearing lube oil and motor windings.

The supply header splits into 4 supply lines inside the Turbine Building (734' El). Two of the supply lines are equipped with motor operated isolation valves which are operated from Control Room Panel 1C06. The components these lines serve include:

- Turbine Lube Oil Coolers
- Reactor Feed Pump Lube Oil and Motor Coolers
- Electrohydraulic Fluid Coolers
- Isolated Phase Bus Duct Coolers
- Main Generator Hydrogen Coolers
- Main Generator Stator Winding Liquid Coolers
- Main Generator Exciter Air Cooler
- Condensate Pump Motor Coolers

- Auxiliary Heating System
- Steam Tunnel Coolers

A separate supply line is used for reactor system components which are the Recirculation Pump MG Set Lube Oil Coolers, the RBCCW Heat Exchangers, and the Steam Tunnel Cooling Unit. The discharge of this line is equipped with a radiation element to detect any leakage of reactor coolant into the GSW System which could occur if there was leakage into the RBCCW System followed by leakage into the GSW System.

A fourth supply line may be used to cool the Instrument and Service Air Compressors, and the Instrument Air Dryers. The Well Water System is the normal supply for these components.

The discharge lines from each group combine into a common header which discharges into the Circulating Water System just downstream of the condenser circulating water outlet valves.

The GSW pumps and strainer are located in the Pump House. The Turbine Building cooled components are located on the 734' and 757' elevations of the Turbine Building. The Reactor Building components are located on the 786' elevation (Recirculation Pump MG Set Lube Oil Coolers), and the 812' elevation (RBCCW Heat Exchangers) of the Reactor Building.

3.2.1.8.2 System Interfaces and Dependencies

The GSW System fault tree model includes support systems required for GSW to function in postulated accident scenarios. The systems which support specific GSW components and their effects on GSW operation are identified in the Dependency Matrix shown in Table 3.2-8.

In addition to the systems identified on a component level in the dependency matrix, the following systems provide support functions to the GSW System:

Cooling Towers and Circulating Water System

Provides a suction and discharge point for the GSW System.

Fire Protection System

In an emergency, the GSW System can supply the Fire Protection System. Normally, GSW is only used for filling and venting of the Fire Protection System. River Water Supply System

The River Water Supply System delivers water from the Intake Structure to the stilling basin in the Pump House from where overflow through a standpipe supplies the Circulating Water System wet pits. The GSW pumps are located in the Circulating Water System wet pits.

Process Radiation Monitoring System

The Service Water Process Radiation Monitor is utilized to detect the inleakage of radioactivity into the General Service Water System.

Well Water System

The Well Water System delivers water to the Circulating Water System wet pits. The GSW pumps are take suction from the Circulating Water System wet pits.

Table 3.2-8

GSW SYSTEM DEPENDENCY MATRIX

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL PSN	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
1P-089A	General Service Water, Pump A	Yard	Running	1A3	Pump fails to run	1D13	Loss of control power	-	-	-	-	-
1P-089B	General Service Water, Pump B	Yard	Not Running	1A4	Pump fails to start	1D23	Loss of control power	-	-	-	-	Pump A low discharge pressure
1P-089C	General Service Water, Pump C	Yard	Running	1A2	Pump fails to run	1D21	Loss of control power	-	-	-	-	-
MO-4208	Motor Operated Valve MO-4208	Yard	Closed	1B62	Valve fails to open	-	-	-	-	-	-	-
MO-4209	Motor Operated Valve MO-4209	Yard	Closed	1B62	Valve fails to open	-	-	-	-	-	-	-
MO-4249	Motor Operated Valve MO-4249	Yard	Closed	1B71	Valve fails to open	-	-	-	-	-	-	-
MO-4250	Motor Operated Valve MO-4250	Yard	Closed	1B71	Valve fails to open	-	-	-	-	-	-	-
MO-4251	Motor Operated Valve MO-4251	Yard	Closed	1B81	Valve fails to open	-	-	-	-	-	-	-
MO-4252	Motor Operated Valve MO-4252	Yard	Closed	1B81	Valve fails to open	-	-	-	-	-	-	-

3.2.1.8.3 System Fault Tree Model Assumptions

This section describes any assumptions specific to the system fault tree.

The following assumptions were used in the development of the GSW System fault tree model:

1. The success criteria used in the fault tree is one of three pumps. Two 50% capacity pumps are normally running during operation with a third in standby. It is assumed that one pump is sufficient to handle the loads expected during the safe shutdown process.

   For the sake of modeling, it is assumed that pumps A and C are running and B is in standby mode.

3.2.1.8.4 Success Criteria

The GSW System is modeled for the IPE as providing two functions - component cooling and alternate injection.

Component Cooling Function

The GSW System as a support system provides cooling water to various heat exchangers. The system functions to transfer heat from front line systems to the plant ultimate heat sink. In this capacity, the GSW success criteria is a function of the operating front line systems and requires two of three pumps operating.

The GSW System provides cooling for the following components modeled in the IPE:

- Reactor feedwater pumps
- Condensate pumps
- RBCCW heat exchangers

Alternate Injection Function

The second function of the GSW System is to provide an alternate injection source to the reactor. Successful GSW injection requires one of three GSW pumps and operator action to initiate alternate low pressure injection via the RHRSW flow path. Additionally, makeup to the circulating water pit from one of four river water pumps or one of four well water pumps is required for alternate injection operation of the GSW system.

3.2.1.9 High Pressure Coolant Injection System

3.2.1.9.1 System Function

The purpose of the High Pressure Coolant Injection (HPCI) System is to ensure that the core is adequately cooled to limit fuel clad temperature (2200°F) during a loss-of-coolant condition which does not result in a rapid depressurization of the reactor vessel, such as a small break in the reactor coolant system or a reactor isolation and failure of the RCIC System. This system permits a complete plant shutdown by maintaining a sufficient water inventory until reactor pressure is reduced to a point where the Low Pressure Coolant Injection mode of the RHR System or the Core Spray System can maintain adequate core cooling.

The following design bases are incorporated into the High Pressure Coolant Injection System:

a. The HPCI System provides adequate core cooling to prevent reactor fuel overheating in the event of a loss-of-coolant accident which does not result in rapid depressurization of the pressure vessel.

b. The HPCI System allows for complete plant shutdown by maintaining sufficient reactor water inventory until the reactor is depressurized to a level where the Core Spray System or Low Pressure Coolant Injection mode of the RHR System can be placed into operation.

c. The HPCI System is capable of fulfilling the objectives of design basis (b) without the Reactor Core Isolation Cooling System in the event of a failure of that system.

d. The HPCI System is capable of operation independent of AC power, plant service air, or external cooling water systems. The system will automatically start upon receipt of a DC system supplied initiation signal, and deliver design flow rate within 30 seconds.

e. The capacity of the system is designed to provide sufficient core cooling to prevent clad failure during the time interval that reactor vessel pressure takes to decrease to a value where the Low Pressure Coolant Injection mode of the RHR System and/or Core Spray System become effective.

f. All active components of the HPCI System can be tested during normal plant operation.

g. The components and piping of the system shall be designed to withstand the physical effects of a loss-of-coolant accident to insure that core cooling is not jeopardized.

The High Pressure Coolant Injection (HPCI) System consists of a turbine driven pump, a booster pump, a barometric condenser, associated piping, valves, instrumentation, and other related equipment to provide a complete and independent Emergency Core Cooling System (ECCS).

Figure 3.2-16 shows a simplified diagram of the High Pressure Coolant Injection (HPCI) System. The 10" HPCI steam supply line taps off main steamline "B"

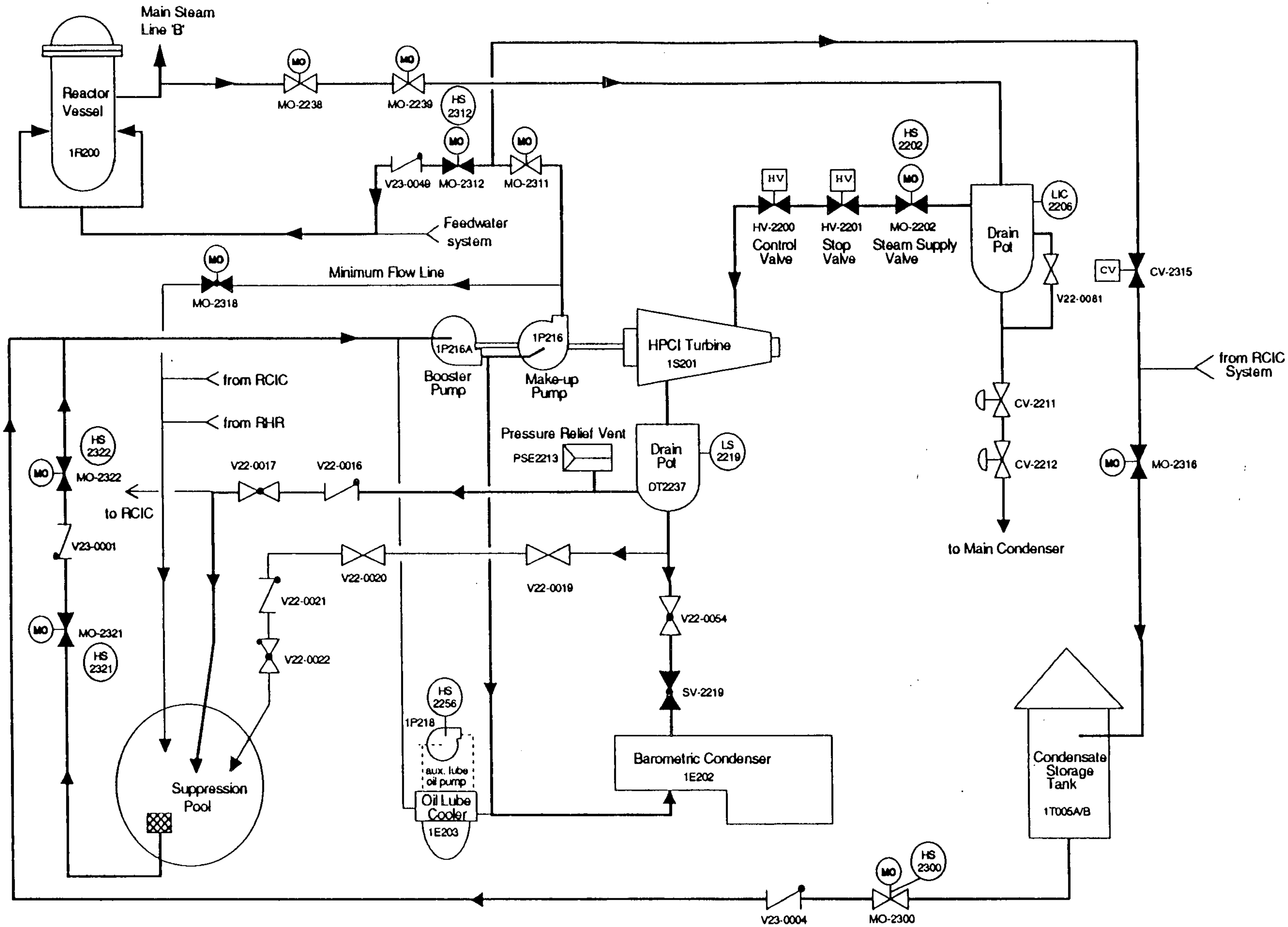


Figure 3.2-16
High Pressure Coolant Injection

upstream of the inboard Main Steam Isolation Valve CV-4415. Two motor operated isolation valves, MO-2238 and MO-2239, are located in the HPCI steamline, one on each side of the primary containment boundary. These valves are normally open to maintain downstream piping near normal operating temperature, thus permitting rapid system startup. A steam supply line drain pot collects condensate which forms in the line during the time the system is in standby/readiness. This condensate is piped through a drain trap to the Main Condenser. Steam is then directed through Turbine Steam Supply valve MO-2202, Turbine Stop valve HV-2201, and Turbine Control valve HV-2200 to the inlet of the turbine. Turbine speed is controlled by the Turbine Control valve which receives signals from the turbine control system. This system is designed to maintain a constant pump flowrate over the entire range of steam pressure.

Steam exhausted from the turbine is directed to the suppression pool via a discharge check valve and a discharge isolation valve. A low point drain pot installed in this line discharges to the suppression pool through a drain trap. Should level in the drain pot increase abnormally high, a solenoid valve will open to bypass drain pot flow to the barometric condenser. Two rupture discs in the turbine exhaust line provide overpressure protection by relieving to the HPCI equipment compartment if exhaust pressure exceeds 175 psig.

Following HPCI turbine operation, the turbine exhaust line cools, allowing any remaining steam to condense, resulting in a reduction of internal exhaust line pressure. This pressure reduction could be sufficient to draw water from the suppression pool into the exhaust line and possibly the HPCI turbine. To prevent this occurrence, two vacuum breaker check valves connect the air space in the suppression chamber to the turbine exhaust line such that a pressure reduction in the exhaust line will act to open the check valves, allowing air pressure in the suppression chamber to "break" the vacuum being formed in the exhaust line. These valves will be automatically isolated on a combination of low reactor pressure and high drywell pressure signals (Group 9 isolation).

Water to be supplied to the reactor vessel, by the HPCI System, is drawn from the Condensate Storage Tanks (CSTs) by the HPCI booster pump. Makeup water may also be drawn from the suppression pool; however, the suppression pool is not the preferred source because of the questionable water purity. If a Condensate Storage Tank low level or a suppression pool high level condition occurs, suction to the HPCI booster pump will automatically shift from the Condensate Storage Tank to the suppression pool.

The HPCI booster pump discharges directly to the suction of the HPCI main pump, providing the main pump with a supply of makeup water under sufficient pressure to meet the main pump NPSH requirements. A small amount of booster pump discharge is also directed to the turbine lube oil cooler for heat removal and to the barometric condenser where it mixes with and condenses the steam contained therein. Water discharged from both the turbine lube oil cooler and barometric condenser is returned to the HPCI booster pump suction.

HPCI main pump discharge is routed to the reactor vessel via feedwater header "A" during normal operation. During system low or no flow conditions, a portion of main pump discharge is directed to the suppression pool through the minimum flow valve. This discharge path is provided to remove pump heat generated in the main and booster pumps to prevent them from overheating. HPCI main pump discharge may also be directed back to the Condensate Storage Tank. This discharge path enables a full flow functional test of the HPCI System to be performed without disturbing reactor operation. During a full flow test, the HPCI Inject valve is shut, isolating HPCI main pump discharge from feedwater header "A".

The barometric condenser is also in service during HPCI System operation to condense steam emitted from turbine gland seals, stop and control valve leakoff, and receive condensate from the turbine exhaust line drain pot. Condensate formed, along with non-condensible gases, is collected in the vacuum tank from which both the vacuum tank

Condensate Pump and Vacuum Pump take a suction.

The vacuum tank Condensate Pump operates intermittently, in response to a vacuum tank level switch, to transfer collected condensate to either the Liquid Radwaste System or back to the HPCI booster pump suction line. A negative pressure (less than atmospheric) is maintained in the vacuum tank by steam condensation and the Vacuum Pump which also transfers non-condensible gases from the vacuum tank to the Standby Gas Treatment System.

Functional testing of the system may be performed during reactor operation; however, upon receipt of an initiation signal, the test configuration is automatically overridden, enabling the system to provide makeup water to the reactor vessel.

3.2.1.9.2 System Interfaces and Dependencies

The HPCI fault tree model includes support systems required for HPCI to function in postulated accident scenarios. The systems which support specific HPCI components and their effects on HPCI operation are identified in the Dependency Matrix shown in Table 3.2-9.

In addition to the systems identified on a component level in the dependency matrix, the following systems provide support functions to the HPCI systems:

<u>Feedwater System</u>

The HPCI System supplies makeup water to the reactor vessel via the HPCI pump discharge line. This line taps into main feedwater heater "A" just upstream of the

Table 3.2-9

HPCI SYSTEM DEPENDENCY MATRIX

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
1S-201	HPCI Pump Drive Turbine	1-E	Off	-	-	1D21 1D23	Loss of automatic and remote turbine trip power Loss of turbine speed control	-	-	-	-	Receives start signal on HPCI Initiation signal.
1VAC014A	HPCI Room Cooling Unit	1-E	Off	1B34	Cooler fails to run	-	-	-	-	ESW	-	Receives start signal on HPCI initiation signal.
1VAC014B	HPCI Room Cooling Unit	1-E	Off	1B44	Cooler fails to run	-	-	-	-	ESW	-	Receives start signal on HPCI initiation signal.
CV-2211	HPCI Steam Line Drain Isol Valve	1-E	Open	-	-	1D13	Fails closed. Fails to allow HPCI pump drive turbine drain pot to drain to the main condenser during standby conditions. No impact on HPCI continued operation.	INST - N2	Fails closed. Fails to allow HPCI pump drive turbine drain pot to drain to the main condenser during standby conditions. Not required for normal system operation.	-	-	Closes when MO-2202 not fully closed.
CV-2212	HPCI Steam Line Drain Isol Valve	1-E	Open	-	-	1D23	Fails closed. Fails to allow HPCI pump drive turbine drain pot to drain to the main condenser during standby conditions.	INST - N2	Fails closed. Fails to allow HPCI pump drive turbine drain pot to drain to the main condenser during standby conditions. Not required for normal system operation.	-	-	Closes when MO-2202 not fully closed.
CV-2315	Test Bypass Shutoff Valve	1-E	Closed	-	-	1D23	Fails closed. Fails to provide system test capability.	INST -N2	Fails closed. Fails to provide system test capability. Not required for normal system operation.	-	-	Closes on HPCI Initiation signal or MO-2321 or MO-2322 full open.
HV-2200	Turbine Control Valve	1-E	Closed	-	-	1D23	Loss of turbine speed control.	-	-	-	-	Receives open signal on HPCI Initiation signal.

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
HV-2201	Turbine Stop Valve	1-E	Closed	-	-	1D23	Failure of turbine trip capability.	-	-	-	-	Receives open signal on HPCI Initiation signal.
LIC-2206	Drain Pot Level Indicator	1-E	N/A	-	-	1D23	Fails to provide accurate drain pot level indication.	-	-	-	-	Cycle on drain pot level.
LS-2219	Exhaust Drain Pot Drain Limit Switch	1-E	Open	-	-	1D23	Fails to cycle on turbine exhaust drain pot level.	-	-	-	-	Cycles on turbine exhaust drain pot level.
LS-4592D	Reactor High Level Indicator	1-E	N/A	-	-	1D23	Failure of automatic HPCI trip.	-	-	-	-	Cycles on reactor coolant level.
MO-2202	Turbine Steam Supply Valve	1-E	Closed	-	-	1D41	HPCI fails to start.	-	-	-	-	Receives open signal on HPCI Initiation signal
MO-2238	HPCI Inbd Steam Line Isol Valve	Cntmt	Open	-	-	1B34	None; desired open	-	-	-	-	Receives open signal on HPCI Initiation signal if isolation and steamline low pressure signals are not present. Closes on isolation or low steamline pressure signals.
MO-2239	HPCI Outbd Steam Line Isol Valve	2-G	Open	-	-	1D41	None; desired open	-	-	-	-	Receives open signal on HPCI Initiation signal if isolation and steamline low pressure signals are not present. Closes on isolation or low steamline pressure signals.

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
MO-2300	CST Suction Valve	1-E	Open	-	-	1D41	Valve fails to close on suction switch to suppression pool. Check valve backup available but not in fault tree	-	-	-	-	Receives open signal on HPCI Initiation signal unless both suppression pool suction valves full open. Closes when both suppression pool suction valves fully open.
MO-2311	Pump Discharge Valve	1-E	Open	-	-	1D41	None; desired open	-	-	-	-	Receives open signal on HPCI Initiation signal.
MO-2312	HPCI Inject Valve	2-G	Closed	-	-	1D41	Failure of HPCI injection.	-	-	-	-	Receives open signal on HPCI Initiation signal after MO-2202 and HV-2201 leave full closed. Closes on MO-2202 or HV-2201 fully closed.
MO-2316	Redundant Shutoff Valve	1-E	Closed	-	-	1D41	None; desired closed	-	-	-	-	Closes on HPCI initiation signal or MO-2321 or MO-2322 full open.
MO-2318	Min Flow Bypass Valve	1-E	Closed	-	-	1D41	Loss of min-flow. See assumption 8.	-	-	-	-	Opens on HPCI pump discharge >125 psig and <300 gpm HPCI pump flow. Closes on HPCI pump flow >600 gpm or MO-2203 or HV-2201 full closed.

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
MO-2321	Inbd Torus Suction Isol Valve	1-A South	Closed	-	-	1D41	Failure of HPCI suction long term.	-	-	-	-	Opens on low CST level or high suppression pool level with no isolation or steamline low pressure signals present. Closes on isolation or low steamline pressure signals.
MO-2322	Outbd Torus Suction Isol Valve	1-E	Closed	-	-	1D41	Failure of HPCI suction long term.	-	-	-	-	Opens on low CST level or high suppression pool level with no isolation or steamline low pressure signals present. Closes on isolation or low steamline pressure signals.
SV-2219	Exhaust Drain Pot Drain Valve		Closed	-	-	1D23	Fails to cycle on turbine exhaust drain pot level.	-	-	-	-	Cycles on turbine exhaust drain pet level.

feedwater piping primary containment penetration. From this point, makeup water is directed to the reactor vessel through the feedwater spargers.

Condensate and Demineralized Water System

Two Condensate Storage Tanks with a total capacity of 400,000 gallons, of which 75,000 gallons are specifically reserved for HPCI and the Reactor Core Isolation Cooling (RCIC) System, supply demineralized water to the suction of the HPCI booster pump. A return line to the Condensate Storage Tanks taps off the HPCI pump discharge header. This return line can be used to perform HPCI System full flow testing during normal reactor operations.

Main Steam System

The HPCI steam line receives steam from main steam line "B" upstream of the Main Steam Isolation Valves. Steam is required for HPCI System operation to power the turbine-driven main and booster pumps.

Primary Containment System

If level in the Condensate Storage Tanks drops to 10,000 gallons, booster pump suction is automatically shifted to the suppression pool. The suppression pool is also used to condense HPCI turbine exhaust and, during system startup, to receive HPCI pump discharge via the minimum flow bypass line.

Auxiliary Heating Boiler

In order to test the HPCI turbine when reactor steam is not available, the Auxiliary Heating Boiler may be connected to the HPCI steam line via a removable spool piece. The spool piece taps in between HPCI outboard steam line isolation valve MO-2239 and the HPCI steam line drain pot. The Auxiliary Heating Boiler was not modeled (credited) as a source of HPCI and RCIC motive steam. This action was not credited in the IPE analysis.

Main Condenser

Condensate in the HPCI steam line is collected by the steam line drain pot. This condensate is transferred to the Main Condenser via a steam trap. The Main Condenser is chosen since the barometric condenser is not in operation when the system is in standby.

3.2.1.9.3 System Fault Tree Model Assumptions

This section describes any assumptions specific to the system fault tree.

The following assumptions were used in the development of the HPCI fault tree model.

1. Includes initial and subsequent start demands as challenges to successful system operation.

2. Includes selected pipe ruptures because leaks may cause system isolation.

3. Replenishing the CST is included in the fault tree but it is assigned a failure probability of 1.0. The actual mechanics (i.e., pumps, valves, etc.) are not modeled.

4. CST to suppression pool transfer logic is not modeled.

5. Lubrication system for turbine not modeled, currently an undeveloped event.

6. Room cooling assumed to be required. One of two room coolers is sufficient for success.

   In addition, operators can open the doors to the HPCI room to provide a natural circulation cooling path.

7. HPCI minimum flow valve assumed not to be required to open for pump start. Because of short period of time during which valve would be open, damage to the pump due to the valves failure is considered highly unlikely.

8. Transfer of suction from the CST to the suppression pool is assumed to be required (may be conservative).

9. Failure of HPCI due to overfill of the vessel is included in the fault tree model.

10. Mission time of 24 hours is consistently reflected by the Nuclear Safety Operational Analysis (NSOA) Auxiliary Diagram. This is then implemented directly in the fault tree model.

3.2.1.9.4 Success Criteria

The HPCI system is a safety related high pressure reactor injection system. The system operates to maintain reactor inventory for events involving inventory losses that are bounded by a medium LOCA. The system is started either manually or automatically.

Automatic actuation of the system requires:

- high drywell pressure signal
- low-low reactor water level signal
- steam supply
- DC power

The HPCI flow path is from the CST to the reactor vessel. Depending on the length of system operation, suction realignment to the torus may be required. HPCI provides inventory makeup, but does not directly provide long term core cooling. Continued operation of HPCI with suction from the suppression pool may require operation of the RHR system in the torus cooling mode. The system is provided with a turbine speed control system. Failure of this control system could result in no HPCI flow, inability to control HPCI flow, or excessive HPCI flow. The system is automatically tripped upon restoration of reactor water level to the level 8 trip setpoint.

Success of the HPCI System is defined for startup, short term, and long term operation. Startup operation of HPCI requires a steam supply greater than 150 psi, a steam exhaust flow path to the suppression pool, hydraulic pressure from the DC powered pump, DC control power, a suction flow path from the CST, and an injection flow path to the reactor. Startup of HPCI does not require operation of the pumps associated with the HPCI turbine seal leakage paths.

Short term operation of HPCI requires the same functions as for startup, except hydraulic pressure can be provided by either the shaft driven or motor driven pumps.

Long term operation of HPCI requires realignment of the suction flow path from the CST to the torus. Torus cooling using the RHR system may be required. The HPCI system can operate independent of plant AC power. However, the battery supply will be depleted as discussed in the electrical system analysis. Therefore, long term operation of HPCI may require the restoration of AC power or the use of an alternate of DC power.

3.2.1.10 Instrumentation System

3.2.1.10.1 System Function

The instrumentation that is evaluated in this system analysis are those necessary to provide actuation signals to the safety-related high and low pressure ECCS systems. Although the RCIC system is not safety-related, its actuation signal is included in this evaluation since the system is credited in the IPE for certain initiating events.

The actuation signals for the following systems are included in this analysis:

- HPCI
- RCIC
- CS
- LPCI
- LPCI Loop Selection Logic
- Diesel Generator

In general, ECCS systems are designed with one out of two taken twice sensor logic, with a logic train powered by each of the essential DC divisions. This logic arrangement is such that loss of individual sensors, or loss of a single DC division will neither cause spurious system initiation nor defeat the initiation logic function.

<u>HPCI Initiation Logic</u>

The HPCI System receives an actuation signal on PS-4310B, PS-4311B, PS-4312B, PS-4313B signal from the Core Spray Initiation Logic or on Lo-Lo Reactor Water Level signal from LIS-4531, LIS-4532, LIS-4533, and LIS-4534. The instrumentation is arranged in a dual bus or 1 of 2 twice logic with normally deenergized relays.

RCIC Initiation Logic

The RCIC System receives an actuation signal on Lo-Lo Reactor Water Level (LIS-4531, LIS-4532, LIS-4533, and LIS-4534), through relay contacts, from the RHR Loop Selection Logic. The instrumentation is arranged in a dual bus or 1 of 2 twice logic with normally deenergized relays.

CS Initiation Logic

The CS System receives an actuation signal on high drywell pressure (PS-4310B, PS-4311B, PS-4312B, and PS-4313B) and on Lo-Lo-Lo Reactor Water Level (LIS-4531, LIS-4532, LIS-4533, and LIS-4534). The actuation signals include a CS System injection valve open permissive on low reactor pressure. All of this instrumentation is arranged in a dual bus or 1 of 2 twice logic with normally deenergized relays.

LPCI Initiation Logic

The LPCI mode of RHR receives actuation signals on high drywell pressure from PS-4310B, PS-4311B, PS-4312B, and PS-4313B and on Lo-Lo-Lo Reactor Water Level from LIS-4531, LIS-4532, LIS-4533, and LIS-4534. This instrumentation is arranged in a dual bus or 1 of 2 twice logic with normally deenergized relays.

Loop Selection Logic

Loop Selection Logic receives actuation signals on high drywell pressure from PS-4310B, PS-4311B, PS-4312B, and PS-4313B and from Lo-Lo Reactor Water Level from LIS-4531, LIS-4532, LIS-4533, and LIS-4534. These variables initiate the loop selection logic. Additional pressure switches are then used to detect the intact recirculation loop so that LPCI injection can be properly directed.

The detection of the intact recirculation loop is based on the measured differential pressure across the Jet Pump risers in each loop.

Differential pressure switches PDIS-4641, PDIS-4642, PDIS-4643, and PDIS-4644 and associated relays are arranged in a 1 out of 2 twice logic matrix.

The output of this logic matrix provides signals to direct the LPCI flow to the intact recirculation loop. If neither loop has ruptured, LPCI flow is directed to loop B.

The instrumentation is arranged in a dual bus or 1 of 2 twice logic with normally deenergized.

Diesel Start Logic

Diesel Start Logic receives actuation signals from PS-4310B, PS-4311B, PS-4312B, and PS-4313B and LIS-4531, LIS-4532, LIS-4533, and LIS-4534. Diesel Start Logic also receives an actuation signal when standby transformer voltage and startup transformer voltage is less than 65% of rated. Their respective essential bus degraded voltage relay or their respective bus loss of voltage relays. This logic also generates an ESW initiation signal.

Each actuation variable is arranged in a dual bus or 1 of 2 twice logic, with normally deenergized relays. Loss of power signals are 1 of 1 signals except the degraded voltage matrix which is 1 of 2 twice. Loss of power logic also uses normally deenergized relays.

3.2.1.10.2 System Interfaces and Dependencies

The Instrumentation System fault tree model includes support systems required for Instrumentation System to function in postulated accident scenarios. The systems which support specific Instrumentation System components and their effects on Instrumentation System operation are identified in the Dependency Matrix shown in Table 3.2-10.

In addition to the systems identified on a component level in the dependency matrix, the following systems provide support functions to the Instrumentation System:

HPCI Initiation Logic

125 VDC - Power for initiation logics "A" and "B" is supplied from Division I and II 125 VDC respectively. (1D1307 and 1D2114)

Core Spray - Core Spray Initiation Logic provides, through relay contacts, high drywell pressure for HPCI Initiation Logic.

Annunciators - The HPCI initiation logic system can actuate the following annunciators:

1C03C, A-3 HPCI AUTO INITIATED

RCIC Initiation Logic

125 VDC - Power for the single initiation logic is supplied from Division I 125 VDC. (1D1317)Residual Heat

Removal - The RHR Initiation Logic provides, through relay contacts, the Lo-Lo reactor water level signal.

Annunciators - The RCIC initiation logic system can actuate the following annunciators:

1C04C, A-4 RCIC AUTO INITIATED

Table 3.2-10

INSTRUMENTATION SYSTEM DEPENDENCY MATRIX

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
PS-4310B PS-4311B PS-4312B PS-4313B	Drywell Pressure Switch	-	De-energized	-	-	1D11 1D21 1D11 1D21	Sensor Fails	-	-	-	-	High Drywell Pressure Signal
PS-4545 PS-4548 PS-4529 PS-4530	Reactor Low Pressure Permissive	-	De-energized	-	-	1D11 1D21 1D11 1D21	Sensor Fails	-	-	-	-	High Drywell Pressure Signal
PDIS-4641 PDIS-4642 PDIS-4643 PDIS-4644	Broken LOOP Detector	-	De-energized	-	-	1D13 1D23 1D13 1D23	Failure of PDIS	-	-	-	-	Loop Selection Logic
PDIS-4625A PDIS-4625B PDIS-4625C PDIS-4625D	Recirculation Pump Status Detector	-	De-energized	-	-	1D13 1D13 1D23 1D23	Sensor Fails	-	-	-	-	Loop Selection Logic
PDIS-4626A PDIS-4626B PDIS-4626C PDIS-4626D	Recirculation Pump Status Detector	-	De-energized	-	-	1D23 1D23 1D13 1D13	Sensor Fails	-	-	-	-	Loop Selection Logic
PS-4555 PS-4556 PS-4557 PS-4558	Reactor Pressure Switch	-	De-energized	-	-	1D13 1D23 1D13 1D23	Sensor Fails	-	-	-	-	Reactor Low Pressure Signal
LIS-4531 LIS-4532 LIS-4533 LIS-4534	Reactor Low Low Water Level	-	De-energized	-	-	1D11, 1D13 1D21, 1D23 1D11, 1D13 1D21, 1D23	Sensor Fails	-	-	-	-	Lo-Lo-Lo Reactor Water Level Lo-Lo Reactor Water Level

CS Initiation Logic

125 VDC - Power for the CS initiation logic is supplied from Division I and II 125 VDC respectively. (1D1115, 1D2125) Power for essential 4160 bus available sensing is supplied by the respective division. (1D1312, 1D2312)

4160 Volt - Permissive for pump start requires buses (1A3, 1A4) be sensed as being greater than 65% of rated voltage.

Annunciators - The CS Initiation Logic system can actuate the following annunciators:

1C03A, A-8 "A" CORE SPRAY SYSTEM AUTO INITIATED
1C03C, A-1 "B" CORE SPRAY SYSTEM AUTO INITIATED

LPCI Initiation Logic

125 VDC - Power for the LPCI Initiation Logic is supplied from Division I and II 125 VDC respectively. (1D1307, 1D2307) Power for essential 4160 bus available sensing is supplied by the respective division. (1D1312, 1D2312)

4160 Volt - Permissive for pump start requires buses (1A3, 1A4) be sensed as being greater than 65% of rated voltage.

Annunciators - The RHR Initiation Logic system can actuate the following annunciators:

1C03B, A-5 LPCI HI DRYWELL PRESS
1C03B, A-4 LPCI RX LO-LO-LO LEVEL INITIATION

Loop Selection Logic

125 VDC - Power for the Loop Selection Logic is supplied from Division I and II 125 VDC respectively. (1D1307, 1D2307)

Annunciators - The RHR Initiation Logic system can actuate the following annunciators:

1C03B, A-5 LPCI HI DRYWELL PRESS
1C03B, A-6 LPCI LOOP SELECT RX LO-LO LEVEL
1C03B, B-5 RHR RX LO PRESSURE PERMISSIVE AT 450 PSIG

Diesel Start Logic

125 VDC - Power for the Diesel Start Logic is supplied from Division I and II 125 VDC respectively. (1D1111 and 1D1112, 1D2111 and 1D2112). Power for loss of bus, LOOP, and degraded voltage sensing is supplied from the division of the respective diesel. (1D1312 and 1D1315, 1D2312 and 1D2315)

4160 Volt - Permissive for pump start requires buses (1A3, 1A4) be sensed as being greater than 65% of rated voltage.

Annunciators - The Diesel Start Logic system can actuate the following annunciators:
1C08A, A-10 and 1C08B, C-1 GENERATOR RUN
1C08A, C-12 and 1C08B, A-3 ENGINE CRANK

<u>Low Pressure Permissive Logic</u>

See Loop Selection and Core Spray Initiation Logics

3.2.1.10.3 System Fault Tree Model Assumptions

This is a summary of the assumptions that have been applied to the instrumentation system. This list of assumptions when combined with the system notebooks and the simplified P&IDs provides the supplementary information needed to understand the fault tree.

1. Initiation logic is broken up into several components:
   - HPCI Initiation Logic
   - RCIC Initiation Logic
   - CS Initiation Logic
   - LPCI Initiation Logic
   - Loop Selection Logic
   - Diesel Generator Start Logic

- Low Pressure Permissive Logic

2. Power is modeled for each individual component (sensor). Power for each system's initiation logic relays is also modeled.

3. Unavailability due to maintenance is assumed to be zero.

4. Failure due to miscalibration is assumed not to occur because:

   - Small errors would have an insignificant effect during accident.

   - Errors large enough to adversely impact the system would be detectable during operation.

3.2.1.10.4 Success Criteria

Since the instrumentation system functions as a support system, the success criteria will be defined on a per train basis. The success criteria for each train of the system is generation of a valid actuation signal upon demand. Since the actuation relays are normally de-energized, DC control power is required. Initiating events which result in CS and LPCI actuation signals involve transients that do not provide adequate time to credit operator backup to a failed signal. Failures of the HPCI and RCIC actuation signal may be credited with operator backup if the vessel transient is such that adequate time to perform the action is available.

Except for the RCIC actuation signal, a high drywell pressure or low vessel level signal (lo-lo for RCIC/HPCI and lo-lo-lo for CS/LPCI) results in system actuation. For RCIC, system actuation occurs only on low vessel level. In the case of the diesel generator, an actuation signal is also generated on loss of its associated 4kV bus voltage. The Loop Selection Logic requires the comparison of loop differential pressures to determine the intact loop for LPCI injection. Failure of the loop selection logic has no adverse impact during events that do not involve failure of the recirculation piping or the selected injection pathway.

3.2.1.11 Reactor Core Isolation Cooling System

3.2.1.11.1 System Function

The Reactor Core Isolation Cooling (RCIC) System is modeled for providing adequate core cooling during a reactor shutdown and isolation should the Feedwater System not be available to provide the required makeup water. Following a reactor shutdown, the RCIC System may be used to reduce reactor temperature and pressure while maintaining sufficient water level in the reactor vessel to prevent the release of radioactive materials from the fuel as a result of inadequate core cooling.

The following design bases are incorporated into the RCIC System:

a. The RCIC System is designed to ensure adequate core cooling in the event of reactor isolation from the Main Steam System accompanied with a loss of feedwater flow without requiring actuation of any Emergency Core Cooling System (ECCS).

b. The RCIC System operates automatically to maintain sufficient coolant inventory in the reactor vessel to prevent compromising the integrity of the radioactive material barrier (fuel clad).

c. System capacity is sufficient to prevent the reactor vessel water level from decreasing to the top of the core (approximately equal to reactor boil-off rate 15 minutes after shutdown).

d. System design is such that all components necessary for the initiation of the RCIC System are capable of startup independent of AC power, plant service air, and external cooling water systems.

e. Piping and equipment are designed to withstand the effects of an earthquake without a failure that could lead to a release of radioactivity in excess of values in published regulations.

f. RCIC is designed as a Class 1E system to improve its reliability.

g. Provision is made for remote manual operation of the system by an operator.

h. Provision is made so that periodic testing can be performed during plant operation to provide a high degree of assurance that the system will operate when necessary.

The RCIC System consists of a turbine-driven pump unit, barometric condenser, and the associated piping, valves, instrumentation, and control circuits required for proper system operation.

Following a reactor shutdown frcm power, the steam generated by decay heat may initially be a significant percentage of normal operating steam flow. The normal flow path for this steam is to the Main Condenser via the Turbine Bypass Valves. Should the Main Condenser be isolated, manual and automatic relief valve actuation can be used to deliver the steam to the suppression pool where it is condensed. This configuration is satisfactory provided the Feedwater System is available for makeup and the suppression chamber temperature or level does not increase significantly.

In the event the Feedwater System is isolated, there will be insufficient makeup water and the level in the reactor vessel will continuously decrease due to the steam generation by decay heat. Upon reaching RPV Level 2, the RCIC System will automatically start to supply makeup water to the reactor vessel. The turbine is driven with residual and decay heat steam from the reactor vessel which is exhausted to the suppression pool. The turbine-driven pump can supply makeup water from the Condensate Storage Tank (CST) or the suppression pool (torus). The RCIC System discharges makeup water to the reactor vessel via the feedwater piping, accomplishing proper mixing and distribution by using the feedwater spargers. The RCIC System discharge may also be directed to a full flow test line which returns the water directly to the CST. In addition, a small amount of flow is continuously directed to the turbine lube oil cooler and barometric condenser.

The barometric condenser functions to condense small amounts of steam from several sources, i.e.; RCIC turbine gland seals, exhaust line drain pot, and valve stem leakoffs.

In addition to the automatic operational features, provisions are included for remote manual startup, operation, testing and shutdown of the RCIC System, provided system initiation or isolation signals do not exist.

Figure 3.2-17 shows the RCIC System in its standby condition. The RCIC System steam supply is taken from Main Steam line "C" between the reactor vessel and the inboard Main Steam Isolation Valve (MSIV). The two normally open RCIC Steam Line Isolation Valves (MO-2400, 2401) allow the RCIC steam piping to be maintained at approximately reactor temperature and pressure upstream of the normally closed Turbine Steam Supply valve (MO-2404), thereby minimizing thermal shock on system startup. The RCIC Turbine Steam Supply valve is capable of opening against a large pressure differential to admit steam to the Turbine Stop valve (MO-2405) and Turbine Control valve (HV-2406). The stop and control valves are mounted on the turbine

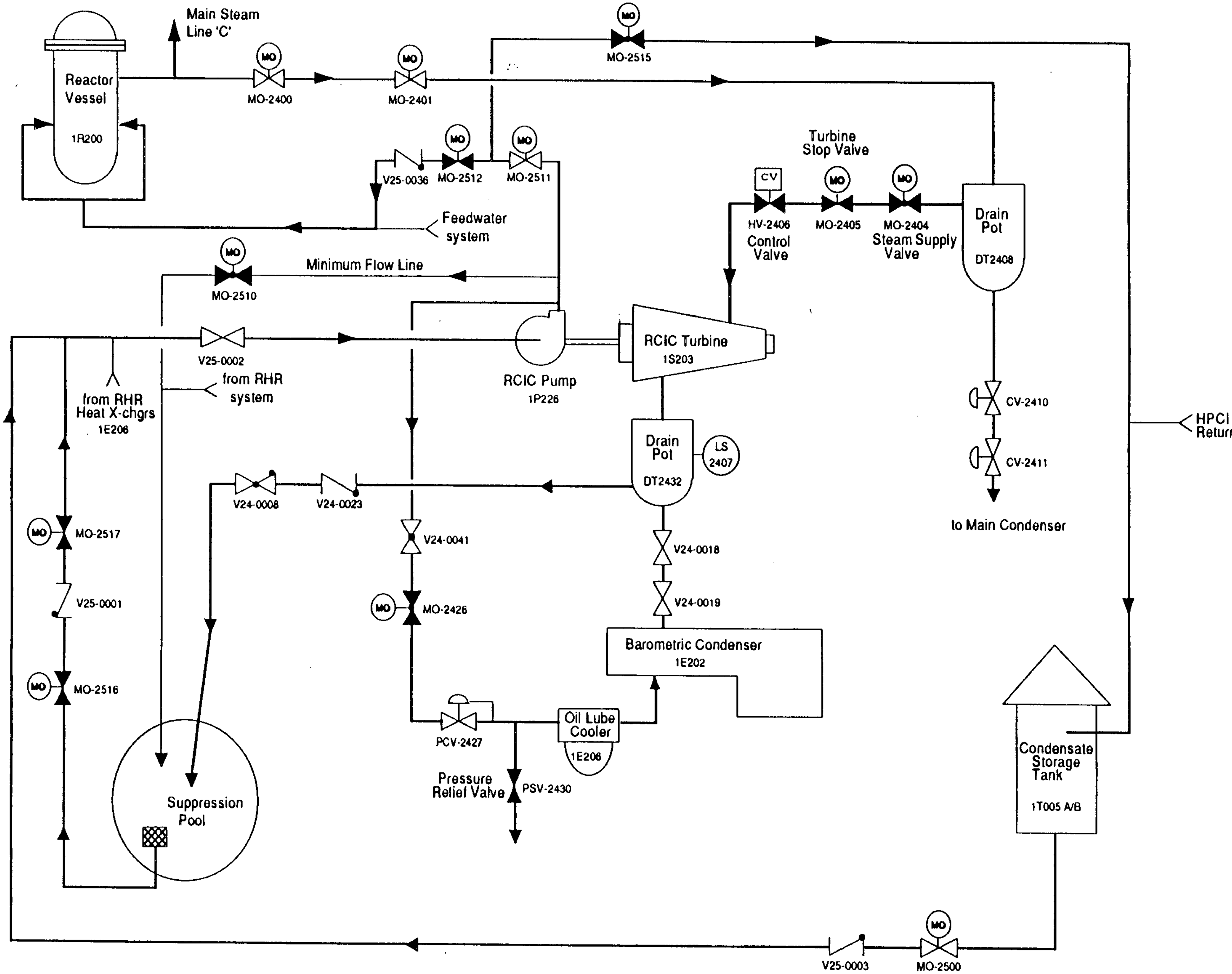


Figure 3.2-17

Reactor Core Injection Cooling

assembly. The control valve position controls turbine speed to maintain the required RCIC pump discharge flow under varying steam pressure conditions. During standby conditions, any condensation formed upstream of MO-2404 is removed via a drain pot to the Main Condenser.

For testing purposes, the RCIC turbine may receive its steam supply from the Auxiliary Heating Boiler. However, use of this source first requires installation of a removable spool piece between the RCIC turbine supply line and the Auxiliary Steam System.

During RCIC turbine operation, turbine exhaust steam is directed to the suppression pool and discharged below the normal water level. Two rupture discs are installed on the exhaust piping. These rupture discs are mounted in series and have pressure switches mounted between them. If the high pressure setpoint between the rupture discs is reached, as would occur due to rupture of the first disc, the pressure switches will act to initiate a RCIC System isolation. Any condensation formed in the exhaust piping is directed to the barometric condenser via a drain pot.

Following RCIC turbine operation, steam condensation in the exhaust piping may reduce its internal pressure sufficiently to draw water from the suppression pool into the exhaust piping. To prevent this occurrence, vacuum breaker check valves are provided between the exhaust piping and the suppression pool free air space. If pressure starts to decrease in the exhaust piping, the vacuum breakers will open to equalize pressure with the suppression pool.

The RCIC pump can take its suction from two sources: the Condensate Storage Tanks (CST), which is preferred source due to its reliable water quality, or the suppression pool. The shift from the CST to suppression pool source of water is automatic on low CST level. The pump discharges through valves (MO-2511 and MO-2512) to the "B" feedwater heater, which directs the RCIC pump discharge into the reactor vessel via the

"C" and "D" feedwater lines and the feedwater spargers.

The discharge from the RCIC pump may also flow to the CST, to the suppression pool, or to the barometric condenser. The full flow test line to the CST is used to ensure adequate system flow rate can be maintained. During testing, RCIC Inject valve MO-2512 remains shut, and the differential pressure across MO-2512 provides the same restriction to flow that would be experienced if the RCIC System were injecting flow to the reactor vessel.

The discharge path to the suppression pool is a minimum flow bypass line that prevents the RCIC pump from overheating during low or no flow conditions. The minimum flow bypass valve (MO-2510) opens when pump discharge pressure exceeds 125 psig and pump discharge flow is less than 40 gpm; it will when close when flow increases to 80 gpm.

RCIC pump discharge is also directed to the turbine lube oil cooler and from there to the barometric condenser. This water is sprayed into the condenser where it mixes with and condenses the collected steam. The condensate is then returned to the RCIC pump suction line by the RCIC condensate pump.

3.2.1.11.2 System Interfaces and Dependencies

The RCIC fault tree model includes support systems required for RCIC to function in postulated accident scenarios. The systems which support specific RCIC components and their effects on RCIC operation are identified in the Dependency Matrix shown in Table 3.2-11.

In addition to the systems identified on a component level in the Dependency Matrix, the following systems provide support functions to the RCIC System.

Table 3.2-11

RCIC SYSTEM DEPENDENCY MATRIX

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
IVAC015A	RCIC Room Cooling Unit	1-F	Off	1B34	Cooler fails to run	-	-	-	-	ESW Loop A	-	-
IVAC015B	RCIC Room Cooling Unit	1-F	Off	1B44	Cooler fails to run	-	-	-	-	ESW Loop B	-	-
CV-2410	RCIC Steam Line Drain Isolation	1-F	Open	-	-	1D13	Valve closes	ISA	None - Not required for system operation	-	-	-
CV-2411	RCIC Steam Line Drain Isolation	1-F	Open	-	-	1D14	Valve closes	ISA	None - Not required for system operation	-	-	-
MO-2515	Test Bypass Valve	1-F	Closed	-	-	1D14	Test line will not isolate if open	-	-	-	-	-
FIC-2509	RCIC Flow Controller	1-F	Energized	-	-	1D13	No controller output - RCIC fails	-	-	-	-	-
LS-4540	Reactor High Level Switch		Open	-	-	1D13	Level Switch has no output	-	-	-	-	-
MO-2404	Turbine Steam Supply Valve	1-F	Closed	-	-	1D14	Valve fails to open - RCIC fails	-	-	-	-	Low-Low Rx Level
MO-2426	Lube Oil Cooler Supply Valve	1-F	Closed	-	-	1D14	Valve fails to open	-	-	-	-	Low-Low Rx Level
MO-2512	RCIC Inject Valve	2-G	Closed	-	-	1D14	Valve fails to open - RCIC fails	-	-	-	-	Low-Low Rx Level

Table 3.2-11
RCIC DEPENDENCY MATRIX
(continued)

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
MO-2516	Inbd Torus Suction Valve	1-F	Closed	-	-	1D14	Valve fails to open - suction switch over fails	-	-	-	-	Low CST Level
MO-2517	Outbd Torus Suction Valve	1-F	Closed	-	-	1D14	Valve fails to open	-	-	-	-	Low CST Level
MO-2510	Min. Flow Bypass Valve	1-F	Closed (open on Low RCIC flow)	-	-	1D14	Min. flow fails to isolate if open	-	-		-	Closes if RCIC discharge flow >80 gpm or MO-2404 or MO-2405 closed
1P226	RCIC Pump	1-F	Off	-	-	-	-	-	-	-	1VAC015A 1VAC015B	-
1S203	RCIC Pump Drive Turbine	1-F	Off	-	-	1D13	Loss of turbine speed control	-	-	-	1VAC015A 1VAC015B	
MO-2400	RCIC Inbd. Steam Line Isol. Valve		Open	1B32	None desired open	-	-	-	-	-	-	-
MO-2401	RCIC Outbd. Steam Line Isol. Valve		Open	-	-	1D14	None desired open	-	-	-	-	-
MO-2405	RCIC Turbine Stop Valve		Open	-	-	1D14	None desired open	-	-	-	-	-
MO-2500	CST Suction Valve		Open	-	-	1D14	None desired open	-	-	-	-	-
MO-2511	RCIC Pump Discharge Valve		Open	-	-	1D14	None desired open	-	-	-	-	-

Feedwater System

The Feedwater System forms part of the discharge path necessary for the RCIC pump to provide makeup water to the reactor vessel. The RCIC pump discharges to the "B" feedwater header and is then directed to the reactor vessel where it is distributed by the "C and D" feedwater spargers.

Main Steam System

Steam necessary for RCIC turbine operation is drawn from Main Steam Line "C". The RCIC turbine steam line upstream of Turbine Supply valve MO-2404, is maintained at an elevated temperature during RCIC System standby periods by the reactor via the Main Steam System.

Condensate and Demineralized Water Systems

The CST provides the primary source of water for reactor vessel makeup used by the RCIC System. The CST is maintained with a reserve of 75,000 gallons specifically for use as makeup water for the reactor vessel. During a full flow test of the RCIC system, RCIC pump discharge is directed back to the CST.

Primary Containment

The suppression pool is the alternate source of makeup water for the reactor vessel should the CST level become low. It is also used to condense RCIC turbine exhaust steam and receives water pumped through the minimum flow bypass line.

3.2.1.11.3 System Fault Tree Model Assumptions

This section describes any assumptions specific to the system fault tree.

The following assumptions were used in the development of the RCIC fault tree model.

1. Includes initial and subsequent start demands as challenges to successful system operation.

2. Includes pipe ruptures.

3. CST modeled in HPCI fault tree.

4. Lubrication system for turbine not modeled in detail.

5. Room cooling assumed to be required. Assume one of two room coolers sufficient for success. In addition, the operators can open the RCIC room doors to establish natural circulation cooling.

6. RCIC minimum flow valve assumed not to be required to open for pump start.

7. Failure of RCIC due to overfill of the vessel is included in the fault tree model.

8. Mission time of 24 hours is consistently reflected by the Nuclear Safety Operational Analysis (NSOA) Auxiliary Diagram. This is then implemented directly in the fault tree model.

3.2.1.11.4 Success Criteria

The RCIC System is a non-safety-related high pressure core cooling system. The system operates to maintain reactor inventory for transient events and events involving inventory losses that are bounded by a small LOCA. The system is started either manually or automatically.

Automatic actuation of the system requires;

- low-low reactor water level
- steam supply
- DC power

The RCIC flow path is from the CST to the reactor vessel. Depending on the length of system operation, suction realignment to the torus may be required. The RCIC provides inventory makeup. Continued operation of the RCIC with suction from the suppression pool may require operation of the RHR system in the torus cooling mode. The system is provided with a turbine speed control system. Failure of this control system could result in no RCIC flow, inability to control RCIC flow, or excessive RCIC flow. The system is automatically tripped upon restoration of reactor water level to Level 8.

Success of the RCIC system is modeled for startup, short term, and long term operation. Startup operation of RCIC is modeled requiring a steam supply greater than 150 psi, a steam exhaust flow path to the suppression pool, hydraulic pressure from the DC powered oil pump, DC control power, a suction flow path from the CST or the suppression pool, and an injection flow path to the reactor. Startup of RCIC does not require operation of the pumps associated with the RCIC turbine seal leakage paths.

Short term operation of RCIC requires the same functions as for startup, except hydraulic pressure can be provided by either the shaft driven or motor driven oil pumps.

Long term operation of RCIC requires the realignment of suction flow path from the CST to the suppression pool. Suppression pool cooling using the RHR system may be required. The RCIC system can operate independent of plant AC power. However, the battery supply will be depleted as discussed in the electrical system analysis. Therefore, long term operation of RCIC may require the restoration of AC power or the use of an alternate DC power source.

RCIC pump room ventilation is required for successful startup, short term and long term RCIC operation. The RCIC pump room cooling units require ESW operation, or operator action to establish natural circulation.

3.2.1.12 Recirculation Pump Trip System

3.2.1.12.1 System Function

Figure 3.2-18 is a simplified diagram of the Recirculation Pump Trip System Logic. The purpose of the Recirculation Pump Trip (RPT) System is to reduce reactor power in response to a turbine trip or generator load rejection with the reactor at a high power level when the core is near the end of cycle. It also mitigates the effects of an ATWS. Trip of the recirculation pumps under these conditions limits the pressure excursion and provides negative reactivity insertion by increasing the core void fraction. RPT coils have been installed between the variable speed MG sets and the Recirculation Pump motors. The RPT feature accomplishes this objective by cutting off power to the recirculating pump motors, resulting in a rapid reduction in recirculation flow which increases the core void content.

The RPT-EOC system provides improved thermal margin for the limiting thermal transients which are either a turbine or generator trip without bypass near end of core life. The RPT-ATWS coupled with Alternate Rod Insertion (ARI) is needed in the unlikely event that the control rods fail to insert upon a scram signal. First RPT-ATWS rapidly reduces reactor power, then ARI opens two solenoid valves to bleed the air off the scram pilot air header, thus producing an alternate method of inserting control rods.

The RPT system utilizes a variety of inputs in order to open the RPT breakers, EPC-RPT logic utilizes the RPS relay contacts for four pressure switches installed in the turbine control valves to sense load reject, four position switches installed in the turbine stop valves to sense a turbine trip, and pressure switches in the first stage of the main turbine to sense pressure equivalent to 30% power. If a stop valve closure or a control valve fast closure occurs and power is above 30%, the logic will trip the

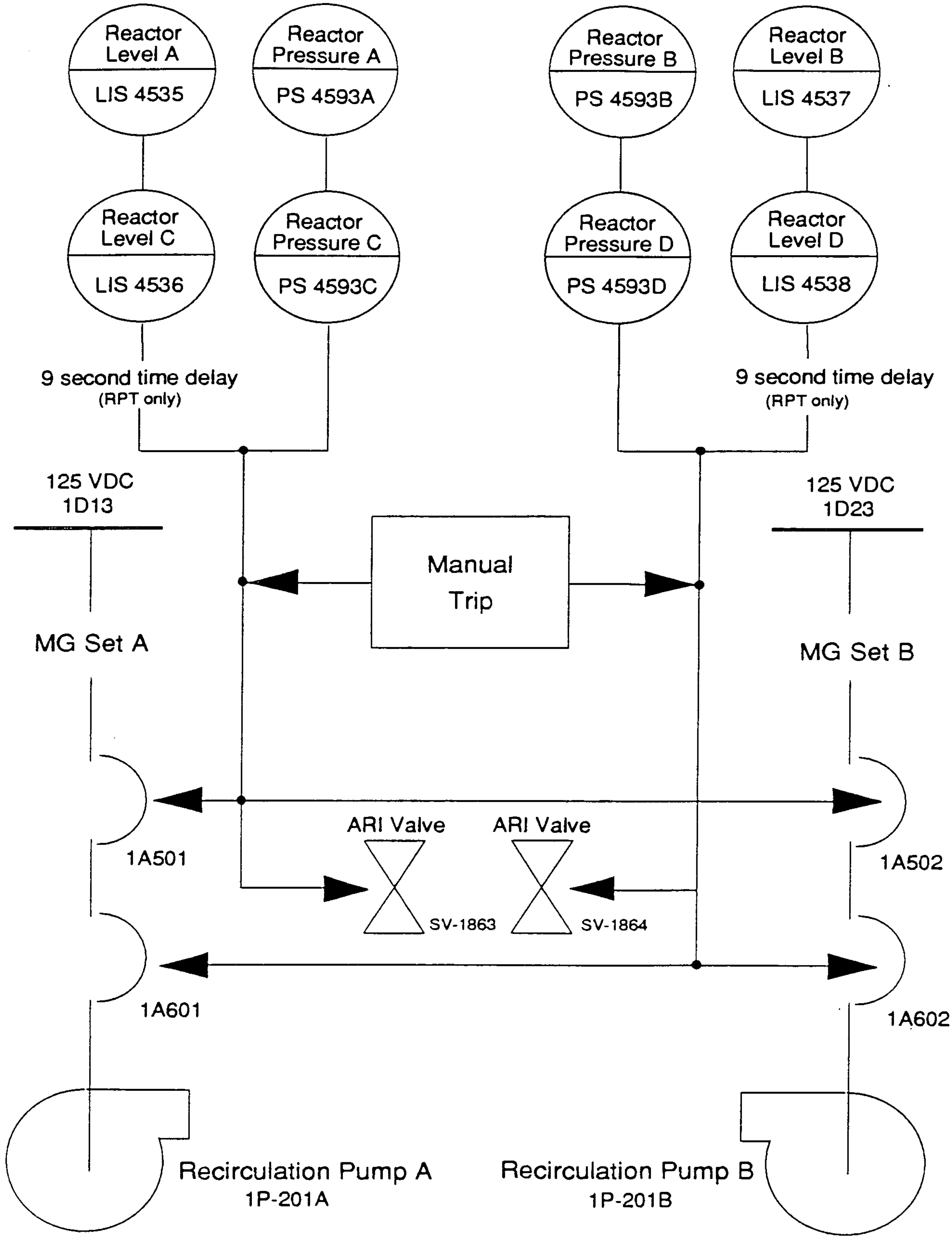


Figure 3.2-18

Recirculation Pump Trip Systems Logic

redundant RPT breakers between the M-G sets and the recirculation pump motors. The pressure switch (turbine control valves) and position switch (turbine stop valves) conditions are combined in a two-out-of-two once logic in two logic systems.

RPT-ATWS/ARI shares four level switches with NSSS for monitoring reactor level (wide range yarway). Reactor vessel pressure is monitored by four pressure switches dedicated to sensing an ATWS high pressure. The level switches and pressure switches are combined in a two-out-of-two once logic in two logic systems. This is the same as the RPT-EOC logic in that either logic trips both recirculation pumps. RPV Level 2 or high RPV pressure will activate the logic tripping the RPT breakers and initiating an ARI scram. When the ATWS/ARI logic is activated on a low level signal a time delay is inserted into the RPT breaker logic. This nine second time delay allows the Low Pressure Coolant Injection loop selection logic signal to be processed before the recirculation pumps trip. The ARI scram due to low level will not be delayed, it is processed immediately. When a high pressure trip signal is activated, it is processed immediately producing both RPT breaker trip and an ARI scram.

Manual tripping of RPT-ATWS/ARI is accomplished from Panel 1C05. Both A and B RPT/ARI Manual Initiate pushbuttons must be depressed in order to activate the logics. In the unlikely event of a Yarway reference leg failure (either side) ATWS/ARI low level would be disabled for both channels. Similarly, the failure of a GEMAC reference leg (either side) would disable the ATWS/ARI high pressure trip.

In addition to the turbine first stage pressure permissive (operating bypass) for RPT operation, a manual bypass (keylock switch on back panel) is provided to allow each RPT division to be disabled for maintenance. Application of 125 VDC to the breaker trip coils gives a designed interruption in power within 135 milliseconds of the switching event.

The RPT-ATWS/ARI trip logic can be tested via remote handswitches in the Recirc. MG room. When in test an RPT trip will not occur from the RPT-ATWS/ARI trip logic. An RPT trip from the EOC-RPT logic can still occur with RPT-ATWS/ARI in test.

3.2.1.12.2 System Interfaces and Dependencies

The RPT fault tree model includes support systems required for RPT to function in postulated accident scenarios. The systems which support specific RPT components and their effects on RPT operation are identified in the Dependency Matrix shown in Table 3.2-12.

In addition to the systems identified on a component level in the Dependency Matrix, the following systems provide support functions to the RPT System.

DC control power from panels ID13 and ID23 is required to provide power to the RPT logic circuit and to energize the RPT breaker trip coils.

3.2.1.12.3 System Fault Tree Model Assumptions

The following assumptions were used in the development of the RPT fault tree model.

1. Manual tripping of the recirculation pumps is given no credit in the analysis due to the very short response time (approximately 10 seconds).

2. Loss of DC power to the breaker trip coils is given a 0.0 failure probability. The few second time window for operability qualifies, for all practical purposes, as an instantaneous time frame. The event would be quantified as 1.0 for loss of DC initiators with mechanical RPS failure and failure to inject SLC; however, this sequence is not considered quantitatively due to the low frequency of the scenario.

Table 3.2-12

RECIRCULATION PUMP TRIP DEPENDENCY MATRIX

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
501	RPT BREAKER	PUMP A RPT	CLOSED	--	RPT FAILURES	1D13	--	--	--	--	--	VESSEL LEVEL PRESSURE
502	RPT BREAKER	PUMP B RPT	CLOSED	--	RPT FAILURES	1D13	--	--	--	--	--	VESSEL LEVEL PRESSURE
601	RPT BREAKER	PUMP A RPT	CLOSED	--	RPT FAILURES	1D23	--	--	--	--	--	VESSEL LEVEL PRESSURE
602	RPT BREAKER	PUMP B RPT	CLOSED	--	RPT FAILURES	1D23	--	--	--	--	--	VESSEL LEVEL PRESSURE

3. The fault tree logic does not model the ARI/RPT logic in detail. The failure of the logic channels is included in the fault tree as a "diamond" event.

4. The quantification of the ARI/RPT logic is based on a high-level fault tree module. The module only considers relay failures; contacts and other such items are disregarded. This should have little effect, as common cause failures are the dominant contributors to the channel logic failure.

5. The basic event for RPT breaker failure includes normal mechanical failures, including failure of the trip coil. No DC to the trip coil is modeled under the loss of DC events and is quantified as stated in Assumption #2.

6. The common cause failure groups for the RPT breakers are restricted to one common cause group per pump; common cause combinations between pumps are not considered. This may be logically non-conservative; however the quantitative probability is judged to be conservative already since the independent failure rate may include common cause "hardened grease" events.

7. Failure to generate low water level or high drywell signals from the primary element channels is not explicitly considered in the model. The relatively small importance on the fault tree results did not warrant the inclusion of these items. Refer to comments #5 and #6.

8. The RPT breaker failure probability is based on a GE survey. The failure probability appears high; however, the magnitude of the estimate is not unfounded considering the industry concern regarding "hardened grease' related failures.

3.2.1.12.4 Success Criteria

For a successful RPT, one breaker on each recirculation pump must open. This is accomplished by energizing the trip coil on each breaker upon receipt of a signal from the ATWS-RPT/ARI logic.

3.2.1.13 Residual Heat Removal System

3.2.1.13.1 System Function

The RHR system may operate functionally in four configurations. The major functional configurations are:

- Low Pressure Coolant Injection (LPCI)
- Containment Spray
- Torus Cooling
- Shutdown Cooling

Each of these modes of operation will be discussed separately. The system can also be used to support:

- Fuel Pool Cooling
- Reactor Vessel Draining
- Torus Draining
- Reactor or Containment flood with RHRSW

The safety design basis objective of the RHR System is to act automatically, in combination with other emergency core cooling systems, to restore and maintain the coolant inventory in the reactor vessel such that the core is adequately cooled to preclude excessive fuel clad temperature following a design-basis LOCA. Figure 3.2-19 depicts the RHR System.

From a safety perspective, the RHR System can provide two major functions - core cooling injection and decay heat removal. The heat removal can be provided via connection to the torus (torus cooling), via connection to the recirculation line (shutdown

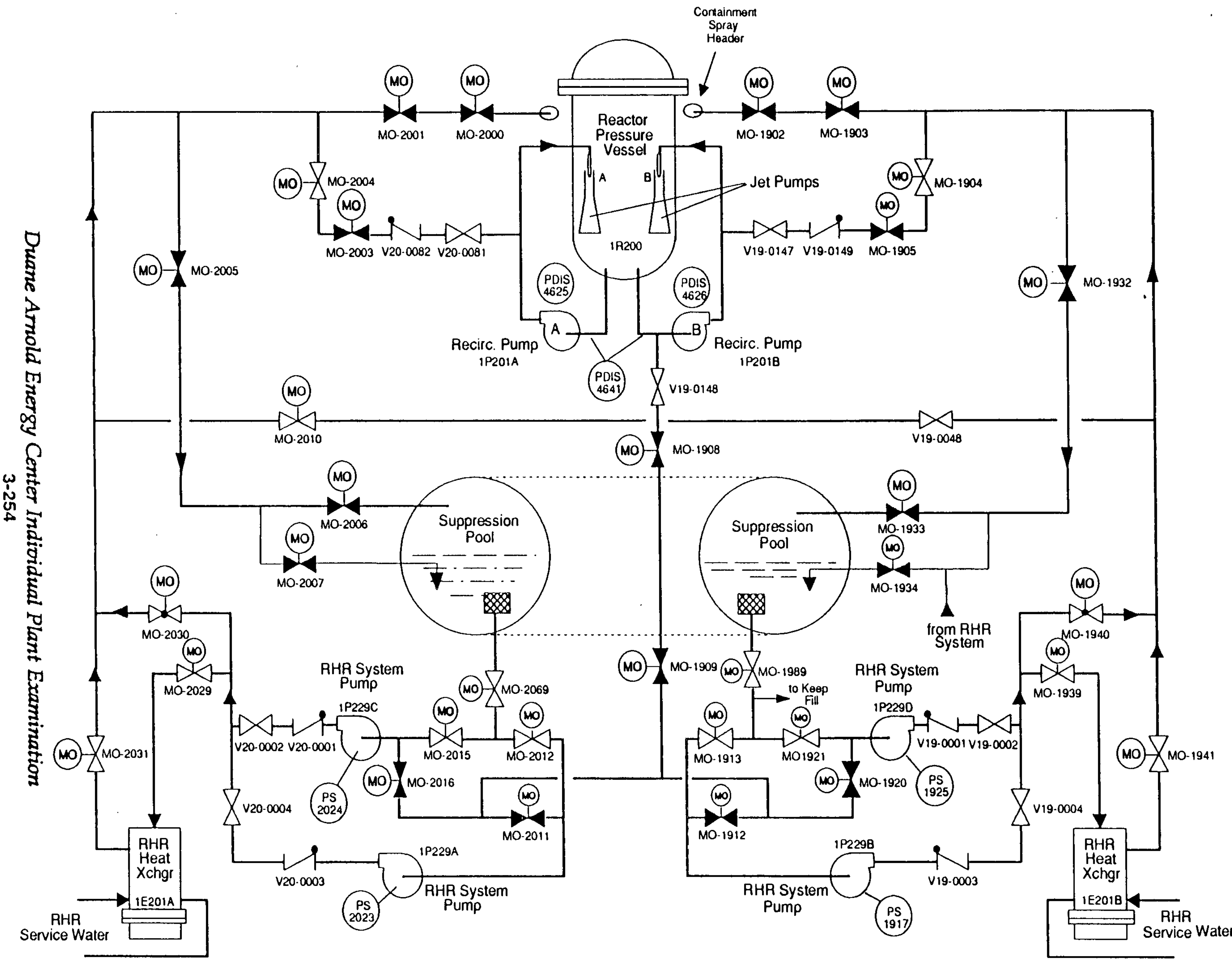


Figure 3.2-19
Residual Heat Removal System

cooling), or via connection to the containment drywell and wetwell (containment spray). The major components of the RHR system are:

- RHR Pump and Motors
- RHR heat exchangers
- RHR Valves
- RHR/Core Spray Fill Pump

RHR Pumps and Motors - The four RHR pumps are 4800 gpm, single stage, vertically mounted, centrifugal type, with mechanical seals. Each pump capacity was chosen based on supplying one third of the design basis LOCA required LPCI rated flow of 14400 gpm. The pumps are located at a lower elevation than the suppression pool to provide adequate net positive suction head under all operating modes. The pump mechanical seals are provided with cool clean water, by use of a cyclone separator driven by pump differential pressure, and a seal cooler which is cooled by the Emergency Service Water System.

Check valves are provided on each pump discharge line to prevent backflow through the pumps and allow the discharge piping to be filled and pressurized up to the closed inboard injection valve by the keepfill system. Since the discharge header is pressurized, the pumps can be started with the discharge valves full open. Each pump is protected from internal overheating at low flows, by a minimum flow bypass line which routes water from the discharge line to the suppression pool.

Each of the motors is cooled by air, forced by fan blading attached to the motor shaft. When the pumps are not in service, motor windings are kept dry by a 600 watt space heater, energized when the motor air circuit breaker is in the open position.

RHR Heat Exchangers - Two vertically mounted, inverted U-tube, heat exchangers remove heat from the reactor or suppression pool water and reject it to the RHR Service Water flowing through the double pass, stainless steel tubes. They are sized on the basis of their required duty for the shutdown cooling mode.The heat exchanger shell and tube side are designed for 450 psig and 32° to 400°F operation.

RHR Valves - The RHR System motor operated valves are controlled to perform various operations both automatically as in LPCI operation, and remote manually, as in containment spray and shutdown cooling operations. All valve motors are provided with thermal overload trips in addition to breaker short circuit current protective trips.

Operating experience has shown that at operating pressures it is possible for the check valves inside containment and the inboard LPCI isolation valves to leak. This would be indicated by the RHR heat exchanger pressure controller indicating above the normal 70 psig reading. This leakage, if not stopped, could lead to over-pressurization of the RHR system piping. Closing the outboard LPCI isolation valve will stop the leakage.

RHR/Core Spray Fill Pump A - The purpose of the keep fill system is to minimize water hammer. The RHR/Core Spray Fill Pump is used to maintain the discharge piping downstream of the pump discharge check valve full of water.

The pump takes a suction from the "B" RHR torus suction strainer and supplies water to the discharge piping in the RHR System and Core Spray System. This pressure is maintained at approximately 70 psig to ensure complete filling. A minimum flow bypass line is installed with a manually operated throttle valve, which allows flow through the pump back to the torus to prevent pump damage.

The following are brief functional descriptions of the major modes of operation:

Low Pressure Coolant Injection - In the event of a need for coolant makeup to the reactor vessel, the RHR System can be used in the LPCI mode to supply water from the torus to the reactor vessel to flood the core and prevent fuel clad damage. LPCI is designed to restore and maintain reactor water level above 2/3 core height following a LOCA. When the break is such that it exceeds the capacity of HPCI, it would be large enough to reduce vessel pressure so that LPCI and Core Spray systems can operate to provide core cooling. For an intermediate size break, where the vessel does not depressurize, and HPCI is unable to maintain level for whatever reason, the ADS system will reduce pressure so that LPCI and Core Spray can operate. For transients, the LPCI system can be operated after depressurization to provide coolant flow.

LPCI is normally in standby when the plant is operating at 100% power. The "A" and "C" RHR pumps and "A" RHR heat exchanger are located in the South East Corner Room. The "B" and "D" RHR pumps and "B" RHR heat exchanger are located in the North West Corner Room. Separation of the pumps, piping, controls, and instrumentation of each loop is such that any single physical event cannot make both loops inoperable.

The automatic actuation of LPCI mode will result from a triple low level in the reactor vessel, 64.5", or high drywell pressure, 2 psig. Each pump has a rated capacity of 4800 gpm. Each pump capacity was chosen based on supplying one third of the required LPCI rated flow of 14,400 gpm against a system head corresponding to a vessel pressure of 20 psig based on individual pump tests. The fourth pump is an installed spare.

There are two distinct logics associated with the LPCI mode; LPCI Loop Selection, and LPCI Injection Initiation (see Section 3.2.1.10).

Torus Cooling - Torus cooling is a mode of RHR which can be used to remove heat from the torus water by using the RHR pumps and heat exchangers in a closed loop. Flow is from the torus, through a heat exchanger where RHRSW provides cooling, and back to the torus via the test return line through MO1932 (2005) and MO1934 (2007). The discharge will normally be underwater in the torus. This mode is manually initiated and is placed in operation to limit the temperature of the water in the torus. The design basis after a LOCA has occurred is the pool temperature to be maintained below 170°F.

Torus cooling is required whenever the water temperature exceeds 95°F, such as following RCIC or HPCI operation, SRV operation, or in a post accident situation. Either loop may be aligned for torus cooling. If LPCI mode of the RHR system has been initiated, LPCI must be allowed to operate for 10 minutes to ensure adequate core cooling before realigning the system for another mode. If the torus temperature is above 95°F after the 10 minute period, one of the RHR system loops would be realigned by the operator for torus cooling.

Containment Spray - In the unlikely event of a nuclear system break within the Primary Containment, after the reactor water level has been restored, the containment spray cooling mode may be manually placed in operation to spray water into the drywell and/or torus to condense steam and cool non-condensible gases to prevent excessive containment pressure and temperature.

Containment spray can be manually initiated only if reactor water level has been restored by the ECCS systems or if this requirement is over-ridden by the operator. The RHR pumps take suction off the torus and may discharge around or through the heat exchangers and through containment spray valves MO2000, MO2001, MO1902, MO1903 and or torus spray valves MO2005, MO2006, MO1932, MO1933.

The containment cooling mode provides containment spray capability as an alternative method of reducing containment pressure following a LOCA. The water pumped through the RHR heat exchangers may be diverted to two spray headers in the drywell and one above the torus. The spray headers in the drywell condense any steam that may exist in the drywell thereby lowering containment pressure. The spray collects in the bottom of the drywell until the water level rises to the level of the pressure suppression vent lines where it overflows and drains back to the torus, completing the flow loop.

Approximately 5% of this flow may be directed to the torus spray ring to cool any non-condensible gases collected in the free volume above the torus.

A postulated condition where containment sprays may be desirable is in the case of a small steam leak in the drywell. The consequence of such an occurrence, assuming no corrective spray action is taken, is the possibility of the containment atmosphere exceeding the containment design temperature, thus presenting the potential to exceed the design temperature of the drywell vessel.

Shutdown Cooling - Shutdown Cooling is a mode of the RHR system which may be used during a normal shutdown and cooldown. The initial phase of reactor cooldown is normally accomplished by dumping steam from the reactor vessel to the main condenser with the main condenser acting as the heat sink. When reactor coolant temperature has decreased to a value where the steam supply pressure is not sufficient to maintain turbine shaft gland seals or vacuum in the main condenser, then RHR is placed in Shutdown Cooling.

The RHR Loop B is the preferred loop for the Shutdown Cooling mode because MO2010, RHR Crosstie allows starting and stopping of SDC without having to enter the Torus. Reactor coolant is pumped by the RHR pumps off the "B" Recirc loop just upstream of the Recirc suction valve and is discharged through one or both of the RHR heat exchangers depending on the decay heat levels. Cooling takes place by transferring heat to the RHRSW. When the decay heat level has decreased sufficiently, the entire SDC load can be shifted to one RHR system heat exchanger leaving the other available for any other cooling loads.

The standard practice at DAEC when in cold shutdown is to operate with shutdown cooling flow greater than 4000 gpm. This provides forced circulation and proper mixing preventing stratification in the vessel. Using adequate flow for forced circulation maintains reactor water temperature uniformly below 212°F. Typically, the temperature is maintained between 150°F and 180°F, except when refueling or performing vessel work. Past operating practice has shown that the 4000 gpm shutdown cooling flow provided by one RHR pump is sufficient to maintain reactor water temperature below 212°F during cold shutdown. Reactor coolant temperature can be maintained by controlling 4000 gpm through the RHR heat exchanger and RHR heat exchanger bypass flow.

Since DAEC is a LPCI loop select plant, the shutdown cooling mode of the RHR system does not meet later plant requirements for safety grade redundancy in this function. Certain components, such as the shutdown cooling recir loop suction valves, are single point vulnerabilities. Other normal and emergency means of decay heat removal are available. One such mode, evaluated under Appendix R, is alternate shutdown cooling. In this mode, SRVs are maintained open with low pressure systems providing flow to the vessel. At reactor pressures below 400 psig, the open SRVs provide subcooled or saturated water flow to the suppression pool which in turn is cooled by either loop of RHR.

3.2.1.13.2 System Interfaces and Dependencies

The RHR fault tree model includes support systems required for RHR to function in postulated accident scenarios. The systems which support specific RHR components and their effects on RHR operation are identified in the Dependency Matrix shown in Table 3.2-13.

In addition to the systems identified on a component level in the dependency matrix, the following systems provide support functions to the RHR system:

Table 3.2-13

RESIDUAL HEAT REMOVAL SYSTEM

COMP	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	ACTUATION SIGNAL
1P229A	RHR PUMP A	1-D	OFF/STBY	1A3	PUMP DEENGZD	1D13/ 1D23	LOSS OF CONTROL POWER	-	-	ESW[1]	V-AC-012	LOW-LOW-LOW RX LEVEL OR HIGH DRYWELL PRESS.
1P229B	RHR PUMP B	1-B	OFF/STBY	1A4	PUMP DEENGZD	1D13/ 1D23	LOSS OF CONTROL POWER	-	-	ESW[1]	V-AC-011	LOW-LOW-LOW RX LEVEL OR HIGH DRYWELL PRESS.
1P229C	RHR PUMP C	1-D	OFF/STBY	1A3	PUMP DEENGZD	1D13/ 1D23	LOSS OF CONTROL POWER	-	-	ESW[1]	V-AC-012	LOW-LOW-LOW RX LEVEL OR HIGH DRYWELL PRESS.
1P229D	RHR PUMP D	1-B	OFF/STBY	1A4	PUMP DEENGZD	1D13/ 1D23	LOSS OF CONTROL POWER	-	-	ESW[1]	V-AC-011	LOW-LOW-LOW RX LEVEL OR HIGH DRYWELL PRESS.
MO-1902	INBD DRYWELL CONT. SPRAY VALVE LOOP B	CONT.	CLOSED	1B44	VALVE FAILS TO OPERATE (OPEN & CLOSE)	-	-	-	-	-	-	-
MO-1903	OUTBD DRYWELL CONT. SPRAY VALVE LOOP B	CONT.	CLOSED	1B44	VALVE FAILS TO OPERATE (OPEN & CLOSE)	-	-	-	-	-	-	-
MO-2000	INBD DRYWELL CONT. SPRAY VALVE LOOP A	CONT.	CLOSED	1B34	VALVE FAILS TO OPERATE (OPEN & CLOSE)	-	-	-	-	-	-	-
MO-2001	OUTBD DRYWELL CONT. SPRAY VALVE LOOP A	CONT.	CLOSED	1B34	VALVE FAILS TO OPERATE (OPEN & CLOSE)	-	-	-	-	-	-	-

COMP	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	ACTUATION SIGNAL
MO-1905	LPCI INBD INJ VALVE LOOP B	2-D	CLOSED	1B34A/ 1B44A	VALVE FAILS TO OPEN	-	-	-	-	-	-	-
MO-2003	LPCI INBD INJ VALVE LOOP A	2-D	CLOSED	1B34A/ 1B44A	VALVE FAILS TO OPEN	-	-	-	-	-	-	-
MO-1904	LPCI OUTBD INJ VALVE LOOP B	2-D	OPEN	1B44	NONE DESIRED OPEN	-	-	-	-	-	-	-
MO-2004	LPCI OUTBD INJ VALVE LOOP A	2-D	OPEN	1B34	NONE DESIRED OPEN	-	-	-	-	-	-	-
MO-1908	SHUTDOWN COOLING SUPPLY ISOLATION VALVE	CONT.	CLOSED	1B34A/ 1B44A	VALVE FAILS TO OPERATE (OPEN & CLOSE)	-	-	-	-	-	-	-
MO-1909	SHUTDOWN COOLING SUPPLY ISOLATION VALVE	2-D	CLOSED	1B34A/ 1B44A	VALVE FAILS TO OPERATE (OPEN & CLOSE)	-	-	-	-	-	-	-
MO-1912	RHR PUMP 1P229B SUCT HDR B	1-B	CLOSED	1B44	VALVE FAILS TO OPEN	-	-	-	-	-	-	-
MO-1920	RHR PUMP 1P229D SUCT HDR B	1-B	CLOSED	1B44	VALVE FAILS TO OPEN	-	-	-	-	-	-	-
MO-2011	RHR PUMP 1P229A SUCT HDR A	1-D	CLOSED	1B34	VALVE FAILS TO OPEN	-	-	-	-	-	-	-
MO-2016	RHR PUMP 1P229C SUCT HDR A	1-D	CLOSED	1B34	VALVE FAILS TO OPEN	-	-	-	-	-	-	-
MO-1913	RHR PUMP 1P229B TORUS SUCT. VALVE LOOP B	1-B	OPEN	1B44	VALVE FAILS TO OPEN	-	-	-	-	-	-	-
MO-1921	RHR PUMP 1P229D TORUS SUCT. VALVE LOOP B	1-B	OPEN	1B44	VALVE FAILS TO OPEN	-	-	-	-	-	-	-

COMP	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	ACTUATION SIGNAL
MO-2012	RHR PUMP 1P229A TORUS SUCT. VALVE LOOP A	1-D	OPEN	1B34	VALVE FAILS TO OPEN	-	-	-	-	-	-	-
MO-2015	RHR PUMP 1P229C TORUS SUCT. VALVE LOOP A	1-D	OPEN	1B34	VALVE FAILS TO OPEN	-	-	-	-	-	-	-
MO-1932	TORUS SPRAY HDR VALVE LOOP B	1-A SOUTH	OPEN	1B44	NONE DESIRED OPEN	-	-	-	-	-	-	-
MO-1933	TORUS SPRAY HDR VALVE LOOP B	1-A SOUTH	CLOSED	1B44	VALVE FAILS TO OPEN	-	-	-	-	-	-	-
MO-1934	LPCI TO TORUS TEST LINE VALVE LOOP B	1-A SOUTH	CLOSED	1B44	VALVE FAILS TO OPEN	-	-	-	-	-	-	-
MO-2005	TORUS SPRAY HDR VALVE LOOP A	1-A NORTH	OPEN	1B34	NONE DESIRED OPEN	-	-	-	-	-	-	-
MO-2006	TORUS SPRAY HDR VALVE LOOP A	1-A NORTH	CLOSED	1B34	VALVE FAILS TO OPEN	-	-	-	-	-	-	-
MO-2007	LPCI TO TORUS TEST LINE VALVE LOOP A	1-A NORTH	CLOSED	1B34	VALVE FAILS TO OPEN	-	-	-	-	-	-	-
MO-1939	RHR HX B INLET VALVE	1-B	OPEN	1B44	NONE DESIRED OPEN	-	-	-	-	-	-	-
MO-1940	RHR HX B BYPASS VALVE	1-B	OPEN	1B44	VALVE FAILS TO OPEN	-	-	-	-	-	-	-
MO-1941	RHR HX B OUTLET VALVE	1-B	OPEN	1B44	NONE DESIRED OPEN	-	-	-	-	-	-	-
MO-2029	RHR HX A INLET VALVE	1-D	OPEN	1B34	NONE DESIRED OPEN	-	-	-	-	-	-	-
MO-2030	RHR HX A BYPASS VALVE	1-D	OPEN	1B34	VALVE FAILS TO CLOSE	-	-	-	-	-	-	-
MO-2031	RHR HX A OUTLET VALVE	1-D	OPEN	1B34	NONE DESIRED OPEN	-	-	-	-	-	-	-

COMP	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	ACTUATION SIGNAL
MO-1989	TORUS OUTLET VALVE LOOP B	1-A NORTH	OPEN	1B44	NONE DESIRED OPEN	-	-	-	-	-	-	-
MO-2069	TORUS OUTLET VALVE LOOP A	1-A SOUTH	OPEN	1B34	NONE DESIRED OPEN	-	-	-	-	-	-	-
MO-2010	RHR LOOPS X-TIE VALVE	1-A NORTH	OPEN	1B34	NONE DESIRED OPEN	-	-	-	-	-	-	-

1. Component cooling is for RHR pump seals only.

RHR Service Water System - RHR Service Water provides the cooling medium to the RHR heat exchangers for the various cooling modes of the RHR System. Additionally, the RHR Service Water System may be cross-connected to the RHR System to provide a source of water for post accident flooding of the reactor or containment.

Emergency Service Water System - The Emergency Service Water System provides the necessary cooling water to the RHR Pump seal coolers and the RHR Pump Room cooling units.

Keep Full System - The RHR/Core Spray Fill Pump maintains the discharge piping in both systems filled and pressurized to 65-75 psig. The Core Spray System minimum flow bypass and full flow test lines return water to the suppression pool via the RHR full flow test lines.

Diesel Generators - The Standby Diesel Generators are the emergency power source to the essential buses should off-site power be lost. During a LOCA accompanied by a loss of off-site power the Standby Diesel Generators provide power to the RHR Pumps.

Automatic Depressurization System - Two pressure switches at the discharge of each RHR Pump provide permissive signals to ADS. These pressure switches indicate to ADS that the RHR Pumps are running when discharge pressure is indicated.

3.2.1.13.3 System Fault Tree Model Assumptions

1. The pump motors are air cooled; no separate lube oil system is required.

2. The suction cross connect is included in the fault tree model.

3. For LPCI mode of RHR system operation, the success criteria requires one of four RHR pumps for inventory makeup. One RHR pump is assumed to be sufficient based on NEDO-24708A.

4. The RHRSW crosstie to RHR for alternate low pressure injection is modeled separately.

5. Room cooling is not required for any mode of RHR operation.

6. Minimum flow lines modeled such that failure does not cause failure of pumps based on observed BWR experience.

7. The shutdown cooling mode of RHR system operation is currently included in the fault tree model, however, it is not utilized by the IPE event tree.

3.2.1.13.4 Success Criteria

The RHR System is a multi-function system that provides low pressure injection for core cooling, decay heat removal, and containment heat removal. There are four functional modes for the system each with its own success criteria.

Low Pressure Coolant Injection

The RHR LPCI mode is a low pressure injection mode that is sized to provide adequate core cooling for all postulated loss of coolant accidents where the reactor is depressurized. The system is designed with a "LOOP Selection Logic" that uses pressure switches to detect a ruptured recirculation Loop and automatically realigns valves such that all of the system flow is directed to the intact loop. Failure of this selection logic during a postulated double ended guillotine break of a recirculation loop results in loss of all LPCI vessel injection capability. Failure of the loop selection logic has no adverse impact for all loss of coolant events where the recirculation piping is intact.

The success criteria for the LPCI mode of RHR operation is injection from 1 of 4 pumps into the reactor vessel. The required flow path consists of suction from the suppression pool, isolation of the full flow test line, and proper alignment of the injection valves based on signal from the Loop Selection Logic. Success of opening the minimum flow line is required and no RHR pump cooling is required.

Automatic actuation of the LPCI mode requires all of the following:

- high drywell pressure signal or lo-lo-lo reactor water level signal
- DC control power
- AC pump and valve power
- low reactor pressure signal to open injection valves

The RHR system is provided with an automatic line fill system which maintains the system discharge piping filled. This is intended to prevent water hammer and other dynamic effects that could occur upon pump start if the line where empty. Failure of this line fill system is a pre-initiator which, if concurrent with operator failure to restore the function before the line empties, may cause RHR system failure due to pipe rupture upon pump start.

Containment Spray - The initiation of containment/suppression pool spray is performed manually by the operator. The system is provided with interlocks to prevent the spray valves from opening unless LPCI signal is present and reactor vessel level has been recovered. The success criteria for containment spray requires one of four RHR pumps taking suction from the torus and discharging to spray headers in the drywell. No RHR pump cooling is required.

Torus Cooling - The torus cooling mode of RHR is used to provide containment heat removal. In this mode, the RHR system is aligned to take suction from the torus and discharges back to the torus, the drywell spray header, or reactor vessel. The flow path includes the RHR heat exchangers. In this operating mode, heat from the containment is rejected to the RHR Service Water System.

The success criteria for torus cooling requires operator action and operation of at least one RHR pump taking suction from the torus and discharging back to the torus, the drywell spray header, or reactor vessel after passing through the RHR heat exchanger. RHR service water flow through the credited heat exchanger is required. One of four RHRSW pumps and makeup to the stilling basin from one of four river water pumps is required. RHR pump cooling is not required.

Shutdown Cooling - The initiation of shutdown cooling is performed manually by the operator. The success criteria for shutdown cooling is the same as that for torus cooling except suction is taken from the suction side of the "B" recirculation pump and the return flow is via one of the LPCI injection paths. RHR pump cooling is not required. Shutdown cooling is used during normal reactor shutdown after the reactor has de-pressurized to approximately 135 psig.

The shutdown cooling mode of RHR system operation is currently not credited in the IPE analysis.

3.2.1.14 Residual Heat Removal Service Water System

3.2.1.14.1 System Function

The purpose of the RHR Service Water (RHRSW) System is to remove heat from the RHR heat exchangers during shutdown cooling, steam condensing, containment and torus cooling modes of RHR. A secondary function of the RHRSW system is to provide an alternate source of water for emergency containment cooling or injection into the reactor vessel, or for containment flooding after a LOCA.

The following design bases are incorporated into the RHR Service Water system:

a. The RHR Service Water pumps are sized to provide a pressure at the cooling water outlet of the RHR system heat exchangers that is at least 20 psi greater than the reactor coolant pressure at the inlet of the heat exchangers during the shutdown cooling and RCIC steam condensing modes of operation. This criterion ensures that in the event of a heat exchanger tube leak, the radioactive coolant does not leak into the service water.

b. The RHR Service Water heat exchangers are sized on the basis of their required duty for the shutdown cooling function.

RHR Service Water is normally pumped through the tube side of the heat exchanger and discharged to the cooling towers via the Circulating Water discharge line from the main condenser. Upstream of the heat exchangers is a cross-tie line that can be used by either or both subsystems to bypass the heat exchanger for post-accident flooding of the primary containment or the reactor.

Each pump discharges into a line that includes a check valve and a locked open gate valve. Each pair of pump discharge lines (A and C, B and D) connect to a common line that runs to the heat exchanger. The common line contains a self cleaning strainer, manually operated bypass gate valve, and a cross-tie line.

Each strainer is automatically placed in a continuous backwash mode of operation upon start of either subsystem pump. A flow control valve, which is operated by air, will divert a small amount of water through the strainer and to the stilling basin. This will continuously flush the strainer. A manually operated gate valve is provided to bypass around the strainer in the event of strainer difficulties.

The RHR Service Water System is a dual (redundant) system. Each subsystem contains two half-capacity, parallel-connected RHR Service Water pumps and associated piping and valving that are operated in conjunction with a single RHR heat exchanger. Subsystem A includes pumps 1P-22A and C and operates with heat exchanger 1E-201A. Subsystem B includes pumps 1P-22B and D and operates with heat exchanger 1E-201B.

The two pairs of RHR Service Water pumps are located in the pump house. They have first priority to water supplied by the River Water Supply System since river water flows into the RHR Service Water and Emergency Service Water pit from the stilling basin which in turn overflows to the Circulating Water pump pit.

Figure 3.2-20 shows a simplified diagram of the RHR Service Water System. The RHR Service Water pumps take a suction on two wet pits. (See Figure 3.2-9 for general water pits layout.) RHR Service Water pumps A & C and ESW pump A take a suction on wet pit "A", while RHR Service Water pumps B & D and ESW pump B take a suction on wet pit "B".

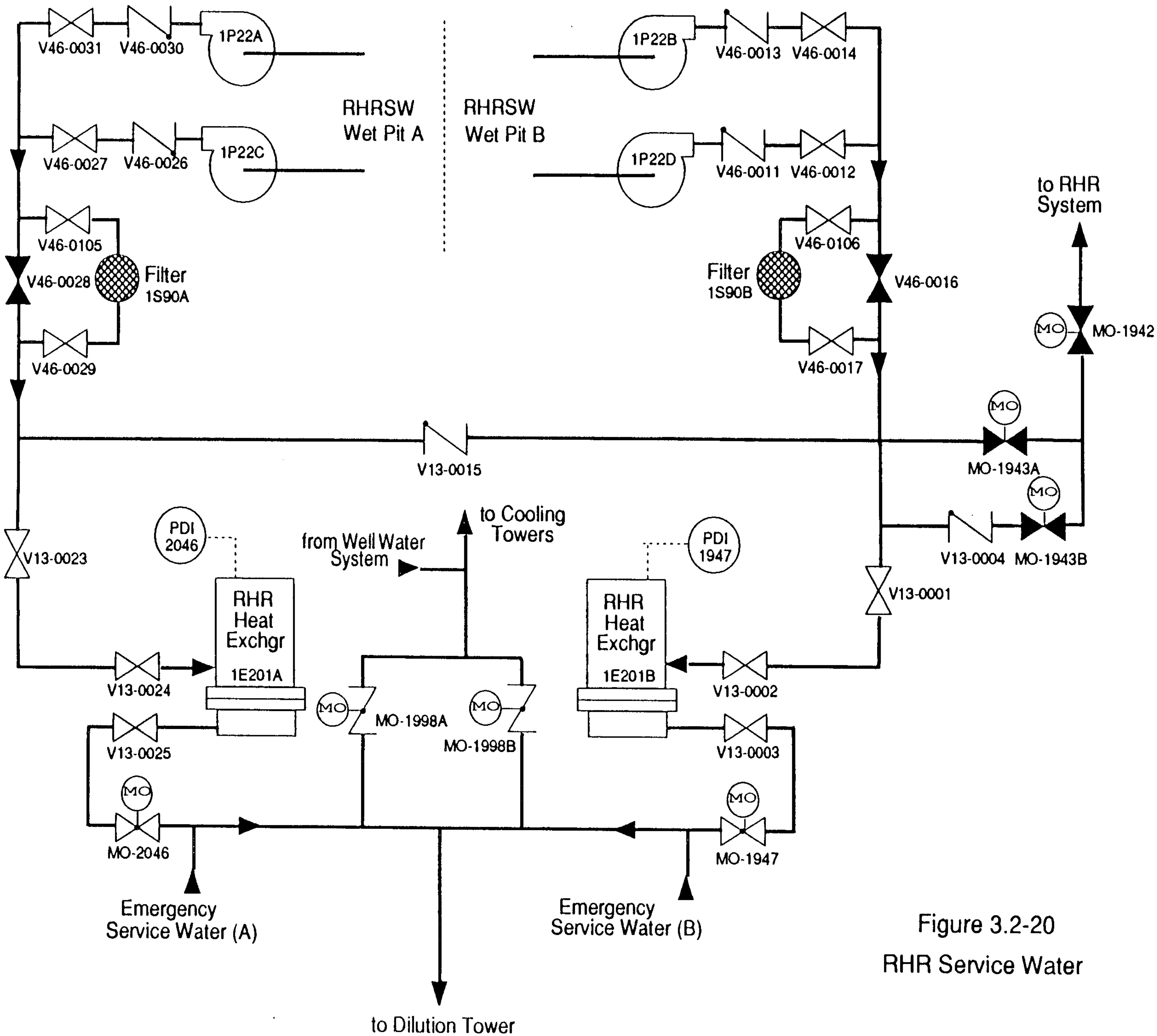


Figure 3.2-20
RHR Service Water

The normal path for RHR Service Water is from the pressure regulating valve through motor operated butterfly valves (MO-1998A or MO-1998B) to the cooling towers. A possible problem with this flow is flooding of the cooling tower basins which could occur because of filling of the Circulating Water pit, which is at the same level as the cooling tower basins, during periods of small evaporative losses. Once level in the controlling cooling tower basin reaches high level, the River Water Supply valve to the stilling basin will be throttled shut. However, the operating RHR Service Water pumps will continue to pump water into the cooling towers and once level in the pit reaches the low level mark, a level switch actuates to force the River Water Supply valves full open. The Circulating Water pit and the cooling tower basin will continue to fill until the overflow mark of 11 inches above the normal level is reached in the cooling tower basin. For this condition, an alternate discharge path is provided.

The alternate discharge path for RHR Service Water is through the rupture disc assemblies (rupture discs removed) and then to the radwaste dilution line. From the dilution line, the service water will flow through butterfly valve V-42-12 to the dilution structure and out to the river via the canal.

The cross-tie for primary containment or reactor flooding consists of a line from each subsystem common pump discharge upstream of the heat exchanger and includes a check valve and a normally closed motor operated gate valve. The subsystem legs terminate in a common line with another normally closed motor operated gate valve and a 1" side leg. The side leg discharges to the open radwaste drain and includes a normally open solenoid controlled gate valve in tandem with a normally open, manually operated gate valve. The motor operated valves are individually and manually operated by keylocked switches in the control room. Thus, two valves must be opened (one in a subsystem leg and the valve in the common line) to set up the flooding alignment. The solenoid operated drain valve closes when any of the motor operated valves moves from 100% closed, and re-opens when all motor operated valves are closed. The motor operated butterfly valves in the normal cooling tower path or the discharge valve in the alternate discharge path (V-42-12) must be shut when RHR Service Water is used for flooding.

During normal plant operations, the RHRSW System is required to support the RHR System in the shutdown cooling mode in order to achieve plant cooldown while maintaining reactor vessel pressure control. RHRSW provides the means of removing the heat from the RPV via the RHR heat exchangers, once RPV pressure is below the shutdown cooling high Rx pressure auto-closure interlock.

3.2.1.14.2 System Interfaces and Dependencies

The RHRSW fault tree model includes support systems required for RHRSW to function in postulated accident scenarios. The systems which support specific RHRSW components and their effects on RHRSW operation are identified in the Dependency Matrix shown in Table 3.2-14.

In addition to the systems identified on a component level in the dependency matrix, the following systems provide support functions to the RHRSW systems:

Emergency Service Water (ESW) System

The ESW System supplies the cooling water for the RHRSW pump motors. ESW pump 1P-99A will supply cooling water to RHR Service Water subsystem "A" (pumps A and C), and ESW pump 1P-99B will supply cooling water to subsystem B (pumps B and D). ESW is started prior to starting an RHR Service Water pump.

The ESW System also utilizes the same discharge piping (same discharge path) as the RHR Service Water System.

Table 3.2-14

RHRSW SYSTEM DEPENDENCY MATRIX

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
CV-4910A	RHRSW to dilution structure control valve		Throttled	-	-	-	-	Instrument Air	Loss of motive power	-	-	SV-4934 & SV-4935 control instrument air. Solenoids deenergize on Hi Drywell Press, Lo Lo Lo Rx Level, ESW Wet Prt Low Level and Loss of Power
CV-4910B	RHRSW to dilution structure control valve		Throttled	-	-	-	-	Instrument Air	Loss of motive power	-	-	SV-4934 & SV-4935 control instrument air. Solenoids deenergize on Hi Drywell Press, Lo Lo Lo Rx Level, ESW Wet Prt Low Level and Loss of Power
CV-4909	RHRSW to dilution structure block valve		Throttled	-	-	-	-	Instrument Air	Loss of motive power	-	-	SV-4934 & SV-4935 control instrument air. Solenoids deenergize on Hi Drywell Press, Lo Lo Lo Rx Level, ESW Wet Prt Low Level and Loss of Power
MO-1942	RHRSW to RHR crosstie		Closed	1B34	None for cooling function cannot be realigned for alternate injection.	-	-	-	-	-	-	
MO-1943A	RHRSW pumps A & C crosstie	16-B	Closed	1B34	None for cooling function cannot be realigned for alternate injection.	-	-	-	-	-	-	
MO-1943B	RHRSW pumps B & D crosstie	16-A	Closed	1B44	None for cooling function cannot be realigned for alternate injection.	-	-	-	-	-	-	

Table 3.2-14
RHRSW DEPENDENCY MATRIX
(continued)

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
MO-1947	RHR HX "B" service water outlet valve	16-A	Closed	1B44	None for alternate injection. Loss of motive power for cooling function.	1D13	Loss of indication and control power	-	-	-	-	
MO-2046	RHR HX "A" service water outlet valve	16-B	Closed	1B34	None for alternate injection. Loss of motive power for cooling function.	1D23	Loss of indication and control power	-	-	-	-	
MO-1998A	"A" ESW/RHRSW discharge to CLG Towers	16-B	Open	1B34	None; fail open	-	-	-	-	-	-	
MO-1998B	"B" ESW/RHRSW discharge to CLG Towers	16-A	Open	1B44	None; fail open	-	-	-	-	-	-	
1P022A	"A" RHRSW pump	16-B	Off	1A3	Fail to start on demand	1D13	Loss of circuit breaker control power	-	-	ESW	-	
1P022B	"B" RHRSW pump	16-A	Off	1A4	Fail to start on demand	1D23	Loss of circuit breaker control power	-	-	ESW	-	
1P022C	"C" RHRSW pump	16-B	Off	1A3	Fail to start on demand	1D13	Loss of circuit breaker control power	-	-	ESW	-	
1P022D	"D" RHRSW pump	16-A	Off	1A4	Fail to start on demand	1D23	Loss of circuit breaker control power	-	-	ESW	-	
1P117A	River water supply pump	17-A	1 of 2 in each loop operating	1B09	Fails to run	1D11	Loss of control power -fails to start	-	-	-	-	

Table 3.2-14
RHRSW DEPENDENCY MATRIX
(continued)

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
1P117B	River water supply pump	17-B	1 of 2 in each loop operating	1B20	Fails to run	1D21	Loss of control power - fails to start	-	-	-	-	
1P117C	River water supply pump	17-A	1 of 2 in each loop operating	1B09	Fails to run	1D11	Loss of control power - fails to start	-	-	-	-	
1P117D	River water supply pump	17-B	1 of 2 in each loop operating	1B20	Fails to run	1D21	Loss of control power - fails to start	-	-	-	-	

Circulating Water System

The Circulating Water System receives the RHR Service Water and depending on which path is used, will either route the service water to the cooling towers, or to the river via the dilution structure.

Condensate Storage and Transfer System

The Condensate Storage and Transfer System is used to flush the RHR heat exchangers when they are not in use.

Process Radiation Monitoring

The service water discharge is monitored for radioactivity in both the radwaste dilution line path and the cooling tower discharge paths.

River Water Supply System

Makeup water to the RHR/ESW wet pits is supplied from the Cedar River by the River Water Supply System. The river water is delivered from the Intake Structure to the stilling basin in the Pump House from where it flows to the wet pits.

3.2.1.14.3 System Fault Tree Model Assumptions

This section describes any assumptions specific to the system fault tree.

Assumptions:

1. The RHRSW tree consists of three top events. The first two top

events represent RHRSW cooling functions as follows:

- Failure of adequate cooling to the Locp B RHR heat exchanger.

- Failure of adequate cooling to the Loop A RHR heat exchanger.

These top events are transferred into the RHR System fault tree. One pump in each loop is assumed adequate.

2. The third top event in the RHRSW System model represents crosstie use of the RHRSW System as an alternate injection source.

3. One of four river water supply pumps are necessary to maintain the water supply in the stilling basin. The RHRSW function fails if all water supply pumps fail. The separate basins in the pump house are treated as equal.

4. Common cause failure of both heat exchangers is included as a basic event.

5. ESW cooling for the RHRSW pumps is required for successful pump operation. No HVAC is required.

3.2.1.14.4 Success Criteria

The RHRSW System serves two purposes for the IPE. RHRSW provides cooling to the

RHR heat exchanger and provides alternate injection to the reactor vessel.

Cooling Function

As a support system, RHRSW provides cooling water to the RHR heat exchangers. The system functions to transfer heat from RHR System, when operating in the shutdown or torus cooling modes, to the plant ultimate heat sink. Successful operation of the RHRSW System for RHR heat exchanger cooling requires one RHRSW pump providing coolant to one RHRSW heat exchanger. Operation of the RHRSW System in this capacity requires AC power and DC power. ESW cooling for the RHRSW pump motors is required. No HVAC is required for successful RHRSW cooling.

Alternate Injection Function

The success criteria for RHRSW as a source of alternate injection requires operator action and one of four RHRSW pumps providing injection to the reactor vessel. Operation of the RHRSW system in the alternate injection mode requires realignment of three crosstie valves, AC power, and DC power. ESW cooling to the RHRSW pump motors is also required. No HVAC is required for successful RHRSW alternate injection.

For both modes of operation, the RHRSW pumps take suction from the ESW wet pits which are fed from the stilling basin. Makeup to the stilling basin from one of the four River Water pumps is required for RHRSW operation. This flow must be delivered to the wet pits within 30 minutes (assuming 4008 gpm demand) in order to prevent damage to the RHRSW pumps due to low NPSH.

3.2.1.15 River Water System

3.2.1.15.1 System Function

The purpose of the River Water System (RWS) is to provide makeup water from the Cedar River for the Circulating Water Systems, General Service Water (GSW), RHR Service Water (RHRSW), Emergency Service Water (ESW) and Radwaste Dilution System to replace that which is lost due to evaporation, blowdown and normal uses.

Water is continuously pumped by one vertical, wet pit, River Water pump from each of the redundant facilities. A simplified diagram of the River Water System is provided in Figure 3.2-21. (See also Figure 3.2-9 for general water pits layout.) River Water pumps 1P-117A/1P-117C and pumps 1P-117B/1P-117D take suction from the same facility. Water is pumped through an 18" discharge line from each pump to a 24" line which leads to the stilling basin of the pumphouse. Part of the water is diverted to the Radwaste Dilution System. The water supplied to the stilling basin serves the closed Circulating Water System, RHRSW, GSW, and ESW systems. Each pump discharge line is provided with a check valve and a butterfly valve in series.

The river water is pumped to the pumphouse via underground pipe to the stilling basin through either of two pneumatically operated control valves (CV4914 and CV4915). An Air Vacuum relief is provided on each header. Normally one of the Control Valves is in automatic control, with the other control valve shut and controlled by a separate Hand Controller. The two headers are cross-connected through two control valves (CV4910 A & B). Radwaste dilution taps off of this line between the control valves and its flow is regulated by another pneumatically operated control valve (CV4909). Therefore, in summary, a water supply pump is operating from each loop (each wet pit) to supply water to the stilling basin through one of the control valves (CV4914 or CV4915), the other control valve is normally shut.

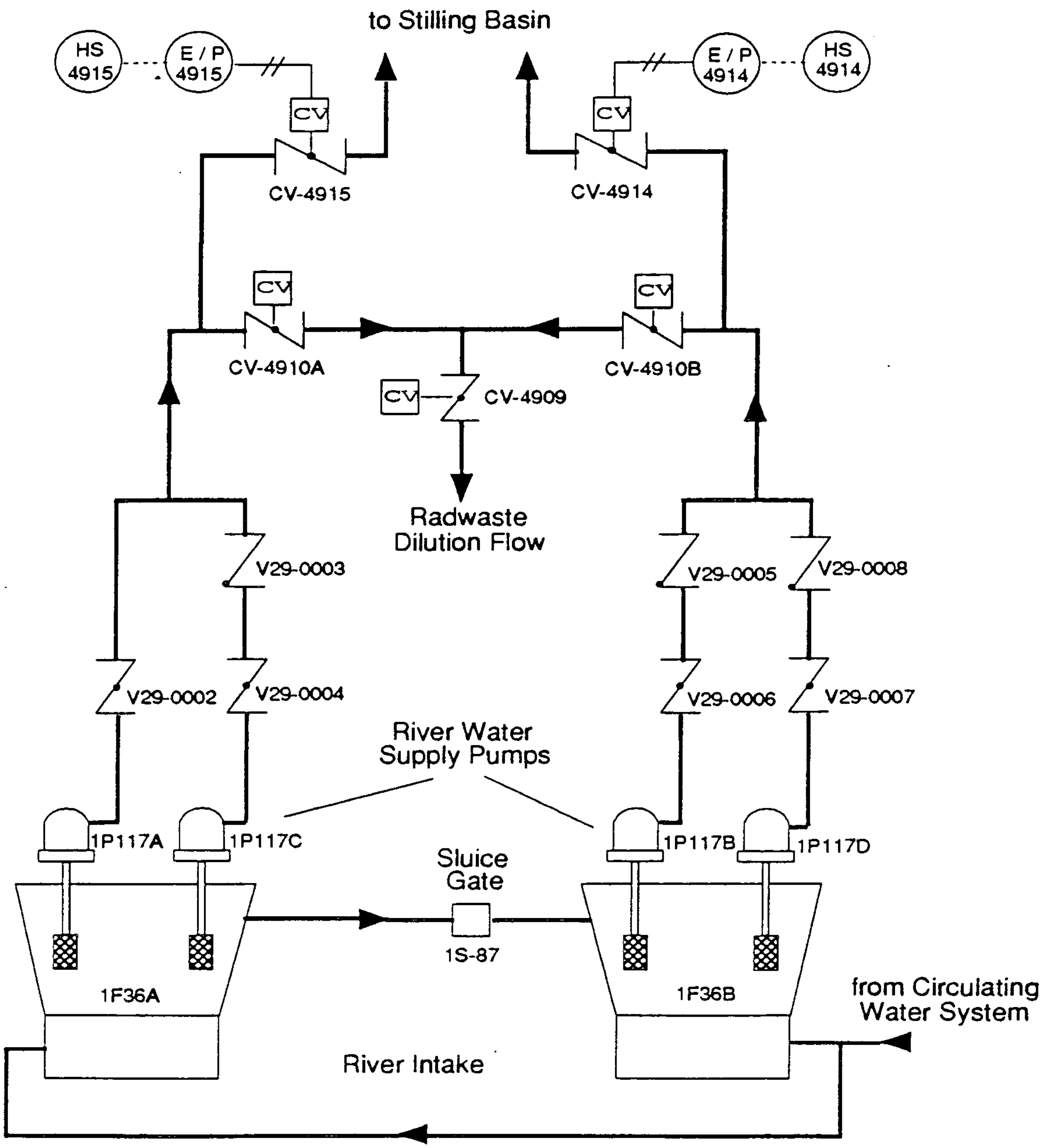


Figure 3.2-21
River Water Supply

The River Water pumps are controlled from Control Room Panel 1C06. The breakers can be operated locally in the intake structure. The Auto-Restart feature is designed to automatically start a River Water pump after a loss of offsite power (UV on 1A3 bus is actual sensing point). The selected pump will automatically restart on a loss of offsite power, after power is regained from respective emergency diesel, unless the selected pump was previously running in which case 2 minute time delay would have to be satisfied.

The only interlock associated with the River Water pumps is that one of the pumps in each loop must be operating before the respective ventilation supply fan can be started. The Supply fan will automatically start if the Fan Control Switch in the intake structure is selected for AUTO when one of the Water Supply pumps is running.

Control Valve 4914 or 4915 is throttled to regulate the proper flow into the stilling basin. They are controlled by one of two Cooling Tower Basin Level Controllers. A control valve selector switch determines which valve will be controlled by the Level Controller. The cooling tower level controller will signal for more makeup water (throttle supply valve open) when the basin level is down to 3 feet, and will continue to signal for makeup until the cooling tower basin has risen 6 inches.

3.2.1.15.2 System Interfaces and Dependencies

The River Water System fault tree model includes support systems required for the River Water System to function in postulated accident scenarios. The systems which support specific River Water System components and their effects on system operation are identified in the Dependency Matrix shown in Table 3.2-15. No additional systems provide support functions.

Table 3.2-15

RIVER WATER SYSTEM DEPENDENCY MATRIX

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF IA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
1P117A	River Water Pump A	Yard	Running	1B9	Pump fails to run	1D11	None. Pump assumed to be operating	-	-	-	-	-
1P117B	River Water Pump B	Yard	Running	1B20	Pump fails to run	1D21	Loss of Control Power	-	-	-	-	-
1P117C	River Water Pump C	Yard	Running	1B9	Pump fails to run	1D11	Loss of Control Power	-	-	-	-	-
1P117D	River Water Pump D	Yard	Running	1B20	Pump fails to run	1D21	Loss of Control Power	-	-	-	-	-
CV4909	Radwaste Dilution Isolation Valve	-	-	-	-	-	-	ISA	None. Valve fails closed.	-	-	Isolation valve closes when SV-4934 and SV-4935 deenergize.

Table 3.2-15
RIVER WATER SUPPLY SYSTEM DEPENDENCY MATRIX
(continued)

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF IA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
CV4915	River Water Control Valve	-	-	-	-	-	-	ISA	None. Valve assumed to be open and fails as is.	-	-	Control valve opens when SV-4934 and SV-4935 deenergize.
CV4914	River Water Control Valve	-	-	-	-	-	-	ISA	Loss of motive power.	-	-	Control valve opens when SV-4934 and SV-4935 deenergize.
SV4934	River water solenoid control valve	-	-	-	-	1D11	Opens CV4914 and CV4915	-	-	-	-	Solenoid valve deenergizes on Hi Drywell Pressure, Lo-Lo-Lo Reactor Vessel level, ESW Wet Pit Low Level and Loss of Power.

Table 3.2-15
RIVER WATER SUPPLY SYSTEM DEPENDENCY MATRIX
(continued)

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF IA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
SV4935	River water solenoid control valve	-	-	-	-	1D21	Opens CV4914 and CV4915	-	-	-	-	Solenoid valve deenergizes on Hi Drywell Pressure, Lo-Lo-Lo Reactor Vessel level, ESW Wet Pit Low Level and Loss of Power.
CV4910A	River water cross-tie control valve	-	-	-	-	-	-	ISA	Loss of motive power	-	-	-
CV4910B	River water cross-tie control valve	-	-	-	-	-	-	ISA	Loss of motive power	-	-	-

3.2.1.15.3 System Fault Tree Model Assumptions

The following assumptions were used in the development of the River Water System Fault tree model.

1. The River Water System will fail to makeup water to the stilling basin if the river freezes and deicing capability is not available.

2. It is assumed River Water Pump 1P-117A is operating, 1P-117B, 1P-117C, and 1P-117D are in standby, control valve CV4914 is closed, control valve CV4915 is open, and cross connect valves CV4910A, B are closed.

3.2.1.15.4 Success Criteria

As modeled for the IPE, successful operation of the River Water System requires one of four River Water pumps operating to provide makeup to the stilling basin. Operation of the River Water System also requires no diversion to the Radwaste Dilution Structure and either river water temperature greater than 32°F or deicing capability. A flow path from the operating River Water pump through CV4915 or CV4914 must also be available. If valve realignment is necessary, instrument air is required.

Operation of the River Water pump requires automatic initiation or operator action for the pumps not assumed to be operating. AC power is required for successful operation of all the pumps. For the pumps assumed to be in standby mode, DC control power is required for starting.

3.2.1.16 Standby Liquid Control System

3.2.1.16.1 System Function

The Standby Liquid Control (SLC) System is operated to provide an alternate means of shutting down the reactor, independent of the Control Rod Drive System. This is accomplished by pumping a neutron-absorbing solution (sodium pentaborate) into the reactor vessel. The design basis is to overcome the maximum positive reactivity resulting from cooldown and xenon decay.

Major system components, which are designed and configured to ensure maximum system reliability in meeting the design objective, are as follows:

- Standby Liquid Control Storage Tank: This is a 3270 gallon cylindrical tank provided with electric heaters and an air sparger system to maintain the boron concentration within Technical Specification limits.

- Standby Liquid Control Injection Pumps: Transfer of the boron solution from the storage tank to the reactor vessel is accomplished by two parallel 100% positive displacement pumps, each rated at 28 gpm.

- Explosive valves: Pump discharge is routed through two parallel injection valves. These valves are normally closed explosive actuated valves designed to assure positive opening when system operation is required. Each valve is capable of passing full system flow.

The following design bases are incorporated into the SLC System:

a. SLC is capable of providing independent normal reactivity control.

b. SLC is sufficiently capable of controlling the reactivity difference between the steady-state rated operating condition and the cold shutdown condition.

c. The time required for SLC to perform its function is consistent with the nuclear reactivity rate of change predicted between rated operating condition and cold shutdown conditions.

d. The SLC System is capable of being tested during operation.

e. The neutron absorber is dispersed within the reactor core in sufficient quantity to provide a reasonable margin for leakage and imperfect mixing.

The SLC System was modified in 1987 to meet the requirements of the NRC rule on Anticipated Transients Without Scram (ATWS), given in 10CFR50.62 and NRC Generic letter 85-03.

The SLC System for ATWS concerns is designed to provide a minimum capacity and boron content equivalent in control capacity to 86 gpm of 13 weight percent natural boron solution.

This was accomplished by doing the following:

1. Modify the SLC control logic so that both pumps are started

whenever the SLC System is manually initiated.

2. Increase the required minimum boron concentration.

When SLC is required to shutdown the reactor, a keylocked switch (HS-2613) on Control Room Panel 1C05 is used to start the pumps. This action causes the following to occur:

1. The SLC pumps start.
2. Both explosive injection valves are fired causing them to open.
3. The Reactor Water Cleanup System isolates automatically.

A simplified drawing of the system is shown in Figure 3.2-22. The absorber solution flows from the storage tank through locked open valve V-26-1 to the combined pump suction lines and into the operating pumps through locked open suction valves V-26-2 or V-26-3. The pumps discharge through the check valve and locked open discharge valves V-26-5 or V-26-7 into the common discharge header that splits into two parallel lines isolated by explosive valves. This parallel discharge path ensures flow in the unlikely event of an explosive valve not opening when initiated. The absorber solution passes through the explosive valves, through the check valves, V-26-8 and V-26-9 and finally through the locked open isolation valve into the reactor vessel lower plenum via the SLC sparger.

System testing may be accomplished by isolating the solution storage tank and initiating system flow using the demineralized water in the SLC test tank as a source.

The system does not function during normal plant operations or during plant transients.The system is manually initiated only as directed by the ATWS Emergency Operating Procedure (EOP) to shutdown the nuclear reaction.

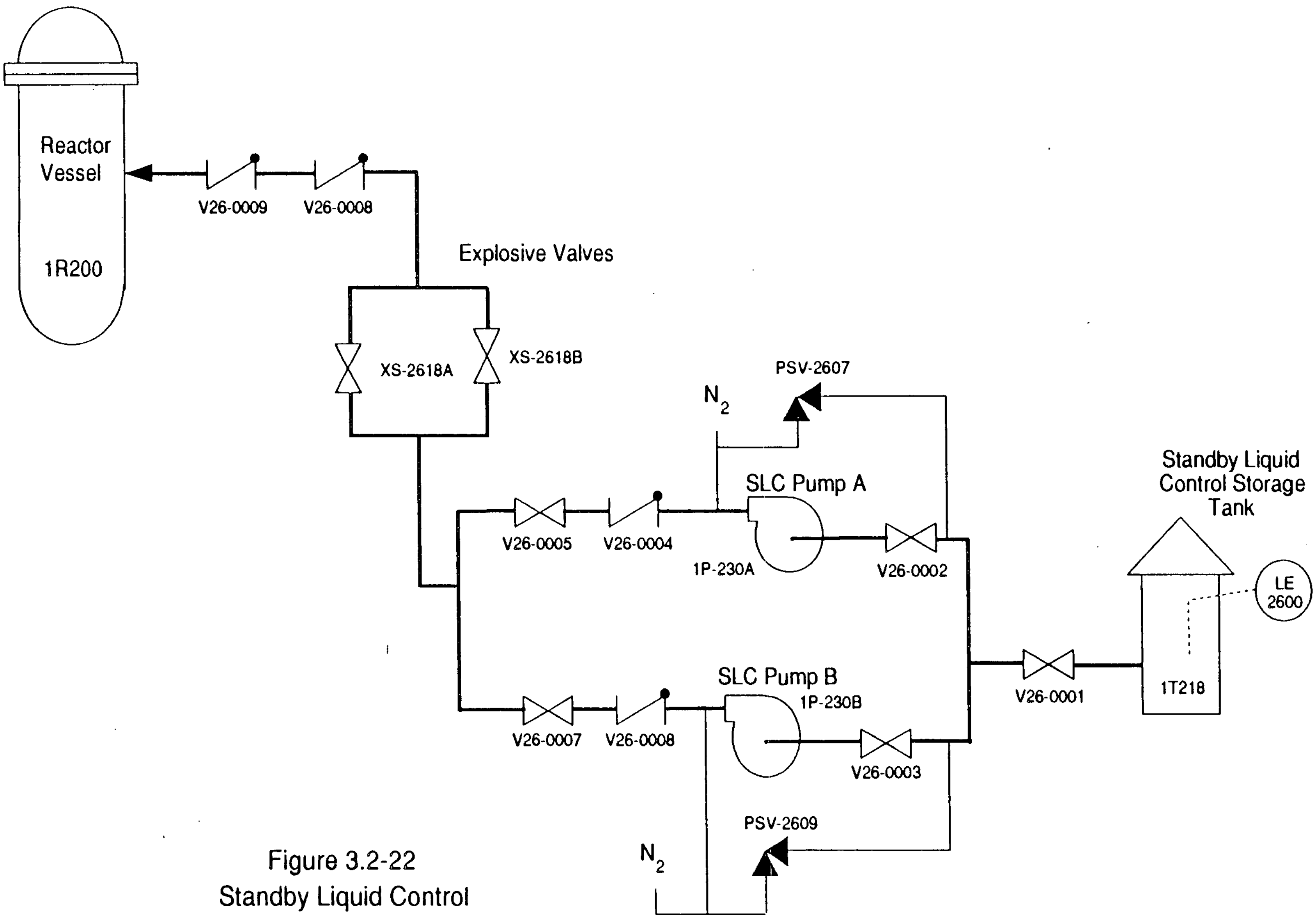


Figure 3.2-22
Standby Liquid Control

NOTE: The system may also be operated, as directed by the EOPs, as an alternate source of reactor vessel makeup water by injecting demineralized water from the SLC Test Tank. This function is not covered by this PRA.

3.2.1.16.2 System Interfaces and Dependencies

The SLC fault tree model includes support systems required for SLC to function in postulated accident scenarios. The systems which support specific SLC components and their effects on SLC operation are identified in the Dependency Matrix shown in Table 3.2-16.

In addition to the systems identified on a component level in the dependency matrix, the following systems provide support functions to the SLC System:

Reactor Vessel and Internals

The SLC System penetrates the reactor vessel through a nozzle in the lower portion of the vessel. The line rises inside the vessel and the solution is dispersed through a sparger in the lower core plenum just below the core support plate. The nozzle used for vessel penetration by this system is also used as an instrumentation tap.

Reactor Vessel Instrumentation

The Standby Liquid Control System injection line serves as an instrument tap for:

1. Core plate differential pressure.
2. CRD high pressure side for drive water pressure control and cooling water flow differential pressure.
3. Jet Pump high pressure side for differential pressure.

Table 3.2-16

STANDBY LIQUID CONTROL SYSTEM DEPENDENCY MATRIX

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
1P-230A	SLC PUMP A	5-A	OFF	1B34	PUMP FAILS TO START/RUN	--	--	--	--	--	--	--
1P-230B	SLC PUMP B	5-A	OFF	1B44	PUMP FAILS TO START/RUN	--	--	--	--	--	--	--
XS-2618A	EXPLOSIVE VALVE A	5-A	CLOSED	1B34	VALVE FAILS TO OPEN	--	--	--	--	--	--	--
XS-2618B	EXPLOSIVE VALVE B	5-A	CLOSED	1B44	VALVE FAILS TO OPEN	--	--	--	--	--	--	--

4. Core spray leak detection.

5. Total Jet Pump Differential pressure high side.

Condensate and Demineralized Water System

The demineralized water header provides a source of water for solution storage tank filling, makeup, and chemical mixing; performance test of SLC using the test tank, and system flushing.

Reactor Water Cleanup System

The Reactor Water Cleanup System is automatically isolated when the keylocked switch on Control Room Panel 1C05 is used to start the SLC pumps. In the event that boron injection is required but cannot be injected using the SLC system, boron is injected into the RPV using the Reactor Water Cleanup System.

3.2.1.16.3 System Fault Tree Model Assumptions

This section describes any assumptions specific to the system fault tree.

The following assumptions were used in the development of the SLC fault tree model.

1. Tank heater operation is not required for successful operation of the SLC System.

2. Heat tracing is not required for successful operation of the SLC System.

3.2.1.16.4 Success Criteria

The SLC System is an emergency boration system that is used to inject negative reactivity during ATWS events. Both pumps will try to inject when SLC is manually initiated. Successful operation of SLC requires either one pump or two pumps depending on the accident:

1. One SLC pump is adequate:
   - turbine trip with the main condenser available and RPV level controlled

2. Two SLC pumps required:
   - Early SLC initiation for accidents in which the main condenser and RPV level control are failed
   - late SLC initiation for accidents in which either the main condenser or RPV level control is successful, but not both.

In both cases, an alternate injection path through the Reactor Water Cleanup System can be utilized, but is not credited in the IPE.

For purposes of the IPE, the following components or functions of the SLC System are not required for successful operation:

1. Tank heater operation.

2. Heat tracing.

3.2.1.17 Torus/Torus Vent System

3.2.1.17.1 System Function

Pressure Suppression Chamber

The pressure suppression chamber is a steel pressure vessel in the shape of a torus located below and encircling the drywell. The pressure suppression chamber contains the suppression pool and the gas space above the pool. The pressure suppression pool serves as a heat sink for postulated transient or accident conditions. Energy is transferred to the pool by either the discharge piping from the reactor pressure safety/relief valves or the drywell vent piping, which discharge below the water level. The pool condenses the steam portion of the flow and collects any water carryover, while non-condensible gases (including any gaseous fission products) are released to the suppression chamber gas space. Energy is removed from the suppression pool when the RHR System is operating in the torus cooling mode.

The suppression pool is also the primary source of water for the Core Spray System and the LPCI mode of the RHR System and the secondary source of water for the RCIC and HPCI Systems. The quantity of water stored in the suppression pool is sufficient to condense the steam from a design basis accident and to provide adequate water for the ECCS. The suppression chamber is subject to the pressure associated with the storage of a minimum of 58,900 cubic feet of water distributed uniformly within the vessel during normal operation.

Vent Piping

Eight 4' 9" diameter vent pipes connect the drywell and the pressure suppression chamber. The drywell vents are connected to a 3' 6" diameter vent header in the form of a torus which is contained within the air space of the suppression chamber. Projecting downward from the header are 48 downcomer pipes, which terminate approximately 3 feet below the water surface of the pool and 7 feet above the bottom of the torus.

Vacuum Relief System

The primary containment is designed for an external pressure not more than 2 psi greater than the concurrent internal pressure. Redundant automatic vacuum relief devices are used to present any unacceptable pressure differential.

There are two groups of vacuum breakers: the torus-to-drywell group, which prevents drywell pressure from being significantly less than torus pressure; and the Reactor Building-to-torus group, which prevent the torus pressure from being significantly lower than the Reactor Building pressure. Only the torus-to-drywell group will be discussed in this system analysis.

The torus-to-drywell group consists of seven vacuum breaker control valves, CV-4327A, B, C, D, F, G, & H, which are located on the vent header within the air space of the suppression chamber. The capacity is adequate to limit pressure differential between the suppression chamber and drywell, during post accident drywell cooling operations, to a value which is within the Suppression System design value of 2 psid in this direction.

Hardened Wetwell Vent

DAEC is in the process of installing a hardened wetwell vent. This hardened wetwell vent will connect the existing piping in the reactor building for the wetwell purge exhaust to existing piping in the turbine building for the discharge from the steam packing exhauster (see Figure 3.2-23). The discharge piping for the steam packing exhauster leads to the plant offgas stack, providing for an elevated release point.

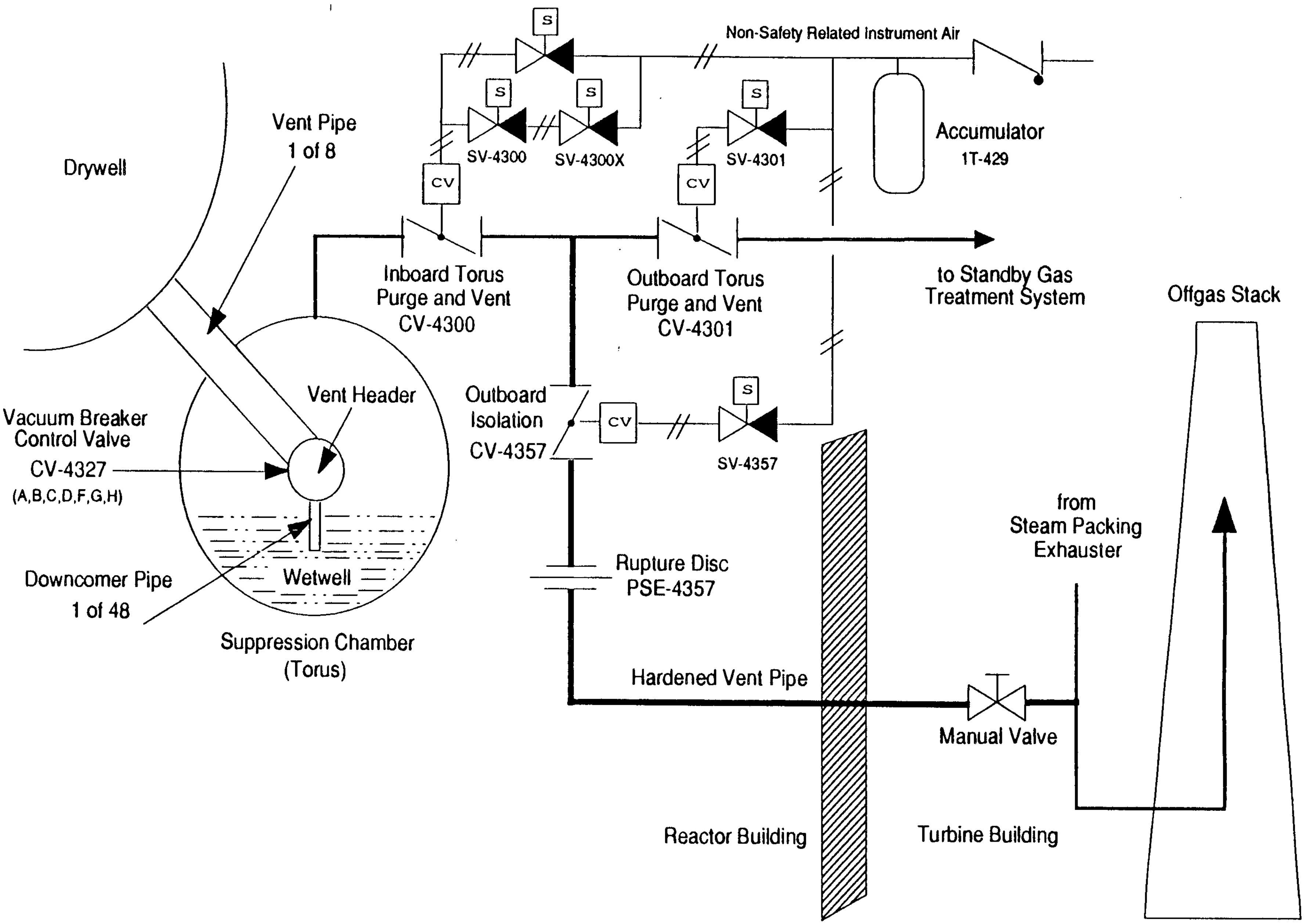


Figure 3.2-23

Torus Vapor Suppression & Vent System

The new piping connects to the wetwell purge exhaust piping between the inboard and outboard torus purge and vent isolation valves. An additional air-operated outboard isolation valve and a rupture disk is installed in the new piping. The design will allow the vent to operate during a station blackout event, i.e. it will be AC independent commensurate with the duration specified by DAEC's compliance with the station blackout rule. The new isolation valve is fail-safe (closed) on loss of air or DC power, but will have a compressed air accumulator to support valve operation following a loss of supply air. Controls for the existing inboard isolation valve are modified to allow it to be operated independent of AC power. Valve control and position indication will be provided in the control room.

The suppression pool is also the primary source of water for the CS System and the Low Pressure Coolant Injection (LPCI) mode of the RHR System and the secondary source of water for the RCIC and HPCI Systems. The quantity of water stored in the suppression pool is sufficient to condense the steam from a design basis accident and to provide adequate water for the ECCS. The hardened pipe vent is not required to function during transient conditions.

The hardened wetwell vent that is being installed provides added assurance that a pathway will be available for venting the primary containment. The installation of the hardened vent was in response to a potential challenge to containment integrity. If reactor vessel isolation occurs and normal suppression pool cooling is lost, transfer of core decay heat to the main condenser must be reestablished in order to maintain containment pressure within acceptable limits. If decay heat removal fails, the containment pressure is expected to be in excess of the nominal 2 psig rating of the "normal" venting pathways. Since failure of the normal pathway due to over-pressurization may occur at undesirable locations within the plant, use of a hardened vent pathway that is capable of withstanding the expected pressures is preferable.

3.2.1.17.2 System Interfaces and Dependencies

The Torus/Torus Vent System fault tree model includes support systems required for Torus/Torus Vent System to function in postulated accident scenarios. The systems which support specific Torus/Torus Vent System components and their effects on Torus/Torus Vent System operation are identified in the Dependency Matrix shown in Table 3.2-17.

In addition to the systems identified on a component level in the dependency matrix, the following systems provide support functions to the Torus/Torus Vent System:

Residual Heat Removal System

The RHR System provides spray cooling for the Primary Containment to limit containment temperature and pressure by condensing steam released within the Primary Containment in event of a LOCA. The RHR System also cools the suppression pool water.

DC Power System

The DC Power System provides control power to pilot valves associated with the torus vent lines.

Instrument Air System

The Instrument Air System provides the compressed air supply to open the vent valves if air is not available from the accumulator.

Table 3.2-17

TORUS VAPOR SUPPRESSION & VENT SYSTEM DEPENDENCY MATRIX

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF IA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
TORUS	TORUS	-	-	-	-	-	-	-	-	RHRSW	-	-
CV-4300	Inboard Torus Purge and Vent Valve	-	Closed	-	-	1D11	Fails Closed	ISA	Accumulator backup supply available	-	-	-
CV-4301	Discharge to SGTS	-	Closed	-	-	1D11	Fails Closed	ISA	Accumulator backup supply available	-	-	-
CV-4357	Discharge through hardened vent	-	Closed	-	-	1D21	Fails Closed	ISA	Accumulator backup supply available	-	-	-

3.2.1.17.3 System Fault Tree Model Assumptions

The following assumptions were used in the development of the Torus/Hardened Wetwell Vent fault tree model:

1. Failure of any individual suppression pool vent pipe or downcomer is assumed to result in failure of the vapor suppression function.

2. Failure of the rupture disk on the hardened wetwell vent line is not modelled.

3.2.1.17.4 Success Criteria

Success of torus for pressure suppression requires maintaining an adequate inventory of water in the torus at sufficiently low temperatures. This requires torus pressure integrity and the RHR System to remove heat from the suppression pool. The water level in the torus must be above the bottom of the downcomers and below the height of the vacuum breakers. Ruptures of any of the 8 vent lines or 48 downcomers, or failure (open) of the vacuum breakers would allow steam to bypass the suppression pool and is considered failure of the pressure suppression function.

Failure of any individual vacuum breaker during small, medium, and/or large LOCAs and IORV/SORV events results in pressure suppression failure.

Success of containment venting via the hardened wetwell vent requires operator action to remotely open the inboard and outboard vent valves. Operation of the valves requires DC control power and an air supply from either the accumulator or the non-safety related Instrument Air System.

3.2.1.18 Well Water System

3.2.1.18.1 System Function

The purpose of the Well Water System is to provide water as required to the following plant demands:

- Cooling water to the various plant ventilation cooling units.
- Supply water to the Makeup Demineralizer System.
- Supply water to the plant Potable Water System.
- Continuous water to the Fire Protection System jockey pump.

The Well Water System also acts as a standby source of water to the Fire Protection System.

Well Water System is designed to meet the following design bases:

1. To remove heat from cooling units and discharge the water into the Circulating Water System as part of makeup for that system during all plant modes of operation.

2. The wells are physically separated from one another by at least 720' to equalize well draw down in the event of more than one pump is used at the same time. The wells are also located at least 1100' from the plant to ensure that any undesirable water sources that enter the ground should not enter the well water.

3. The wells are dug at a minimum depth of 120' and sealed to prevent the collection of less desirable ground water from the more shallow

water sources.

4. A backflow preventer is provided to ensure that contaminated water cannot flow into the wells or into the potable water supply.

The Well Water System, as shown in Figure 3.2-24 (also Figure 3.2-9), consists of four independent wells and supply headers combining into a common supply header at the plant buildings. Each well and pump is protected from the weather by its own building, and space heaters are used during cold weather. Transient/surge absorbers (metal oxide varistors) are installed at each of the flow transmitters (FT-4414 A-D) in the well houses and on the flow controllers (FC-4414 A-D) located in Panel 1C23 in the Control Room. The individual supply headers from each well are also heat traced to prevent freezing. Three of the pumps (1P-58A, B and C) have a 750 gpm pump capacity, while the other pump (1P-58D) has a 1650 gpm pump capacity. To maintain the 1200-1500 gpm flow requirements for the system, 1P-58D is normally run while a combination of two of the remaining three pumps are used as a backup source.

A backflow preventer is provided to ensure that contaminated water cannot flow back into the wells or potable water system.

The system also includes a chemical injection system downstream of the backflow preventer to add chemicals to mitigate the formation of calcium carbonate buildup on piping and related components.

The Well Water System consists of the following major components:

1. Well Pumps 1P-58A, 1P-58B, 1P-58C, 1P-58D
2. Beeco Backflow Preventer 1S-84, 1S-86

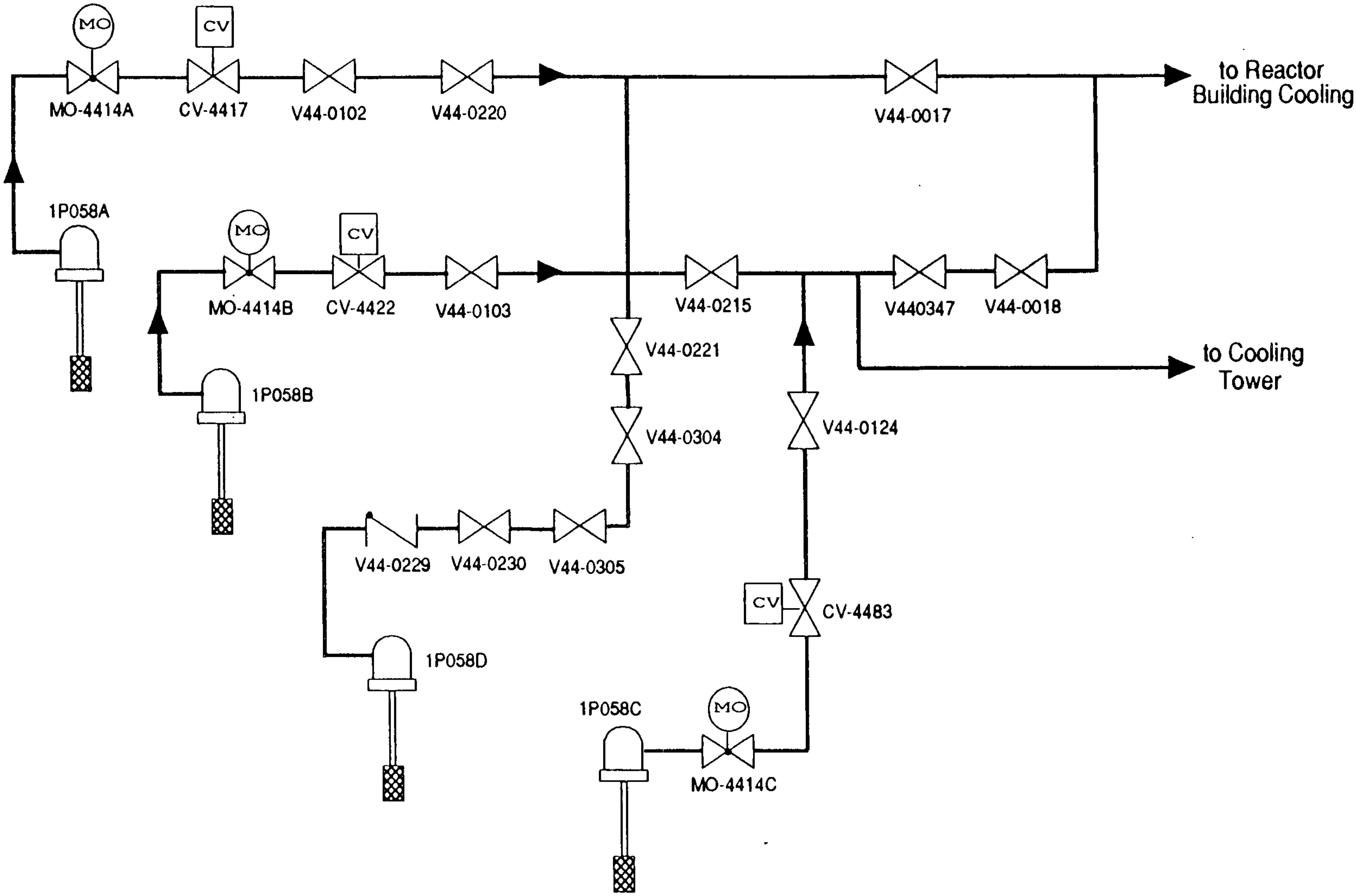


Figure 3.2-24
Well Water Supply

During normal operation, the Well Water System supplies the following equipment:

- Plant Ventilation System
- Potable and Sanitary Water System
- Makeup demineralizer
- Radwaste and machine shop rotoclones
- Offgas recombiner
- Offgas glycol refrigeration unit
- Containment N_2 compressor
- Fire Protection System jockey pump

The Well Water System acts as a standby source of water to the Fire Protection System.

3.2.1.18.2 System Interfaces and Dependencies

The Well Water System fault tree model includes support systems required for the Well Water System to function in postulated accident scenarios. The systems which support specific Well Water System components and their effects on system operation are identified in the Dependency Matrix shown in Table 3.2-18. No additional systems provide support functions.

3.2.1.18.3 System Fault Tree Model Assumptions

The well water system is normally running.

3.2.1.18.4 Success Criteria

The purpose of this system is to supply cooling water for plant ventilation units, potable water for drinking and sanitary uses, and water as required for the makeup

Table 3.2-18

WELL WATER SYSTEM DEPENDENCY MATRIX

COMPONENT	DESCRIPTION	FIRE ZONE	NORMAL POS.	AC BUS	LOSS OF AC EFFECT	DC BUS	LOSS OF DC EFFECT	INST. AIR	LOSS OF ISA EFFECT	COMP COOLING	HVAC	AUTO ACTUATION SIGNAL
1P-058A	Well Water Pump A	Yard	Running	1B33	Pump fails to run	-	-	-	-	-	-	-
1P-058B	Well Water Pump B	Yard	Running	1B45	Pump fails to run	-	-	-	-	-	-	-
1P-058C	Well Water Pump C	Yard	Running	1B33	Pump fails to run	-	-	-	-	-	-	-
1P-058D	Well Water Pump D	Yard	Running	1A2	Pump fails to run	1D21	Loss of Control Power	-	-	-	-	-

demineralizers. Although supplying water to these systems does not constitute a safety-significant action, maintaining a minimum flow to them is required for continued plant operation.

In order to maintain the 1200-1500 GPM flow requirements and prevent a forced plant shutdown, the four wells/pumps should be used in a combination which will meet plant water demands. The normal flow capacity for well nos. 1, 2, and 3 is 750 gpm and well no. 4 is 1650 gpm. The four wells are used in varying combinations to meet plant water demands.

3.2.2 Top Logic Description

When developing functional event trees, it is necessary to describe the functions in terms of system success (or failure). This section provides a mapping of the functions used as headings in the DAEC Level I Event Trees and the DAEC Level I System Fault Trees. Figure 3.2-25 presents a graphical description of the top logic.

3.2.2.1 Reactivity Control

Most of the event trees use a point estimate for the reactor SCRAM function. The exceptions are the ATWS trees. These provide models for both automatic and manual SCRAMs and the Standby Liquid Control (SLC) injection.

C(SCRAM)

The SCRAM function is modeled in the fault tree linking process as the combination of mechanical, electrical, and Alternate Rod Insertion (ARI) failures. Included in this is the automatic recirculation pump trip. The description of this function is described in detail in Section 3.2.1.13 Turbine Trip with Bypass ATWS.

C(SLC)

The SLC function is modeled using either the one pump or two pump fault trees. The one pump models are used in early SLC nodes with the condenser available, and in late SLC nodes. The two pump model is included in nodes where SLC injection is required early and the main condenser is not available. Each of these modeled is "OR-ed" with the appropriate operator action for the time phase in which the node is required. The late injection operator action is split into the early injection failure "AND-ed" with the conditional probability that late failure occurs given early failure. This allows the computer models to properly account for the dependencies between these events.

3.2.2.2 Primary Pressure Control

M(SRVSOPEN)

A point estimate is used for the common cause failure of all SRVs and SVs failing to open to control the initial pressure spike of a transient. There is no fault tree mapping for this function.

P(SRVSCLOSE)

This function is modeled as the combination of the failure probability of 1 of 2 SRVs to reseat at high pressure, combined with the probability that it fails to reseat after RPV pressure is reduced to below 200 psid.

Q(FW:RUNBACK)

In ATWS cases, it is desirable for feedwater to "run back" to a lower flow rate in order to reduce the insertion of positive reactivity. A point estimate is used for this node.

3.2.2.3 Containment Pressure Control

D(VAPOR:SUPP)

In most scenarios, there are two systems capable of providing containment pressure control immediately following an accident. These are the pressure suppression function of the torus and containment sprays. This function is modeled by the "AND" of the fault trees for these systems.

3.2.2.4 Reactor Coolant Inventory

Q(MC:AVAIL)

This function does not model the success of the inventory function directly. It is used to set the stage for the possible success of feedwater as an injection source, and the possible success of the main condenser for containment heat removal. It is modeled using the "Main Condenser Initially Available" gate in the Condensate System fault tree.

Q(FW:CND)

The normal makeup source for the reactor is the Feedwater system. The Condensate system is required to be running for success. This node is modeled by a direct link of the Feedwater system fault tree. In cases where the Feedwater system needs to be restarted, e.g. after a high water level trip, the fault tree logic is changed accordingly.

U(H:R)

The High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC) provide a similar function of AC power independent coolant makeup. The main difference between these systems is capacity. This node is modeled by the logical "AND" of these fault trees. In cases where RCIC is not capable of providing sufficient flow, the RCIC gate is set to TRUE.

In the Loss of Offsite Power and ATWS trees, there are several event tree nodes that explicitly model the operators taking specific actions to maintain the HPCI and RCIC systems available under adverse Reactor Building and support system conditions. These are modeled as point estimates.

X(TIME:RX:DEP)

In the event that the high pressure coolant systems are unavailable and the initiating event does not provide depressurization, the operators must take manual action to depressurize the reactor via the SRVs, the main condenser, or HPCI/RCIC in CST-to-CST mode.

This function is questioned only after HPCI and RCIC have failed, so it is unlikely that the CST-to-CST mode can be successful. Reactor level will likely be low if all high pressure systems have failed, so it is likely that the MSIVs will have isolated, therefore successful depressurization to the condenser is not likely. Per procedure, the operators will defeat the automatic initiation of ADS as soon as the ADS timer starts, so the automatic initiation of ADS is not considered likely. None of these modes of depressurization are considered for this function.

This event tree node is modeled by the direct transfer from the manual depressurization fault tree.

V(CS:LPCI)

This node models the modes of low pressure injection that take a suction from sources internal to the containment. These are the Core Spray and LPCI systems. This node is modeled as the "AND" of the "One Core Spray Pump Fails to Inject" version of the Core Spray system fault tree and the "One RHR Pump - Zero RHRSW Pumps for Cooling - Fails to Inject" version of the RHR system fault tree.

Versions of these trees for modeling Large LOCAs are handled within the system fault tree logic.

V(C:R:G:E)

This node models the modes of low pressure injection that take a suction from sources outside the containment. These are the Condensate system, which injects from the hotwell, and the RHRSW, ESW, and GSW systems, which take suction from the river. This node is modeled as the logical "AND" of these systems' failure to inject versions. The Condensate tree used requires that a Condensate pump be restarted.

3.2.2.5 Level Control

These nodes are explicitly modeled in the ATWS event trees. They represent the operators taking control of the reactivity situation by controlling reactor water level. Each of these nodes is represented by a point estimate.

3.2.2.6 Containment Heat Removal

Z(MC:RECOV)

This node models the ability to use the main condenser as the ultimate heat sink. It is contingent upon the condenser being available early in the event scenario. This node is modeled by a direct transfer to the Main Condenser system fault tree.

W(TCOOL)

This node models the torus cooling mode of the RHR system. One division of RHR is necessary for success of this function. In all but the ATWS cases, only one RHRSW pump is required to provide river water flow to the heat exchangers. In the ATWS cases, two RHRSW pumps are required.

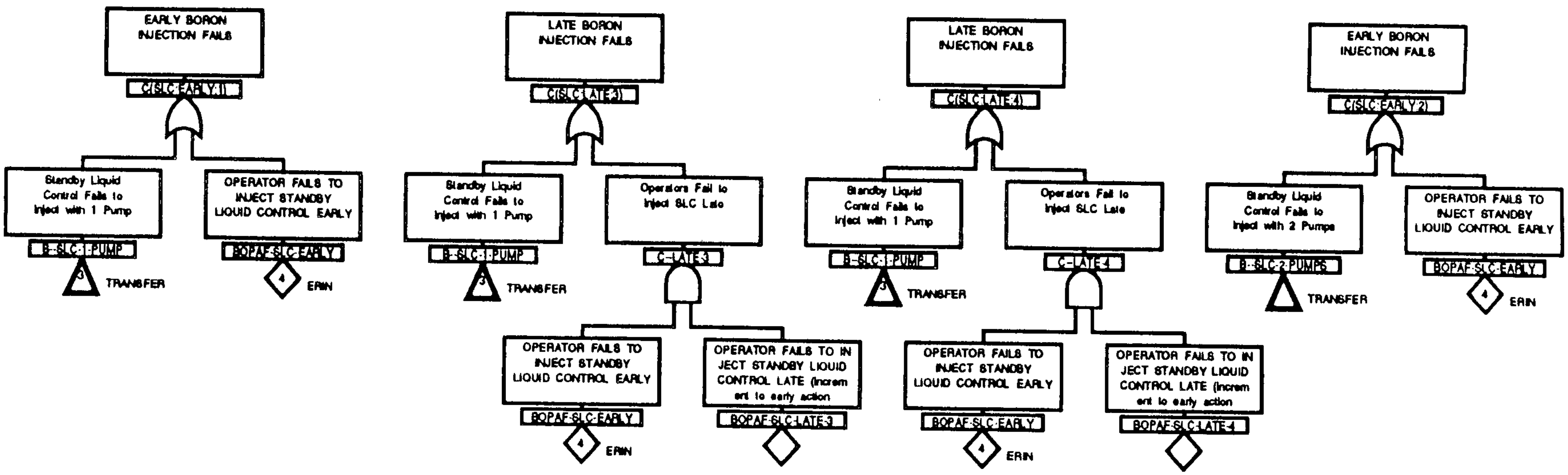
W(VENT)

This node models the venting of containment via either the Torus vent to the Standby Gas Treatment System or the Hard Piped Vent. The drywell venting modes are not considered. The Vent system fault tree models both of the venting methods considered.

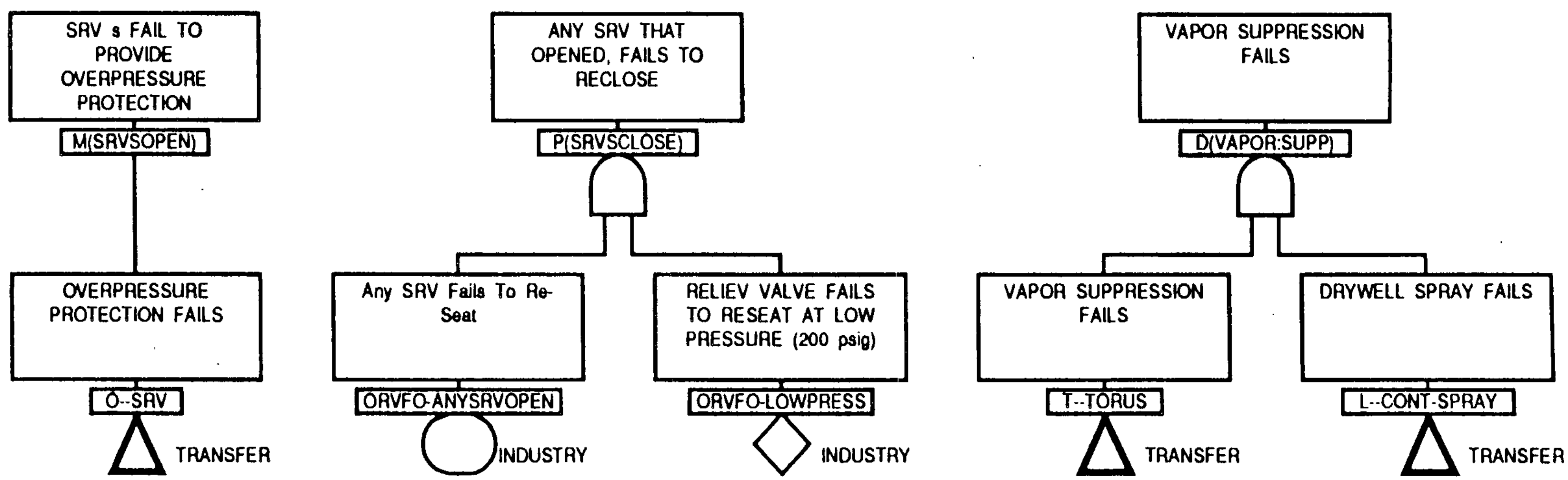
3.2.2.7 Injection Post Containment Challenge

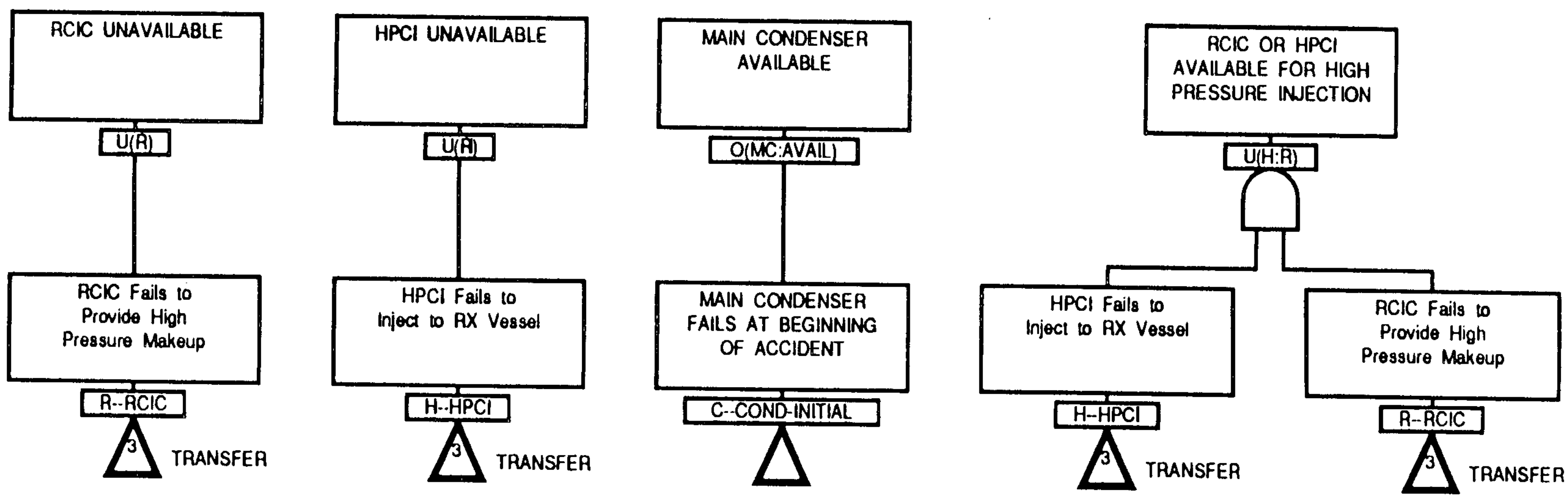
QUV(PST:CNT:CHL)

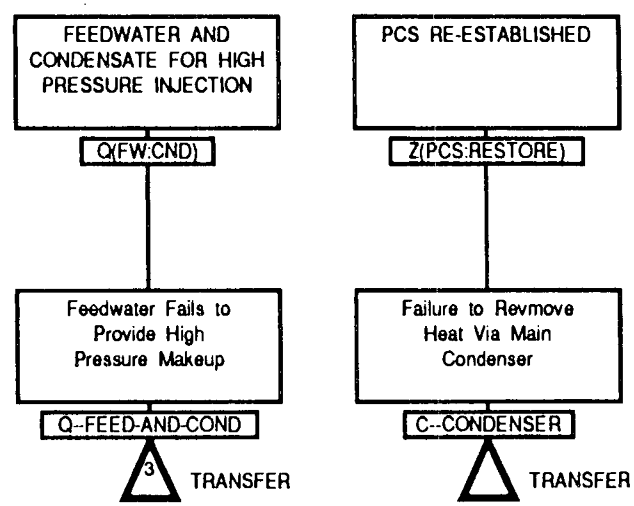
This node is modeled by a point estimate. A complete description of this node is provided in Section 3.1.2.0 Event Sequence Analysis - General Methodology.

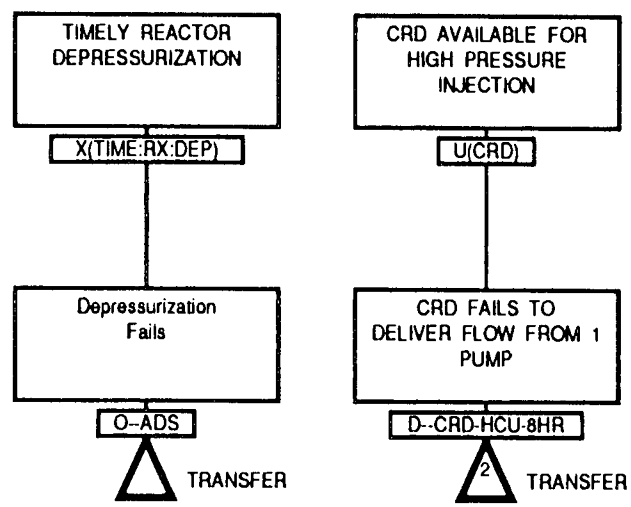


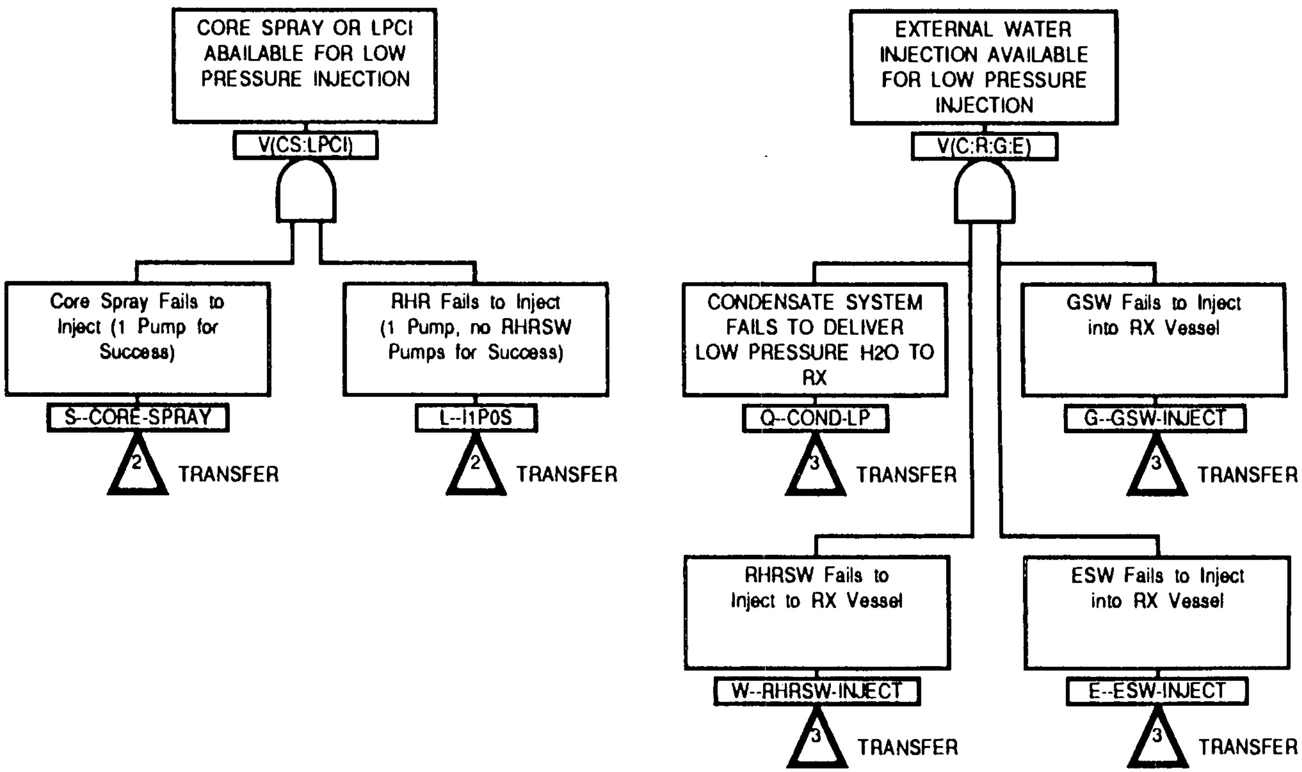
Top Logic for DAEC Event Trees	FIGURE 3.2-25	11-03-92	Page 313


SRV s FAIL TO PROVIDE OVERPRESSURE PROTECTION
M(SRVSOPEN)
OVERPRESSURE PROTECTION FAILS
O--SRV
TRANSFER
ANY SRV THAT OPENED, FAILS TO RECLOSE
P(SRVSCLOSE)
Any SRV Fails To Re-Seat
ORVFO-ANYSRVOPEN
INDUSTRY
RELIEV VALVE FAILS TO RESEAT AT LOW PRESSURE (200 psig)
ORVFO-LOWPRESS
INDUSTRY
VAPOR SUPPRESSION FAILS
D(VAPOR:SUPP)
VAPOR SUPPRESSION FAILS
T--TORUS
TRANSFER
DRYWELL SPRAY FAILS
L--CONT-SPRAY
TRANSFER

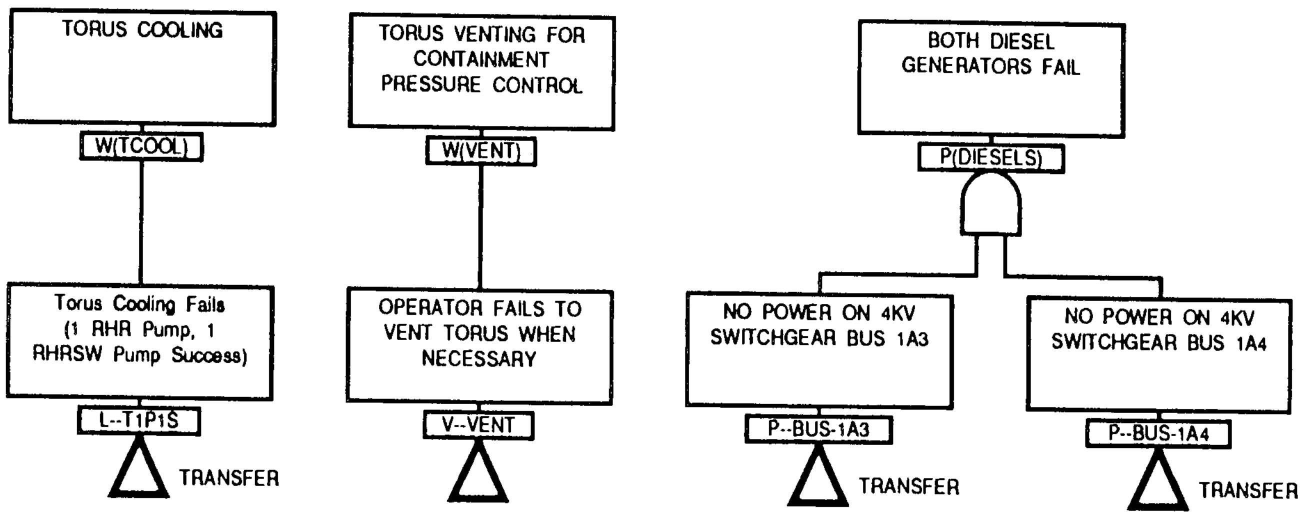

RCIC UNAVAILABLE
U(R)
RCIC Fails to Provide High Pressure Makeup
R--RCIC
3
TRANSFER
HPCI UNAVAILABLE
U(H)
HPCI Fails to Inject to RX Vessel
H--HPCI
3
TRANSFER
MAIN CONDENSER AVAILABLE
O(MC:AVAIL)
MAIN CONDENSER FAILS AT BEGINNING OF ACCIDENT
C--COND-INITIAL
RCIC OR HPCI AVAILABLE FOR HIGH PRESSURE INJECTION
U(H:R)
HPCI Fails to Inject to RX Vessel
H--HPCI
3
TRANSFER
RCIC Fails to Provide High Pressure Makeup
R--RCIC
3
TRANSFER


FEEDWATER AND CONDENSATE FOR HIGH PRESSURE INJECTION
Q(FW:CND)
Feedwater Fails to Provide High Pressure Makeup
Q--FEED-AND-COND
3
TRANSFER
PCS RE-ESTABLISHED
Z(PCS:RESTORE)
Failure to Revmove Heat Via Main Condenser
C--CONDENSER
TRANSFER


TIMELY REACTOR DEPRESSURIZATION
X(TIME:RX:DEP)
Depressurization Fails
O--ADS
TRANSFER
CRD AVAILABLE FOR HIGH PRESSURE INJECTION
U(CRD)
CRD FAILS TO DELIVER FLOW FROM 1 PUMP
D--CRD-HCU-8HR
2
TRANSFER



Top Logic for DAEC Event Trees	FIGURE 3.2-25 (CONT.)	11-03-92	Page 318


TORUS COOLING
W(TCOOL)
Torus Cooling Fails (1 RHR Pump, 1 RHRSW Pump Success)
L--T1P1S
TRANSFER
TORUS VENTING FOR CONTAINMENT PRESSURE CONTROL
W(VENT)
OPERATOR FAILS TO VENT TORUS WHEN NECESSARY
V--VENT
BOTH DIESEL GENERATORS FAIL
P(DIESELS)
NO POWER ON 4KV SWITCHGEAR BUS 1A3
P--BUS-1A3
TRANSFER
NO POWER ON 4KV SWITCHGEAR BUS 1A4
P--BUS-1A4
TRANSFER

3.2.3 System Dependency Matrix

The System Dependency Matrix is presented in Table 3.2-19. This matrix represents all the system interdependencies at the DAEC plant. Refer to Section 3.2.1, where dependency matrices for individual systems are presented.

Room and Component Cooling Requirements

A review of cooling requirements for components analyzed in the Duane Arnold Individual Plant Examination (IPE) has been performed to ensure support systems are properly accounted for in the fault tree models. Table 3.2-20 summarizes the findings of this review.

Table 3.2-20 lists each of the components evaluated, indicates the primary method the components are cooled and lists any alternate method available (e.g. running fans without chillers). The table also lists any other options available to maintain component temperatures if the primary and alternate methods of component cooling are unavailable.

Table 3.2-19
System Dependency Matrix

SUPPORT SYSTEM	REACTIVITY CONTROL						HIGH PRESSURE COOLANT MAKEUP						RPV DEPRESSURIZATION				LOW PRESSURE COOLANT MAKEUP													CONTAINMENT PRESSURE/ TEMPERATURE CONTROL				
	RPS		RPT		SLC		FEED-WATER		HPCI	RCIC	CRD		ADS/SRV LOGIC		SRVs		COND		LPCI		CS		RHRSW		ESW		GSW			MSIV REMAIN OPEN	MAIN CONDENS	TORUS COOLING		TORUS VENTING
	A	B	A	B	A	B	A	B			A	B	A	B	A	B	A	B	A	B	A	B	A	B	A	B	A	B	C			A	B	
4160 VAC Bus 1A1							X[2]										X[Mw]														P[9]			
4160 VAC Bus 1A2								X[2]										X[2]										X[2]			P[9]			
4160 VAC Bus 1A3											X[2]								X[2]		X[2]		X[2]				X[2]					X[2]		
4160 VAC Bus 1A4												X[2]								X[2]		X[2]		X[2]				X[2]					X[2]	
480 VAC Bus 1B34					X				P[4]	P[4]									X[1]		X[1]		P[33]		P[33]			P[33]				X[1]		
480 VAC Bus 1B43															P[10]	P[10]																		
480 VAC Bus 1B44						X			P[5]	P[5]										X[1]		X[1]	P[33]		P[33]			P[33]					X[1]	
480 VAC Bus B32																									X[2]									
480 VAC Bus B42																										X[2]								
480 VAC MCC 1L80					X[32]																													
480 VAC MCC Bus 1B12							P[27]										P[27]													P[27]	P[27]			
480 VAC MCC Bus 1B22								P[27]										P[27]												P[27]	P[27]			
120 VAC Panel Y30A	X																													P[28]	P[28]			
120 VAC Panel Y30B		X																												P[28]	P[28]			
125 VDC Panel 1D11							P[18]										P[18]				P[12]										P[8]			X[7]

Table 3.2-19
System Dependency Matrix

SUPPORT SYSTEM	REACTIVITY CONTROL			HIGH PRESSURE COOLANT MAKEUP				RPV DEPRESSURIZATION		LOW PRESSURE COOLANT MAKEUP						CONTAINMENT PRESSURE/ TEMPERATURE CONTROL			
	RPS A B	RPT A B	SLC A B	FEED-WATER A B	HPCI	RCIC	CRD A B	ADS/SRV LOGIC A B	SRVs A B	COND A B	LPCI A B	CS A B	RHRSW A B	ESW A B	GSW A B C	MSIV REMAIN OPEN	MAIN CONDENS	TORUS COOLING A B	TORUS VENTING
125 VDC Panel 1D13		X[20]				X[31]	P[21]	P	P B		P[12]	P[12]	P[12]			P[28]	P[28]	P[12]	
125 VDC Panel 1D14						X													
125 VDC Panel 1D21				P[18]	P[30]					P[18]		P[12]					P[8]		X[7]
125 VDC Panel 1D23		X[20]			X		P[21]	B P	B P		P[12]	P[12]	P[12]	P[12]	P[12]	P[28]	P[28]	P[12]	
250 VDC MCC 1D41					X														
GSW				X[17]			X[22]			X[17]					X X X	D[26]	P[37]		
RBCCW							X[22]												
ESW (A)					P[3][6]	P[3][6]					(16)	X[13]	X[13]					(16)	
ESW (B)					P[3][6]	P[3][6]					(16)	X[13]	X[13]					(16)	
RHRSW (A)													X					X[24]	
RHRSW (B)													X					X[24]	
Stilling Basin													X[15]	X[15]					
River Water				(36)						(36)			X[15]	X[15]	P[35]		P[34]		
Well Water				(36)						(36)					P[35]		P[34]		
Keep Full Pump											(25)	(25)						(25)	
Instrument Nitrogen				P[29]	(11)				P	P[29]						P[29]	P[29]		

Table 3.2-19
System Dependency Matrix

SUPPORT SYSTEM	REACTIVITY CONTROL			HIGH PRESSURE COOLANT MAKEUP				RPV DEPRESSURIZATION		LOW PRESSURE COOLANT MAKEUP						CONTAINMENT PRESSURE/ TEMPERATURE CONTROL			
	RPS A B	RPT A B	SLC A B	FEED-WATER A B	HPCI	RCIC	CRD A B	ADS/SRV LOGIC A B	SRVs A B	COND A B	LPCI A B	CS A B	RHRSW A B	ESW A B	GSW A B C	MSIV REMAIN OPEN	MAIN CONDENS	TORUS COOLING A B	TORUS VENTING
SRV N2 Accumulator									B										
Instrument Air				(19)			(23)												P
Air/N2 Accumulators				B(29)						B(29)						B(29)	B(29)		B
1V-AC-011											(14)	(14)						(14)	
1V-AC-012											(14)	(14)						(14)	
1V-AC-014A					P(6)														
1V-AC-014B					P(6)														
1V-AC-015A						P(6)													
1V-AC-015B						P(6)													
Circ. Water Pumps																	X		

X = COMPLETE DEPENDENCE — COMPLETE DEPENDENCE. NO BACKUP. THE FRONTLINE SYSTEM IS IMMEDIATELY AND COMPLETELY UNAVAILABLE.

P = PRIMARY DEPENDENCE — PARTIAL DEPENDENCE. NORMAL BACKUP EXISTS.

B = BACKUP DEPENDENCE — BACK-UP SOURCE OF SUPPORT FOR THE FRONTLINE SYSTEM.

D = DELAYED DEPENDENCE — IMPACT ON THE FRONTLINE SYSTEM NOT IMMEDIATE (E.G. LOSS OF ROOM COOLING). CODE CAN BE USED IN CONJUNCTION WITH CODES P AND B IF BACKUP SUPPORT SYSTEMS EXIST. IF NO BACKUP SUPPORT SYSTEMS EXIST, THE CODE D IS USED BY ITSELF.

Table 3.2-19
SYSTEM DEPENDENCY MATRIX

(1) Bus required for valve motive power.

(2) Bus required for pump motive power.

(3) ESW required for room cooling.

(4) Loss of Bus 1B34 results in loss of HPCI/RCIC room cooler IV-AC-15A. Room cooler IV-AC-15B can be used as a backup.

(5) Loss of Bus 1B44 results in loss of HPCI/RCIC room cooler IV-AC-15B. Room cooler IV-AC-15A can be used as a backup.

(6) HPCI/RCIC room temperature can also be maintained by opening pump room door 30 minutes into scenario (AOP 301.1).

(7) Torus Vent valve fails closed on loss of DC power.

(8) Loss of 1D11 and 1D21 results in loss of circulating water pump breaker control. Breakers can be manually operated.

(9) 1A1 and 1A2 provide motive power for the circulating water pumps. Both must fail to disable main condenser.

(10) Loss of Bus 1B43 results in loss of normal N_2 supply to all SRVs. Backup accumulator available.

(11) Instrument nitrogen required for steam line drain isolation and test bypass shutoff valves. Loss of nitrogen does not affect HPCI operability.

(12) 125 VDC power required for RHRSW, GSW, CS and RHR breaker closure. DC Panels 1D11/1D13 and 1D21/1D23 can be crosstied to provide breaker control power if loss of either DC Bus occurs. Breakers can also be manually operated at cabinets.

(13) ESW required for pump motor cooling.

(14) Loss of RHR/CS pump room cooling will not result in pump failure.

(15) River Water makeup to stilling basin required for successful RHRSW and ESW

operation. One of four River Water pumps is sufficient.

(16) ESW is normally provided for RHR pump seal cooling. Per pump manufacturer, loss of seal cooling will not preclude pump from performing required safety function.

(17) GSW is required for feedwater and condensate pump motor cooling.

(18) Loss of 125 VDC Panels 1D11 and 1D21 results in loss of feedwater and condensate breaker control. Breakers can be manually operated at the cabinet.

(19) Feedwater regulating valves fail as is on loss of instrument air.

(20) DC power is required to provide power to the RPT logic circuit and to energize the RPT breaker trip coils.

(21) Loss of 125 VDC Panels 1D13 and 1D23 results in loss of CRD pump breaker control. The breakers can be manually operated at the cabinet.

(22) RBCCW is required for CRD pump cooling. GSW provides RBCCW heat exchanger cooling. CRD pumps assumed to fail if cooling is lost.

(23) Control valve CV-1821 on the cooling water header fails closed on loss of instrument air. The charging line can also be used for injection.

(24) RHRSW train A cools RHR heat exchanger A; RHRSW train B cools RHR heat exchanger B.

(25) The RHR/Core Spray keep Full Pump maintains the discharge piping in both systems filled and pressurized. IT is assumed there is a conditional probability that failure of the keep fill system results in a pipe failure due to water hammer. Failure of the keep fill system does not necessarily result in the failure of LPCI, CS or RHR torus cooling.

(26) Loss of General Service Water results in loss of cooling to instrument nitrogen compressor 1K-14 and after-coolers.

(27) 480 VAC MCC 1B12 and 1B22 are required to open turbine bypass valves BPV1 and BPV2, respectively. One of two bypass valves must be open for successful

condensing function.

(28) One of four MSIVs must be open for successful condensing function. Each MSIV is provided with an AC (Y30A or Y30B) and a DC (1D13 or 1D23) solenoid vent valve. closure of an MSIV requires venting of both AC and DC solenoid valves.

(29) Containment instrument nitrogen supply is required to keep MSIVs open. Nitrogen accumulators are available as a backup pneumatic supply.

(30) 125 VDC Panel 1D21 required for automatic HPCI initiation. Manual operator initiation can be utilized.

(31) 125 VDC Panel 1D13 required for automatic RCIC initiation. Manual operator initiation can be utilized.

(32) 1L80 required for SLC tank piping heat tracing capability.

(33) Required for RHRSW, GSW and ESW re-alignment for alternate low pressure reactor injection.

(34) Makeup to the circulating water pit for condenser operation can be provided by the Well Water System or the River Water System.

(35) For successful utilization of GSW as an alternate injection source, makeup to the circulating water pit via the Well Water System or the River Water System must be available.

(36) Operation of GSW for component cooling purposes only does not require makeup to the circulating water pit.

(37) Loss of GSW induces a loss of circ water which causes a loss of main condenser vacuum.

1. HPCI Pump

The HPCI pump room can be cooled by either HVAC unit 1V-AC-014A or 1V-AC-014B.

A calculation was performed to determine the transient and steady state temperature in the HPCI room during a loss of ventilation event due to Station Blackout. The calculation assumes the door to the HPCI room is opened 30 minutes into the transient. The final room temperature at the end of the 4 hour Station Blackout scenario is 139°F.

Abnormal Operating Procedure AOP 301.1, "Station Blackout", Attachment 6 provides a list of doors which will be blocked open immediately following initiation of the AOP.

For purposes of the IPE, it is assumed if normal ventilation fails, operator action must be taken within 30 minutes to open the HPCI room door.

2. RCIC Pump

The RCIC pump room can be cooled by HVAC unit 1V-AC-015A or 1V-AC-015B.

A calculation was performed to determine the transient and steady state temperature in the RCIC room during a loss of ventilation event due to Station Blackout. Unlike the HPCi calculation, it is not assumed the RCIC room door will be opened during the scenario. The resulting RCIC room temperature at the end of the 4 hour Station Blackout event is 129°F. It should be noted the insulation on the RCIC turbine and associated steam piping has recently been upgraded and may account for the differences in results as compared to the HPCI pump room.

Because the IPE evaluates a 24 hour scenario, and the operator action to open the RCIC room doors is proceduralized in Attachment 6 of AOP 301.1, it is consecutively assumed the RCIC room door must also be opened within 30 minutes if normal ventilation fails.

3. Core Spray Pumps

The core spray pump rooms are normally cooled by HVAC units 1V-AC-011 and 1V-AC-012. Additionally, Emergency Service Water (ESW) provides motor bearing cooling for the core spray pumps.

Approximately 3 gpm of ESW for motor bearing cooling is required for successful core spray pump operation. The estimated time to motor failure given a loss of ESW is about thirty minutes.

HVAC unit 1V-AC-011 provided cooling for core spray pump B and HVAC unit 1V-AC-012 provides cooling for core spray pump A. With respect to room cooling for the IPE, indicates that due to the large CS pump room volume and the fact that ESW is provided for motor cooling, loss of HVAC will not result in pump failure during the 24 hour mission time.

In summary, only ESW for motor bearing cooling is required for successful operation of the core spray pumps.

4. LPCI/RHR Pumps

For IPE purposes, the Residual Heat Removal (RHR) pumps are required for Low Pressure Coolant Injection (LPCI) as well as Torus Cooling and Containment Spray operation. The RHR pump seals are cooled by the ESW system and the RHR pump motors are air-cooled. Additionally, the RHR/Core Spray pump room cooling units (1V-AC-011 and 1V-AC-012), require ESW for successful operation.

The RHR pumps can successfully perform their design function with loss of ESW to the seal coolers. Failure of the seals will most likely occur, however, the amount of bypass flow out of the seals is not enough to preclude the pumps from performing their safety function.

With respect to room cooling requirements, the RHR pump rooms (ECCS corner rooms) are very large, and the IPE mission time is short (24 hours). Based on operating experience, the RHR pump rooms are not expected to reach a temperature high enough to preclude successful operation of the pumps if room cooling is lost.

In summary, it is assumed for the IPE that neither ESW seal cooling nor RHR pump room HVAC is required for any mode of RHR pump operation.

5. Essential Switchgear Rooms

The essential switchgear rooms are cooled by the Control Building Chillers which are connected to both ESW and Well Water (WW). The system having the highest pressure will supply the chillers. There are two Control Building HVAC systems, one supplied by Division 1 power and one supplied by Division 2. Either system is capable of maintaining conditions in the switchgear rooms.

The Control Building Ventilation fans can be operated with or without the chillers available. For purposes of the IPE, it is assumed 2 days per year the chillers must operate in order to keep the switchgear rooms cool enough to prevent component damage. This assumption is based on input from the HVAC system engineer as well as operating experience when the chillers have been unavailable.

In the event all Control Building ventilation is lost, an alternate ventilation path is established by opening security doors and manually energizing two permanently mounted fans. This action is proceduralized in emergency Operating Procedure EOP-6. The fans are fed from the emergency bus and provide air flow through the Division I and II Switchgear rooms.

6. Non-Essential Switchgear Rooms

The non-essential switchgear rooms are cooled by the Turbine Building HVAC Unit 1V-AC-020. The HVAC unit chiller requires support from the Well Water system for successful operation. It is conservatively assumed the HVAC units can successfully cool the non-essential switchgear rooms only if the chiller is available.

There is not proceduralized backup method of ventilation.

7. RHR Service Water Pumps

The RHRSW pump motors are cooled by the ESW system. Each RHRSW motor cooler requires approximately 4 gpm of ESW flow. Successful operation of the RHRSW pumps requires the availability of the ESW system.

The RHRSW pump house HVAC system maintains ambient air temperature near the pumps within their design range of 40°F to 104°F. The impact of a loss of HVAC is dependent on the outside air temperature however, for the IPE it is assumed the RHRSW pumps can operate for 24 hours following a loss of normal ventilation. This assumption has been confirmed with the RHRSW system engineer.

8. Emergency Diesel Generators

The Emergency Diesel Generator (DG) engines have the largest demand on the ESW system during transient/accident conditions. Each DG lube oil cooler requires approximately 565 gpm of ESW flow. Each DG has a series arrangement of three heat exchangers whose heat-rejection requirements and cooling flow rates are determined by the unit's lubrication oil cooler. To prevent overheating of the diesel engines, the two ESW pumps start immediately after the DGs come up to speed and the essential bus is energized.

For purposes of the IPE, it is assumed each Emergency Diesel Generator must be cooled by the associated ESW train or by the redundant ESW train provided operator action is taken to cross-tie the ESW trains.

9. Feedwater and Condensate Pumps

The Feedwater and condensate pump motors are cooled by the GSW system. Successful operation of these pumps requires the availability of the GSW system.

The successful operation of one of three GSW pumps is sufficient to provide adequate cooling water to the feedwater and condensate pump motor coolers. Heat from these coolers is rejected to the circ water pit which, even with a loss of river water makeup, having 2 million gallons of water, is essentially an infinite heat sink for the purposes of pump motor cooling.

TABLE 3.2-20

COMPONENT COOLING SUMMARY

	Primary Cooling	Alternate	Other Method
HPCI	1VAC014A or 1VAC014B (ESW required)	HPCI room door can be opened 30 minutes into scenario	Bypass Trips (LOOP only)
RCIC	1VAC015A or 1VAC015B (ESW required)	RCIC room door can be opened 30 minutes into scenario	Bypass Trips (LOOP only)
CS	ESW for motor cooling	None	None
LPCI/RHR	Not required	Not require	None
Essential Switchgear	Division 1 Ventilation (1VAC030A) or Division 2 Ventilation (1VAC030B) (Well Water or ESW required for both)	Division 1 Fans (1VAC030A, 1VEF030A, or 1VEF030C) or Division 2 Fan (1VAC030B or 1VEF030B) (Acceptable 363 of 365 days per year)	EOP-6 provides alternate Control Building cooling options
Non Essential Switchgear	1VAC020 (Well Water required)	None	None
RHRSW	ESW for motor cooing required	None	None
Emergency Diesel Generators	1 Train of ESW required for lube oil cooling of each diesel generator	None	None

3.3.1 List of Generic Data

Generic sources are used for component failure rate data for the DAEC IPE. (See Table 3.3-1.) Plant specific data is used for (1) system/train unavailabilities due to maintenance or testing, (2) common cause failure data, (3) human reliabilities, and (4) initiating events when feasible. (See Sections 3.3.4, 3.3.3, and 3.3.1 for further discussion with regard to the last three items.) Use of generic component failure rate data is conservative and justified in Section 3.3.2. The references for generic data are, in order of preference to the DAEC PRA:

1) PSA Procedures Guide (NUREG/CR-2815)
2) NUREG/CR-1150 (NUREG/CR-4550)
3) NUCLARR
4) Shoreham PRA
5) Oconee PRA, NSAC/60
6) IEEE 500
7) Calvert Cliffs PRA
8) Oak Ridge National Laboratories Reliability Data
9) ALWR Reliability Data
10) Monticello PRA

The generic database of component failure rates was assembled using a wide variety of sources, indicated above, that includes NUREGs, plant-specific PRAs, and a previous failure rate database acquired from February 1987. The data from the Shoreham PRA is a composite of three basic sources: (1) acutal operating experience data, as reported to the NRC in the Licensee Event Reports, and analyzed by EG&G for the NRC to obtain failure rate values; (2) WASH-1400 which provided median failure rates based on plant operating experience and other applications of similar components; and (3) data collected by General Electric. The other database was the source for the Oconee PRA, Calvert Cliffs IREP, and Oak Ridge National Laboratory data.

Several criteria were used in the selection of values for use in component type database. These include:

1) priority of information source;
2) applicability of units (i.e., per demand or per hour);
3) applicability to BWRs; and
4) consistency with other data sources.

Data sources were prioritized as follows: highest priority was given to NUREG/CR-2815, due to the fact that it was referred to as a valid data source in Generic Letter 88-20; of next highest priority was NUREG/CR-4550, as it is the latest PRA published by the NRC; EGG-SSRE-8875 (NUCLARR database) is the third highest priority because it is the latest NRC published database; the Shoreham PRA is the fourth highest priority; and finally, the remaining sources were used when no other information was provided in the previous references.

In addition to priority of data source, applicability of units was important in selecting values, in that if the units did not correspond to the failure mode, the value was not selected. For example, for the failure mode "failure to start," only demand failures were considered to be applicable. Similarly, only hourly failures were considered for "failure to run."

A final consideration in the selection of data was its consistency compared to other data sources. Generally, if a value was within a factor of five of other data, it was considered applicable. Occasionally, a priority reference presents an unusually low value when compared against all other references. In that case, it is conservative to use the next priority value.

Generic initiating events are used when no plant data is available to estimate an initiating event frequency. The following generic initiatiing events are used for the Duane Arnold Energy Center IPE:

Large LOCA
Medium LOCA
Small LOCA
LOCA Outside Containment
Loss of Offsite Power (combination of generic and plant specific data)
ATWS (combination of generic and plant specific data)
Loss of River Water

For further information on generic and plant specific initiating event frequencies, refer to Table 3.1-1.

Table 3.3-1
Generic Components Failure Rates Data

Component		Failure Mode	PSA Procedures Guide (Ref. 1)	NUREG/CR-1150 (Ref. 2)	NUCLARR (Ref. 3)	Shoreham PRA (Ref. 4)	Oconee PRA NSAC/60 (Ref. 5)	IEEE 500 (Ref. 6)	Calvert Cliffs (Ref. 7)	Oak Ridge Natl. Lab. Rel. Data (Ref. 8)	ALWR Reliability Data (Ref. 9)	Monticello PRA (Ref. 10)	Remarks
Accumulator	AT	FE			5E-6/h (10)								
Annunciator	AI	LO			1E-6/h (10)			7.8E-7/h					
Valve- Air Operated	AV	MB		8E-4/d (10)									
Valve- Air Operated	AV	MF		8E-4/d (10)									
Valve- Air Operated	AV	ML		8E-4/d (10)									
Valve- Air Operated	AV	FB		1E-7/h (3)									
Valve- Air Operated	AV	FL		5E-7/h (10)									
Valve- Air Operated	AV	FC		2E-3/d									
Valve- Air Operated	AV	FO	1E-5/h	2E-3/d (3)		3E-3/d	9E-4/d			PWR-4.8E-3/d BWR-2.8E-3/d			NUREG/CR-4550: hdwr. faults & Control ckt. cmd. faults included; [1]
Bus- AC Power	AB	FB	3E-8/h								2E-7/h		
Bus- AC Power	AB	FL	3E-8/h								2E-7/h		
Battery Charger	BC	LO						1.15E-6/h		5.5E-6/h			IEEE-Composite of no output & low output
Bus - 125 VDC	DB	FB		1E-7/h (5)									
Bus - 125 VDC	DB	FL		1E-7/h (5)									NUREG/CR-4550: hardware failure
Battery	BT	LO	2E-6/h	1E-6/h (3)	2E-5/h (5)	3.75E-6/h (3)	9E-8/h	2.65E-6/h		3.84E-6/h	**5E-4/d**		Fails to provide proper output; [2]
Control Breaker	CB	FO	1E-5/h	3E-3/d (10)		1.25E-3/d (3)							EG&G-indoor/outdoor composite; [2]
Control Breaker- >4160V	CB	FC					7.9E-4/d						

Table 3.3-1
Generic Components Failure Rates Data

Component		Failure Mode	PSA Procedures Guide (Ref. 1)	NUREG/CR -1150 (Ref. 2)	NUCLARR (Ref. 3)	Shoreham PRA (Ref. 4)	Oconee PRA NSAC/60 (Ref. 5)	IEEE 500 (Ref. 6)	Calvert Cliffs (Ref. 7)	Oak Ridge Natl. Lab. Rel. Data (Ref. 8)	ALWR Reliability Data (Ref. 9)	Monticello PRA (Ref. 10)	Remarks
Control Breaker	CB	FB	3E-5/h	1E-6/h (3)	6E-7/h (10)	1.25E-6/h (3)							EG&G-indoor/outdoor composite; [2]
Control Breaker-4160V	CB	FC			1E-3/d			4.29E-5/d					EG&G- Indoor /outdoor composite
Control Breaker-480V	CB	FL					1.6E-7/h						[10]
Control Breaker-≤120V	CB	FL					1.6E-7/h						[10]
Protective Breaker->4160V	CB	FC					7.9E-4/d						
Protective Breaker->4160V	CB	FO					4.3E-5/d						
Protective Breaker->=460V	CB	FB		1E-6/h (3)									[2]
Protective Breaker-480V	CB	FL					1.6E-7/h						[10]
Protective Breaker-≤120V	CB	FO		3E-3/d (10)									[2]
Protective Breaker-≤120V	CB	FB		1E-6/h (3)									[2]
Protective Breaker-≤120V	CB	FL					1.6E-7/h						[10]
Chiller	CH	FB		5.7E-6/h									
Chiller	CH	FS									8.1E-3/d		
Chiller	CH	FR						4.01E-6/h			5E-6/h		
Compressor	CP	FS		8E-2/d (3)	5E-3/d (5)						1E-2/d		

Table 3.3-1
Generic Components Failure Rates Data

Component		Failure Mode	PSA Procedures Guide (Ref. 1)	NUREG/CR -1150 (Ref. 2)	NUCLARR (Ref. 3)	Shoreham PRA (Ref. 4)	Oconee PRA NSAC/60 (Ref. 5)	IEEE 500 (Ref. 6)	Calvert Cliffs (Ref. 7)	Oak Ridge Natl. Lab. Rel. Data (Ref. 8)	ALWR Reliability Data (Ref. 9)	Monticello PRA (Ref. 10)	Remarks
Compressor	CP	FR		2E-4/h (10)	1E-4/h (10)						1E-4/h		
Valve- Check	CV	FO	2E-6/h	1E-4/d (3)	1E-3/d (5)	1.6E-6/h					2E-4/d	2E-6/h	[3] [16]
Valve- Check	CV	FC	2E-7/h	1E-4/d (3)	5E-5/d (5)	1E-4/d					2E-4/d	2E-7/h	[1] [16]
Valve- Check	CV	FB									2E-7/h		
Valve- Check	CV	LB									2E-7/h		
Valve- Check	CV	FL			3.55E-8/h (10)	1.07E-6/h					6E-7/h		Composite of leakage & rupture; [1]
Diesel Generator	DG	FR	3E-3/h	2E-3/h (10)	5E-3/h (10)	3E-2/h		1.7E-7/h		3.6E-4/h	2.4E-3/h		[1]
Diesel Generator	DG	FS	6E-5/h	3E-2/d (3)	1E-2/d (5)	4E-2/d		1.82E-2/d		2.9E-3/d	1.4E-2/d		[1]
Electro-pneumatic Contrlr	EP	LO					1.6E-4/h (3)						
Electro-Hyraulic Valve	EV	FB		1E-7/h (3)									
Electro-Hydraulic Valve	EV	FC		2E-3/d (3)									
Electro-Hydraulic Valve	EV	FO		2E-3/d (3)									
Filter	GF	FB	3E-6/h	3E-5/h (10)	1E-5/h (10)						2E-6/h	3E-5/h	For clear fluids. [16]
Fan / Blower	BL	FR		1E-5/h (3)	3E-5/h (10)						1E-5/h	1.8E-6/h	[18]
Fan / Blower	BL	FS		3E-4/d (3)	5E-3/d (5)						6E-4/d	2.2E-3/d	[18]
Flow Indicator-Sensor	FI	HO						3.9E-7/h					Comp of zero/max & low output [8]
Flow Indicator-Transmitr	FI	HO						1.91E-6/h					Comp of zero/max & high output [8]

Table 3.3-1
Generic Components Failure Rates Data

Component		Failure Mode	PSA Procedures Guide (Ref. 1)	NUREG/CR-1150 (Ref. 2)	NUCLARR (Ref. 3)	Shoreham PRA (Ref. 4)	Oconee PRA NSAC/60 (Ref. 5)	IEEE 500 (Ref. 6)	Calvert Cliffs (Ref. 7)	Oak Ridge Natl. Lab. Rel. Data (Ref. 8)	ALWR Reliability Data (Ref. 9)	Monticello PRA (Ref. 10)	Remarks
Flow Indicator-Switch	FI	HO		3E-6/h (10)		2.6E-7/h							[3] [8]
Flow Indicator-Meter	FI	HO & LO						4.82E-6/h					All modes. [8] [9]
Flow Indicator-Sensor	FI	LO						4.2E-7/h					Comp of zero/max & low output [9]
Flow Indicator-Transmitr	FI	LO						1.49E-6/h					Comp of zero/max & low output [9]
Flow Indicator-Switch	FI	LO		3E-6/h (10)		2.6E-7/h							[9]
Fuse	FU	FO			1E-6/d (10)	1.25E-5/d (3)						3E-6/h	[2] [16]
Gas Bottle	GT	FF						6E-7/h (10)					
Heat Exchanger (Shell)	HX	FE	3E-6/h	3E-6/h (10)	3E-7/h (10)	5.7E-6/h					1E-6/h		Composite of leakage and rupture; [3]
Heat Exchanger (Tube)	HX	FL	3E-9/h	3E-6/h (10)	1E-8/h (10)	5.7E-6/h					1E-6/h		Composite of leakage and rupture; NUREG/CR-4550 does not specify whether shell or tube H/X; [3]
Heat Exchanger (Tube)	HX	FB		5.7E-6/h (10)	3E-7/h (10)						1E-6/h	5.7E-6/h	Not specified as shell or tube in NUREG/CR-4550, ALWR. [20]
Inverter	IN	LO	6E-5/h		1E-2/d (5) or 5E-6/h (5)		1.3E-4/h	5.4E-6/h		2.1E-5/h	2E-5/h	6E-5/h	[16]
Level Indicator-Sensor	LI	FF		3E-6/h (10)	1E-6/h (10)	3.9E-6/h							[3] Generic failure[10]
Level Indicator-Transmtr	LI	FF		3E-6/h (10)	3E-6/h (10)						5E-6/h	2.7E-6/h	[10] [17]

Table 3.3-1
Generic Components Failure Rates Data

Component		Failure Mode	PSA Procedures Guide (Ref. 1)	NUREG/CR -1150 (Ref. 2)	NUCLARR (Ref. 3)	Shoreham PRA (Ref. 4)	Oconee PRA NSAC/60 (Ref. 5)	IEEE 500 (Ref. 6)	Calvert Cliffs (Ref. 7)	Oak Ridge Natl. Lab. Rel. Data (Ref. 8)	ALWR Reliability Data (Ref. 9)	Monticello PRA (Ref. 10)	Remarks
Level Indicator-Switch	LI	FF		3E-6/h (10)							3E-7/h		[10]
Level Indicator-Meter	LI	FF						4.62E-6/h				2.7E-6/h	All modes[10].General instrumentation[17].
Level Indicator-(see failure mode FF)	LI	LO											Generic Failure: All modes [11]
Pump-Motor	MP	FF			3E-6/h (10)								EG&G-leakage & rupture composite
Pump-Motor	MP	FR	1E-4/h	3E-5/h (10)	3E-5/h (10)	2.8E-5/h					2.5E-5/h	1E-4/h	Shoreham: Comp. of normal, standby, & post-accident scenarios; [1][6][16]
Pump-Motor	MP	FS	1E-5/h	3E-3/d (10)	3E-3/d (5)	2E-4/d	5E-4/d			5.3E-3/d	2E-3/d	1E-5/h	Ckt. bkr. cmd. faults & hdwr. faults included in NUREG/ CR-4550 value;[1][7][16]
Valve-Motor Operated	MV	FB	2E-7/h	1E-7/h (10)		1.25E-4/d (3)	2.3E-7/h				1.3E-7/h		
Valve-Motor Operated	MV	FL	1E-7/h	5E-7/h (3)		1.25E-4/d (3)							[2]
Valve-Motor Operated	MV	FO	1E-5/h	3E-3/d (10)		1.25E-3/d (3)					4E-3/d		[2]
Valve-Motor Operated	MV	MB		8E-4/d (10)									
Valve-Motor Operated	MV	MF		8E-4/d (10)									
Valve-Motor Operated	MV	ML		8E-4/d (10)									
Valve-Motor Operated	MV	FC	1E-5/h	3E-3/d (10)		1.25E-3/d (3)					4E-3/d		[2]

Table 3.3-1
Generic Components Failure Rates Data

Component		Failure Mode	PSA Procedures Guide (Ref. 1)	NUREG/CR -1150 (Ref. 2)	NUCLARR (Ref. 3)	Shoreham PRA (Ref. 4)	Oconee PRA NSAC/60 (Ref. 5)	IEEE 500 (Ref. 6)	Calvert Cliffs (Ref. 7)	Oak Ridge Natl. Lab. Rel. Data (Ref. 8)	ALWR Reliability Data (Ref. 9)	Monticello PRA (Ref. 10)	Remarks
Switch- Manual	HS	FF	1E-6/h			1.25E-5/d (3)						1.25E-5/d	[2] [17]
Valve- Pneumatic	NV	FC			1E-3/d (10)	3E-3/d		4.7E-7/d					[1]
Valve- Pneumatic	NV	FB				1.25E-4/d (3)	2.3E-7/h	3.7E-7/h (10)					[2]
Pipe	LP	FB			5E-10/h (30)								1-3 inch (per foot)
Pipe	BP	FE			1.05E-9/h (30)	8.59E-10/h (30)						8.6E-10/h	> 3 in; EG&G-leakage & rupture composite; [2] [17]
Pipe	LP	FE			1.05E-8/h (30)	8.59E-9/h (30)						8.6E-9/h	≤ 3 in; EG&G-leakage & rupture composite; [2] [17]
Pressure Indicator-Sensor	PI	HO & LO		3E-6/h (10)	1E-6/h (10)	1.25E-4/d (3)		5.15E-6/h					[2] [12] [13]
Pressure Indicator-Transmitter	PI	HO						7.5E-7/h				2.7E-6/h	Zero or max output [12] [17]
Pressure Indicator-Transmitter	PI	LO						1.5E-7/h					[13]
Pressure Indicator-Switch	PI	HO & LO	2E-7/h	3E-6/h (10)		1.25E-4/d (3)					3E-7/h	2E-7/h or 8.04E-8	[12] [13] [16] [17]
Pressure Indicator-Meter	PI	HO & LO						4.62E-6/h					All modes. [12] [13]
Relay	RE	LO			5E-7/h (10)			1.0E-7/h			6E-7/h	1E-6/h	[16]
Relay	RE	FF	1E-6/h			1.25E-4/d (3)					1E-4/d	1E-6/h or 2.7E-4/d	[16] [17]
Relief Valve	RV	FL				6E-6/h						6E-6/h	[1] [19]

Table 3.3-1
Generic Components Failure Rates Data

Component		Failure Mode	PSA Procedures Guide (Ref. 1)	NUREG/CR -1150 (Ref. 2)	NUCLARR (Ref. 3)	Shoreham PRA (Ref. 4)	Oconee PRA NSAC/60 (Ref. 5)	IEEE 500 (Ref. 6)	Calvert Cliffs (Ref. 7)	Oak Ridge Natl. Lab. Rel. Data (Ref. 8)	ALWR Reliability Data (Ref. 9)	Monticello PRA (Ref. 10)	Remarks
Relief Valve	RV	FO			6E-3/d (5)	5E-3/d					3E-3/d		[1] [21]
Relief Valve	RV	FC			3.3E-3/d (5)	8E-3/d					7.9E-3/d		[1]
Speed Indicator-Sensor	SI	HO						9.3E-7/h					Comp of zero/max & high output [14]
Speed Indicator-Transmitter	SI	HO		1E-6/h (3)									[14]
Speed Indicator-Switch	SI	HO						5.7E-7/h					[14]
Speed Indicator-Meter	SI	HO						4.62E-6/h					All modes. [14]
Valve- Solenoid	SV	FB				1.25E-4/d (3)		1.1E-7/h					[2]
Valve- Solenoid	SV	FC	2E-6/h	2E-3/d (3)	5E-4/d (10)	1.25E-3/d (3)				BWR: 2.3E-3/d		1.25E-4/d	see SV FF
Valve- Solenoid	SV	FF	2E-6/h	2E-3/d (3)	5E-4/d (10)	1.25E-3/d (3)				BWR: 2.3E-3/d		2E-6/h 1.25E-4/d	Hdwr. faults & Cntrl. ckt. cmd. faults in NUREG/CR-4550 value;[2][16][17]
Transformer	TR	LO	6E-7/h					5.1E-7/h					Composite of no & low output
Tank	TT	FE			5E-7/h (10)						1E-7/h	2.7E-7/h	ALWR-rupture; EG&G -leakage & rupture composite. [16]
Pump- Turbine	TP	FR			1E-4/h (10)								
Pump- Turbine	TP	FS	1E-4/h	3E-2/d (10)	3E-2/d (5)	3.75E-3/d (3)	4E-3/d				1.5E-2/d		[2]
Cable Wire	CW	FB	1E-5/h		1E-7/h (10)	3.75E-6/h (3)		7.54E-6/h					EG&G/IEEE-All modes; [2]

Table 3.3-1
Generic Components Failure Rates Data

Component		Failure Mode	PSA Procedures Guide (Ref. 1)	NUREG/CR -1150 (Ref. 2)	NUCLARR (Ref. 3)	Shoreham PRA (Ref. 4)	Oconee PRA NSAC/60 (Ref. 5)	IEEE 500 (Ref. 6)	Calvert Cliffs (Ref. 7)	Oak Ridge Natl. Lab. Rel. Data (Ref. 8)	ALWR Reliability Data (Ref. 9)	Monticello PRA (Ref. 10)	Remarks
Cable Wire	CW	FL	3E-8/h			2.68E-8/h (10)							Short to power; [2]
Cable Wire	CW	FL	1E-6/h			8.04E-7/h (10)							Short to ground; [2]
Temperature Indicator- Sensor	TI	HO		3E-6/h (10)	1E-6/h (10)								Failure [15]
Temperature Indicator-Transmitter	TI	HO						8.4E-7/h					Max or zero output [15]
Temperature Indicator-Switch	TI	HO		3E-6/h (10)		2.3E-6/h							[3] [15]
Temperature Indicator- Meter	TI	HO						4.62E-6/h					All modes. [15]
Vacuum Breaker	VV	FC							1E-5/d (3)			2E-7/h	[16]
Vacuum Breaker	VV	FL					4.3E-7/h						Fitzpatrick:1E-6/h (10)
Vacuum Breaker	VV	FO							1E-5/d (3)				
Valve- Manual	HV	FC		1E-4/d (3)	5E-4/d (10)	1E-4/d				BWR: 6.1E-4/d		1.25E-5/d	[1] [17]
Valve- Manual	HV	LC		1E-4/d (3)	5E-4/d (10)	1E-4/d				PWR: 4.2E-4/d BWR: 6.1E-4/d			[1]
Valve- Manual	HV	MB		8E-4/d (10)									
Valve- Manual	HV	MF		8E-4/d (10)									
Valve- Manual	HV	ML		8E-4/d (10)									
Valve- Manual	HV	FO		1E-4/d (3)	5E-4/d (10)	1E-4/d				PWR: 4.2E-4/d BWR: 6.1E-4/d			[1]
Valve- Manual	HV	FL			8E-8/h (10)	1E-7/h							EG&G leakage & rupture compos. [1]

Table 3.3-1
Generic Components Failure Rates Data

Component		Failure Mode	PSA Procedures Guide (Ref. 1)	NUREG/CR -1150 (Ref. 2)	NUCLARR (Ref. 3)	Shoreham PRA (Ref. 4)	Oconee PRA NSAC/60 (Ref. 5)	IEEE 500 (Ref. 6)	Calvert Cliffs (Ref. 7)	Oak Ridge Natl. Lab. Rel. Data (Ref. 8)	ALWR Reliability Data (Ref. 9)	Monticello PRA (Ref. 10)	Remarks
Valve- Manual	HV	LL			8E-8/h (10)	1E-7/h							EG&G leak/rupt. [1]
Valve- Manual	HV	FB		1E-7/h (3)	3E-9/h (10)	-					3.7E-8/h		
Valve- Manual	HV	LB		1E-7/h (3)	3E-9/h (10)						3.7E-8/h		
Switch- Limit	LS	FF	6E-6/h			3.75E-4/d (3)							[2]
Valve- Explosive	XV	FC		3E-3/d (3)									NUREG/CR-4550

Table 3.3-2
Addenda: Generic Component Failure Rates Data

Name	Description	Probability	Remarks
A-INST-AIR-LOSS	Loss of instrument air	1E-4	[22] [23]
P-SWING-BUS-XFER	Swing bus transfer logic failure	1E-3	[22]
LIMT	Level instrument out for testing/maintenance	6E-4	testing for 6 hours [22]
LUFF	Lubrification system failure	1E-3	[22]
LMPFF	Keep fill pump failure	1E-2	[22]
TTFF	SBLC tank failure	1E-3	[22]
RRCIC-TRIPPED	RCIC failure due to false turbine trip	2.5E-3	[22] [23]
RTPSI-SPD-CNTL	RCIC turbine speed controller failure	1E-3	[22]
HHPCI-SPD-CNTL	HPCI pump speed controller failure	1E-3	[22]
TSTFE	Suppression pool water unavailable	1E-6	[22]
TSTFF	Torus level failure	1E-5	[22]
TSTFF-TORUS-TEMP	Torus water unavailable due to high temperature	1E-5	[22]
TVVFF-TAIL	Tailpipe vacuum breaker sticks open	1E-3	[22]
WCRICBD	Conditional probability that circulatory water blowdown not available	1E-3	[22]
WGAF-1S087	Sluice gate fails to remain open	1E-3	[22]
Q-GROUP1	Conditional probability that a Group 1 isolation is received	1E-1	[22]
Q-LOAD-SHED	Conditional probability that load shedding has occurred and pumps must be restarted	5E-2	[22]
QFLOWCONT	Feedwater flow controller failure causes feedwater reg valves to fail close	1E-3	Generic failure of component to operate is 1E-4/demand, per NUREG/CR-4550.

NOTES:

( ) The values in the parethesis represent the error factor.

[1] Shoreham PRA reference - NRC LER data

[2] Shoreham PRA reference - WASH-1400

[3] Shoreham PRA reference - General Electric BWR data

[4] Bolded value indicates units are inconsistent with failure mode. Also see [5].

[5] To convert from an hourly failure rate to a demand failure rate, use:

$$D = (HxT)/2$$

where D = demand failure rate, H = hourly failure rate, and T = time interval between component tests.

[6] Specific motor-driven pump failure rates for failure to run were given in the ALWR reference: BWR CRD-2.4E-6/h; CCW-5E-6/h; Containment Spray-5E-5/h; Emergency Feed-1.5E-4/h; LPI/RHR-2E-5/h; SI-5E-5/h; SW-2.8E-5/h.

[7] Specific motor-driven pump failure rates for failure to start were given in the ALWR reference: BWR CRD-2.4E-3/d; CCW-1.3E-3/d; Containment Spray-5E-3/d; Emergency Feed-3E-3/d; LPI/RHR-3E-3/d; SI-1.3E-3/d; SW-4E-3/d.

[8] Failure rates for subcomponents should be added together to arrive at failure rate for Flow Indicator failure mode: FI HO.

[9] Failure rates for subcomponents should be added to get component failure rate: FI LO.

[10] Failure rates for subcomponents should be added together to arrive at failure rate for Level Indicator failure mode: LI FF.

[11] Failure rates for subcomponents are for generic failure mode. Use values from LI FF.

[12] Failure rates for subcomponents should be added together to arrive at failure rate for Pressure Indicator failure mode: PI HO.

[13] Failure rates for subcomponents should be added for Pressure Indicator mode PI LO.

[14] Failure rates for subcomponents should be added for Speed Indicator mode SI HO.

[15] Failure rates for subcomponents should be added for Temperature Indicator mode TI HO.

[16] Monticello PRA reference- NUREG/CR-2815

[17] Monticello PRA reference- WASH-1400

[18] Monticello PRA reference- Palisades PRA

[19] Monticello PRA reference- NUREG/CR-1363

[20] Monticello PRA reference- NUREG/CR-4550

[21] Based on industry experience, given that a relief valve has opened successfully, it will fail to reseat 15% of the time.

[22] Data was obtained from a conservative estimate.

[23] This is an estimate of loss of system without modeling it.

3.3.2 Plant Specific Data and Analysis

A comparison has been made between DAEC plant specific component failure rate data and failure rate data reported by the remaining commercial nuclear power industry to NPRDS. A summary of the comparison is provided in Table 3.3-3. The trend demonstrates component failure rates at DAEC are consistently as low or lower than those submitted by the industry to NPRDS. That is, for all components investigated, DAEC experienced lower failure rates than industry experienced. Therefore, generic failure rate data is conservative and is justified.

Generic data has been obtained for diesel generator failures. Two pieces of generic data apply to the plant at DAEC. Although the lower number can be justified, the average between the two generic values, which is conservative, is used in the DAEC PRA model. A consistent review of DAEC operating history demonstrates that the diesel generators have a lower failure rate than the average generic number used in the PRA model.

Plant specific unavailabilities due to testing or maintenance are modeled at the system/train level in the DAEC IPE-PRA. These data are summarized in Table 3.3.3-4. DAEC plant specific data for human reliability is discussed in Section 3.3.3. Plant specific initiating events data is used in the DAEC IPE when it was feasible to do so.

An evaluation of DAEC plant specific failure rates has been developed by comparing the DAEC failure reports submitted to NPRDS with the industry average values for key components. The results of that comparison indicate DAEC plant specific component failure rates are as low or lower than the failure rates reported to NPRDS by the rest of the industry. This demonstrates that generic data is valid and suitable for use in the DAEC IPE.

Table 3.3-3

DAEC Plant Specific Data
6/87 - 11/91

Component	DAEC-experienced Failure Rate	Industry-reported Failure Rate	Higher Failure Rate	Generic data OK?
HPCI Pump	0.0	2.01E-6	Industry	Yes
HPCI Turbine	0.0	1.74E-5	Industry	Yes
RCIC Pump	0.0	2.63E-6	Industry	Yes
RCIC Turbine	0.0	1.67E-5	Industry	Yes
DG Engine	0.0	6.70E-5	Industry	Yes
DG Breaker	1.27E-5	8.45E-6	DAEC	Yes [1]
MSIV	0.0	1.41E-5	Industry	Yes
SLC	0.0	1.87E-5	Industry	Yes
RHR Pump	0.0	9.44E-6	Industry	Yes
RHR Heat Exchanger	0.0	3.05E-6	Industry	Yes
ESW Pump	2.82E-6	2.39E-5	Industry	Yes
Relief Valve	0.0	2.53E-6	Industry	Yes
Check Valve	6.01E-7	3.36E-6	Industry	Yes
Motor-Operated Valve	1.03E-6	4.14E-6	Industry	Yes
125V Battery	0.0	3.45E-6	Industry	Yes
250V Battery	0.0	1.65E-6	Industry	Yes
125V Charger	0.0	1.85E-5	Industry	Yes
250V Charger	0.0	1.82E-5	Industry	Yes

[1] DAEC experienced a failure rate 50% higher than reported by industry (NPRDS). However, a small sample size of 1 failure allows the inustry value to be used.

Table 3.3-4
System Unavailabilities per Train

System Description	Unavailability per Train or Division	Comments
Automatic Depressurization & Safety Relief	less than 1E-4	[5] per circuitry realignment[6]
Circulation Water	less than 0.0048	[2] no evnt in last 3-4yrs [6]
Condensate	0.0064	2 weeks in 6 train-years [6]
Main Condenser	0.12	Loss of condenser vacuum [4]
Control Rod Drive	less than 0.0048	CRD pumps out of service only during refuel outage during 1988-1992 (assume 2 wk outage) [6]
Core Spray	0.0023	Tracking for last year [6]
Electric Power	0.005	Average emergency generator unavailability [1]
Emergency Service Water	0.0254	4 hrs. PM per year, 3 days CM per 24 months [6] [7]
Feedwater	0.0027	1 train for 2 weeks non-optimal over 3-4 years [6]
General Service Water	0.00365	1-day swap of spare pump on 18 mo. cycle [6][8]
High Pressure Coolant Injection	0.031	[1]
Reactor Core Isolation Cooling	0.03	[5] [6]
Recirculation Pump Trip	1E-4	[5] Testing will not affect circuit availability[6].
Residual Heat Removal	0.006	[1]
RHR Service Water	0.006	[1]
River Water Supply	0.005	[6]
Standby Liquid Control	0.0137	[5] 1 week (5 days) per year [6]
Torus/Torus Vent	2.3E-4	Open vacuum breakers, no maintenance performed. Testing procedure explicitly modeled in PRA model.
Well Water Supply	0.004566	2 weeks per 18 mo. cycle[6] [9]

Notes

[1] IE Comparative Performance Indicator Summary, September 1991.

[4] Initiator data.

[5] Best estimate.

[6] Information obtained from the assigned System Engineer.

[7] ESW unavailability was obtained by combining preventive maintenance and corrective maintenance as:

4 hours per year per train for PM, and a 3-day failure (CM) of one train over a 24-month period. Therefore,

4/(365*24) + 3/(2*30) = 0.00254.

[8] System engineer states that GSW pumps are taken out of service by swapping them with a spare reserve GSW pump when necessary. Each pump is swapped on an 18-month cycle, requiring 1 day (typically 1 day but never more than 1 week) per swap per pump. Therefore,

2d / (1.5*365d) = 0.00365.

[9] Again system engineer states that Well Water Supply pumps are taken out of service by repairing them on an 18-month cycle, procedure performed couple months apart per pump. The procedure typically requires 1 day per pump. However, the SE indicated on previous conversation that procedure might take 2 weeks for all pumps during the cycle. Therefore,

10 days / (4*1.5*365d) = 0.004566.

3.3.3 Human Failure Data

3.3.3.1 Scope and Treatment of Different Error Types

The types of human interactions that can influence a probabilistic evaluation of nuclear plant safety have been classified in the past as follows:

- Type A - Test and maintenance
- Type B - Actions causing initiating events
- Type C - Procedural actions during the course of an accident
- Type D - Actions leading to inappropriate actions (Sometimes referred to as errors of commission)
- Type E - Recovery actions.

These five types of human interactions are included in the DAEC Individual Plant Examination (IPE). Table 3.3-5 is the disposition of these five types of human interactions in the DAEC IPE and the bases for the disposition. Also addressed are any supplementary requirements imposed by the Generic Letter 88-20 or the IPE Guidance Document NUREG-1335.

The focus of the present evaluation is on qualitative and quantitative assessment of DAEC IPE post-accident (Type C and E) and pre-accident (Type A) human actions. Key human actions in the DAEC IPE have been assessed using operator interviews, the latest revision of the Emergency Operating Procedures (EOPs), the Annunciator Response Procedures (ARPs), and plant specific information as input to available human error rate models.

3.3.3.2 Post-Accident Human Actions

Table 3.3-6 provides a list of the DAEC post-accident human actions that are analyzed and quantified for the IPE model. As the "Quantification Method" column in this table shows, each of these post-accident human actions is quantified either through the use of a detailed Human Reliability Analysis (HRA) or one of several screening techniques. The selection of a quantification methodology (screening or a detailed HRA) is based on an assessment of the "importance" of the human action to either core damage frequency or public risk.

The RMIEP methodology is used as the method of choice in quantifying the post-accident human actions for which a detailed analysis was considered appropriate. The RMIEP methodology is considered to be an acceptable HRA tool and one that produces consistent results when used within its limitations.

The use of the EPRI model for quantification of some of the post-accident human actions was necessary. This is because the time-reliability curves for the human action "groups" used in the RMIEP model do not represent some of the post-accident human actions needed for the DAEC IPE; in addition, the time frames for these human actions are beyond the time frames for which data was available to support the time-reliability curves in the RMIEP model.

A number of post-accident human actions, that are quantified by screening techniques, are quantified using the simple quantification methodology provided in EPRI Research Project 3206-03 (Modeling of Recovery Actions in PRAs). These post-accident human actions are judged to be of sufficient importance with respect to their impact on the overall model that their quantification involved a screening technique that is based on plant-specific information obtained during the interview sessions with the plant operators. Among the remaining post-accident human actions that are quantified by a screening technique, there are a number of human actions that are judged to be of less "importance" with respect to core damage frequency or public risk, and therefore are quantified using ASEP post-accident screening HRA.

To summarize, human actions considered at DAEC were evaluated with regard to their "importance". Those that were deemed to be of relatively high importance were evaluated using detailed models. The RMIEP model is used as a first preference as long as the human action and time frames required are within the bounds of the RMIEP model. When it was deemed that a human action was outside the bounds of the RMIEP model then EPRI methodology was used. Human actions that were determined to be of lesser importance such that detailed models did not have to be performed, but of significant concern that the ASEP screening was considered inappropriate, used the approach specified by EPRI Research Project 3206-03. All remaining human actions, that were considered, used the ASEP screening approach. The details of the breakdown of the various human actions are given in the following sections and associated tables.

3.3.3.3 Pre-Accident Human Actions

Table 3.3-7 provides a list of the DAEC pre-accident human actions that are quantified. All human actions listed are quantified using ASEP pre-accident screening HRA. The quantification of each of these human actions is based on plant-specific information obtained during the interview session with the plant operators.

3.3.3.4 Quantification of Post-Accident Human Actions

Table 3.3-8 provides the human error probabilities for the DAEC post-accident human actions that are quantified using a detailed HRA model (RMIEP). The variables used in quantification of these human actions are provided in Table 3.3-9.

Table 3.3-10 provides the human error probabilities for the DAEC post accident actions that were also quantified using the EPRI model. The variables used in quantification of these human actions are provided in Table 3.3-11.

Table 3.3-12 provides the human error probabilities for the DAEC post-accident human actions that are quantified using the screening quantification method provided in EPRI Research Project 3206-03. Tables 3.3-13 and 3.3-14 provide the bases for quantification of these human actions. The quantification bases for these human actions consists of the information obtained from the operating staff during interview sessions.

Table 3.3-15 provides the human error probabilities for the DAEC post-accident human actions that are quantified using ASEP post-accident screening HRA. The variables that were used in quantification of the HEPs for these human actions are provided in Table 3.3-16.

3.3.3.5 Quantification of Pre-Accident Human Actions

Table 3.3-17 provides the human error probabilities for DAEC pre-accident operator actions quantified using ASEP pre-accident screening HRA. The initial basic HEP and the recovery factors used in quantifying these human actions are shown in Table 3.3-18. The bases for the recovery factors, provided in Table 3.3-18, are given in Table 3.3-19.

Table 3.3-5

HRA SUMMARY TABLE

HRA ACTIONS	METHODS OF HRA		
	Alternative Available	Additional NRC Guidance	Method Chosen
Type A: Test and Maintenance	· Use of Operating Experience Data · WASH-1400 estimates · THERP Analysis NUREG/CR-1278 · ASEP Pre-accident Screening HRA (NUREG/CR -4772)	None	ASEP pre-accident screening HRA is used for quantifying Type A human actions. The quantification is based on the information obtained from plant operators and the information contained in the plant procedures.
Type B: Actions Causing Initiating Events	· Operating Experience · Rare Initiators from Master Logic Diagram (THERP)	None	Initiating event frequencies are based upon operating experience data which include as one of the root cause contributors human errors. These are included directly in the data.
Type C: Procedural Actions During Course of Accident	· SWAIN: Time dependent performance shaping factors · OAT: Time reliability curve · HCR: Time reliability curve (SHARP) (EPRI RP 2847-1 December 1989.) · SLIM: Subjective evaluation by expert opinion averaging . IREP: Quantitative Guidance · Direct Numerical Estimation · Paired Comparison Technique · Simulator Data (EPRI or RMIEP) · Performance shaping factor methodology · ASEP post-accident screening HRA (NUREG/CR -4772) · Simple quantification method outlined in EPRI Research Project 3206-03 (Modeling of Recovery Actions in PRAs).	Quantitative evaluation of operator actions taken without a written procedure are difficult because: · They are dependent upon training interpretation or individual memories which may change with time, · They may not be clear and recalled under high stress conditions, and · The mechanical implementation may not be easily carried out without a written procedure. Therefore, consistent with the guidance in the NUREG-1335 IPE guidance document (Pg. C-19), the failure probability (in the base case analysis) for those actions without written procedures is taken to be 1.0.	A detailed review of the DAEC procedures is preformed to establish the bases for the quantitative evaluation. The quantification of the Type C human actions are performed using the following models: · RMIEP data compiled by the NRC from LaSalle simulator data. · ASEP post-accident screening HRA, and · The simple quantification method outlined in EPRI research project 3206-03 (Modeling of Recovery Actions in PRAs).

Table 3.3-5 (con't)

HRA SUMMARY TABLE

HRA ACTIONS	METHODS OF HRA		
	Alternative Available	Additional NRC Guidance	Method Chosen
Type D: Actions Leading to Inappropriate Actions	Confusion Matrix	The NRC in NUREG-1335 [pg. C-19] has indicated that for the IPE it is not the intent of the analysis program to "break new ground" regarding the development of a methodology for modeling errors of commission. Therefore, consistent with this guidance errors of commission are treated in a manner similar to past published PRAs, i.e., those that result from misleading instrumentation are treated, those which have no apparent cause and involve a purposeful violation of procedures to create a new problem are not included.	Only those actions caused by failures of instrumentation are included in the evaluation.
Type E: Recovery	· SWAIN: Time dependent performance shaping factors · OAT: Time reliability curve · HCR: Time reliability curve (SHARP) (EPRI RP 2847-1, December 1989) · SLIM: Subjective evaluation by expert opinion averaging · IREP: Quantitative Guidance · Direct Numerical Estimation · Paired Comparison Technique · Simulator Data (EPRI or RMIEP) · Performance Shaping Factor Methodology · ASEP HRA Model (NUREG/CR-4772) · Simple quantification method outline in EPRI research project 3206-03 (Modeling of Recovery Actions in PRAs.)	Quantitative evaluation of operator actions taken without a written procedure are difficult because: · They are dependent upon training interpretation or individual memories which may change with time · They may not be clear or recalled under high stress conditions, and · The mechanical implementation may not be easily carried out without a written procedure. Therefore, consistent with the guidance in the NUREG-1335 IPE guidance document (Pg. C-19) the failure probability (in the base case analysis) for those actions without written procedures is taken to be 1.0.	A detailed review of the DAEC procedures is preformed to establish the bases for the quantitative evaluation. The quantification of the Type C human actions are performed using the following models: · RMIEP data compiled by the NRC from LaSalle simulator data. · ASEP post-accident screening HRA, and · The simple quantification method outlined in EPRI research project 3206-03 (Modeling of Recovery Actions in PRAs).

Table 3.3-6

LIST OF POST-ACCIDENT (TYPE C) OPERATOR ACTIONS FOR DAEC PRA

PRA Sequence Function	Operator Action Description	Model Designator	Quantification Method: Detailed	Quantification Method: Screening	Basis for Quantification: Operator Interviews	Basis for Quantification: Simulator Observation
RPV Depressurization	Operator Fails to Bypass Group 3 Isolation for Backup N2	OOPPF-GROUP3BYP	X		X	X
	Operator Inhibits ADS	O-ADS-INHIBIT	X		X	
	Operator fails to Inhibit ADS					
	- Transient[3]	Not Used				
	- ATWS with High Pressure Makeup	OOPAF-ADSBLOCK1	X		X	X
	- ATWS with Insufficient High Pressure Makeup	OOPAF-ADSBLOCK2	X		X	
	Operator Fails to Manually Initiate Emergency Depressurization					
	- Transients and Small LOCAs	OOPAF-MANUAL-DEP	X		X	
	- Medium LOCAs	OOPAF-MANUAL-DEP	X		X	
	- ATWS (Heat Capacity Temperature Limit exceeded)	OOPAF-MANUAL-DEP	X		X	
	- ATWS (level below -30")	OOPAF-MANUAL-DEP	X		X	
	- In-vessel core degradation (CET Node OP) (Level 2)	OOPAF-LVL2-ADS	X		X	
Containment Heat Removal	Operator Fails to Reopen MSIVs and Restore Condenser for Containment Heat Removal					
	- Non-ATWS	COPAF-CD03	X		X	
	- ATWS	Not Used	X		X	
	-- SLC Initiated, level control implemented					
	-- SLC not initiated, level control implemented					
	-- SLC not initiated, no level control					
	Operator Fails to Control Pressure with Bypass Valve	COPAF-PRES-CNTL	X		X	
	Operator Fails to Bypass MSIV Isolation Interlocks (ATWS)	Q (MC:AVAIL)	X		X	

Table 3.3-6

LIST OF POST-ACCIDENT (TYPE C)
OPERATOR ACTIONS FOR DAEC PRA

PRA Sequence Function	Operator Action Description	Model Designator	Quantification Method		Basis for Quantification	
			Detailed	Screening	Operator Interviews	Simulator Observation
	Operator Fails to Initiate RHR in Suppression Pool Cooling (SPC) Mode - Transient and LOCAs - ATWS	LOPPF-TORUS-COOL LOPPF-TORUS-COOL	X X		X X	X X
	Operator Fails to Manually Locally Open MO1947/MO 2046 in RHRSW LOOP BD and AC for torus cooling	WOPAF-MO1947 WOPAF-MO2046		X		
Containment Heat Removal (con't)	Operator Fails to Locally Manually Close Breaker to Start Pump (RHRSW/RHR/GSW/CS)	SOPAF-LOCALSTART WOPAF-LOCALSTART LOPAF-LOCALSTART GOPAF-LOCALSTART POPAF-LOCALSTART		X		
	Operator Fails to Vent the Torus - Transients or LOCA - ATWS	VOPAF-TORUS-VENT	X X		X X	
	Operator Fails to Initiate Vent (Wetwell) Per Procedure (CET Node GV) (Level 2)	NOPPF-GASVENT	X		X	
	Staff does not check for H_2/O_2 indication (CET Node GV) (Level 2)	UOPAF-H2O2INDN		X		
High Pressure Injection (FW/Condensate or HPCI/RCIC Operation)	Operator failure to Manually Initiate System After Auto Actuation Signal Failure (Feedwater/Condensate or HPCI/RCIC)					

Table 3.3-6

LIST OF POST-ACCIDENT (TYPE C)
OPERATOR ACTIONS FOR DAEC PRA

PRA Sequence Function	Operator Action Description	Model Designator	Quantification Method		Basis for Quantification	
			Detailod	Screening	Operator Interviews	Simulator Observation
High Pressure Injection (FW/Condensate or HPCI/RCIC Operation) [continued]	HPCI/RCIC - Small LOCA or transient with MSIVs closed - Medium LOCA - Large LOCA	 HOPAF-HPCI/RCIC HOPAF-HPCI/RCIC HOPAF-HPCI/RCIC	X		X	
	Feedwater/Condensate - MSIV closure transient - Small or medium LOCA with HPCI/RCIC failure - Large LOCA condensate initiation, control.	 FW: OOPAF-FW99 FW: QOPAF-FW99 FW: OOPAF-FW99	X		X	
	Operator Fails to Manually Transfer HPCI/RCIC Suction From CST to Torus on Low CST Level	HOPAF-15	X		X	
	Operator Fails to Prevent Transfer of HPCI Suction From CST to Torus on High Torus Level[2]	Not Used	X		X	X
	Operator Fails to Shutoff HPCI or RCIC if required	HOPAF-14 HOPAF-HC14		X		
	Operator Fails to Replenish CST Supply	HOPAF-5		X		
	Operator Fails to Close HPCI/RCIC Minimum Flow Valve	ROPAF-RCIC8 HOPAF-MO2318		X		
	Operator Inadvertently Opens RCIC Minimum Flow Valve MO2510	ROPAF-RCIC7		X		
	Operator Fails to Ventilate HPCI and RCIC Rooms (under SBO)	HOPAF-OPEN-DOORS	X		X	
	Operator Fails to Reduce FW Flow Given an Isolation ATWS	Q-OP-RUNBACK		X		
	Operator Fails to Control Reactor Water Level Following SCRAM	OOPAF-LEVEL		X		
	Operator Fails to Open Valve MO1546 in Control Room	OOPAF-FW05		X		

Table 3.3-6

LIST OF POST-ACCIDENT (TYPE C) OPERATOR ACTIONS FOR DAEC PRA

PRA Sequence Function	Operator Action Description	Model Designator	Quantification Method		Basis for Quantification	
			Detailed	Screening	Operator Interviews	Simulator Observation
High Pressure Injection (FW/Condensate or HPCI/RCIC Operation) [continued]	Operator Fails to Open Makeup Line Bypass Valve from CST to Hotwell	COPAF-CD04	X		X	
	Operator Fails to Open Bypass Valve in Control Room	COPAF-CD02		X		
RPV Coolant Makeup (CRD Operation)	Operator Fails to Align CRD for Injection - At transient initiation (enhanced flow) - Following containment failure	 DOPAF-1P209B DOPAF-1P209B	 X	 X	 X	 X
	Operator Fails to Restart CRD Pump 1P209A after Load Shed	DOPAF-1P209A		X		
	Operator Fails to Locally Close Stored Energy Breaker for CRD Pump Start	DOPAF-LOCASTART		X		
	Operator Fails to Reset SCRAM Signal (CRD)	DOPAF-SCRAMRESET		X		
	Operator Fails to Align Alternate Suction (CRD)	DOPAF-SUCT		X		
	Operator Fails to Open V17-0014 (CRD)	DOPAF-V17-0014		X		
Low Pressure Injection	Operator Fails to Restore LPI Post RPV Depressurization (ATWS)	L (LEVEL:NOT:LOW)	X		X	
	Operator Fails to Prevent Overfilling the RPV, Given Automatic Depressurization, Without LPI Inhibit (ATWS) - Inadvertent ADS[4] - Emergency depressurization (pumps successfully terminated) - Inadvertent or slow depressurization (SORV)	 L (LEVEL:NOT:HI) L (LEVEL:NOT:HI) L (LEVEL:NOT:HI)	X		X	
	Operator Fails to Manually Initiate LPCI/Core Spray Following Auto Initiation Failure	LOPPF-LPCI-INIT	X		X	
	Operator Fails to Open LPCI Injection Valves After Failure of Automatic Signal	LOPPF-LPCIINJECT		X		

Table 3.3-6

LIST OF POST-ACCIDENT (TYPE C) OPERATOR ACTIONS FOR DAEC PRA

PRA Sequence Function	Operator Action Description	Model Designator	Quantification Method		Basis for Quantification	
			Detailed	Screening	Operator Interviews	Simulator Observation
Low Pressure Injection [continued]	Low Pressure System Manually Shutoff on High Level	LOPAF-LPCI3		X		
	Operator Distracted for 2 to 6 hours Fails to Restart Low Pressure Systems	LOPAF-LPCI4	X			
	Operator Fails to Follow Procedure and Restart Low Pressure Systems	LOPPF-LPCI5	X			
	Operator Fails to Perform Actions to Open Flowpath From CST to Core Spray	SOPAF-LPCS-CST		X		
	Operator Fails to Recognize the Need for Core Spray	SOPIF-INATENTION		X		
	Operator Fails to Locally Align RHRSW Injection to RPV (Action occurs in rector building corner rooms and torus room)	WOPPF-RHRSWINJ		X		
	Operator Fails to Initiate RHRSW	WOPAF-RHRSW		X		
	Operator Closes CV 4910A or B and prevents flow through alternate flow path to the stilling basin	WOPFF-WS04 WOPFF-WS03		X		
	Lack of Operator Action (to Open Manual Valve V46-0009)	GOPAF-469		X		
	Operator Fails to Manually Locally Close MO1947/MO2046 in RHRSW System for RPV Injection	WOPAF-MO1947-C WOPAF-MO2046-C		X		

Table 3.3-6

LIST OF POST-ACCIDENT (TYPE C)
OPERATOR ACTIONS FOR DAEC PRA

PRA Sequence Function	Operator Action Description	Model Designator	Quantification Method		Basis for Quantification	
			Detailed	Screening	Operator Interviews	Simulator Observation
Low Pressure Injection [continued]	Operator Fails to Align Alternate Injection Systems (RHRSW/GSW/ESW/Well Water) for Injection <u>Core Damage Prevention</u> - Loss of makeup sequences - Large LOCAs - ATWS (Loss of injection) <u>In-vessel Recovery</u> - Loss of makeup (Level 2) - Large LOCAs (Level 2) - ATWS (Level 2) <u>Shell Failure Prevention</u> - Loss of makeup sequences (Level 2) - Large LOCAs (Level 2) - ATWS (Loss of injection) (Level 2) <u>Prevention of Containment Overtemperature Failure</u> - All sequences (Level 2)	WOPAF-ALTINJ COPAF-ALTINJ	X		X	
	Operator Fails to Manually Open Valve CV4915 or CV4914 (Water Supply Pumps Discharge Valves)	WOPAF-WS02		X		
	Operator Intervenes and Terminates Injection (CET Node RX) (Level 2)	XOPZF-TERMINJ		X		
	Alignment Not Completed Prior to Containment Failure (CET Nede RX) (Level 2)	WOPAF-ALGNTME		X		
	Operator Fails to Recover Injection Before RPV Melt (CET Node RX) (Level 2)	XOPAF-RECVRINJ		X		

Table 3.3-6

LIST OF POST-ACCIDENT (TYPE C) OPERATOR ACTIONS FOR DAEC PRA

PRA Sequence Function	Operator Action Description	Model Designator	Quantification Method		Basis for Quantification	
			Detailed	Screening	Operator Interviews	Simulator Observation
	Operator Restores Coolant Injection After Control Rods Are Melted (CET Node CZ) (Level 2)	XOPAF-RESTRINJ		X		
	Operator Fails to Recover Low Pressure Systems (CET Node SI) (Level 2)	XOPAF-LPSYS		X		
	Operator Fails to Recover Low Pressure Systems (CET Node TD) (Level 2)	XOPAF-TDCONTFL XOPAF-TDLPREC		X		
Sprays	Operator Fails to Override Drywell Spray Interlock	LOPIF-DW-INTRLOK		X		
	Operator Fails to Initiate Drywell Spray - Small LOCAs - Medium LOCA, IORV, SORV - Large LOCAs	LOPPF-DW-SPRAY		X		
	Operator Fails to Initiate Torus Spray	LOPPF-TORUS-SPRY		X		
Reactivity Control (ATWS Response)	Operator Fails to Initiate SLC Boron Injection - Condenser Unavailable (within 6 min.) - Condenser Unavailable (within 20 min.) - Condenser Available (within 40 min.)	BOPAF-SLC-EARLY BOPAF-SLC-LATE-3 BOPAF-SLC-LATE-4	X		X	
	Operator Fails to Initiate SLC Via Alternate Boron Injection Path (RWCU)	ROPAF-SLC-INIT				
	Operator Fails to Lower RPV Water Level to TAF for Power Control OR Operator Fails to Restore RPV Level at End of SLC Injection or When Reactor is Shutdown	L (CONTROLLED)	X		X	
Containment Flood	Operator Fails to Implement Contingency EOP to Flood Containment (CET Node FC) (Level 2)	XOPPF-CONTFLD	X		X	

Table 3.3-6

LIST OF POST-ACCIDENT (TYPE C) OPERATOR ACTIONS FOR DAEC PRA

PRA Sequence Function	Operator Action Description	Model Designator	Quantification Method		Basis for Quantification	
			Detailed	Screening	Operator Interviews	Simulator Observation
Containment Flood [continued]	Operator Fails to Close Wetwell Vent (CET Node FC) (Level 2)	NOPAF-WWVENT		X		
	Operator Suspends Flooding Based on Erroneous Indication (CET Node FD) (Level 2)	MOPCF-INDICTN		X		
	Operator Fails to Implement Drywell Vent Path (CET Node FD) (Level 2)	MDWPF-DWVENT		X		
	Operator Fails to Vent RPV (Level 2)	XOPPF-RPV-VENT	X		X	
	Operator Fails to Initiate Venting (Drywell) Given Indication (CET Node CV) (Level 2)	CV-02-01	X		X	
Containment Isolation	Operator Fails to Isolate Path Given Isolation Signal Failure (CET Node IS) (Level 2)	XOPAF-DWPURGE XOPAF-DWVENT XOPAF-MKUPN2DW XOPAF-EOUIPDRN XOPAF-FLOORDRN XOPAF-WWPURGE XOPAF-WWVENT XOPAF-MKUPN2WW		X		
DC Power	Operator Fails to Shed Loads from DC Buses (Station Blackout)	SH (OP:LD:SHED)	X		X	
Diesel Generator Cooling	Operator Fails to Cross-tie ESW Trains	EOPAF-CROSS-TIE		X		
	Operator Fails to Manually Bleeds Air Supply to CV2080/CV2081 for ESW	EOPAF-CV2080 EOPAF-CV2081		X		
	Operator Fails to Initiate ESW (for diesel cooling)	EOPAF-ES10		X		
	Operator Fails to Bypass Filter 1S089B/1S089A for ESW	EOPAF-V46-0019 EOPPF-V46-0024		X		
Control Building Ventilation	Operator Fails to Establish Alternate Control Room Building Cooling	POPAF-ALTCOOLING	X			

Notes to Table 3.3-6

(1) Maximum Makeup to CST is only 90 gpm.
(2) Current model does not include this effect. High torus temperature failure of HPCI/RCIC is not treated.
(3) This action is used to derive O-ADS-INHIBIT, but is not used explicitly in the model.
(4) Treated in event tree logic.
(5) The designator HOPAF-2 is replaced with HOPAF-HPCI/RCIC.

Table 3.3-7

LIST OF PRE-ACCIDENT (TYPE A) OPERATOR ACTIONS FOR DAEC PRA

PRA Sequence Function	Operator Action Description	Model Designator	Quantification Method		Basis for Quantification	
			Detailed	Screening	Operator Interviews	Simulator Observation
High Pressure Injection	HPCI auto reset not reset	HOPAF-1		X		
	Operator fails to take action to empty drain pot	HOPAF-IN-DRN-POT ROPAF-IN-DRN-POT		X		
	RCIC mechanical overspeed trip not reset following test and maintenance	ROPAF-MECH-OVSPD		X		
Low Pressure Injection	Operator fails to notice low basin water level	WOPIF-RWS-START		X		
Reactivity Control (ATWS Response)	Operator fails to respond to low level indications in the SLC tank	BOPAF-SL10		X		
	Tour by operator fails to uncover low level in tank (once a day)	BOPPF-SL09		X		
Level/Pressure Instrumentation	Miscalibration of level instrumentation used to initiate the HPCI/RCIC/LPCI/CS pumps	ILIMF-RXLVL		X		
	Miscalibration of pressure instrumentation used to initiate the HPCI/RCIC/LPCI/CS/Recirc. pumps	ILIMF-DWPRS I-LOW-PRES-PERM IPIMF-RCPA IPIMF-RCPB I-LPCI-SELECT ILIMF-LOOPSEL		X		

Table 3.3-8

RMIEP METHOD
HEPs FOR POST-ACCIDENT OPERATOR ACTIONS QUANTIFIED USING A DETAILED HRA MODEL

HUMAN ACTION DESCRIPTION	HEP	DESIGNATOR
Operator Fails to Bypass Group 3 Isolation for Backup N2	2.7E-3	OOPPF-GROUP3BYP
Operator Inhibits ADS	1.0	O-ADS-INHIBIT
Operator Fails to Inhibit ADS		
- Transient[4]	5.8E-3	Not Used
- ATWS with High Pressure Makeup	1.4E-2	OOPAF-ADSBLOCK1
- ATWS with Insufficient High Pressure Makeup	3.2E-1	OOPAF-ADSBLOCK2
Operator Fails to Manually Initiate Emergency Depressurization		
- Transients and Small LOCAs	2.2E-3[5]	OPAF-MANUAL-DEP
- Medium LOCAs	2.2E-3[5]	OPAF-MANUAL-DEP
- ATWS (Heat Capacity Temperature Limit Exceeded)	5.6E-2[5]	OPAF-MANUAL-DEP
- ATWS (level below -30")	2.2E-3[5]	OPAF-MANUAL-DEP
- In-vessel Core Degradation (CET Node OP) (Level 2)	2.2E-3[5]	OOPAF-LVL2-ADS
Operator Fails to Reopen MSIVs and Restore Condenser for Containment Heat Removal		
- Non-ATWS	1.7E-3	COPAF-CD03
- ATWS		Not Used
-- SLC initiated, level control implemented	1.0	
-- SLC not initiated, level control implemented	1.0	
-- SLC not initiated, no level control	1.0	
Operator Fails to Control Pressure with Bypass Valve	3.1E-3	COPAF-PRES-CNTL
Operator Fails to Bypass MSIV Isolation Interlocks (ATWS)	7.2E-1	Q (MC:AVAIL)

Table 3.3-8

RMIEP METHOD
HEPs FOR POST-ACCIDENT OPERATOR ACTIONS QUANTIFIED USING A DETAILED HRA MODEL

HUMAN ACTION DESCRIPTION	HEP	DESIGNATOR
Operator Fails to Initiate RHR in Suppression Pool Cooling (SPC) Mode - Transients and LOCAs - ATWS	 1E-6[8] 1.6E-2	 LOPPF-TORUS-COOL LOPPF-TORUS-COOL
Operator Fails to Vent the Torus - Transients or LOCAs or ATWS (with SLC) - ATWS (without SLC)	 2.2E-3 1.0	 VOPAF-TORUS-VENT VOPAF-TORUS-VENT
Operator Fails to Initiate Vent (Wetwell) Per Procedure (CET node GV) (Level 2)	2.2E-3[5]	NOPPF-GASVENT
Operator Fails to Manually Initiate System After Auto Actuation Signal Failure (Feedwater/Condensate or HPCI/RCIC) <u>HPCI/RCIC</u> - Small LOCA or transient with MSIVs Closed - Medium LOCA - Large LOCA <u>Feedwater/Condensate</u> - MSIV closure transient - Small or medium LOCA with HPCI/RCIC failure - Large LOCA condensate initiation, control	 3.1E-3 3.1E-3 N/A 5.2E-3 5.2E-3 3.3E-1	 HOPAF-HPCI/RCIC[6] HOPAF-HPCI/RCIC[6] HOPAF-HPCI/RCIC FW: QOPAF-FW99 FW: QOPAF-FW99 FW: QOPAF-FW99

Table 3.3-8

RMIEP METHOD
HEPs FOR POST-ACCIDENT OPERATOR ACTIONS QUANTIFIED USING A DETAILED HRA MODEL

HUMAN ACTION DESCRIPTION	HEP	DESIGNATOR
Operator Fails to Manually Transfer HPCI/RCIC Suction from CST to Torus on Low CST Level	7.7E-2	HOPAF-15
Operator Fails to Prevent Transfer of HPCI Suction from CST to Torus on High Torus Level(2)	8.4E-3	Not Used
Operator Fails to Ventilate HPCI and RCIC Rooms (Under SBO)	3.1E-3	HOPAF-OPEN-DOORS
Operator Fails to Open Makeup Line Bypass Valve from CST to Hotwell	3.1E-3	COPAF-CD04
Operator Fails to Align CRD for Injection - At transient initiation (enhanced flow) - Following containment failure	 1.0 (1)	 DOPAF-1P209B DOPAF-1P209B
Operator Fails to Restore LPI Post RPV Depressurization (ATWS)	1.6E-2	L (LEVEL:NOT:LOW)
Operator Fails to Prevent Overfilling the RPV, Given Automatic Depressurization Without LPI Inhibit (ATWS) - Inadvertent ADS(3) - Emergency depressurization (pumps successfully terminated) - Inadvertent or slow depressurization (SORV)	 1.0 1.5E-1 4.6E-1	 L (LEVEL:NOT:HI) L (LEVEL:NOT:HI) L (LEVEL:NOT:HI)
Operator Fails to Initiate LPCI/Core Spray Following Auto Initiation Failure	3.6E-3(5)	LOPPF-LPCI-INIT
Operator Distracted for 2 to 6 Hours Fails to Restart Low Pressure Systems	3.6E-3(5)	LOPAF-LPCI4
Operator Fails to Follow Procedure and Restart Low Pressure Systems	3.6E-3(5)	LOPPF-LPCI5

Table 3.3-8

RMIEP METHOD
HEPs FOR POST-ACCIDENT OPERATOR ACTIONS QUANTIFIED USING A DETAILED HRA MODEL

HUMAN ACTION DESCRIPTION	HEP	DESIGNATOR
Operator Fails to Align Alternate Injection Systems (RHRSW/ESW/GSW/Well Water System) for Injection		WOPAF-ALTINJ COPAF-ALTINJ(7)
Core Damage Prevention		
- Loss of Makeup Sequences	1.4E-1	
- Large LOCAs	2.5E-1	
- ATWS (Loss of Injection)	2E-1	
In-vessel Recovery		
- Loss of makeup (Level 2)	7.5E-2	
- Large LOCAs	8.7E-2	
- ATWS (Level 2)	8.3E-2	
Shell Failure Prevention		
- Loss of Makeup Sequences (Level 2)	3.8E-2	
- Large LOCAs (Level 2)	5.1E-2	
- ATWS (Loss of Injection) (Level 2)	4.4E-2	
Prevention of Containment Overtemperature Failure		
- All Sequences (Level 2)	1.4E-2	

Table 3.3-8

RMIEP METHOD
HEPs FOR POST-ACCIDENT OPERATOR ACTIONS QUANTIFIED USING A DETAILED HRA MODEL

HUMAN ACTION DESCRIPTION	HEP	DESIGNATOR
Operator Fails to Initiate SLC Boron Injection - Condenser Unavailable (within 6 min.) - Condenser Unavailable (within 20 min.) - Condenser Available (within 40 min.)	 1.1E-1 8.3E-3 3.1E-3	 BOPAF-SLC-EARLY BOPAF-SLC-LATE-3 BOPAF-SLC-LATE-4
Operator Fails to Lower RPV Water Level to TAF for Power Control <u>OR</u> Operator Fails to Restore RPV Level at End of SLC Injection or When Reactor is Shutdown	1.5E-2	L (CONTROLLED)
Operator Fails to Implement Contingency EOP to Flood Containment (CET Node FC) (Level 2)	3.2E-3	XOPPF-CONTFLD
Operator Fails to Vent RPV (Level 2)	3.1E-3	XOPPF-RPV-VENT
Operator Fails to Initiate Venting Given Indication (CET Node CV) (Level 2)	6.6E-3[5]	CV-02-01
Operator Fails to Shed Loads From DC Buses (Station Blackout)	5E-2	SH (OP:LD:SHED)
Operator Fails to Establish Alternate Control Room Building Cooling	3.1E-3	POPAF-ALTCOOLING

Notes to Table 3.3-8

(1) See Table 3.3-16 for quantification.

(2) Current model does not include this effect. High torus temperature failure of HPCI/RCIC is not treated.

(3) Treated in event tree logic.

(4) This action is used to derive O-ADS-INHIBIT, but is not used explicitly in the model.

(5) Not used as the recommended value.

(6) The designator HOPAF-2 replaced with HOPAF-HPCI/RCIC.

(7) COPAF-ALTINJ is the portion of WOPAF-ALTINJ that is the "operator fails to recognize need for alternate injection" (this is proceduralized) and is 6E-4.

(8) Although the HRA modeling supports a 1E-6 failure rate, a 1E-4 number was used in the quantification of the DAEC PRA.

Table 3.3-9

RMIEP METHOD
VARIABLES ASSESSED IN QUANTIFICATION OF HEPs FOR POST-ACCIDENT (TYPE C) OPERATOR ACTIONS USING A DETAILED HRA MODEL

Description of Action	Crew Recovery Action Group	Maximum Time for Crew to Complete Action (T_M)	Time for Crew to Perform Action (T_A)	Time for Crew to Diagnose Action (T_D)	Median Failure Probability $P(ND)_{med}$ at T_D	Upper Error Factor (UEF)	Point Estimate for Diagnose Failure Probability $P(ND)_{mean}$ at T_D	Failure Probability for Accomplishing Recovery Action P(NA)	Total Failure Probability for Recovery Action P(NR)
Operator Fails to Bypass Group 3 Isolation for Backup N2	8	4 hrs.[6]	30 sec.	239.5 min.	6.5E-4	10	1.7E-3	1E-3	2.7E-3
Operator Inhibits ADS	---	---	---	---	---	---	---	---	1.0
Operator Fails to Inhibit ADS									
- Transient[23]	1	23 min.[7]	15 sec.	22.75 min.	3E-3	5	4.8E-3	1E-3	5.8E-3
- ATWS with High Pressure Makeup	1	16 min.[8]	30 sec.	15.5 min.	9.3E-3	3.7	1.3E-2	1E-3	1.4E-2
- ATWS with Insufficient High Pressure Makeup	1	3 min.[9]	30 sec.	2.5 min.	3.2E-1	1.3	3.2E-1	1E-3	3.2E-1
Operator Fails to Manually Initiate Emergency Depressurization									
- Transients and Small LOCAs	2	64 min.[1]	15 sec.	63.75 min.	8.1E-4	10	2.2E-3	9E-6	2.2E-3
- Medium LOCAs	2	55 min.[2]	15 sec.	54.75 min.	8.1E-4	10	2.2E-3	9E-6	2.2E-3
- ATWS (Heat Capacity Temperature Limit Exceeded)	2	16 min.[3]	30 sec.	15.5 min.	2.1E-2	10	5.6E-2	9E-6	5.6E-2
- ATWS (level below -30")	2	43 min.[3]	30 sec.	42.5 min.	8.1E-4	10	2.2E-3	9E-6	2.2E-3
- In-vessel Core Degradation (CET Node OP) (Level 2)	2	87 min.[17]	30 sec.	86.5 min.	8.1E-4	10	2.2E-3	9E-6	2.2E-3

Table 3.3-9

RMIEP METHOD
VARIABLES ASSESSED IN QUANTIFICATION OF HEPs FOR POST-ACCIDENT (TYPE C) OPERATOR ACTIONS USING A DETAILED HRA MODEL

Description of Action	Crew Recovery Action Group	Maximum Time for Crew to Complete Action (T_M)	Time for Crew to Perform Action (T_A)	Time for Crew to Diagnose Action (T_D)	Median Failure Probability $P(ND)_{med}$ at T_D	Upper Error Factor (UEF)	Point Estimate for Diagnose Failure Probability $P(ND)_{mean}$ at T_D	Failure Probability for Accomplishing Recovery Action P(NA)	Total Failure Probability for Recovery Action P(NR)
Operator Fails to Reopen MSIVs and Restore Condenser for Containment Heat Removal									
- Non-ATWS	8	> 10 hrs.[10]	10 min.	590 min.	6.5E-4	10	1.7E-3	1E-5	1.7E-3
- ATWS									
-- SLC initiated, level control implemented	8	15 min.[18]	1 hr.	0 min.	1.0	1.0	1.0	6E-3	1.0
-- SLC not initiated, level control implemented	8	15 min.[18]	1 hr.	0 min.	1.0	1.0	1.0	6E-3	1.0
-- SLC not initiated, no level control	8	10 min.[18]	1 hr.	0 min.	1.0	1.0	1.0	6E-3	1.0
Operator Fails to Control Pressure with Bypass Valve	1	> 10 hrs.	30 sec.	599.5 min.	1E-3	7.4	2.1E-3	1E-3	3.1E-3
Operator Fails to Bypass MSIV Isolation Interlocks (ATWS)	12	15 min.	30 sec.	14.5 min.	2.7E-1	10	7.2E-1	6E-3	7.2E-1
Operator Fails to Initiate RHR in Suppression Pool Cooling (SPC) Mode									
- Transients and LOCAs	---	---	---	---	---	---	---	---	1E-6[11]
- ATWS	1	15 min.[15]	30 sec.	14.5 min.	1.1E-2	3.45	1.5E-2	1E-3	1.6E-2
Operator Fails to Vent the Torus									
- Transients or LOCAs or ATWS (with SLC)	1	5 hrs.[3]	30 sec.	299.5 min.	1E-3	7.4	2.1E-3	1E-4	2.2E-3
- ATWS (without SLC)	1	0 min.[3]	30 sec.	0 min.	1.0	1.0	1.0	1E-4	1.0
Operator Fails to Initiate Vent (Wetwell) Per Procedure (CET Node GV) (Level 2)	1	5 hrs.[3]	30 sec.	299.5 min.	1E-3	7.4	2.1E-3	1E-4	2.2E-3

Table 3.3-9

RMIEP METHOD
VARIABLES ASSESSED IN QUANTIFICATION OF HEPs FOR POST-ACCIDENT (TYPE C) OPERATOR ACTIONS USING A DETAILED HRA MODEL

Description of Action	Crew Recovery Action Group	Maximum Time for Crew to Complete Action (T_M)	Time for Crew to Perform Action (T_A)	Time for Crew to Diagnose Action (T_D)	Median Failure Probability $P(ND)_{med}$ at T_D	Upper Error Factor (UEF)	Point Estimate for Diagnose Failure Probability $P(ND)_{mean}$ at T_D	Failure Probability for Accomplishing Recovery Action P(NA)	Total Failure Probability for Recovery Action P(NR)
Operator Fails to Manually Initiate System After Auto Actuation Signal Failure (Feedwater/Condensate or HPCI/RCIC)									
HPCI/RCIC									
- Small LOCA or transient with MSIVs Closed	1	66 min.(3)	30 sec.	65.5 min.	1E-3	7.4	2.1E-3	1E-3	3.1E-3
- Medium LOCA	1	35 min.(3)	30 sec.	34.5 min.	1E-3	7.4	2.1E-3	1E-3	3.1E-3
- Large LOCA	1	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Feedwater/Condensate									
- MSIV closure transient	1	28 min.(3)	30 sec.	27.5 min.	1.7E-3	6.5	3.2E-3	2E-3	5.2E-3
- Small or medium LOCA with HPCI/RCIC failure	1	28 min.(3)	30 sec.	27.5 min.	1.7E-3	6.5	3.2E-3	2E-3	5.2E-3
- Large LOCA condensate initiation, control	1	3 min.(3)	30 sec.	2.5 min.	3.2E-1	1.3	3.2E-1	2E-2	3.3E-1
Operator Fails to Manually Transfer HPCI/RCIC Suction from CST to Torus on Low CST Level	3	13 min.(3)	30 sec.	12.5 min.	5.7E-2	3.5	7.6E-2	1E-3	7.7E-2
Operator Fails to Prevent Transfer of HPCI Suction from CST to Torus on High Torus Level(24)	1	20 min.(16)	30 sec.	19.5 min.	4.8E-3	4.6	7.4E-3	1E-3	8.4E-3
Operator Fails to Ventilate HPCI and RCIC Rooms (Under SBO)	1	2 hrs.(14)	35 min.(19)	85 min.	1E-3	7.4	2.1E-3	1E-3	3.1E-3
Operator Fails to Open Makeup Line Bypass Valve from CST to Hotwell	1	5 hrs.(14)	3 min.	297 min.	1E-3	7.4	2.1E-3	1E-3	3.1E-3

Table 3.3-9

RMIEP METHOD
VARIABLES ASSESSED IN QUANTIFICATION OF HEPs FOR POST-ACCIDENT (TYPE C) OPERATOR ACTIONS USING A DETAILED HRA MODEL

Description of Action	Crew Recovery Action Group	Maximum Time for Crew to Complete Action (T_M)	Time for Crew to Perform Action (T_A)	Time for Crew to Diagnose Action (T_D)	Median Failure Probability $P(ND)_{med}$ at T_D	Upper Error Factor (UEF)	Point Estimate for Diagnose Failure Probability $P(ND)_{mean}$ at T_D	Failure Probability for Accomplishing Recovery Action P(NA)	Total Failure Probability for Recovery Action P(NR)
Operator Fails to Align CRD for Injection									
- At transient initiation (enhanced flow)	--	--	--	--	--	--	--	--	1.0[20]
- Following containment failure	--	--	--	--	--	--	--	--	(5)
Operator Fails to Restore LPI Post RPV Depressurization (ATWS)	1	15 min.[13]	30 sec.	14.5 min.	1.4E-1	2.1	1.5E-1	1E-3	1.6E-2
Operator Fails to Prevent Overfilling the RPV, Given Automatic Depressurization Without LPI Inhibit (ATWS)									
- Inadvertent ADS[25]	--	--	--	--	--	--	--	--	1.0[21]
- Emergency depressurization (pumps successfully terminated)	8	15 min.[14]	10 sec.	890 sec.	1.3E-1	2.2	1.5E-1	1E-3	1.5E-1
- Inadvertent or slow depressurization (SORV)	8	5 min.[14]	10 sec.	290 sec.	4.5E-1	1.3	4.6E-1	1E-3	4.6E-1
Operator Fails to Initiate LPCI/Core Spray Following Auto Initiation Failure	3	64 min.[1]	30 sec.	63.5 min.	9.9E-4	10	2.6E-3	1E-3	3.6E-3
Operator Distracted for 2 to 6 Hours Fails to Restart Low Pressure Systems	3	64 min.[1],[22]	30 sec.	63.5 min.	9.9E-4	10	2.6E-3	1E-3	3.6E-3
Operator Fails to Follow Procedure and Restart Low Pressure Systems	3	64 min.[1]	30 sec.	63.5 min	9.9E-4	10	2.6E-3	1E-3	3.6E-3

Table 3.3-9

RMIEP METHOD
VARIABLES ASSESSED IN QUANTIFICATION OF HEPs FOR POST-ACCIDENT (TYPE C) OPERATOR ACTIONS USING A DETAILED HRA MODEL

Description of Action	Crew Recovery Action Group	Maximum Time for Crew to Complete Action (T_M)	Time for Crew to Perform Action (T_A)	Time for Crew to Diagnose Action (T_D)	Median Failure Probability $P(ND)_{med}$ at T_D	Upper Error Factor (UEF)	Point Estimate for Diagnose Failure Probability $P(ND)_{mean}$ at T_D	Failure Probability for Accomplishing Recovery Action P(NA)	Total Failure Probability for Recovery Action P(NR)
Operator Fails to Align Alternate Injection Systems (RHRSW/ESW/GSW/Well Water System) for Injection									
Core Damage Prevention									
- Loss of Makeup Sequences	10	30 min.[26]	1 min.	29 min.	9.8E-2	4.2	1.4E-1	1E-3	1.4E-1
- Large LOCAs	10	10 min.[3]	1 min.	9 min.	2.2E-1	2.4	2.5E-1	1E-3	2.5E-1
- ATWS (Loss of Injection)	10	15 min.[3]	1 min.	14 min.	1.6E-1	2.9	2E-1	1E-3	2E-1
In-vessel Recovery									
- Loss of Makeup (Level 2)	10	120 min.[3]	1 min.	119 min.	2.8E-2	10	7.4E-2	1E-3	7.5E-2
- Large LOCAs (Level 2)	10	100 min.[3]	1 min.	99 min.	3.4E-2	9.4	8.6E-2	1E-3	8.7E-2
- ATWS (Level 2)	10	110 min.[3]	1 min.	109 min.	3.1E-2	10	8.2E-2	1E-3	8.3E-2
Shell Failure Prevention									
- Loss of Makeup Sequences (Level 2)	10	240 min.[3]	1 min.	239 min.	1.4E-2	10	3.7E-2	1E-3	3.8E-2
- Large LOCAs (Level 2)	10	180 min.[3]	1 min.	179 min.	1.9E-2	10	5E-2	1E-3	5.1E-2
- ATWS (Loss of Injection) (Level 2)	10	220 min.[3]	1 min.	219 min.	1.6E-2	10	4.3E-2	1E-3	4.4E-2
Prevention of Containment Overtemperature Failure									
- All Sequences (Level 2)	10	600 min.[3]	1 min.	599 min.	4.7E-3	10	1.3E-2	1E-3	1.4E-2

Table 3.3-9

RMIEP METHOD
VARIABLES ASSESSED IN QUANTIFICATION OF HEPs FOR POST-ACCIDENT (TYPE C) OPERATOR ACTIONS USING A DETAILED HRA MODEL

Description of Action	Crew Recovery Action Group	Maximum Time for Crew to Complete Action (T_M)	Time for Crew to Perform Action (T_A)	Time for Crew to Diagnose Action (T_D)	Median Failure Probability $P(ND)_{med}$ at T_D	Upper Error Factor (UEF)	Point Estimate for Diagnose Failure Probability $P(ND)_{mean}$ at T_D	Failure Probability for Accomplishing Recovery Action P(NA)	Total Failure Probability for Recovery Action P(NR)
Operator Fails to Initiate SLC Boron Injection									
- Condenser Unavailable (within 6 min.)	1	6 min.[14]	30 sec.	5.5 min.	1E-1	1.7	1.1E-1	1E-3	1.1E-1
- Condenser Unavailable (within 20 min.)	1	20 min.[14]	30 sec.	19.5 min.	4.9E-3	4.3	7.3E-3	1E-3	8.3E-3
- Condenser Available (within 40 min.)	1	40 min.[14]	30 sec.	39.5 min.	1E-3	7.4	2.1E-3	1E-3	3.1E-3
Operator Fails to Lower RPV Water Level to TAF for Power Control OR Operator Fails to Restore RPV Level at End of SLC Injection or When Reactor is Shutdown	1	17 min.[3]	2 min.	15 min.	1E-2	3.6	1.4E-2	1E-3	1.5E-2
Operator Fails to Implement Contingency EOP to Flood Containment (CET Node FC) (Level 2)	2	1.8 hrs.[14]	1 min.	107 min.	8.1E-4	10	2.2E-3	1E-3	3.2E-3
Operator Fails to Vent RPV (Level 2)	1	2 hrs.[14]	5 min.	115 min.	1E-3	7.4	2.1E-3	1E-3	3.1E-3
Operator Fails to Initiate Venting (Drywell) Given Indication (CET Node CV) (Level 2)	8	2 hrs.[14]	2 min.	118 min	2.1E-3	1.0	5.6E-3	1E-3	6.6E-3
Operator Fails to Shed Loads from DC Buses (Station Blackout)	--	--	--	--	--	--	--	--	5E-2[12]
Operator Fails to Establish Alternate Control Room Building Cooling	1	2 hrs.[14]	20 min.[14]	100 min.	1E-3	7.4	2.1E-3	1E-3	3.1E-3

Notes to Table 3.3-9

(1) Time from transient initiation until RPV level reaches below 1/3 core height obtained from DAEC MAAP case LII-1A2.

(2) Estimated to be lower than that of transients and small LOCAs. A time frame of 55 minutes is conservatively assumed based on engineering judgement.

(3) Based on engineering judgement and information from a surrogate plant MAAP runs

(4) Time takes to reach TAF.

(5) See Table 3.3-16.

(6) Conservative estimate based on accumulator bleeddown.

(7) 20 minutes to reach Level 1 plus 2 minutes for the ADS timer and 1 minute for the blowdown process.

(8) Based on times from NEDE 24222 for operator to lower water level.

(9) Based on times from NEDE 24222 for water level to be lowered because of mismatch in injection and power production.

(10) Time by which the containment pressure reaches a level which would preclude opening the MSIVs because of inadequate pressure differential.

(11) Based on previous PRAs.

(12) The action is considered to be relatively easy to perform and only a function of being aware (for 30 minutes to 2 hours into a SBO) that such load shedding is appropriate. Because the actions take place over a period of several hours it is judged that they can be characterized as events required to be performed as part of a step by step procedure over a long period of time. This is judged to be given to intervening distractions and therefore a relatively high failure probability of 5E-2 is assigned to this human action.

(13) The allowable time is defined as the time it takes for existing inventory to be discharged to the pool during depressurization and to boil off from the TAF to 1/3 core height. Assuming the worst case ATWS scenario and disregarding boron injection status, the time before depressurization and the allowable time for operator action is estimated to be 13 minutes, plus a short time for failure to restore the level, assumed here to be 2 minutes. This is a total of 15 minutes.

(14) Conservative estimate based on engineering judgement.

(15) NEDE-24222 assumes RHR initiation occurs within 15 minutes. It is judged that this allowable time for action may be significantly longer, but it is conservatively chosen to be 15 minutes.

(16) The time for the operator to ensure no suction from the torus is assumed to be 20 minutes. This is judged to be a conservative assumption, since HPCI failure will not occur immediately.

[17] The onset of core melting is approximately at 1.28 hours (76.8 minutes) from DAEC MAAP run LII-1A2. The allowable time is estimated to be approximately 10 minutes following the onset of core melt (i.e., ~ 87 minutes).

[18] The allowable time for ATWS conditions are judged to be substantially lower than non-ATWS scenarios.

[19] One operator stated that it would take 10 minutes; the other stated that it would take 60 minutes. The time frame of 35 minutes if the average of these two number.

[20] The operators interviewed stated that they would not rely upon CRD to provide low pressure or high pressure RPV injection initially during a transient event. As such, failure probability for operator aligning CRD for enhanced injection during a transient event is considered to be approximately 1.0.

[21] No recovery judged possible; therefore, failure probability is set to 1.0.

[22] The allowable time frame is judged to be approximately 2 hours. However, the applicable table from the RMIEP model does not include extrapolation beyond 62 minutes.

[23] This action is used to derive O-ADS-INHIBIT, but is not used explicitly in the model.

[24] Current model does not include this effect. High torus temperature failure of HPCI/RCIC is not treated.

[25] Treated in event tree logic.

[26] Time from when level is at TAF to when it reaches 1/3 core height. The operator is assumed not to recognize the need for operator action until level is at TAF.

Table 3.3-10

EPRI METHOD (EPRI-NP-6560-L)
HEPs FOR POST-ACCIDENT OPERATOR ACTIONS QUANTIFIED USING A DETAILED HRA MODEL

Description of Action	HEP	Designator
Operator Fails to Manually Initiate Emergency Depressurization		
- Transients and small LOCAs	2.1E-4	OOPAF-MANUAL-DEP
- Medium LOCAs	2.1E-4	OOPAF-MANUAL-DEP
- ATWS (Heat Capacity Temperature Limit Exceeded)	9.7E-3	OOPAF-MANUAL-DEP
- ATWS (Level below -30")	2.1E-4	OOPAF-MANUAL-DEP
- In-vessel Core Degradation (CET Node OP) (Level 2)	1.1E-4	OOPAF-LVL2-ADS
Operator Fails to Initiate LPCI/Core Spray Following Auto Initiation Failure	1.2E-3	LOPPF-LPCI-INIT
Operator Distracted for 2 to 6 Hours Fails to Restart Low Pressure Systems[(1)]	3E-4	LOPAF-LPCI4
Operator Fails to Follow Procedure and Restart Low Pressure Systerns[(1)]	3E-4	LOPPF-LPCI5
Operator Fails to Vent the Torus		
- Transients or LOCAs	1E-2	VOPAF-TORUS-VENT
- ATWS	1.0	VOPAF-TORUS-VENT
Operator Fails to Initiate Vent (Wetwell) per Procedure (CET Node GV) (Level 2)	1E-2	NOPPF-GASVENT
Operator Fails to Initiate Venting Given Indication (CET Node CV) (Level 2)	3.5E-2	CV-02-01

[(1)] The operator actions "operator distracted for 2 to 6 hours fails to restart low pressure systems" and "operator fails to follow procedure and restart low pressure systems" are redundant and therefore only one of these human actions should be used in the model.

Table 3.3-11

EPRI METHOD (EPRI-NP-6560-L)
VARIABLES ASSESSED IN QUANTIFICATION OF HEPs FOR POST-ACCIDENT (TYPE C) OPERATOR ACTIONS USING A DETAILED HRA MODEL

Description of Action	Non-recoverable Failure Probability (P_1)	Non-response Failure Probability (P_2)							Manipulative Error Probability (P_3)	Total HI Failure Probability ($P_1 + P_2 + P_3$)
		Type of HI (CPI-5)	Average Log Standard Deviation	Crew Median Response Time (T)	Time Window for Cognitive Response			Probability of Crew Non-Response (P_2)		
					T_{SW}	T_M	T_W			
Operator Fails to Manually Initiate Emergency Depressurization										
- Transients and Small LOCAs	1E-4	CP2	0.58	6.7 min.	64 min.	15 sec.	63.75 min.	1E-4	9E-6	2.1E-4
- Medium LOCAs	1E-4	CP2	0.58	6.7 min.	55 min.	15 sec.	54.75 min.	1E-4	9E-6	2.1E-4
- ATWS (Heat Capacity Temperature Limit Exceeded)	1E-4	CP2	0.58	4 min.	16 min.	30 sec.	15.5 min.	9.6E-3	9E-6	9.7E-3
- ATWS (Level below -30")	1E-4	CP2	0.58	5.2 min.	43 min.	30 sec.	42.5 min.	1E-4	9E-6	2.1E-4
- In-vessel core degradation (CET Node OP) (Level 2)	1E-4	CP2	0.58	9 min.	87 min.	30 sec.	86.5 min.	0.0	9E-6	1.1E-4
Operator Fails to Initiate LPCI/Core Spray Following Auto Initiation Failure	1E-4	CP1	0.70	5 min.	64 min.	30 sec.	63.5 min.	< 1E-4	1E-3	1.2E-3
Operator Distracted for 2 to 6 Hours Fails to Restart Low Pressure Systems*	1E-4	CP1	0.70	5 min.	2 hrs.	30 sec.	119.5 min.	< 1E-4	1E-4	3E-4
Operator Fails to Follow Procedure and Restart Low Pressure Systems*	1E-4	CP1	0.70	5 min.	2 hrs.	30 sec.	119.5 min.	< 1E-4	1E-4	3E-4
Operator Fails to Vent the Torus										
- Transients or LOCAs or ATWS (with SLC)	1E-2	CP1	0.70	25 min.	5 hrs.	30 sec.	299.5 min.	2E-4	1E-4	1E-2
- ATWS (without SLC)	1E-2	CP1	0.70	25 min.	0 hrs.	30 sec.	0 min.	1.0	1E-4	1.0
Operator Fails to Initiate Vent (Wetwell) per Procedure (CET Node GV) (Level 2)	1E-2	CP1	0.70	25 min.	5 hrs.	30 sec.	299.5 hrs.	2E-4	1E-4	1E-2
Operator Fails to Initiate Venting (Drywell) Given Indication (CET Node CV)	1E-4	CP3	0.75	30 min.	2 hrs.	2 min.	118 min.	3.4E-2	1E-3	3.5E-2

* The operator actions "operator distracted for 2 to 6 hours fails to restart low pressure systems" and "operator fails to follow procedure and restart low pressure systems" are redundant and therefore only one of these human actions should be used in the model.

Table 3.3-12
HEPs FOR POST-ACCIDENT OPERATOR ACTIONS QUANTIFIED USING
EPRI RESEARCH PROJECT 3206-03 SCREENING METHODOLOGY

HUMAN ACTION DESCRIPTION	HUMAN ERROR PROBABILITY (HEP) (Point Estimate)	DESIGNATOR
Operator fails to manually locally open MO1947/MO2046 in RHRSW loop BD and AC for torus cooling		WOPAF-MO1947 WOPAF-MO2046
- Transients	0.03	
- LOCAs	1.0	
- ATWS	1.0	
Operator fails to manually locally close MO1947/MO2046 in RHRSW system for RPV injection	0.1	WOPAF-MO1947-C WOPAF-MO2046-C
- Transients	1.0	
- LOCAs	1.0	
- ATWS		
Operator fails to locally manually close breaker to start RHRSW pump (Action occurs in control room building)		WOPAF-LOCALSTART
RPV Injection	0.05	
- Transients	1.0(2)	
- LOCAs	1.0(2)	
- ATWS		
Torus Cooling	0.01(1)	
- Transients	0.01(1)	
- LOCAs	0.01(1)	
- ATWS		
Operator fails to start core spray pump by locally manually closing breaker (Action occurs in control building)	0.05	SOPAF-LOCALSTART
- Transients	1.0(2)	
- LOCAs	1.0(2)	
- ATWS		

Table 3.3-12
HEPs FOR POST-ACCIDENT OPERATOR ACTIONS QUANTIFIED USING
EPRI RESEARCH PROJECT 3206-03 SCREENING METHODOLOGY

HUMAN ACTION DESCRIPTION	HUMAN ERROR PROBABILITY (HEP) (Point Estimate)	DESIGNATOR
Operator fails to locally manually close breaker to start RHR pump (Action occurs in control building)		LOPAF-LOCALSTART
RPV Injection	0.05	
- Transients	1.0[2]	
- LOCAs	1.0[2]	
- ATWS		
Torus Cooling	0.01[1]	
- Transients	0.01[1]	
- LOCAs	0.01[1]	
- ATWS		
Operator fails to start GSW pump by locally manually closing breaker (Action occurs in control building)		GOPAF-LOCALSTART
RPV Injection	0.05	
- Transients	1.0[2]	
- LOCAs	1.0[2]	
- ATWS		
	0.01[1]	
Torus Cooling	0.01[1]	
- Transients	0.01[1]	
- LOCAs		
- ATWS		

Table 3.3-12
HEPs FOR POST-ACCIDENT OPERATOR ACTIONS QUANTIFIED USING EPRI RESEARCH PROJECT 3206-03 SCREENING METHODOLOGY

HUMAN ACTION DESCRIPTION	HUMAN ERROR PROBABILITY (HEP) (Point Estimate)	DESIGNATOR
Operator fails to locally manually close breaker to start a pump (Loss of DC Power Event Tree)		POPAF-LOCALSTART
RPV Injection		
- Transients	0.05	
- LOCAs	1.0[2]	
- ATWS	1.0[2]	
Torus Cooling		
- Transients	0.01[1]	
- LOCAs	0.01[1]	
- ATWS	0.01[1]	
Operator fails to replenish CST supply	1.0	HOPAF-5
Operator fails to perform actions to open flowpath from CST to core spray	1.0	SOPAF-LPCS-CST
Operator fails to locally align RHRSW injection to RPV (Action occurs in reactor building corner rooms and torus rooms)	0.3	WOPPF-RHRSWINJ
- Transients	1.0	
- LOCAs	1.0	
- ATWS		
Operator fails to manually open pneumatic control valve CV 4915 or CV 4914 (water supply pumps discharge valves)		WOPAF-WS02
RPV Injection	0.3	
- Transients	1.0[2]	
- LOCAs	1.0[2]	
- ATWS		
Torus Cooling	0.05	
- Transients	0.05	
- LOCAs	0.05	
- ATWS		

Table 3.3-12
HEPs FOR POST-ACCIDENT OPERATOR ACTIONS QUANTIFIED USING
EPRI RESEARCH PROJECT 3206-03 SCREENING METHODOLOGY

HUMAN ACTION DESCRIPTION	HUMAN ERROR PROBABILITY (HEP) (Point Estimate)	DESIGNATOR
Operator fails to cross-tie ESW trains - SBO without HPCI/RCIC - SBO with HPCI/RCIC availability for 4 hours	 1.0 0.1	EOPAF-CROSS-TIE
Operator fails to manually bleed air supply to CV-2080 and CV-2081 for ESW - SBO without HPCI/RCIC - SBO with HPCI/RCIC availability for 4 hours	 0.1 0.05	EOPAF-CV2080 EOPAF-CV2081
Operator closes CV4910A or CV4910B and prevents flow through alternate path to the stilling basin RPV Injection - Transients - LOCAs - ATWS Torus Cooling - Transients - LOCAs - ATWS	 0.1 1.0[2] 1.0[2] 0.03 0.03 0.03	WOPFF-WS04 WOPFF-WS03

(1) Because the fault tree model only includes one designator for both RPV injection and torus cooling functions, the value of 0.05 should be conservatively used for both functions.

(2) For LOCA and ATWS cases because of the limited time, state of confusion and possibly limited number of AOs, the HEP is assumed to be 1.0.

Table 3.3-13
VARIABLES ASSESSED IN POST-ACCIDENT OPERATOR ACTIONS USING EPRI RESEARCH PROJECT 3206-03 SCREENING METHODOLOGY
DAEC INTERVIEWEE: SRO No. 1

Human Action Description (Designator)	What is the Time Available During Which Recovery Action Must Be Performed?	What is the time required to perform the action?	Location/Accessibility: Where is the location of the equipment?	Location/Accessibility: Is equipment location accessible during accident? (Yes/No)	Location/Accessibility: Is equipment accessible without special equipment?[1] (Yes/No)	Location/Accessibility: If special equipment is required for access to equipment, does it exist in sufficient proximity to equipment?[2] (Yes/No)	Special Equipment: If special equipment is needed to perform recovery action, is it available within the time available?[3] (Yes/No)	Function: Can the required equipment be put into a functional condition by plant personnel in the time available?[3] (Yes/No)	Training/Practice: Are plant personnel trained/practiced on the recovery action? (Yes/No)	Training/Practice: Is the action included in the plant JPM[7] list? (Yes/No)	Training/Practice: What is the number of operators required to perform the action?	Procedures: Is there a procedure for carrying out the recovery action? (Yes/No)	Complexity: Is the action simple[3] or complex[4]?	Environmental Factors: Is the environment in the vicinity of equipment good[5] or poor[6]?
Operator fails to manually locally open MO1947/MO2046 in RHRSW loop BD and AC (WOPAF-MO1947, WOPAF-MO2046)	Transients: 28 hrs.[28] LOCAs. 0 hrs.[29] ATWS: 0 hrs.[29]	9 min.[26]	Reactor Building Corner Rooms	YES[9]	YES	N/A	YES	YES	NO[8]	NO	1	NO[10]	Simple	Good[17]
Operator fails to manually locally close MO1947/MO2046 in RHRSW system (WOPAF-MO1947-C, WOPAF-MO2046-C)	Transients: 0.64 hrs.[30] LOCAs. 0 hrs.[31] ATWS. 0 hrs.[31]	12 min.[11]	Reactor Building Corner Rooms	YES	YES	N/A	YES	YES	NO[8]	NO	1	NO[12]	Simple	Good
Operator fails to locally manually close breaker to start RHRSW pump (WOPAF-LOCALSTART)	Injection Transients 0.64 hrs[30] LOCAs. 10 min.[31] ATWS: 10 min.[31] Torus Cooling Transients. 28 hrs[27] LOCAs 15 - 20 hrs[32] ATWS: 4 - 10 hrs[33]	10 min.[26]	Switchgear 1A3/4 Central Switchgear Room - Control Building First Floor	YES	YES[13]	N/A	YES	YES	YES[14]	YES	1	YES[15]	Simple	Good
Operator fails to start core spray pump by locally manually closing breaker (SOPAF-LOCALSTART)	Transients: 0.64 hrs[30] LOCAs: 10 min.[31] ATWS: 10 min.[31]	10 min.[26]	Switchgear 1A3/4 Central Switchgear Room - Control Building First Floor	YES	YES[14]	N/A	YES	YES	YES[14]	YES	1	YES[15]	Simple	Good

Table 3.3-13
VARIABLES ASSESSED IN POST-ACCIDENT OPERATOR ACTIONS USING EPRI RESEARCH PROJECT 3206-03 SCREENING METHODOLOGY
DAEC INTERVIEWEE: SRO No. 1

			Location/Accessibility				Special Equipment	Function	Training/Practice			Procedures	Complexity	Environmental Factors
Human Action Description (Designator)	What is the Time Available During Which Recovery Action Must Be Performed?	What is the time required to perform the action?	Where is the location of the equipment?	Is equipment location accessible during accident? (Yes/No)	Is equipment accessible without special equipment?[1] (Yes/No)	If special equipment is required for access to equipment, does it exist in sufficient proximity to equipment?[2] (Yes/No)	If special equipment is needed to perform recovery action, is it available within the time available?[3] (Yes/No)	Can the required equipment be put into a functional condition by plant personnel in the time available?[4] (Yes/No)	Are plant personnel trained/practiced on the recovery action? (Yes/No)	Is the action included in the plant JPM[7] list? (Yes/No)	What is the number of operators required to perform the action?	Is there a procedure for carrying out the recovery action? (Yes/No)	Is the action simple[8] or complex[9]?	Is the environment in the vicinity of equipment good[8] or poor[9]?
Operator fails to locally manually close breaker to start RHR pump (LOPAF-LOCAL-START)	Injection Transients 0.64 hrs.[20] LOCAs 10 min.[21] ATWS 10 min.[21] Torus Cooling Transients 28 hrs[42] LOCAs: 15 - 20 hrs.[27] ATWS 4 - 10 hrs[27]	10 min.[24]	Switchgear 1A3/4 Central Switchgear Room - Control Building First Floor	YES	YES[13]	N/A	YES	YES	YES[14]	YES	1	YES[16]	Simple	Good
Operator fails to manually locally open breaker to start GSW pump	Injection Transients. 0.64 hrs.[20] LOCAs 10 min.[21] ATWS: 10 min.[21] Torus Cooling Transients: 28 hrs.[42] LOCAs 15 - 20 hrs.[27] ATWS. 4 - 10 hrs.[27]	10 min.[24]	Switchgear 1A3/4 Central Switchgear Room - Control Building First Floor	YES	YES[13]	N/A	YES	YES	YES[14]	YES	1	YES[16]	Simple	Good
Operator fails to locally manually close breaker to start a pump (Loss of DC Power avont tres)	Injection Transients: 0.64 hrs[20] LOCAs: 10 min.[21] ATWS: 10 min.[21] Torus Cooling Transients: 28 hrs[42] LOCAs 15 - 20 hrs[27] ATWS: 4 - 10 hrs.[27]	10 min.[24]	Switchgear 1A3/4 Central Switchgear Room - Control Building First Floor	YES	YES[13]	N/A	YES	YES	YES[14]	YES	1	YES[16]	Simple	Good

Table 3.3-13
VARIABLES ASSESSED IN POST-ACCIDENT OPERATOR ACTIONS USING EPRI RESEARCH PROJECT 3206-03 SCREENING METHODOLOGY
DAEC INTERVIEWEE: SRO No. 1

Human Action Description (Designator)	What is the Time Available During Which Recovery Action Must Be Performed?	What is the time required to perform the action?	Location/Accessibility				Special Equipment	Function	Training/Practice			Procedures	Complexity	Environmental Factors
			Where is the location of the equipment?	Is equipment location accessible during accident? (Yes/No)	Is equipment accessible without special equipment?[1] (Yes/No)	If special equipment is required for access to equipment, does it exist in sufficient proximity to equipment?[2] (Yes/No)	If special equipment is needed to perform recovery action, is it available within the time available?[3] (Yes/No)	Can the required equipment be put into a functional condition by plant personnel in the time available?[3] (Yes/No)	Are plant personnel trained/practiced on the recovery action? (Yes/No)	Is the action included in the plant JPM[7] list? (Yes/No)	What is the number of operators required to perform the action?	Is there a procedure for carrying out the recovery action? (Yes/No)	Is the action simple[5] or complex[6]?	Is the environment in the vicinity of equipment good[8] or poor[9]?
Operator fails to replenish CST supply (HOPAF-5)[18]	---	---	---	---	---	---	---	---	---	---	---	---	---	---
Operator fails to perform actions to open flowpath from CST to core spray (SOPAF-LPCS-CST)[18]	---	---	---	---	---	---	---	---	---	---	---	---	---	---
Operator fails to locally align RHRSW injection (WOPPF-RHRSWINJ)	Transients: 0.64 hrs [20] LOCAs. 0 hrs.[21] ATWS: 0 hrs.[21]	20 min.[17]	Reactor Building Corner Rooms and Torus Room	YES	YES	N/A	YES	YES	NO[18]	YES	3[20]	NO[18]	Complex	Good
Operator fails to manually open pneumatic control valve CV 4915 or CV 4914 from the control room (WOPAF-WS02)	Injection Transients: 0.64 hrs.[20] LOCAs: 30 min.[22] ATWS: 30 min.[22] Torus Cooling Transients: 28 hrs [20] LOCAs: 15-20 hrs.[20] ATWS: 4-10 hrs.[22]	6 min.[17]	Basement of pumphouse	YES	YES	N/A	YES	YES	NO[23]	NO	1	NO[24]	Medium Complexity[23]	Good
Operator fails to cross-tie ESW trains[27] (EOPAF-CROSS-TIE)	SBO w/o HPCI/RCIC: 30 min.[34] SBO with HPCI/RCIC available for 4 hrs ≤ 4 hrs.	> 1 hr.	ESW Pump Room	YES	YES	N/A	YES[34]	YES[34]	NO	NO	2	NO	Complex	Good
Operator fails to manually (local) bleed air supply to CV 2080 and CV 2081 for ESW[27] (EOPAF-CV2080, EOPAF-CV2081)	SBO w/o HPCI/RCIC: 30 min [34] SBO with HPCI/RCIC available for 4 hrs: ≤ 4 hrs.	10 min.[25]	Diesel Generator Room	YES	YES	N/A	YES[34]	YES[35],[34]	NO[27]	NO	1	NO[26]	Simple	Good

Table 3.3-13
VARIABLES ASSESSED IN POST-ACCIDENT OPERATOR ACTIONS USING EPRI RESEARCH PROJECT 3206-03 SCREENING METHODOLOGY
DAEC INTERVIEWEE: SRO No. 1

Human Action Description (Designator)	What is the Time Available During Which Recovery Action Must Be Performed?	What is the time required to perform the action?	Location/Accessibility: Where is the location of the equipment?	Location/Accessibility: Is equipment location accessible during accident? (Yes/No)	Location/Accessibility: Is equipment accessible without special equipment?[1] (Yes/No)	Location/Accessibility: If special equipment is required for access to equipment, does it exist in sufficient proximity to equipment?[2] (Yes/No)	Special Equipment: If special equipment is needed to perform recovery action, is it available within the time available?[2] (Yes/No)	Function: Can the required equipment be put into a functional condition by plant personnel in the time available?[2] (Yes/No)	Training/Practice: Are plant personnel trained/practiced on the recovery action? (Yes/No)	Training/Practice: Is the action included in the plant JPM[7] list? (Yes/No)	Training/Practice: What is the number of operators required to perform the action?	Procedures: Is there a procedure for carrying out the recovery action? (Yes/No)	Complexity: Is the action simple[3] or complex[4]?	Environmental Factors: Is the environment in the vicinity of equipment good[5] or poor[6]?
Operator error for CV 4910A or 4910B (WOPFF - WS04, WOPFF-WS03)	Injection Transients: 0.64 hrs.[24] LOCAs: 30 min.[33] ATWS 30 min.[33] Torus Cooling Transients: 28 hrs [28] LOCAs. 15 - 20 hrs [27] ATWS. 4 - 10 hrs.[27]	5 min.[41]	Pump House	YES	YES	N/A	YES	YES	NO	NO	1	NO[40]	Simple	Good

Notes to Table 3.3-13

1. If your answer is "yes" then skip the next question.

2. If your answer is "no", STOP, the action is infeasible.

3. Simple (e.g., opening/closing MOVs, starting electrical equipment, no special tools required, no special clothing required, etc.)

4. Complex (e.g., re-alignment of non-safety related systems to safety-related systems, cross-connection to other units, etc.)

5. Good (e.g., no restricted access and easy to reach, good lighting, no excessive heat, no radiation, no excessive noise, enough space).

6. Poor (e.g. restricted access and difficult to reach, poor lighting, excessive heat, radiation, excessive noise, tight space).

7. Job Performance Measure (JPM).

8. For accidents that need torus cooling, the equipment is accessible. The accessibility generally depends on the accident.

9. Plant personnel are trained in closing/opening valves. However, they are not trained to open/close these specific valves.

10. The operator specified that the applicable procedure is the procedure for manual startup of RHRSW (OI 416). This procedure specifies manipulation of these valves from the control room. Manual local actions for valve manipulation are not included in the procedure; however, operator would assume that local valve operation is part of the procedure.

11. The operator specified that closing valves for injection into the RPV would take longer (7 min.) since the operators at this stage of the accident are going through the EOPs. A time frame of 5 min. was added to take into account the diagnosis time.

12. The operator specified that the applicable procedure is "RPV Injection with RHRSW" (IPOI 7). IPOI 7 specifies manipulation of these valves from the control room. Manual local actions for valve manipulation are not included in the procedure.

13. The operator specified that no special equipment is required unless there are problems with the breakers. In this case, rubber gloves and face shield are needed which are next to the breakers.

14. According to the operator interviewed, there is simulator training and actual plant training at AO level.

15. The applicable procedure is AOP 302.1 (Loss of 125 VDC Power). This procedure specifies local operation of the affected switchgear to stop and start equipment, as required.

16. The operator interviewed stated that he would not replenish CST on low CST level. Therefore, the HEP for this action is considered to be 1.0.

17. The environmental conditions in the vicinity of equipment is generally accident-dependent.

18. The operator stated that this action is not performed except during refueling. He also stated that he generally would not use core spray during an accident, since use of core spray would create saturation conditions which could result in inaccuracy of instrumentation. As such, the HEP is considered to be 1.0.

19. Operator stated that the plant personnel are trained on how to lineup RHRSW for injection from the control room; However, there is no training on local alignment of RHRSW for injection.

20. The operator stated that 3 operators are needed to perform the action; 1 operator at the pumphouse; 1 operator at cooling towers; and 1 operator at control room panels.

21. The allowable time for LOCAs and ATWS is estimated at zero. This is because of high radiation during these scenarios which would preclude access.

22. Plant personnel in general are trained on valve manipulation both at SRO and AO level. However, they are not specifically trained on this action.

23. Assumed to be a complex action.

24. The operator specified ARP 1C06A-D1, D2, and D11 (A/B RHRSW & ESW pit Low Level; circulating water Low Level) as applicable procedures. However, these procedures do not appear to direct the operators to manually open CV4915 or CV4914.

25. Can run diesel generators without cooling for at least 10 minutes.

26. The operator specified Standby Diesel Generator, OI 324, as the applicable procedure. However, this procedure does not appear to direct the operators to manually bleed air supply to CV2081 and CV2080.

27. It is assumed that the fault tree model accounts for preserving the alternate DG despite lack of cooling on initial restart.

28. Conservative estimate based on DAEC MAAP case LII-2T1. The allowable time for LOCAs and ATWS is estimated at zero. This is because of high radiation during these scenarios which would preclude access.

29. General training and practice exists for valve manipulation. However, there are no specific training or practice for this action.

30. Based on MAAP case LII-1D4.

31. Best estimate judgement of the time to core damage.

32. Based on engineering judgement.

33. Based on the assumption that the stilling basin has enough inventory for approximately 30 minutes.

34. Special equipment needed are available within the available time frame only if HPCI/RCIC are available. For SBO without HPCI/RCIC the HEP estimated to be 1.0.

35. The operator stated that it would take approximately 4 minutes for travel time and valve manipulation. A time frame of 5 minutes was added to take into account the diagnosis time.

36. The operator stated that it would take approximately 1 minute to perform the action. A time frame of 4 minutes for travel time and a time frame of 5 minutes for diagnosis time was added.

37. The operator stated that it would take approximately 15 minutes for travel time and to perform the action. A time frame of 5 minutes was added to take into account the diagnosis time.

38. The operator stated that it would take approximately 1 minute to perform the action. A time frame of 5 minutes was added to take into account the diagnosis time.

39. The operator stated that it would take approximately 5 minutes for travel time and performing the action. A time frame of 5 minutes was added to take into account the diagnosis time.

40. The operator cited AOP 410 and 518 as applicable procedures. However, these procedures do not specify manual operation of CV4910A or B.

41. Based on engineering judgement. This time frame takes into account the diagnosis time, travel time, and manipulation time.

42. Conservative estimate based on DAEC MAAP case LII-2T-1.

Table 3.3-14

VARIABLES ASSESSED IN POST-ACCIDENT OPERATOR ACTIONS USING EPRI RESEARCH PROJECT 3206-03 SCREENING METHODOLOGY
DAEC INTERVIEWEE: SRO No. 2

Human Action Description (Designator)	What is the Time Available During Which Recovery Action Must Be Performed?	What is the time required to perform the action?[5]	Location/Accessibility: Where is the location of the equipment?	Location/Accessibility: Is equipment location accessible during accident? (Yes/No)	Location/Accessibility: Is equipment accessible without special equipment?[1] (Yes/No)	Location/Accessibility: If special equipment is required for access to equipment, does it exist in sufficient proximity to equipment?[2] (Yes/No)	Special Equipment: If special equipment is needed to perform recovery action, is it available within the time available?[2] (Yes/No)	Function: Can the required equipment be put into a functional condition by plant personnel in the time available?[2] (Yes/No)	Training/Practice: Are plant personnel trained/practiced on the recovery action? (Yes/No)	Training/Practice: Is the action included in the plant JPM[7] list? (Yes/No)	Training/Practice: What is the number of operators required to perform the action?	Procedures: Is there a procedure for carrying out the recovery action? (Yes/No)	Complexity: Is the action simple[3] or complex[4]?	Environmental Factors: Is the environment in the vicinity of equipment good[6] or poor[6]?
Operator fails to manually locally open MO1947/MO2046 in RHRSW loop BO and AC (WOPAF-MO1947, WOPAF-MO2046)	Transients: 28 hrs.[20] LOCAs: 0 hrs.[20] ATWS: 0 hrs.[20]	7 min.[9]	Reactor Building Corner Rooms	YES[8]	YES	N/A	YES	YES	NO[10]	NO	1	NO[11]	Simple	Good (Depends on Accident)
Operator fails to manually locally close MO1947/MO2046 in RHRSW system (WOPAF-MO1947-C, WOPAF-MO2046-C)	Transients: 0.64 hrs.[22] LOCAs: 0 hrs.[18] ATWS: 0 hrs.[18]	7 min.[9]	Reactor Building Corner Rooms	YES[8]	YES	N/A	YES	YES	NO[10]	NO	1	NO[12]	Simple	Good (Depends on Accident)
Operator fails to locally manually close breaker to start RHRSW pump (WOPAF-LOCALSTART)	Injection Transients: 0.64 hrs.[22] LOCAs. 10 min.[24] ATWS: 10 min.[24] Torus Cooling Transients: 28 hrs.[20] LOCAs: 15-20 hrs.[25] ATWS: 4-10 hrs.[26]	10 min.[26]	Switchgear 1A3/4- Central Switchgear Room - Control Building - First Floor	YES	YES	N/A	YES	YES	YES[13]	NO	1	YES[14]	Simple	Good
Operator fails to start core spray pump by locally manually closing breaker (SOPAF-LOCALSTART)	Transients: 0.64 hrs.[22] LOCAs: 10 min.[24] ATWS 10 min.[24]	10 min.[26]	Switchgear 1A3/4 - Central Switchgear Room - Control Building First Floor	YES	YES	N/A	YES	YES	YES[13]	NO	1	YES[14]	Simple	Good
Operator fails to locally manually close breaker to start RHR pump (LOPAF-LOCAL-START)	Injection Transients: 0.64 hrs.[22] LOCAs. 10 min.[24] ATWS: 10 min.[24] Torus Cooling Transients: 28 hrs.[20] LOCAs: 15-20 hrs.[25] ATWS: 4-10 hrs.[26]	10 min.[26]	Switchgear 1A3/4 Central Switchgear Room - Control Building - First Floor	YES	YES	N/A	N/A	YES	YES[13]	NO	1	YES[14]	Simple	Good

Table 3.3-14

VARIABLES ASSESSED IN POST-ACCIDENT OPERATOR ACTIONS USING EPRI RESEARCH PROJECT 3206-03 SCREENING METHODOLOGY
DAEC INTERVIEWEE: SRO No. 2

Human Action Description (Designator)	What Is the Time Available During Which Recovery Action Must Be Performed?	What is the time required to perform the action?[5]	Location/Accessibility				Special Equipment	Function	Training/Practice			Procedures	Complexity	Environmental Factors
			Where is the location of the equipment?	Is equipment location accessible during accident? (Yes/No)	Is equipment accessible without special equipment?[1] (Yes/No)	If special equipment is required for access to equipment, does it exist in sufficient proximity to equipment?[2] (Yes/No)	If special equipment is needed to perform recovery action, is it available within the time available?[2] (Yes/No)	Can the required equipment be put into a functional condition by plant personnel in the time available?[2] (Yes/No)	Are plant personnel trained/practiced on the recovery action? (Yes/No)	Is the action included in the plant JPM[7] list? (Yes/No)	What is the number of operators required to perform the action?	Is there a procedure for carrying out the recovery action? (Yes/No)	Is the action simple[3] or complex[4]?	Is the environment in the vicinity of equipment good[6] or poor[6]?
Operator fails to manually locally open breaker to start GSW pump	Injection Transients: 0.64 hrs.[22] LOCAs: 10 min.[24] ATWS. 10 min.[24] Torus Cooling Transients: 28 hrs.[26] LOCAs. 15-20 hrs.[26] ATWS. 4-10 hrs.[26]	10 min.[23]	Switchgear 1A3/4 Central Switchgear Room - Control Building - First Floor	YES	YES	N/A	N/A	YES	YES[18]	NO	1	YES[14]	Simple	Good
Operator fails to locally manually close breaker to start one pump (Loss of DC Power event tree)	Injection Transients: 0.64 hrs.[22] LOCAs: 10 min.[24] ATWS: 10 min.[24] Torus Cooling Transients: 28 hrs.[26] LOCAs: 15-20 hrs.[26] ATWS. 4-10 hrs.[26]	10 min.[23]	Switchgear 1A3/4 Central Switchgear Room - Control Building - First Floor	YES	YES	N/A	N/A	YES	YES[18]	NO	1	YES[14]	Simple	Good
Operator fails to replenish CST supply[14] (HOPAF-5)	----	----	----	----	----	----	----	----	----	----	----	----	----	----
Operator fails to perform actions to open flowpath from CST to core spray [14] (SOPAF-LPCS-CST)	----	----	----	----	----	----	----	----	----	----	----	----	----	----
Operator fails to locally align RHRSW injection (WOPPF-RHRSWINJ)	Transients: 0.64 hrs.[22] LOCAs. 0 hrs.[18] ATWS: 0 hrs.[18]	13 min.[24]	Reactor Building Corner Rooms	YES[9]	YES	N/A	YES	YES	NO[19]	YES	1 or 2	NO[12]	Complex	Good (Depends on Accident)

Table 3.3-14

VARIABLES ASSESSED IN POST-ACCIDENT OPERATOR ACTIONS USING EPRI RESEARCH PROJECT 3206-03 SCREENING METHODOLOGY
DAEC INTERVIEWEE: SRO No. 2

			Location/Accessibility				Special Equipment	Function	Training/Practice			Procedures	Complexity	Environmental Factors
Human Action Description (Designator)	What is the Time Available During Which Recovery Action Must Be Performed?	What is the time required to perform the action?[a]	Where is the location of the equipment?	Is equipment location accessible during accident? (Yes/No)	Is equipment accessible without special equipment?[1] (Yes/No)	If special equipment is required for access to equipment, does it exist in sufficient proximity to equipment?[2] (Yes/No)	If special equipment is needed to perform recovery action, is it available within the time available?[d] (Yes/No)	Can the required equipment be put into a functional condition by plant personnel in the time available?[2] (Yes/No)	Are plant personnel trained/practiced on the recovery action? (Yes/No)	Is the action included in the plant JPM[7] list? (Yes/No)	What is the number of operators required to perform the action?	Is there a procedure for carrying out the recovery action? (Yes/No)	Is the action simple[b] or complex[c]?	Is the environment in the vicinity of equipment good[d] or poor[e]?
Operator fails to manually open pneumatic control valve CV 4915 or CV 4914 from the control room (WOPAF-WS02)	Injection Transients: 0.64 hrs.[33] LOCAs: 30 min.[34] ATWS: 30 min.[34] Torus Cooling Transients: 28 hrs.[36] LOCAs. 15-20 hrs.[36] ATWS 4-10 hrs.[36]	6 min.[31]	Basement of Pump House	YES	YES	N/A	YES[31]	YES	NO[32]	NO	At least 1 mechanic	NO	Complex	Good
Operator fails to cross-tie ESW trains[34] (EOPAF-CROSS-TIE)	SBO without HPCI/RCIC: 30 min[37] SBO with HPCI/RCIC Available for 4 hrs.: ≤ 4 hrs.	> 1 hr.	ESW Pump Room	YES	YES	N/A	YES[35]	Depends on the scenario	NO	NO	2	NO	Complex	Good
Operator fails to manually (local) bleed air supply to CV 2080 and CV 2081 for ESW (EOPAF-CV2080, EOPAF-CV2081)	SBO without HPCI/RCIC: 30 min[37] SBO with HPCI/RCIC Available for 4 hrs.: ≤ 4 hrs.	9 min.[37]	Diesel Generator Room	YES	YES	N/A	N/A	-	NO[37, 38]	NO	1	NO	Simple	Good
Operator error for CV 4910A and CV 4910B (WOPFF - W504, WOPFF - W503)	Injection Transients: 0.64 hrs.[33] LOCAs. 30 min.[34] ATWS: 30 min.[34] Torus Cooling Transients: 28 hrs.[36] LOCAs: 15 - 20 hrs.[36] ATWS: 4 - 10 hrs.[36]	5 min.[39]	Pump House	YES	YES	N/A	YES	YES	NO	NO	1	NO[17]	Simple	Good

Notes to Table 3.3-14

1. If your answer is "yes" then skip the next question.

2. If your answer is "no", STOP, the action is infeasible.

3. Simple (e.g., opening/closing MOVs, starting electrical equipment, no special tools required, no special clothing required, etc.)

4. Complex (e.g., re-alignment of non-safety related systems to safety-related systems, cross-connection to other units, etc.)

5. Good (e.g., no restricted access and easy to reach, good lighting, no excessive heat, no radiation, no excessive noise, enough space).

6. Poor (e.g. restricted access and difficult to reach, poor lighting, excessive heat, radiation, excessive noise, tight space).

7. Job Performance Measure (JPM).

8. According to the operator, it would take 1 min. to get to the corner room and 1 min. to open the valve. A time frame of 5 minutes was also added to take into account the diagnosis time.

9. Accessibility generally depends on accident. May not be accessible if radiation is present.

10. The operator stated that the plant personnel are trained in closing/opening valves in general. He stated that the 2nd assistant who will perform the action has probably performed opening/closing valves many times and that there is training at SRO and AO level. However, there is no training for this specific action.

11. According to the operator, the applicable procedure is manual startup of RHRSW (OI 416). This procedure specifies manipulation of these valves from the control room. Manual actions for valve manipulation are not included in the procedure.

12. The manual operation of RHRSW valves is only directed to be performed from the control room (IPOI 7). No local manual action is specified.

13. AOs are trained in manual manipulation of breakers.

14. Abnormal operating procedures for loss of 125 VDC power (AOP 302.1) directs the operator to locally operate affected switchgear to start and stop equipment, as required.

15. The operator stated that this action would probably not be performed. This action requires manipulation of one valve in the basement of the turbine building.

16. The allowable time frame for LOCAs and ATWS is estimated to be zero. This is because of high radiation which may which may preclude access.

17. The operator cited AOP 410 and 518 as applicable procedures. However, these procedures do not specify manual operation of CV4910 or B.

18. Operator stated that he would not perform this action and that he would use RHRSW instead. As such, the HEP for this action is considered to be 1.0.

19. Plant personnel are trained to manipulate MOVs in general. However, there is no specific training for this action.

20. Based on engineering judgement. This time frame takes into account the travel time, the diagnosis time and the manipulation time.

21. According to the operator interviewed, performing this action requires a variety of wrenches and etc.

22. Plant personnel are not trained on this specific action. However, they are trained in manipulating valves in general.

23. To cross-tie the ESW trains, the spool piece is needed. Also need wrenches to install the spool piece. Installing the spool piece is not possible within the available time for cases with no HPCI/RCIC.

24. It is assumed that the fault tree model accounts for preserving the alternate DG despite lack of cooling on initial restart.

25. Plant personnel are trained in isolating and bleeding air to air operated valves.

26. Based on DAEC MAAP case LII-2T1. The allowable time for LOCAs and ATWS is estimated at zero because of high radiation during these scenarios which would preclude access.

27. General training and practice exists for valve manipulation. However, there are no specific training or practice for this action.

28. The operator stated that it would take 5 minutes for travel and manipulation. A time frame of 5 min. was added to take into account the diagnosis time.

29. The operator stated that it would take 5 minutes for travel and manipulation. A time frame of 5 minutes was added to take into account the diagnosis time.

30. The operator stated that it would take 6 minutes for valve manipulation (3 valves). A time frame of 2 minutes for travel time and a time frame of 5 minutes for diagnosis was added.

31. The operator stated that it would take 1 minute for valve manipulation. A time frame of 5 minutes was added to take into account the diagnosis time.

32. The operator stated that it would take 2 minutes to perform the action. A time frame of 2 minutes for travel time and a time frame of 5 minutes for diagnosis time was added.

33. Based on DAEC MAAP case LII-1D4.

34. Best estimate judgement of the time to core damage.

35. Based on engineering judgement.

36. Based on the assumption that the stilling basin has inventory for approximately 30 minutes.

37. Special equipment needed are available within the available time frame only if HPCI/RCIC are available. For SBO without HPCI/RCIC the HEP is estimated to be 1.0.

38. Based on DAEC MAAP case LII-2T1.

Table 3.3-15

HEPs FOR POST-ACCIDENT OPERATOR ACTIONS QUANTIFIED
USING ASEP (NUREG/CR-4772) SCREENING METHODOLOGY

HUMAN ACTION DESCRIPTION	HEP	DESIGNATOR
Operator Fails to Shutoff HPCI/RCIC if Required	1.0	HOPAF-14 HOPAF-HC14
Operator Fails to Close Minimum Flow Valve (HPCI/RCIC)	3.5E-1	ROPAF-RCIC8 HOPAF-MO2318
Operator Inadvertently Opens RCIC Minimum Flow Valve MO2510	1E-2	ROPAF-RCIC7
Operator Fails to Reduce FW Flow Given an Isolation ATWS	1E-2	QPAF-FW-RUNBACK
Operator Fails to Control Reactor Water Level Post SCRAM	(1)	QOPAF-LEVEL
Operator Fails to Open Valve MO1546 in Control Room	(1)	QOPAF-FW05
Operator Fails to Open Bypass Valve in Control Room	(1)	COPAF-CD02
Operator Fails to Align CRD for Injection - Following containment failure	 2.5E-1	DOPAF-1P209B
Operator Fails to Restart CRD Pump 1P209A After Load Shed	1.1E-2	DOPAF-1P209A
Operator Fails to Locally Close Stored Energy Breaker for CRD Pump	1.0	DOPAF-LOCALSTART
Operator Fails to Reset SCRAM Signal	1.0	DOPAF-SCRAMRESET
Operator Fails to Align Alternate Suction	1.0	DOPAF-SUCT
Operator Fails to Open V17-0014	1.0	DOPAF-V17-0014
Operator Fails to Open LPCI Injection Valves After Failure of Automatic Signal	(1)	LOPPF-LPCIINJECT

Table 3.3-15

HEPs FOR POST-ACCIDENT OPERATOR ACTIONS QUANTIFIED USING ASEP (NUREG/CR-4772) SCREENING METHODOLOGY

HUMAN ACTION DESCRIPTION	HEP	DESIGNATOR
Low Pressure System Manually Shutoff on High Level	1.0	LOPAF-LPCI3
Operator Fails to Recognize the Need for Core Spray	(1)	SOPIF-INATENTION
Operator Fails to Initiate RHRSW	(1)	WOPAF-RHRSW
Lack of Operator Action (To Open Manual Valve (V46-0009)	1.0	GOPAF-469
Operator Intervenes and Terminates Injection (CET Node RX) (Level 2)	1E-4	XOPZF-TERMINJ
Alignment not Completed Prior to Containment Failure (CET Node RX) (Level 2)	5E-1	WOPAF-ALGNTME
Operator Fails to Recover Injection Before RPV Melt (CET Node RX) (Level 2)	9E-1	XOPAF-RECVRINJ
Operator Restores Coolant Injection After Control Rods are Melted (CET Node CZ) (Level 2)	1E-4	XOPAF-RESTRINJ
Operator Fails to Recover Low Pressure Systems (CET Node SI) (Level 2)	9E-1	XOPAF-LPSYS
Operator Fails to Recover Low Pressure Systems (CET Node TD) (Level 2)	9E-1[(2)]	XOPAF-TDCONTFL XOPAF-TDLPREC
Operator Fails to Override Drywell Spray Interlock	(1)	LOPIF-DW-INTRLOK
Operator Fails to Initiate Drywell Spray - Small LOCAs - Medium LOCA, IORV, SORV - Large LOCAs	 1.3E-2 3.7E-2 1.0	LOPPF-DW-SPRAY
Operator Fails to Initiate Torus Spray	1.0	LOPPF-TORUS-SPRAY

Table 3.3-15

HEPs FOR POST-ACCIDENT OPERATOR ACTIONS QUANTIFIED USING ASEP (NUREG/CR-4772) SCREENING METHODQLOGY

HUMAN ACTION DESCRIPTION	HEP	DESIGNATOR
Operator Fails to Initiate SLC via Alternate Boron Injection Path (RWCU)	1.0	BOPAF-SLC-INIT
Staff Does Not Check for H_2/O_2 Indication (CET Node GV) (Level 2)	(2)	UOPAF-H202INDN
Operator Fails to Close Wetwell Vent (CET Node FC) (Level 2)	(3)	NOPAF-WW-VENT
Operator Suspends Flooding Based on Erroneous Indication (CET Node FD) (Level 2)	1E-2	MOPCF-INDICTN
Operator Fails to Implement Drywell Vent Path (CET Node FD) (Level 2)	1E-2	MDWPF-DWVENT
Operator Fails to Isolate Path Given Isolation Signal Failure (CET Node IS)	1E-1	XOPAF-DWPURGE XOPAF-DWVENT XOPAF-MKUPN2DW XOPAF-WWPURGE XOPAF-WWVENT XOPAF-MKUPN2WW
Operator Fails to Initiate ESW (for Diesel Cooling)	1.0	EOPAF-ES10
Operator Fails to Bypass Filter 1S089A and B for ESW	1.0	EOPAF-V46-0019 EOPPF-V46-0024
Operator fails to Bypass Area High Temperature Trips on HPCI and RCIC	1E-2	TR(OP:BYP:TRIP)
Operator Fails to Maximize Well Water to CST	0.1	ZOPAF-MAXWELLWTR

(1) The HEP = 0.0. See Table 3.3-16.

(2) Recovery is not considered for scenarios involving Level 1 containment failures. As such, the HEP for XOPAF-TDCONTFL is set to 1.0 (see CET node TD writeup).

Table 3.3-16

VARIABLES ASSESSED IN QUANTIFICATION OF HEPs FOR POST-ACCIDENT OPERATOR ACTIONS USING ASEP (NUREG/CR-4772) SCREENING METHODOLOGY

Human Action Description	Maximum Allowable Time (T_m)	Travel and Manipulation Time (T_a)	Diagnosis Time (T_d) ($T_d = T_m - T_a$)	Median Failure Probability for Diagnosis from Figure 7-1 of NUREG-4772	Mean Failure Probability for Diagnosis P_{mean} (Td)	Failure Probability for Accomplishing the Action P(Ta) From Table 7-1 of NUREG-4772	Estimated Total Failure Probability (= P(Td) + P(Ta))
Operator Fails to Shutoff HPCI/RCIC if required	---	---	---	---	---	---	1.0(4)
Operator Fails to Close Minimum Flow Valve (HPCI/RCIC)	5 min.(5)	1 min.(22)	4 min.	2.5E-1	3.4E-1(6)	1E-2(7)	3.5E-1(25)
Operator Inadvertently Opens RCIC Minimum Flow Valve MO2510	---	---	---	---	---	---	1E-2(19), (25)
Operator Fails to Reduce FW Flow Given an Isolation ATWS	---	---	---	---	---	---	1E-2(26)
Operator Fails to Control Reactor Water Level Post SCRAM	---	---	---	---	---	---	(11), (12)
Operator Fails to Open Valve MO1546 in Control Room	---	---	---	---	---	---	(8)
Operator Fails to Open Bypass Valve in Control Room	---	---	---	---	---	---	(8)
Operator Fails to Align CRD for Injection							
- Following containment failure	2 hrs.(23)	1 min.(22)	119 min.	7E-5	5.4E-4(21)	0.25(2)	0.25
Operator Fails to Restart CRD Pump 1P209A after Load Shed	1 hr.(9)	1 min.(22)	59 min.	1E-4	8.5E-4(1)	1E-2(7)	1.1E-2
Operator Fails to Locally Close Stored Energy Breaker for CRD Pump Start	---	---	---	---	---	---	1.0(3)

Table 3.3-16

VARIABLES ASSESSED IN QUANTIFICATION OF HEPs FOR POST-ACCIDENT OPERATOR ACTIONS USING ASEP (NUREG/CR-4772) SCREENING METHODOLOGY

Human Action Description	Maximum Allowable Time (T_m)	Travel and Manipulation Time (T_a)	Diagnosis Time (T_d) ($T_d = T_m - T_a$)	Median Failure Probability for Diagnosis from Figure 7-1 of NUREG-4772	Mean Failure Probability for Diagnosis P_{mean} (Td)	Failure Probability for Accomplishing the Action P(Ta) From Table 7-1 of NUREG-4772	Estimated Total Failure Probability (= P(Td) + P(Ta))
Operator Fails to Reset SCRAM Signal (CRD)	---	---	---	---	---	---	1.0(10)
Operator Fails to Align Alternate Suction (CRD)	---	---	---	---	---	---	1.0(3)
Operator Fails to Open V17-0014 (CRD)	---	---	---	---	---	---	1.0(3)
Operator Fails to Open LPCI Injection Valves After Failure of Automatic Signal	---	---	---	---	---	---	(46)
Low Pressure System Manually Shutoff on High Level	---	---	---	---	---	---	1.0(24)
Operator Fails to Recognize the Need for Core Spray	---	---	---	---	---	---	(12)
Operator Fails to Initiate RHRSW	---	---	---	---	---	---	(16)
Lack of Operator Action (to Open Manual Valve V46-0009)	---	---	---	---	---	---	1.0(27)
Operator Intervenes and Terminates Injection (CET Node RX) (Level 2)	---	---	---	---	---	---	1E-4(28)
Alignment not Completed Prior to Containment Failure (CET Node RX) (Level 2)	---	---	---	---	---	---	5E-1(29)

Table 3.3-16

VARIABLES ASSESSED IN QUANTIFICATION OF HEPs FOR POST-ACCIDENT OPERATOR ACTIONS USING ASEP (NUREG/CR-4772) SCREENING METHODOLOGY

Human Action Description	Maximum Allowable Time (T_m)	Travel and Manipulation Time (T_a)	Diagnosis Time (T_d) ($T_d = T_m - T_a$)	Median Failure Probability for Diagnosis from Figure 7-1 of NUREG-4772	Mean Failure Probability for Diagnosis P_{mean} (Td)	Failure Probability for Accomplishing the Action P(Ta) From Table 7-1 of NUREG-4772	Estimated Total Failure Probability (= P(Td) + P(Ta))
Operator Fails to Recover Injection Before RPV Melt (CET Node RX) (Level 2)	---	---	---	---	---	---	9E-1[30]
Operator Restores Coolant Injection After Control Rods are Melted (CET Node CZ) (Level 2)	---	---	---	---	---	---	1E-4[31]
Operator Fails to Recover Low Pressure Systems (CET Node SI) (Level 2)	---	---	---	---	---	---	9E-1[33]
Operator Fails to Recover Low Pressure Systems (CET Node TD) (Level 2)	---	---	---	---	---	---	9E-1[32]
Operator Fails to Override Drywell Spray Interlock	---	---	---	---	---	---	[13]
Operator Fails to Initiate Drywell Spray							
- Small LOCA	30 min.[5]	1 min.[22]	29.0 min.	1.2E-3	3.2E-3[17]	1.0E-2	1.3E-2
- Medium LOCAs, IORV, SORV	20 min.[5]	1 min.[22]	19.0 min.	1E-2	2.7E-2[18]	1.0E-2	3.7E-2
- Large LOCAs	0 min.[5]	1 min.[22]	---	---	---	---	1.0
Operator Fails to Initiate Torus Spray	---	---	---	---	---	---	[14]
Operator Fails to Initiate SLC Via Alternate Boron Injection Path (RWCU)	---	---	---	---	---	---	1.0[39]
Staff Does Not Check for H_2/O_2 Indication (CET Node GV) (Level 2)	---	---	---	---	---	---	[35]

Table 3.3-16

VARIABLES ASSESSED IN QUANTIFICATION OF HEPs FOR POST-ACCIDENT OPERATOR ACTIONS USING ASEP (NUREG/CR-4772) SCREENING METHODOLOGY

Human Action Description	Maximum Allowable Time (T_m)	Travel and Manipulation Time (T_a)	Diagnosis Time (T_d) ($T_d = T_m - T_a$)	Median Failure Probability for Diagnosis from Figure 7-1 of NUREG-4772	Mean Failure Probability for Diagnosis P_{mean} (Td)	Failure Probability for Accomplishing the Action P(Ta) From Table 7-1 of NUREG-4772	Estimated Total Failure Probability (= P(Td) + P(Ta))
Operator Fails to Close Wetwell Vent (CET Node FC) (Level 2)	---	---	---	---	---	---	[36]
Operator Suspends Flooding Based on Erroneous Indication (CET Node FD) (Level 2)	---	---	---	---	---	---	1E-2[37]
Operator Fails to Implement Drywell Vent Path (CET Node FD) (Level 2)	---	---	---	---	---	---	1E-2[38]
Operator Fails to Isolate Path Given Isolation Signal Failure (CET Node IS) (Level 2)	---	---	---	---	---	---	1E-1[40]
Operator Fails to Initiate ESW (for diesel cooling)	---	---	---	---	---	---	1.0[15]
Operator Fails to Bypass Filter 1SO89A and B for ESW	---	---	---	---	---	---	1.0[15]
Operator Fails to Bypass Area High Temp Trips on HPCI and RCIC	---	---	---	---	---	---	1E-2[43]
Operator Fails to Maximize Well Water to CST	---	---	---	---	---	---	0.1[42]

Notes To Table 3.3-16:

(1) Based on error factor of 30 (Error factor = Upper Bound/Median = 3E-3/1E-4 = 30) and the assumptions of lognormal distribution.

(2) Based on the assumption that the operator has to perform a critical procedural action correctly under extremely high stress (see NUREG/CR-4772).

(3) Required action outside of control room; therefore, probability is set to 1.0 (see NUREG/CR-4772).

(4) The probability of failure of automatic high level trip of HPCI is very low. A conservative screening value is used in this assessment.

(5) Based on engineering judgement.

(6) Based on error factor of 3.6 (Error factor = upper bound/median = 0.9/0.25 = 3.6) and the assumption of lognormal distribution.

(7) Based on performing a post-diagnosis immediate emergency action for the reactor vessel/containment critical parameter, when a) it can be judged to have been committed to memory, b) it can be classified as skill-based action for Table 2-1 of NUREG/CR-4772, and c) there is a backup written procedure. (see Table 7-3 of NUREG/CR-4772)

(8) This operator action is included in the derivation of "operator failing to manually initiate FW/condensate." See Table 3.3-9.

(9) Time takes to go from normal water level to 1/3 core height. Conservative estimate.

(10) Resetting the scram signal is cited in IPOI5 (Reactor Scram). It is believed that the operator will not perform the action in time and therefore a HEP of 1.0 is conservatively assigned to this action.

(11) It is assumed that this operator action applies to scenarios in which RPV makeup is being supplied by only low pressure injection systems (i.e., high pressure ECCS have an automatic high RPV water level trip). Refer to quantification of operator action "Low Pressure System Manually Shutoff on High Level."

(12) This operator action is considered an integral part of a manual action to initiate low pressure core spray. Refer to operator action "Operator Fails to Initiate Core Spray" in the RMIEP quantification table. (Table 3.3-9).

(13) This operator action is considered an integral part of a manual action to initiate DW spray. Refer to operator action "Operator Fails to Initiate Drywell Spray" in this table.

(14) Torus spray is not considered a sufficient means to establish containment vapor suppression. Refer to functional success criteria for LOCA events.

(15) The automatic startup of ESW for diesel cooling is judged to be reliable. The HEP for failure to manually initiate ESW for diesel cooling is conservatively judged to be 1.0.

(16) This operator action is considered an integral part of a manual action to initiate alternate low pressure makeup systems upon failure of low pressure ECCSs. Refer to operator action "Operator Fails to Align Alternate Injection Sources" in the RMIEP quantification table. (Table 3.3-9).

(17) Based on error factor of 10 (Error factor = upper bound/median= 1E-2/1E-3 = 10) and the assumption of lognormal distribution.

(18) Based on error factor of 10 (Error factor = upper bound/median = 1E-1/1E-2 = 10) and the assumption of lognormal distribution.

(19) Approximate HEP for error of commission.

(20) Not used.

(21) Based on error factor of 28 (Error Factor = Upper Bound/Median = 2E-3/7E-5 - 28) and the assumption of lognormal distribution.

(22) The time frame of 1 minute includes travel and manipulation time.

(23) Since it does not appear to be any limitation to provide makeup to the RPV at elevated containment pressure, assuming that the operator has been unable to establish containment heat removal, it is possible that the operator has been able to maintain RPV coolant inventory using the CRD system until containment breach. Upon containment failure, and depending on the failure mode, operating injection systems may be assumed to be temporarily unavailable. In this situation, it is judged that the operator could have several hours to restore makeup to the RPV before core damage is induced. However, for this assessment, a time frame of two hours post containment failure is conservatively assumed to be the limit for the operator restoring injection to the RPV using CRD.

(24) System is of sufficient capacity to generally result in a guaranteed restoration of water level to a high RPV level. Operator in an effort to control water level will terminate the low pressure system injection.

(25) The HEP for this human action is judged not to have any impact on the overall model.

(26) This is a proceduralized and practiced action. Judgement is utilized in this number. A detailed evaluation would be expected to yield at least an order of magnitude lower number.

(27) Manual operation of a manual valve outside control room; therefore, the HEP is set to 1.0.

(28) This is the crucial mistake with the operators at Three Mile Island made in the 1979 accident. It considered an unlikely action; operator training has improved greatly since that time. In addition, this is an act of commission, typically not included in a PRA. However, since it may be considered a "classic" or highly visible action, it is included in the analysis and assigned a low probability similar to SHARP "skill" based assessment, by using the ASEP model as the source for the mean estimate (i.e., a time frame of approximately 2 hours is assumed).

(29) Judgement is used to assign a 50% chance that the cross-tie valves in the reactor building have not been aligned prior to the containment failure and, as a result, have been rendered inoperable due to the harsh environment.

(30) Judgement: If failed systems have not been repaired/restored at this point in the accident, it is assumed relatively unlikely that the system(s) will be restored before RPV melt-through (~ 1 - 2 hours depending on the sequence). It is assumed that system restoration is more involved than simply bypassing a system trip from the control room. Recovery is considered impossible for accidents in which the primary containment has already failed.

(31) A small time window exists between the time when the control rods begin to melt until the fuel rods also begin to melt. Injection of water during this time frame could create a large reactivity excursion. This event models the possibility that the operator restores injection within this small time frame. Judgement is used to assign a low probability of coincidental occurrence.

(32) If the accident has progressed to this point with injection systems failed, it is assumed that some major problem exists (e.g., equipment failure, debris plugging of suction lines, during AC power) precluding the operator from repairing these systems in the time frame of interest. This estimate is also supported by the fact that many areas of the reactor enclosure may be inaccessible at this point in the accident. The probability of failure is conservatively estimated assuming that there are many hours (i.e., 6 - 10 hours) available prior to the postulated containment over-temperature failure.

(33) Following failure to recover injection systems to prevent RPV vessel melt-through, a small time window is available (~ 30 minutes) for system recovery before shell attack occurs. A value of 0.9 is assigned for the conditional probability that recovery is not successful in the short time window. Recovery is not considered possible for accidents in which the primary containment has already failed.

(34) Not used.

(35) This event is included in the model for illustrative purposes. It is not quantified explicitly because it is considered subsumed by NOPPF-GASVENT.

(36) This event is assumed to be subsumed by the overall operator action XOPPF-OPACTION (See Section C.9 of CET writeup). As such, this event is shown for illustrative purposes and is not included in the quantitative analysis.

(37) As the containment is flooded, the operating crew must monitor containment pressure and water level to ensure that the PCPL and MPCWLL limits are not exceeded. It is conceivable that if either indication failed, the operator would suspend flooding due to the uncertainty as to the status of the containment. Therefore, the conditional probability of 1.0E-2/demand assigned to this event primarily accounts for the possibility that containment conditions are erroneously indicated in the control room before the containment water level reaches TAF (i.e., the conditional probability that the operator would suspend flooding is conservatively assessed as 1.0/demand).

(38) Implementation of the containment flooding contingency procedure does not alleviate the responsibility of the operator from maintaining containment conditions within acceptable limits

throughout the evolution. In fact, as containment water level rises, the possibility that non-condensible gases become concentrated in the drywell to the point where overpressure becomes a concern also increases. Therefore, this action is defined as the operation of drywell vent path(s) to relieve containment overpressure and maintain containment integrity during the course of the flooding evolution. The time frame available to the operator to successfully implement containment pressure control is defined by the point at which unmitigated overpressure conditions result in containment breach. This time period is conservatively estimated to be 30 min. (Note that combustible gas concentration and the potential for hydrogen combustion was not considered when determining the allowable time frame for operator action, since the containment is assumed to have remained isolated.) Given the dependence on two senior control room operators to recognize the conditions and initiate this action under stressful conditions, it is determined that the conditional probability for failing to vent the drywell is 1.0E-2/demand. This assessment is considered to be a conservative application of the ASEP methodology.

(39) The action is an outside control room action and as such the HEP is conservatively set to 1.0

(40) This estimate is based on the WASH-1400 HRA methodology. The operator has approximately 1 to 2 hours following accident initiation in most cases before containment isolation becomes critical. It is assumed the situation would be moderately stressful; thereby, affecting operator performance. This assumption may not apply to faster developing accidents, such as ATWS, but the action was not quantified for these different cases because this action is a negligible contributor to risk - even at a probability of failure value of 1.0

(41) This action is judged to be subsumed by the operator action "operator fails to manually initiate LPCI following auto initiation failure".

(42) This action is proceduralized and has a long duration to implement. A detailed evaluation could justify a much lower number. However, a conservative estimate of 0.1 for failure was used since the quantification showed that a more accurate evaluation was not needed.

(43) This is a proceduralized and practiced action. If it is not done and the pumps trip, the temperature trips can then be bypassed and the pumps restarted. Using previous experience the value of 1E-2 is used and is felt to be conservative.

Table 3.3-17

HEPs FOR PRE-ACCIDENT OPERATOR ACTIONS QUANTIFIED USING ASEP (NUREG/CR-4772) SCREENING METHODOLOGY

HUMAN ACTION DESCRIPTION	HEP (mean)	DESIGNATOR
HPCI auto reset not reset	2.7E-5	HOPAF-1
Operator fails to take action to empty drain pot	2.7E-5	HOPAF-IN-DRN-POT ROPAF-IN-DRN-POT
RCIC mechanical overspeed trip not reset following test and maintenance	2.7E-5	ROPAF-MECH-OVSPD
Operator fails to notice low basin water level	8E-4	WOPIF-RWS-START
Tour by operator fails to uncover low level in SLC tank (once a day)	2.7E-5	BOPPF-SL09
Operator fails to respond to low level indications in the SLC tank	2.7E-5	BOPAF-SL10
Miscalibration of level instrumentation used to initiate the HPCI/RCIC/LPCI/CS pumps	8E-5	ILIMF-RXLVL
Miscalibration of pressure instrumentation used to initiate the HPCI/RCIC/CS/LPCI/Recirc. Pumps	8E-5	ILIMF-DWPRS I-LOW-PRES-PERM IPIMF-RCPA IPIMF-RCPB I-LPCI-SELECT ILIMF-LOOPSEL

Table 3.3-18

VARIABLES ASSESSED IN QUANTIFICATION OF PRE-ACCIDENT OPERATOR ACTIONS USING ASEP (NUREG/CR-4772) SCREENING METHODOLOGY [1]

Human Action Description (Designator)	Applicable Procedure	Initial Basic HEP[1]	Recovery Factor (RF) Determination				Final Basic HEP (= Initial Basic HEP x RFs) (median)[2]	Final Basic HEP (= Initial Basic HEP x RFs) (mean)[3]
			Compelling Signals?	Post-Test not Performed or Not Effective?	Written Check or Verification?	Shift of Daily Check of Component Status?		
HPCI auto reset not reset (HOPAF-1)	ARP 1C03C-A3	0.03	0.1	0.01	0.1	0.1	1E-5	2.7E-5
Operator fails to take action to empty drain pot (HOPAF-IN-DRN-POT) (ROPAF-IN-DRN-POT)	ARP 1C03C-D8	0.03	0.1	0.01	0.1	0.1	1E-5	2.7E-5
RCIC mechanical overspeed trip not reset following test and maintenance (ROPAF-MECH-OVSPD)	ARP 1C04C-A5	0.03	0.1	0.01	0.1	0.1	1E-5	2.7E-5
Operator fails to notice low basin water level (WOPIF-RWS-START)	ARP 1C06A-D1, D2, D11	0.03	0.1	N/A	N/A	0.1	3E-4	8E-4

Table 3.3-18

VARIABLES ASSESSED IN QUANTIFICATION OF PRE-ACCIDENT OPERATOR ACTIONS USING ASEP (NUREG/CR-4772) SCREENING METHODOLOGY [1]

Human Action Description (Designator)	Applicable Procedure	Initial Basic HEP[1]	Recovery Factor (RF) Determination				Final Basic HEP (= Initial Basic HEP x RFs) (median)[2]	Final Basic HEP (= Initial Basic HEP x RFs) (mean)[3]
			Compelling Signals?	Post-Test not Performed or Not Effective?	Written Check or Verification?	Shift of Daily Check of Component Status?		
Tour by operator fails to uncover low level in tank (once a day) (BOPPF-SL09)	ARP 1C05A-E3	0.03	0.1	0.01	0.1	0.1	1E-5	2.7E-5
Operator fails to respond to low level indications in the SLC tank (BOPAF-SL10)	ARP 1C05A-E3	0.03	0.1	0.01	0.1	0.1	1E-5	2.7E-5
Miscalibration of level instrumentation used to initiate the HPCI/RCIC/LPCI/CS pumps (ILIMF-RXLVL)	STP 42 Series	0.03	N/A	0.01	0.1	None	3E-5	8E-5

Table 3.3-18

VARIABLES ASSESSED IN QUANTIFICATION OF PRE-ACCIDENT OPERATOR ACTIONS USING ASEP (NUREG/CR-4772) SCREENING METHODOLOGY (1)

Human Action Description (Designator)	Applicable Procedure	Initial Basic HEP(1)	Recovery Factor (RF) Determination				Final Basic HEP (= Initial Basic HEP x RFs) (median)(2)	Final Basic HEP (= Initial Basic HEP x RFs) (mean)(3)
			Compelling Signals?	Post-Test not Performed or Not Effective?	Written Check or Verification?	Shift of Daily Check of Component Status?		
Miscalibration of pressure instrumentation used to initiate the HPCI/RCIC/LPCI/CS/Recirc pumps (ILIMF-DWPRS) (I-LOW-PRES-PERM) (IPIMF-RCPA) (IPIMF-RCPB) (I-LPCI-SELECT) (ILIMF-LOOPSEL)	STP 42 Series	0.03	N/A	0.01	0.1	None	3E-5	8E-5

(1) The initial basic HEP is estimated at 3E-2 for all pre-accident actions based on NUREG/CR-4772. The procedures are considered to be excellent and easy to follow; therefore, no upward adjustment was performed on the basic HEPs.

(2) For cases where all the RFs apply, a negligible HEP is assessed due to the excellence of the RFs. For these cases, a conservative HEP of 1E-5 is used.

(3) The conversion of median HEPs to mean HEPs is based on the assumption of lognormal distribution with error factor of 10.

Table 3.3-19

VARIABLES ASSESSED IN PRE-ACCIDENT OPERATOR ACTIONS USING ASEP (NUREG/CR-4772) SCREENING METHODOLOGY
DAEC INTERVIEWEES: SRO No. 1 and No. 2

Human Action Description	What are the Applicable Procedure for this Human Action?	Are the procedures well written, clear, and easy to follow?	Recovery Factor (RF) Determination			
			Is component status indicated in the control room by some "Compelling Signals" such as an annunciator when the maintenance or calibration task or subsequent test is finished or before normal power operation can be resumed?	Is component status verified by a post-maintenance or a post-calibration test?	Is there written check by a second person to directly verify component status after completion of a maintenance or calibration task?	Is there a requirement for a shiftly or daily check of component status (in or outside of the control room) using a written list?
HPCI auto reset not reset	ARP 1C03C-A3	YES	Yes, Amber Light on Panel 1C03	YES[1]	YES[1]	Yes, Shiftly
Operator fails to take action to empty drain pot	ARP 1C03C-D8	YES	Yes, High Level Alarm	YES[1]	YES[1]	Yes, Valve Position Checked Shiftly
RCIC mechanical overspeed trip not reset following test and maintenance	ARP 1C04C-A5	YES	Yes, RCIC Trip Annunciator	YES[1]	YES[1]	Yes, Shiftly
Operator fails to notice low basin water level	ARP 1C06A-D1, D2, D11	YES	Yes, RHRSW/ESW/Circulating Water Pit Low Level Alarm	N/A	N/A	Yes, Shiftly
Tour by operator fails to uncover low level in SLC tank (once a day)	ARP 1C05A-E3	YES	Yes, Low Level Alarm and Local Indications	YES[1]	YES[1]	Yes, Shiftly Check on SLC Level and Temperature[2]
Operator fails to respond to low level indications in the SLC tank	ARP 1C05A-E3	YES	Yes, Low Level Alarm	YES[1]	YES[1]	Yes, Shiftly Check on SLC Level and Temperature[2]
Miscalibration of level instrumentation used to initiate the HPCI/RCIC/LPCI/CS pumps	STP 42 Series	YES	N/A	YES[1]	YES[1]	Yes, Periodically Per Surveillance Test Procedure
Miscalibration of pressure instrumentation used to initiate the HPCI/RCIC/LPCI/CS/Recirc. pumps	STP 42 Series	YES	N/A	YES[1]	YES[1]	Yes, Periodically Per Surveillance Test Procedure

[1] Maintenance post test and surveillance test (Partial) is performed for safety systems. It required 2 licensed operators.
[2] Daily.

3.3.4 Common Cause Failure Analysis

3.3.4.1 Purpose

The unavailability of complex redundant systems (such as those at a nuclear power plant) will be understated if common cause is not considered. The purpose of this section is to describe the approach taken by the DAEC PRA to evaluate the common cause contribution to system unavailability.

3.3.4.2 General Discussion and Background

In analyzing the potential for common cause failures, a search is made for common cause component groups and mechanisms which could lead to common cause failures of those component groups. Generally, PRAs restrict the analysis of common cause failures to redundant active components. For example, the following types of components are examined as part of the common cause failure analysis:

* Motor operated valves
* Motor driven pumps
* Safety relief valves
* Air operated valves
* Diesel generators
* Batteries
* Etc..

The complexity of common cause failure mechanisms, and the difficulty of observing and measuring these mechanisms, have led PRA analysts to treat common cause failures in almost an exclusively quantitative way. That is to say, current PRAs attempt to estimate the probability of common cause failures due to all possible mechanisms without specifically analyzing the nature of those underlying mechanisms. Although this analytical approach fails to specifically address the physical mechanisms which can result in common cause failures, it is based on actual operating data and seems to lead to reasonable quantitative estimates of the probabilities of common cause failures.

There have been a number of parametric models used to estimate common cause failure probabilities. NUREG/CR-4780 splits these models into "Single" and "Multiple" groups. There is essentially only one single parametric model and this is referred to as the "Beta Factor" method. A variant of the beta factor model is the "C" factor method. The multiple parametric models include the "Alpha Factor", "Multiple Greek Letter", and the "Binomial Failure Rate" methods.

Of these models, the beta factor method has come to be used most often. As stated from NUREG/CR-4780, "Although historical data collected from the operation of nuclear power plants indicate that common cause events do not always fail all redundant components, experience from using this simple model shows that, in many cases, it gives reasonably accurate (only slightly conservative) results for redundancy levels up to about three or four items. However , beyond such redundancy levels, this model generally yields results that are conservative." Because of its extensive use, simplicity, and conservatism the beta factor method has been chosen for use in modeling common cause in the DAEC PRA.

3.3.4.3 Description of the Beta Factor Method

A beta factor is defined as the conditional probability of a common cause failure of a component given that a "similar" component previously failed. In this context, two components are deemed to be similar if they are redundant components of the same type, operating under the same conditions. Beta factors are developed based on the following equation:

$$\beta = \frac{\text{\# common cause failures}}{\text{total component failures}}$$

EPRI NP-3967 indicates that, in practice, when data is collected to calculate the beta factor, common cause failures tend to be more completely reported in data sources than do individual independent failures. Therefore, the denominator of the above equation tends to be slightly smaller than it should be. This gives a larger (and therefore a more conservative) beta factor than is the actual case.

To demonstrate how a beta factor would be calculated, a simple example is given. Imagine that two redundant motor driven pumps are tested a total of 10,000 times and that a total of 100 pump failures are observed. In those 100 failures there were 10 occasions on which both of the pumps failed in common mode. The beta factor for this type of motor driven pump would be calculated as:

$$\beta = 10/100 = 0.1$$

Again, a beta factor of 0.1 should be interpreted as a conditional probability that a second motor driven pump will fail given that the first pump already is failed. Thus, the probability of two redundant components failing due to common cause contributors is calculated as:

$$C = \lambda \times \beta$$

where:

c = common cause failure rate
λ = independent failure rate
β = beta factor

Applying this formula to the above example, the probability that both motor driven pumps fail due to common cause, given an independent failure rate of a motor driven pump of 1E-2 is calculated as:

$$C = 1E\text{-}2 \times 0.1 = 1E\text{-}3$$

3.3.4.4 Beta Factors to Be Used in the DAEC PRA

Based on reviews of key components and initial quantifications of the Level 1 models Table 3.3-20 was produced for components to be evaluated for common cause in the DAEC PRA.

The beta factors given in the table will be used to calculate a common cause failure rate for the components identified. This will be done in a fashion as described earlier in this section (in some cases a common cause failure rate is given in the table instead of a beta factor). The common cause failure rates are then entered into the fault tree models as a basic event for common cause failure of a given component. In the case of the DAEC PRA, the modes of common cause failures that are modeled are those associated with an active component's failure to start or change its state at a required time.

Table 3.3-20
Common Cause Beta Factors

FAILURES	β Factor Proposed	SOURCE
Diesel Generators	0.077	NUREG/CR-2099, & K.N. Fleming
Batteries	mission 0.04 standby 0.001	ASEP NUREG-0666
ESW Pumps	0.03	EPRI NP-3967
ESW Discharge Valves	0.08	EPRI NP-3967
MOVs on Cooling Water Suction and Discharge (2 sets of Valves)	0.08	EPRI NP-3967
Safety Relief Valves	0.22	EPRI NP-3967
ADS Function	1E-4	Based on Precursor Events
SLC Pumps	0.17	Based on review of SLC LER's 1980 - 1990
SLC Squib Valves	+	EPRI-3967
Feedwater Pumps (Restart)	0.05	See RHR/CS Pumps
HPCI/RCIC (Turbine Driven)	0.1	Judgement; EPRI NP-3967 and ASEP
Core Spray Pumps	0.05	Judgement; EPRI NP-3967 and ASEP
RHR Pumps	0.05	Judgement; EPRI NP-3967 and ASEP
RHRSW Pumps	0.03	EPRI NP-3967
River Water Supply Pumps	0.0375	EPRI NP-3967
Control Building HVAC Fans	0.13	NUREG/CR-4780
Chillers	0.11	NUREG/CR-4780

+ A value of 0.014 has been derived from operating experience for the combination of 2 squib valve failures. This is not the β factor, it is the total SQUIB valve unavailability. This valve is directly the increment to be added to the fault tree.

The approach used to evaluate common cause for the DAEC PRA is considered to be applicable and conservative. DAEC PRA project team members have evaluated the types of failures reported to the NPRDS data base for the Duane Arnold plant and have compared them to those reported for other plants. The conclusion was reached that Duane Arnold was reasonably well represented by generic data. In addition to the conservatism indicated by EPRI NP-3967, the independent failure rates (λs) used for the DAEC PRA include all failures (including those due to common cause).

3.3.4.5 Conclusions

The DAEC PRA uses the beta factor method for modeling common cause. The beta factor method is an inherently conservative approach that is widely used in the industry. The DAEC PRA models the common cause failure to start or change state for a list of selected active components. The selected components were arrived at by reviewing the plant design and initial quantifications of the Level 1 PRA. Generic data was used after a review by project personnel indicated that it was appropriate. Additional conservatism is modeled into the common cause failure analysis of DAEC PRA in that the independent failure rates include all failures (including common mode failures).

3.3.5 Quantification of Sequence Frequencies

The purpose of this section is to describe the methods and computer codes used to quantify the event sequence frequencies for the DAEC IPE study.

3.3.5.1 General Methodology

The basic methodology used in the DAEC Level I IPE modeling is the "large fault tree - small event tree" methodology. Each event tree function was modeled by a combination of fault tree system models, including full linking to the support system models. The modeling was carried out to the level of detail supported by the data available, i.e. down to major components. Human actions, system availability, and common cause contributions were explicitly included in each of the system fault trees.

3.3.5.2 Systems Quantification

The system models were built and maintained in the CAFTA[1] computer code. The database of basic event values was also maintained using CAFTA. Each of these models contain all of the logic necessary to create all versions of the system fault trees used in the quantification of the DAEC IPE. Each of the system fault trees were merged with the top logic files to create a single logical model of the DAEC. This ensured that all external system transfers were resolved prior to event tree quantification.

A CAFTA macro file was developed for the quantification of each version. (See Section 3.1.2 Event Sequence Analysis for a discussion of the node quantifications.) Cutsets were then generated for each version of the top logic used in the event sequence

[1]The CAFTA code was developed by SAIC. The version used is 2.0e. DAEC used the version of SAIC's codes that were written specifically for the INTEL 80386 CPU in all cases unless no specific 80386 version was available.

quantification by the CUT386 code. These are stored in the format used by CUT386; a report of the cutsets for each nodal quantification was also generated.

Combinations of events that are not possible or are prohibited by the DAEC Technical Specifications, or mutually exclusive events, are treated in the event tree quantification process.

All system models were quantified with a truncation value of $3.0x10^{-11}$. No modularization was used.

3.3.5.3 Event Tree Quantification

The event trees were quantified using a fault tree linking process. This ensures that dependencies between systems, and their associated support systems, are explicitly accounted. Each of the core damage sequences were represented by the logical "AND" of each of the failed (down branch) nodes. The success (up branch) nodes were credited by deleting those cutsets that would be sufficient to fail a success branch from the final list of sequence cutsets. Mutually exclusive cutsets are also removed during the sequence quantification.

The computer code used to perform the fault tree linking was SEQUENCE. It is a code developed by DAEC personnel specifically for linking CAFTA generated cutsets. It was developed after size limitaticns in the SAIC fault tree linking functions were discovered. It was developed using the Lahey Protected Mode FORTRAN Compiler, F77l3-EM32.

Results from SEQUENCE are routed back into the CAFTA cutset editor, CSED386. Reports of the cutsets, event tree sequence frequencies, and basic event importance were generated by this code. Summaries of these reports are found in the various graphical representations of the results found through out this document.

Once again, all sequences were quantified with a truncation value of 3.0×10^{-11}. No modularization was used.

3.3.6 Internal Flooding Analysis

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 and Surry IPE-PRA (both PWRs), however, concluded flooding was a dominant contributor to the total core damage frequency. Subsequently, the plant made modifications as a result. Other plants have experienced maintenance events which have resulted in the flooding of equipment. All these factors provide the basis for performing the DAEC internal flooding analysis.

The purpose of the internal flooding analysis is to determine potential vulnerabilities due to flooding from sources such as tank overfilling, hose and pipe ruptures, and pump seal leaks. The analysis uses bounding, frequently conservative assumptions while still demonstrating a low potential for core damage. Attention is focused on the major flood sources in the plant which could affect multiple systems and flood initiators which are bounded by other flooding events are given less consideration.

The study, utilizing very conservative estimates, produced no credible flooding sequences having a frequency greater than 1E-7 per year. It was, therefore, concluded that there were no significant contributions to core damage frequency by internal flooding events.

Six areas/events were identified as posing potential risk to plant operations. The first flooding event involves condenser bay events. The second event involves flooding events in the turbine building basement. The third event involves main feedwater events within the steam tunnel. The fourth event is inadvertent actuation of the fire deluge system in the reactor building (786 elevation). The fifth event is flooding events in the torus area. The sixth event is a catastrophic rupture of the circ water piping in the pumphouse. No flooding initiators were identified that by themselves disabled core cooling.

3.3.6.1 Background

Considerable review of the DAEC plant design and operating procedures has been performed in the past with respect to the potential and effects of internal flooding.

The UFSAR High-Energy Line Break (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 and RCIC could 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.

INPO requested utilities perform an analysis for vulnerabilities due to internal flooding events in Significant Operating Event Report (SOER) 85-5. The DAEC response to SOER 85-5 indicates that an analysis of localized internal flooding was performed at DAEC. This review did not identify any credible events where safe shutdown capability would be impaired. Additionally, the document indicates that the concerns raised by INPO are also addressed in several previous documents, such as in various amendments to FSAR and NRC Circular 78-06 and Information Notice 83-44 (potential for common mode flooding of redundant equipment via drain lines). The document indicates that all such concerns have been resolved. Finally, it indicates that the EQ Program and response to Bulletin 79-01B questions have been addressed.

The Fire Hazards Analysis performed by DAEC (NG-84-4027) in 1984 revealed two areas in the plant where both divisions of safe shutdown equipment could be affected. Modifications to those areas have since addressed this matter and the issue is resolved.

Plant design safety features and general plant configuration was re-examined for vulnerabilities due to internal floods. Additionally, various related documents, such as the Abnormal Operating Procedures on flooding and the Emergency Operating Procedures were reviewed.

In the course of this study events analyzed and evaluated previously in other evaluations, for example, high-energy line breaks in the steam tunnel, were used. And these analyses were not re-evaluated. Information obtained from such documents was considered true and accurate of plant configuration.

3.3.6.2 Process

For the purpose of performing the DAEC 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.

Plant walkdowns were conducted for each zone and each potential flooding source to qualitatively review 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, communication pathways for propagation to and from other areas, door arrangement, curbs, capacity of sources to initiate flooding, and credible proposed localized flood levels. 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. In determining potential for initiating events for flooding normally running systems, systems with automatic start features, or systems that could drain by gravity were given most consideration while systems that are normally in standby and do not have automatic start capability were given less consideration as potential flood sources.

For each flood zone for which drainage is credited, analyses were performed to estimate the flooding rate an area could tolerate considering factors such as floor drains, sump capacity, and door leakage. This review is addressed in the Fire Hazards Analysis Report as per the Appendix R Fire Protection effort.

The screening process used for the DAEC internal flooding analysis, very conservative assumptions were used, while much evaluation was qualitative to conserve resources expended the internal flooding analysis effort. The screening process used is described below.

Screening Process

(1) Initial screening

The facility and flooding potential was viewed from several overlapping perspectives: source oriented, target oriented, and special topic oriented. Source oriented evaluations made an assessment of potential for flooding in various zones and where such water might propagate. Target oriented evaluations identified zones where safety-related equipment is located and examined the potential for flood water to propagate into these zones. Special topics for common mode failures or spatial/system interactions (including operator action, drains, and venting) were investigated in regard to flood initiation, propagation, detection, and mitigation.

(2) Review of areas

A detailed review of areas/zones not immediately eliminated by Step 1 was performed. Zones were reviewed for general configuration and potential significant sources without drainage. Consequence of flooding were also evaluated. Potential and consequences were rated qualitatively as high, medium, or low. Areas rated of at least medium potential for flooding and at least medium consequences were screened for further evaluation.

(3) Evaluation of risk-significant areas and topics

Risk-significant zones were evaluated in greater detail. In some instances, a simple fault-tree was constructed, engineering judgement was used to estimate initiating event frequencies, documentation of independent flooding studies consulted. The following six scenarios were examined in this step.

Six scenarios examined in more detail as follows.

Potentially Risk Significant Flood Zones

Turbine Building Condenser Bay

Flooding events within the condenser bay area are bounded by the loss of feedwater initiator. In this sequence a large break in the main condenser is also considered to occur. Due to a loss of condenser vacuum early closure of the Main Steam Isolation Valves will occur. The large flood from the condenser will directly affect the following systems: loss of 1A2 electrical supply, loss of instrument air (due to the flooding of the air dryers), loss of the condensate storage tanks (except for the dedicated safety system volume), and loss of minor systems.

Turbine Building Basement

Flooding events within the turbine building basement are bounded by the loss of condenser vacuum on the circ water side. Direct flooding of this zone will lead to an early loss of condensate/feedwater and loss of 1A2 electrical supply. Loss of instrument air will occur due to flooding of the air dryer. Pumphouse draining will lead to a loss of GSW and a loss of fire water. A loss of minor system is considered to occur.

Steam Tunnel

Flooding events within the steam tunnel are addressed in the HELB analyses performed by the Bechtel Corporation as well as in the HELB analysis for the UFSAR. A feedwater line break bounds other such high energy/capacity line breaks in the steam tunnel. Because this event is identified in several documents for analysis, these documents provide the basis for identifying this event for evaluation. Investigation of the P&IDs relating to plant layouts and utilizing plant knowledge by several key personnel led to the conclusion that a credible flood in the steam tunnel due to a feedwater line break would cause minimal damage to equipment and piping systems located in the steam tunnel area. However, the flood would propagate into the turbine building basement areas, particularly the heater bay. Such events have been considered in a separate evaluation that would bound any high-energy line break in the steam tunnel and are postulated as sequences evaluated for the turbine building.

Reactor Building, 757' Elevation

Flooding events in the reactor building at Elevation 757' are thought to be bounded by the inadvertent actuation of the fire deluge system. The Fire Hazards Analysis performed to specifications for Appendix R takes this event into account. The conclusions of this report (at the time of publication) indicate certain modifications necessary to prevent loss of redundant trains of equipment needed for safe plant shutdown capability. The aforementioned modifications have since taken place and pose no risk to the core damage frequency.

Torus Area

The torus room area is an open room area that encloses the primary containment torus (suppression pool). The torus room is isolated from the surrounding ECCS corner rooms by concrete walls and watertight doors. This arrangement makes the torus room floodable to above the ECCS suction and quencher elevation given a torus rupture or leak below the water line. The torus room wall on elevation 716' is marked for water level measurement purposes in the torus room. Some ventilation penetrations exist between the torus and adjacent ECCS rooms but are at elevations substantially above the floor. The torus room itself has the capacity to retain substantial volumes of water equivalent to the full volume of torus water without propagation to adjacent rooms. Multiple systems pass through this area, but little equipment is on the torus room floor elevation. Level instrumentation exists in the room for detection of flooding. The torus area can receive flood water from the RHR valve room and the 757' general area (access plates). An opening in the south wall allows direct communication to the radwaste tank room (just above the RCIC room).

Due to the size of the room, the fact that it is built for flooding, and the fact that the area it has flooding indications and alarms in it, the torus area would require a flooding incident of such magnitude and duration to propagate from this room to affect the adjacent areas that this event did not seem credible as a flooding event that could jeopardize safe shutdown of the plant. Therefore, this event is screened from quantification.

Pumphouse

A flooding event initiated by a large rupture of the pumphouse circ water piping can occur, and had been identified for further evaluation. This event could cause a loss of circulatory water and/or GSW and possibly a loss of ESW. Eventually a loss of condenser vacuum and loss of feedwater would occur. Feedwater would be lost due to the loss of GSW cooling to the FW system. Additionally, the flooding source is of such magnitude and duration that some other safety-related equipment could be lost.

Further review of the pumphouse revealed that the bellows on the circ water pumps is the most likely mode of failure for the circ water system. However, due to the configuration of the pumphouse, isolation of the circ water system from either of two isolated trains of essential water supplies, consideration of likely propagation paths for waters to the environment, and location of essential equipment substantially above flood levels considered credible, it was concluded that the initiating frequency for a flooding event of any consequence is of such low magnitude (1E-7) that further evaluation is not warranted.

In evaluating the zones identified, the area was reviewed against existing applicable documentation and plant knowledge used to bound the most limiting or credible events. From the final two sequences identified for quantification, the Level 1 PRA results were modified to reflect the effects toward the total loss of the local equipment contained in the zone and propagation to affect equipment in adjacent areas. Discussion of the two dominant sequences is provided in Section 3.3.6.3.

Only two of these six events survived this screening criterion. These were the two turbine building events.

(4) Quantification

The two surviving events were quantified, using the existing PRA model, by proposing a most limiting sequence for the event identified. Very conservative assumptions were made about the quantification of these sequences. For example, the flood zones affected were evaluated by considering maximum consequences for flooding in the zone and maximum potential for flooding propagation. Moreover, it was conservatively assumed that all safety-related systems located in the flood zones would fail, with no chance for survival. (In fact, several systems are located above the flood level for the two turbine building events and would likely survive.)

Based on conservative assumptions and plant-specific knowledge,including plant walkdowns, reduced these scenarios to only two as described below.

Event 1: **Turbine building condenser bay event**

For the turbine building event quantified, very conservative assumptions were made. These assumptions were maximum consequences of flooding the zone evaluated, maximum potential for flood propagation, complete failure of all safety-related equipment located in the flood zone. No credit is taken for other systems that might help mitigate the flooding event, nor was credit taken for recovery of any safety-related systems. Safety-related equipment in these areas, for the most part, are located several feet above the floor, substantially above the flood level. Despite these conservative assumptions, quantification of this event yields about 1E-7 toward core damage frequency. Because this sequence was not modeled in detail sufficient to be representative of plant abilities to withstand such a flooding event, it is expected that this event would actually contribute significantly less to plant risk than 1E-7 per reactor year.

<u>Event 2:</u> **Turbine building basement event**
Please refer to the preceding paragraph.

3.3.6.4 Event Evaluation

Each of the two flood initiated accident sequences was quantified by using the existing PRA model or by modifying the model as necessary. This task was performed in a preliminary form as described above. Initiating frequencies were generated using very conservative assumptions which created event sequences that contributed less than 1E-7. The conclusion reached is that is not necessary to pursue the flood events in greater detail.

3.3.6.5 Results and Conclusions

Via the screening process described, we got down to the two events we thought would be the highest contributors. Based on the preliminary evaluation of those two events in the PRA model, a very conservative initial estimate indicated that each of these scenarios would contribute less than 1E-7 per reactor-year. It was, therefore, concluded that further detailed evaluation of internal flooding was not required.

The overall conclusion of this evaluation is that flood initiators do not contribute significantly to the risk of core damage at DAEC.

3.4.1 Application of Generic Letter Screening Criteria

Generic Letter 88-20 Appendix 2 presents the screening criteria to be applied to the results of the IPE for establishing reportability to the NRC. These criteria are repeated as follows:

1. Any functional sequence that contributes 1E-6 or more per reactor-year to core damage,

2. Any functional sequence that contributes 5% or more to the total core damage frequency,

3. Any functional sequence that has a core damage frequency greater than or equal to 1E-6 per reactor-year and that leads to containment failure which can result in a radioactive release magnitude greater than or equal to BWR-3 or PWR-4 release categories of WASH-1400,

4. Functional sequences that contribute to a containment bypass frequency in excess of 1E-7 per reactor-year, or

5. Any functional sequences that the utility determines from previous applicable PRAs or by utility engineering judgement to be important contributors to core damage frequency or poor containment performance.

Figure 3.4-1 gives a list of the most significant sequences with regard to their contribution to core damage. By applying the above criteria five sequences are identified as reportable. The are as follows:

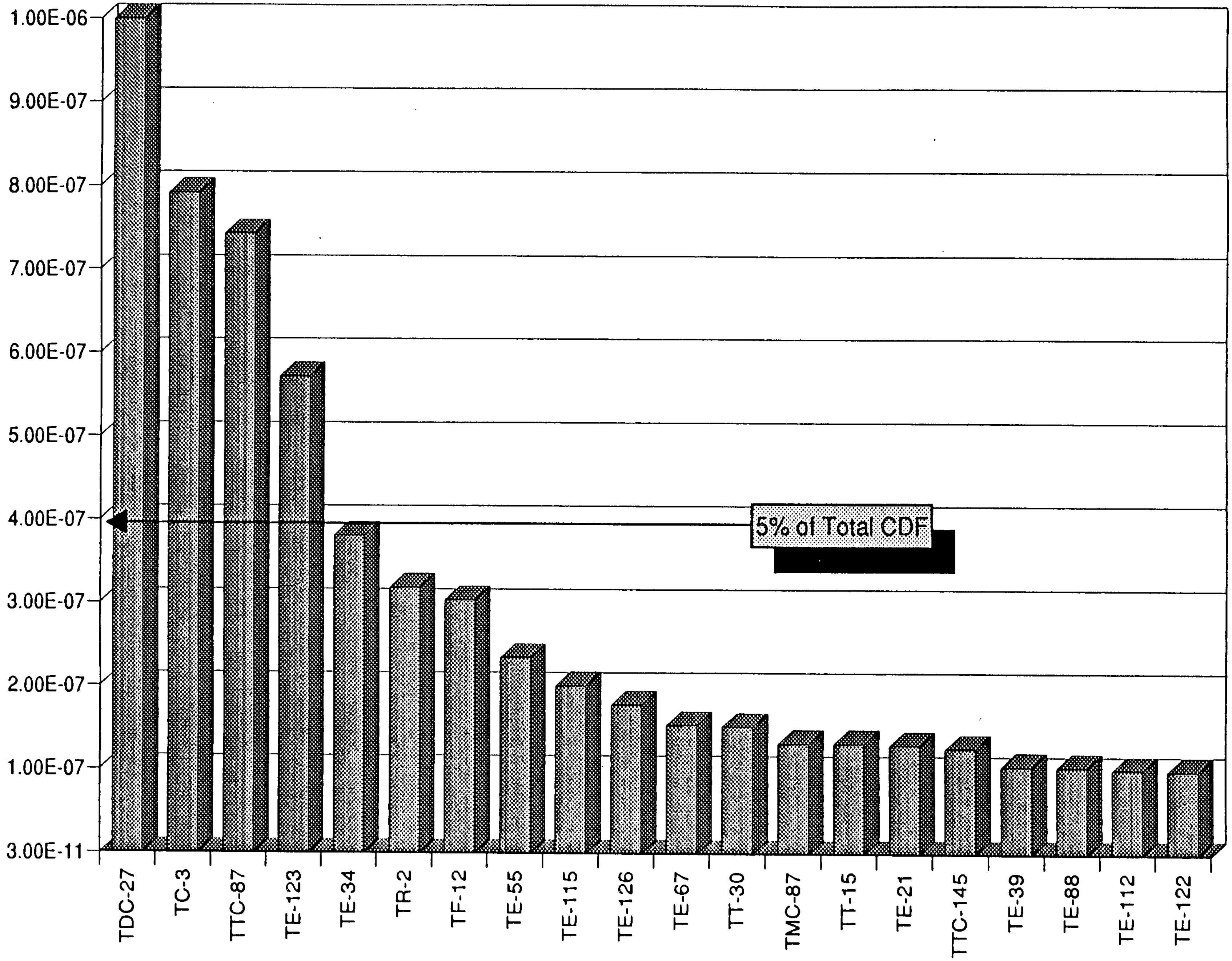


Figure 3.4-1
Most Significant Sequences

- At 1E-6, sequence TDC-27 (Loss of all 125 V DC power) is identified by both criteria 1 and 2. This sequence contributes 13% to the CDF (7.8E-6).

- At 7.9E-7, sequence TC-3 (Loss of decay heat removal) contributes 10% to CDF, and so it is identified by criterion 2.

- At 7.4E-7, sequence TTC-87 (ATWS with failure of SLC) is identified by criterion 2, since it contributes 9% to the CDF.

- At 5.7E-7, sequence TE-133 (Station blackout greater than 15 hours) is also identified by criterion 2, since it contributes 7% to the CDF.

- At 3.8E-7, sequence TE-34 (Loss of offsite power with early HPCI/RCIC failure) contributes 5% to CDF and is the last sequence to be identified by criterion 2.

The quantification was run at 3E-11 and all sequences were reviewed against the criteria listed. Only these five were identified as reportable. A more detailed description of each sequence is given in Section 6.

A comment during the formal review process questioned (see review comments in Section 5.3) whether the requirements of item 6. under Section 2.1.6 of NUREG-1335 had been carried out. Item 6. states "Identification of sequences that, but for low human error rates in recovery actions, would have been above the applicable core damage frequency screening criteria, any sequence that drops below the core damage frequency criteria because the frequency has been reduced by more than an order of magnitude by credit taken for human recovery actions should be discussed." By the definition used in this IPE analysis, (see Section 3.3.3.1 Type E - recovery actions) the only "recovery actions" used were power restoration in the loss of offsite power scenarios. In performing the restoration / recovery of power, methodology based on NUREG-1032 is used. In this methodology explicit human failures are not modeled. Instead, the technique develops weighted cumulative probabilities to recover power. These are basically the result of historic data and generic information that are adapted to consider site specific characteristics. As a consequence the DAEC IPE did not treat these recovery probabilities as human failures even though there is a component of human actions that will be required to accomplish them (the methodology does not explicitly identify what they are, nor does it separate them from equipment failures). Therefore, there are no recovery actions based on human actions as defined in this report. However, if a broader definition of "recovery actions" (i.e. including any human reliability action as a recovery action, as implied by the NRC response to questions at the Ft. Worth G.L. 88-20 meeting) is used, the DAEC IPE has several sequences to consider.

Using the broader definition of "recovery actions" an analysis was performed to evaluate the effect of "low human error rates" on the DAEC IPE. This analysis was done by setting all human error rates that were below 0.1 to 0.1 and evaluating their effect on the study. By doing this, an additional 28 sequences beyond the original five already discussed are identified. These additional sequences are shown on Figure 3.4-2. By referring to Section 3.1.2 the sequences identified on Figure 3.4-2 can be studied in more detail as to the success and failures of various systems in each individual sequence. As indicated by Figure 3.4-3, the contribution increase due to raising human failure rates to 0.1 is the result of three human reliability basic events.

OOPAF-MANUAL-DEP is the manual depressurization of the primary system. By the way the BWROG EPGs are implemented at DAEC, when a signal is received in the control room that starts the automatic ADS initiation sequence, (there is a 120 sec. delay before initiation) the operators immediately defeat the automatic initiation signal by locking out ADS. If it becomes necessary to go ahead and depressurize, the operators will have to do so manually. OOPAF-MANUAL-DEP is the failure to perform this manual

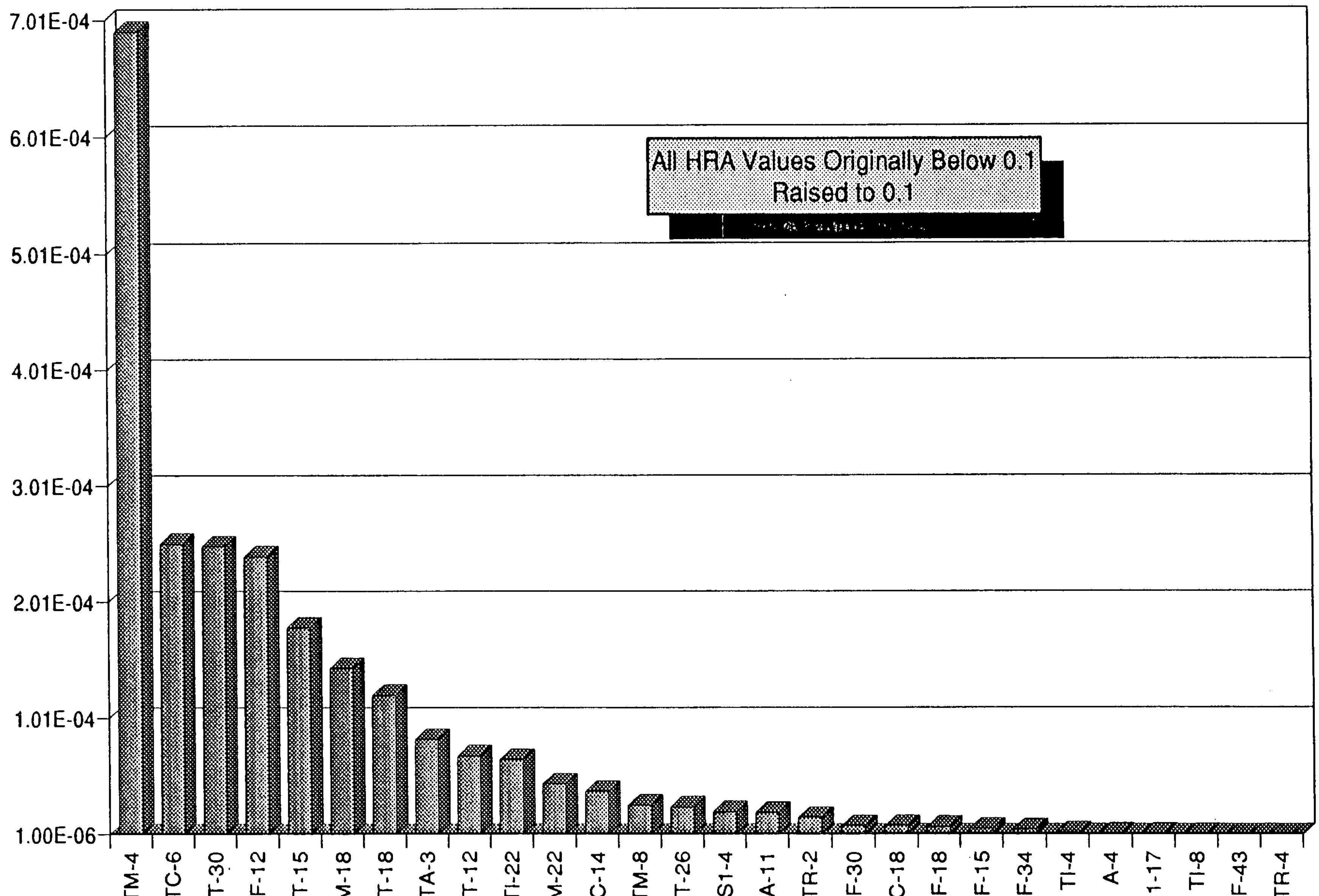


Figure 3.4-2
Sequences Identified Due to Human Reliability Sensitivities

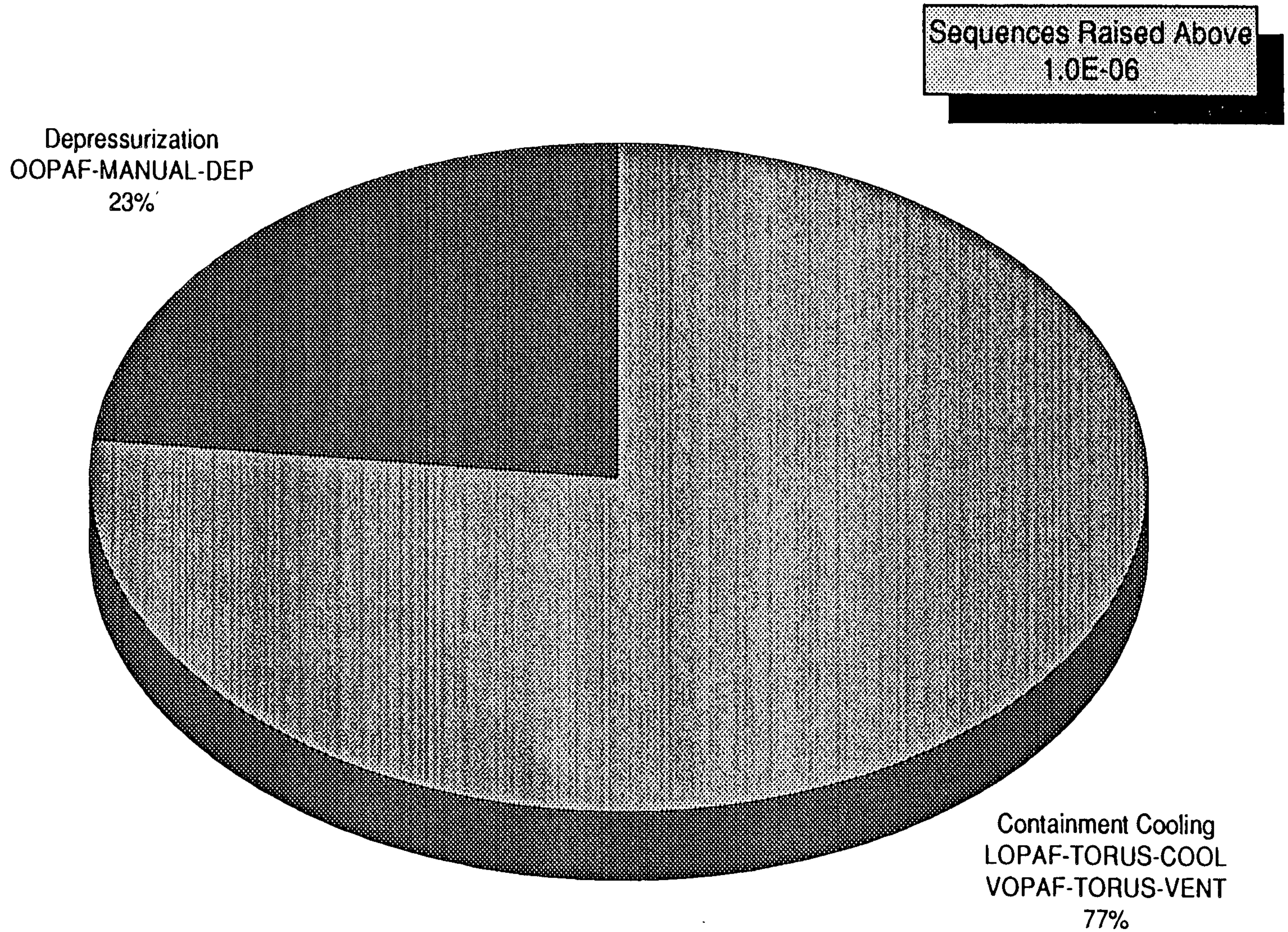


Figure 3.4-3
Contribution of Specific Operator Actions to HRA Sensitive Sequences

depressurization. As indicated by Figure 3.4-2, 23% of the added contribution is due to this human reliability basic event. This action is performed from the control room and is proceduralized and practiced. It also has, on the order of, one hour for the operators to take action. Therefore, a low failure rate should be expected and is justifiable. A value of 2.1E-4 is used in the quantification. Sequences TF-12, TM-18, TT-12, TT-26, TA-11, S1-17, and TF-43 are those that are affected by this human reliability basic event.

The remaining 77% of the added contribution is due to LOPAF-TORUS-COOL and VOPAF-TORUS-VENT. All the remaining sequences are affected by both these human reliability events with the exception of TR-2, and TR-4 which are only affected by VOPAF-TORUS-VENT. LOPAF-TORUS-COOL is the failure to manually establish torus cooling. VOPAF-TORUS-VENT is the failure to manually vent the torus. Both these events are proceduralized, practiced, and performed from the control room manual actions that have on the order of several hours to accomplished. Therefore, low failure rates should be expected and are justifiable. The value used in the quantification is 2E-3 for VENTING , and is 1E-4 for COOLING.

As an additional point of interest, TC-3, which was already identified by the previous screening criteria, went from 7.9E-7 to 1.24E-3 as a result of raising the LOPAF-TORUS-COOLING, and VOPAF-TORUS-VENT failures to 0.1.

In summation, it is clear that the three human reliability basic events discussed here are important. However, the lower failure rates used in the quantification are justifiable.

3.4.2 Vulnerability Screening

No vulnerabilities were identified as part of the IPE process for DAEC. 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 DAEC core damage frequency would not be able to meet the NRC's safety goal for core damage?

3. Are there any single failures of components that lead directly to a core damage state. This does not include the common cause failure of multiple components of similar types.

None of these criteria lead to the identification of potential vulnerabilities for the DAEC. 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. Also, while it does not include the contribution from external events, the overall core damage frequency of 7.84×10^{-6} per year is only a fraction of the NRC's safety goal of 1.0×10^{-4} per year. This leaves ample margin for accommodating external events.

Even though no "vulnerabilities" were uncovered by this evaluation, many insights were gained. In general, a significant insight was a system, component, or action which influenced the results of the study to a greater level than other events evaluated. A significant insight may involve:

1. A unique safety feature which drove risk either by limiting the core damage frequency or contributing a significant fraction to core damage frequency.

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.

Section 6.0 Plant Improvements and Unique Safety features discusses these areas in more detail.

3.4.3 Decay Heat Removal Evaluation

3.4.3.1 Introduction

This section provides a discussion of the adequacy of the DAEC shutdown decay heat removal (DHR) systems. This section summarizes the DAEC effort to satisfy the Generic Letter 88-20 requirement for a plant specific evaluation of USI A-45.

3.4.3.2 Historical Perspective

The generic issue of decay heat removal capability was approved as an unresolved safety issue (USI A-45) by the NRC in 1980. Prior to becoming a USI, Task A-45 as it was referred to, focused on the adequacy of steam generator auxiliary feedwater systems and alternative means of decay heat removal at PWRs. When the issue was approved as a USI, it was broadened to also investigate the need and possible design requirement for improving reliability of decay heat removal systems for BWRs.

In NUREG-1289, the staff defines the systems related to the decay heat removal function as those components and systems required to maintain primary and secondary coolant inventory control and to transfer heat from the reactor coolant system to an ultimate heat sink following shutdown of the reactor for transients, such as loss of feedwater, loss of offsite power, and small- break LOCAs. The A-45 program was not concerned with ATWS, ISLOCA, or those emergency core cooling systems that are required only during the reflood phase to maintain coolant inventory following a large or medium LOCA. The USI A-45 program did consider supporting systems that are required for various modes of decay heat removal. The reliability of the reactor protection system was not addressed, and therefore, successful shutdown of the reactor is assumed.

3.4.3.3 Definitions

The staff definition of DHR in A-45 is an expanded version of the functional definition of decay heat removal in a BWR. This expansion results in the inclusion of inventory makeup systems in addition to the function of decay heat removal. Therefore, in discussions of contributors to the core damage frequency in the DAEC IPE, it is important to ensure which definition of DHR is being used -- that of the NRC in A-45 or that traditionally assigned to the DHR function in PRAs.

3.4.3.4 NRC Staff Criteria (NUREG-1289)

The purpose of USI A-45 regarding shutdown decay heat removal requirements is to evaluate the adequacy of current designs to ensure that LWRs do not pose unacceptable risk as a result of DHR system failures. The primary objectives of the USI A-45 program are to evaluate the safety adequacy of DHR systems in existing LWR power plants and to assess the value and impact of alternative measures for improving the overall reliability of the DHR function.

At the time the USI A-45 program commenced, the NRC also started to develop a set of qualitative safety goals and quantitative design objectives. To aid progress in the USI A-45 program, some interim objectives were defined with the knowledge that these might have to be changed later in the program to conform with those finally decided on by the Commission. The principal quantitative design objective selected for USI A-45 is the frequency of core damage due to failure of the DHR function. An interim value of 1.0×10^{-5} per reactor year was recommended for examining the DHR related risk at individual plants.

The NRC staff in NUREG-1289 (pp.1-14) states that: "(a) limited scope[1] plant specific PRA could demonstrate the adequacy of the existing decay heat removal function by documenting that its contribution to core damage frequency was relatively low, on the order of 1E-5 per reactor year or less."

The decay heat removal criteria is structured such that at least the initial screening is based solely on the core damage frequency as the figure of merit. This is based on the NRC's conclusion that if this screening value is met, then little if any, cost beneficial modifications would be warranted.

The experience gained from application of PRAs to US LWRs in the USI A-45 and other programs suggests that, when the systematic examinations for severe accident vulnerabilities have been completed, the existing plants will fall into three broad categories. The following quantitative values have been used by the staff as a basis for categorization of these events:

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[1]Limited scope PRA as defined here includes the following initiating events:

1. Small LOCAs
2. Loss of Offsite Power Transients
3. Transients caused by loss of the power conversion systems
4. Transients with offsite power and power conversion systems initially available
5. Transients caused by the loss of an AC or DC bus

The following initiating events are not included in a limited-scope PRA as defined here:

1. Large and Medium LOCAs
2. Reactor vessel ruptures
3. Interfacing systems LOCAs
4. Anticipated transients without SCRAM
5. Steam generator tube rupture
6. External Events

Category	Classification of Level of DHR Vulnerability	Criterion
1	Frequency of core damage due to failures of the DHR function acceptably small, or reducible to an acceptable level by simple improvements.	less than $3.0x10^{-5}$
2	DHR performance characteristics intermediate between Categories 1 and 3.	between $3.0x10^{-5}$ and $3.0x10^{-4}$
3	Frequency of core damage so large that prompt action to reduce the probability of core damage to an acceptable level is necessary.	greater than $3.0x10^{-4}$

3.4.3.5 Containment Decay Heat Removal Configuration at DAEC

This subsection describes the containment decay heat removal configuration at DAEC. There are also numerous RPV coolant injection systems that provide coolant makeup to the RPV and are of interest when investigating the broader question of A-45; these are described in Section 2.

Decay heat removal is accomplished during normal plant operation using the main condenser and the Condensate/Feedwater system. Feedwater is a normally operating, unshared system, which gives it a high likelihood of being available after a transient.

The next method for decay heat removal following a plant trip or shutdown is the shutdown cooling mode of RHR. When the reactor is depressurized, the RHR system is used in the SDC mode. Reactor water is drawn from the vessel and sent through heat exchangers prior to being returned to the vessel. The heat exchangers are cooled by the Residual Heat Removal Service Water (RHRSW) system.

During abnormal conditions, the EOPs specify the use of a number of methods for providing decay heat removal, either directly or indirectly:

- Maintain the main condenser by reopening the MSIVs
- Maintain the drywell coolers
- Establish torus cooling
- Initiate containment sprays
- Maximize cooling using the Reactor Water Cleanup System
- Use HPCI and RCIC with or without injection to the vessel
- Open the containment vent

Containment Venting Configuration

DAEC has adopted the NRC request, as part of Supplement 1 to Generic Letter 88-20 and Generic Letter 89-16, to include a hard piped vent from the torus. This plant modification further enhances the containment heat removal capability of the DAEC design. The DAEC hard piped vent directs primary containment effluent directly out of the reactor building, rather than through the Standby Gas Treatment System. This bypasses the possibility of rupturing the SGTS ductwork inside the reactor building, and exposing key safety components to harsh environmental conditions. The DAEC hard piped vent is a DC operated system.

RHR/RHRSW Configuration

The RHR system is comprised of two loops with two pumps and two heat exchangers in parallel. The RHR system operates in one of four modes:

1. Low Pressure Cooling Injection (LPCI)
2. Shutdown Cooling (SDC)
3. Suppression Pool Cooling
4. Containment Sprays

The RHRSW system is equipped with four pumps; each pump can accommodate the heat removal requirements of one RHR heat exchanger. (Section 3.2.1 contains a more detailed discussion of these systems.)

3.4.3.6 Incorporation of Plant Features in the DAEC IPE Model

The DAEC IPE explicitly models the functions of containment heat removal and reactor inventory control. As part of the DAEC analysis, the dedicated hard piped containment vent has been included in the baseline models. This is consistent with the NRC request in Generic Letter 89-16. Therefore, for containment heat removal, the use of the main condenser, torus cooling, shutdown cooling, wetwell sprays, and the wetwell vent are the primary methods considered. In the case of the main condenser, the EOPs specify bypassing low level interlocks to re-open the MSIVs if they have previously closed. The drywell fans, Reactor Water Cleanup, the head vent, and the use of HPCI and/or RCIC in CST-to-CST mode are inadequate by themselves in most cases to remove containment heat in order to prevent containment overpressure.

The torus cooling mode of RHR is modeled under the W(TORUS) node of the Level I PRA event trees. Use of the main condenser is modeled under the Z(MC:RECOV) node. If these methods fail or are inadequate, the W(VENT) node of the event trees questions the use of the wetwell vent.

3.4.3.7 "TW" (Class II) Accident Sequences

A more traditional definition of decay heat removal for BWRs would examine those accident sequences that result in core damage due to the loss of only the decay heat removal function. These sequences tend to be long duration sequences in which the inability to remove heat from containment results in either containment overpressure failure and subsequent core damage, or core damage induced by the incipient containment failure.

These sequences were referred to in WASH-1400 as "TW" sequences, and are categorized in the DAEC evaluation as Class II sequences.

The results of the Class II analysis were compared to the results for Peach Bottom, Unit 2, presented in NUREG/CR-4550, the NUREG-1150 BWR/4 Mark I surrogate plant, the Monticello IPE, and the Fitzpatrick IPE. This is presented in Table 3.4-1.

Table 3.4-1

"TW" Sequence Core Damage Frequencies

Study	Core Damage Frequency for Loss of DHR[2] Sequences (per reactor year)
WASH-1400	1.0×10^{-5}
NUREG-1150 (BWR/4)	$< 1.0\times10^{-8}$
Monticello IPE	7.1×10^{-8}
Fitzpatrick IPE	2.0×10^{-7}
DAEC IPE	1.8×10^{-6}

3.4.3.8 Other Studies

In addition to simply considering this relatively low contribution to core damage frequency, DHR related insights may be uncovered by comparison to the vulnerabilities raised in the USI A-45 BWR case studies. Table 3.4-2 summarizes the specific "vulnerabilities", which were stated to exist at the two case study plants. The applicability to DAEC is also presented. The following are the Generic Insights published in the study.

<u>Generic Insights</u>

- At the support system level, there is often less redundancy, less separation and

[2]Traditional TW Sequences

independence between trains, poorer overall general arrangement of equipment from a safety viewpoint, and much more system sharing compared to the higher level systems.

- Human errors were found to be of special significance. The six studies modeled errors of omission (e.g., delays or failures in performing specified actions), and it was found that in many cases the resulting risk was very sensitive to the assumptions used and the way that the errors were modeled. Consequently, great care is warranted in the development of human error models. In addition, it is likely that errors of commission are also important. Although such "cognitive" errors are much more difficult to model, efforts to take them into account will result in a more complete picture of DHR-related risk.

- Of equal importance to human errors is the credit that is allowed for recovery actions, which can have a very significant effect upon the resulting risk. Some of the more important recovery actions are restoring offsite power, fixing local faults of batteries and diesel generators, actuating safety systems manually , and opening locally failed motor-operated valves. Considering the importance of such human recovery actions, considerable effort is justified in the development of the methods and assumptions used in these areas.

- Transient events that are initiated or influenced by a loss of offsite power were found to contribute significantly to risk. A new rule, 10CFR50.63 has been issued June 21, 1988 as a resolution to USI A-44, "Station Blackout". Implementation of this rule will reduce the risk from such events.

All of the generic insights presented in Generic Letter 88-20 Appendix 5 have been considered in the DAEC IPE assessment to ensure an accurate treatment of the DHR issue.

3.4.3.9 A-45 Conclusions for DAEC

The assessment of the adequacy of decay heat removal systems was performed as part of the DAEC IPE.

The IPE used both quantitative design objectives from the NRC staff and qualitative insights from past A-45 studies as input for the analysis. The IPE evaluation supports the conclusion that no vulnerabilities exist at DAEC to adversely affect the operators' ability to accomplish the DHR function during an accident. Specifically, the results of this evaluation indicate that 23% (1.8×10^{-6} per Rx-year) of the total core damage frequency is due to sequences involving loss of containment heat removal, and that the contribution to core damage frequency from the loss of inventory accidents is approximately 52% (4.1×10^{-6} per Rx-year). Therefore, the frequency of core damage associated with DHR failures is 5.9×10^{-6} per reactor year using the NRC definition of DHR related sequences in A-45. This is below the acceptance level set by the NRC staff in NUREG-1289 of 3.0×10^{-5} per Rx-year, and much lower than the 3.0×10^{-4} level set for prompt corrective action. This comparatively low core damage frequency results in the determination that no plant modifications are judged to be cost beneficial.

Table 3.4-2

Vulnerabilities Raised in the USI A-45 BWR Case Studies

A-45 Case Study Insight	Applicability to DAEC
Following a Loss of Offsite Power, the diesel's configuration may not successfully shutdown both units given an extended loss of offsite power.	Not applicable to DAEC. DAEC has only one unit, with a dedicated diesel generator for each division of safeguards equipment.
Loss of Offsite power combined with loss of the 125V DC batteries will prevent the diesels from starting due to loss of field flashing.	This sequence is explicitly modeled in the AC Power models for the DAEC.
If each EDG has a dedicated jacket cooling system with only one cooling water pump, then there are several single failures that can fail the cooling system and , ultimately, the diesel generator.	The individual emergency diesel generators are not designed to be single failure proof. Loss of cooling water can occur and result in eventual failure of a diesel in the event ESW is lost. Design provisions include the ability to crosstie, with a spool piece, the ESW loops.
Following a Loss of Offsite Power, failure of one diesel generator, and the failure of the opposite division of 125V DC will leave the plant without low pressure ECCS systems.	True, if a 125 VDC division is unavailable, the associated diesel will not load due to a lack of field flashing.
Following a Loss of Offsite Power, diesel generator faults are dominant contributors to core damage frequency.	True, if the LOSP is not recovered within a reasonable time period.
Failure of certain motor-operated valves in the RBCCW system could isolate cooling water to the ECCS room coolers or divert cooling water to non-critical loads.	Not applicable to DAEC. ESW is provided to critical components. These are separate from the EDG cooling water headers. In addition, each load path is in parallel, so a single failure will not disable all cooling.

3.4.4 USI and GSI Screening

With the exception of USI A-45 no USIs or GSIs are proposed to be resolved by the DAEC IPE submittal. With regard to USI A-45, Section 3.4.3 presents the results of the DAEC evaluation of decay heat removal. The conclusion reached in the evaluation is that sequences regarding DHR are well below the screening value and, as a result, no plant modifications are judged to be cost beneficial.

4. BACK END ANALYSIS

4.0 INTRODUCTION AND OVERVIEW

The purpose of this section is to describe the approach for the performance of the containment analysis for the DAEC Individual Plant Examination. This section outlines a complete Level 2 PRA methodology which satisfies the request made by the NRC in the IPE Generic Letter 88-20 and its companion guidance document, NUREG-1335. This section also develops a framework within which future questions regarding DAEC containment performance can be addressed.

The DAEC IPE evaluation includes consideration of severe accident behavior recognizing the DAEC containment capability and incorporating the role of DAEC mitigating systems in responding to an accident. This information can furnish input for any future development of accident management procedures and/or the revision of current emergency operating procedures.

4.0.1 Application of the Level II IPE Requirements by Iowa Electric

This section documents the Iowa Electric effort to establish a thorough, scrutable, and technically sound methodology for examining the DAEC containment capability under severe accident conditions in response to Generic Letter 88-20. It has involved the development of a detailed set of containment event trees as a framework for examining severe accident phenomena including both active and passive mitigation functions of the DAEC Mark I containment. This effort is based upon previous methods used in the Shoreham PRA, the Limerick IPE, the Peach Bottom Containment Evaluation, and the Vermont Yankee Containment Safety Study.

Figure 4.0-1 provides a simple flow chart that describes the relationship between the Level 1 PRA and the Level 2 containment evaluation. The interface between the two evaluations requires the transfer of information describing the key aspects of the postulated severe accident scenarios.

The principal technical advances that have been incorporated into the DAEC containment evaluation effort include the following:

- A containment event tree that includes sufficient detail to quantify the effects of plant modifications and changes in procedures.

- Established success criteria for recovery of degraded core conditions within the reactor vessel (e.g., TMI-2 events), including those that involve recovery actions during in-vessel core melt progression accidents.

- Incorporation of the emergency procedures at DAEC. This includes containment flooding which is a major model perturbation from previous studies.

- Consideration of the BWROG/NUMARC containment safety study to incorporate the latest input on severe accident phenomenological issues as they affect containment response (e.g., direct containment heating, heat management, seal performance).

- Plant specific deterministic calculations using MAAP to support the improved success criteria.

DEVELOPMENT OF AN INTEGRATED RISK MODEL

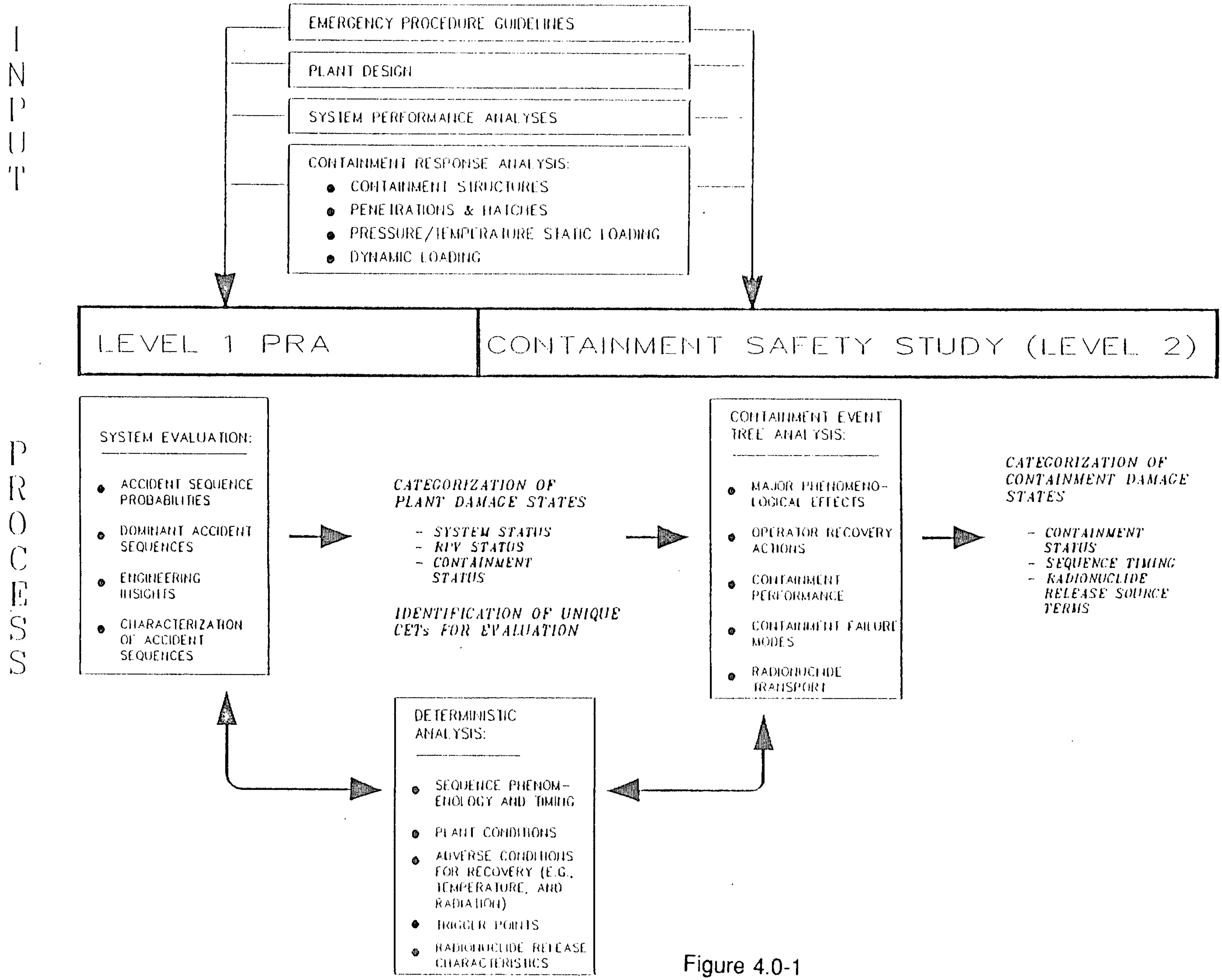


Figure 4.0-1

- A traceable documentation path through the containment event tree so that both qualitative and quantitative insights can be developed. This facilitates both communication with the NRC and internal use within Iowa Electric.

- Responses to issues raised by the NRC contractors in NUREG-1150 in a more visible manner.

The desire to use a thorough approach that emphasizes functional response and uses fault trees to develop the detail of each node was reinforced by NUREG-1150 (Draft) peer review comment that stated:

> *By using the large event tree approach (of NUREG-1150) the containment event tree becomes less of a tool for understanding containment functions. The fact that there are front line containment functions supported by various individual systems and features becomes somewhat lost. The use of a smaller functional event tree with supporting fault tree logic would probably provide a more manageable product capable of providing greater insights.*

This comment was further expanded in the Special Committee report:

> *It seemed to us that this level of detail exceeded understanding of the phenomena involved, and implied greater insight into the processes assumed to be taking place than was justified. When confronted by the need to quantify poorly understood phenomena, it is certainly necessary to dissect the problem carefully to ensure that important aspects are not overlooked. But this practice should be restricted to assisting the thought process, and the final quantification should be at a scale commensurate with the overall understanding.*

In addition to the need to have detail commensurate with the available data and the ability to communicate that data, another important insight has been gleaned from the NUREG-1150 peer Review:

> *The containment event tree should make allowance for accident management actions, for example, attempting to recover cooling and to protect vessel and containment integrity. Such actions are likely to substantially change the course of events and could significantly affect risk.*

4.0.2 Objectives of the Level 2 Analysis

The primary objective of this analysis is to perform a comprehensive containment evaluation of the DAEC plant. The approach addresses the requirements of Generic Letter 88-20 and NUREG-1335, utilizing plant-specific analyses and referenceable calculations (i.e., previously performed analyses and available data), which have been determined to be appropriate for the application to the DAEC study.

Another objective of this plant specific evaluation is to provide a framework within which the following questions can be considered:

- Do any unusual containment vulnerabilities exist in the DAEC containment that are required to be modified by procedural or hardware changes?

- Can severe accident behavior information be presented to engineering, operations, and maintenance in a way that will assist these organizations in preventing or mitigating severe accidents through forward thinking approaches?

- Are there severe accident management techniques that should be developed?

Other objectives of this plant specific back-end evaluation are to allow the following:

- Provide a consistent interface with the accident scenarios postulated in the Level 1 analysis,

- Represent possible containment failure mechanisms,

- Represent uncertainties in severe accident phenomenology,

- Identify controlling plant features,

- Incorporate technical information from many sources,

- Provide a methodology that may be updated with new plant information and severe accident technology, and

- Highlight the time windows for recovery actions to be integrated into an accident management program.

As a result of meeting these objectives, the DAEC Level 2 containment safety analysis is considered to be a comprehensive Individual Plant Examination (IPE) for investigating and evaluating containment and severe accident evaluations as requested by the NRC in Generic Letter 88-20 and NUREG-1335.

4.0.3 Approach to the Level 2 Analysis

The process of performing the containment analysis begins with an evaluation of the DAEC Level 1 sequences. These sequences are categorized in terms of the type of challenge to containment posed by each sequence and the operability of systems that could mitigate these effects. Since risk is additive, it is possible to bin, or group, similar sequences based on these criteria, and consider each bin collectively as representing one challenge type to the containment. While each Level 1 accident sequence is explicitly treated in the computer model of the DAEC plant, the rules and split fractions used in the assessment take advantage of the recognition of similar accident challenges or classes from the Level 1 analysis.

Plant structural and physical information is required in order to evaluate the response of the containment systems to the core damage event. This information is used to perform the plant-specific analyses, as well as, to characterize or modify the results of studies from other similar plants for use within the DAEC study.

The determination of ultimate containment failure capability is required to assess the timing, size, and location of possible failure modes. A containment analysis of the DAEC Mark I containment by CB&I is included in the analysis to provide insights on the containment failure pressure, temperature, and location.

An integrated deterministic evaluation of containment response to accident challenges involves the comparison of the calculated thermal-hydraulic response of structure with the ultimate containment capability to identify possible containment failure points for a particular accident scenario. Therefore, an assessment of the physical response of plant and containment systems to each challenge is performed using a deterministic code.

The containment event tree (CET) is a device for representing these various accident scenarios in terms of system capability and human interaction to arrest core damage and prevent an undesirable outcome. The DAEC analysis will treat both the systems and required operator actions together in the CET.

The objective of obtaining a realistic, analytical result can be achieved by including necessary detail regarding system capability and human intervention. In this context, the containment event tree allows for the consideration of the operating staff implementing active mitigation strategies which might reduce the severity of release or delay the time of the release. Such actions would allow additional time for the implementation of other actions which might terminate the event. Actions which prevent major releases, reduce the consequences, or delay a radionuclide release are effective in reducing the overall plant risk. Consequently, these operating staff actions are treated explicitly in the containment event tree for each sequence type.

The containment event tree nodes are quantified by developing functional fault trees to describe the various factors which influence the nodal failure probability. These detailed, plant-specific nodal fault trees are then converted to a system equation format similar to the Level 1 PRA model for input to CAFTA for the quantification of the Level 2 model.

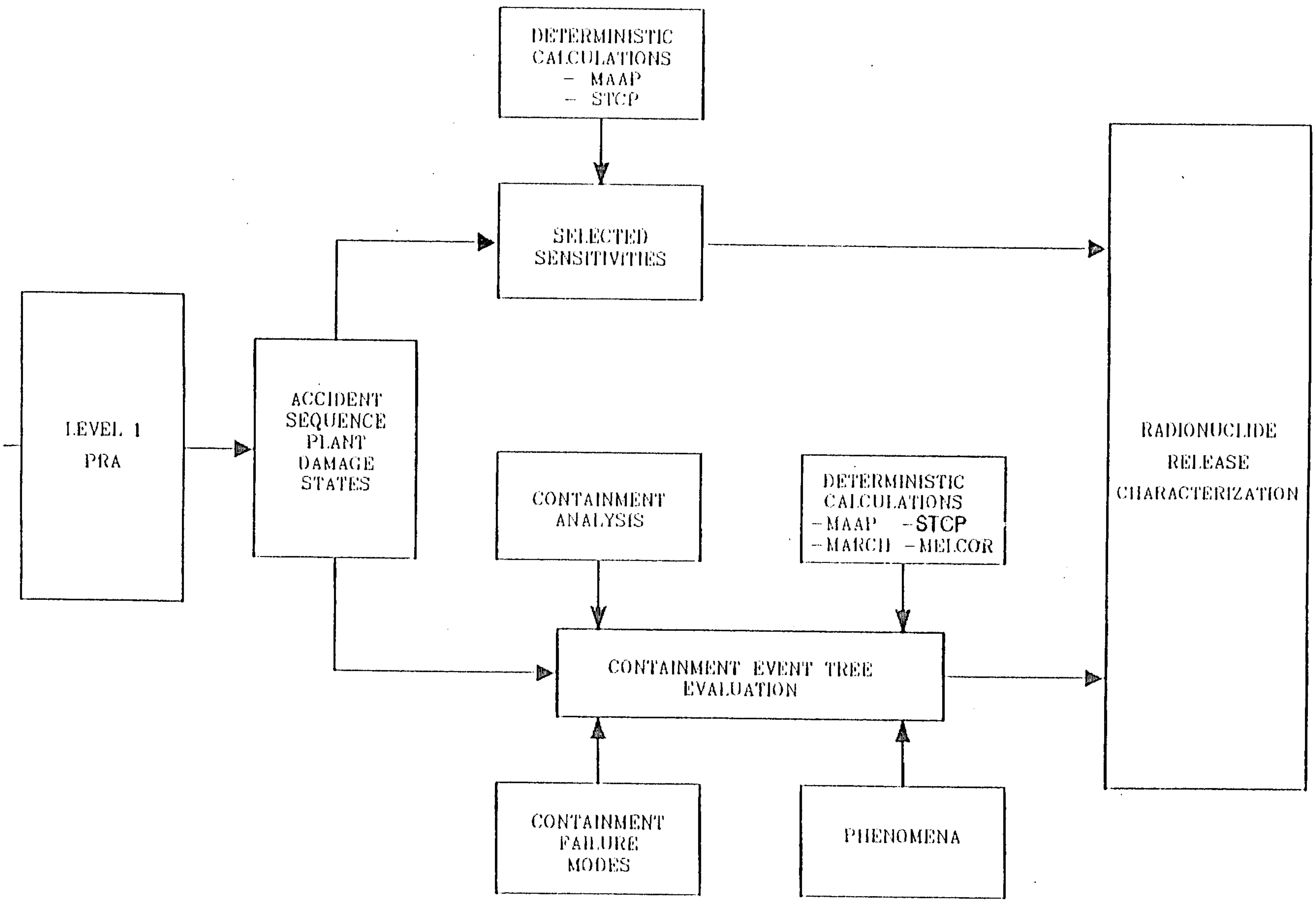
The DAEC IPE also includes an assessment of phenomenological matters considering NRC positions on these issues and related uncertainties including the issues as summarized in IDCOR Technical Report 86.1 (i.e., letters from T. Speis, NRC, to A. Buhl, ITC, "Position Papers for the NRC/IDCOR Technical Issues," dated September 22, 1986; November 26, 1986; and March 11, 1987).

The DAEC Level 1 system analysis is integrated with the containment analysis so that initiating events and system failures (resulting in core damage) that also impair containment systems are accounted for through the direct coupling of the Level 1 and

Level 2 event trees. This direct linking on a sequence-by-sequence basis of the front-end to back-end portions ensures that the support state conditions (e.g., dependencies) are properly accounted for throughout the front-end and back-end trees. These trees and their direct linking include preventive or mitigative features as well as timing considerations. Three different containment event tree structures, each linked to the appropriate front-end event tree sequence, are used to properly handle the various combinations of accident sequences that may occur (e.g., containment failure before core damage cases as well as vice-versa and containment bypass sequences).

Figure 4.0-2 provides a simplified overview of the DAEC Level 1 and Level 2 PRA model identifying the nomenclature of the various elements of the event tree models and their interfaces.

Figure 4.0-3 provides a simplified flow chart of the major technical tasks involved in the DAEC Level 2 evaluation and where each of these elements is discussed in this report.



FLOW OF INFORMATION FROM LEVEL 1 PRA TO LEVEL 2 PRA

Figure 4.0-2

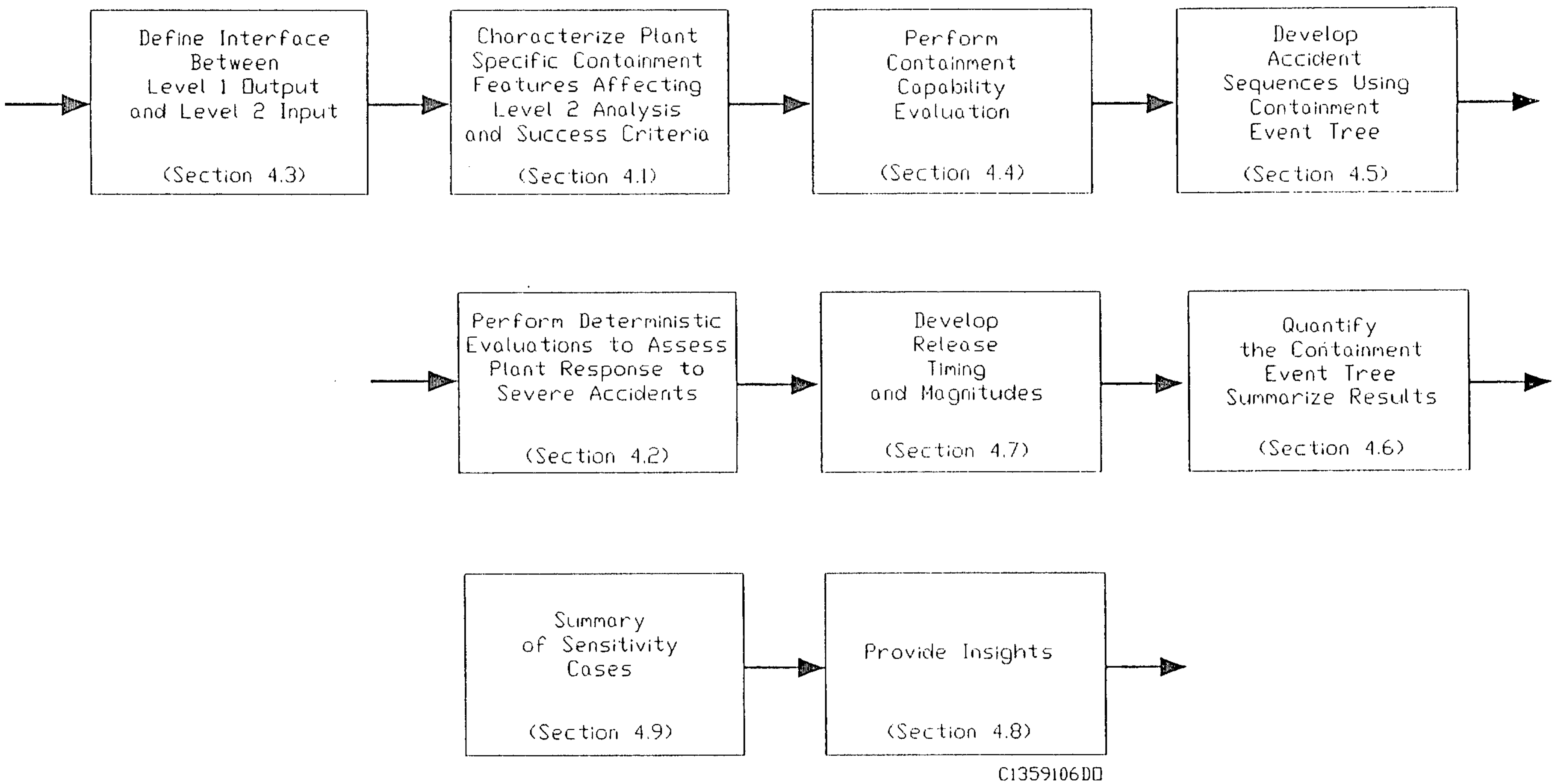


Figure 4.0-3 Major Technical Elements in Level 2 Evaluation

4.1 DAEC CONTAINMENT DESIGN DESCRIPTION AND DATA

This subsection provides the following data and design descriptions:

- Primary containment
- Secondary containment
- Functional capability of containment.

4.1.1 Summary of Primary Containment Features

The primary containment structure is a low leakage, pressure suppression system. It forms a fission product barrier which, in conjunction with the secondary containment system, will contain the radioactive fission products generated during all modes of plant operation and any postulated design basis accident so that off-site doses will not exceed the requirements of 10CFR100.

DAEC employs a Mark I pressure suppression containment system which houses the reactor vessel, the reactor recirculation loops, and other branch connections of the Reactor Coolant System, a pressure suppression chamber that stores a large volume of water, a vent system connecting the drywell and the pressure suppression chamber, isolation valves, containment cooling systems, and other service equipment.

The DAEC primary containment system consists of two major structural components: (1) the drywell and (2) the suppression chamber. The drywell surrounds the Reactor Pressure Vessel, and it is connected by 8 vent pipes, each 4 ft. 9 in. in diameter, to the torus-shaped suppression chamber. The suppression chamber, also called the torus or wetwell, contains a large volume of water affording an effective means of primary containment pressure control if steam is released from the reactor coolant pressure boundary into the drywell.

The general configuration of the primary and secondary containments and equipment locations are shown in Figures 4.1-1 through 4.1-4. The principal design parameters and characteristics are given in Table 4.1-1.

The main functions of the primary containment system as stated in the DAEC UFSAR are:

- To withstand the pressures and temperatures resulting from any of the postulated design-basis accidents,

- To provide an essentially leak tight barrier against uncontrolled release of radioactivity,

- To house and support reactor vessel and support equipment.

In addition to those functions specified above, the containment also provides the following functions:

- A heat sink using the suppression pool

- A potential scrubbing mechanism in the radionuclide release path using the suppression pool and the drywell sprays.

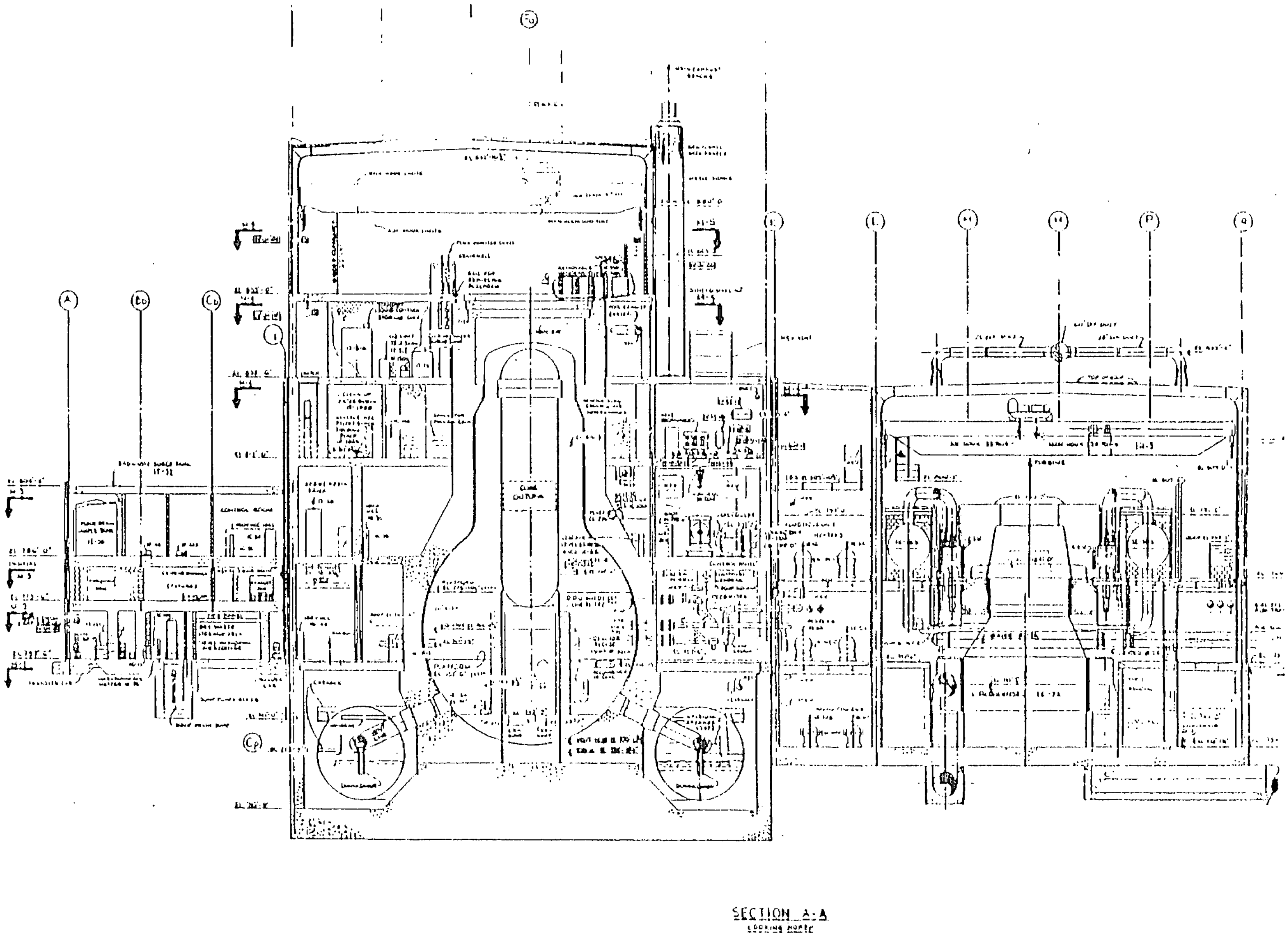
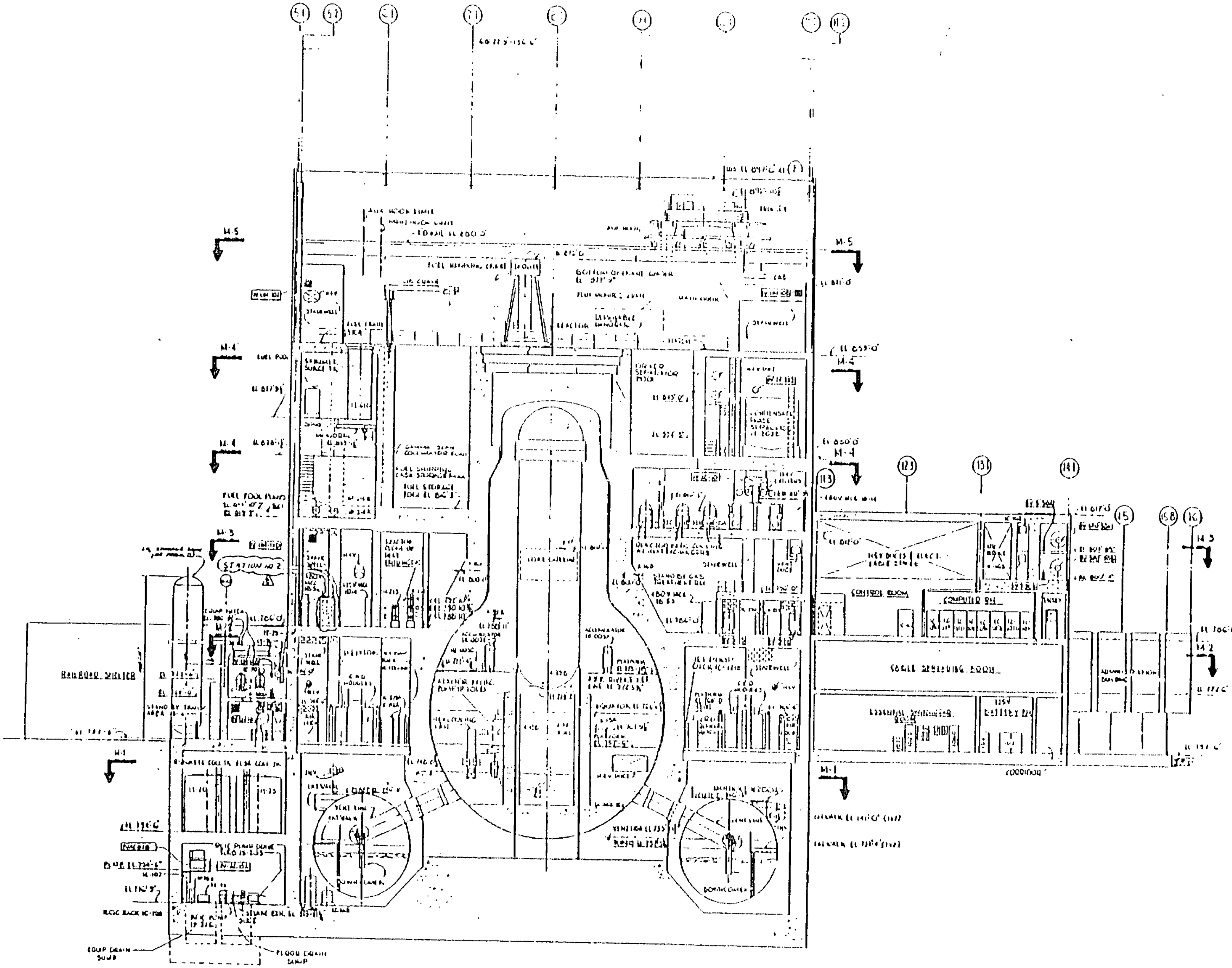
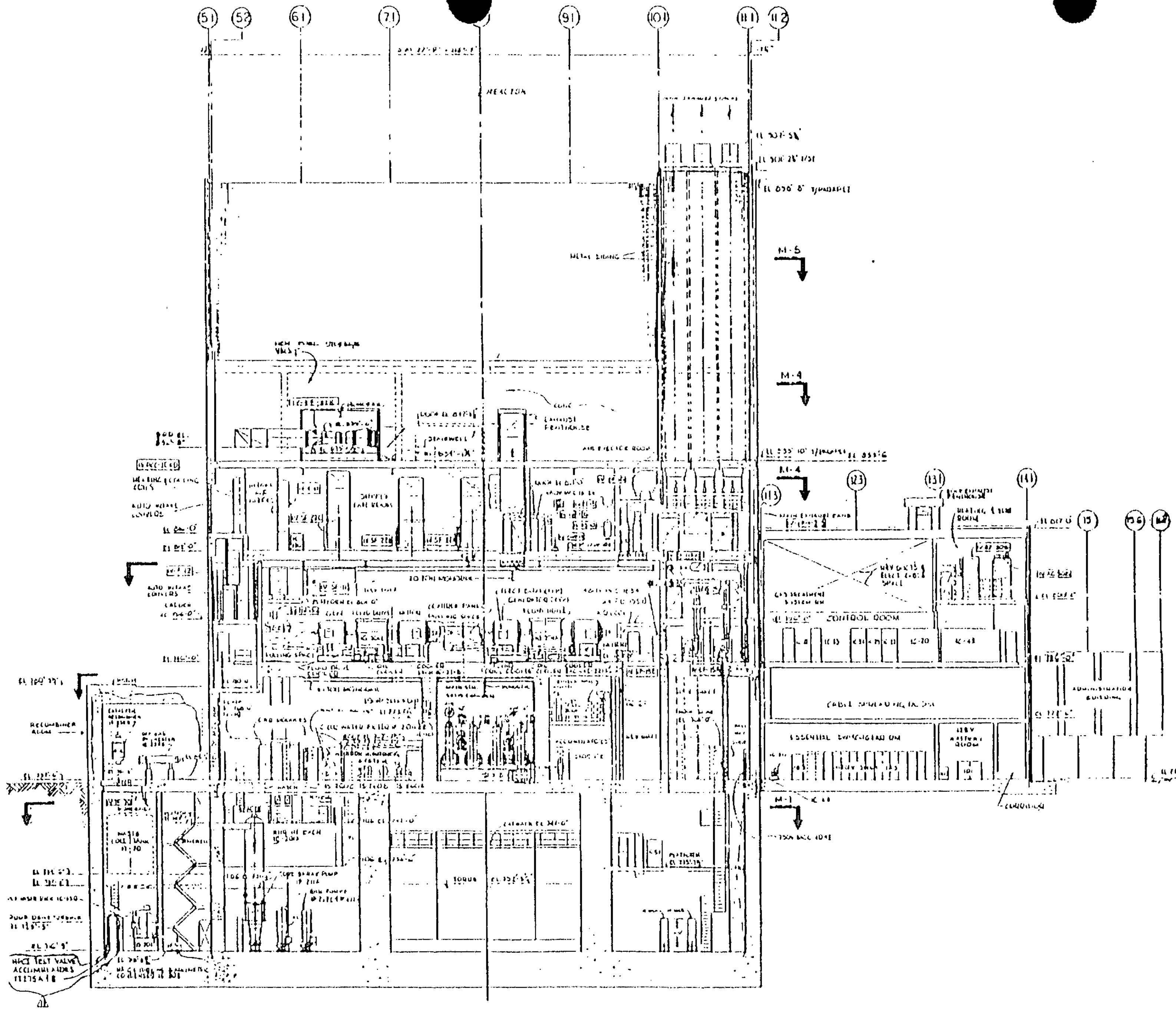


Figure 4.1-1 DAEC Containment Cutaway Diagram and Equipment Locations



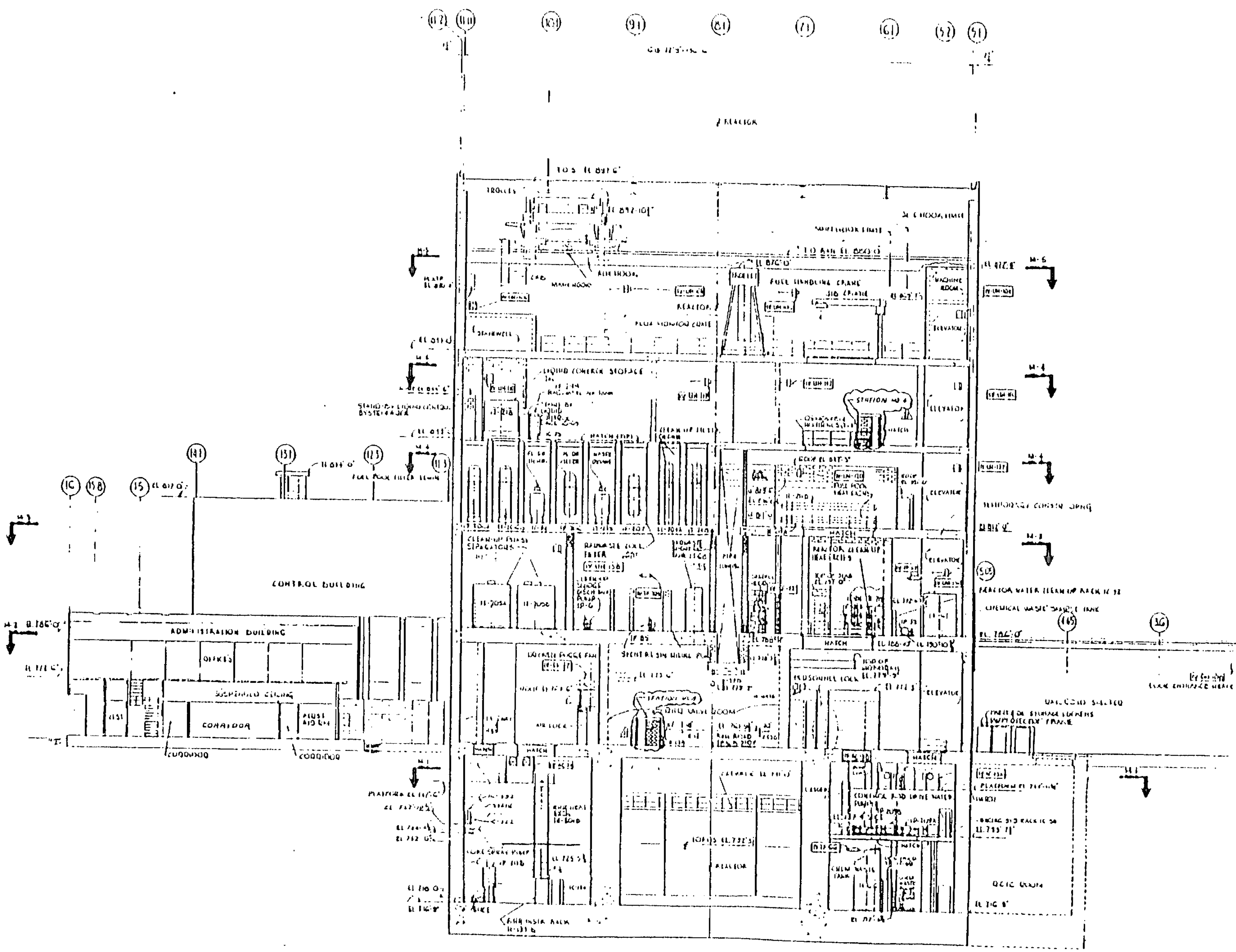
SECTION G-G

Figure 4.1-2 Containment Cutaway Diagram and Equipment Locations



SECTION E-E

Figure 4.1-3 Containment Cutaway Diagram and Equipment Locations



SECTION H-H

Figure 4.1-4 DAEC Containment Cutaway Diagram and Equipment Locations

Table 4.1-1

PRINCIPAL DESIGN PARAMETERS AND CHARACTERISTICS OF THE DAEC PRIMARY CONTAINMENT

Pressure Suppression Chamber Internal Design Pressure External Design Pressure	 56 psig 2 psid
Drywell Internal Design Pressure External Design Pressure	 56 psig 2 psid
Drywell Free Volume, including vent system - Minimum (approximate) - Gross (approximate)	 109,400 $ft.^3$ 144,000 $ft.^3$
Pressure Suppression Chamber Free Air Volume - Minimum (approximate) - Gross (approximate)	 94,270 $ft.^3$ 162,400 $ft.^3$
Pressure Suppression Pool Water Volume - Minimum (approximate) - Maximum (approximate)	 58,900 $ft.^3$ 61,500 $ft.^3$
Design Temperature of Drywell	281°F
Design Temperature of Pressure Suppression Chamber	281°F
For the DBA LOCA: · Effective Accident Break Area · Vent Loss Coefficient · Break Area/Total Vent Area · Calculated Maximum Pressure After Blowdown - Drywell - Pressure Suppression Chamber · Design Leak Rate	 2.515 $ft.^2$ 4.4 0.0177 42.7 psig 23.6 psig 2% Weight/Day

Component Description

The following containment system components will be discussed:

- Drywell
- Pressure suppression chamber
- Vent system
- Vacuum breakers
- Primary containment penetrations

Drywell

The drywell is a steel pressure vessel with a spherical lower portion 63 ft. in diameter and a cylindrical upper portion 32 ft. in diameter. The overall height is approximately 108 ft. 9 in. (see Figure 4.1-5). The drywell is designed for an internal pressure of 56 psig coincident with a temperature of 281°F with applicable dead, live, and seismic loads imposed on the shell. The maximum internal drywell design pressure is 62 psig. Design external pressure is 2 psig at 281°F.

The drywell is surrounded by a reinforced-concrete structure for shielding purposes. The concrete provides no drywell structural support for design conditions. In areas where it backs up the drywell shell, this reinforced concrete provides additional resistance to deformation and buckling of the shell when pressures and temperatures exceed design. Above the transition zone, and below the flange, the drywell is separated from the reinforced concrete by a gap of approximately 2 in. Shielding over the top of the drywell is provided by removable, segmented, reinforced-concrete shield plugs.

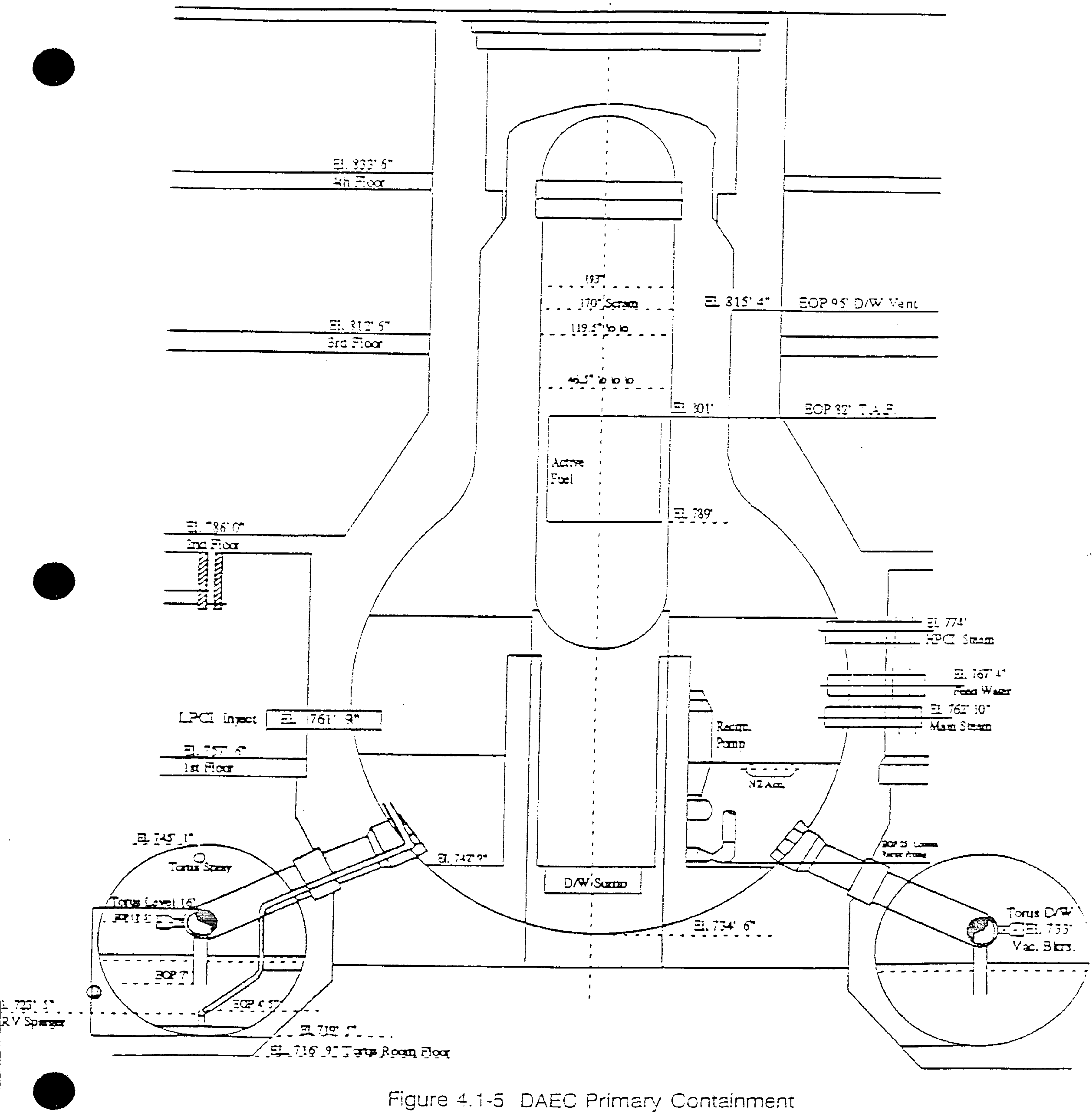


Figure 4.1-5 DAEC Primary Containment

The general configuration of the drywell floor can affect the course of many accident sequences (i.e., those in which the core melts through the RPV). Analysis of the materials, dimensions, and configurations encountered in the DAEC Mark I containment is important in determining the potential for ex-vessel debris to fill the drywell sumps, spread across the containment floor and contact the steel containment vessel.

Access to the drywell is through the equipment hatch, through the equipment/personnel air lock (both 12 ft. in diameter), and through the double-gasketed drywell head, with a 24 inch manhole, all of which have provisions for individual leak testing.

The bolted top closure of the drywell is 27 ft. 2 in. in diameter and is made with a double tongue and groove seal having a test connection between, which will permit periodic checks for tightness without pressurizing the entire vessel.

Jet deflectors are provided at the inlet of each vent pipe to prevent possible damage to the pipes or bellows assemblies from a jet force that might accompany a pipe break in the drywell and to prevent overloading any single vent.

After the initial leak rate and overpressure testing, the drywell was embedded in concrete to elevation 741 ft. 8.5 in. An embedment transition is provided for the shell from elevation 741 ft. 8.5 in. to elevation 742 ft. 9 in. The embedment transition is filled with sand and covered with an 18 gauge galvanized steel plate which is sealed to the drywell shell. Any leakage onto the cover plate is directed into the Torus Room basement via four 4-inch drain lines. The transition area is provided with four 2-inch sand-filled drain lines.

Pressure Suppression Chamber

The pressure suppression chamber is a steel pressure vessel in the shape of a torus located below and encircling the drywell, with a major diameter of 98 ft. 8 in. and a cross-sectional diameter of 25 ft. 8 in. (see Figure 4.1-7). The pressure suppression chamber contains the suppression pool and the gas space above the pool. The suppression chamber will transmit seismic loading to the reinforced-concrete foundation slab of the reactor building. Space is provided outside the chamber for inspection.

Inside the suppression chamber is the vent system distribution header. Projecting downward from the header are 48 downcomer pipes that terminate below the water surface of the pool. Connecting to the vent header are eight vent lines from the drywell. Columns extending from and attached to the bottom of the suppression chamber support the vent header and downcomers and also resist the upward reaction from the downcomers during blowdown. The columns are pinned at the top and bottom to accommodate the differential horizontal movement between the header and the suppression chamber.

The pressure suppression chamber is supported on 16 pairs of equally spaced columns. These supports transmit vertical loading to the reinforced-concrete foundation slab of the reactor building. Lateral loads due to an earthquake are transmitted to the foundation by four symmetrically placed earthquake ties.

Access to the pressure suppression chamber from the reactor building is through two manholes with double-gasketed bolted covers with a test connection between, which can be tested for leakage.

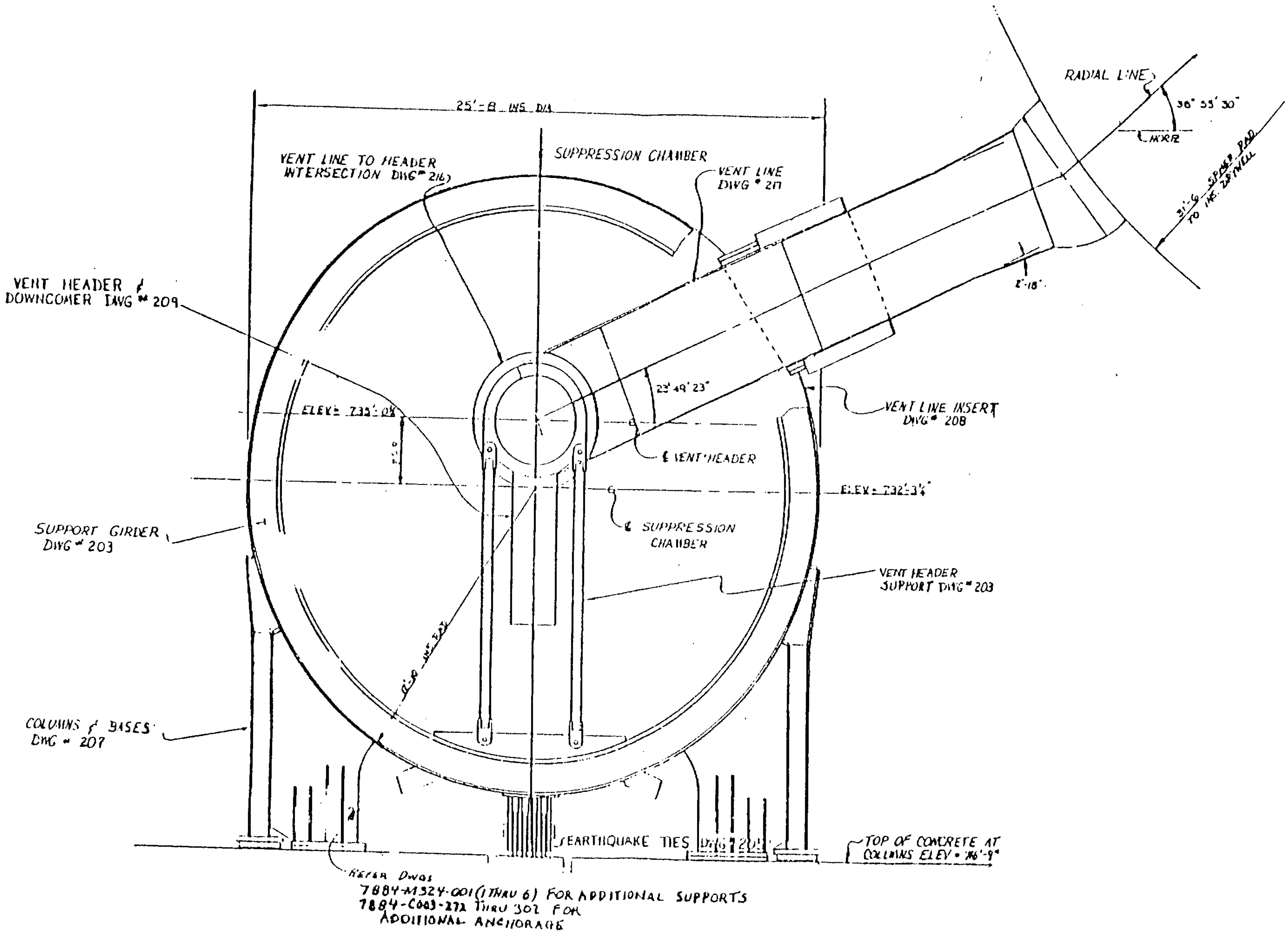


Figure 4.1-7 Suppression Chamber General Arrangement

The pressure suppression pool contains approximately 58,900 $ft.^3$ of demineralized water at low water level and approximately 61,500 $ft.^3$ at high suppression pool water level. Air volume above the pool changes as water volume changes from high and low levels. The suppression chamber is designed to serve many purposes, including the following:

- The suppression pool acts as the heat sink for all of the following:

  - A Loss of Coolant Accident (LOCA) within the drywell;
  - a safety/relief valve lift;
  - the HPCI and RCIC turbine exhaust.

  Energy is transferred to the suppression pool by the discharge piping from the safety/relief valves, the drywell vent system, the HPCI and RCIC system turbine exhaust pipes. The exhaust steam is discharged below the water surface and is condensed. The SRV discharge piping is used as the energy transfer path for any condition which requires relief valve operation. The drywell vent system is the energy transfer path for energy released to the drywell during a LOCA.

- The suppression pool serves as a source of water for the Reactor Core Isolation Cooling (RCIC), High Pressure Coolant Injection (HPCI), Core Spray (CS), and Low Pressure Coolant Injection (LPCI) systems.

- The suppression pool acts as an intermediate heat sink for transferring heat from the reactor and then to the RHR system in the suppression pool cooling mode.

The Reactor Recirculation piping rupture represents the most rapid energy addition to the pool. For this design basis accident, the vent system, which connects the drywell and suppression chamber, conducts a flow from the drywell to the suppression chamber without excessive resistance and distributes this flow effectively and uniformly in the pool. The pressure suppression pool receives this flow, condenses the steam portion, and releases the non-condensible gases to the pressure suppression chamber air space. An additional benefit, not part of the design basis, is that the suppression pool acts as an effective scrubber of fission products other than nobles gases when the release pathway is through the suppression pool.

The suppression pool receives steam and water energy from the reactor relief valve discharge piping or the drywell vent system downcomers which discharge under water. The steam, and any water carryover, cause an increase in pool volume and temperature. Energy can be removed from the suppression pool by the RHR system operating in suppression pool cooling mode.

The suppression pool water level and temperature are continuously monitored in the control room and are maintained within strict limits imposed by Technical Specification requirements.

In addition, as will be seen in the Level 2 analysis, the containment can also be flooded with water in response to an accident. This is a specified operator response action in the DAEC Emergency Operating Procedures.

Vent System

Large vent pipes connect the drywell and the pressure suppression chamber. A total of eight circular vent pipes are provided, each having a diameter of 4 ft. 9 in. The vent pipes are designed for the same pressure and temperature conditions as the drywell and suppression chamber. Jet deflectors are provided in the drywell at the entrance of each vent pipe to prevent possible damage to the vent pipes from jet forces that might accompany a pipe break in the drywell. The vent pipes are provided with two-ply expansion bellows to accommodate deferential motion between the drywell and suppression chamber.

The drywell vents are connected to a 3 ft. 6 in. diameter vent header in the form of a torus, which is contained within the airspace of the suppression chamber. Projecting downward from the header are 48 downcomer pipes, 24 in. in diameter and terminating not less than 3 ft. below the water surface of the pool.

Seven 20 in. vacuum breakers relieves pressure from the suppression chamber through the vent lines to the drywell to prevent a significant pressure differential between the drywell and suppression chamber. These vacuum breakers also prevent a backflow of water from the suppression pool into the vent system and prevent excessive water level oscillation within the downcomer pipes. Two 20 in. vacuum breakers are provided from the reactor building to the torus to prevent an external pressure on the containment greater than 2 psi.

There are six safety relief valve (SRV) vent lines (10 in. diameter) extending into the center of the torus and terminating below the pool water level in tee connections (the T-quenchers). The discharge from these tees is horizontal in two directions along the torus centerline.

Vacuum Breakers

There are two groups of vacuum breakers: the torus-to-drywell group, which is connected to the vent header inside the torus and prevents drywell pressure from being significantly less than torus pressure; and the reactor building-to-torus group, which prevents the torus pressure from being significantly lower than building pressure.

Reactor Building-Torus Group

A vacuum breaker and an air-operated butterfly valve are located in series on each of two lines that run from the reactor building to a common line (20-in. diameter) that penetrates the torus. The butterfly valve is actuated by a differential pressure and requires pneumatic pressure and DC power. The vacuum breaker is self-actuating, and it can be locally operated for testing purposes.

The vacuum breaker system is of adequate size to prevent pressure in either the drywell or the pressure suppression chamber from exceeding their negative design pressure (2 psi) as a result of the most-rapid-cooldown transient that can occur during normal operation or postulated accident condition assuming the failure of a single valve to open.

Suppression Chamber-Drywell Group

The internal torus-drywell vacuum breakers consists of seven 20-inch inside diameter (ID) swing check valves that relieve a negative pressure differential between the torus and the drywell. Vacuum breaker valves are located on the vent header within the airspace of the suppression chamber. These valves prevent excessive water level variations in the submerged portion of the vent downcomer lines. The capacity is adequate to limit the pressure differential between the suppression chamber and the drywell during post accident drywell cooling operations to a value that is within the suppression system design values.

Primary Containment Penetrations

In order to maintain design containment integrity, containment penetrations have the following design characteristics:

- They are designed for the same pressure and temperature conditions as the drywell and pressure suppression chamber.

- They are capable of withstanding the forces caused by the impingement of the fluid from the rupture of the largest local pipe or connection without failure.

- They are capable of accommodating the thermal and mechanical stresses that may be encountered during all modes of operation without failure.

- They are capable of withstanding the maximum reaction that the pipe to which they are attached is capable of exerting.

The types of containment penetrations are as follows:

- Pipe penetrations

- Electrical penetrations

- Traversing In-Core Probe (TIP) penetrations

- Personnel and Equipment Access Locks and Hatches

- Access to the Pressure Suppression Chamber

- Access for Refueling Operations.

4.1.2 Summary of Secondary Containment Features

The secondary containment system consists of four subsystems, which are the reactor building, the reactor building isolation and control system, the standby gas treatment system, and the offgas stack. The secondary containment system surrounds the primary containment system and is designed to provide secondary containment for the postulated LOCA. The secondary containment system also surrounds the refueling facilities and is designed to provide primary containment for the postulated refueling accident (see Figures 4.1-1 through 4.1-4).

The safety design bases of the secondary containment system as described in the DAEC UFSAR are as follows:

1. The secondary containment system is designed to provide secondary containment when the primary containment is operable and when the primary containment is open.

2. The secondary containment system is designed with sufficient redundancy so that no single active system component failure can prevent the system from achieving its safety objective.

3. The secondary containment system is designed in accordance with Seismic Category I design criteria.

4. The secondary containment is designed to provide a filtered, elevated release of airborne radioactive materials so that offsite doses from a design-basis fuel-handling accident or LOCA will be below the guideline values stated in 10 CFR 100.

5. The reactor building is designed to contain a positive internal pressure of at least 7 in. of water.

6. The secondary containment system is designed to be sufficiently leak-tight to allow the standby gas treatment system (SGTS) to maintain the reactor building pressure at a sub-atmospheric pressure of 0.25 in. of water when the standby gas treatment system is exhausting reactor building atmosphere.

7. The reactor building isolation and control system is designed to isolate the reactor building fast enough to prevent fission products from the postulated fuel-handling accident from being released to the environments through the normal discharge paths.

8. The secondary containment system is provided with means to conduct periodic tests to verify system performance.

The secondary containment system uses four different features to mitigate the consequences of a postulated LOCA (pipe break inside the drywell) and the refueling accident(fuel-handling accident). The first feature is a negative pressure barrier that minimizes the ground-level release of fission products by ensuring that all leakage relative to the environment is into the secondary containment. The second feature is a low-leakage containment volume that provides a holdup time for fission product decay before release. The third feature is the removal of particulate and iodines by filtration before release, and the fourth feature is the exhausting of the secondary containment atmosphere through an elevated release point, which aids in the dispersion of the effluent by atmospheric diffusion. Each of the features is provided by a different combination of subsystems: the first by the reactor building, the reactor building isolation and control system, and the standby gas treatment exhaust system; the second by the reactor building and the reactor building isolation and control system; the third by the standby gas treatment system filters; and the fourth by the offgas stack.

Reactor Building

The reactor building completely encloses the reactor and its pressure suppression primary containment system. The reactor building houses the refueling and reactor servicing equipment, new and spent-fuel storage facilities, and other reactor auxiliary and service equipment. Also housed within the reactor building are the emergency core cooling systems, reactor cleanup filter-demineralizer system, RCIC system, ventilation and exhaust systems, standby liquid control system, CRD system, reactor protection system, and electrical equipment components (see Figures 4.1-1 through 4.1-4).

The DAEC reactor building is compartmentalized below the ground elevation (el. 757'). The various ECCS corner rooms form these compartments at elevation 716'. The reactor building is comparatively open on the elevations above the ground floor. Entrance to the reactor building is through personnel airlock on the north wall. Entrance to each of the corner rooms is via a stairwell in each corner of the reactor building. All stairwells to the corner rooms have doors (except SW corner) that open into the stairwells from the ground elevation in the reactor building. Watertight doors cover the access to the torus room from each of the corner rooms and from the SE corner room to the HPCI room. From the ground floor to the refuel floor (el. 855'), a communication path exists between the reactor building elevations (el. 757', 786', 812', and 833') in the form of a large equipment hoistway. The dimensions of the equipment hoistway is 20' x 18' and forms the main communication path in the reactor building below the refuel floor. This equipment hoistway has a metal cover which is closed during normal plant operation. Equipment hatches to the corner rooms and the torus room are located on the ground floor of the reactor building. Normally closed airlock railroad doors (one interior and one an exterior door) are located in the southwest corner of the reactor building on the ground floor. Access from the reactor building to drywell is through the containment equipment hatch, personnel airlock to drywell, and the CRD removal hatch on the ground elevation.

Access to the refueling floor is through a stairwell from the 4th floor, or the main elevator. The access door opens into the stairwell. The walls of the refueling floor are made of sheet metal siding instead of concrete. There are 4 (~ 5' x 10') blowout panels (one in each wall) that fail at 0.25 psi. These blowout panels appear to be the primary path from the reactor building to the environment. All other doors to the environment are pressure doors or airlocks. In addition, the equipment hoistway cover will have to lift for a break anywhere in the reactor building to release into the refuel floor.

Reactor Building Isolation and Control System

The reactor building isolation and control system serves to trip the reactor building supply and exhaust fans, isolate the normal ventilation system, and provide the starting signals for the standby gas treatment system in the event of the postulated LOCA inside the drywell or the postulated fuel-handling accident in the reactor building. Three signals will automatically initiate the secondary containment isolation system. Two signals, high drywell pressure and low reactor water level, indicate a LOCA inside the drywell. Radiation monitors in the operating (refueling) floor ventilation exhaust duct, which indicate a fuel-handling accident, can also initiate the secondary containment isolation system. Secondary containment isolation can also be initiated manually from the control room.

Normally open air-operated isolation dampers are provided on the discharge side of the reactor building and operating floor supply fans. Similar isolation dampers are located in the intakes to the operating floor ventilation exhaust fans and to the contaminated area exhaust fans. Two dampers in series are provided throughout the isolation system to provide the required redundancy. Both dampers fail closed on a loss of power to the solenoids, or on a loss of instrument air to the dampers. The isolation dampers are spring operated and designed to close before fission products from the design-basis refueling accident can travel distance between radiation monitors and the isolation dampers.

Penetrations of the secondary containment are designed to have leakage characteristics consistent with secondary containment leakage limitations. Electrical penetrations in the reactor building are designed to withstand environmental conditions and to retain their integrity during the postulated fuel-handling accident and the LOCA inside the drywell. Two doors on the equipment/personnel access lock are provided with interlocks to ensure that building access can not interfere with maintaining the secondary containment integrity. All normally open drains that are open both to the secondary containment and outside atmosphere are provided with water seals to maintain containment integrity.

Standby Gas Treatment System

The standby gas treatment system consists of two identical parallel air filtration assemblies located at elevation 786 ft. of the reactor building. Each of the filtration assemblies has full capacity. With the reactor building isolated, each train can hold the building at a sub-atmospheric pressure of 0.25 in. of water.

The physical arrangement of the standby gas treatment system is such that redundant units are physically separated by both space and structural components in the reactor building.

The standby gas treatment system dampers are designed to fail to positions that provide open flow paths through the filter trains. The dampers are arranged such that a loss of air, a loss of dc power, or a failure of a logic channel will not prevent exhausting all reactor building areas through one of two filter trains.

The standby gas treatment system is protected from overpressurization by a relief damper installed in the standby gas treatment system suction ductwork. The relief damper is actuated within 0.1 seconds when the differential pressure between the suction ductwork and the secondary containment exceeds 10 inches of water.

Upon receiving the required initiation signal, all normal reactor building ventilation is isolated, both standby gas treatment system filter trains start, all standby gas treatment isolation dampers open, and each fan draws air from the isolated reactor building.

In order to achieve the design differential pressure as rapidly as possible and reduce the possibility of exfiltration, both trains start initially. After the operation of each train is verified by flow indicators in the control room, the control room operator switches one train to OFF and then back to AUTO. In this condition, the train will not start unless low flow in the operating train occurs.

Each train has maximum flow rate of 4000 cfm. Automatically operated exhaust fan inlet vane controls maintain the required flow rate to establish 0.25 in. of water sub-atmospheric pressure. The filtration system has a capability of removing in excess of 99% of the iodine in the air stream with 10% of the iodine in the form of methyl iodide under entering conditions of 70% relative humidity. HEPA filters having an efficiency of 99.97% for particles greater than 0.3 μm are located upstream and downstream of the iodine absorber in each train.

The HEPA filters are Cambridge Type 1E-1000 with aluminum separators, galvanized steel frames, fiberglass-asbestos filter media, and a fire-resistant rubber-base sealant compound. The sealant will withstand an accumulated radiation exposure of at least 1 x 10^8 rads at a continuous operating temperature of 250°F. This radiation exposure is approximately two orders of magnitude greater than the exposure that could be accumulated by the upstream HEPA filter of the standby gas treatment system, assuming that it collects the particulate leaking from the primary containment as a result of a LOCA releasing the hypothetical TID-14844 fission product source term.

The activated carbon iodine filter is a high-efficiency deep-bed type with a 6 in. layer of charcoal, activated for trapping elemental iodine and radioiodine in the form of organic compounds.

The deep-bed filter in each train contains 1240 lb of potassium iodide impregnated, activated charcoal.

Each lot of charcoal is tested to ensure that its quality meets design requirements. Representative samples from each lot are shown to be capable of removing at least 99.95% of molecular iodine-131 and 99.9% of methyl iodine-131. Both removal efficiencies are determined in the presence of 50 mg/m^3 of nonradioactive iodide. This performance level is maintained until the amounts retained have reached the equivalent of 2500 g of molecular iodine and 200 g of methyl iodide in the full-scale systems. Following these loadings, air at 70% relative humidity and 150°F is drawn through the samples at rated flow for 2 hours. The intergrade removal efficiencies including both feed and air flow periods are at least 99.9% for molecular iodine and 90% for methyl iodide.

The filter unit consists of six individual charcoal beds connected in parallel to a common inlet plenum. Each charcoal bed is contained and formed by a rectangular-shaped perforated metal inlet enclosure within an outer perforated metal enclosure. These enclosures form the inlet and outlet boundaries, respectively, of the beds. The perforated metal inlet enclosure is completely submerged within the bed of charcoal.

The reactor building ventilation supply and exhaust fans are tripped, the normal reactor building ventilation is isolated, and both trains of the standby gas treatment systems are started on the receipt of a signal that indicates that either a fuel-handling accident or a LOCA have taken place. Any of the following conditions is sufficient to initiate this action: the detection of high radiation in the refueling floor ventilation exhaust duct or in the reactor building exhaust duct; high drywell pressure; or low reactor water level. The system can also be manually started from the control room. On the receipt of any one of these signals, each damper required for reactor building isolation and standby gas treatment system initiation is designed to go to its required position within 10 seconds except as noted in Table 9.4-2 of DAEC UFSAR. The reactor building ventilation supply and exhaust fans trip and the SGTS fans start immediately. The SGTS fans will be up to speed in less than 10 seconds after the receipt of a start signal.

When system flow has been verified, one train is manually stopped and placed in a standby condition. Cross-connections between the filter trains are provided to maintain the required decay heat removal cooling air flow on the charcoal filters in the inactive train. The system discharges to the offgas stack. The standby gas treatment system is powered from independent emergency service portions of the auxiliary power distribution system.

Offgas Stack

The offgas stack provides an elevated release point for airborne activity during the postulated loss-of-coolant and refueling accidents for the SGTS and the new hard pipe vent. The top of the stack is 100m above plant grade.

4.1.3 Containment Functional Capabilities

Previous PRAs, IDCOR, and the NRC (NUREG-1150) have demonstrated that BWR containments are capable of preventing radionuclide releases to the environment under severe accident conditions. The capability is a result of a series of multiple barriers, including both active and passive barriers. Figure 4.1-1 illustrates the DAEC BWR Mark I passive containment structure.

The primary mission of the containment barriers under severe accident conditions, as assessed here, is to protect the health and safety of the public. This mission can be carried out through the following functional capabilities:

- Protect the containment boundary
- Preserve water injection capability to the RPV and containment
- Minimize radionuclide releases.

An important aspect of the containment capability is the interface with possible operator actions. This interface will be shown to be important in the containment event tree analysis, i.e., specific operator actions using active mitigation measures can modify the radionuclide release timing and magnitude. Therefore, the investigation of containment capability is integrally tied to operator action for a BWR.

The containment functional capabilities are discussed under the topics of BWR Mark I passive and active barriers:

Passive Mitigation Measures

Primary containment

Secondary containment: Reactor Building

Active Mitigation Measures

Post core melt coolant injection and spray

Post core melt containment spray

Containment heat removal including containment venting from the wetwell

Suppression pool scrubbing through venting

4.1.3.1 Primary Containment

The primary containment can assist in both preventing and mitigating severe accidents. The design of the BWR pressure suppression containment uses large quantities of water inside containment to provide a passive heat sink for post-accident mitigation. The principal design basis accident governing many of the design features of the containment is the Design Basis Accident (DBA) LOCA. This double ended shear of a primary system pipe is explicitly in the deterministic calculations supporting the design basis of the containment. The DAEC containment is, therefore, explicitly designed to function adequately with coolant injection occurring to the RPV and containment heat removal occurring via the RHR system despite the possibility of a large break in the primary system piping. Substantial margin exists beyond this design basis event which allows adequate containment response for even more severe accidents.

The DAEC primary containment has the following design functions which are included to prevent releases and preserve critical safety functions during design basis accidents:

- Containment isolation to prevent releases from the RPV or the containment.

- Low leakage capability to assure substantial retention of accident products within the containment.

- High internal pressure capability (Design pressure of 56 psig).

- Vapor suppression capability to condense large quantities of steam in the water volume located in the wetwell below the reactor vessel.

- Inerted containment to prevent hydrogen detonation or deflagration during accidents.

- Source of water for coolant injection which can be continuously recirculated to the core for cooling.

- Containment heat removal capability through the RHR system.

- Drywell and wetwell sprays that can be used for temperature and pressure control within containment.

The above features are part of the design basis of the DAEC plant. In addition, there is substantial margin included that establishes higher realistic estimates for these capabilities.

This substantial margin associated with the design and construction of the containment features are translated into the following realistic severe accident mitigation capabilities based upon available analyses:

- High internal pressure capability of approximately 140 psig at temperatures below 500°F.

- High internal temperature capability of over 700°F at design pressures.

In addition, the DAEC containment also offers other features which are useful in the mitigation of postulated severe accident phenomena:

- The DAEC suppression pool provides an effective means to scrub fission products from possible release when the releases are directed to the pool via the SRV discharge from the RPV or through the downcomers from the drywell.

- The drywell and wetwell sprays provide a means to scavenge fission products from the containment atmosphere and prevent or substantially delay their release (not applicable to noble gases).

- Containment venting capability for both combustible gas control and containment overpressure protection is included in the DAEC emergency procedures.

These elements and their interrelationships with other mitigation features such as the reactor building and post-core melt coolant injection to the containment are described in more detail in the containment event tree development.

4.1.3.2 Secondary Containment

The secondary containment or reactor building provides potential mitigation of radionuclide releases during design basis and severe accidents. The reactor building completely encloses the reactor and its pressure suppression primary containment.

The reactor building provides secondary containment when the primary containment is closed and in service, and provides primary containment when the primary containment is open, as it is during refueling. The reactor building houses the refueling and reactor servicing equipment, new and spent fuel storage facilities, and other reactor auxiliary and service equipment. Also housed within the reactor building are the emergency core cooling systems, reactor cleanup filter-demineralizer system, RCIC system, ventilation and exhaust systems, standby liquid control system, CRD system, reactor protection system, and electrical equipment components.

The reactor building includes the steam tunnel containing the outboard main steam isolation valves (MSIVs), the main steam lines up to the turbine building, the feedwater lines, and the outboard feedwater line isolation valves.

A significant reduction in the magnitude of radioactive releases can potentially occur in the reactor building. However, reactor buildings or secondary containment configurations differ greatly among plants. For the secondary containment to retain a significant quantity of fission products, one of two conditions must occur:

- First, in many cases what might be loosely refer to as "active" decontamination measures should be available. This would include scrubbing due to the passage of fission products through deep water pools, decontamination by ventilation system filters, or scrubbing due to wide-coverage fire sprays. If such measures are functional, they would generally overwhelm the natural settling processes and result in relatively small environmental releases of all fission products except for noble gases. A few qualifications to this statement must be offered, however. First, ventilation filters are not usually designed for the large aerosol loadings they would see in a severe accident and consequently may tear, overheat, or clog. Second, fire sprays may not cover the area of all the affected secondary containment regions. Finally, while aerosol behavior is relatively well understood, there are significant uncertainties associated with the effectiveness of scrubbing fission product vapors in water pools; these might impact the release when the source of fission products is at a very high temperature.

If no such active measures are at work, a natural settling processes must be relied upon. For this to be effective, the fission products may require a relatively long residence time in the secondary containment before they can be swept to the environment. This in turn requires that the ventilation systems be secured, that the flowrate from the primary system or containment be relatively small, and that vigorous natural circulation be avoided between the secondary containment and the environment. The last of these requirements is often the most difficult to confirm. Vigorous natural circulation between the secondary containment and the environment can be set up if one large hole is opened (leading to large counter-current flows through the one opening), or if two holes are opened, one low in the building and one higher up. This latter configuration gives rise to a "chimney-like" flow pattern.

Consequently, given the possible different removal mechanisms and the associated effectiveness of each, the reactor building is examined and modeled in performing fission product release calculations as part of the DAEC IPE effort.

The effectiveness of the reactor building in mitigating radionuclide releases is dependent upon the following key features:

- Reactor building isolation
- Prevention of hydrogen detonation or deflagration induced failures
- Prevention of bypass mechanisms
- Prevention of the SGTS flow path leading to a forced circulation discharge with failed filters
- Actuation of fire suppression sprays (to a lesser degree).

The reactor building isolation system is designed to protect against possible post-accident contamination for the range of design basis accidents.

Successful severe accident mitigation by the reactor building is characterized if the DAEC secondary containment significantly reduces the radionuclides released to the environment during a severe accident. This radiation release mitigation can result from passive features of the reactor building such as:

- Large, cold surface areas
- Tortuous paths

- No recirculation paths to enhance effluent discharge

- No forced circulation, high flow rate discharge paths that can lead to blowing effluent from the reactor building

- Reactor building isolation.

Figures 4.1-8 and 4.1-9 display the nodal modeling used in the deterministic assessment of reactor building response. The MAAP deterministic nodal modeling of the reactor building shows that the pathways from the reactor building are through:

- The steam tunnel blowout panels

- The railroad door

- The refuel floor blowout panels.

These pathways will be important in the discussion of the reactor building effectiveness in limiting radionuclide releases to the environment. The DAEC reactor building is compartmentalized below the ground elevation. The various ECCS corner rooms form these compartments at elevation 716'. The reactor building is comparatively open on the elevations above the ground floor. Entrance to the reactor building is through personnel airlock on the north wall. Entrance to each of the corner rooms is via a stairwell in each corner of the reactor building. All stairwells to the corner rooms have doors (except SW corner) that open into the stairwells from the ground elevation in the reactor building. Watertight doors cover the access to the torus room from each of the corner rooms and from the SE corner room to the HPCI room. From the ground floor to the refuel floor, communication paths between the reactor building elevations exist in the form of a large equipment hatch. This equipment hatch forms the main communication path in the

Notes

J-1 HPCI to First Floor Thru Stairwell $A \approx 1\ ft^2$

J-2 RCIC to First Floor Thru Stairwell $A \approx 1.5\ ft^2$

J-3 HPCI to Torus Vent $A \approx 1\ ft^2$

J-4 RCIC to Torus Vent $A \approx 1\ ft^2$

J-5 Torus to Each Corner Room $A \approx 4 \times 1\ ft^2$

J-6 Corner Room Vents to First Floor Thru Stairwell $A \approx 1\ ft^2$

J-7 Equipment Hatch $A = 361\ ft^2$

J-8 Equipment Hatch $A = 361\ ft^2$

J-9 Normal Leakage $A = 0.79\ ft^2$

J-10 Equipment Hatch $A = 361\ ft^2$

J-11 Normal Leakage $A = 1.19\ ft^2$

J-12 Torus Room 786' Pipe Chase

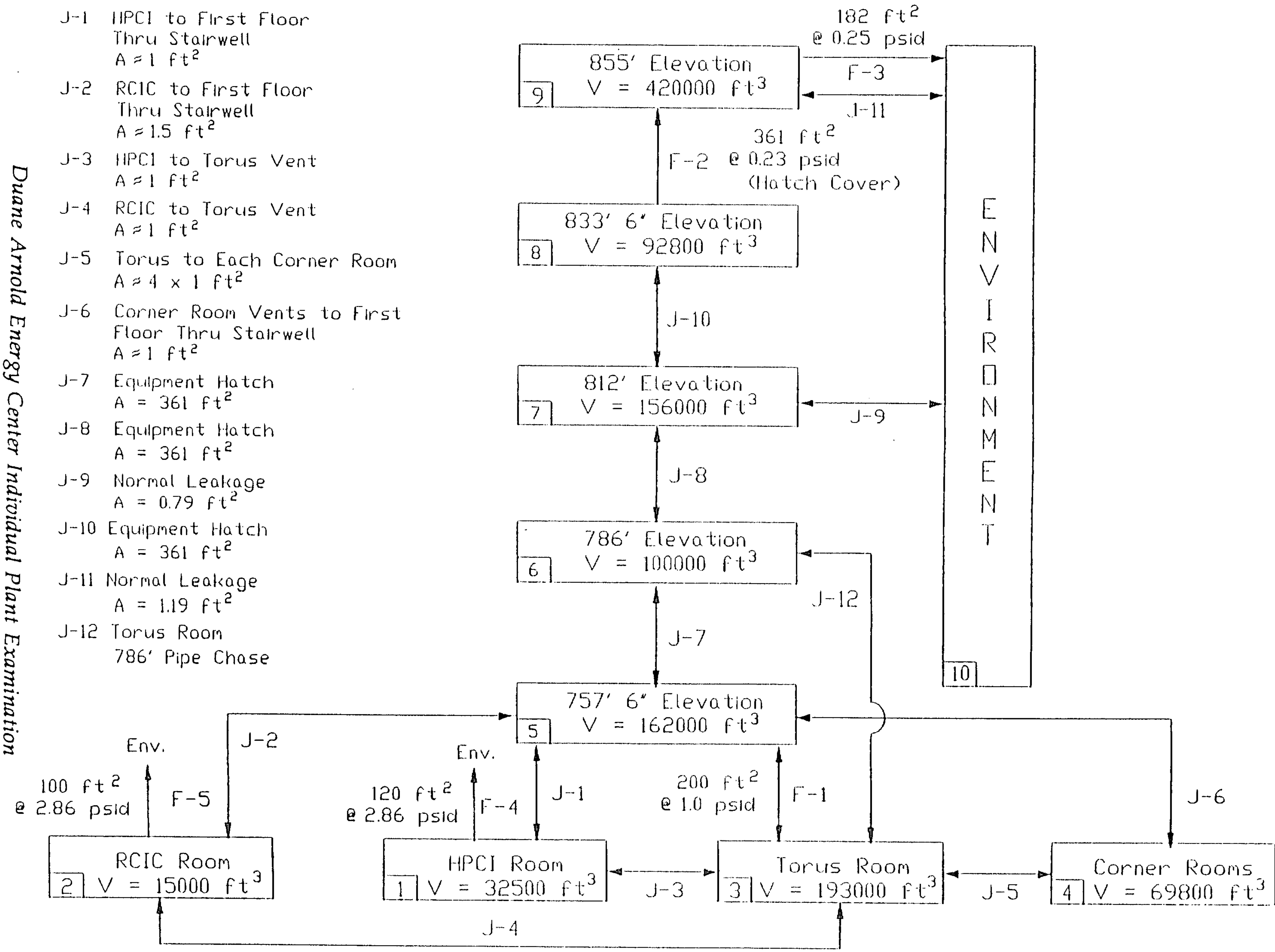
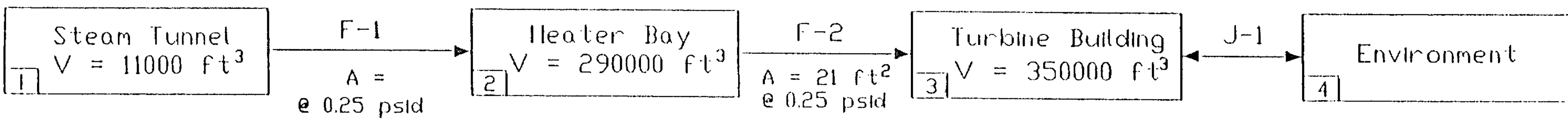


Figure 4.1-8 Reactor Building - Default Nodalization



NOTES

F-1 Represents Steam Tunnel Blowout Panel to Heater Bay

F-2 Represents an Assumed Door Opening to Turbine Building Proper @ 0.25 psid

J-1 Represents Normal Leakage to Environment Nominally Set to Size of Door (This is Large Enough to Prevent Pressurization in Turbine Building)

Figure 4.1-9 - Reactor Building - Steam Line Break Nodalization reactor building below the refuel floor. This equipment hatch has a metal cover which is closed during normal plant operation. Equipment hatches to the corner rooms and the torus room are located on the ground floor of the reactor building. Normally closed airlock railroad doors (an interior and an exterior door) are located in the southwest corner of the reactor building on the ground floor.

The reactor building steam tunnel is included as part of the secondary containment. The main steam and main feedwater lines run through the reactor building steam tunnel to the turbine building steam tunnel. The outboard MSIVs are in the steam tunnel as well as the steam line drain valves. In the event of a line break in the steam tunnel, no communication to the main reactor building is expected.

The reactor building steam tunnel penetrates the primary containment at the first floor, elevation 757'6" and then runs upward where it continues on to the turbine building. A blowout panel in the steam tunnel is designed to fail at approximately 0.25 psig in the event of a break in the steam tunnel. The release would be directed to the Heater Bay. The Heater Bay is connected to the turbine building via a regular steel door that opens into the reactor building. In the event of a break in the steam tunnel and subsequent failure of the blowout panels, the Heater Bay would be pressurized until the door to the turbine building fails (at approximately 0.25 psig). The turbine building is not expected to pressurize but to leak fluid/gas to the environment due to the nature of the sheet metal walls.

Fire sprinkler systems are only credited in the HPCI and RCIC rooms. These fire sprinklers are assumed to activate at a room temperature of 212°F. The fire protection spray system would only increase fission product retention in the event of HPCI/RCIC steam line break into either of these rooms. No other spray systems are credited in the reactor building, steam tunnel, or turbine building.

Blowout panels on the refuel floor are assumed to open to the environment once the pressure differential exceeds 0.25 psid (Failure Path #3, Figure 4.1-8). The equipment hatch on the refuel floor is modeled (failure path #2, Figure 4.1-8) and will lift at approximately 0.25 psid.

Finally, the vent path is to the environment directly. Containment venting is modeled in the DAEC IPE. A hardpipe containment vent is currently being installed. Therefore, containment torus vent releases are modeled as though they are released directly to the environment. In the event of a drywell vent, the refuel floor blowout panels will open, and the stairwell doors and equipment hatch cover will remain shut. This flow path will offer a minimum amount of retention because only the refuel floor volume will be available to provide residence time for fission product removal. No retention will occur during a wetwell vent.

4.1.3.3 Post-Core Melt Coolant Injection

Post core melt coolant injection to the containment provides a method for:

- RPV and containment temperature control
- Containment pressure control
- Debris coolability; limiting debris movement in the containment and limiting the amount of debris-concrete interaction and consequential non-condensible gas generation
- Scavenging of fission products from debris or containment atmosphere.

DAEC has many coolant injection systems which are capable of supplying water to the reactor from sources internal or external to the primary containment. These water injection sources are useful in satisfying many containment control functions.

The systems available for these operations include:

- Control Rod Drive Pumps
- Low Pressure Coolant Injection
- Core Spray
- Standby Liquid Control (SLC) System
- RHR Service Water
- Fire System
- General Service Water
- Emergency Service Water
- Condensate/Demin. Service Water

Core melt accidents are postulated for a large number of reasons, for example:

- Loss of high pressure coolant makeup and failure to depressurize the RPV; and

- Failure of adequate AC power.

These two examples involve cases in which the coolant injection sources can be made effective after RPV melt-through due to either depressurization, recovery of AC power, or completion of required operator actions within the time following core melt but before containment failure (e.g., 6 hours). Therefore, for certain accident sequences the likelihood of containment coolant injection may be found to be very high.

4.1.3.4 Post-Core Melt Containment Spray

The DAEC containment drywell sprays provide a means for both pressure and temperature control using a number of sources with a variety of pumps. When RHR pumps are available, the containment sprays can be manually initiated rapidly from the control room.

4.1.3.5 Containment Heat Removal

The DAEC IPE assessed the effectiveness of several decay heat removal systems including potential inter-system dependencies and identifies the specific functions being performed and potential failure causes. This includes consideration of the use of systems and actions not normally intended for decay heat removal. The IPE also considers the availability of procedures to accomplish such use.

Containment heat removal can be accomplished using a variety of methods including:

Safety Related

- RHR

Non-Safety Related

- Main Condenser
- RWCU
- Spent fuel pool heat exchanger (this can only be accomplished after shutdown with reactor cavity flooded up and fuel pool gates removed)
- Venting (see Section 4.1.3.6 below)

The principal method of containment heat removal under severe accident conditions is RHR in the suppression pool cooling mode. This mitigation function can be effective in preventing containment overpressure challenges either before core melt or during core melt progression.

4.1.3.6 Containment Venting

Containment venting has been identified as a possible mitigative action for certain postulated accident sequences beyond the plant licensing design basis. The DAEC Emergency Operating Procedures specify that containment venting can be used as a last resort operator action in the event of rising containment pressure. The action is intended to prevent an uncontrolled breach of containment.

Containment venting has also been specified for combustible gas control to ensure that hydrogen deflagration will not cause a containment failure. Combustible gas control is through Standby Gas Treatment only.

4.2 PLANT MODELS AND METHODS FOR PHYSICAL PROCESSES

The modeling of physical processes and phenomena is carried out in two principal ways:

- First, computer code models, principally MAAP, are used to calculate plant response and radionuclide releases

- Secondly, generic issues which are not modeled by such codes are treated in a probabilistic manner within the containment event tree.

Subsection 4.2.1 documents the approach taken in the deterministic calculations to support the modeling of core melt progression in the containment event tree:

- Key event timing

- Containment pressure and temperatures

- Radionuclide releases.

The MAAP code is the primary tool used in the analysis; extensive MAAP calculations were performed to gain insights from possible variations in modeling or assumptions.

Subsection 4.2.2 documents the phenomena not included explicitly in the MAAP model. Subsection 4.2.3 provides a list of key assumptions used in the analysis. Subsection 4.9 summarizes some of the formal sensitivity evaluations in support of the IPE.

4.2.1 Deterministic Evaluation in Support of Severe Accident Analysis

Primary and secondary containment response to pressures, temperatures, flow rates, and timing of actions was evaluated using the BWR Mark I version of the MAAP thermal hydraulic code (version 3.0B, revision 7.03). This code, using DAEC specific parameters as input, provided reactor and containment pressures, levels (water and radiation), and temperatures. Also calculated were the time windows between key events such as the onset of core melt and containment failure.

The MAAP results are used in the Level 2 analysis to determine success criteria, release timing and magnitude, the location of the containment failure, and time available for critical accident management actions.

Additionally, the result of several NRC sponsored Mark I containment analyses were used qualitatively in the evaluation.

4.2.1.1 Purpose of Using Deterministic Analysis

The assessment of plant response under postulated severe accident scenarios is a complex integrated evaluation. The primary and secondary containment building responses are sensitive to pressures, temperatures, flows, and event timings. These parameters also affect the timing of operator actions, the radionuclide releases, and the performance of mitigating systems. Therefore, the proper DAEC characterization of a severe accident progression is important to the realistic representation of the plant's response during the accident. These deterministic calculations are performed to estimate key plant parameters (e.g., pressures and temperatures in the RPV, the drywell, the wetwell, and the reactor building) as a function of time for various accident scenarios and the source term magnitude and timing of an impending radionuclide release.

This information is critical to the determination of the benefit associated with postulated recovery actions that could be implemented to mitigate specific effects of a severe accident. In addition, performing DAEC deterministic analysis helps to maximize the understanding of severe accident progression within DAEC engineering and operations departments to support future accident management programs.

4.2.1.2 Tools Available

There are several codes available which can be utilized to determine a plant specific response. Included in the list of codes are; MELCOR, STCP, BWRSAR, LTAS, and MAAP. Among these, only MAAP, MELCOR, and STCP are fully integrated codes capable of modeling all aspects of a severe accident while representing all important interrelationships between phenomena.

The computer codes required for such analyses must be able to address several fundamental needs. These fundamental needs include the following:

1. Quantification and refinement of system success criteria (primary and containment systems);

2. Quantification of containment response to severe accident phenomena including the performance of containment systems, mission times, and response intervals;

3. Quantification of fission product releases (radionuclide magnitude, release timing, or con-current energy release);

4. Quantification of operator/recovery actions;

5. Ability to integrate the systems (frontend) and containment (backend) assessments.

Several codes are available that have been suggested for use for containment performance analyses. These codes are MAAP, MELCOR, STCP (Source Term Code Package), and BWRSAR (in combination with CONTAIN). In order to select the appropriate code to accomplish these tasks, their various attributes must be compared.

The MAAP code compares well when considering capability of such a tool for accomplishing the tasks described above. Furthermore, user support, QA requirements, NRC "acceptability," and the required user's knowledge of severe accident phenomena are other attributes that should be considered when choosing a tool. MAAP is unique in that EPRI supports it and provides direct user support via the MAAP Users Group (MUG) and will maintain an archived and controlled version in support of QA. However, most importantly, it is an integrated code package that can most completely model the widest spectrum of severe accidents.

4.2.1.3 Advantages of MAAP

MAAP is judged to be one of the most appropriate tools to use in support of the IPE. Factors that reinforce this decision are:

- According to NUMARC and EPRI estimates, approximately 40 utilities representing 60 plants are expected to use MAAP for IPEs or PRAs to meet the requirements of the generic letter. A program is currently underway by NUMARC, EPRI, and DOE to bring NRC up to speed on MAAP and thus to make the IPE submittal process more orderly.

- Among the competing tools, MAAP has the highest level of QA documentation. This documentation includes two EPRI-sponsored efforts: a recently-completed formal design review by respected independent authorities and an independent validation and verification program which included a line-by-line review of the source code.

- MAAP is being aggressively developed and maintained. Continued development is being funded by EPRI and US DOE, and included in the development was an extensive thermal-hydraulic benchmarking activity sponsored by EPRI.

- In comparison to some of the new NRC tools, the MAAP code is fast-running and relatively mature, with a considerable history of successful use at utilities. It is quite practicable and very common to run the code on 386 or 486-type personal computers.

- An active MAAP User's Group consisting of over 40 members exist through which helpful information is shared between utilities and other MAAP users.

- EPRI has developed a guideline document to provide the users with recommendations on selected parameter values. These recommendations will assist the user in addressing many of the key areas of uncertainty.

- EPRI has performed numerous sensitivity analyses using MAAP to better address some of the NRC questions on important phenomenology.

4.2.1.4 DAEC Unique Features Incorporated Into MAAP

The DAEC MAAP model includes several plant specific features that could not be handled by the generic parameter file. Several of these DAEC specific features were incorporated into the MAAP assessment methodology. These features include the following:

- The Heat Capacity Temperature Limit curve is modeled with the use of a MIPS user function.

- EOPs for containment/RPV venting, pool cooling, drywell sprays and containment flood were all modeled using MIPS input files.

- The drywell spray initiation limit was controlled with a MIPS user function to vary with drywell pressure per EOPs.

- The RHRSW cross-tie was used as the alternate injection source to the reactor vessel. Alternate injection was included in the engineered safeguard section of the parameter file.

4.2.1.5 MAAP Model Initialization

This subsection provides a brief summary of the background on the MAAP code initialization. Detailed information is retained on-site regarding the calculational files for the MAAP parameter file.

<u>DAEC MAAP Parameter File</u>

The DAEC MAAP parameter file has been assembled based on plant specific data. In some selected cases, there have been parameters used from the Peach Bottom reference parameter file.

Radioactive Inventory of Reactor Core

The potential radioactive source (fission products, transuranics, and activation products), in the reactor immediately preceding the initiation of an accident was obtained from analyses performed with the ORIGEN computer program. ORIGEN is used to describe the formation, transmutation and decay of nuclides.

The ORIGEN method includes:

1) treating the full range of transmutations that might occur.

2) computing nuclide concentrations for decay chains in which
   a) a nuclide decays to produce one of its precursors (e.g., neutron capture followed by alpha decay); or

   b) a nuclide decays to produce a daughter that is present in another decay chain.

Reactor Composition, Design, and Operating History

Radionuclide inventories were obtained from the Peach Bottom reference parameter file (MAAP 3.0B). These values were developed from an ORIGEN run and scaled based on the difference in core power between DAEC and Peach Bottom.

The inventory of isotopes that reach equilibrium during irradiation (short-lived) is directly proportional to the power density (neutron flux). The inventory of the long-lived radionuclides, however, is proportional to burnup (i.e., neutron flux times time) and is not sensitive to power density at any given exposure. Iodine is termed a "short-lived" isotope and the core inventory at shutdown is therefore proportional to power density. Cs is termed a "long-lived" isotope and the core inventory at shutdown is therefore proportional to burnup.

Selection of Radionuclides

The ORIGEN program calculates the time-dependent concentration of a very large number of nuclides: 246 activation products, 461 fission products, and 82 transuranics. Although many of these nuclides are not radioactive, the total number of radionuclides is quite large and significant amounts of computer storage and computational time would be required to handle all of them in the consequence model. At a very small sacrifice in the precision of the radiation dose calculations, the number of radionuclides considered can be reduced to a manageable size.

The elimination of radionuclides from consideration in radiation dose calculations was based on a number of parameters, such as quantity (curies), release fraction, radioactive half-life, emitted radiation type and energy, and chemical characteristics. In addition, it is possible to eliminate radionuclides with half-lives shorter than 25.7 minutes (decay constants greater than 4.5×10^{-4} sec^{-1}) because, the minimum delay time between termination of the chain-reaction (start of the accidents and the release of radioactive material to the atmosphere) would be at least 0.5 hour and could be greater than 10 hours.

Core Peaking Factors

The DAEC core peaking factors are characteristics of the beginning of cycle data. Actual plant data was used from the Periodic NSS Core Performance Log (9/25/90 07:30:10).

4.2.1.6 Deterministic Results

Section 4.7 describes the deterministic results that are used in the CET evaluation to determine the following:

- Success criteria
- Timing of release
- Failure location
- Radionuclide release magnitude.

4.2.2 Phenomena Not Included in MAAP Model

While the MAAP code is the primary deterministic assessment tool used in the containment evaluation, there are accident sequences and phenomena that the MAAP code is judged not to be effective. For these sequences and phenomena, separate effects analyses, experiments, or expert judgement are used in the evaluation process. This subsection is a brief review of these sequences and phenomena that fall into this category and the disposition of them for the DAEC IPE.

First, the DAEC IPE includes the assessment of phenomenological matters considering NRC positions on issues and related uncertainties including the issues as summarized in IDCOR Technical Report 86.1 (e.g., letters from T. Speis, NRC, to A. Buhl, ITC, "Position Papers for the NRC/IDCOR Technical Issues," dated September 22, 1986; November 26, 1986; and March 11, 1987). The IDCOR regulatory interaction program was devoted to the definition and resolution of open technical issues related to the assessment of severe accidents. Great progress has been made between the NRC and the Industry in resolving these issues through a variety of technical exchange meetings. Many of these issues manifest themselves as NRC concerns with specific models used in MAAP.

Table 4.2-1 lists the current status of each of the issues. While the status reflects the "resolution" of the issue for the DAEC IPE baseline calculation, Iowa Electric has investigated the possible sensitivities to the results that may occur as a result of these issues. Section 4.9 summarizes the results of these sensitivity evaluations.

Table 4.2-2 summarizes additional phenomenological issues of potential impact on the Level 2 results and their disposition in the DAEC IPE. These issues were not treated by MAAP.

4.2.3 Assumptions in the Modeling

In the course of a complex analysis, it is usually necessary to make assumptions or interpretation in order to model a system or group of systems.

Assumptions can introduce effects into the analysis that are realistic, conservative, or non-conservative.

The assumptions made in the analysis are meant to provide a realistic, best estimate basis for the evaluation. However, because of uncertainties, assumptions that may not be known to be a best estimate could be used. In general, conservative values will be chosen for this type of value.

Finally, there may be exceptions to the above two rules in which apparent non-conservatisms have been included in the analysis. These apparent non-conservatisms are generally present because of modeling simplicity. This section discusses both the conservatisms and non-conservatisms in the analysis. Analysis not highlighted here are considered to be "best estimate".

Table 4.2-1

PHENOMENA DISCUSSED BY NRC AND IDCOR

ISSUE	NRC POSITION AND CURRENT STATUS	DAEC IPE RESOLUTION
1. Fission product release prior to vessel failure	No substantial differences between NRC and industry models; no compelling evidence for volatile iodine release.	MAAP model used
2. Recirculation of coolant in the RPV	Mostly an issue for PWR sequences at very high pressure (e.g., blackout).	N/A
3. Release models for control rod material	No substantial differences between NRC and industry positions; no compelling evidence that B_4C affects iodine chemistry or hydrogen production.	MAAP model used
4. Fission product and aerosol deposition in primary system and containment	NRC concerned that MAAP aerosol correlations might be inadequate when transport times are short, e.g., when early containment failure occurs. Subsequent EPRI and USDOE sponsored comparisons of the model to detailed methods indicate that model is suitable for IPE use.	MAAP model used
5. In-Vessel Hydrogen Generation	NRC concerned that MAAP "blockage" model may seriously under-predict hydrogen production; EPRI is currently recommending that the model that may under predict H_2 not be used for base-case IPE calculations.	H_2 production maximized in MAAP base calculations; sensitivity performed
6. Core melt progression and vessel failure	NRC and IDCOR agreed that the mass of molten material available at vessel failure was uncertain and sensitivities to this quantity should be investigated when calculations are performed.	Sensitivity performed
7. In-vessel steam explosions leading to Alpha mode failure of containment	Steam Explosion Review Group as well as Industry experts subscribe to the view that steam explosions sufficient to fail containment do not contribute significantly to risk; this is consistent with NUREG-1150; issue considered largely resolved for IPEs.	Probabilistically treated
8. Direct containment heating	Considered primarily a PWR issue.	Probabilistically treated
9. Ex-vessel fission product release	MAAP model improved to include more chemical species; not likely to be major issue for IPEs.	MAAP model used
10. Ex-vessel heat transfer models from molten core to concrete	While uncertainties exist, MAAP model compares relatively well to experiment; unlikely to be major issue for IPEs.	MAAP model used for base case; sensitivities performed.

Table 4.2-1

PHENOMENA DISCUSSED BY NRC AND IDCOR

ISSUE	NRC POSITION AND CURRENT STATUS	DAEC IPE RESOLUTION
11. Revaporization of deposited fission products	IDCOR and NRC agree that MAAP uncertainty calculations should be performed to treat possibility of chemical reactions between volatile fission products and steel surfaces.	MAAP model used
12. Amount and timing of suppression pool bypass	Mainly an issue for Mark III; use of Vaughan model to assess plugging of leakage path by aerosols (as in MAAP) acceptable for flowpaths less than 1 cm wide.	N/A
13. Retention of fission products in ice beds	PWR ice condenser issue.	N/A
14. Modeling of emergency response	Not an issue for IPEs; issue resolved if analyst assumes that a fraction (e.g., 5 percent) of the population does not evacuate.	N/A
15. Containment performance	IDCOR and NRC agreed that a spectrum of failure sizes should be considered to address spectrum of failure pressures be considered; recent EPRI report lends credence to IDCOR leak-before-break assumption.	Spectrum of Failures included in both MAAP assessment and probabilistically in CET evaluation.
16. Secondary containment performance	In response to NRC concerns, MAAP model was made much more detailed. While NRC concerns focused on dependence of aerosol residence time, hydrogen burns in the secondary containment, and the rate of concrete off-gas production in containment, in most plants it appears that fission product retention will mainly depend on failure mode(s) of secondary containment and scrubbing.	MAAP model and probabilistic assessment used
17. Hydrogen Ignition and Burning	Not an issue for Mark I and II. NRC concerned with MAAP models for global burns and burns at igniters. MAAP models were significantly updated to address NRC concerns; not likely to be a major issue for IPEs.	N/A

Table 4.2-2

Additional Phenomena of Potential Impact on Level 2 Analysis

Sequence or Phenomena	Disposition in DAEC IPE
Ex-vessel Steam Explosion	Treated probabilistically in the IPE assessment
Mark I Shell Failure	Treated probabilistically using work by Theofanous (NUREG/CR-5423)
Direct Impingement Induced Failure	Treated probabilistically
Direct Containment Heating	Separate effects analysis and treated probabilistically
Reactivity Insertion during Core Melt Progression	Separate effects analysis and treated probabilistically

4.2.3.1 Assumptions

This subsection provides a list of general assumptions that have been used in the Level 2 PRA analysis.

Level 1 Interface

- Each of the Level 1 end states represents a core damage situation in which the RPV water level is below 1/3 core height and decreasing as a result of insufficient coolant makeup to the RPV, or a loss of containment integrity that is presumed to severely challenge the continued operation of these injection systems.

Level 2

- The containment event tree structure has been structured to be as concise as possible, but at the same time sufficiently detailed to represent important functional events that can result in significant differences in containment survivability, or the magnitude or timing of radionuclide releases.

- The list of containment failure modes considered in the Level 2 assessment is believed to be comprehensive, including all published failure modes; however, there may be other failure modes not currently postulated or known that could also compromise the containment. Section 4.4 summarizes the failure modes and their treatment.

- The containment capability has been assessed based on extrapolation of detailed deterministic calculations at "low" temperatures. The further extrapolation of the containment capability to high temperatures and pressures has been performed using separate effects assessments and engineering judgement.

- The response of the containment to severe accidents (i.e., calculated pressures and temperatures) is modeled using the MAAP code. The results have been checked against other published deterministic codes from similar plants.

- The calculated source terms (i.e., radionuclide release magnitude and timing) have been determined using the MAAP code. These accident source terms have been compared with other deterministic code calculations for similar plants.

- Generally, the treatment of hardware repair and recovery is explicitly treated in the Level 1 analysis. The Level 2 model considers repair and recovery of systems that primarily affect the ability of the operator to maintain RPV coolant inventory (e.g., high pressure and low pressure injection systems, and EDGs). Level 1 scenarios that result in the loss of containment integrity prior to core damage usually do not include any additional opportunity to restore injection system in the Level 2 analysis to prevent core damage (i.e., due to minimum time frames and severe secondary containment conditions).

- Revision 7.0.3 of MAAP 3.0B was not yet capable of modeling the response to a containment flood sequences without some modeling intervention to match-up plant features that MAAP can treat. Therefore, changes in the gas space volume and the location of the vacuum breaker in the wetwell were used to allow the MAAP code to converge. These changes are believed to have an insignificant effect on the results.

- CS and LPCI are not assumed to have high reliability if the equipment is available following containment venting. Both the NPSH and reactor building environmental conditions are considered contributors.

- The DAEC containment is normally inerted; therefore, hydrogen combustion is not a dominant contributor to the release frequency. It was assumed that hydrogen combustion occurs due to the presence of numerous electrical components whenever the core is damaged and the containment is not inerted and combustible gas control actions are not taken. Furthermore, it is assumed that hydrogen combustion always produces a large release of radioactivity (i.e., no credit was taken for frequent periodic burning of small amounts of combustible gases to limit the pressure rise).

- The containment vent valves remain operational after containment venting.

- Representative sequences for MAAP evaluation are chosen conservatively, and a number sequences are calculated to lead to similar release bins.

- Retention of debris in-vessel even after substantial core degradation has been included in the IPE assessment. This assessment has been included based on the time available for adequate recovery, and the insights from NRC sponsored computer models, e.g., BWRSAR, MARCH, MELCOR.

- The DAEC IPE treats phenomenological uncertainties through sensitivity studies performed with MAAP as well as using insights from other studies. Selection of the sensitivity runs are generally consistent with those given in the EPRI draft report "Recommended Sensitivity Analyses for an Individual Plant Examination using MAAP 3.0B."

- The containment response assessment and the evaluation of radionuclide release timing and magnitude is performed with DAEC specific MAAP calculations.

- The DAEC IPE considers the possible outcomes resulting from the potential of direct containment heating and shell meltthrough. This flexibility allows a baseline quantification and sensitivities to each of these phenomena for different accident management actions.

- The potential for hydrogen combustion in the reactor building has been assessed using the MAAP code. In addition, a probability that extensive reactor building hydrogen combustion occurs regardless of MAAP calculations has also been accounted for in the reactor building effectiveness node of the containment event tree.

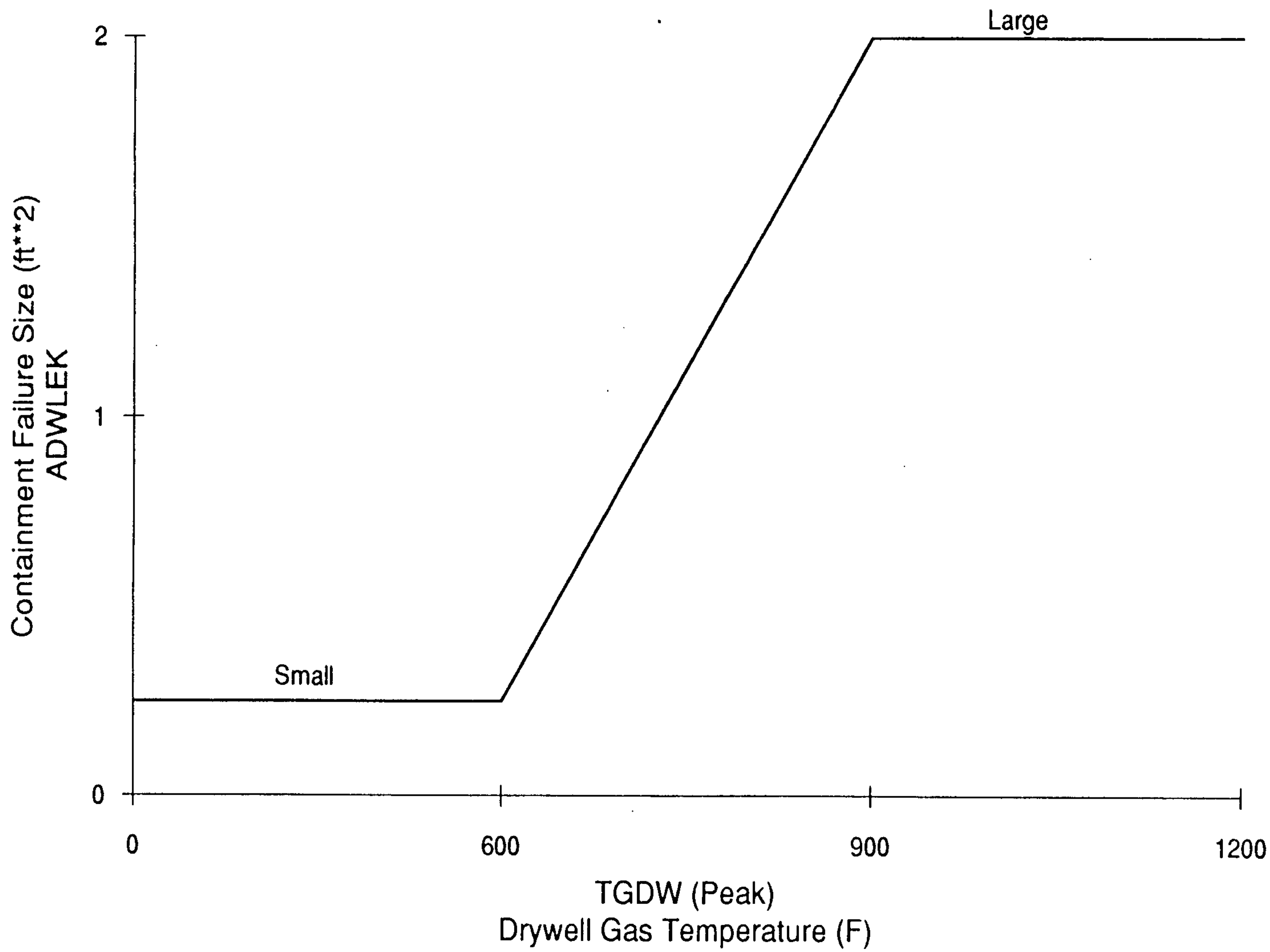
- Decontamination factors for the secondary containment in the analyses consider the possibility of natural circulation and localized hydrogen burns causing loss of secondary containment building effectiveness.

- The DAEC IPE includes a sufficient number of release categories to adequately account for the potential individual source terms, taking into account severity and timing.

MAAP cases were performed with containment failure size chosen by the following:

- Large (~ 2 $ft.^2$)
- Small (~ 27 in^2)
- Variable

The MAAP model implements the variable containment failure size as a function of temperature using the simple relationship shown in Figure 4.2-1.



Note: Imposition of "Large" or "Small" failures was usually achieved by bypassing the above relationship

Figure 4.2-1
Containment Failure Size vs. Temperautre

The determination of the size of the failure is included probabilistically in the containment event tree analysis.

4.2.3.2 Conservatisms

This subsection provides a list of known conservative assumptions that have been used in the Level 2 analysis:

- The containment failure curve is more limiting than calculated by CB&I on a plant specific basis for DAEC or that developed in NUREG-1150 for Peach Bottom. This could result in slightly shorter times to containment failure than if the CB&I plant specific DAEC curve is used. In addition, there may be sequences that would be considered as resulting in "no containment failure" if the CB&I curve were used to assign a containment ultimate capability in this analysis.

- Dynamic containment failures are postulated at a calculated bulk temperature of 260°F in the pool for ATWS (included explicitly in the analysis to determine failure locations).

- Little credit is allowed for the reactor building DF. The reactor building DF is limited to no more than a factor of 10 and is determined by MAAP. It is also applied probabilistically such that for drywell head failures no credit is given to the reactor building.

- No credit is allowed for lower release due to small containment failures.

- Drywell venting as part of the containment flood process is evaluated to include the coupled effects of RPV venting to the condenser and the drywell vent. As noted by the MAAP sensitivity cases in Section 4.9, the radionuclide release for each individual case is a medium (M) as long as the condenser provides reasonable retention of radionuclides. For situations in which the condenser is ineffective (noted by RB failure cases), the release is assumed to be high (H).

  Because the deterministic containment flood modeling has been developed as a first of kind model, conservative models have been used resulting in an overall overestimate of the radionuclide release.

- The radionuclide release for a given sequence may have releases which occur over a long period of time. For the bin scheme used in the DAEC IPE, the releases of sequences are grouped such that the earliest time of release (even if that is only noble gas) is used to set the <u>time</u> of release (e.g., early or late). This conservative bin scheme may result in some overestimation of the releases associated with bins such as High/Early which have been referred to as a "large" release.

- For "dry" sequences, the core melt progression is considered to fail the drywell shell at approximately the time of vessel breach due to direct debris contact. The following two aspects of these sequences are considered conservative:

  - The delay before shell failure is assumed to be very short (7 minutes) The delay could be significantly longer resulting in more scrubbing of releases and less energy to the Reactor Building.

- High temperatures in the drywell, i.e., in excess of 900°F can occur. The drywell head is then taken to be a failure pathway at these high temperatures despite the fact that 0 psid exists under such a postulated scenario.

This latter conservatism was included to avoid the situation that shell failures would be <u>beneficial</u> in the containment performance assessment because they would preclude high releases.

4.2.3.3 Non-conservatisms

This subsection provides a list of potential non-conservative assumptions that could influence the Level 2 results.

- Residual debris remaining in the RPV could result in high DW temperatures. MAAP cases indicate that even when water injection is available to the drywell (via LPCI injection) that high drywell temperatures are possible. This is considered to be an analysis anomaly because the flow to the vessel would provide vessel cooling (currently not accounted for in the MAAP runs) and therefore reduce temperatures in the drywell. Additionally, the use of core spray or drywell spray would provide the desired drywell cooling.

- A drywell equipment mass of 2.7 million lbs. appears to be an overestimation for DAEC, resulting in additional heat sinks and longer times to high drywell temperature (i.e., which affects both containment failure and revaporization source term contributions). Sensitivity cases have been performed with a reduced drywell equipment mass (1 million lbs.) and the results of that evaluation are factored into the radionuclide release results.

4.3 BINS AND PLANT DAMAGE STATES

The interface between the Level 1 and Level 2 analyses is important to ensure that the information and data from the Level 1 analysis is properly transferred and interpreted in the CET.

In some PRA analyses (e.g., WASH-1400, NUREG-1150), the coupling of the front-end analysis to the back-end is through the binning of the multitude of front-end sequences into a group of plant damage states with similar back-end characteristics. For such analyses, it is important that the bins be justified on the basis of such factors as timing of important events or operability of key features.

The DAEC assessment involves the direct coupling of each sequence from the Level 1 to the CET evaluation. The Level 1 end state bins have a valuable use as a summary point.

Specifically, the DAEC IPE directly links the front-end to back-end portions of severe accident sequences through directly linked event trees. These trees ensure that the support state conditions are properly accounted for throughout the front-end and back-end trees. These trees and their direct linking include preventive or mitigative features as well as timing considerations. Three different containment event trees, each properly handle containment failure before core damage cases as well as vice-versa, and containment bypass sequences.

4.3.1 Input to CET: Interface Between Accident Sequence Classes (Level 1 End States and Containment Challenges)

Binning may have two purposes:

- First as a necessity in order to perform the probabilistic evaluation (e.g., WASH-1400).

- Second as a method of allowing discrete evaluations using deterministic codes and for display of Level 1 results.

The DAEC evaluation has used the bin scheme for the second purpose, that is:

- The Level 1 results have been chosen to be usefully displayed in functional groupings having similar challenges to containment and operator response

- The deterministic calculations used to calculate pressure and temperature responses are also conveniently characterized as related to these types of challenges.

Each of these aspects are discussed in this section.

4.3.2 Level 1 PRA End States Classification Scheme

An event sequence classification into five accident sequence functional classes can be performed using the functional events as a basis for selection of end states. The description of functional classes is presented here to introduce the terminology to be used in characterizing the basic types of challenges to containment. The reactor pressure vessel condition and containment condition for each of these classes at the time of initial core damage is noted below:

Core Damage Functional Class	RPV Condition	Containment Condition
I	Loss of effective coolant inventory (includes high and low pressure inventory losses)	Intact
II	Loss of effective containment pressure control, e.g., heat removal	Breached or Intact
III	LOCA with loss of effective coolant inventory makeup	Intact
IV	Failure of effective reactivity control	Breached or Intact
V	LOCA outside containment	Breached (bypassed)

In assessing the ability of the containment and other plant systems to prevent or mitigate radionuclide release, it is desirable to further subdivide these general functional categories. In the second level binning process, the similar accident sequences grouped within each accident functional class are further discriminated into subclasses such that the potential for system recovery can be modeled. These subclasses define a set of functional characteristics for system operation which are important to accident progression, containment failure and source term definition. Each subclass contains front-end sequences with sufficient similarity of system functional characteristics that the containment accident progression for all sequences in the group can be considered to behave similarly in the period after core damage has begun. Each subclass defines a unique set of conditions regarding the state of the plant and containment systems, the physical state of the core, the primary coolant systems, and the containment boundary at the time of core damage, as well as vessel failure.

The important functional characteristics for each subclass are determined by defining the critical parameters or system functions which impact key results. The sequence characteristics which are important are defined by the requirements of the containment accident progression analysis. These include the type of accident initiator, the operability of important systems, and the value of important state variables (e.g., reactor pressure) which are defined by system operation. The interdependencies that exist between plant system operation and the core melt and radionuclide release phenomena are represented in the release frequencies through the binning process involving these subclasses, as shown in past PRAs and PRA reviews. The binning process, which consolidates information from the systems' evaluation of accident sequences leading to core damage in preparation for transfer to the containment-source term evaluation, involves the identification of 13 classes and subclasses of accident sequence types. Table 4.3-1 provides a description of these subclasses that are used to summarize the Level 1 PRA results.

Published BWR PRAs have identified that there may be a spectrum of potential contributors to core melt or containment challenge that can arise for a variety of reasons. In addition, sufficient analysis has been done to indicate that the frequencies of these sequences are highly uncertain; and therefore, the degree of importance on an absolute scale and relative to each other, depends upon the plant specific features, assumptions, training, equipment response, and other items that have limited modeling sophistication.

This uncertainty means that the analyst can neither dismiss portions of the spectrum from consideration nor emphasize a portion of the spectrum to the exclusion of other sequence types. This is particularly true when trying to assess the benefits and competing risks associated with a modification of a plant feature.

Table 4.3-1

SUMMARY OF THE CORE DAMAGE ACCIDENT SEQUENCE SUBCLASSES

Accident Class Designator	Subclass	Definition	WASH-1400 Designator Example
Class I	A	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high.	TQUX
	B	Accident sequences involving a station blackout and loss of coolant inventory makeup.	T_EQUV
	C	Accident sequences involving a loss of coolant inventory induced by an ATWS sequence with containment intact.	T_TC_MQU
	D	Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced to 200 psi.; i.e., accident sequences initiated by common mode failures disabling multiple systems (ECCS) leading to loss of coolant inventory makeup.	TQUV
	E	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high and DC power is unavailable.	---
Class II	A	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage induced post containment failure	TW
	L	Accident sequences involving a loss of containment heat removal with the RPV breached but no initial core damage; core damage after containment failure.	AW
	T	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage induced post high containment pressure	N/A
	V	Class IIA or IIL except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	TW

Accident Class Designator	Subclass	Definition	WASH-1400 Designator Example
Class III (LOCA)	A	Accident sequences leading to core damage conditions initiated by vessel rupture where the containment integrity is not breached in the initial time phase of the accident.	R
	B	Accident sequences initiated or resulting in small or medium LOCAs for which the reactor cannot be depressurized prior to core damage occurring.	S_1QUX
	C	Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is a low pressure and no effective injection is available.	AV
	D	Accident sequences which are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate, challenging the containment integrity with subsequent failure of makeup systems.	AD
Class IV (ATWS)	A	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post containment failure.	$T_TC_MC_2$
	L	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially breached (e.g., LOCA or SORV); core damage induced post containment failure.	N/A
	T	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post high containment pressure.	N/A
	V	Class IV A or L except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	N/A
Class V	--	Unisolated LOCA outside containment	N/A

This end state characterization of the Level 1 PRA in terms of accident subclasses is usually sufficient to characterize the CET entry states for most purposes. However, when additional refinement is required in the CET quantification, it may be useful to further discriminate among the contributors to the core damage accident classes. This discrimination can be performed through the use of the individual accident sequence characteristics.

For DAEC, functional based plant damage states are used to summarize Level 1 results and to ensure that the Level 2 CETs are sufficient to allow each functional sequence to be addressed.

4.3.3 Summary of Specific Aspects of the Level 1 - Level 2 Interface

This subsection provides a brief summary of particular aspects of the interface that are useful to highlight.

- Equipment failures in Level 1: Equipment failures that have been assessed in Level 1 are carried by the computer into the Level 2 analysis. Therefore, failed equipment cannot be used in the Level 2 assessment, unless an explicit evaluation has been performed as part of the Level 2 to support repair or recovery. This would include consideration of adverse environments where appropriate. This includes support systems, accident prevention systems, and mitigation systems.

- Human errors: There is a check performed on all sequences to ensure that Level 1 sequences that result from human errors have only those recoveries that can be justified as consistent with operating staff recoveries given human failures in the Level 1 analysis.

- RPV status: The RPV pressure condition is explicitly transferred from the Level 1 analysis to the CET.

- Containment status: The containment status is explicitly transferred from the Level 1 analysis to the CET. This includes recognition of whether the containment has previously failed, is intact, or is at elevated pressure conditions.

- Containment isolation: All support system dependencies are transferred as part of the individual Level 1 sequences such that the containment isolation evaluation is performed on a sequence by sequence basis.

- Differences in accident sequence timing are also transferred with the Level 1 sequences. These timings affect such sequences as:

  - station blackout
  - loss of containment heat removal
  - ATWS
  - vapor suppression failure

  This allows the timing to be properly assessed in the Level 2 CET.

- Thermal hydraulic deterministic assessments:

The use of deterministic codes in the characterization of accident sequences has been performed in a manner similar to the functional sequence binning classification scheme identified above. These functional sequences then have various additional failures applied to determine containment response for various postulated scenarios through the CET. Variations in timing and assumptions regarding subtle sequence variations have been explicitly calculated to ensure that the sequence representations using the thermal-hydraulic code is representative.

- Dual Usage: Because the Level 1 and Level 2 models are directly coupled on a sequence basis the accountability of common water sources or common power sources falls out of the combined sequence analysis when it is run from initiating event to release point.

- Mission Times: The mission times for the entire sequence from initiating event to release point are considered.

- Timing of Recovery: Equipment or power recovery is accounted for at various phases in the Level 1 and 2 analyses. Each sequence includes a consistent recovery model to ensure no double counting.

4.4 CONTAINMENT FAILURE CHARACTERIZATION

A knowledge of the pressure and temperature capability of the containment, as well as the probable location and size of a containment failure, is fundamental in determining the timing and magnitude of a potential radionuclide release under postulated severe accident conditions. Consequently, several studies of the Mark I containment design have been performed to advance the state-of-knowledge in terms of its ability to withstand severe accident conditions.

The Level 2 analysis is strongly influenced by the containment failure modes and their timing.

This subsection includes an assessment of:

- Primary containment failure modes (Subsection 4.4.2)
    - pressure and temperature dependent
    - dynamic failure modes
    - phenomenological induced failures
- Summary of severe accident challenges to containment (Subsection 4.4.3)
- Primary containment ultimate capability (Subsection 4.4.4)
    - low temperatures
    - moderate temperatures
    - high temperatures
    - dynamic load induced

- Summary of DAEC containment evaluation (Subsection 4.4.5)

- Secondary containment failure modes (Subsection 4.4.6).

4.4.1 DAEC Mark I Containment

Although the frequency of core damage events is very low, severe accidents may present a threat to the integrity of the containment. Primary containment (see Figure 4.4-1) is one of the boundaries preventing the release of radionuclide fission products to the environment. Therefore, to assess accident management actions that could be implemented as part of contingency planning for severe accidents, containment response under severe accident conditions must be considered.

The DAEC containment is of the Mark I pressure-suppression type. The steel "Light-bulb-shaped" drywell is a steel spherical shell intersected by a circular cylinder (see Figure 4.4-1). The top of the cylinder is closed by a head bolted to the drywell. The pressure-suppression chamber, or the wetwell, is a toroidal steel vessel located below and encircling the drywell. The wetwell and drywell are interconnected by eight circular vent pipes. The containment is enclosed by the reactor building, which also contains the refueling area, fuel storage facilities, and other auxiliary systems. In the event of a primary system pipe failure within the drywell, a mixture of drywell atmosphere and steam would be forced through the vents into the suppression pool resulting in steam condensation and pressure reduction. The design internal pressure of the containment is 56 psig. The minimum and gross free volume of the wetwell are 94,270 $ft.^3$ and 162,400 $ft.^3$, respectively. The minimum and gross free volume of the drywell are 109,400 $ft.^3$ and 144,000 $ft.^3$, respectively.

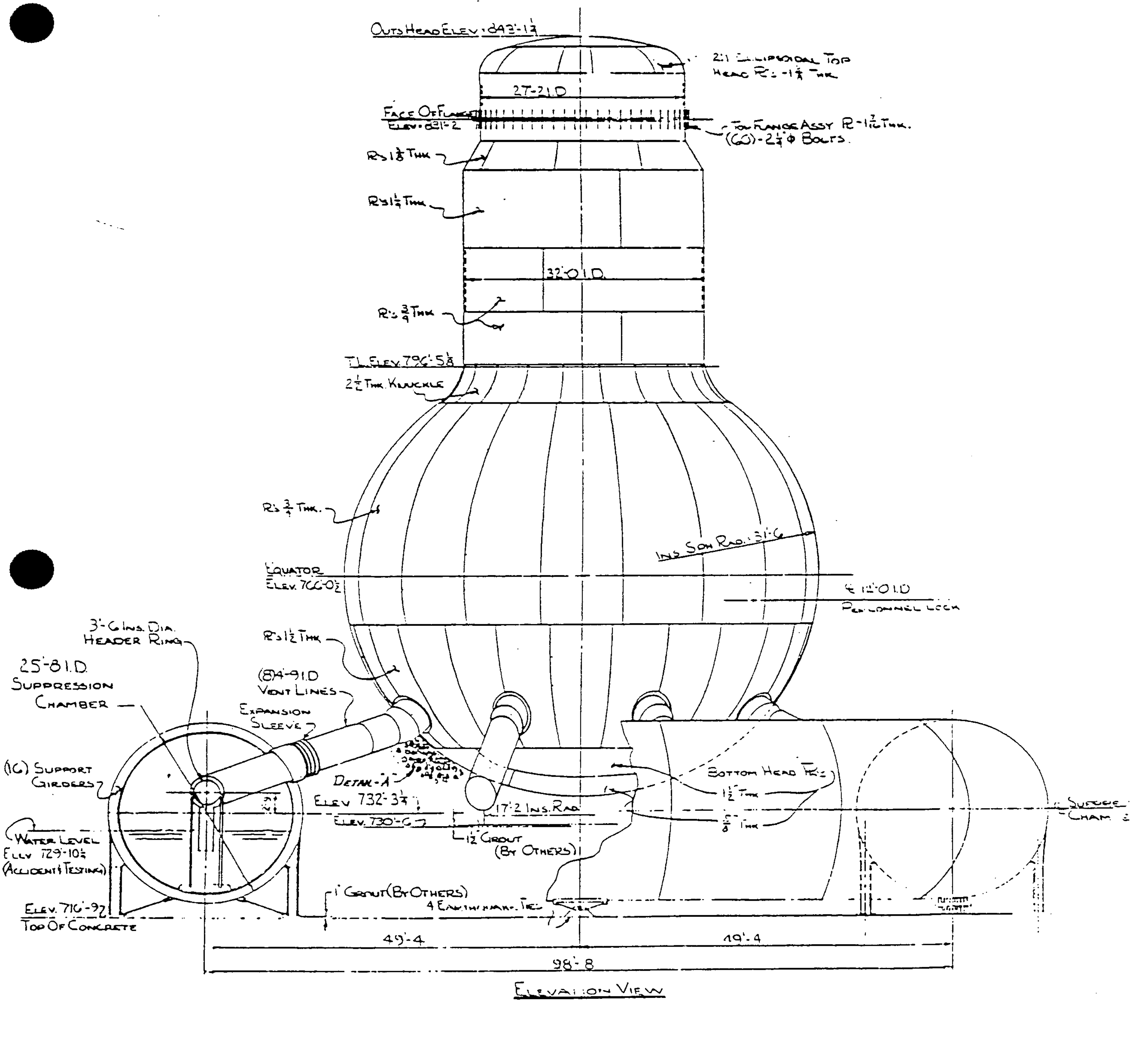


Figure 4.4-1 DAEC Containment Structure

Section 4.1 briefly described the key features of the containment. This section discussed postulated failure modes.

4.4.2 Postulated Containment Challenges/Failure Modes

To determine the effectiveness of the requirements in assuring adequate containment performance, a systematic review of the containment challenges associated with a spectrum of severe accident types has been assembled. Radionuclide releases are associated with a containment failure (location and size) and accident sequence.

Plant specific MAAP evaluations can be used to determine the release magnitude and timing for the various release categories corresponding to some of the postulated containment failure modes (e.g., slow containment overpressure).

The containment capability is combined with the deterministic MAAP calculations described in Sections 4.2 and 4.7 to determine the timing and location of many of the containment failure modes. Figure 4.4-2 is a simplified flow chart showing the combination of MAAP deterministic results for one sequence overlaid on the containment capability curve determined by the containment structural analysis. This comparison of calculated deterministic results with the capability curve results in identifying the time it takes conditions to degenerate to the point that jeopardizes the containment integrity.

However, as will be seen in this section, not all postulated containment failure modes are easily calculated with existing deterministic codes. For such cases, separate effects analyses or engineering judgement is used to provide the magnitude of the threat and timing. MAAP can still be used to bracket the general

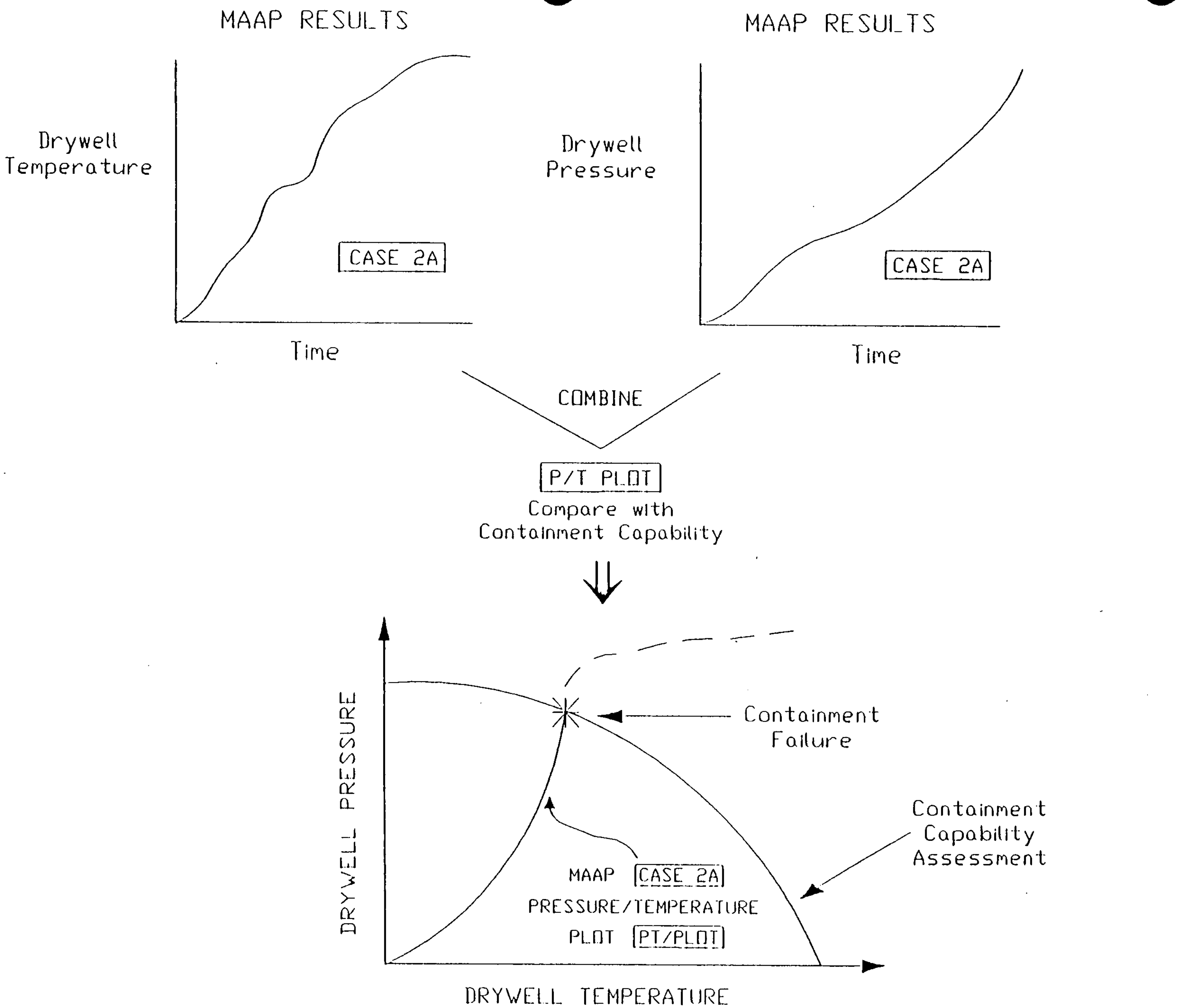


Figure 4.4-2 Simplified Flow Chart Showing the Relationship of MAAP Deterministic Results Compared with the Ultimate Containment Capability

time frame by characterizing when the necessary and sufficient conditions are present (e.g., when steam explosions might be possible).

One of the basic reasons for focusing on containment failure mode and timing is that it can immediately make obvious the type of response that can either mitigate or reduce containment failure probability. Accident management actions will be dependent primarily on the containment failure assessment and supplemented by the radionuclide release magnitude.

Table 4.4-1 presents the various functions associated with plant response to accident and transient conditions which either preclude expected challenges, or allow the containment to accommodate challenges.

Associated with combinations of success or failure of each of these functions during transient or accident conditions are potential challenges to the integrity of the fuel, reactor coolant system and containment. These functions are assessed probabilistically and deterministically in the containment event tree analysis.

Table 4.4-2 identifies either postulated containment challenges or the corresponding failure modes that have previously been identified in severe accident or Design Basis Accident (DBA) analyses. These challenges/failure modes span a range from those historically considered by regulations to those beyond traditional design bases, including severe accident conditions. To ensure that a comprehensive list of challenges is investigated, the important containment functions listed in Table 4.4-1 were reviewed and an assessment made of their impact on containment integrity. In addition, challenges/failure modes were selected to encompass the following:

- initiating events which by definition result in bypass of containment,

- random system or equipment failures which could lead to breach of the containment boundary independent of any severe accident challenges, and

- potential dependent failures that could be caused by phenomena which challenge the structural integrity of containment as a result of the accident. Challenges to containment integrity as identified in the General Design Criteria of the Standard Review Plan are incorporated into the study. The list also covers those postulated accident initiators or phenomena that have been identified as important in past industry and NRC studies, such as the Industry Degraded Core Rulemaking (IDCOR) program, NUREG-1150, and various BWR probabilistic risk studies. Also cited are the failure modes provided in the PRA Procedures Guide regarding potential containment failure modes. This last item was explicitly suggested in the IPE Generic Letter 88-20.

Table 4.4-1 IMPORTANT FUNCTIONS FOR PREVENTION AND ACCOMMODATION OF CONTAINMENT CHALLENGES
· Reactivity Control
· Reactor Pressure Control
· Fuel/Debris Coolant Inventory Control
· Containment Pressure/Temperature Control
· Combustible Gas Control
· Containment Isolation
· Vapor Suppression
· Containment Structural Capability for External Loading

Table 4.4-2

POSTULATED CONTAINMENT CHALLENGES/FAILURE MODES

		Challenges/Failure Modes That May Precede a Severe Accident
1.	*	Containment Isolation Failure
2.	*	Interfacing System Loss of Coolant Accident (LOCA)
3.	*	Blowdown Forces Due to Rupture or Containment Overpressure Due to Catastrophic Reactor Pressure Vessel (RPV) Failure
4.	*	Pipe Whip/Steam Jet Impingement
5.	*	Containment Overpressure Due to Anticipated Transient Without Scram (ATWS) or Loss of Containment Heat Removal
6.		Containment Overpressure from Pool Bypass (BWR)
7.		External Pressure Loading Due to Partial Vacuum Conditions
8.	*	Missiles from Internal (Plant) Sources
9.	*	Tornado and Tornado Missiles[1]
10.		Seismic Induced Failure[1]
11.		Containment Venting
		Challenges/Failure Modes Potentially Resulting from a Severe Accident
12.		High Pressure Core Melt Ejection
13.	*	Hydrogen Related Issues (Deflagration/Detonation)
14.	*	In-vessel Steam Explosion
15.	*	Ex-vessel Steam Explosion
16.	*	Reactor Pressure Vessel Support Failure and Containment Basemat Penetration
17.		Containment Sump Failure from Core Debris
18.	*	Containment Shell Failure from Core Debris
19.		Containment Overtemperature Due to Debris
20.	*	Containment Overpressurization Due to Core Debris Decay Heat Steam Generation
21.	*	Noncondensible Gas Generation
22.		Reactivity Insertion During Core Melt Progression
23.		N_2 Pressure
24.		Direct Impingement

* Identified from PRA Procedures Guide
[1] These failure modes are treated in the IPEEE and are not addressed in this report.

Table 4.4-3 summarizes the disposition of these failure modes in the DAEC Level 1 and 2 assessments after extensive evaluations of the DAEC containment, the severe accident spectrum, and current published information.

The containment failure modes have specific characteristics that allow them to be associated with the critical parameters governing radionuclide release determination. Three of these critical parameters are:

- Time of containment failure
- Size of containment failure
- Location of containment failure.

Table 4.4-4 summarizes the general relationships considered in the Level 2 analysis to represent the failure modes from Table 4.4-3. Note that multiple failure sizes and locations are possible for many of these failure modes. These relationships are used to construct the containment event trees and the functional fault tree for each node.

In addition to identified containment failure modes, there are also a number of related phenomenological issues. The IDCOR regulatory interaction program was devoted to the definition and resolution of open technical issues related to the assessment of severe accidents. These are discussed in Section 4.2.2. Consideration of each of these issues has been included in the development of the containment capability curves and the CET nodal evaluations. Specific sensitivity studies have been performed on selected issues (see Section 4.9).

Table 4.4-3

SUMMARY OF TREATMENT OF CHALLENGES IN THE DAEC CONTAINMENT SAFETY STUDY

Postulated Containment Challenges	Disposition
Containment Initial Conditions	
1. Containment Isolation Failure	Included in CET Treatment assumes inerting has substantial benefit in assuring isolation
Sequence Dependent Failure Modes	
2. Interfacing System LOCA	Included in Level 1 evaluation
3. RPV Rupture Overpressure	Included in Level 1 evaluation
4. Pipe Whip/Steam Jet Impingement	Dismissed based on low probability
5. ATWS - Overpressure TW - Overpressure	Included in Level 1 evaluation
6. Vapor Suppression Failure (Suppression Pool Bypass)	Included in Level 1 evaluation
7. Containment Implosion Due to Drywell Sprays	Low Probability Due to Mark 1 Structural Capability and EPG procedural guidance and external vacuum breakers
8. Missiles from Internal Sources	To be evaluated in IPEEE
9. Tornado and Tornado Missiles	To be evaluated in IPEEE
10. Seismic Induced	To be evaluated in IPEEE
11. Containment Venting and Combustible Gas Vents	Included in Level 1 and 2 Event Trees

Table 4.4-3

SUMMARY OF TREATMENT OF CHALLENGES IN THE DAEC CONTAINMENT SAFETY STUDY

Postulated Containment Challenges	Disposition
<u>Phenomenological Failure Modes</u>	<u>Addressed In Level 2 CET</u>
12. Direct Containment Heating	Included (low probability)
13. Hydrogen Effects:	
- Quantity of H_2 Produced In-Vessel	Range of values examined
- $H_2 + O_2$ Deflagration Effects	Conditional probability of deflagration included
- Introduction of O_2	None considered possible except operation deinerted
- RPV Blowdown Failure + H_2 Causes containment failure	Calculated not to cause containment failure at DAEC
14. In-vessel Steam Explosions	Included in Level 2 analyses
15. Ex-vessel Steam Explosions	Included in Level 2 analyses
16. Structural Failure Due to RPV Collapse and Tear Out of Penetration	Included in high temperature induced pedestal/skirt failures
17. Containment Sump Line Failure	Considered in Level 2 analysis (does not apply to DAEC sump line configuration)
18. Direct Contact of Molten Material W/Steel Shell	Included in Level 2 analyses
19. DW Head Seal Performance at Elevated Temperature (High Temp Failure)	Included as a potential leak path
20. Containment overpressure due to decay heat	Included in Level 2 analyses
21. Non-Condensable Gas Generation (Core Concrete Attack)	Included (range of modeling assumptions examined)
22. Reactivity Insertion During Core Melt Progression	Included in Level 2 quantification

4.4.3 Summary of Severe Accident Challenges to Containment

Subsection 4.4.2 identified in tabular form the individual containment failure modes identified in the literature.

For the purposes of this analysis, the entire spectrum of accident conditions that could challenge containment integrity can be categorized into four regimes. The following four regimes are used to determine those areas where containment ultimate capability is assessed:

1) Pressure Induced Containment Challenge: Containment pressures may increase from normal operating pressure along a saturation curve to very high pressures (i.e., beyond 100 psi), during accidents involving:

   - Insufficient long term decay heat removal; and
   - inadequate reactivity control and consequential inadequate containment heat removal.

2) Temperature Induced Containment Challenge: Containment temperatures can rise without substantial pressure increases if containment pressure control measures (e.g., venting) are available, but debris temperature control is inadequate. In such cases, containment temperature induced failure at less than design pressure may occur during accidents involving core melt progression because temperature can increase to greater than 900°F.

Table 4.4-4

SUMMARY OF TIMING, SIZE, AND LOCATION FOR POSTULATED CONTAINMENT FAILURE MODES

Postulated Containment Challenge	Timing	Size	Location[1]
Sequence Dependent Failure Modes			
· ATWS Without Mitigation	Early	Large	DW, WW
· RPV Rupture Large Enough to Cause Containment Failure	Early	Large	DW
· TW-Overpressure	Late	Small, Large	DW, WW
· Vapor Suppression Failure + LOCA	Early	Large	DW, WW
· N_2 Overpressurization	Intermediate	Small, Large	DW, WW[2]
· Combustible Gas Vent	Early[3]	Large	WW
· Containment Implosion Due to DW Spray Initiation	Early	Large	DW
· Containment Overpressure Vent	Late	Small	WW
Phenomenological Failure Modes			
· Non-Condensable Gas Generation	Intermediate[3]	Small, Large	DW, WW
· Direct Containment Heating	Early[3]	Large	DW
· DW Temperature Rise	Intermediate[3]	Small, Large	DW
· Steam Explosions	Early[3]	Large	DW
· Hydrogen Explosions	Early[3]	Large	WW, DW[2]
· Structural Failure due to Penetration Tearout	Intermediate[3]	Large	DW
· Vessel Thrust Forces	Early[3]	Large	DW
Containment Initial Conditions			
· Containment Isolation Failure	Early	Large	DW
· Containment Leakage	Early/Late[3]	Small	WW

Notes To Table 4.4-4

(1) WW = Wetwell, DW = Drywell

(2) Always treated as a drywell failure in the simplified CET evaluation.

(3) These times are relative to RPV breach, which of course may be delayed significantly from accident initiation depending on the accident sequence.

3) Combined Pressure and Temperature Induced Containment Challenge: Containment pressures and temperatures can both rise during a severe accident due to molten debris effects following RPV failure and subsequent core concrete interaction. For instance:

- Drywell temperatures can rise from approximately 300°F at core melt initiation to above 1000°F in time frames on the order of 10 hours.

- Pressure can rise due to non-condensible gas generation and RPV blowdown in the range of 40 psig to 100 psig over this same time frame.

4) Dynamic Loads: In addition to these "steady state" challenges, failure modes associated with dynamic loading resulting from high steam flow to a saturated pool or from energetic phenomena (e.g., steam explosions) are also postulated.

It is clear from analyses of the severe accident challenges to containment, that the containment response and capability both vary substantially over a spectrum of possible challenges in terms of temperature and pressure. Therefore, the definition of adequate containment performance proposed explicitly considers these regimes. The following subsection addresses the containment ultimate capability for the four regimes.

4.4.4 Primary Containment Ultimate Capability

The primary containment ultimate structural integrity is important in severe accident analysis due to its key role as a fission product barrier. As noted above there is a broad spectrum of postulated severe accidents that may challenge containment.

The DAEC Mark I containment has been analyzed by Chicago Bridge and Iron (CB&I) to predict its ability to withstand severe accident conditions, i.e., pressures and temperatures imposed on containment during core melt progression accidents.

Features of the DAEC Mark I containment that were investigated include the steel containment structure (drywell, drywell head, and torus), containment hatches, hatch seals, penetrations and isolation valves.

This subsection provides a summary of the following issues related to containment performance:

- Capability at low temperature (below 500°F)
- Capability at intermediate temperature (between 500°F and 800°F)
- Capability at high temperatures (above 900°F)
- Capability for high suppression pool temperatures and high SRV discharge flow rates.

Because the containment failure size and location are influential in quantifying the radionuclides released to the reactor building and subsequently to the environment, the size and location are important features to identify in the analysis and include in the probabilistic CET evaluation. The failure size can be divided into three size regimes: negligible, small and large.

4.4.4.1 Low Temperature Containment Performance (< 500°F)

Chicago Bridge and Iron (CB&I) has performed a plant specific scoping analysis of the DAEC containment. This analysis forms the bases for the ultimate containment capability assessment. In addition, the DAEC IPE includes an assessment of containment strength and containment loads from other plants and extracts those results most applicable to DAEC. This includes an extrapolation of the detailed analysis of ultimate pressure capability for Peach Bottom performed by CB&I. CB&I Peach Bottom analysis is considered to be the most comprehensive Mark I containment evaluation performed to date. The CB&I scoping study comparison of DAEC to the Peach Bottom containment concluded that the DAEC is generally as strong or stronger than the Peach Bottom containment. At the reported containment failure pressure of 159 psig for Peach Bottom (163 psig for DAEC) where the membrane strain exceeded 1% for the wetwell air space, the calculated maximum wetwell displacement was 2 inches, but structural restraints such as piping and support attachments were not included in the model. Therefore, a mean value of 140 psig failure pressure is used in the DAEC IPE at low temperature to provide a best estimate representation of the containment ultimate pressure capability, considering the effects of restraints.

The containment studies that have been reviewed represent a significant body of knowledge on the ultimate capability of the Mark I containment under high pressure conditions.

Primarily, these studies have focused on low-temperature conditions (i.e., a temperature range from 70°F to 340°F). Much beyond this range, other issues such as material property degradation and performance of seals become a concern.

It has been concluded that for temperatures below 340°F a reasonable assessment of the mean containment pressure capability is 140 psig. As pressure exceeds this limit, material yielding in the wetwell begins, and displacement increases rapidly with pressure. Large displacements are anticipated to result in tears in the wetwell shell or at piping and support attachments. In addition, the closure head may begin to leak substantially. Therefore, the pressure capability of the DAEC containment near the design temperature limit of 340°F is expected to be in the range of 125 to 163 psig, with the predicted dominant failure mode occurring in the wetwell air space.

The distribution of failure locations and leak sizes are judged by CB&I to be similar to those estimated by CB&I for Peach Bottom over the low temperature range.

The general conclusions pertaining to low-temperature (i.e., less than 340°F) containment performance that can be drawn from the studies evaluated are:

- The ultimate pressure capability of the Mark I containment is two to three times greater than design.

- The concrete biological shield encasing the drywell, by limiting the expansion of the shell, extends the pressure capacity of the drywell shell by 30 to 50 psig.

- The most probable containment failure modes due to pressurization are: (1) a rupture in the wetwell airspace; or (2) rupture in the wetwell water space; or (3) a break in the drywell head.

- The key areas of uncertainty in containment analyses to date are the performance of the drywell head closure, vent line expansions bellows, local restraints, penetrations, and hatches; and associated temperature and pressure effects on failure pressure and location.

4.4.4.2 Intermediate Temperature Containment Performance (500 to 800°F)

In an intermediate temperature range (500 to 800°F), the high-pressure performance of the containment is expected to degrade due to reductions in material strength and seal properties. The large DAEC containment penetrations have silicone rubber seals. For the purpose of discussion, the intermediate temperature range can be characterized by a temperature of 700°F, which is the projected temperature at which silicone seals begin to fail in non-steam environments. (Outer seals are considered to "see" minimal steam environments initially). At this temperature the results of the reports surveyed would indicate that the drywell closure head seal would be the principal containment failure location.

The Peach Bottom containment study predicted that a gap in the drywell head closure would occur at pressures significantly below the reported "low temperature" failure pressure. The study concluded that the potential for leakage was controlled by the resiliency of the silicone head gasket. Silicone rubber gaskets are rated for long-term temperatures up to 450°F and tested for 4 months at 500°F. However, seal performance can be degraded by accident conditions. The outer seal is expected to be intact for many hours after the inner seal reaches 500°F in a wet environment and much longer in a dry environment).

Under "dry" severe accident conditions where water is not available to cool the slumped core debris, containment heatup is expected. For this scenario, the silicone rubber seals are expected to maintain resiliency up to 700°F, beyond 700°F the seal is assumed to fail.

For "wet" or steam severe accident conditions where debris cooling is available, the containment environment is expected to be at saturated conditions at maximum containment pressure (~150 psig or ~500°F to 600°F). Testing of silicone rubber seals showed reduced performance in steam environments. For these conditions the seals are assumed to fail between 500°F and 700°F. The outer seal will be somewhat protected from the environment even if the inner seal fails.

The CB&I study for DAEC reported similar conclusions. At 800°F, it is expected that the flange seal material will significantly deteriorate. The curing temperature for this material is approximately 500°F. Beyond this temperature, the material will harden and crack and will not provide any significant amount of elasticity. This provides for a higher probability of leakage through the bolted closure. At this temperature, the yield strength of the bolts also drop to the point that yield could occur over the range of probable failure pressures. This will increase the leakage area; and is likely to provide adequate area to qualify for rupture.

For this case, the pressure strain required to close the gap between the steel containment and concrete building is less. (Some of the gap will be closed by thermal expansion.) This will reduce the probability of rupture in those areas of the containment vessel that are backed up by concrete. At these temperatures this material is quite ductile, reducing the chances of fracture and resulting rupture failure. At this temperature, the drywell has grown axially to the point that some of the penetration necks have made contact with the concrete wall. But there should be adequate strain to start tearing the shell until higher temperatures are reached.

NUREG-1037 calculations for Peach Bottom indicate leakage through the purge and vent valves is present at approximately 60 - 80 psig and that significant leakage through the drywell head begins in this range and dominates through the higher pressure ranges. The CB&I calculations for DAEC provide subjective probabilities indicating leakage becomes significantly likely at around 135 psig, dominated by drywell flange and bellows leakage.

Based on this information, primary containment failure in the 500°F to 800°F range is estimated, for the DAEC IPE, to occur at around 88 psig (see Figure 4.4-3) and to be dominated by drywell head flange leakage. This is a leakage dominated failure mode, i.e., a small failure of approximately 18 in.^{2} to 19.5 in.^{2} which is considered to be a small leakage failure. Rupture failures are also possible, but are judged to be less likely for most scenarios.

4.4.4.3 High Temperature Containment Performance

Drywell temperatures well in excess of 900°F have been calculated by the MAAP computer code for accidents in which core melt has occurred, the core has slumped to the drywell floor, and core debris cooling is unavailable. Such events are of very low frequency, but without the addition of water to cool the core debris there is little confidence that the containment can withstand such extreme temperatures without significant material degradation. The determination of the containment failure location is important for determining the magnitude of the radionuclide release from the containment and estimating the reactor building effectiveness in reducing the source term.

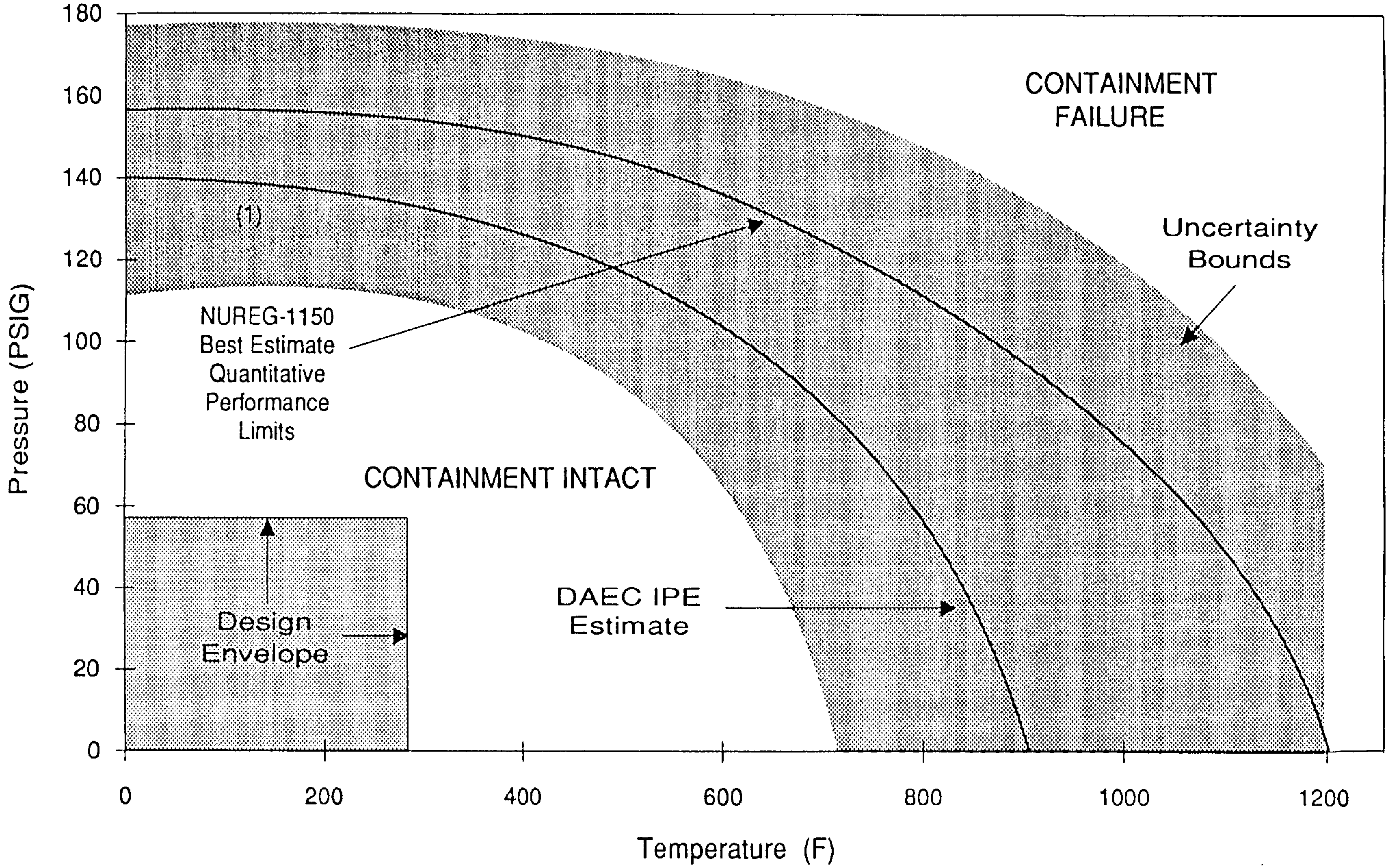


Figure 4.4-3
Primary Containment Performance

NOTES TO FIGURE 4.4-3:

(1) The ultimate pressure capability identified here is not judged to be applicable to ATWS conditions in which high steam flows to the wetwell occurs at elevated pool temperatures. As discussed in Section E.4, the failure criteria is assumed at a calculated bulk pool temperature of 260°F and is related to combinations of hydrodynamic loads and failure to fully condense high flow discharge from the SRVs or downcomers.

The Industry Degraded Core Rulemaking (IDCOR) program recently examined the effect of very high temperatures on drywell shell integrity (IDCOR Task 17.5). As a first-order assessment, this evaluation calculated the pressure at which the drywell shell would contact the concrete biological shield. Although the calculated strain at the time of contact was below the ultimate limit, this criterion did allow an understanding of how less restrained structures, such as the drywell head, may behave. It should be noted that local leakage due to degradation of seals was not addressed in this study.The conclusion reached by IDCOR Task 17.5 is that failure is more likely to occur between 700°F to 900°F due to large upward and radial thermal growth of the containment. Some of the numerous small and large penetrations are expected to bind on the biological shield wall and fail. Radial growth of the containment may also cause the seismic stabilizers to punch through the upper portion of the drywell at these temperatures.

Based on the CB&I study for DAEC, sealing material is expected to completely degenerate at 1200°F. The leakage through bolted flange connections becomes a higher probability.

The general conclusion that may be formulated from these results is that the strength of the containment becomes suspect as temperatures exceed 900°F. In applying these results, the BWROG Mark I containment evaluation program and IDCOR Task 17.5 (using the Peach Bottom Atomic Power Station as the model) assumed the drywell would fail under any appreciable pressure load at a temperature of 900°F.

4.4.4.4 High Pool Temperatures, High Containment Pressures and High SRV Discharge Rates

Based on the limited amount of data to support containment integrity at high SRV discharge rates and at elevated containment temperatures, pressures and water levels, the containment torus is considered to be failed if temperatures exceed 260°F in the suppression pool and substantial power is being produced in the core and discharged to the pool. This is further supported by approximately eight issues each of which identifies a potential area of containment failure when subjected to these conditions.

These eight issues are the following:

- Condensation phenomena
- Temperature profile at the quencher device
- Limitation of calculational models
- Vacuum breaker performance with cycling drywell sprays
- Containment structural capability under hydrodynamic loads
- Cyclic pressure effects
- Elevated torus water levels affecting hydrodynamic loads.

4.4.5 Summary of DAEC Containment Evaluation

Based on the information discussed above, a summary of those components or structures that govern the containment capability is presented below:

- At low temperatures the capability is dominated by structural members such as the torus above the water line and the containment closure head leakage.

- At intermediate temperatures, the containment leak potential is increased due to potential seal degradation, and the containment pressure capability is assessed as lower than at low temperatures.

- At extremely high temperatures, leakage is guaranteed, structural capacities are lower, and interference failures of penetrations with the biological shield wall are found to dominate.

- For cases of high suppression pool temperatures (greater than 260°F) and high SRV discharge rates, the containment capability is limited by hydrodynamic loads.

Table 4.4-5 summarizes the results of this evaluation.

Among the numerous containment performance analyses performed to date, there is a range of predicted dominant failure modes associated with extreme conditions affecting containment integrity. This is considered to be predominantly due to the differences in assumed analysis boundary conditions and failure criteria. However, by reviewing the various containment analyses, it is possible to identify dominant containment failure modes in three pressure-temperature regimes:

- High pressure and low temperature,

- High pressure and high temperature, and

- Low pressure and high temperature.

Table 4.4-5

SUMMARY TABLE OF DAEC CONTAINMENT CAPABILITY AND CONTROLLING FEATURES

Components	Failure Mode	Pressure Capability (psig) 350°F	< 500°F	500°F to 800°F	> 800°F
Structural Members	Rupture	140[9]*	120 - 140	52 - 120	0 - 52
Purge & Vent Valves	Leak	---	---	40 - 60[1]*	0[1]
	Rupture	>> 200	>> 200	---	---
Hatches					
- Personnel Airlock	Leak[2]	90-150*	90-150	90-150	90-150
	Rupture	>> 200	>> 200	>> 200	>> 200
- CRD Removal	Leak/Rupture[4]	---	---	---	---
- Suppression Pool Access	Leak/Rupture[6]	---	---	--	--
- Drywell Equipment Hatch	Negligible Leak[3]	80-140	80-140	80-140	80-140
	Small Leak	140	140	140	140
	Rupture	>> 200	> 200	> 200	see "Structural Member"
Drywell Head	Leak[5]	80-140*	80-140	> 80*	< 80
	Rupture	>> 140[5]	see "Structural Member"	see "Structural Member"	see "Structural Member"
Pipe Penetrations	Shear/Rupture	N/A	N/A	N/A	Rupture* (~900°F)

* Governing Feature

NOTES TO TABLE 4.4-5

(0) The estimate of containment structural capability is based on the CB&I evaluation of DAEC, which calculated the median containment capability to be 163 psig. The CB&I estimate is covered to 140 psig to account for pipe attachments and anchor points outside containment that were not included in the CB&I analysis. This is consistent with the NUTECH analysis of Monticello referred to in Section E.2.

(1) Purge and vent valves are judged to have a high pressure capability except in the presence of high temperatures where the seal material can be considered failed (i.e., > 700°F drywell temperature) for which the leak area is taken at 18 in^2. However, because of the redundant valve design the outer valves are considered not to be subject to these severe conditions until much later in the scenario.

(2) Personnel airlock is judged to have a high pressure capability (> 200 psig) except in the presence of high temperatures where the seal material can be considered ineffective (i.e., > 700°F drywell temperature for dry cases and > 500°F drywell temperature for steam environment). The leak area with no seal present is a maximum of 2 in^2 at 160 psig (NUREG-1037 pg. D-10). This is assumed to apply to DAEC also.

(3) Drywell equipment hatch is judged to have a high pressure capability (> 200 psig) except in the presence of high temperatures, in which case the seal material can be considered ineffective (i.e., > 700°F drywell temperature for dry cases and > 592°F drywell temperatures for steam environment). The leak area with no seal present is a maximum of 3 in^2 at 140 psig (NUREG-1037 pg. D-40)

(4) CRD removal hatch does not leak appreciably at any accident pressures or temperatures considered in this analysis (NUREG-1037 pg. D-42).

(5) Drywell head leakage produces negligible leakage below 82 psig for Peach Bottom (75 for DAEC) with no gasket or seal present because it was calculated that a minimum of 82 psig for Peach Bottom (75 for DAEC) internal pressure was required to separate the drywell head flange to initiate leakage. However, at elevated temperatures the drywell head seals (double O-ring seals of silicone rubber) are expected to degrade and become ineffective (i.e., > 700°F drywell temperature for dry cases and > 592°F drywell temperatures for steam environments). The leak area with no seal present is a maximum of 68 in.^{2} at 140 psig (NUREG-1037 pg. D-35) for Peach Bottom. The leak area for DAEC could be substantially less because of the net higher flange separation pressure.

(6) The suppression pool access is a normally bolted cover plate flange seal. The access hatch is judged not to breach at pressures considered in this evaluation. Additionally, the seal is determined not to be affected by the relatively low temperatures in the wetwell air space.

The purpose of assimilating the information available concerning the performance of Mark I containments is to develop an integrated containment performance profile describing the pressure and temperature conditions inside the containment that can cause a larger than negligible breach in the containment. Figure 4.4-3 provides the best estimate failure pressure as a function of the drywell temperature. In addition, Figure 4.4-3 also indicates the uncertainty about the best estimate and provides the NUREG-1150 Peach Bottom containment median estimate of pressure versus temperature capability. Note that the additional failure mode due to hydrodynamic loads at elevated pool temperatures and high SRV discharge flow rates is not represented by the curve in Figure 4.4-3. It represents an additional failure criterion that must be considered. Once the analyst has determined the best estimate performance capability of the containment, the probabilistic split fractions, describing in general terms the size and location of the containment breach, are developed for each severe accident scenario postulated in the Level 2 assessment. This task requires that the analyst understand the accident signature of each scenario before the capability of the containment to respond to deteriorating conditions can be assessed probabilistically. Therefore, the MAAP code is used to determine the accident signature (i.e., containment pressure and drywell gas temperature), by modeling the scenario initially assuming that the containment has infinite capacity to remain intact. The analyst then superimposes the scenario signature onto the containment performance profile and compares the accident conditions with the probability tables to estimate the following event node split fractions describing the possible locations and magnitudes of the impending containment breach: (The uncertainty range is considered in the comparative evaluation of the plant response for calculated pressure and temperature traces that may approach the limits.)

- NO LARGE CONTAINMENT FAILURE
- DRYWELL INTACT

- WETWELL AIRSPACE FAILURE

Tables 4.4-6 through 4.4-9 summarize the results of the DAEC containment evaluation. The DAEC evaluation can be compared with the assessed failure probabilities at various temperatures with those estimated in NUREG-1150 and the BWROG Mark I analysis. The recommended profile of the DAEC Mark I containment pressure response as a function of containment temperature is presented graphically in Figure 4.4-3. Figure 4.4-4 illustrates the breach locations that represent failure in a particular containment zone. These locations are assigned to each zone to facilitate the calculation of the source term associated with a particular accident sequence that results in containment failure and radionuclide release to the Reactor Building.

4.4.6 Secondary Containment Failure Modes

The DAEC secondary containment (reactor building) surrounds the Mark I containment. The failure modes and locations of the DAEC reactor building are as follows:

Failure Location: The reactor building is a concrete structure with blowout panels located in the refuel floor roof. Therefore, overpressurization of the reactor building has been found to result in failure of the blowout panels and a release path through one of these blowout panel pathways.

Failure Modes: Reactor building overpressure failure at the refuel floor blowout panels is the dominant failure mode. No other failure modes have been assessed as probable in the reactor building failure mode assessment.

Nevertheless, the Level 2 assessment has assigned a high probability of a lack of reactor building effectiveness (i.e., a DF = 1.0) due to a number of possible phenomena such as:

- Hydrogen burning
- High flow rates
- Model inadequacy.

Table 4.4-6

SUMMARY OF DUANE ARNOLD CONTAINMENT CONDITIONAL FAILURE PROBABILITY AT LOW INTERNAL TEMPERATURES

$T \leq 500°F$[3]

Containment Failure Location	Failure Type	Mean Conditional Failure Probability		
		DAEC CB&I Analysis (@ 163 psig)	Peach Bottom NUREG-1150 (@ 147 psig)	Level 2 PRA Expert Assessment (@ 140 psig)
Zone C1 DW Head	Leak	0.30	0.25	0.32[5]
	Rupture	0.06		0.06
Zone C2 DW Upper Body	Leak	N/A		
	Rupture	N/A		
Zone C3 DW Main Body	Leak	0.02		
	Rupture	0.00		
Zone C4 WW Above Water Line	Leak	0.26	0.15	0.34
	Rupture	0.14	0.25	0.20
Zone C5 WW Below Water Line	Leak	0.16		0.001
	Rupture	0.06	0.35	0.08

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(1) Drywell failures in these zones are considered less likely, and are treated as conservatively represented by failures at Zone C1.

(2) Failures due to debris contact with the drywell shell are treated on a sequence by sequence basis; see Appendix C for the method of treatment.

(3) Containment pressurization caused by water vaporization and non-condensible gas generation post RPV breach. Containment interior conditions: $T \leq 400°F$, $P \sim 140$ psig.

(4) Two ply bellows.

(5) DAEC drywell head unseating pressure is less than Peach Bottom (75 psig vs. 82 psig)

Table 4.4-7

SUMMARY OF DUANE ARNOLD CONTAINMENT CONDITIONAL FAILURE PROBABILITY AT INTERMEDIATE INTERNAL TEMPERATURES
T = 500°F to 800°F[3]

Containment Failure Location	Failure Type	Mean Conditional Failure Probability		
		DAEC CB&I Analysis (@ 135 psig)	Peach Bottom NUREG-1150 (@ 112 psig)	Level 2 PRA Expert Assessment (@ 88 psig)
Zone C1 DW Head	Leak	0.42	0.40	0.7
	Rupture	0.24	0.12	0.2
Zone C2 DW Upper Body	Leak	N/A		(1)
	Rupture	N/A		(1)
Zone C3 DW Main Body	Leak	0.12		(1), (2)
	Rupture	0.06		(1), (2)
Zone C4 WW Above Water Line[4]	Leak	0.08	0.30	0.1
	Rupture	0.02		Epsilon
Zone C5 WW Below Water Line	Leak	0.04		Epsilon
	Rupture	0.02		Epsilon

(1) Drywell failures in these zones are considered less likely and are treated as conservatively represented by failures at Zone C1.

(2) Failures due to debris contact with the drywell shell are treated as sequences - by - sequence basis; see Appendix C for the method of treatment.

(3) Containment pressurization caused by water vaporization and non-condensible gas generation post RPV breach. Containment drywell interior conditions: T = 700°F, P ~88 psig.

(4) Two ply bellows.

(5) DAEC drywell head unseating pressure is less than Peach Bottom (75 psig vs. 82 psig)

Table 4.4-8

SUMMARY OF DUANE ARNOLD CONTAINMENT CONDITIONAL FAILURE PROBABILITY AT HIGH INTERNAL TEMPERATURES

$T \geq 900^{\circ}F$ [3]

Containment Failure Location	Failure Type	Mean Conditional Failure Probability		
		DAEC CB&I Analysis (@ 47 psig)	Peach Bottom NUREG-1150 (@ 36 psig)	Level 2 PRA Expert Assessment (@ 0 psig)
Zone C1 DW Head	Leak	0.32		0.5[4]
	Rupture	0.24		0.5[4]
Zone C2 DW Upper Body	Leak	N/A	0.5	[1][2]
	Rupture	N/A	0.5	[1][2]
Zone C3 DW Main Body	Leak	0.14		[1][2]
	Rupture	0.30		[1][2]
Zone C4 WW Above Water Line	Leak	0.00		Epsilon
	Rupture	0.00		Epsilon
Zone C5 WW Below Water Line	Leak	0.00		Epsilon
	Rupture	0.00		Epsilon

(1) Drywell failures in these zones are considered less likely and are treated as conservatively represented by failures at Zone C1.

(2) Failures due to debris contact with the drywell shell are treated as sequences - by - sequence basis; see Appendix C for the method of treatment.

(3) Containment pressurization caused by the water vaporization and non-condensible gas generation post RPV breach. Containment drywell interior conditions: $T \geq 900^{\circ}F$. Suppression pool airspace temperature is estimated to be $< 400^{\circ}F$, which is judged not to affect the integrity of penetrations inside the suppression chamber.

(4) The DAEC model has been constructed to assume that high temperature induced failures of the containment will be considered to have the impact of a large failure.

Table 4.4-9

SUMMARY OF THE DUANE ARNOLD CONTAINMENT CONDITIONAL FAILURE PROBABILITY UNDER DYNAMIC LOADING

Containment Location	Failure Type	LEVEL 2 PRA EXPERT ASSESSMENT	
		Unmitigated ATWS(1)	Failure at CET node CZ or CX(2)
Zone C1 DW Head	Leak	Epsilon	Epsilon
	Rupture	0.01	1.0
Zone C2 DW Upper Body	Leak	(3)	(3)
	Rupture	(3)	(3)
Zone C3 DW Main Body	Leak	(3)	(3)
	Rupture	(3)	(3)
Zone C4 WW Above Water Line	Leak	Epsilon	Epsilon
	Rupture	0.99	Epsilon
Zone C5 WW Below Water Line	Leak	Epsilon	Epsilon
	Rupture	Epsilon	Epsilon

(1) Suppression pool water temperature > 260°F and high power discharge rates to the pool combine large "chugging" loads with rapid containment pressure increase.

(2) These containment challenges (e.g., ex-vessel steam explosions) are conservatively assumed to result in large drywell failures.

(3) Drywell failures are in these zones are considered less likely, and are treated conservatively as failures in Zone C1.

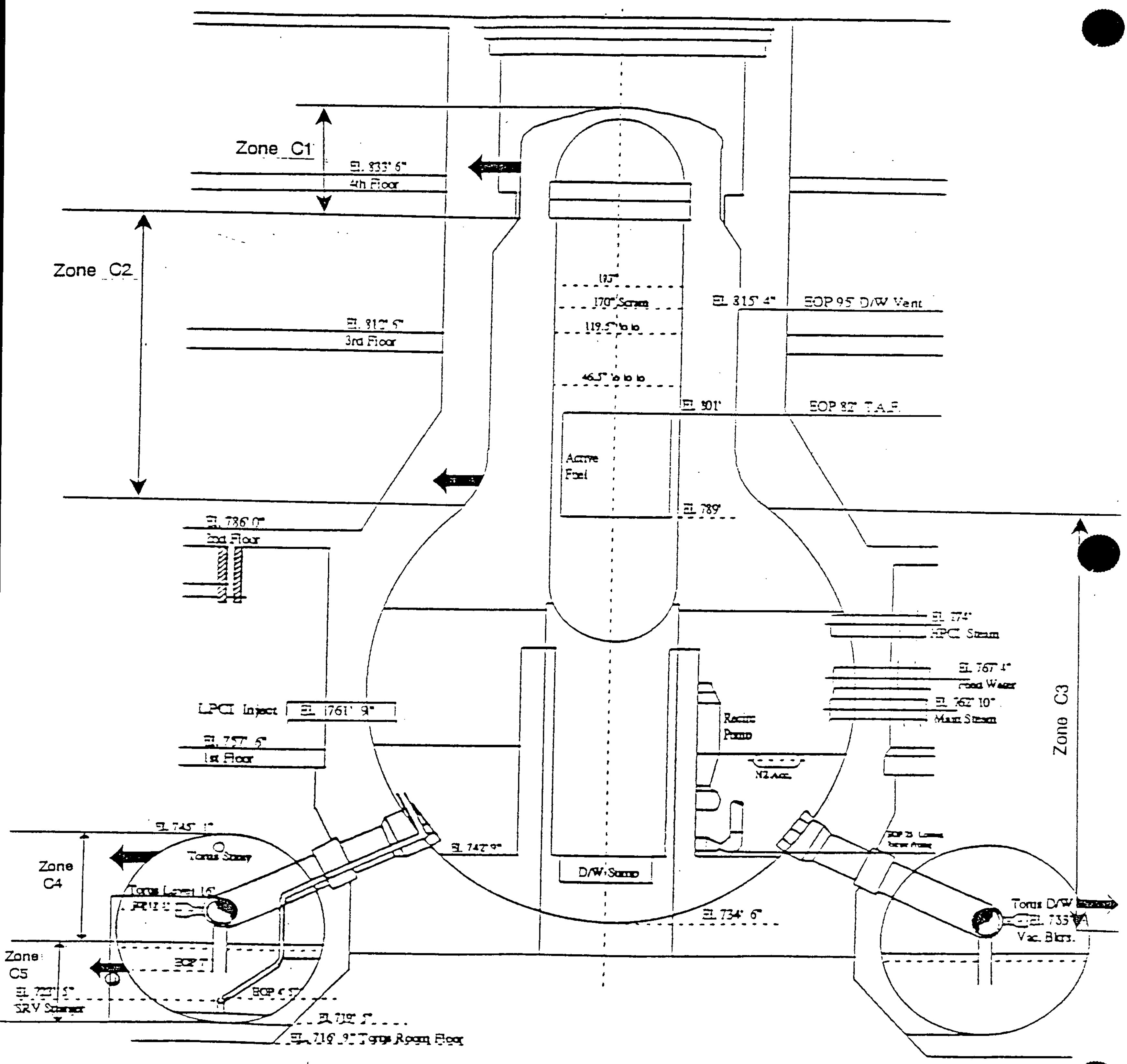

Figure 4.4-4 General Arrangement of DAEC Primary Containment Showing Postulated Failure Locations

4.5 DAEC CONTAINMENT EVENT TREE REPRESENTATION

4.5.1 Containment Evaluation Process

DAEC Containment Event Trees (CETs) are developed to provide the link between: (1) the Level 1 event tree core damage end states; and (2) safe shutdown or radionuclide release end states that describe release magnitude and timing. The CET is used to map out the possible containment conditions affecting the radionuclide releases associated with a given core damage sequence. The DAEC IPE uses containment event trees that integrate system/human responses with phenomenological aspects of a severe accident. The potential for recovery actions based on the accident management philosophy of the EOPs is included. Additionally, these models describe the various potential radionuclide release paths to the environment and provide an estimate of their relative likelihoods. The approach chosen focuses on the treatment of containment failure mechanisms and the timing of such failures. Application of this approach makes use of a number of deterministic and probabilistic risk assessment tools to establish a framework for radionuclide release evaluation. The spectrum of radionuclide releases which could result from these end states is then calculated for the postulated discrete end states of the CET.

The DAEC containment event tree structure has the following features:

- Represents the time sequence of events and divides the CET into major time periods;

- Incorporates all important system, human and phenomenological occurrences including possible recovery;

- Maintains a simplified representation;

- Avoids the necessity of intermediate binning;

- Preserves the nature of the challenge throughout the analysis;

- Divides sequence treatment based on whether the RPV is at high or low pressure and whether the accident progression is in-vessel or ex-vessel;

- Explicitly recognizes the effect of postulated containment failure modes;

- Allows the identification of recovery and repair actions that can terminate or mitigate the progression of a severe accident (note that prevention measures have been addressed in the system evaluation of core damage frequency); and

- Categorizes the end states of the resulting sequences into groups that can be assessed for their affect on public safety.

The first objective was achieved by representing the containment event tree as a series of chronological occurrences based upon MAAP runs and NRC code timing results. Some compromise to time phasing occurs where two events are mutually dependent upon each other. These are minimized, however, and the event tree generally represents the timed sequence of events from initiator to sequence end state.

The analysis implements the containment event tree assessment in a time phased approach. The first time phase involves occurrences up to vessel breach i.e., including opportunities for in-vessel recovery. The second time phase covers the period from vessel failure or arrest in-vessel until the intermediate term phenomena have occurred. This can be visualized as being approximately 3 to 15 hours after vessel challenge. The third time phase includes longer term phenomena such as containment heat removal and reactor building response. These time phases may overlap in certain accident scenarios.

The remaining objectives were satisfied by using a sufficiently large number of top events and through the use of functional fault trees to describe qualitatively and quantitatively the interrelationship among mitigating systems, operator actions, and the resulting end states.

The containment event tree (CET) is a tool for identifying and analyzing the spectrum of accident scenarios which may evolve following postulated core damage accidents. By considering the active and passive mitigating functions which can occur after a significant amount of core degradation, end states are identified in which the primary containment maintains its integrity or functionality. It has been recognized, since the publication of WASH-1400, that there can be a significant conservatism in the reactor plant risk estimates if the containment functionality is assumed to be ineffective following postulated core degradation or melt sequences. Nevertheless, CETs are developed and quantified in order to provide a realistic and systematic assessment of:

- The relative possibility of successfully mitigating postulated accidents

- The severity and timing of associated radionuclide releases from a degraded core accident.

The following were used as input to the CET models:

- Containment Walkdown Results

- P&IDs of Containment Control Systems

- Drawings of Containment Structure and Penetrations

- Technical Specifications

- Containment Leak Data

- Operating Experience

- Containment Structural Analyses

- EOPs (Including Containment Control)

- Level 1 Analysis and Results
- Deterministic Model (e.g., MAAP).[1]

The CET provides a characterization of the state of containment from the time of the initial core damage to either mitigation of the accident within the RPV or penetration of the RPV. The core melt progression sequences are also followed through their potential interaction with the containment to states involving either successful mitigation within the containment, or a radionuclide release.

Given the entry states, the CETs model the containment response (i.e., the core and containment conditions which could affect the accident progression paths and challenge the containment), plus the active and passive mitigative capability of the plant systems to terminate or reduce the radionuclide release.

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[1] The IPE utilizes the MAAP code for plant specific analyses of containment challenges. However, industry experience and staff positions on phenomenological uncertainties are also taken into account.

The CET allows a detailed characterization of the state of containment from the time of the initial core damage to either mitigation of the accident within the RPV or penetration of the RPV. The core melt progression sequences are also followed through their potential interaction with the containment to states involving either successful mitigation within the containment or a radionuclide release.

In the development of the CET, the important factors which affect the consequences for an accident are considered. Consequences in this context are measured in terms of the magnitude and timing of the radionuclide release. The primary focus of the back-end analyses is on containment failure mode and release timing rather than on source term analysis. The release and transport of both radioactive material inside containment and that released to the environment are tracked for future input to the accident management process. The identification of the containment failure mode and timing is generally used as an indicator of the type of response that can either mitigate or reduce containment failure probability.

The CET structure includes event tree nodes that address the following four aspects of severe accidents that are considered important in characterizing a radionuclide release:

- Core damage accident class (i.e., the entry state to the CET);
- Mitigating system response including operator actions (post core melt);
- Containment response, including pressures, temperatures and possibly failure location, path, and size, if appropriate;
- Reactor building response including failure size and location which are sequence dependent; and

- Phenomenological effects that can alter any of the above.

Because of the large number of interrelated degraded core accident phenomena which must be considered, the process of evaluating the severe core damage events and their effects on containment can be a complex and iterative task. Given the entry states, the CETs model the containment response (i.e., the core and containment conditions which could affect the accident progression paths and challenge the containment), plus the active and passive mitigative capability of the plant systems to terminate or reduce the radionuclide release.

4.5.2 Description of Containment Event Tree Models

Several types of containment event trees are necessary to characterize various containment challenges. The DAEC IPE directly links the front-end to back-end portions of severe accident sequences through directly linked event trees. These trees convey the support state conditions throughout the front-end and back-end trees and include considerations of preventive or mitigative features, as well as timing considerations. Three different containment event trees, each linked to the front-end trees, are used to properly handle cases with containment failure before core damage, core damage before containment failure, and containment bypass sequences. The CETs include:

- CET1: Class I and III CETs: Containment initially intact. These sequences are characterized by an initial loss of coolant makeup to the reactor vessel that leads to core damage.

- CET2: Class II and IV CETs: Containment initially failed or seriously challenged before core melt. For these classes of accidents, the primary containment boundary would fail before or at the time the molten core penetrates the reactor vessel. In Class II accident sequences, the inability to remove heat from the containment results in heat up of the suppression pool and a gradual containment pressurization. A more rapid pressurization is expected for Class IV accidents (e.g., ATWS). Reactor power remains above decay heat levels so that the amount of energy transferred to the suppression pool exceeds its heat removal capacity.

- CET3: Class V: Containment bypassed and direct release path established from the RPV to the reactor building. The Class V CET is used to evaluate two distinct core melt scenarios. LOCAs outside containment for which coolant makeup to the reactor vessel has failed leads to a core melt event with a direct release pathway from the vessel to the reactor building, and an interfacing LOCA or drywell bypass.

Examples of the three generalized types of CETs are given in the following figures:

Initiating Accident	CET Characteristic	Example CET	
		Figure	Table
Class I, IIIA, B, C, & IIT	Containment initially intact	4.5-1	4.5-1
Class IIID, IV, IIA, IIL, & IV	Containment initially failed at time of core melt initiation	4.5-3	4.5-2
Class V	Containment bypassed	4.5-4*	None

* See Section 4.5.4

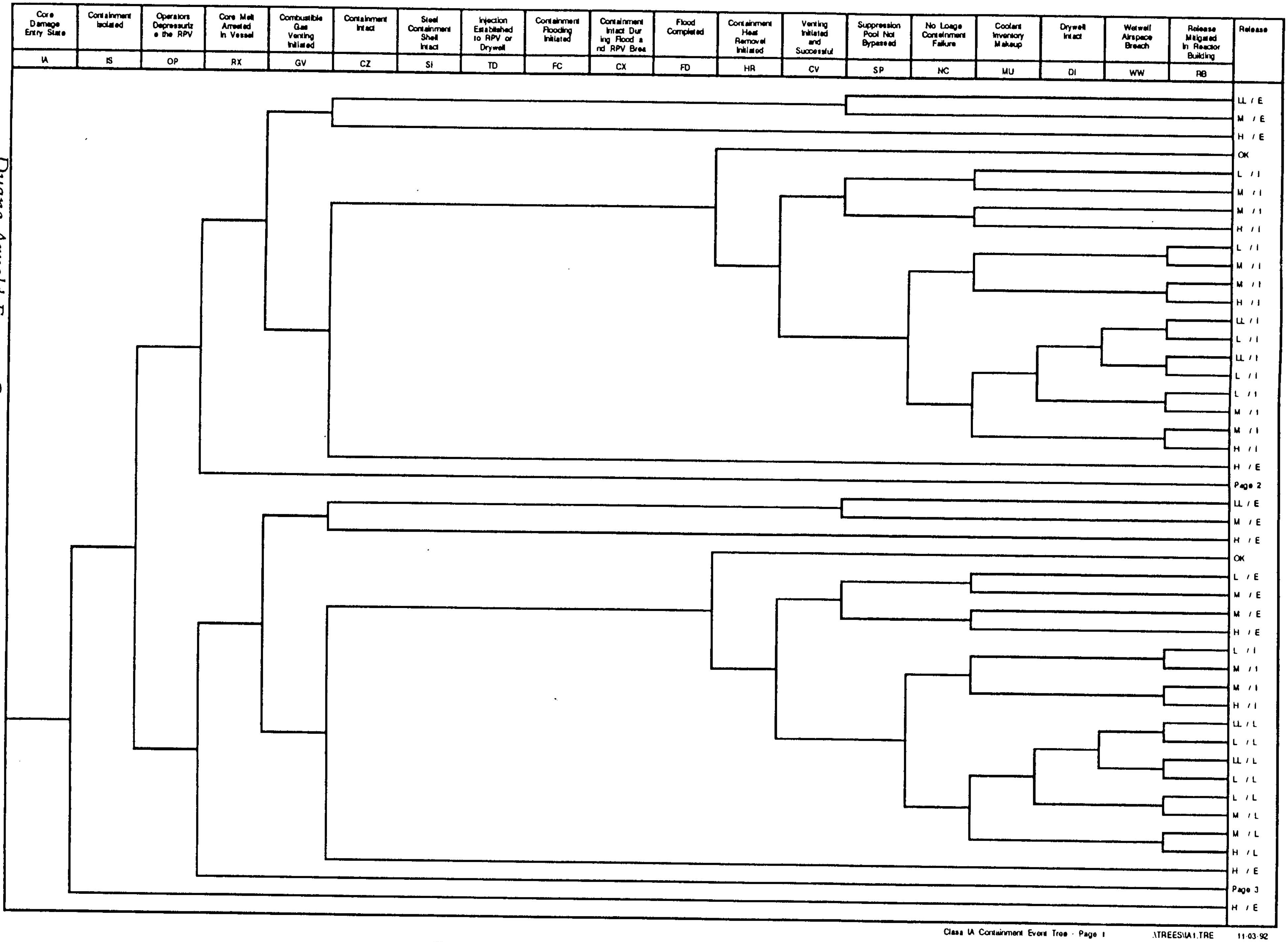

Figure 4.5-1

Duane Arnold Class ontainment Event Tree

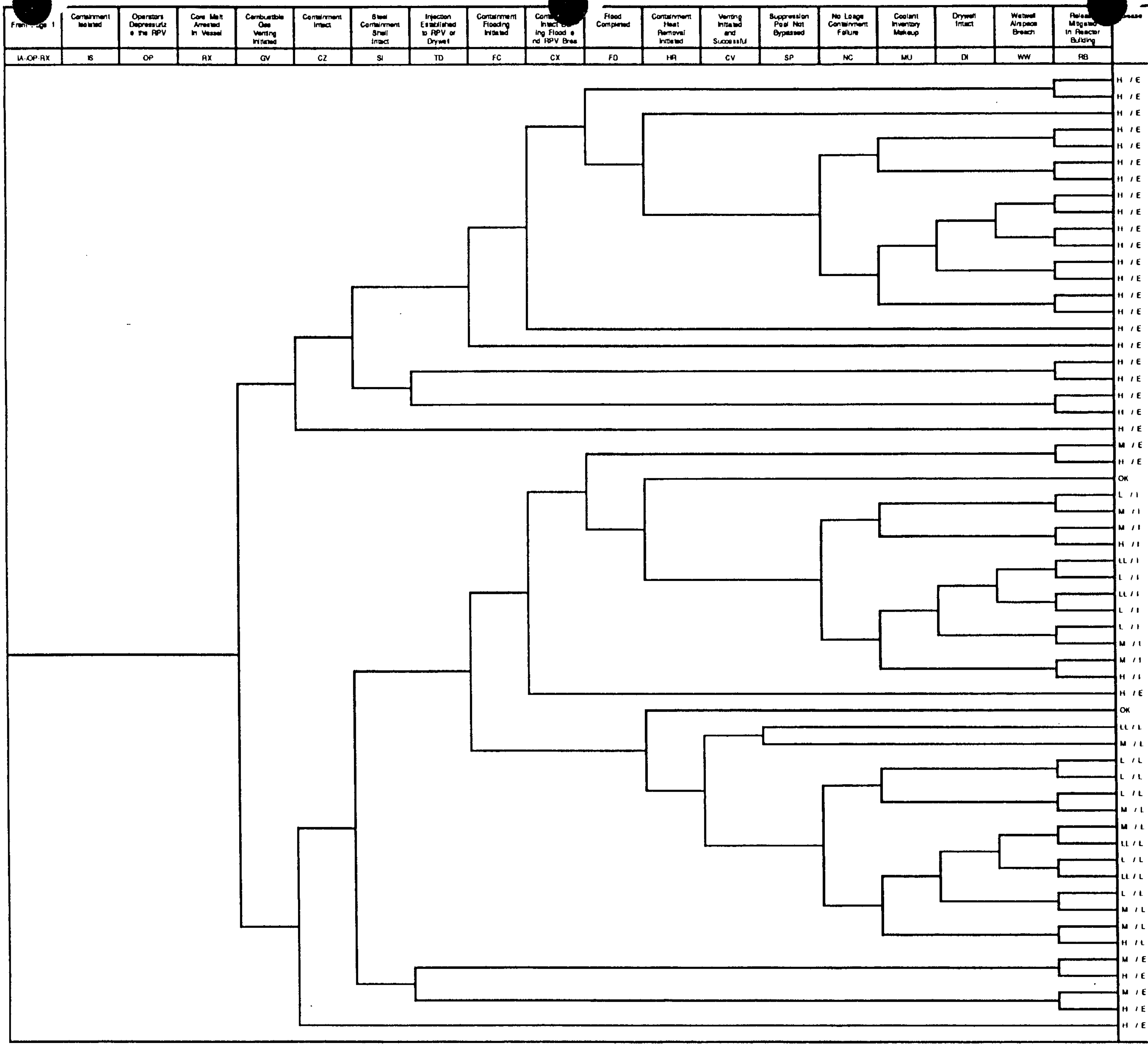

Figure 4.5-1 (con't)

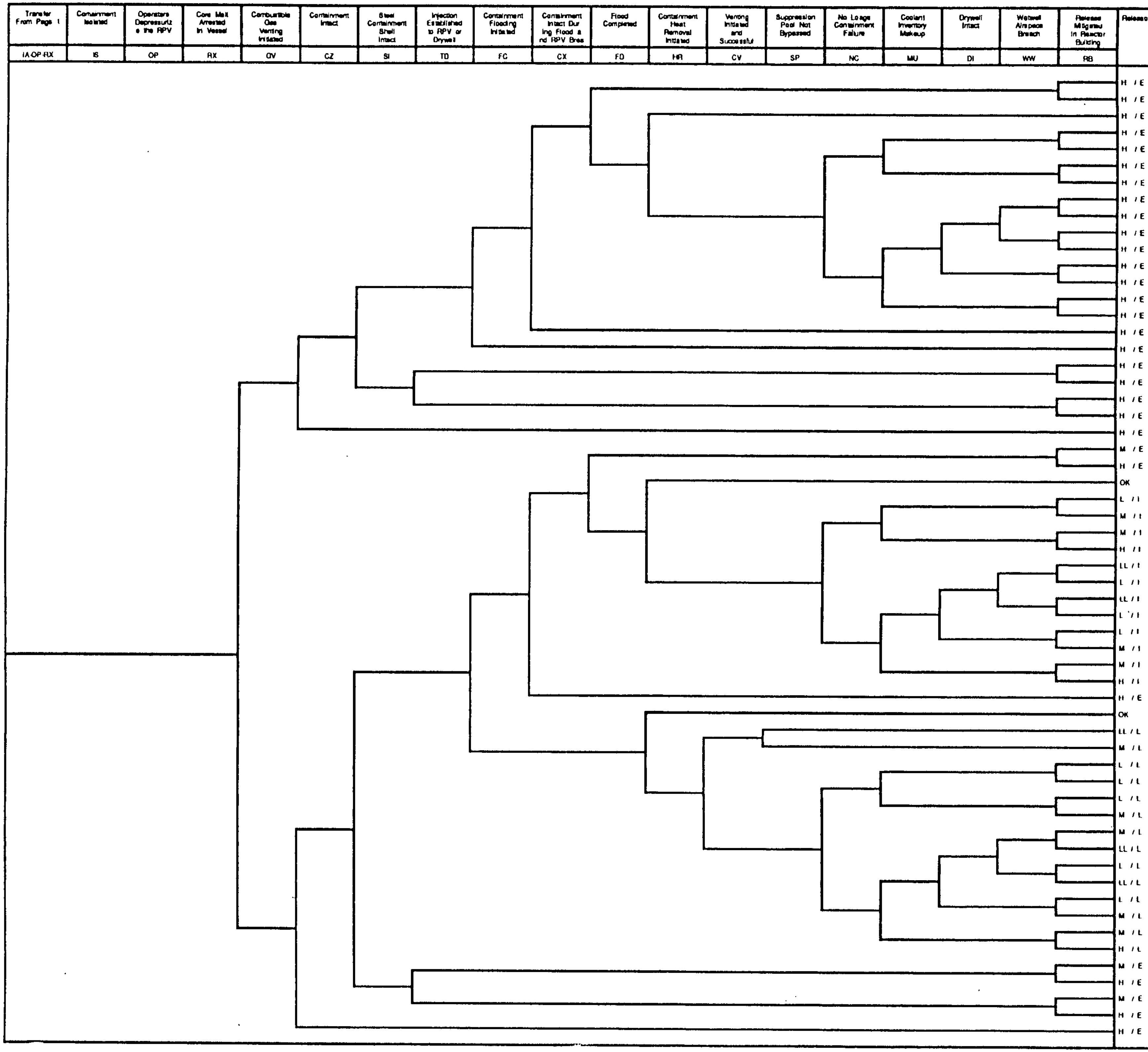

Figure 4.5-1 (con't) Duane Arnold Clas Containment Event Tree

Table 4.5-1

LEGEND FOR FIGURE 4.5-1

Top Event Designator	Top Event Description
IA	Core Damage Entry State
IS	Containment Isolated
OP	Operator Depressurized the RPV
RX	Core Melt Arrested In-vessel
GV	Combustible Gas Venting Initiated
CZ	Containment Intact
SI	Mark I Shell Failure Precluded
TD	Injection Established to RPV or Drywell
FC	Containment Flooding Initiated
CX	Containment Intact During Flood or RPV Breach
FD	Flood Completed
HR	Containment Heat Removal Initiated
CV	Venting Initiated and Successful
SP	Suppression Pool not Bypassed
NC	No Large Containment Failure
MU	Coolant Inventory Makeup
DI	Drywell Intact
WW	Wetwell Airspace Breach
RB	Release Mitigated in Reactor Building

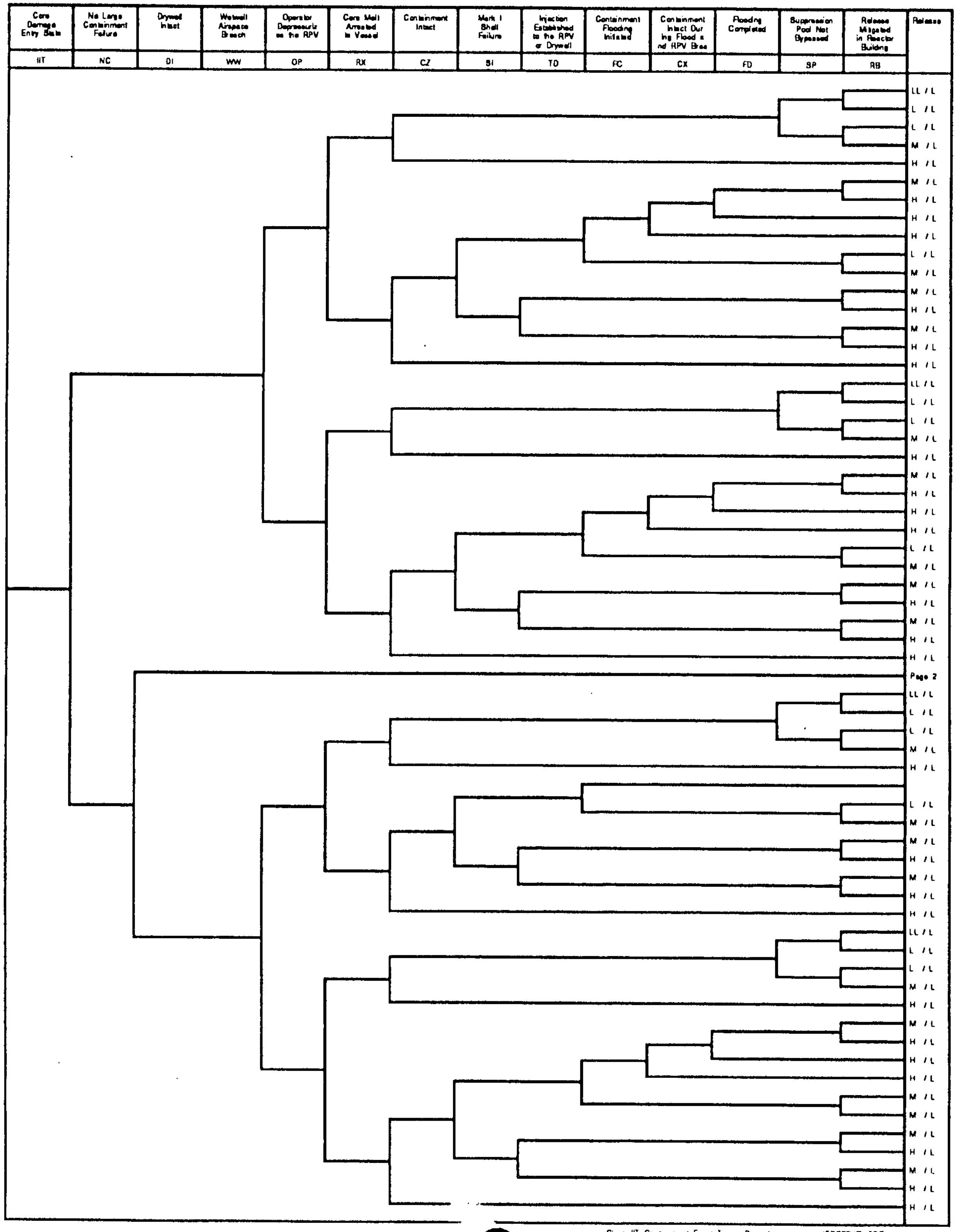

Figure 4.5-2 Duane Arnol[illegible]ss IIT Containment Event Tree

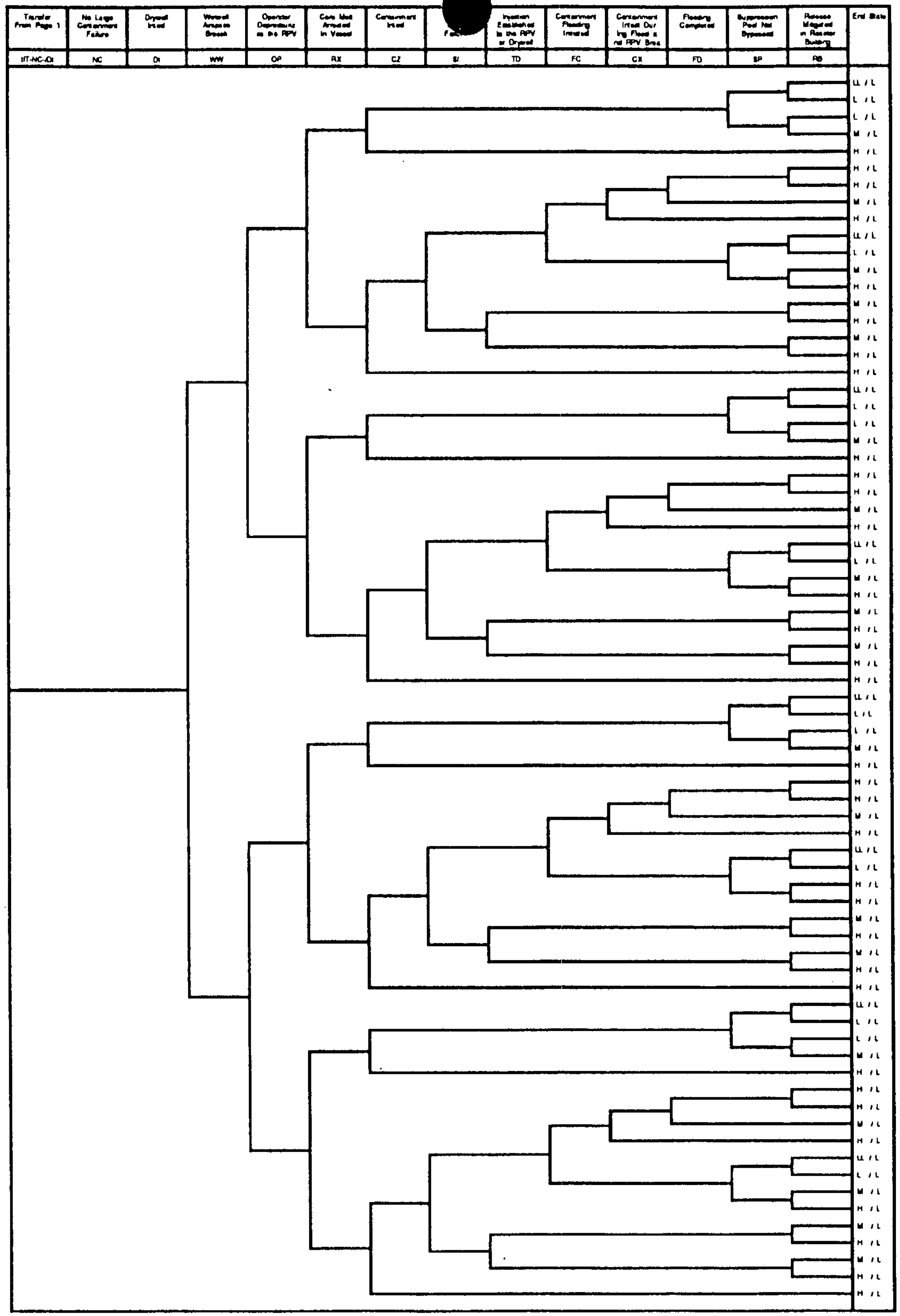

Figure 4.5-2 (con't) Duane Arnold Class IIT Containment Event Tree

Table 4.5-2

LEGEND FOR FIGURE 4.5-2

Top Event Designator	Top Event Description
IIA	Core Damage Entry State
NC	No Large Containment Failure
DI	Drywell Intact
WW	Wetwell Airspace Breach
OP	Operator Depressurizes the RPV
RX	Core Melt Arrested In-Vessel
CZ	Containment Intact
SI	Mark I Shell Failure Precluded
TD	Injection Established to the RPV or Drywell
FC	Containment Flooding Initiated
CX	Containment Intact During Flood and RPV Breach
FD	Flooding Completed
SP	Suppression Pool Not Bypassed
RB	Release Mitigated in Reactor Building

4.5.3 Description of CET1 and 2 Functional Nodes (All Classes except Class V)

The functional event nodes of the CETs are:

- Containment Isolation (IS)
- Operator Depressurizes RPV (OP)
- Core Melt Progression Arrested In-Vessel (RX)
- Combustible Gas Venting (GV)
- Early Containment Failure (CZ)
- Drywell steel shell intact (SI)
- Coolant injection for temperature control of molten debris (TD)
  - Containment Flood (FC, CX, FD)
- Containment heat removal
  - RHR (HR)
  - Venting (CV)
- Suppression Pool Bypass (SP)
- Containment breach size (NC)
  - Leakage
  - Overpressure failures

- Coolant inventory makeup (MU)

- Location of containment breach

  - Drywell (DI)
  - Wetwell airspace (WW)

- Reactor Building effectiveness (RB)

These top level functional events are described in more detail below.

Containment Isolation (IS)

Consistent with the NRC preference indicated in NUREG-1335, containment isolation is among the first nodal decision points of the CET. The "IS" node is used to assess whether the DAEC containment has been successfully isolated given the core damage challenge identified in the Level 1 PRA. Because the DAEC containment is required to be inerted, there is high confidence that the containment is also isolated.

The IPE examines in detail the status of the containment isolation systems prior to core melt. This node considers:

- The pathways that could significantly contribute to containment isolation failure,

- The signals required to automatically isolate the penetration,

- The potential for generating the signals for all initiating events,

- Consideration of testing and maintenance, and

- The quantification of each containment isolation mode (including common-mode failure).

Initiating events that include containment isolation failure are binned as part of the Level 1/Level 2 interface process and are transferred to CETs that bypass the IS node (i.e., Class IIA, IIL, IIV, IV, IIID, and V sequences fall into this category).

Operator Depressurizes the Reactor Vessel (OP)

This heading represents the manual or automatic action of depressurizing the RPV. The operator recovery action to depressurize the reactor allows low pressure system injection to the RPV if the low pressure systems are available. The upward path at this node represents successful depressurization and the down path models failure.

The status of RPV pressure can have a profound impact on the ability to successfully mitigate a severe accident and the subsequent containment response. Therefore, the determination of the RPV pressure is key to understanding subsequent active and passive mitigation capability.

Core Melt Progression Arrested In-vessel (RX)

The containment event tree (RX) node addresses the ability to arrest core melt within the reactor vessel. Specifically, success requires recovery of coolant makeup to the reactor vessel so that cooling may be reestablished to prevent further degradation of the fuel integrity. For debris to be coolable, a source of in-vessel injection with flow in excess of that required to remove decay heat must be available. When the debris is cooled in-vessel the threat to containment integrity will be reduced since: 1) the potentially large containment and drywell loads resulting from mechanisms at vessel failure (fuel coolant interaction, and vessel blowdown) will not be present, and 2) the production of combustible gases (e.g., H_2) resulting from debris/concrete attack will be avoided. The time window for successful recovery of coolant inventory occurs between core melt initiation and the time when the core melt progression cannot be halted within the RPV. This can be one hour to several hours depending upon the sequence of events and the analytic model used to model the situation.

The assessment addresses:

- The operator action to inject to the RPV
- The equipment availability
- Phenomena which may preclude successful arrest of the core melt progression in-vessel.

The makeup sources to ensure debris cooling in-vessel consist of the same sources examined in the Level 1 system evaluation. Therefore, the "RX" node is primarily an examination of repair and recovery actions that can occur in the time window of in-vessel core melt progression. Note that "RX" success is also strongly dependent on the successful RPV depressurization of the previous node, (OP). In turn, RX also has strong influences on subsequent CET nodes such as "TD", availability of water injection to the containment after RPV breach. The "TD" node examines water recovery over a much longer time frame of 2 hours up to 10-20 hours.

Combustible Gas Venting (GV)

This node addresses the possibility that the containment may have a combustible gas mixture and no operator actions would be taken to mitigate the condition.

This CET heading characterizes the potential for venting the containment during accident sequences in which combustible gases may be present. The upward branch defines the path where the vent has been opened to control combustible gas mixtures, given the unlikely situation that the containment is de-inerted. The downward path represents cases in which the containment remains inerted and vent is not required.

Early Containment Failure (CZ)

This node addresses postulated severe accident phenomena that can result in an energetic failure of containment during the core melt progression.

Energetic containment failure modes resulting from the core melt accident sequence initiator and the subsequent core melt phenomena at the time of initial RPV breach due to debris attack are estimated to have potentially high radionuclide releases.

Event heading (CZ) describes the condition of the containment after a failure of the primary system. In the upward path, the containment has remained intact during the initial stages of core melt progression up through RPV breach and blowdown, while the downward path depicts an overpressure failure of the drywell induced by a loss of primary system integrity.

The containment is the primary defense in retaining core melt fission products. The failure modes considered in the early containment failure model include the following:

- Containment pressurization due to RPV blowdown causes rapid containment pressure rise above capability,
- Steam explosion,
- Direct containment heating,
- Recriticality,
- Core/concrete interactions,
- Hydrogen deflagration in a de-inerted containment, and
- Debris impingement.

The structure of the CZ model is divided into in-vessel and ex-vessel phenomena, depending upon the success or failure of the RX node. Shell failure due to debris contact is explicitly modeled under the SI node.

Wherever possible, the MAAP code is used for plant specific analyses of containment challenges. However, deterministic analyses regarding the capability of the DAEC containment to withstand the various energetic accident phenomena were not performed. Rather, industry studies and staff positions on phenomenological uncertainties were taken into account to assign failure probabilities that are deemed appropriate for the DAEC Mark I containment.

An assessment of the DAEC containment capability in response to slower developing overtemperature and overpressure scenarios (e.g., loss of debris cooling, loss of containment heat removal) was performed and is described in Section 4.4. Those slow developing scenarios are inherently different than the energetic failures modeled by the CZ node, and thus, are modeled under a separate node (NC) so that the different potential releases can be accounted for.

Drywell Shell Fails (SI)

The Mark I drywell steel shell provides the primary containment boundary. The drywell steel shell interfaces directly with the concrete floor of the drywell. The drywell floor is also the location where a substantial fraction of the core debris may be deposited if core damage cannot be arrested in-vessel and the RPV is subsequently breached. Without substantial water injection to the containment during the core melt progression, it is found that the steel shell would likely fail from extremely high local temperature.

This node addresses whether adequate water is available to the drywell to prevent drywell steel shell failure. This is contingent on equipment availability, an assessment of the phenomena of debris coolability and liner integrity, and the operator action to initiate debris cooling.

Active Mitigation Temperature Control (TD)

Subsequent to debris attack of the RPV, containment challenge may occur from high temperatures in the drywell, or a combination of high temperatures coupled with high pressures due to non-condensible gas generation. Injection of water into the containment and/or the RPV can mitigate the consequences of a core melt and prevent either of these failure modes. Each of these are discussed below:

- Drywell Sprays

Drywell sprays can mitigate the consequences of a potential core melt accident. The sprays can perform three functions, the two most important of which are: (1) scrubbing fission products that are not otherwise scrubbed (i.e., in the case where the suppression pool is bypassed); and (2) providing water to cool the core debris on the drywell floor. In this mode of operation, containment failure could be prevented by termination of drywell wall heating and the associated temperature induced containment failure, and non-condensible gas generation due to core concrete reaction.

- Vessel Water Injection

RPV water injection can perform some of the same functions as spray operation mentioned above (i.e., scrub fission products from the debris), prevent containment overtemperature failure, and reduce the core concrete reaction by quenching the debris. The systems that might perform the function of coolant injection post core melt at DAEC include:

- Fire system
- Control rod drive pumps
- Low pressure coolant injection (RHR)
- Core spray
- Standby liquid control (SLC system)
- Condensate pumps
- Well water systems
- Emergency service water system

- General service water system

Operation of the vessel water injection systems after vessel failure will act to cool the core debris that remains on the drywell floor, cool the drywell atmosphere as a result of steam generation[1] and cool the RPV internal structure (i.e., this cooling may prevent fission product revaporization from the RPV.) The post-core melt water injection and associated steam will prevent the drywell from reaching very high temperatures. For Class II, IV and V with the containment already failed, preventing the drywell from overheating will prevent overtemperature failure of the drywell head or seal, the drywell shell, or the penetrations. An added benefit for vessel water injection after vessel breach is the potential to scrub ex-vessel fission products via the water overburden.

Containment failure size and location is dependent on the status of this CET function.

<u>Containment Flood (FC, CX, FD)</u>

These nodes address the question of whether the procedures and operator actions will be taken to flood the containment with external water during the core melt progression, or whether the actions will be to maintain suppression pool level at approximately the LCO limits. The availability of an external injection water source instrumentation to monitor the injection, and vent capability are all included.

Success indication flooding of the containment above the vessel normal core height is achieved. The three segments of the evolution are initiation of flooding, containment integrity during venting of RPV during flood process, and completion of flooding.

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[1] Ex-vessel debris coolability is highlighted in NUREG-1335 as an issue to be considered.

Containment Heat Removal (HR)

This node addresses the availability of the RHR system and the operator action to initiate the system for containment heat removal.

The DAEC Mark I containment system is provided with significant heat capacity and heat management capabilities. The management of heat in the containment prior to, during, and following a severe core damage event directly affects containment response. The DAEC containment heat capacity can be classified as both active and passive. The passive capacities include the suppression pool and the containment structure. The active heat management capabilities include the RHR system, the RWCU system, venting, and containment fan coolers. This event tree node addresses all heat management capabilities, but the dominant influence on successful containment heat removal post core melt is the RHR system. (Note containment venting is discussed separately below.) Severe accident effects on the performance of the RHR system (e.g., steam binding) are considered in the model.

The RHR system, operating in the suppression pool cooling mode, can maintain long term containment integrity through adequate containment heat removal if other failure modes can also be mitigated. With the RHR system operating during the course of a core melt accident, containment pressure and temperature can be maintained within the structural failure criteria of the containment. As a result, the consequences of a radioactive release to the environment can be prevented.

The upward branch at this event tree node represents successful containment heat removal via the RHR system operating in the suppression pool cooling mode. The downward branch models failure of containment heat removal.

Wetwell Vent (CV)

This event heading characterizes use of the wetwell vent to relieve containment pressure, and provide an alternate path for containment pressure control. Venting provides the operator a means of removing decay heat and non-condensible gases, and maintain the integrity of the containment. At this node, the upward path represents successful use of the vent, while the downward path represents venting failure due to mechanical faults, inadequate procedures, or operator error. Severe accident effects on the performance of the wetwell vent (e.g., high differential pressure prevents valve operation) are considered in the model.

Suppression Pool Bypass (SP)

This node is an assessment of hardware availability to preserve the suppression function of the torus.

If the operator is unsuccessful in maintaining the heat management functions as described in the preceding section, wetwell venting would be required to maintain containment integrity. In this situation, this event heading examines the potential for suppression pool bypass that would allow the release of radionuclides from the reactor vessel to pass directly from the drywell to the wetwell air space without the benefit of suppression pool scrubbing during venting. Suppression pool bypass may result from a number of causes. These include: 1) structural failure of the drywell, 2) drywell vacuum breaker failure, 3) loss of suppression pool water below the level of the SRV quenchers, and 4) excessive leakage through drywell penetrations.

The upward branch at this event tree node represents no bypass, while the downward branch models a scenario in which release effluent passes directly to the wetwell air space and out the wetwell vent or a wetwell air space failure. This node is only asked following successful venting.

Containment Response Integrity (NC, DI, WW)

These nodes address only the size and location of the containment failure. NC questions whether the breach is large. If there is a large breach, DI determines if the breach is in the drywell or wetwell. If the breach is in the wetwell, WW determines whether the breach is above or below the water line.

For the purposes of a containment performance evaluation under severe accident conditions, it is useful to have a criterion to describe the adequacy of containment integrity as a function of pressure and temperature within containment. Using severe accident profiles from published Mark I severe accident analyses, criteria describing the containment capability to withstand these conditions, can be established. These criteria are based on published BWR containment assessments using the following priorities:

- DAEC specific
- Mark I specific
- BWR specific.

A containment response profile which represents a reasonable interpretation of published analyses for characterizing the severe accident performance of the DAEC Mark I containment is presented in Section 4.4. As shown in that section, the containment pressure and temperature capability limits have a considerable uncertainty associated with them. Assessments have been performed which identify the likely containment failure modes and conditions causing these failures for three distinct extreme cases; high temperature, high pressure, and an intermediate point for each. These three cases are used to establish the realistic performance limits describing the capability of the DAEC containment.

The containment failure location and its size will impact the calculated radionuclide releases. Failure location and size also depend on the core melt accident sequence and the operability of mitigating systems. Section 4.4 provides additional detail on the derivation of these failure mode locations, and discusses the basis for estimating the size of containment breach. The containment analysis meets the IPE requirement that plant-specific containment analyses be performed. The analysis considers the effects of high temperatures and pressures on seals, valves, hatches, and other key areas of the containment structure (e.g., drywell head area). When studies of reference plants were used, their applicability to DAEC was taken into consideration and explicitly discussed.

A more complex CET could examine the possibility of a small containment breach subsequent to any severe accident, even those that are adequately mitigated by coolant injection and containment heat removal. The simplification that is used here is that the resultant leakage would be relatively small and within the capacity of the SGTS, such that little if any release greater than the DBA would be calculated. NUREG-1150 studies have shown that such small leakages make no measurable contribution to the assessed public risk.

Continued Inventory Makeup (MU)

This node evaluates the availability of combined makeup to the primary containment following failure of venting. This node considers the effect of harsh environment (e.g., humidity, temperature) following containment failure or venting on the availability and survivability of injection systems and components.

Reactor Building (RB)

This node includes an assessment of the active and passive features of the secondary containment (reactor building), along with phenomena that may cause bypass of the secondary containment.

The reactor building can act to retain a significant fraction of the radionuclides released from containment for certain severe accident scenarios. Time averaged decontamination factors for the reactor building vary between 1.0 and much greater than 10. The determination of whether the reactor building is effective is determined at this node.

Contributors to the determination of reactor building effectiveness include the following:

- Reactor building integrity after containment failure,
- Standby gas treatment system (SGTS) operation,
- Fire sprinkler operation (water curtains),
- Hydrogen combustion in the reactor building, and
- Reactor building integrity after hydrogen combustion.

The down branch of the reactor building node implies minimal effectiveness of the reactor building to retain fission products due to primarily two failure mechanisms:

1) Combustion of gases in the reactor building causing high temperature and minimum or zero retention.

2) Direct pathway from the containment failure location to the blowout panels with minimal interaction within the reactor building before release to the outside atmosphere.

4.5.4 LOCA Outside Containment (CET3)

The interfacing system LOCA outside containment Level 2 evaluation begins by developing an event tree sequence diagram which portrays accident progression following core damage. This event tree is given in Figure 4.5-3. The entry states are the core damage-containment bypass sequences from the Level 1 event tree end states ISLOCA (V-sequences). The Level 2 event tree end states are states involving radionuclide release magnitude and timing as defined in Section 4.6 of this report.

Containment event trees for other containment failure modes or plant damage states may include more detail on in-containment phenomena, however, for the LOCA outside containment plant damage state, the focus is primarily on the response of the reactor building and containment failure due to high temperatures. Therefore, this tree is simplified relative to the other containment event trees because its purpose is principally to assess the secondary containment building effects.

The LOCA outside containment Level 2 event tree nodes are discussed individually in the following paragraphs.

LOCA Outside Containment Entry State (I)

The Level 2 LOCA outside containment model is used to evaluate a spectrum of accident scenarios involving an unisolated leak or rupture outside containment in either a high pressure rated line connected directly to the RCS, or a low pressure rated line that is normally isolated from a high pressure system that interfaces with the RPV. Additionally, these scenarios are further defined as including the operator's inability to both isolate the LOCA and compensate for the inventory loss from the RPV using available coolant makeup systems before the level in the RPV reaches one-third core height. Therefore, it is presumed for the Level 2 entry state, that RPV water level decreases to the point of impending breach in the RPV bottom head due to debris attack, and that the LOCA remains unisolated for the duration of the scenario. This is the end of the Level 1 evaluation. The initiators include V sequence events and no additional recovery is included in the Level 2 analysis. This may be slightly conservative.

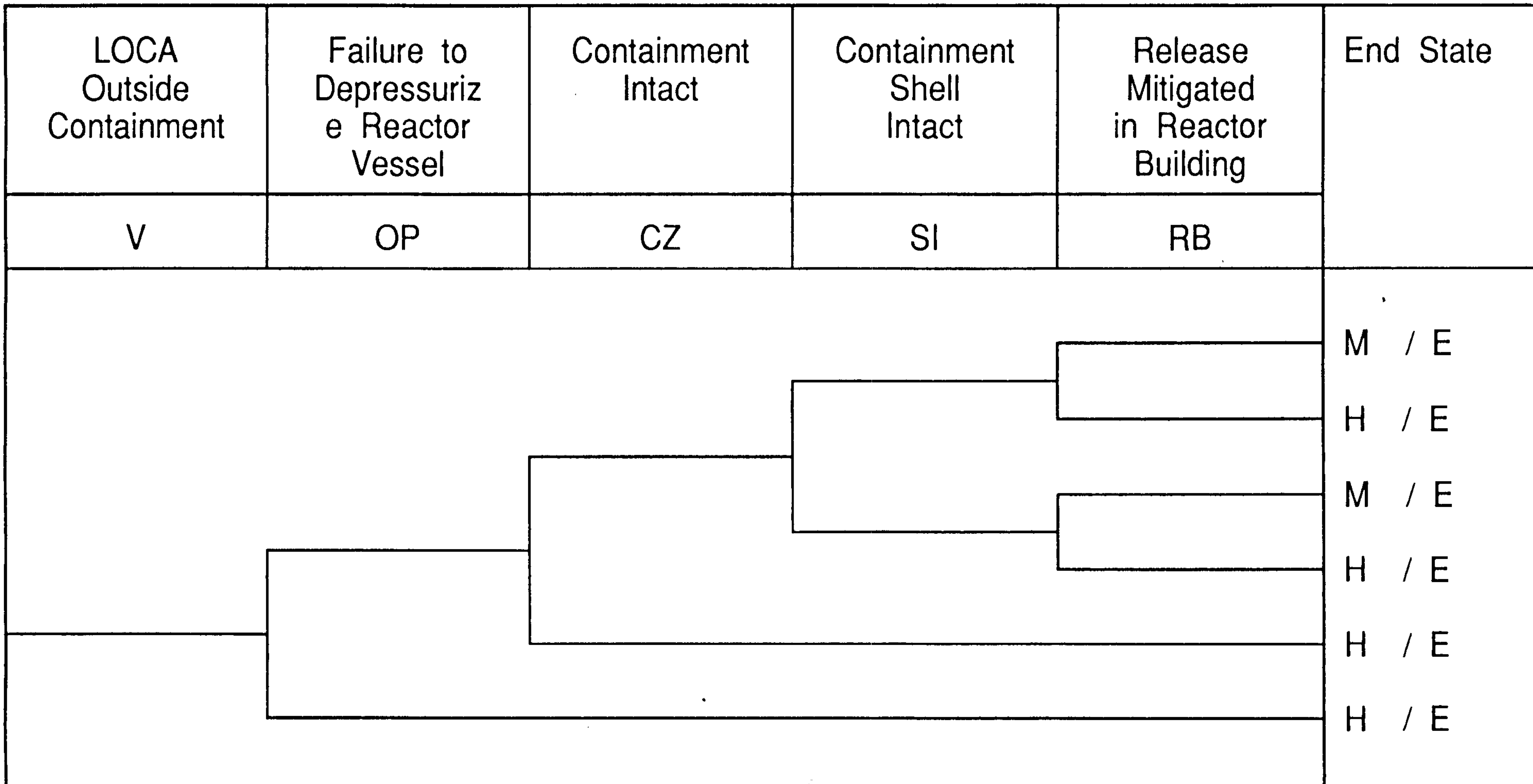
Operator Depressurizes the Reactor Pressure Vessel (OP)

Following the initiation event, the next event tree node examines the potential for the operator to depressurize the reactor vessel. The upward path at this node represents successful RPV depressurization before its impending failure, while the downward path depicts a failure to depressurize due to equipment failure or operator inaction. For all large LOCAs outside containment this node is always successful - i.e., always depressurized. The only time a non-success would occur would be for small LOCAs outside containment. This is determined on a sequence by sequence basis.

This event tree node may have substantial influence on the ability to mitigate a small break, minimize adverse impacts on the containment due to high pressure blowdown of the RPV, and minimize radionuclide releases. The primary effects associated with the ability to depressurize the RPV include the following:

- Allows injection to the RPV from the low pressure injection systems.

- Increases the flow from CRD pumps as pressure decreases.



Class V Containment Event Tree .\TREES\V.TRE 11-03-92

Figure 4.5-3 Duane Arnold Class V Containment Event Tree

- Reduces the stresses on the primary system components; therefore, increasing the likelihood that the primary system will remain intact.

- Reduces the likelihood that a high pressure blowdown coupled with inadequate vapor suppression would lead to an immediate containment failure.

- Reduces the likelihood that molten material will be finely dispersed in the containment atmosphere leading to a direct containment heating failure mode.

The success of the depressurization function for the RPV following core damage initiation is similar to the criterion established in the Level 1 analysis. However, there are additional phenomena (i.e., non-condensible gas generation and potentially very high internal temperatures) which can occur during the accident progression beyond core damage that pose further challenge to the operator's ability to depressurize the RPV. However, it is presumed that there are no additional failure modes affecting the operator's use of the ADS solely introduced by the occurrence of a LOCA outside containment than would be expected to challenge the operator employing the same equipment during a LOCA inside containment.

<u>In-vessel Recovery (RX)</u>

No in-vessel recovery is included.

<u>Containment Remains Intact and Isolated Post RPV Breach (CZ)</u>

The assessment of the radionuclide release associated with a LOCA outside containment event can be considered to have two distinct pathways of release:

- First, the release through the break or leak outside containment results in a release directly to the reactor building. This is the principal release pathway and the one which has been focused on in published PRAs.

- The second release pathway is of importance, if and only if, the first pathway results in very small release. In such cases the continued core melt progression inside containment may inevitably lead to a second containment failure that could compound the releases from the first mechanism.

This event heading characterizes the behavior of the containment when the molten core penetrates the reactor vessel. A breach of the reactor vessel would result in containment pressurization that could pose an early challenge to containment integrity. An early containment overpressure failure, caused by energetic effects associated with core melt or RPV blowdown post vessel breach, would provide, in this instance, a second direct release path to the reactor building.

The CZ node is used to identify those potential containment failure modes that have the following characteristics:

- They are generally energetic in nature,

- They result in potentially large drywell failures, and

- They occur at or near time of core melt progression or RPV breach.

These containment failure modes provide fission product pathway characteristics that result in the potential for substantial radionuclide releases to the reactor building; although, they are of low conditional failure probability.

The upward path at this node indicates that the containment drywell remains intact, or while the downward path depicts an energetic breach of the containment drywell. Successful containment integrity results from preventing early, phenomenologically induced containment failure modes. The mechanisms for containment failure at the time the molten core debris breaches the RPV primarily include extreme pressure and temperature effects. Secondary effects, such as those caused by either energetic interaction between debris with water (either in-vessel or ex-vessel) or debris dispersed throughout containment, may also severely challenge containment integrity. Tertiary effects (i.e., missile impingement) are also considered as potential mechanisms for causing containment failure. Containment isolation failure is also included here.

Containment Shell Intact (SI)

This node considers the possibility that the ex-vessel core debris contacts the drywell shell and melts through it. Considering that sufficient water injection has not been available to prevent core damage and vessel melt-through, it is assumed that the core debris fails the drywell shell in all cases.

Reactor Building Effectiveness (RB)

The last event tree node is the reactor building effectiveness. The potential issues that influence the determination of reactor building effectiveness are the strong dependence of the radionuclide residence time in the reactor building on the following events or features of the secondary containment:

- The mode of containment failure,
- The location of containment failure relative to the reactor building point of failure,
- The location of any water flooding in or into the reactor building,
- The rate of gas production in the primary containment,
- The parallel flow path of containment shell failure and late containment drywell failure
- The status of the SGTS,
- The status of the railroad doors or other reactor building bypass paths, and
- The potential for hydrogen burning in the reactor building.

4.5.5 Level 2 PRA Success Criteria

The Level 2 containment event tree (CET) describes accident progression from initiation of core damage to successful mitigation or release. Each node within the CET requires a definition of what success implies. As part of the containment event tree development and the Level 2 evaluation of core melt progression and mitigation there will also be successful end states in which radionuclide releases from the reactor building will be prevented or substantially contained. As part of the evaluation, these success states require a consistent criteria to be applied in order to allow the quantitative assessment of the radionuclide release frequency and magnitude.

For these reasons it is important to establish the success criteria to be used in the qualitative and quantitative analysis. Failure to meet these criteria may result in extreme containment conditions (i.e., excessive temperature and/or pressure) that could challenge containment structural integrity.

4.5.5.1 Overview of Level 2 Success Criteria

The overall success criteria for the Level 2 evaluation can be defined in terms of successful achievement of the following safety functions:

- RPV integrity,
- Containment integrity, and
- Reactor Building effectiveness.

These overall success criteria describe the bases upon which each protective barrier is examined to determine its effectiveness in radionuclide release mitigation. Table 4.5-5 summarizes the success criteria used for these top level safety functions.

4.5.5.2 Functional Success Criteria

Using the overall success criteria established above, the next step is to interpret these success criteria in terms of key functions that can be explicitly defined and quantified within the Level 2 model.

Table 4.5-3

OVERALL LEVEL 2 SUCCESS CRITERIA

Protective Barrier	Success Criteria
RPV Integrity	RPV integrity is assured if no LOCA has occurred and the following criteria are met: a) RPV internal temperature must be maintained below 4000°F; if not, then RPV structural failure and depressurization is assumed at the location of high temperature. b) Bottom head temperatures and local penetration are maintained below their melting temperature. If molten debris is calculated to fail instrument tubes or CRD penetrations, then RPV structural failure and depressurization is assumed.
Containment Integrity	The containment integrity is assured if the following criteria are met: · No energetic, early containment failure modes occur;[1] · Pressure and temperature within the containment must be maintained below the best estimate containment capability; · Containment is isolated (i.e., no unisolated openings greater than 2 inches in diameter); and · Uncooled core debris must be prevented from contact with the drywell shell. If these criteria are not met the containment is assumed to fail.

Table 4.5-3
OVERALL LEVEL 2 SUCCESS CRITERIA

Protective Barrier	Success Criteria
Reactor Building Effectiveness	Substantial radionuclide release (i.e., a factor of 5 to 10 reduction in the radionuclide release magnitude) requires either of the following: . Primary containment failures low in the reactor building (e.g., torus room) for which the release pathway consists of a torturous pathway through the reactor building and no natural circulation pathway is created which can sweep fission products out of the reactor building. . Submerged releases from the wetwell where the release pathway remains submerged during the release process.

Table 4.5-4 summarizes which CET nodes (top events) impact the three protective barriers and their success criteria. It also provides a cross-reference among these protective barriers and the specific success criteria, developed for each CET functional node, that the conditions required for their successful implementation during each accident scenario.

Table 4.5-5 summarizes the success criteria for each functional node in the containment event tree (CET).

Table 4.5-4

Summary of CET Top Events Which Affect Primary Success Criteria[1]

Protective Barriers	Cont. Isolation (IS)	RPV Depressurized (OP)	RPV Intact (RX)	Combustible Gas Control (GV)	Cont. Intact (CZ)	Cont. Shell Intact (SI)	Injection Recovered (TD)	Cont. Flooding (FC)	Cont. Intact During Flood (CX)	Cont. Flooded and Drywell Vented (FD)	RHR Heat Removal (HR)	Cont. Vent (CV)	Suppression Pool Not Bypassed (SP)	No Large Bypass (NC)	Inventory Makeup Preserved (MU)	Drywell Intact (DI)	Wetwell Airspace Failure (WW)	Reactor Building Effective (RB)
RPV Integrity		X	X	X														
Containment Integrity	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	
Reactor Building Integrity and Effectiveness	X	X	X	X	X	X		X	X	X				X		X	X	X

[1] "X" denotes that the CET functional node has an impact on the success or failure of the respective protective barrier. The CET functional nodes are discussed more completely in Appendix C subsections.

Table 4.5-5

FUNCTIONAL SUCCESS CRITERIA

CET FUNCTIONAL NODE	SUCCESS CRITERIA
Containment Isolation (IS)	The success of the containment isolation node (IS) is satisfied if the containment penetrations that communicate between the drywell (or wetwell) atmosphere and the reactor building (or environment) are "closed and isolated". The criteria used to satisfy this requirement of "closed or isolated" is that no line, hatch, or penetration has an opening greater than 2 inches in diameter. This implies that all containment penetrations are adequately sealed and isolated during the entire accident progression until either: (1) a safe stable state is reached; or, (2) the accident conditions exceed the ultimate capability of containment as determined in the plant specific evaluation.
RPV Depressurization (OP)	This function questions whether the operator depressurizes the RPV after core damage but before vessel breach has occurred. Success of this action would allow low pressure injection, if available, and would minimize the challenge to containment due to a high pressure RPV rupture. The functional success criterion for this node is defined as having the RPV depressurized (i.e., less than 100 psig) until core melt is arrested in-vessel or until the RPV is breached by debris attack. The success of the depressurization function for the RPV following core damage initiation is similar to the criterion established in the Level 1 analysis, i.e., prior to core damage. However, there are additional phenomena (i.e., non-condensible gas generation contributing to a high containment pressure that prevents SRV operation, and potentially very high containment temperatures which could fail electrical and mechanical components of the SRVs) which can occur during the accident progression beyond core damage and pose further challenge to the operator's ability to depressurize the RPV. The success criteria is to depressurize the RPV to less than 100 psig. The success criteria, in terms of systems, is the same as that used prior to core damage, i.e., • Any single SRV[6] or • Failure of the primary system due to high temperature during core melt progression.[1] or • A large or medium LOCA. Other alternatives[2] may be available but are not credited in this analysis.

Table 4.5-5

FUNCTIONAL SUCCESS CRITERIA

CET FUNCTIONAL NODE	SUCCESS CRITERIA
Arrest Core Melt Progression In-vessel (RX)	In-vessel recovery or arrest of core melt progression addresses the ability of the operating staff to restore adequate core cooling from the time the end state of the Level 1 PRA occurs (i.e., RPV water level less than 1/3 core height and decreasing) until restoration of water injection make-up cannot prevent the breach of the RPV bottom head by debris. As part of the definition of success, it is also useful to define what constitutes failure to maintain the RPV intact. The two primary failure medes that have been identified in the literature include: • Local penetration seal failure due to debris heat up and local failure at welds, • Creep rupture failure of the entire bottom head The MAAP evaluation calculates that the RPV integrity would be challenged by debris contact with local penetration welds. This is supported by experiments by R. Leahey (RPI) which indicate for PWRs that drain plug configurations are susceptible to failure. This configuration correlates to the BWR instrument tubes or CRD seals. The base quantification assumes that RPV failure occurs at local penetrations. The large, bottom head failure scenario is treated as a sensitivity case. Preventing the core melt from progressing outside the reactor pressure vessel requires the timely introduction of water onto the debris and intact fuel assemblies. Both timing and system requirements must be defined as part of the success criteria. There are differences in core melt progression models regarding the ability to recover adequate cooling under different circumstances. These vary from no credit for retention of debris in-vessel after core melting has begun (MAAP), to substantial credit for recovery even after debris has accumulated in the bottom head (MARCH). The best estimate success criteria used in this evaluation are based on the time available from the initiation of core degradation until just before substantial core relocation occurs. This typically is on the order of 30-40 minutes. In terms of system requirements, coolant injection is assumed necessary to re-flood the RPV to above 1/3 core height. It is judged, based on deterministic calculations, that this can be accomplished using makeup systems (identified in the EOPs) with capability greater than approximately 1000 gpm.[5]

Table 4.5-5

FUNCTIONAL SUCCESS CRITERIA

CET FUNCTIONAL NODE	SUCCESS CRITERIA
Combustible Gas Venting (GV)	The functional success criterion at this node is that the containment vent and purge lines are opened to allow combustible gas mixtures to be removed from containment. The downward path of GV in the CET implies that combustible gas venting has not been initiated. Therefore, on the downward path either of two conditions may exist: • The containment is inerted[3] or • A combustible gas mixture is present The probabilistic evaluation of these two states on the downward branch are treated in the Containment Remains Intact Early (CZ) node. Hydrogen combustion that could lead to containment failure is prevented by either of the following: • Deinerted operation with no oxygen intrusion during the accident • Combustible gas purging and venting through the purge and vent lines If both these success paths fail, the hydrogen detonation is assumed to occur, resulting in containment failure. The location of the failure is assumed to be in the drywell head region and is classified as a large failure.

Table 4.5-5

FUNCTIONAL SUCCESS CRITERIA

CET FUNCTIONAL NODE	SUCCESS CRITERIA
Containment Remains Intact (CZ)	The functional success criteria for the containment intact node are that the containment retains its pressure capability and that no early containment failure modes compromise the containment integrity. The early containment failures modeled by the CZ node are characterized by phenomenological events (e.g., steam explosions, missile generation, direct containment heating) that are estimated to challenge containment integrity relatively quickly following core melt. Late containment failures, modeled in subsequent nodes, are characterized by extreme pressure and temperature conditions that develop slowly over the course of the accident due to inadequate containment heat removal. Note that successful prevention of early containment failure does not necessarily precludo late containment failure. Therefore, successful prevention of early containment failure requires the following: · No direct containment heating (direct containment heating is precluded if the RPV is already depressurized) • No ex-vessel steam explosion • No failure of vapor suppression (i.e., the suppression pool is not bypassed no more than 1 drywell to wetwell vacuum breaker fails open) • No in-vessel steam explosion (i.e., in-vessel steam explosions are precluded if either the RPV is at high pressure, e.g., greater than 100 psig or the core does not fragment into fine particles before dropping onto the bottom head) • No high pressure spike sufficient to cause containment failure occurs at the time of vessel melt-through (i.e., extreme pressure spikes are precluded if the RPV bottom head penetration fails locally; or the RPV remains at low pressure) • No hydrogen deflagration or detonation (i.e., if the containment remains inert or effective combustible gas vent was operated successfully; then, hydrogen detonation or defiagration is guaranteed not to occur). • No RPV blowdown from high pressure with the suppression pool temperature above 240° F • No recriticality due to an unusual core configuration that may be achieved during the melt progression. If these failure modes cannot be prevented, containment failure is assumed to occur. The failure location is assumed to be in the drywell head region and is classified as a large failure.

Table 4.5-5

FUNCTIONAL SUCCESS CRITERIA

CET FUNCTIONAL NODE	SUCCESS CRITERIA
Drywell Shell Remains Intact (SI)	Success at this node requires that water is available (greater than 1000 gpm) to the core debris at the time of vessel failure. Shell failure can occur relatively quickly (i.e., minutes) following RPV failure if water is not available to quench the core debris. It is assumed in the model that the core debris will come in contact with and fail the drywell shell if water is not available.
Ex-vessel Debris Coolability (TD)	Ex-vessel core debris coolability can be considered to be successful if very high containment temperatures, core concrete ablation, and substantial non-condensible gas generation that can result from poorly cooled debris can be prevented. These are considered preventable if either of two situations exist: (a) on a best estimate basis a continuous water supply is available to the debris with a flow rate of greater than 1000 gpm; or (b) the passive nature of containment prevents overtemperature failure. The two methods that may provide adequate coolant injection to the debris bed include continued make-up to the RPV and initiation of drywell sprays. However, there are some models that indicate that concrete attack and non-condensible gas generation will not be terminated even if substantial water injection is available to the debris. The temperatures in the drywell will be acceptable, but continued non-condensible gas generation will occur. In addition, it turns out that a passive method of preventing drywell head failures also exists when a drywell shell breach has previously occurred, i.e., prevention of a second drywell failure mode. This "passive" method of prevention is represented by the extended time it takes to heat up the drywell. This time is well beyond that which is being evaluated in the IPE (i.e., 36 hours past RPV breach). Plant specific MAAP runs confirm that the drywell temperature remains below the failure pressure and temperature of the drywell for those cases in which debris attack has previously failed the drywell shell. Therefore, for shell failure cases, there is a high probability that multiple containment failures will not result even though no additional active systems may become available. Failure at this node could result in either of the following occurring: • High temperatures in the drywell, or • Excessive concrete ablation causing pedestal structural failure or basemat penetration. These effects would influence the integrity of containment. This node differs from SI in that water need not be present coincident with vessel failure. Time is available in which to restore debris cooling before very high containment temperatures develop and threaten additional containment failures. Note that TD is considored successful if the SI function is successful.

Table 4.5-5

FUNCTIONAL SUCCESS CRITERIA

CET FUNCTIONAL NODE	SUCCESS CRITERIA
Containment Flooding Initiated (FC)	Success at this node implies that the containment flooding contingency procedure has been initiated by the operating staff and that a system of adequate flow capacity from external sources is available to implement the procedure. In addition to these two requirements, the instrumentation must be available to initiate the flood operation.
Containment Remains Intact (CX)	The success branch of the CX node occurs if two situations can be prevented: • Blowdown of the RPV into a reduced free volume (i.e., the increased water level creates a reduced free volume that results in a decreased capability of the containment to accept blowdown loads. and • Core melt progression causing RPV failure and a large steam vaporization. These two failure modes are somewhat dependent upon the relative timing of containment fill versus core melt progression. In addition, the effects are dependent on the following: • Whether the RPV is depressurized allowing injection of external water sources (Node OP), and • Whether containment flooding is accomplished through injection nozzles outside of the RPV (i.e., drywell sprays and RHR suppression pool return lines).
Containment Flooded Above Debris (FD)	This node evaluates the pessibility that the operator suspends containment flooding because the staff is unable to maintain containment conditions within prescribed limits described in the EOPs. Success at FD includes drywell venting. Since it is presumed that containment pressurization will occur during the latter stage of flooding as a result of a diminishing drywell volume, the operator will be required to establish a drywell vent path (i.e., > 8 inch equivalent diameter). Drywell venting can have varying degrees of releases associated with it depending on the following: • When in the containment flood process drywell venting is required, and • Whether success of RHR suppression peol cooling and injection is effective in controlling containment pressure Success at this juncture in the model is defined as the continuation of the flooding evolution with containment conditions remaining within the limits of the Maximum Primary Containment Water Level Limit (MPCWLL).

Table 4.5-5

FUNCTIONAL SUCCESS CRITERIA

CET FUNCTIONAL NODE	SUCCESS CRITERIA
Containment Pressure Control (see node descriptions HR and CV below)	Successful containment pressure control is achieved if either of two functional nodes are successfully satisfied; (1) RHR containment heat removal or (2) Containment venting. Because these have different potential impacts on the radionuclide releases they are treated in separate nodes.
(1) RHR Containment Heat Removal (HR)	Successful containment pressure control is unattainable using RHR[4] suppression pool cooling if the following conditions are not satisfied: • Debris cooling (in-vessel or ex-vessel) • No "Early" containment failure modes.

Table 4.5-5

FUNCTIONAL SUCCESS CRITERIA

CET FUNCTIONAL NODE	SUCCESS CRITERIA
(1) RHR Containment Heat Removal (HR) (con't)	RHR has the capability to remove heat from containment through the RHR heat exchangers. This capability requires: • A flow path from the suppression pool • One RHR pump • One RHR heat exchanger • RHRSW to cool the heat exchanger • A return flow path to: - The suppression pool - The RPV - The drywell spray (wetwell spray flow rate is considered to low). • Bypass of the low RPV water level (2/3 core height) interlock if not using RPV return • Not using injection path from service water through the RHR cross tie Failure at this juncture in the sequence implies insufficient containment heat rejection to the environment and that the continued decay heat generation could subject the containment to continued pressurization. This condition may eventually cause structural failure, which could subsequently threaten continued successful core coolant injection. Note that RHR success is a moot point if adequate injection to the core or debris has failed. This is because of high temperatures from debris radiative heating or high pressure from non-condensible gases will cause drywell failure.

Table 4.5-5

FUNCTIONAL SUCCESS CRITERIA

CET FUNCTIONAL NODE	SUCCESS CRITERIA
(2) Containment Venting (CV)	The capability to vent the wetwell is a valuable supplement to the containment pressure control systems. As pressure and temperature increase, there is decreasing confidence in the ability to maintain the integrity of the containment pressure boundary. By instituting a controlled vent of the containment atmosphere, it is possible to maintain long term containment integrity by providing a viable means of containment pressure control and heat removal. Venting also constitutes a viable mitigative action to minimize the source term released to the environment. Containment venting is successful if it can remove the excess heat and non-condensible gases from the containment and, thereby, maintain the containment pressure within acceptable limits. Adequate pressure control can be obtained by containment venting if the following conditions are satisfied: • Reactivity control exists • No "early" containment failure modes occur • Containment flooding does not eliminate the venting pathways • Vent pathways can be opened and controlled. Besed upon deterministic calculations, a containment vent of approximately 8 inches in diameter will provide sufficient vent capability to prevent containment failure for sequences involving the loss of containment heat removal or severe accidents. Currently, no vent capability is considered successful for unmitigated ATWS or failure to scram events.
No Suppression Pool Bypass (SP)	This node in the CET is used to characterize the magnitude of radionuclides that may escape the containment if wetwell failure or venting occurs. Success means that radionuclides are directed through the suppression pool. Subsequent headings address specific release paths. Success in preventing suppression pool bypass requires that: • No more than one vacuum breaker remains stuck open • The suppression pool water level remains above the bottom of the downcomers • The vent pipes, downcomers, or ring header do not rupture.

Table 4.5-5

FUNCTIONAL SUCCESS CRITERIA

CET FUNCTIONAL NODE	SUCCESS CRITERIA
No Large Containment Failure (NC)	This event examines the size of containment leakage that may be induced by extreme pressure and temperature conditions. The downward path at this event tree node is defined as large leakage or failure, while the upward path depicts either no leakage or the existence of drywell leak paths that prevent further containment pressurization. Any failure of the containment structure greater than 1 $ft.^2$ is considered to be a large containment failure and is modoled as a 2 ft^2 break in the MAAP runs. A small break is assumed to be 1 $ft.^2$ or less in size, and is modeled in MAAP with a leak size of 27 $in.^2$. A small containment break may be characterized by any of the following breach of containment: • Electrical penetration leak, • Hatch seal leak, • Bellows seal leak, or • Drywell head seal leak: - Thermal degradation - Inadequate pre-load Leak sizes up to 3 $in.^2$ in equivalent area are assumed to present a negligible impact on the course of the accident. The downward branch of the "No Large Containment Failure" node is probabilistically based on the plant specific structural analysis. However, there are certain cases in which failure (i.e., large break) is guaranteed. These cases include the following: • Failure to scram sequences with continued injection and no SLCS. • No injection to containment, causing high temperature induced failure, • Any early containment failure (e.g., steam explosion, etc.), or • LOCA plus failure of vapor suppression

Table 4.5-5

FUNCTIONAL SUCCESS CRITERIA

CET FUNCTIONAL NODE	SUCCESS CRITERIA
Coolant Makeup Remains Available Post Containment Failure (MU)	This event node is used to examine the availability of water injection to the drywell and RPV following containment failure. Failure of coolant makeup to the debris results in delayed fission product release due to heat up and revaporization of fission products on the RPV internals and containment structures. Releases are reduced if coolant injection can be maintained. The success of coolant makeup following containment failure may be compromised by any of the following: • Harsh environment in reactor building • Steam binding of pumps. • Disruption of injection pathways due to catastrophic containment failure. The same success criteria established for accomplishing ex-vessel debris coolability (node "TD") and averting shell melt-through (node "SI") influence the analysis of whether functional success is achieved at this node. Alignment of the following injection sources external to the reactor building (these systems are not hindered by steam binding or harsh conditions in the reactor building) may be used to achieve success: • RHRSW • GSW • ESW • Well Water • Fire System

Table 4.5-5

FUNCTIONAL SUCCESS CRITERIA

CET FUNCTIONAL NODE	SUCCESS CRITERIA
Drywell Intact (DI)	Containment failure has already been asked in the CET. If containment failure has not occurred, this node is bypassed. If containment failure is determined to have occurred, then "DI" node is included to distinguish whether the failure occurred in the drywell (failure branch) or wetwell ("success" branch). The probabilistic determination of the location of the failure is determined based on the plant specific structural analysis for slow overpressure events. Additional guidance is also provided for other accident scenarios as follows: • High temperature induced failures result in drywell failures • Rapid or energetic failure modes are assumed to occur in the drywell (e.g., steam explosions, etc.)
Wetwell Airspace Failure (WW) (Scrubbed Release)	This node appears after the Drywell Intact (DI) node. If the DI node determines that the containment failure occurred in the drywell this node is bypassed. If the containment failure occurred in the wetwell, this node distinguishes whether the wetwell failure occurred above or below the wetwell water line. As in the previous node, successfully avoiding a large containment failure requires successful containment heat removal. The probabilistic determination of the location of the failure is determined based on the plant specific structural analysis for slow overpressurization events.
Reactor Building Effectiveness (RB)	The reactor building provides a substantial capability to remove particulate fission products from the release pathway for scenarios where the containment has failed. Success of the reactor building to provide a substantial radionuclide reduction (i.e., a factor of 5 to 10 reduction in the radionuclide release magnitude) is based upon any of the following: • Very small containment failures (i.e., 2 inch equivalent diameter) for which the reactor building remains substantially intact • Primary containment failures low in the reactor building for which the release pathway consists of a circuitous route through the reactor building. • Cases in which substantial fire protection spray is occurring during the release (not credited due to limited area coverage at DAEC).

Notes to Table 4.5-5

[1]Primary system failure may be induced by very high internal temperatures generated by molten debris in an uncooled state within the RPV. Such high temperatures coincident with high RPV pressures may lead to localized failures at weak points high within the RPV.

[2]Opening MSIVs or the use of HPCI/RCIC steam lines are not credited because these are not directed by the EOPs, or are of insufficient capacity to lead to depressurization, respectively.

[3]For this situation the containment remains inerted and venting would not have been required. Therefore, in this case, the down branch is not considered as a failure of combustible gas venting but as a continuation of the sequence.

[4]Other modes of containment heat removal are <u>not</u> considered effective because of interlocks or procedural restrictions under severe accident conditions. (e.g., RWCU, Main Condenser).

[5]The 1000 gpm criterion is an approximation. There is a comparatively large degree of uncertainty surrounding this issue. However, ORNL and GE calculations seem to indicate that an injection rate close to 1000 gpm initiated at thirty minutes may be sufficient.

[6]A plant specific assessment of the DAEC response to a high pressure core melt with a late malfunction of a single SRV has shown that the RPV depressurizes well before RPV failure.

4.6 ACCIDENT PROGRESSION AND CET QUANTIFICATION

This subsection summarizes two important features of the Level 2 modeling:

- The accident progression description
- The CET quantification.

4.6.1 Characterization of Containment Performance Based on Accident Progression

The role of the containment as a vital barrier to the release of fission products to the environment has been widely recognized. The public safety record of nuclear power plants has been fostered by applying the "defense-in-depth" principle, which relies on a set of independent barriers to fission product release. The containment and its supporting systems are one of these barriers. Containment design criteria are based on a set of deterministically derived challenges. Pressure and temperature challenges are usually based on the design basis loss-of-coolant accident; radionuclide challenges are based on the source term of 10 CFR Part 100. Also, criteria based on external events such as earthquakes, floods, and tornadoes are considered. The margins of safety provided by such practices have been the subject of considerable research and evaluation, and these studies have shown the ability of many containment systems to survive pressure challenges of two to three times design levels.

Section 4.2 identified the deterministic models used in the DAEC containment performance assessment.

Section 4.4 identified those containment failure modes for which the DAEC containment would result in breach.

Section 4.4 further identified the containment capability limits over the range of assumed containment challenges:

- Pressure
- Pressure and temperature
- Excessive temperature
- Dynamic loads.

This section examines the types of challenges identified in Section 4.4 and the corresponding containment pressures and temperatures (where applicable) to identify the containment performance in the CET sequence evaluation.

Containment challenges are determined for each of the sequences identified for the various plant damage state bins. In this context, "challenges" refer to the potential for elevated pressures and temperatures, missiles, direct contact of containment by core debris, containment bypass, and the like. The magnitude of these challenges when compared with the containment capacity will determine if containment failure will occur and, if it does, the time at which failure is reached. Figure 4.6-1 shows the comparison of the individual sequence pressures and temperatures versus the containment capability curve. This information is therefore extremely important and is needed to quantify the CETs.

4.6.3.3 Examination of the Baseline Quantified Results of the DAEC CET

Different organizations may have different opinions on what are the most important issues related to the protection of the public health and safety. For example,

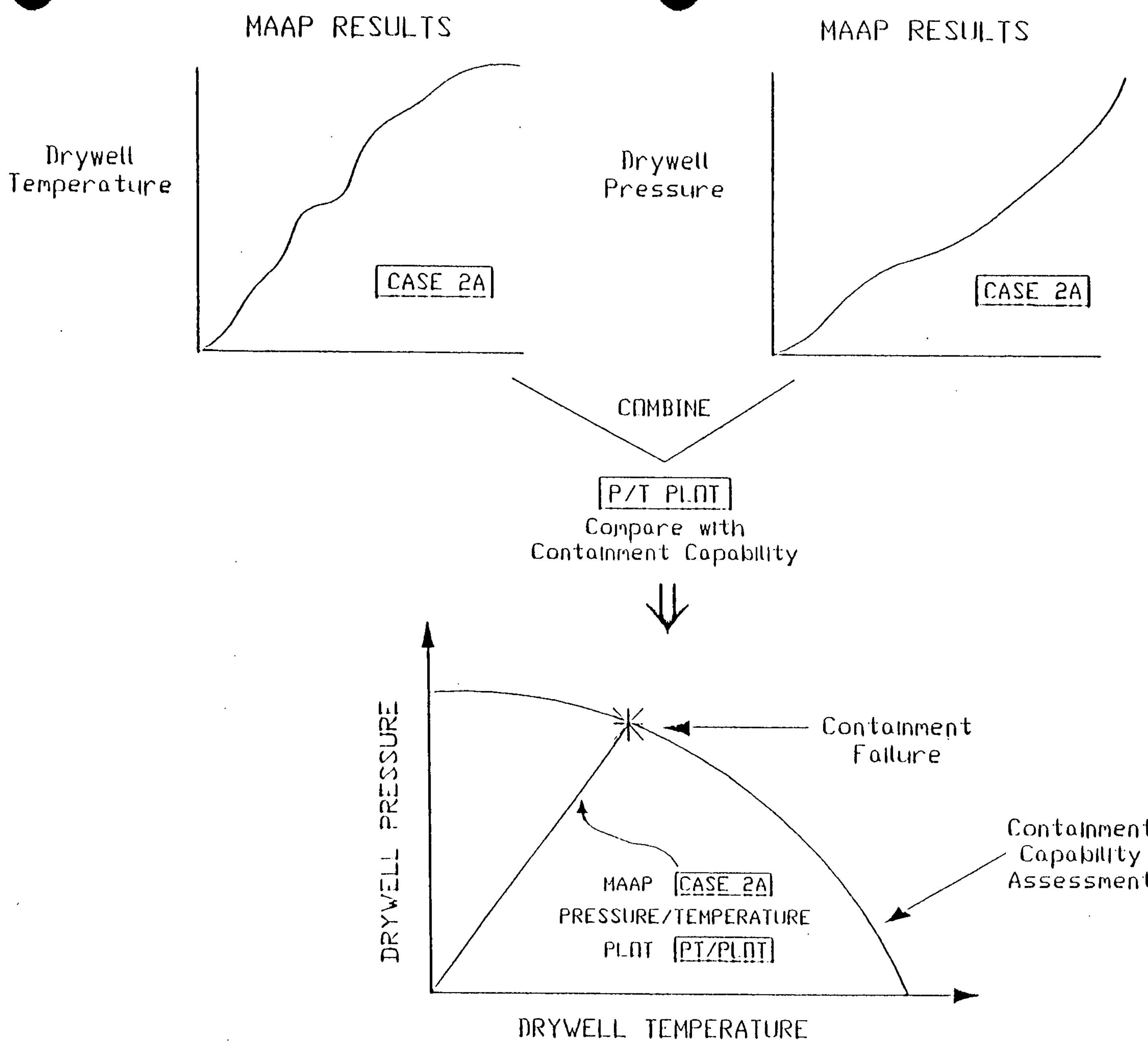


Figure 4.6-1 Simplified Flow Chart Showing the Relationship of MAAP Deterministic Results Compared with the Ultimate Containment Capability

Federal Emergency Management Administration (FEMA) may consider the understanding, prevention, and mitigation of accidents that could result in changing evacuation plans or evacuation effectiveness as the key issue. For such determinations both the magnitude and timing of the accident sequence is of importance. On the other hand, some organizations may consider latent health effects to be the dominant contributor to public risk and therefore the magnitude of the release is of principal importance regardless of the timing. To account for these different viewpoints, the radionuclide release binning is summarized in different ways such that various organizations can make the most effective use of the information for their specific purpose. Therefore, this subsection examines the quantitative results of the DAEC Level 1 and Level 2 PRA evaluations from a number of different viewpoints.

The DAEC IPE considers a full spectrum of severe accidents that have been postulated and which may challenge accident management actions in unique or special ways. These unique challenges include:

- Core melt progression with containment intact
  - At high RPV pressure
  - At low RPV pressure
  - With and without adequate reactivity insertion
- Core melt progression with the containment breached or not isolated

The containment performance discussion can be usefully divided among the different time phases addressed in the Level 1 and Level 2 analysis:

- Before core damage

- In-vessel core degradation
- Ex-vessel core melt progression

Each of these phases and the corresponding containment performance is discussed as follows:

Before core damage

This phase is covered in the Level 1 IPE and the containment performance can be assessed for the following types of challenges:

- Slow containment overpressure failure (Class IIT)
- Rapid drywell pressurization (Class IIID)
  - RPV Rupture
  - Vapor Suppression Failure
- Rapid energy disposition to the pool (i.e., ATWS)
- Containment isolation failure (treated in Level 2 IPE)
- Containment bypass (Class V)
- Venting (Class IIV).

In-vessel Core Degradation Phase

- In-vessel steam explosion
- H_2 Deflagration
- Venting

Ex-vessel Core Melt Progression

- Overpressurization due to decay heat and non-condensible gas generation
- Ex-vessel steam explosion
- Direct containment heating
- H_2 deflagration
- Vapor suppression failure
- Overtemperature failure

The DAEC deterministic MAAP calculations provide the technical baseline for the determination of:

- Success criteria and plant response at each node
- Containment survivability under the postulated severe accident

- The source terms

- Postulated sensitivities.

It is useful to indicate the general trends that can be anticipated for the different postulated accident sequences that may be encountered in the evaluation of accident management actions under severe accident conditions. Therefore, a small sample of the calculated MAAP responses used in the characterization of the DAEC containment are included here for reference.

Included in this section are MAAP calculated containment response pressures and temperatures. These conditions can be compared with the ultimate containment capability to ascertain the status of containment (see Figure 4.6-1).

The representative severe accidents that are discussed here for example include the following:

- Class IA: Loss of adequate makeup at high RPV pressure with the containment initially intact

- Class ID: Loss of adequate makeup at low RPV pressure with the containment initially intact

- Class II: Loss of adequate containment heat removal

- Class IV: ATWS event with containment failure preceding the occurrence of core damage.

Table 4.6-1 provides the designators for these sequences. Many other postulated accident scenarios are performed to determine the variations in timing associated with change in the sequence (see Section 4.7 for a summary of MAAP runs). These examples are only shown for illustration; numerous other MAAP runs are used as part of the DAEC IPE to characterize plant response and radionuclide release. The following is a brief discussion of each representative sequence:

Table 4.6-1

Example Representative Accident Sequences

MAAP Case	Accident Class	Description	Figure Numbers
LII-IA-1	IA	- MSIV closure initiator - No HPCI, RCIC, CRD or LPCI - ADS inhibited - High pressure core melt - RPV breach - Only CS available for injection to debris - Containment pressure remains below venting pressure	4.6.1-2(Pressure) 4.6.1-3 (Temperature)
LII-ID-7	ID	- Core melt at low RPV pressure - No injection available - No containment venting - No RHR available - Drywell failure induced by high pressure and temperature	4.6.1-4 (Pressure) 4.6.1-5 (Temperature)
LII-2L-1	II	- Loss of containment heat removal - Large LOCA - ADS inhibited - No feedwater - No high pressure injection available	4.6.1-6 (pressure) 4.6.1-7 (Temperature)
LII-4A-1	IV	- Failure to scram - No effective boron injection - Inadequate containment heat removal - Failure of core makeup after containment fails	4.6.1-8 (Pressure) 4.6.1-9 (Temperature)

Class IA: Loss of Makeup at High RPV Pressure with the Containment Initially Intact

The key events for this sequence can be summarized as follows:

Event	Timing (Hrs.)
Core Uncovered	0.69
Initiation of Core Damage	1.06
Initiation of Core Melt	1.28
RPV Failure/Breach	3.38
Containment Failure Location: N/A Size: N/A	Containment Intact
Radionuclide Release Magnitude: Negligible	N/A

Figures 4.6-2 and 4.6-3 demonstrate that a high pressure core melt sequence (e.g., TQUX) can result in high containment pressure and temperature spikes at approximately 3.4 hours into the accident. These spikes will not in and of themselves cause containment failure. Following these temperature and pressure spikes, the containment temperature and pressure increase slowly up to twenty hours into the accident and then start to level off. Therefore, containment challenge does not seem to be an issue for this accident scenario.

Class ID: Loss of Adequate RPV Makeup at Low RPV Pressure

The key events in this postulated sequence involves the plant response when the RPV has been successfully depressurized, but no injection is available to the RPV.

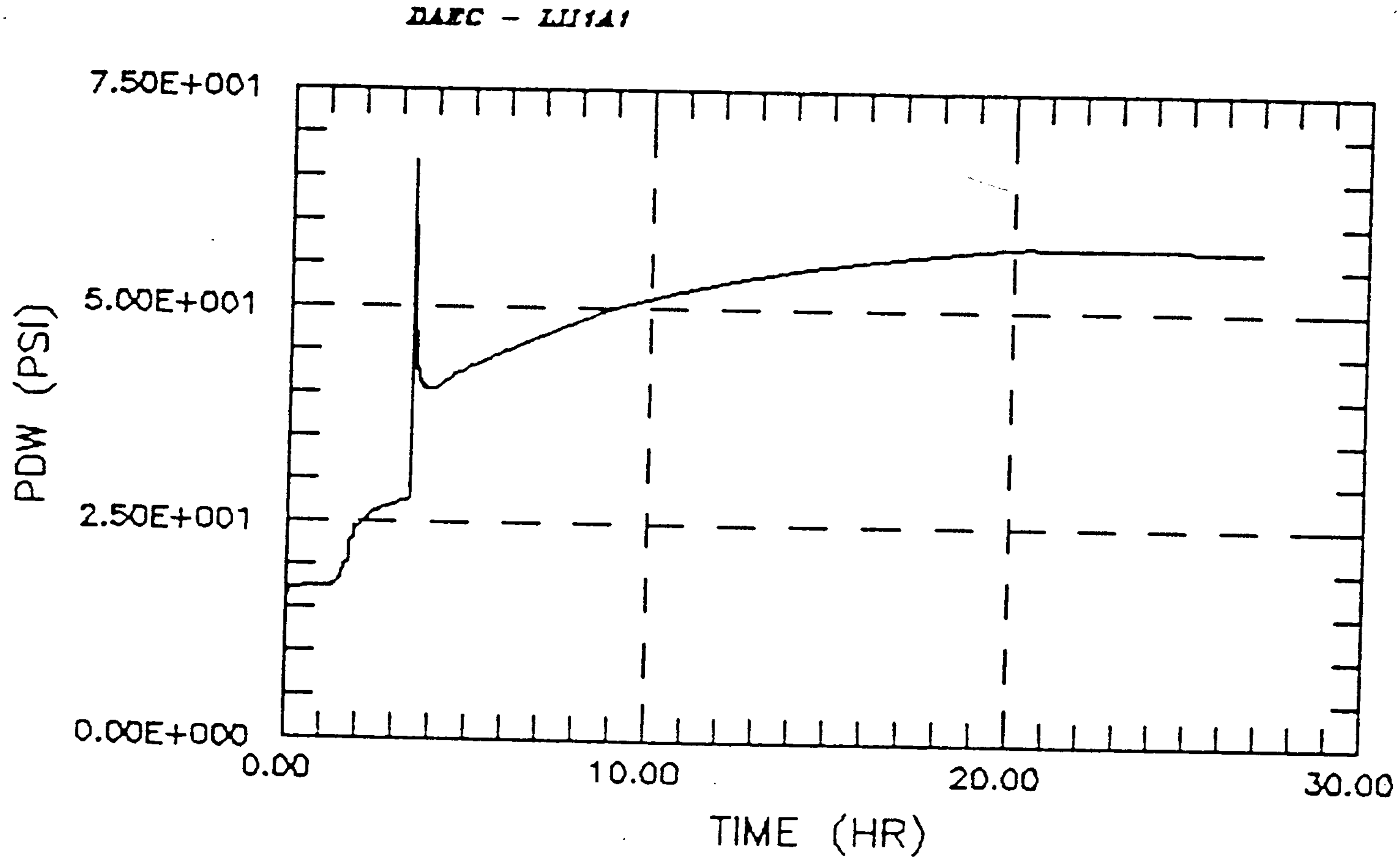


Figure 4.6-2

Sequence LII-IA-1, Drywell Pressure Trace

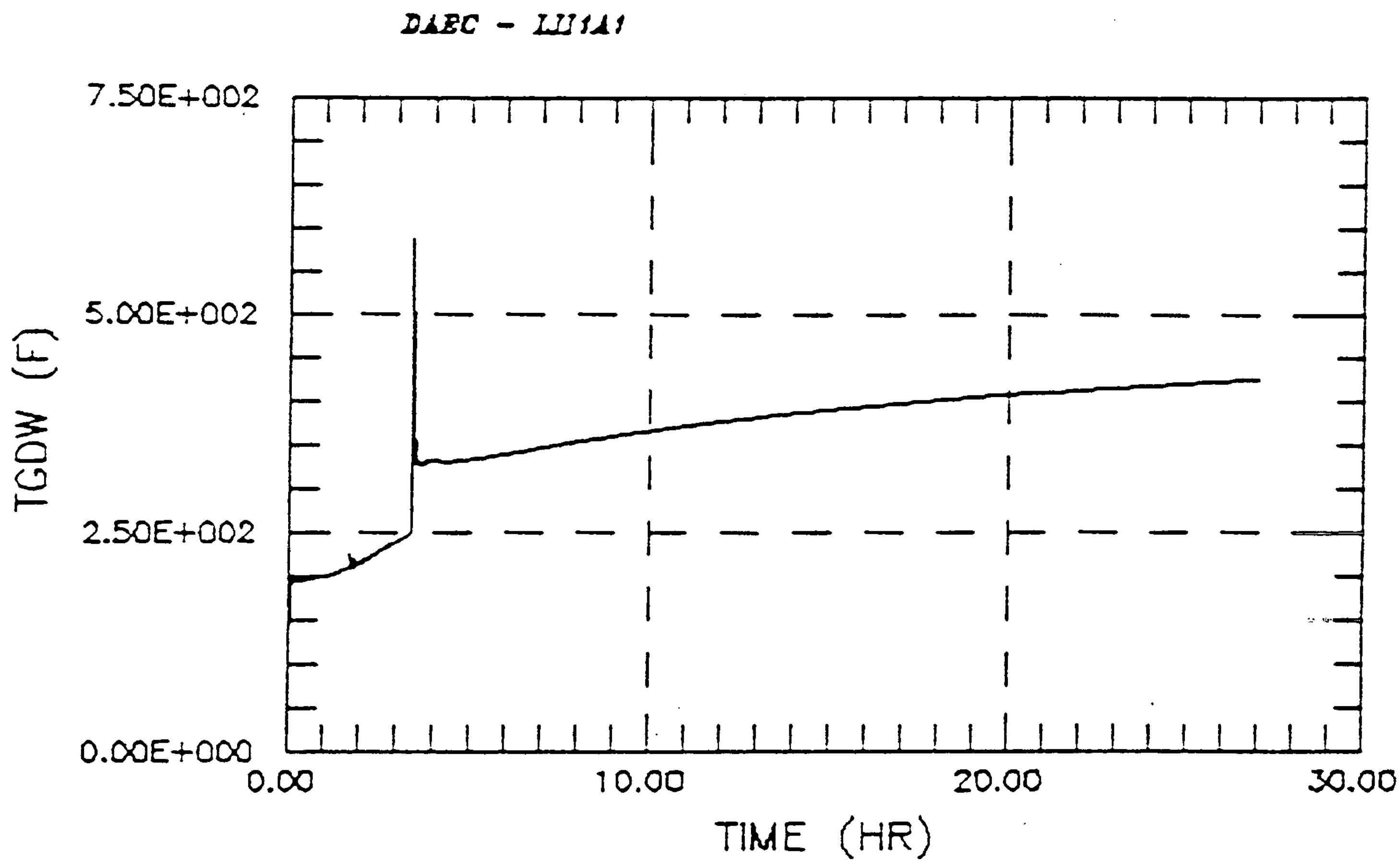


Figure 4.6-3

Sequence LII-1A-1, Drywell Temperature Trace

Event	Timing (Hrs.)
RPV Depressurization	0.59
Core Uncovered	0.62
Initiation of Core Damage	0.64
Initiation of Core Melt	1.39
RPV Failure/Breach	1.91
Containment Failure Location: DW Size: Large	~27
Radionuclide Release Magnitude: Moderate	27/

Figures 4.6-4 and 4.6-5 provide the drywell pressure and temperature traces for the case in which the RPV fails at low pressure and no injection or heat removal capability exists. A pressure spike occurs at the time of RPV failure (approximately 2 hours into the accident), but is less than that for a high RPV pressure blowdown. Over the next twenty hours the containment pressure rises with containment temperature until the primary containment fails at about 27 hours into the accident.

Class II: Loss of Adequate Containment Heat Removal

This accident sequence involves core damage only after containment failure occurs. This is substantially different in timing and response from the Class I sequences. Key events for this sequence are shown below.

Event	Timing (Hrs.)
Core Uncovered	0.02
Initiation of Core Melt	26.5
RPV Failure/Breach	28.9
Containment Failure Location: DW Head Size: Large	25.0
Radionuclide Release Magnitude: High	28.9

Figures 4.6-6 and 4.6-7 provide the drywell pressure and temperature for a sequence in which containment heat removal is postulated to fail, but coolant injection to the RPV remain available until containment failure occurs. This sequence is similar to the WASH-1400 "TW" sequence. The containment failure occurs at relatively low containment temperatures at a time of approximately 25 hours after scram and loss of containment heat removal. This represents an exceedingly long time.

Class III: LOCAs with Inadequate Makeup

No example sequences are presented. The MAAP results for DAEC demonstrate that the characteristics are similar to those of Class I for similar system availability cases.

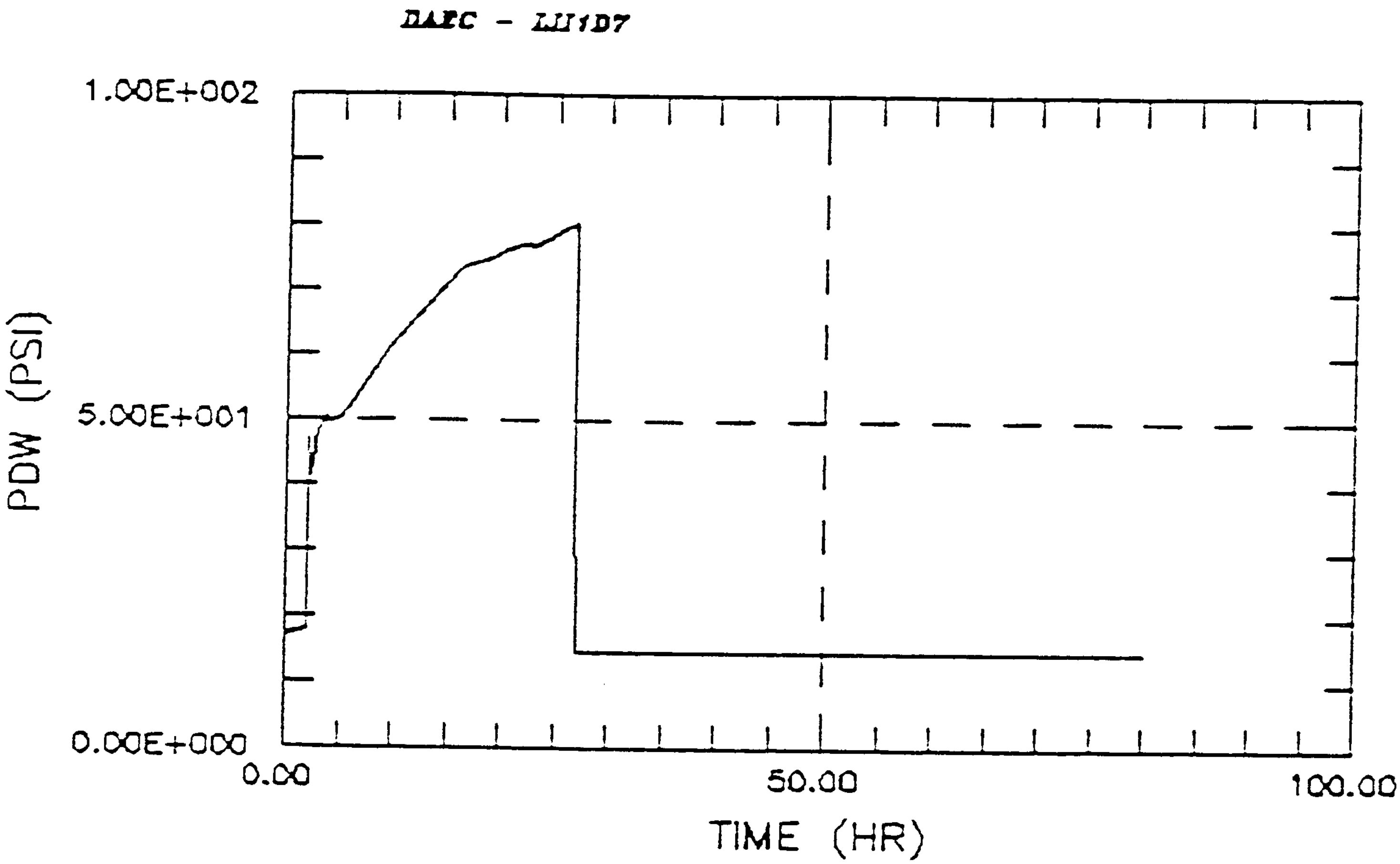


Figure 4.6-4

Sequence LII-ID-7, Drywell Pressure Trace

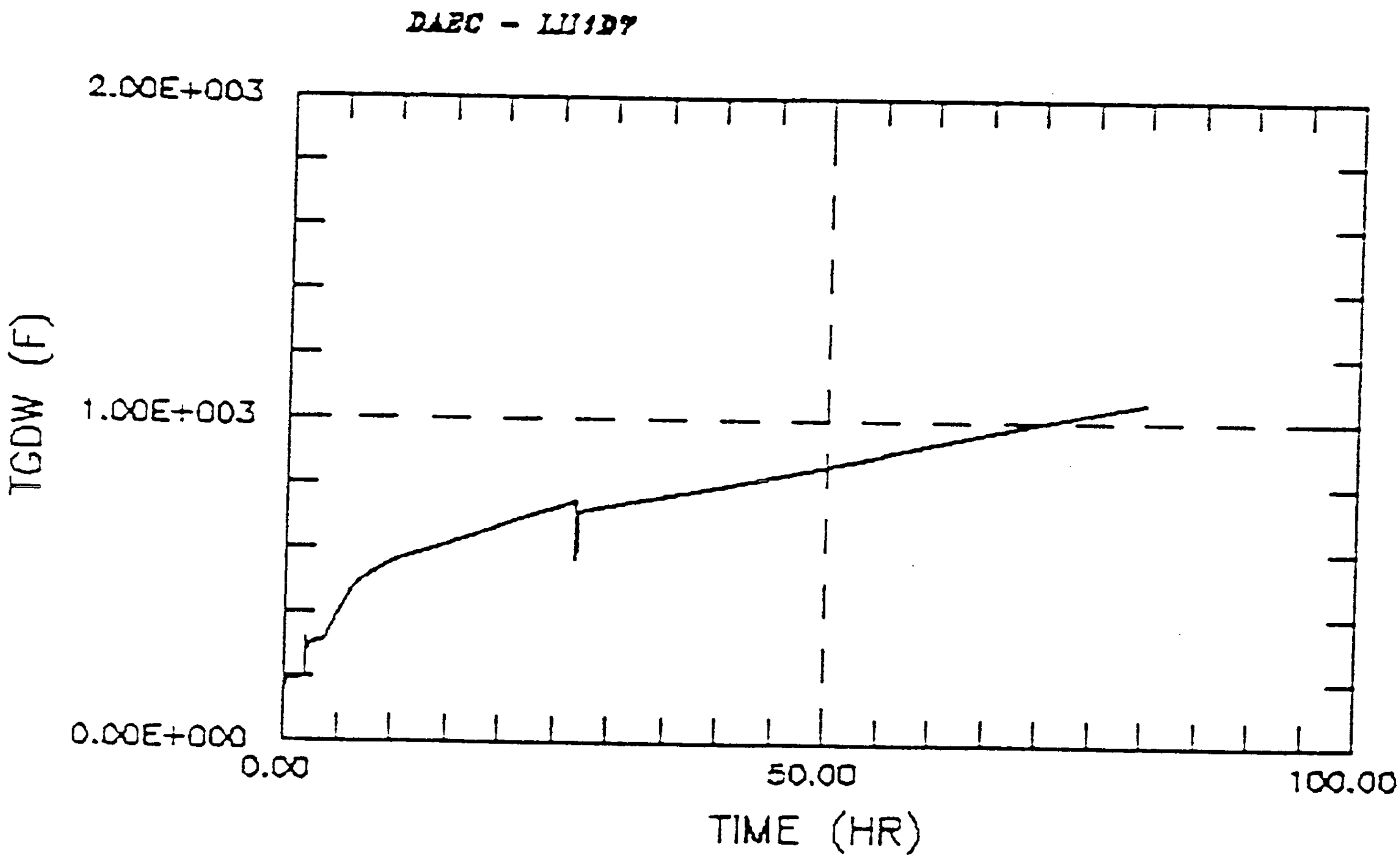


Figure 4.6-5

Sequence LII-ID-7, Drywell Temperature Trace

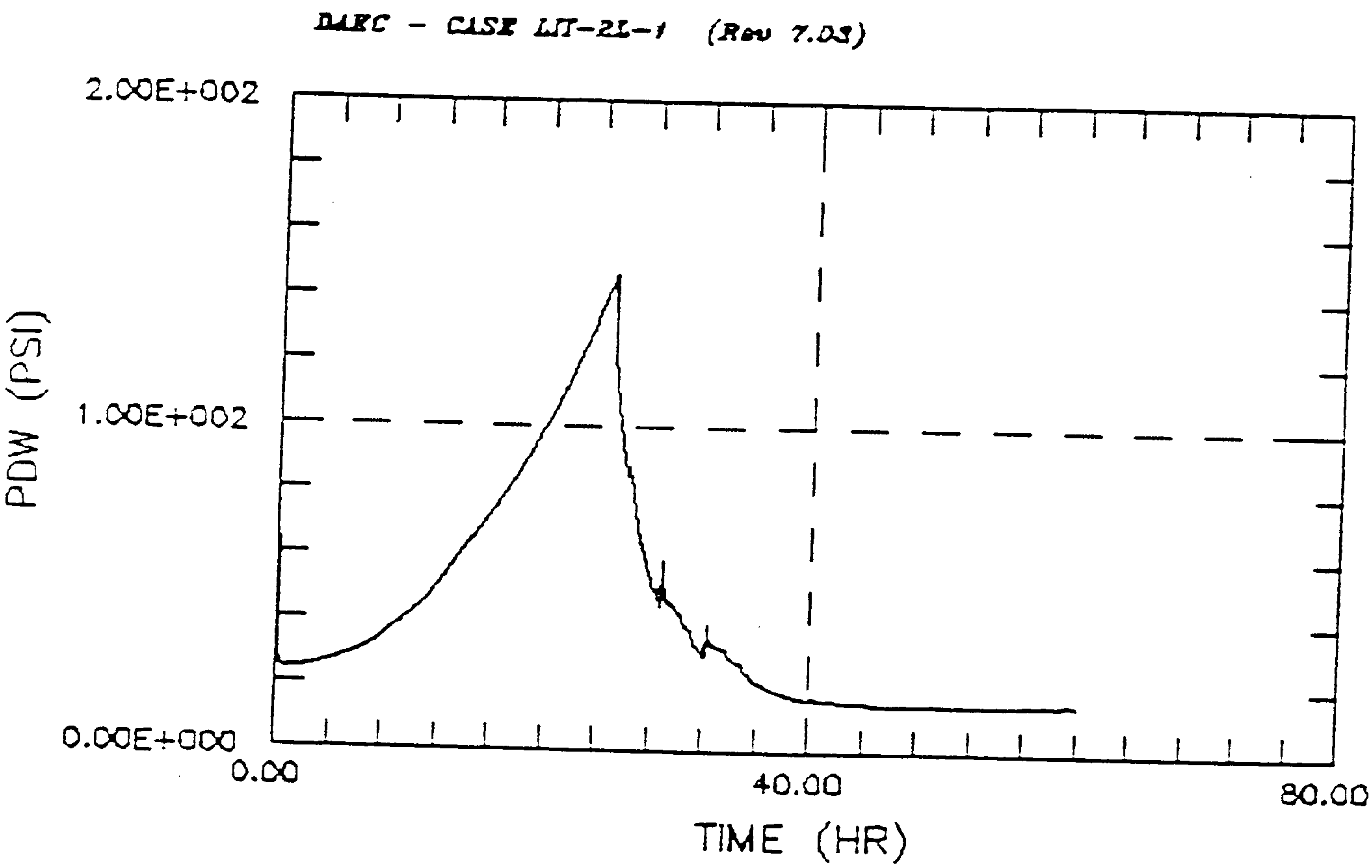


Figure 4.6-6

Sequence LII-2L-1, Drywell Pressure Trace

Duane Arnold Energy Center Individual Plant Examination

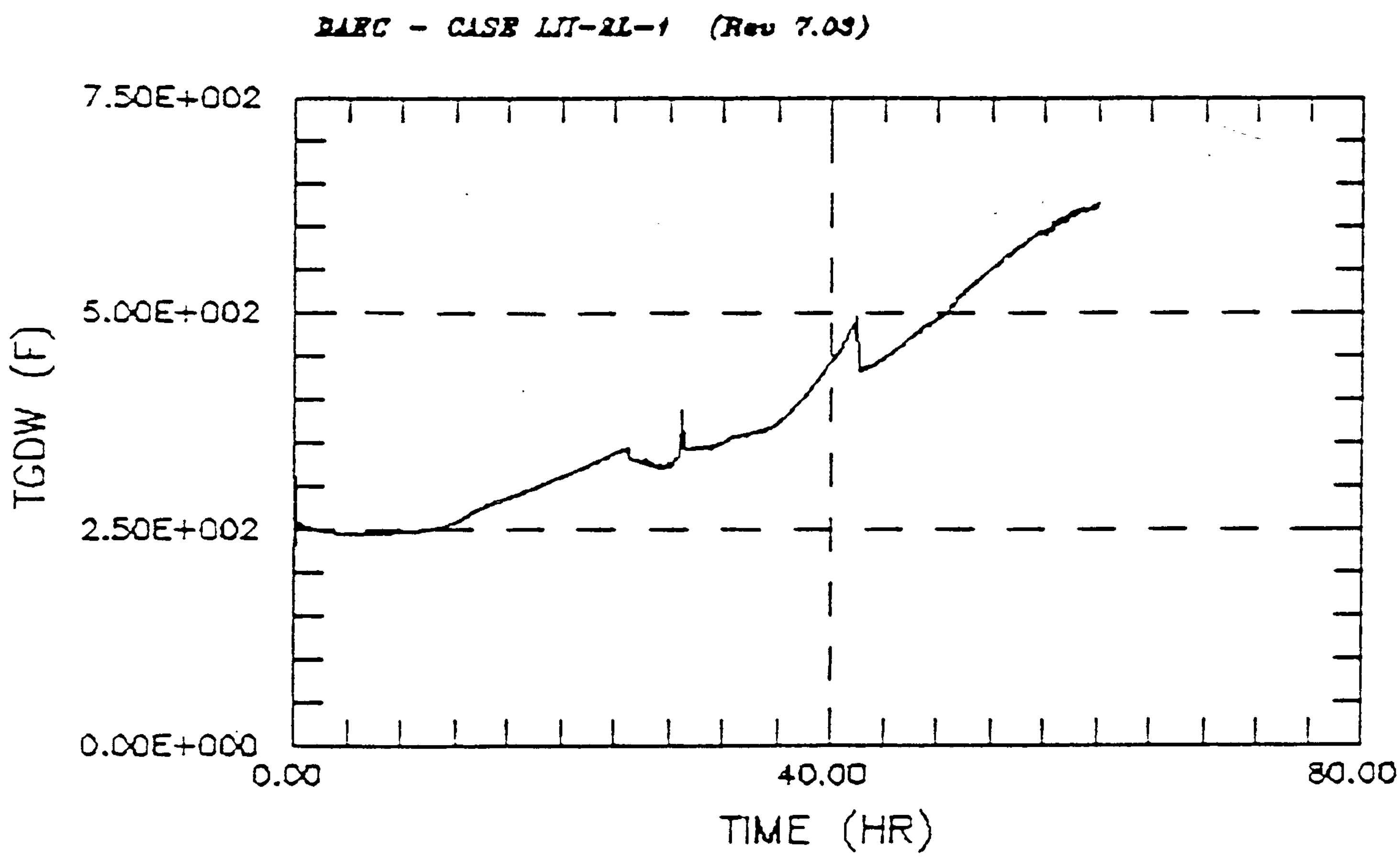


Figure 4.6-7

Sequence LII-2L-1, Drywell Temperature Trace

Duane Arnold Energy Center Individual Plant Examination

Class IV: ATWS Induced Containment Failure Followed by Core Damage

In this accident sequence containment failure is induced by a rapid increase in containment pressure which precedes core damage. The key events for this postulated scenario is:

Event	Timing (Hrs.)
Core Uncovered	0.12
Initiation of Core Damage	0.99
Initiation of Core Melt	1.39
RPV Failure/Breach	4.20
Containment Failure Location: DW Head Size: Large	1.0
Radionuclide Release Magnitude: Moderate	1.0

Figures 4.6-8 and 4.6-9 provide the containment drywell pressure and temperature traces for a postulated ATWS. For this "worst case" scenario containment failure occurs "early", i.e., in the 1 to 2 hour time frame, and core damage follows soon after.

4.6.2 Quantification Process

The quantification of the Level 2 IPE model merges all the deterministic thermal hydraulic calculations, the postulated containment failure modes, the assessment of the containment ultimate strength, the assessment of mitigation and the probabilistic

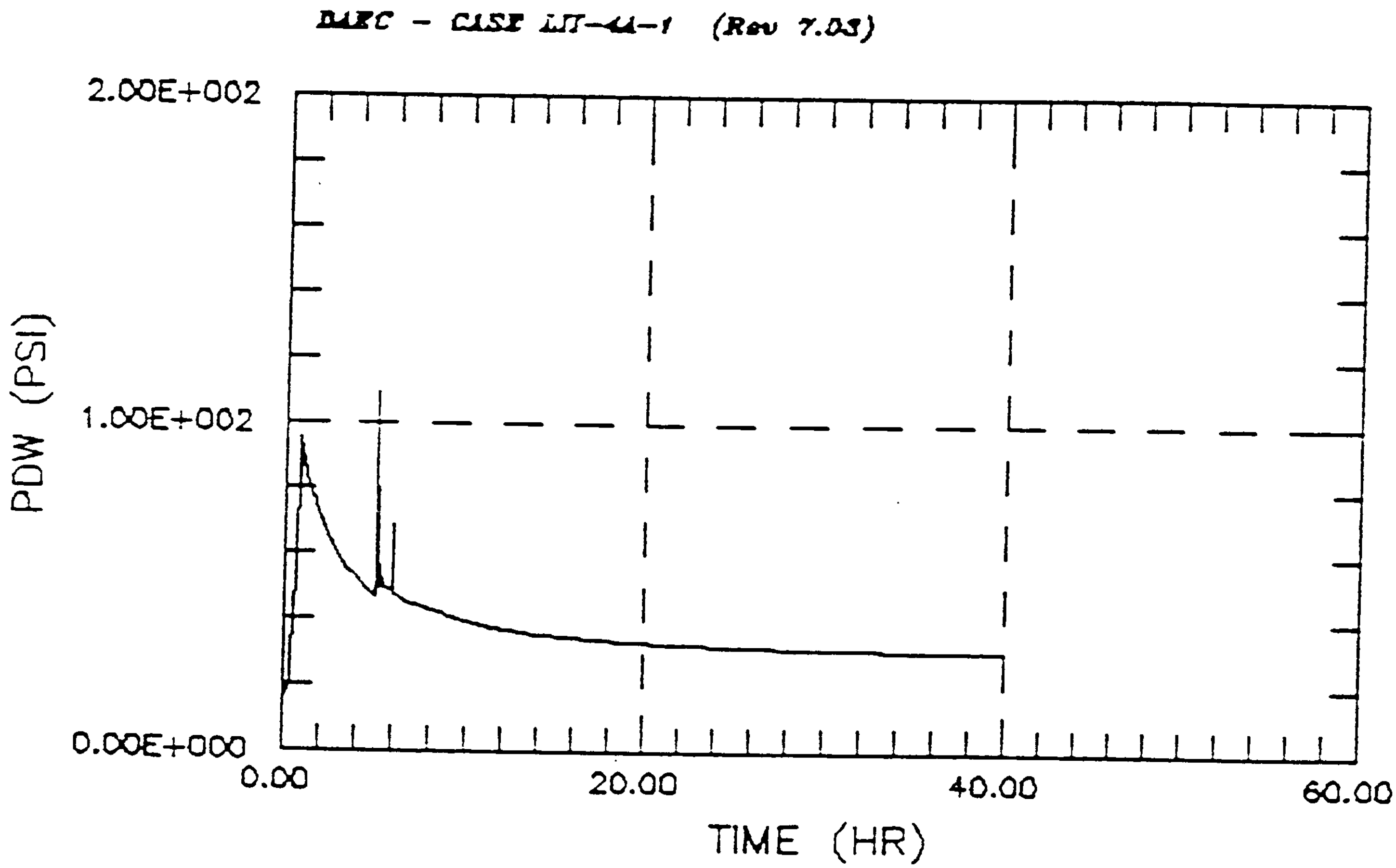


Figure 4.6-8

Sequence LII-4A-1, Drywell Pressure Trace

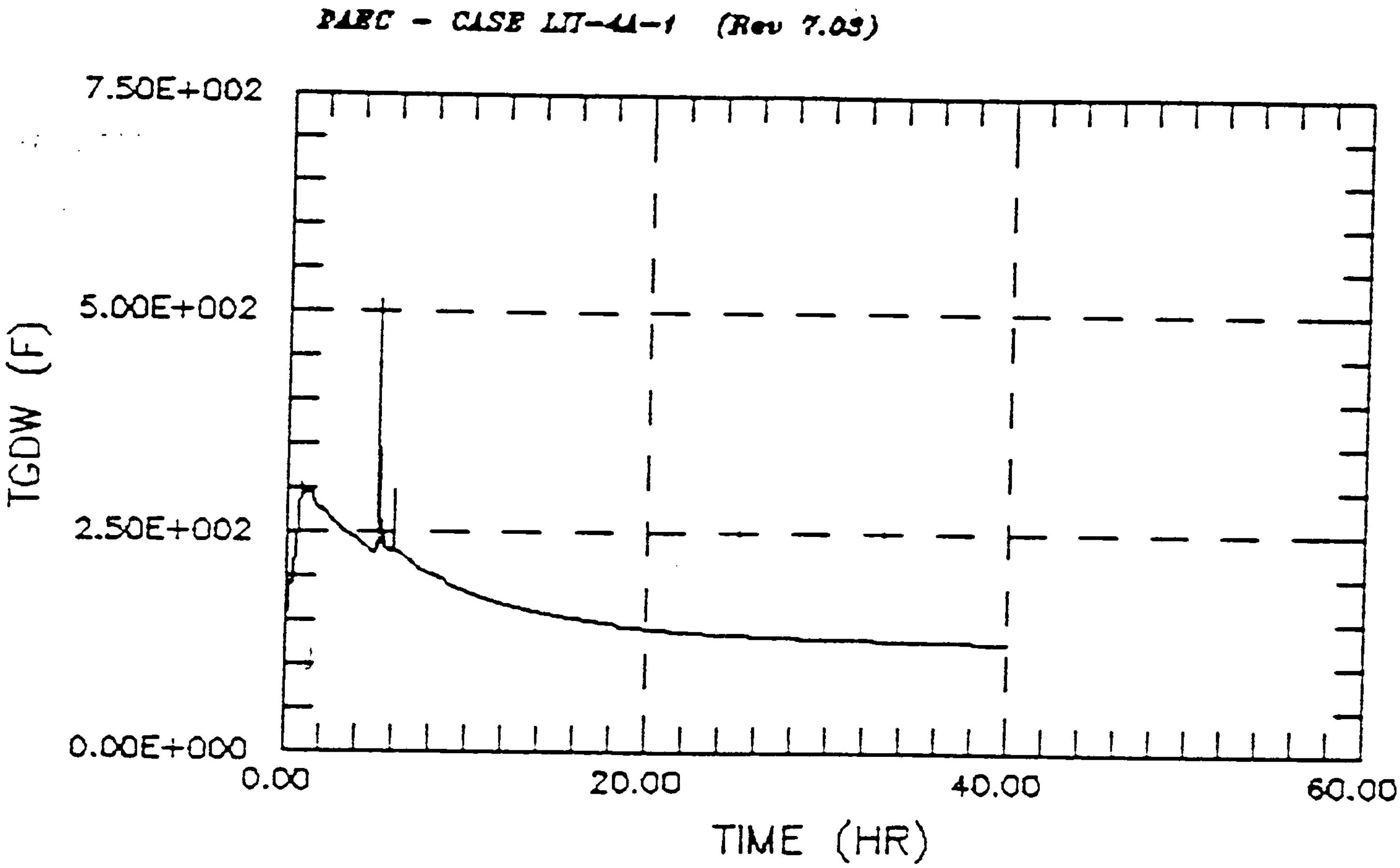


Figure 4.6-9

Sequence LII-4A-1, Drywell Temperature Trace assessment of the likelihood of each. This subsection provides an overview of the main elements included in the quantification process.

4.6.2.1 Postulated Containment Failure Modes

A comprehensive list of containment failure mechanisms is developed and presented in Section 4.4. The CET was used to structure these failure mechanisms so they could be probabilistically assessed given the severe accident challenges which are determined from the Level 1 analysis and considering the recovery and mitigation in the Level 2 analysis.

This process resulted in the identification of the most probable potential containment failure mechanisms for DAEC.

4.6.2.2 Containment Ultimate Strength

As described in Section 4.4 the ultimate containment capability for the spectrum of severe accidents is determined. This ultimate capability is then overlayed on top of the containment pressure and temperature response determined (see 4.6.1) to assess the probability of containment failure. The ultimate containment capability is based on a plant specific assessment by CB&I for static loading and a separate effects analysis for dynamic loads.

4.6.2.3 Equipment Survivability

The quantification process includes an examination of the impact of severe accident conditions on equipment required for accident prevention and mitigation. However, formal environmental qualification requirements are not applicable to the IPE and accident management process. When credit is taken for equipment in severe accidents, an assessment is made of the ability of the equipment to perform the function for a specific period of time considering exposure to temperature, pressure, aerosol loading, radiation, and moisture. The degree of credit is based upon review of studies concerning the capacity of equipment to survive or operate in various environments. If the available data do not cover the range of conditions expected during a severe accident, then the data are extrapolated. The DAEC IPE considers the survivability/operability of equipment, systems, structures relied upon in a severe accident relying principally upon engineering judgment coupled with some limited data.

Research studies and tests of equipment survivability were reviewed for the following components:

- Cables
- Electrical penetration assemblies
- Electrical connections
- Solenoid valves
- Motor-operated valves
- Motor-driven pumps
- MCCs.

In general, components located in the reactor building have a fairly high reliability rate. The reactor building is estimated to experience temperatures of a couple of hundred degrees in worst cases, and most components can survive in this type of environment for tens of hours. Cable connections (specifically terminal blocks) appear to be the weakest links, exhibiting high failure rates in steam environments of approximately 200°F. However, no critical terminal blocks are considered to be present in the DAEC containment.

Susceptibility of individual components was not modeled; for example, injection systems were grouped into a single basic event that considered failure due to harsh environment. Due to like components among systems, the assumption was made that if components failed in one system due to harsh environment then so did components in the other injection systems.

4.6.2.4 Containment Isolation

Consistent with NUREG-1335, containment isolation is modeled as the first node in the containment event tree. The modeling of containment isolation is based on a fault tree model. The fault tree for containment isolation incorporates modeling of containment hatches and large lines that penetrate the containment and open to the containment atmosphere (e.g., purge and vent lines). The fault tree considers automatic isolation signals, pre-existing open pathways, manual isolation, and component failures.

Any failure of containment isolation is modeled as a large failure (2 $ft.^2$) in the drywell. Containment isolation failure is conservatively characterized as a high radionuclide release at the time of initial core damage (i.e., H/E release categorization).

4.6.2.5 Human Intervention

The Level 2 PRA considers important human interaction events that can affect containment performance and radionuclide release frequency, magnitude, or timing, and establish the risk profile of DAEC. Therefore, it is necessary to consider the human tasks that are performed under normal operating conditions, and those actions performed in response to accidents or abnormal occurrences. These are actions that may occur during the course of an accident as the operator interprets the incoming diagnostic information and implements the task determined to be appropriate. However, whether during normal operation or during responses to an accident situation, only human errors, defined as mistakes in the performance of assigned tasks, are modeled. It is assumed that any intentional deviation from the operating procedures is made because of misdiagnosis or misleading indication for which the operators believe their method of operation to be safer or more efficient.

The Level 2 analysis incorporates the consideration of operator actions. In general, the actions considered in the analysis are confined to those that are proceduralized (i.e., actions directed by current EOPs), although, occasionally operator actions are included that are not explicitly directed by the EOPs. Refer to Table 4.6-2 for a list of types of operator actions included in the Level 2 analysis and their associated procedures (a list of specific operator action basic events is not provided here). Those actions that are EOP-directed are quantified considering the operators are trained in their implementation. The quantification method for operator actions is the same as that employed in the Level 1 analysis.

Table 4.6-2

PROCEDURE-BASED HUMAN INTERVENTION INCLUDED IN LEVEL 2 ANALYSIS

Class of Operator Action	Procedure
Isolate Primary Containment Pathway Given Failure of Automatic Isolation	Primary Containment Control, Step PC/H specifies executing procedure for Containment Isolation
RPV Depressurization	ED-Emergency Depressurization
Injection Recovery	EOPs do not specify system recovery, although the implications to do so can be inferred
Offsite Power Recovery	Station Blackout Procedure
EDG Recovery	Station Blackout Procedure
Combustible Gas Vent	Primary Containment Control, Steps PC/H and PC/P specify executing emergency vent procedure
Containment Flooding	Primary Containment Flooding
Containment Venting	Primary Containment Control, Step PC/P specifies executing emergency vent
Manual Alignment of Alternate Injection Systems	Alternate Level Control Procedure
RPV Venting	Primary Containment Flooding

4.6.3 Quantification Results

This section includes the following summaries:

- The quantification of the plant damage states from the Level 1 PRA for input to the CET

- The output radionuclide release frequencies from the CET quantification for the baseline evaluation

- Graphical comparisons of the radionuclide release magnitudes and timing, including their major contributors.

4.6.3.1 Input

Table 4.6-3 summarizes the core damage frequency contributions from the DAEC Level 1 PRA (1992) by subclass which in turn provides the input to the Level 2 containment evaluation.

As discussed earlier, the different accident types represent substantially different challenges to containment, containment mitigating systems, and the operating staff. Therefore, each of these accident subclasses has been treated separately in the containment event tree evaluation.

This treatment consists of:

- Using the CET structure that best describes the chronology of events

- Including the appropriate dependencies as a function of the sequence type.

4.6.3.2 Output Summary

Table 4.6-3 summarizes the following results:

- <u>Radionuclide Release End States</u>: The release categories used to discriminate among the CET end states are identified.

Table 4.6-3

SUMMARY OF THE CORE DAMAGE ACCIDENT SEQUENCE SUBCLASSES

Accident Class Designator	Subclass	Definition	Level 1 Frequency (per Rx Yr)
Class I	A	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high.	4.14E-7
	B	Accident sequences involving a station blackout and loss of coolant inventory makeup.	1.92E-6
	C	Accident sequences involving a loss of coolant inventory induced by an ATWS sequence with containment intact.	1.49E-7
	D	Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced to 200 psi.	5.18E-7
	E	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high and DC power is unavailable.	1.01E-6
Class II	A	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage induced post containment failure.	---
	L	Accident sequences involving a loss of containment heat removal with the RPV breached but no initial core damage; core damage induced post containment failure.	2.64E-7
	T	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage induced post high containment pressure.	1.64E-6
	V	Class IIA or IIL except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	---
Class III (LOCA)	A	Accident sequences leading to core damage conditions initiated by vessel rupture where the containment integrity is not breached in the initial time phase of the accident.	< 3E-11

Table 4.6-3

SUMMARY OF THE CORE DAMAGE ACCIDENT SEQUENCE SUBCLASSES

Accident Class Designator	Subclass	Definition	Level 1 Frequency (per Rx Yr)
Class III (LOCA) (con't)	B	Accident sequences initiated or resulting in small or medium LOCAs for which the reactor cannot be depressurized prior to core damage occurring.	2.34E-10
	C	Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is a low pressure and no effective injection is available.	2.62E-8
	D	Accident sequences which are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate, challenging the containment integrity with subsequent failure of makeup systems.	1.35E-7
Class IV (ATWS)	A	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post containment failure.	1.68E-6
	L	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially breached (e.g., LOCA or SORV); core damage induced post containment failure.	8.48E-8
	T	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially intact, core damage induced post high containment pressure	---
	V	Class IVA or IVL except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	---
Class V	--	Unisolated LOCA outside containment	< 3E-11

Table 4.6-4

SUMMARY OF CONTAINMENT EVALUATION

INPUT			OUTPUT
DAEC PRA LEVEL 1		CET EVALUATION	
Core Damage Frequency	Characterize Release	Release Bin	Release Frequency (per Year)
7.84E-6/yr	Leakage or No Release	OK	1.68E-6
	Low Public Risk Impact	LL & Late	3.26E-7
		LL & I	2.60E-9
		LL & E	1.78E-7
		L & Late	8.62E-7
		M & Late	2.20E-7
		L & I	2.27E-8
		L & E	1.37E-6
	Moderate Release	M & I	4.51E-7
		M & E	1.61E-6
	High Release	H & Late	5.00E-7
		H & I	1.07E-7
		H & E	5.02E-7

- Output: The output frequencies of the CETs as a function of the end state bins are identified.

The individual accident class contributors to the radionuclide release frequency may also provide insights into the containment performance as a function of the type of severe accident. The contributors to each of the radionuclide release end states can be broken down by the type of accident class from the Level 1 analysis as shown in Table 4.6-5.

The quantification provides a yardstick to measure the best estimate of containment performance given that severe accidents could progress to beyond core damage. The quantification may include some conservatisms to account for the inability of current models and experiments to predict certain severe accident related phenomena.

A significant fraction (21%) of the accidents transferred from the Level 1 PRA are substantially mitigated such that releases are contained within an intact containment (i.e., No Release bin). Whereas, only 6% of the postulated severe accidents have "large" releases occurring before protective action can be taken.

Table 4.6-5 shows that the largest contributors to the worst release category, high (H) release magnitude and early release (E) following initiation, are Classes IA, IE, IB, and IIID.

4.6.3.3 Examination of the Baseline Quantified Results of the DAEC CET

Different organizations may have different opinions on what are the most important issues related to the protection of the public health and safety. For example, Federal Emergency Management Administration (FEMA) may consider the understanding, prevention, and mitigation of accidents that could result in changing evacuation plans or evacuation effectiveness as the key issue. For such determinations both the magnitude and timing of the accident sequence is of importance. On the other hand, some organizations may consider latent health effects to be the dominant contributor to public risk and therefore the magnitude of the release is of principal importance regardless of the timing. To account for these different viewpoints, the radionuclide

Table 4.6-5

SUMMARY TABLE OF RELEASE VS. ACCIDENT CLASS

Class[1]	NO RLEASE	LL/E	LL/I	LL/L	L/E	L/I	L/L	M/E	M/I	M/L	H/E	H/I	H/L	Total Release (per/yr)	Total
IA	1.73E-7	1.54E-9	6.29E-13	7.25E-11	1.28E-12	6.18E-10	---	1.49E-7	3.91E-11	---	8.93E-8	3.52E-12	---	2.41E-7	4.14E-7
IB	7.46E-7	5.05E-9	8.21E-10	6.24E-10	4.07E-12	2.37E-9	1.39E-12	4.89E-7	4.41E-7	1.28E-12	1.37E-7	9.58E-8	---	1.17E-6	1.92E-6
IC	7.72E-8	6.81E-10	4.77E-13	5.84E-11	---	3.67E-10	---	5.77E-8	2.41E-11	---	1.30E-8	1.65E-12	---	7.18E-8	1.49E-7
ID	2.26E-7	2.62E-9	---	3.95E-9	---	1.73E-8	6.84E-11	1.82E-7	9.50E-9	8.18E-11	3.04E-8	8.24E-9	---	2.54E-7	5.16E-7
IE	4.02E-7	3.33E-9	4.39E-12	3.12E-10	1.47E-12	2.00E-9	5.19E-12	4.82E-7	1.41E-10	6.35E-12	1.21E-7	1.97E-11	---	6.09E-7	1.01E-6
IIL	---	---	---	---	---	---	---	---	---	4.46E-8	---	---	2.19E-7	2.64E-7	2.64E-7
IIT	---	---	---	3.21E-7	---	---	8.62E-7	---	---	1.75E-7	---	---	2.81E-7	1.64E-6	1.64E-6
IIIB	1.23E-8	---	---	---	---	---	---	8.96E-11	---	---	2.11E-11	---	---	1.11E-10	2.34E-10
IIIC	2.44E-8	---	4.11E-14	---	---	1.27E-10	9.22E-15	7.60E-10	1.22E-12	5.42E-15	8.91E-10	2.65E-15	---	1.78E-9	2.62E-8
IIID	---	---	---	---	---	---	---	3.00E-8	---	---	1.05E-7	---	---	1.35E-7	1.35E-7
IVA	---	1.57E-7	---	---	1.31E-6	---	---	2.10E-7	---	---	5.22E-9	---	---	1.68E-6	1.68E-6
IVL	---	7.00E-9	---	---	6.50E-8	---	---	1.24E-8	---	---	3.55E-10	---	---	8.48E-8	8.48E-8
V[2]	---	---	---	---	---	---	---	---	---	---	---	---	---	0.00E+0	0.00+0

[1] Classes are defined in Table 4.6-3.

[2] All sequences truncated in Level I analysis (3.0E-11).

release binning is summarized in different ways such that various organizations can make the most effective use of the information for their specific purpose. Therefore, this subsection examines the quantitative results of the DAEC Level 1 and Level 2 PRA evaluations from a number of different viewpoints.

4.6.3.3.1 Plant Damage States

The input to the Level 2 PRA evaluation comes from the output of the Level 1 DAEC PRA. Each of the accident subclasses represents different challenges to containment and therefore will have different impacts on public safety. The characteristics of the dominant contributing classes can be summarized as follows:

DOMINANT CORE DAMAGE FREQUENCY CONTRIBUTOR PLANT DAMAGE STATES FROM LEVEL 1 PRA		
Accident Class	% of Core Damage Frequency	Characteristic
IB	24%	Station blackout with inability to supply adequate makeup. No AC power available at the time of core damage initiation.
IVA	21%	Failure of adequate reactivity control results in overpressure failure of containment before core melt.
IIT	21%	Accident sequences involving loss of containment heat removal in which the containment fails prior to vessel failure.

The impact of these subclasses on public safety are summarized in the following subsections.

4.6.3.3.2 Radionuclide Release Magnitude Frequency

The frequency of radionuclide release is characterized by the quantification of the Level 1 and Level 2 PRA models. The Level 2 containment event tree end states are further delineated by the magnitude and timing of the calculated radionuclide release. Using the end state release magnitude and timing, a comparison can be developed to identify the overall frequency of the various end state release magnitudes, from very low to high.

See Sections 4.7.4.1 and 4.7.4.2 for the two term matrix defining release magnitude and timing.

Figure 4.6-10 summarizes in bar-graph form a comparison of the total core damage frequency (i.e., the results of the Level 1 IPE) with the end state frequencies of the Level 2 analysis, i.e., High (H), Moderate (M), Low (L) and Low-low (LL) release magnitudes plus those severe accident sequences that result in an intact containment (OK). A substantial fraction (55%) of the core damage end states are either of; (1) low release; or (2) the containment remains intact.

These results can also be plotted in a pie-chart format to show the relative contributions from the various Level 2 release magnitude end states (see Figure 4.6-11).

4.6.3.3.3 Timing of Radionuclide Releases

Another parameter in the evaluation of the impact of radionuclide releases is related to the timing of the release. This parameter is of importance to identify the time available for:

- Accident management response actions

- Public safety measures, such as sheltering or evacuation.

Figures 4.6-12 and 4.6-13 summarize the frequency of radionuclide release for the DAEC accident analysis as a function of the timing of the radionuclide release initiation.

Table 4.6-6

RELEASE SEVERITY AND TIMING CLASSIFICATION SCHEME
(SEVERITY, TIMING)

Release Severity Source Term Release Fraction		Release Timing	
Classification Category	Cs Iodide % in Release	Classification Category	Time of Release[†] (noble gases or CsI)
High (H)	greater than 10	Late (L)	greater than 24 hours
Moderate (M)	1 to 10	Intermediate (I)	6 to 24 hours
Low (L)	0.1 to 1	Early (E)	less than 6 hours
Low-low (LL)	less than 0.1		
No iodine (No Release)	0		

The three categories of timing used in the end state quantification are as follows:

- Early: Releases are initiated within 6 hours of the accident initiation.

- Intermediate: Releases are initiated between 6 hours and 24 hours after accident initiation.

- Late: Releases are initiated more than 24 hours after the accident initiation.

[†]Time relative to initiating event; closely related to exceeding the Emergency Action Level (EAL) for General Emergency for the worst case accidents analyzed in the Level 2 PRA.

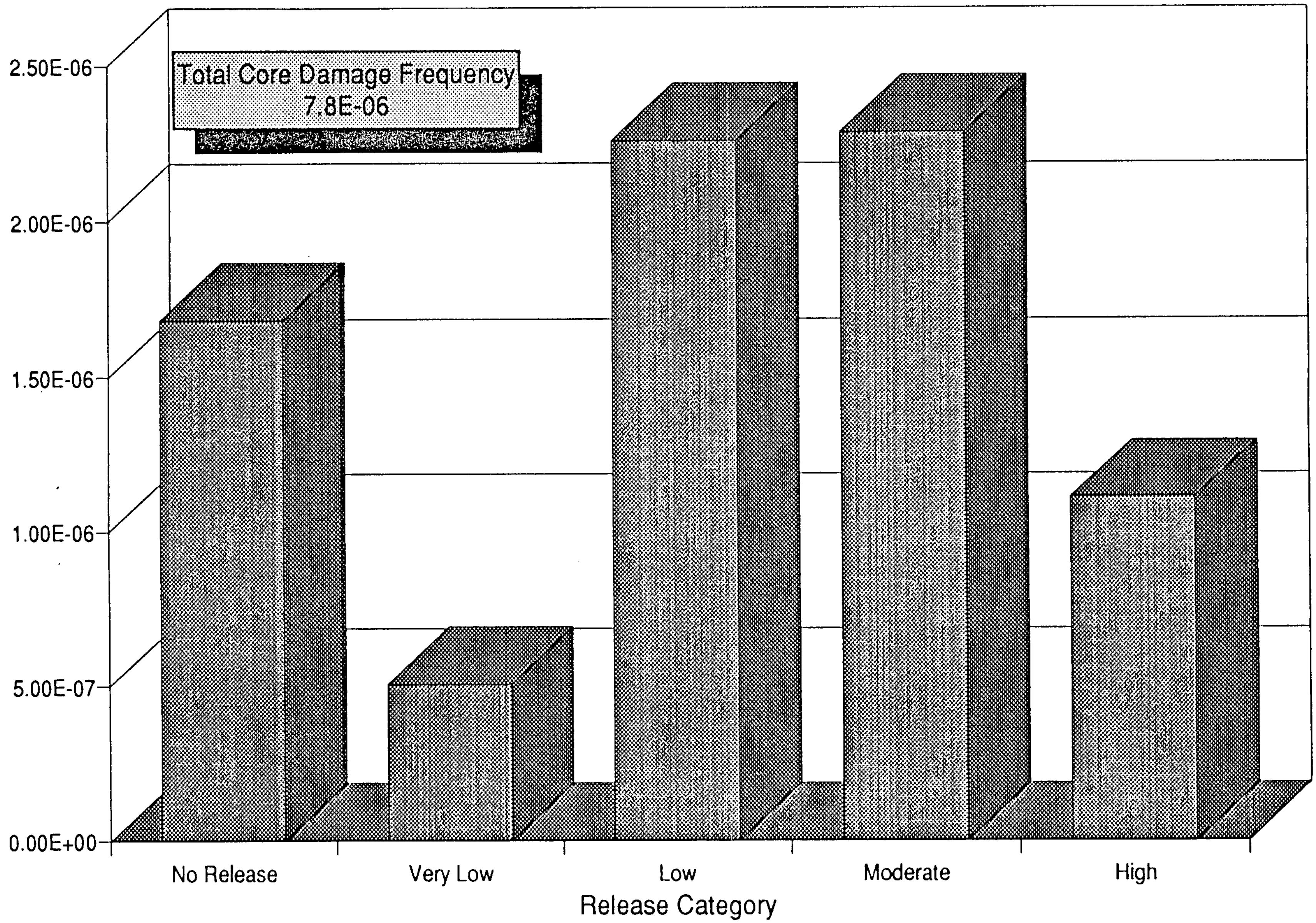


Figure 4.6-10
Summary of Release Magnitude

4.6.3.3.4 Relationship to Safety Goal

It is important that the above selection of release bins be flexible enough to be used to answer important questions that may be raised by the NRC in IPE evaluation or application. Therefore, a review of available published NRC directives and staff recommendations was performed. The primary purpose of the IPE is to perform a systematic evaluation of each plant for vulnerabilities to severe accidents, not to assess nuclear power plant risk relative to the safety goals. In addition, the strength of PRAs or similar examinations is not in determining absolute risk, but in better understanding plant operations and in determining relative risks and how to reduce them.

The Commission's Severe Accident Policy Statement stated:

> *...formulate an integrated systematic approach to an examination of each nuclear power plant now operating or under construction for possible significant risk contributors (sometimes called "outliers") that might be missed absent a systematic search.*

In SECY-88-205 dated July 15, 1988, which forwarded the IPE program to the Commission for approval, the staff indicated how the IPE results would be used with the Safety Goal Policy.

> *. . . we intend to review the IPE results as an aggregate to identify severe accident vulnerabilities generic to a class or several classes of plants. Such generic vulnerabilities would be used to determine if deficiencies in the regulations existed. If deficiencies were identified, the benefits of modifying the regulations would be assessed against the safety goal policy as part of determining whether modifications to the regulations were needed.*

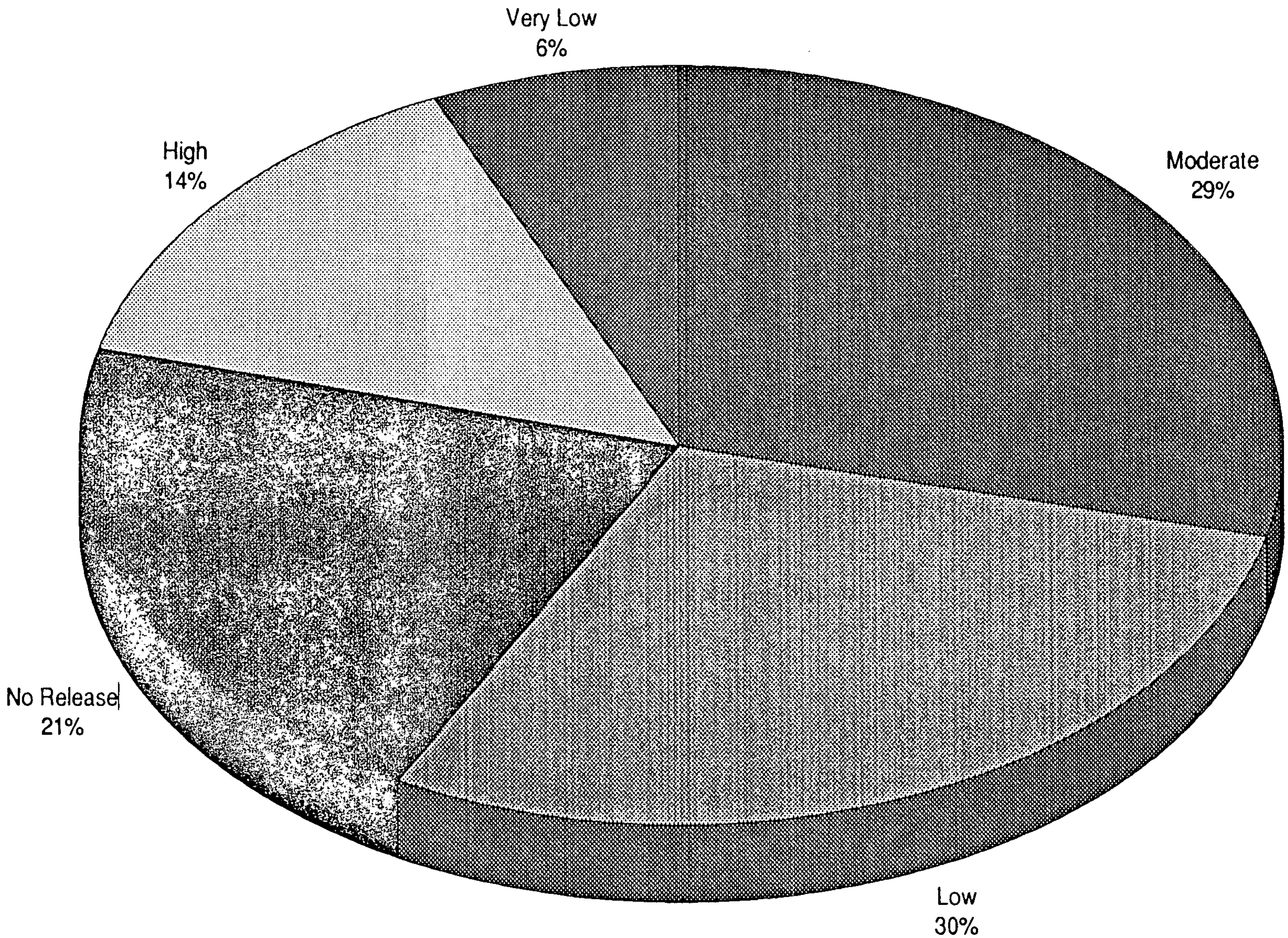


Figure 4.6-11
Summary of Radionuclide Release Magnitudes

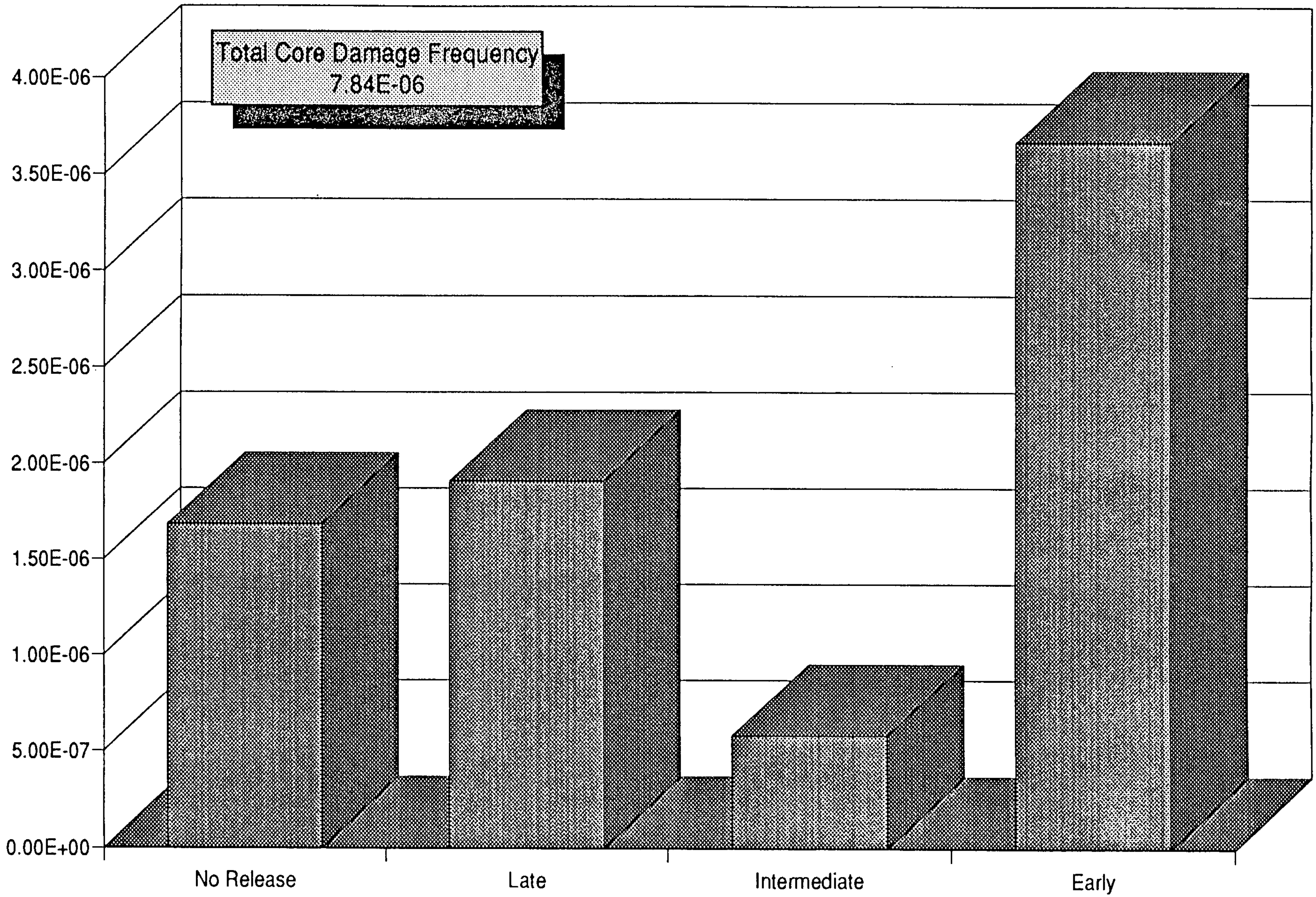


Figure 4.6-12
Summary of Release Timings

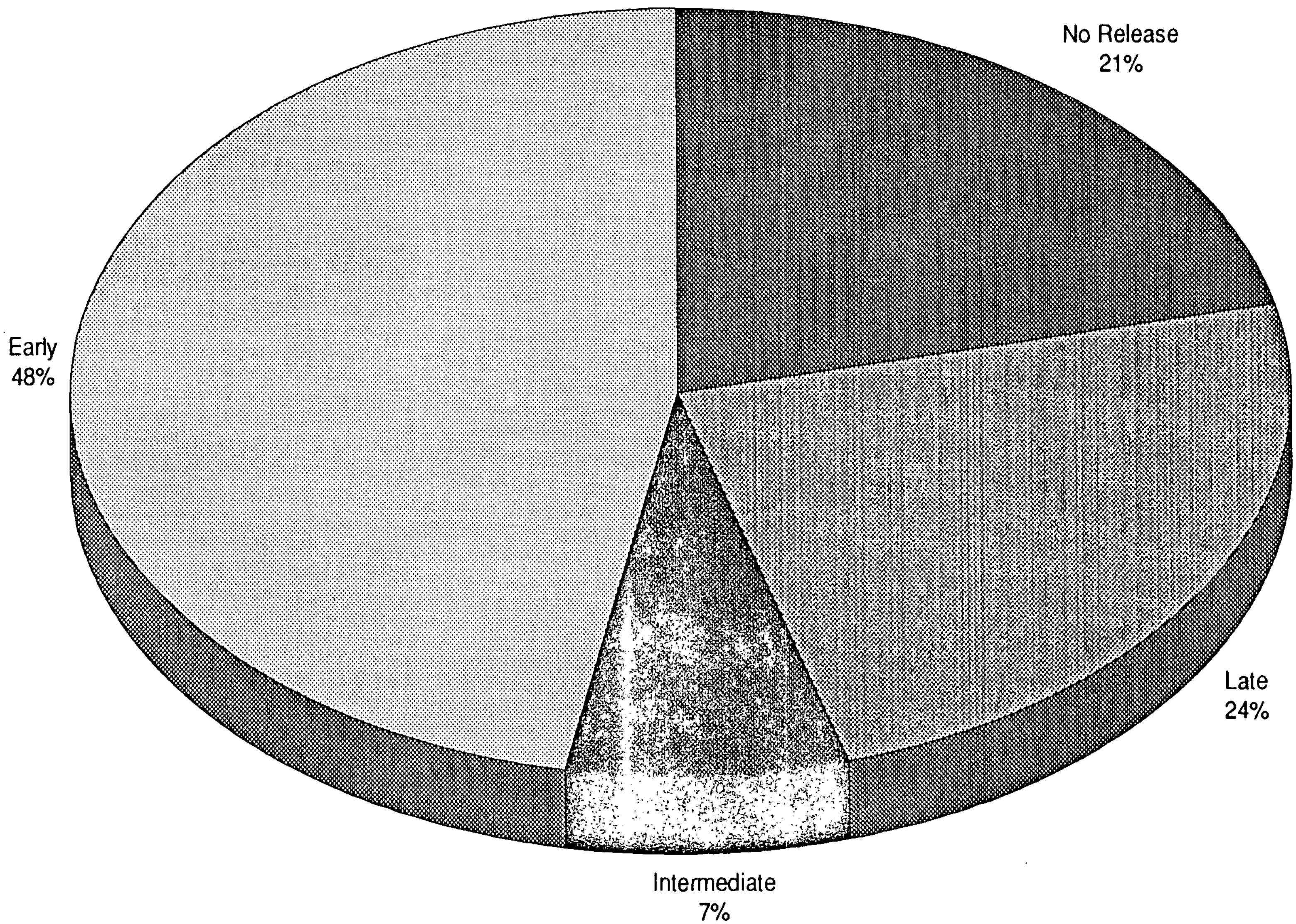


Figure 4.6-13
Summary of Radionuclide Release Timings

The use of the IPE results in this fashion is consistent with the Commission's Safety Goal Policy.

The Commission recognizes that the safety goal can provide a useful tool by which the adequacy of regulations or regulatory decisions regarding changes to the regulations can be judged.

It appears that one of the items of interest in the assessment (either on a plant specific or a generic basis) is a comparison with the safety goal general performance guideline stated by the Commission as follows:

Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation.

The difficulty is in clearly defining a "large" release. The following discussion provides the basis for connecting this "large" release with the release bin characterization in the DAEC PRA as High/Early.

The NRC staff recommendation on the definition of "large" release (SECY 90-405 dated December 14, 1990) is:

A large release is a release of radioactivity from the containment to the environment of a magnitude equal to or greater than: (An amount, to be determined by the staff, expressed in curies or fraction of the core inventory, which has the potential, based on representative site characteristics, for causing one or more offsite prompt fatalities.)

This definition of "large" release is based on offsite consequences. However, rather than comparing plant specific offsite consequences, the staff proposes that a spectrum of sites be considered to establish representative site characteristics. These site characteristics would take into account factors such as meteorology and population distribution. From these site characteristics, the staff will determine a value for an accidental radioactive release to the environment that would have the potential for causing doses high enough that one or more prompt fatalities are probable near the representative site. In other words, Safety Goal Objective Level Three would define a large release to be a release of predetermined magnitude.

In this definition, the magnitude of the source term release may be expressed as curies (or "equivalent curies") or fraction of the core inventory of chemical elements that represent the radionuclides present at full power operation. Appropriate provision will need to be made to address significant variations in power levels, if the definition is stated in terms of fraction of core inventory released.

The effort to determine the release magnitude would focus on highly exposed individuals to determine the release required for a prompt fatality in a fashion identical to that used in NUREG-1150. That is, the weighted probability of a prompt fatality over the exposed population, given site and source term factors, would be determined. The source term factors include the timing of the release, its path to the environment and energy content, and the biological effectiveness of the various radionuclides. The site factors include population distribution and meteorology. It is expected that the assumptions used for emergency planning early in the accident sequence will not be critical and that the magnitude selected for a large release will be independent of emergency planning assumptions early in the accident sequences. The staff intends to confirm this by evaluating the effect of various emergency planning assumptions as part of the analysis.

Therefore, the definition of "large" release can be correlated to the High/Early DAEC release category because of the following:

High: Releases less than High have been shown to have little chance of causing prompt fatalities.

Early: The issue is to define those releases that can lead to prompt fatalities before effective emergency planning can be implemented.

The determination of the likelihood of a "large" release to the environment can then be answered by examining those releases that are both early and high in magnitude.

From Table 4.6-4 it can be seen that, if "large" release is defined as any release to the environment of sufficient radionuclide material to be life threatening and within a time frame too short to allow protective action (i.e., H/E), then the frequency of such "large" releases would be 5.02E-7/year which is less than the NRC staff generic safety goal objective. Because of the emphasis in the NRC Safety Goal Policy on the secondary objective of maintaining "large" releases below 1E-6/reactor year on a generic basis, it may be useful to compare the DAEC results with this objective (see Figure 4.6-14). The H/E frequency shown here represents a best estimate upper bound, in that the H/E release category is preferentially assigned to accident sequences (when appropriate) when a significant degree of uncertainty exists in the modeling of the sequence.

4.6.3.4 Containment Integrity

In the assessment of radionuclide release, the mechanisms for releases include:

- Containment failures

- Containment venting.

The reason for separating the release modes between containment venting and containment failure is to provide an indication of those release pathways that are controllable, and therefore for which containment integrity can be restored.

Figure 4.6-15 provides an interesting division of the releases. Based on the DAEC quantification, releases associated with venting represent a large fraction of the releases. "Containment venting" as referred to in this comparison refers only to those Level 2 end states for which no other release pathway is induced during the core melt progression. The large fraction of drywell venting is due to successfully completing drywell flooding.

Another informative division of release is based on containment failure modes. Figure 4.6-16 provides a pie graph illustrating the division of release and no release sequences; the release sequences are subdivided into various containment "failure" modes:

- Overtemperature/Overpressure
- Vented Containment
- Shell Melt-through
- Other.

There are a number of containment failure modes that can occur coincidentally. For example, a postulated ATWS scenario may induce a torus dynamic failure. The subsequent loss of injection to the RPV can result in drywell shell failure due to direct contact with debris or high temperatures in the drywell causing drywell head failure. These consequential failure modes are more severe in terms of release potential because the pool is bypassed.

4.6.3.5 Combination of Release Magnitude and Timing

In Sections 4.6.3.3.2 and 4.6.3.3.3, the radionuclide release as a function of magnitude and timing were examined separately. These two viewpoints can be combined to determine if there are relationships or impacts associated with the release magnitude and the timing that may influence accident management decisions.

Figure 4.6-17 summarizes graphically the radionuclide release magnitude in a manner similar to Figure 4.6-10 except that it is augmented to also show the time of the release associated with the contributions to each release magnitude.

Figure 4.6-18 summarizes the radionuclide release timing in a manner similar to Figure 4.6-12 except that it is augmented to also show the magnitude of the release associated with the contributions to each release time phase.

4.6.3.6 Containment Radionuclide Releases as a Function of Accident Type

Important insights into possible accident management strategies can be obtained by identifying the types of accident sequences that are contributing to the radionuclide release bins.

Figure 4.6-19 provides a graphical summary of the "large" release contributors by accident class. As can be seen from the figure, Class IB, Class IA, Class IE, and Class IIID accidents are the dominant contributors to High-Early releases. The station blackout sequences are the largest contributor to High/Early releases (approximately 53.5%).

It is also useful to examine the difference between the contributors to core damage and those to high or high/early releases. Figure 4.6-20 compares the contributors to core damage frequency and those that contribute to "high" releases. This comparison graphically shows that the sequences that dominate core damage frequency are not necessarily those which dominate the high release. For example, Class IIID accidents are small contributors to core damage frequency but make up a significant fraction of the H/E releases.

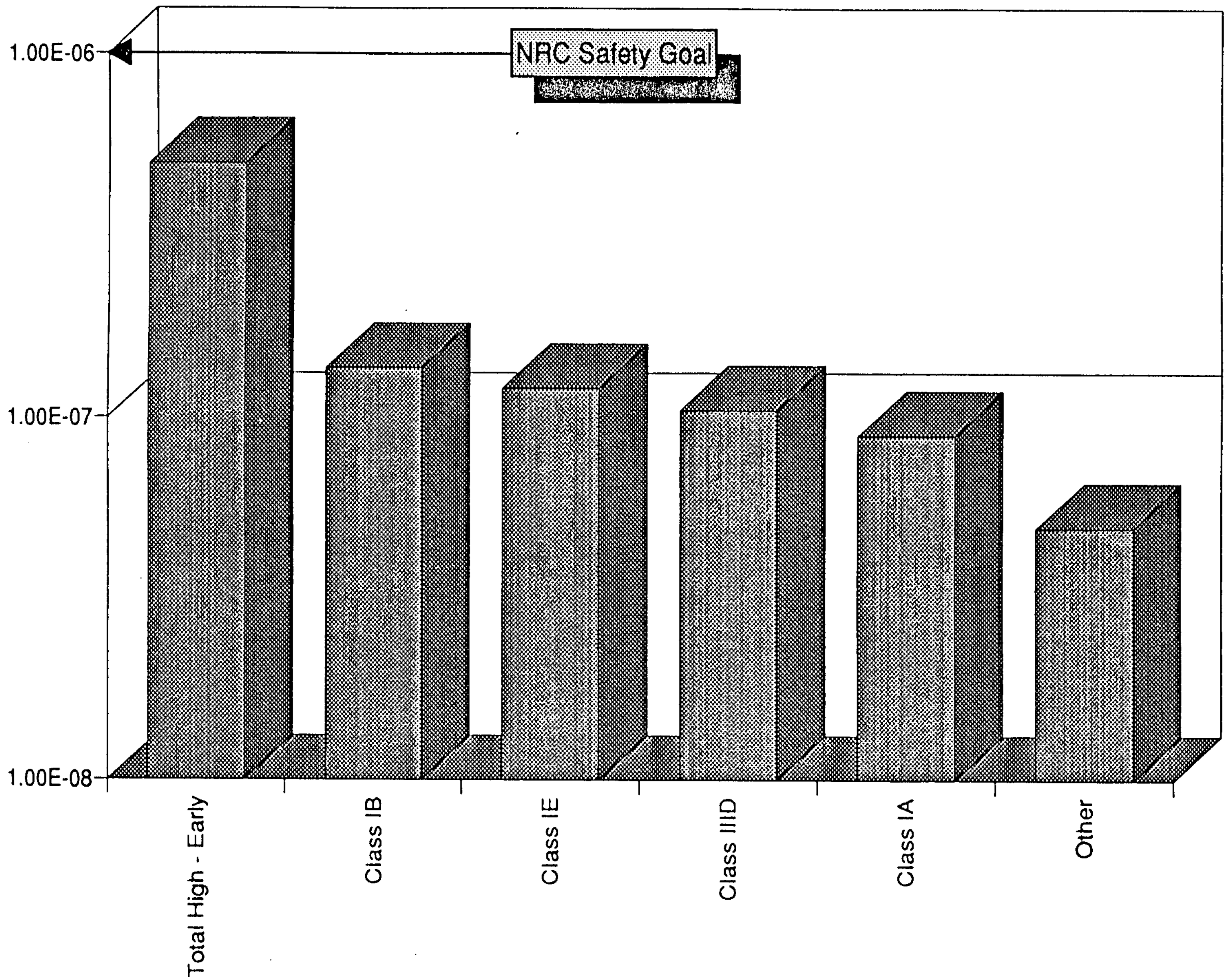


Figure 4.6-14
Comparison of Contributors to the Large Release Category

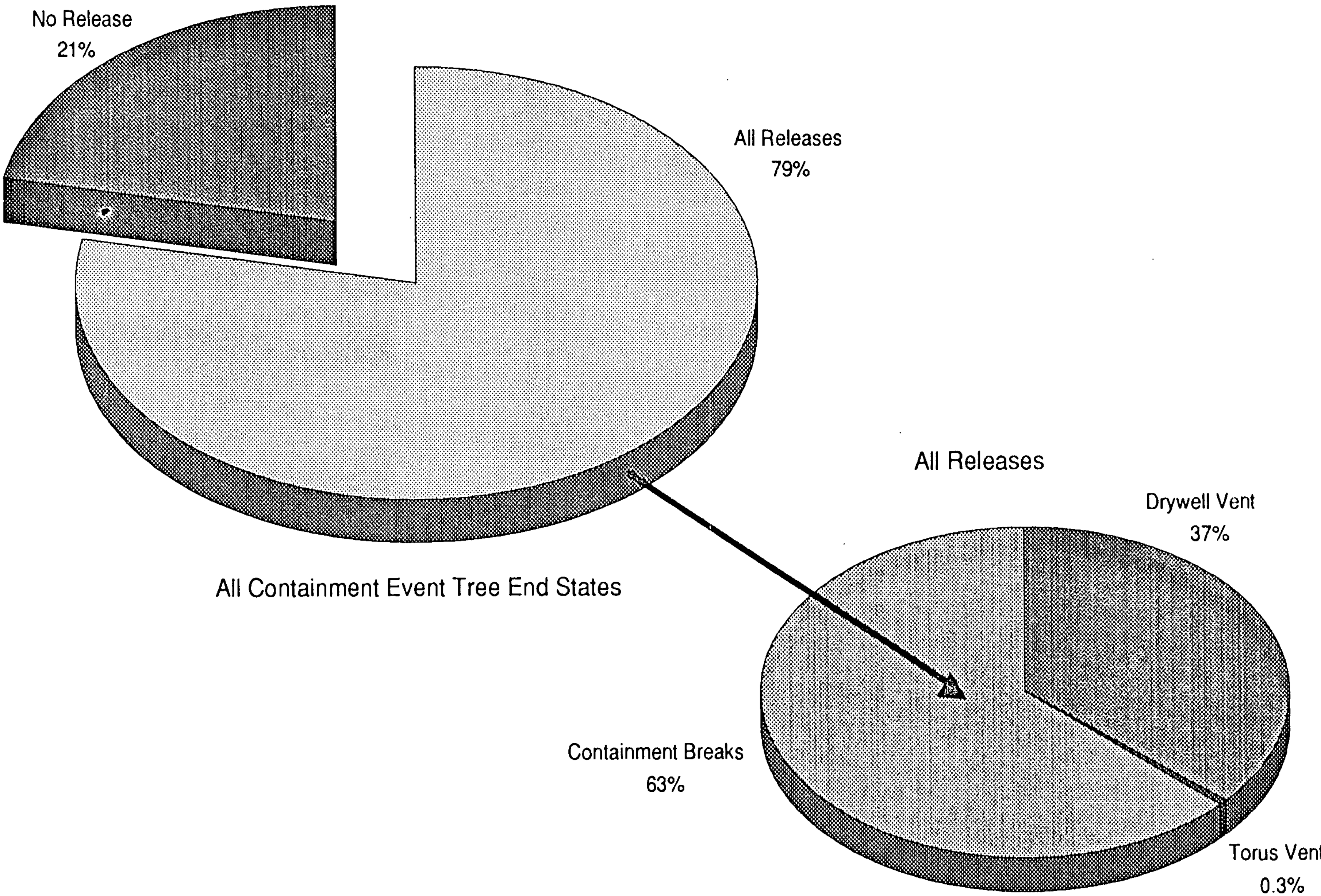


Figure 4.6-15
Ratio of Vent Sequences to Containment Failure Sequences

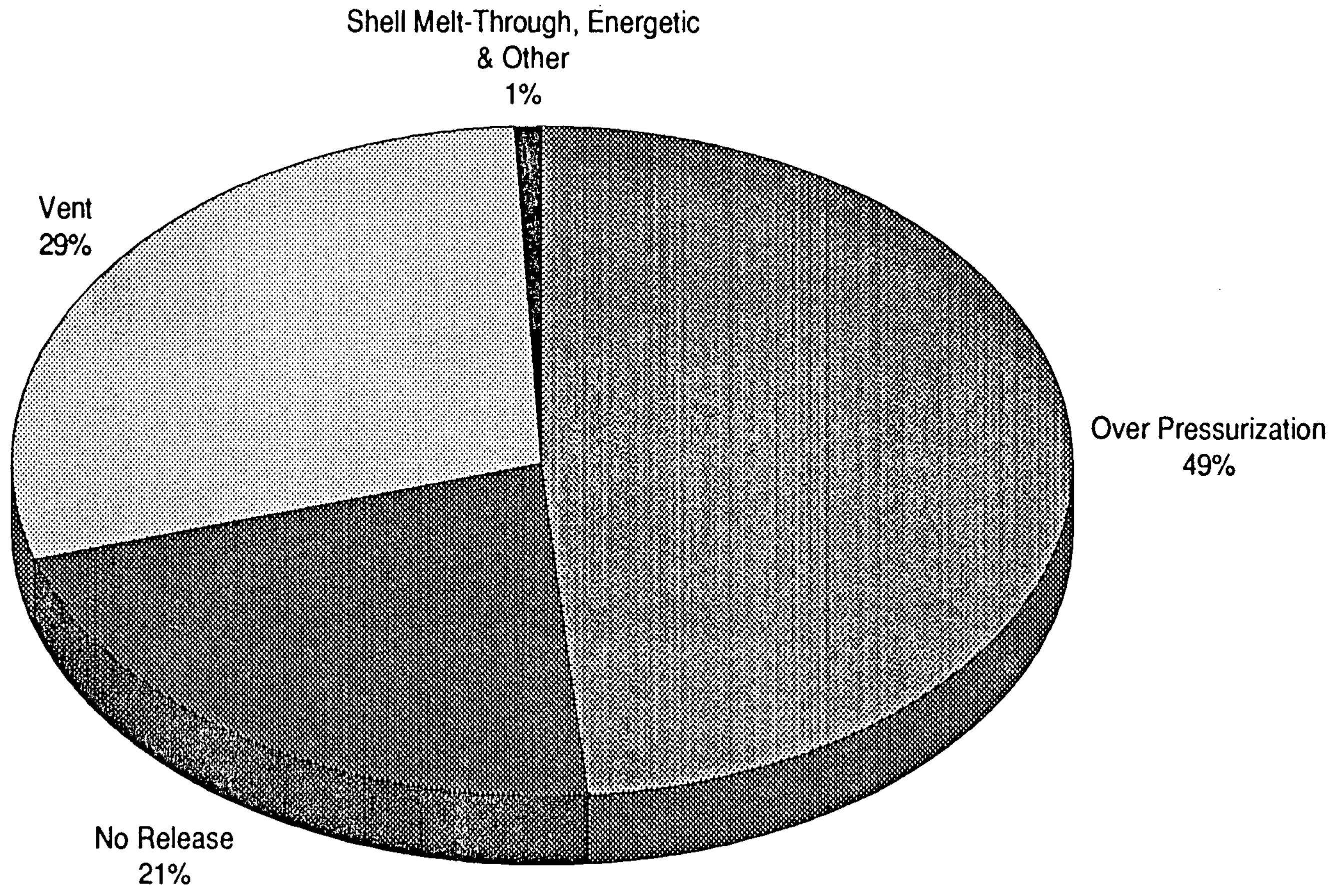


Figure 4.6-16
Summary of Containment Failure Modes

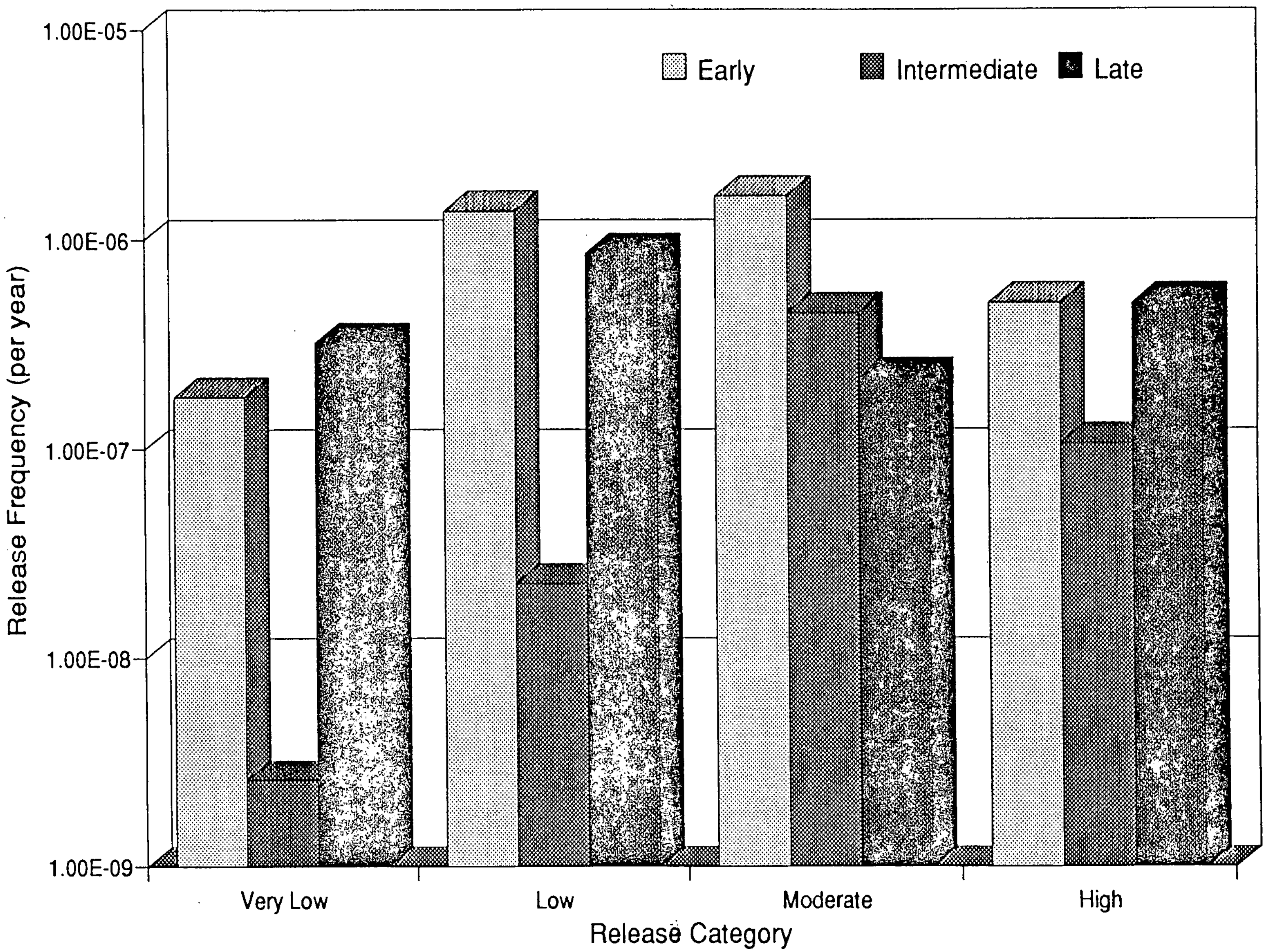


Figure 4.6-17
Release Magnitude Contributions

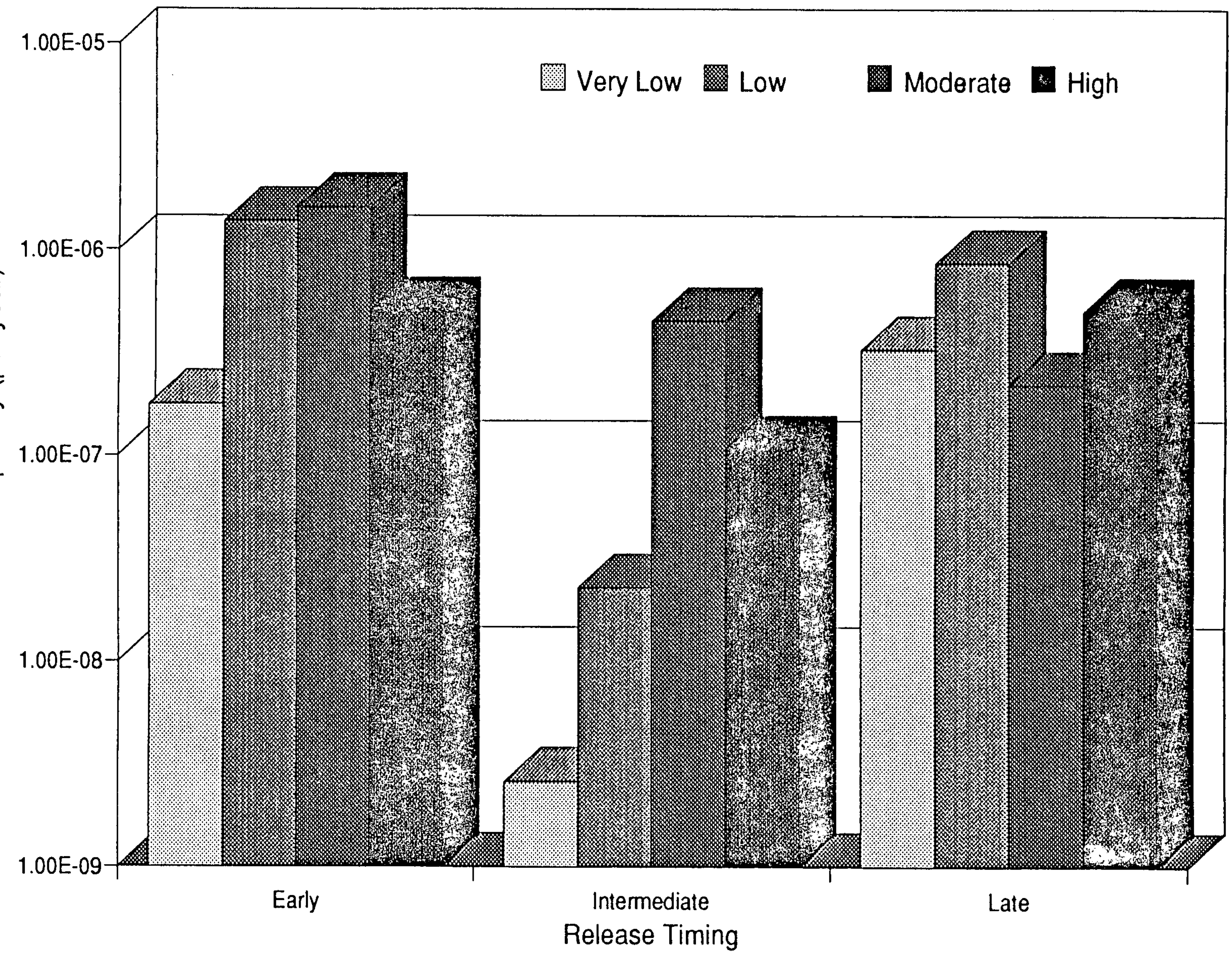


Figure 4.6-18
Release Timing Contribution

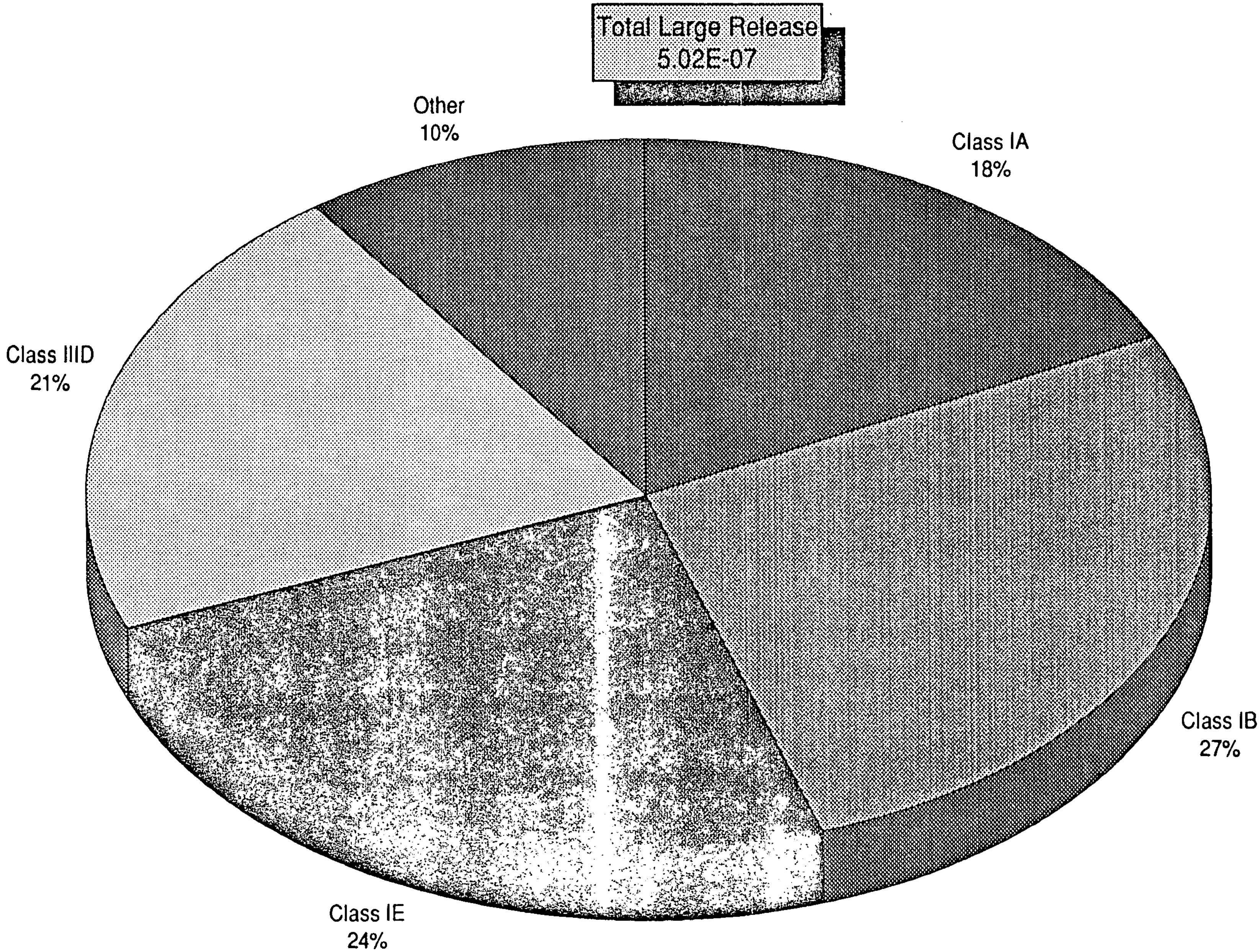


Figure 4.6-19
Contributors to "Large" Release Category

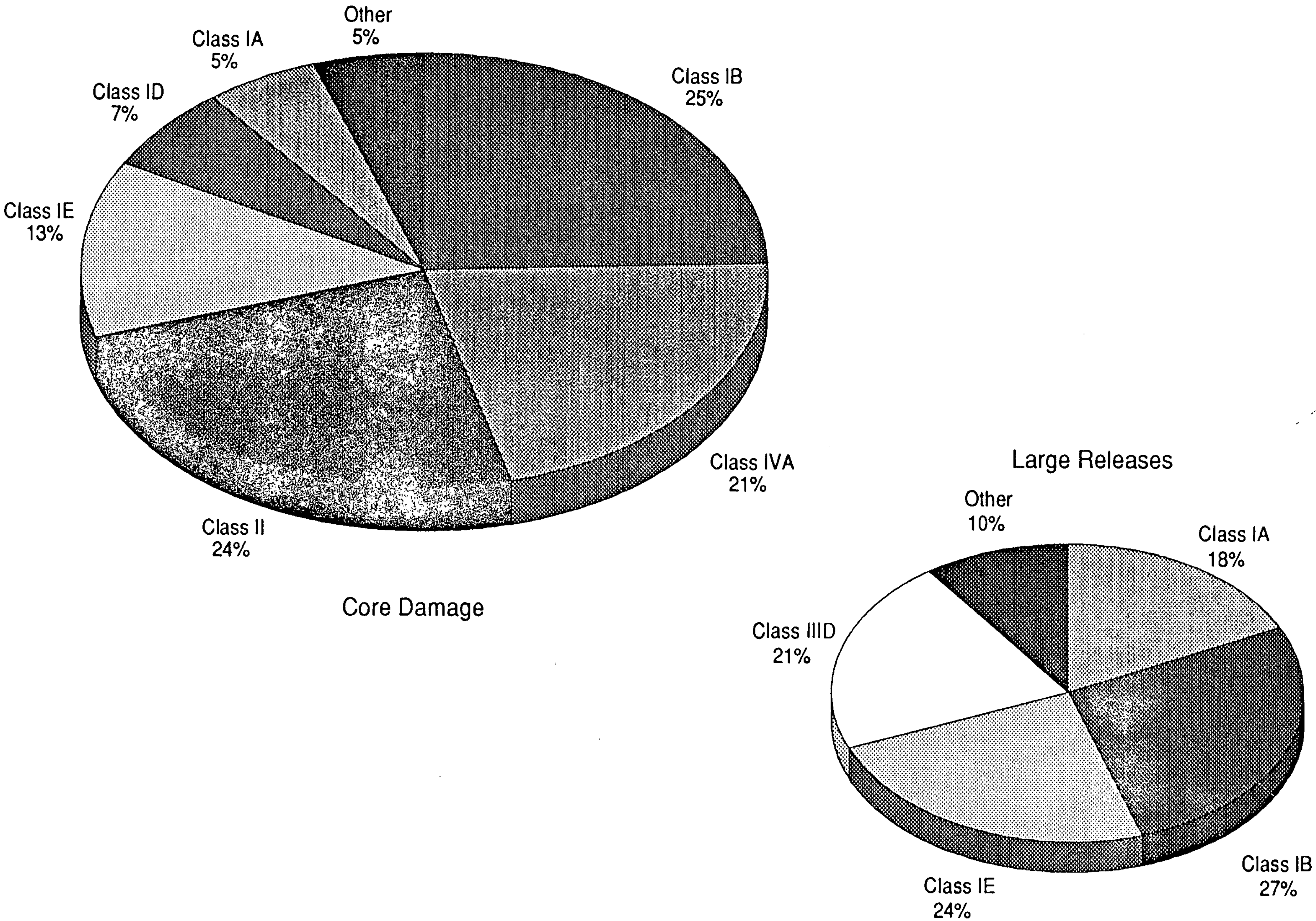


Figure 4.6-20
Comparison of Core Damage and Large Release Contributors

4.7 RADIONUCLIDE RELEASE CHARACTERIZATION

The radionuclide release sequences determined from the CET evaluation that exceed the screening criteria frequency (i.e., reporting criteria) have been assessed to determine their radionuclide release magnitude. The timing is also reported as discussed in Section 4.6.

The determination of the radionuclide release magnitude for the DAEC IPE has taken two approaches both of which were identified as viable options in NUREG-1335. The two approaches are:

- Use of existing Mark I radionuclide releases for a similar plant to characterize some release sequences,

- Use of plant specific DAEC calculations to both confirm the surrogate plant calculations and to fill in missing sequence calculations.

This subsection includes the following important discussion items regarding radionuclide release characterization:

- Overview (Section 4.7.1)

- Governing features (Section 4.7.2)

  - Removal processes and pathways
  - Containment failure modes
  - Phenomenology
  - Timing

- Release Bins (CET End States) (Section 4.7.3)

- Criteria for Release Bins (Section 4.7.4)

  - Magnitude
  - Timing

- MAAP calculational results for the Release Bins (Section 4.7.5).

4.7.1 Overview of Potential Release Characterization

Each CET end state can be associated with a radionuclide source term bin which covers a spectrum of similar potential scenarios and timing. Theoretically, it would be desirable in determining the point estimates of risk to evaluate the source terms for each sequence or each accident plant damage state. However, because of the very large number of systemic sequences and for purposes of risk presentation, the CET end states are characterized in such a manner as to combine similar "consequence impact" release event sequences within a CET end state.

4.7.2 Governing Features in Radionuclide Release Characterization

There are a number of plant features or accident progression features that can substantially increase or decrease the ability to retain fission products or mitigate their release. This subsection reviews some of the more important of these features in the following areas:

- Radionuclide removal processes

- Containment failure modes

- Phenomenology

- Timing.

4.7.2.1 Removal Processes in Containment

Radionuclide release processes are initiated when the core overheats and melts. These release processes involve transport from the fuel, from the RPV, and from primary and secondary containment. These release processes when categorized into end states can indicate the amounts and types of radionuclide material that could potentially be released to the environment. It should be noted that, depending on the kind of accident in progress, there are inherent removal mechanisms that can occur to remove and retain these fission products. These deposition mechanisms include plateout and retention on the vessel surfaces (at least as long as RPV temperatures remain relatively low).

Once the fission products are airborne in the containment, there are removal mechanisms that reduce the magnitude of the source terms that are available for leakage to the environment. These removal mechanisms include plateout and settling in containment. The degree of attenuation is determined to a large extent by the time available for these processes to occur. The time between fission product release from the fuel to containment failure determines the residence time of the radionuclides within containment. The containment failure modes and failure location also contribute to determining the radionuclide removal mechanisms that are operating along the exit path to the environment.

Given that radionuclides are released from the fuel, the removal of fission products from any leakage pathway varies with the kind of accident sequence in progress, the containment failure mode, and the type of fission product being transported. These removal mechanisms may be categorized in terms of the following:

- Natural removal - Radionuclides may be removed by natural deposition (plateout) or settling mechanisms.

- Active Safety System - The systems that can potentially "wash-out" or filter particulate radionuclides:

  - Containment sprays
  - RPV and containment injection
  - SGTS.

- Passive Safety System - The suppression pool provides a removal mechanism for radioactivity during a core melt progression accident. The effectiveness of pool decontamination depends on the characteristic of the aerosol source (e.g., particle size distribution), the temperature of the water, and whether pool bypass pathways exist.

These removal mechanisms are each included in the DAEC specific deterministic modeling.

4.7.2.2 Containment Failure Modes

For each of the accident sequence classes, there is a set of containment failure modes and release pathways that affect the magnitude of the radionuclide releases (See Section 4.4.2). Briefly, the principal methods in which the containment failure modes affect the radionuclide release are:

- Size of the Containment Breach - The size of postulated containment failure determines the usefulness of the reactor building with regards to the capability of the structure and systems to affect the release source term.

- Location of the Breach - The location of the postulated containment failure affects the degree of radionuclide release decontamination along the path; the more torturous the pathway for release, the greater the likelihood that deposition reduces the radionuclide release mass distribution. However, the most important aspect of the location is in relation to the suppression pool. For some sequences that include drywell failure, the radionuclide release after containment failure could bypass the suppression pool, thus eliminating this valuable fission product removal mechanism.

4.7.2.3 Phenomenology

The CET includes an assessment of the probability of occurrence of energetic phenomenological effects that can result in containment failure and add energy to the radionuclide release. Examples of such phenomena include the following:

- Steam explosions

- Hydrogen detonation

- Direct containment heating

- Excessive blowdown pressure.

Such phenomena, while of low probability even given a severe accident, may have a substantial influence on the containment integrity, radionuclide removal processes, and the radionuclide release source term. Therefore, the end states of the Level 2 PRA are also influenced directly by the occurrence of these phenomena.

4.7.2.4 Timing

The length of time over which the accident progresses can influence the degree of retention and the pathway through which the release propagates. In addition to the accident initiation time, there is also a requirement to define the duration, i.e., time over which the accident release will be calculated.

The assessment of radionuclide release duration for the purposes of calculating release magnitudes and the assignment of accident sequences to release categories includes two considerations:

1) The compensatory measures that can be taken to significantly reduce or prevent dose to the public, and

2) The characteristics of radionuclide release.

It is incumbent upon the PRA analyst to determine the end point of deterministic calculations that describe the impact of an accident scenario, with respect to potential off-site consequences after all measures prescribed in the EOPs are postulated to be ineffective in mitigating the accident.

These two principal considerations are discussed below along with the conclusion regarding the selection of an appropriate release duration used to determine the magnitude of the source term assigned to the severe accident end state.

4.7.2.4.1 Compensatory Measures

The consideration of MAAP calculational results for determining and assigning a radionuclide release category to an accident sequence is based in part on off-site accident response which is not examined directly in a Level 2 PRA. Some of these response actions are prescribed in the facility's emergency response plan; however, these actions, which are routinely practiced, are geared to mobilizing utility resources to implement emergency procedures, assessing the potential off-site consequence of an accident, and recommending to government officials appropriate action for protecting the public. Usually, an emergency plan does not include direction for the emergency response organization to effect supplemental actions (i.e., in addition to the EOPs) to mitigate the accident and return the reactor plant to a stable, albeit damaged condition. Utilities are currently actively pursuing programs to develop severe accident management strategies for implementing these actions.

The scenario end point might represent the time at which the accident poses minimal additional off-site dose to the population. This time frame is difficult to precisely assess because the analyst must extrapolate beyond existing emergency procedures to forecast the utility and government's ability to implement effective mitigation measures to either terminate the radionuclide release to the environment or remove the affected population.

There are primarily two ways to minimize the accident's impact on the population and regain control of any offsite releases:

1) Evacuate the affected public; and

2) Mitigate the radionuclide release from the facility.

The evaluation of safe stable states in a PRA has generally involved the assessment of equipment operation and operator actions over an extended period of time. This time frame is nominally taken to be sufficient to marshall additional resources and mitigate the accident progression. The considerations that dominate the choice of mission time are as follows:

- Beyond the time frame of 24 hours, "ad hoc" procedures can be developed to utilize additional hardware and personnel resources, and implement system recovery and alignments that are not presently considered part of plant practices. Training for such extreme and unlikely situations is not considered to be an effective use of limited available resources.

- The emergency planning organization and procedures could potentially accelerate the time to implement such heroic actions if a catastrophic event were to occur at a nuclear facility in the United States.

- During the course of the accident, the TSC and EOF would become operational, and additional expertise could be available to successfully help mitigate the accident.

- It is considered highly likely that off-site resources (e.g., equipment, power, vehicles) would become available.

From a risk perspective, actual data from natural and man-caused disasters indicate that public evacuations can be effectively carried out well within time frames of less than 36 hours.

For instance, even during the disaster at Chernobyl, effective protective measures (i.e., evacuating the population and undertaking heroic actions to mitigate the radionuclide release) were being implemented within 16 hours after the hydrogen explosion disrupted the reactor. This is well within the 36 hour time discussed here. Figure 4.7-1 shows the approximate timeline of events to regain control of the facility and minimize the off-site effects from the radionuclide release.

It is expected that if a similar catastrophic event were to develop at a U.S. facility, that the governing Emergency Plan would be implemented soon after the declaration of an emergency. The Chernobyl example can only be used to illustrate the upper bound on timeliness of emergency response as a result of unplanned, but heroic actions by numerous individuals and agencies. The incident at TMI-2 and subsequent enhancements in US Emergency Planning indicates that the emergency command structure and on-site and off-site resources would be available to the government to protect the public and the utility to mitigate the accident.

4.7.2.4.2 Important Characteristics of Radionuclide Release Timing

The radionuclide release characterization has been postulated to have two primary characteristics:

- An early release component that occurs at or near RPV and/or containment breach

- A very long duration release component that is characteristic of either:

  - revaporization of radionuclides from "hot" internal deposition surfaces; or,

- revaporization of material deposited in water pool.

Using the MAAP code, it has been found that the revaporization term can occur over time frames of many days. However, it is also found that the predominant release term generally can be found to occur within a time frame of 36 hours past RPV breach for "dry" cases, i.e., cases with no water injection. These observations from MAAP are based primarily on BWR MAAP assessments where temperatures of internal surfaces may be substantially higher than in PWRs.

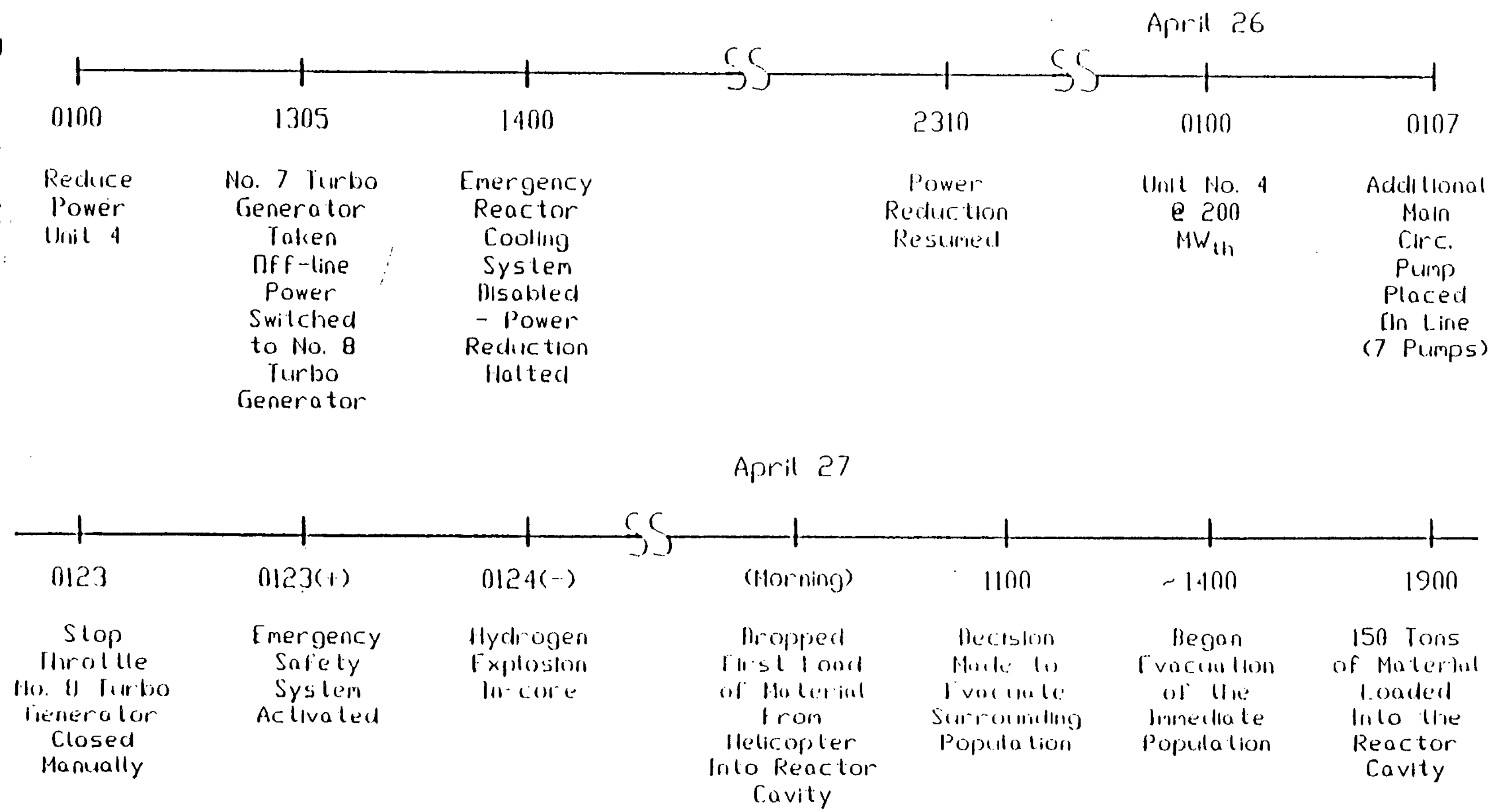


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Figure 4.7-1 Chernobyle Unit 4 Accident Timeline

Examples of this characteristic time frame for CsI release for the following four different accident types are shown in Figure 4.7-2:

- Class IA: Loss of makeup injection with the RPV remaining at high pressure until RPV breach
- Class IB: Loss of makeup injection due to station blackout
- Class IIIC: Large LOCA with no injection
- Class IVA: Failure to scram with containment pressurization and failure

4.7.2.4.3 Conclusion Regarding Release Duration

Based on this information, the scenario end point in the Level 2 PRA is defined as 36 hours after RPV breach.

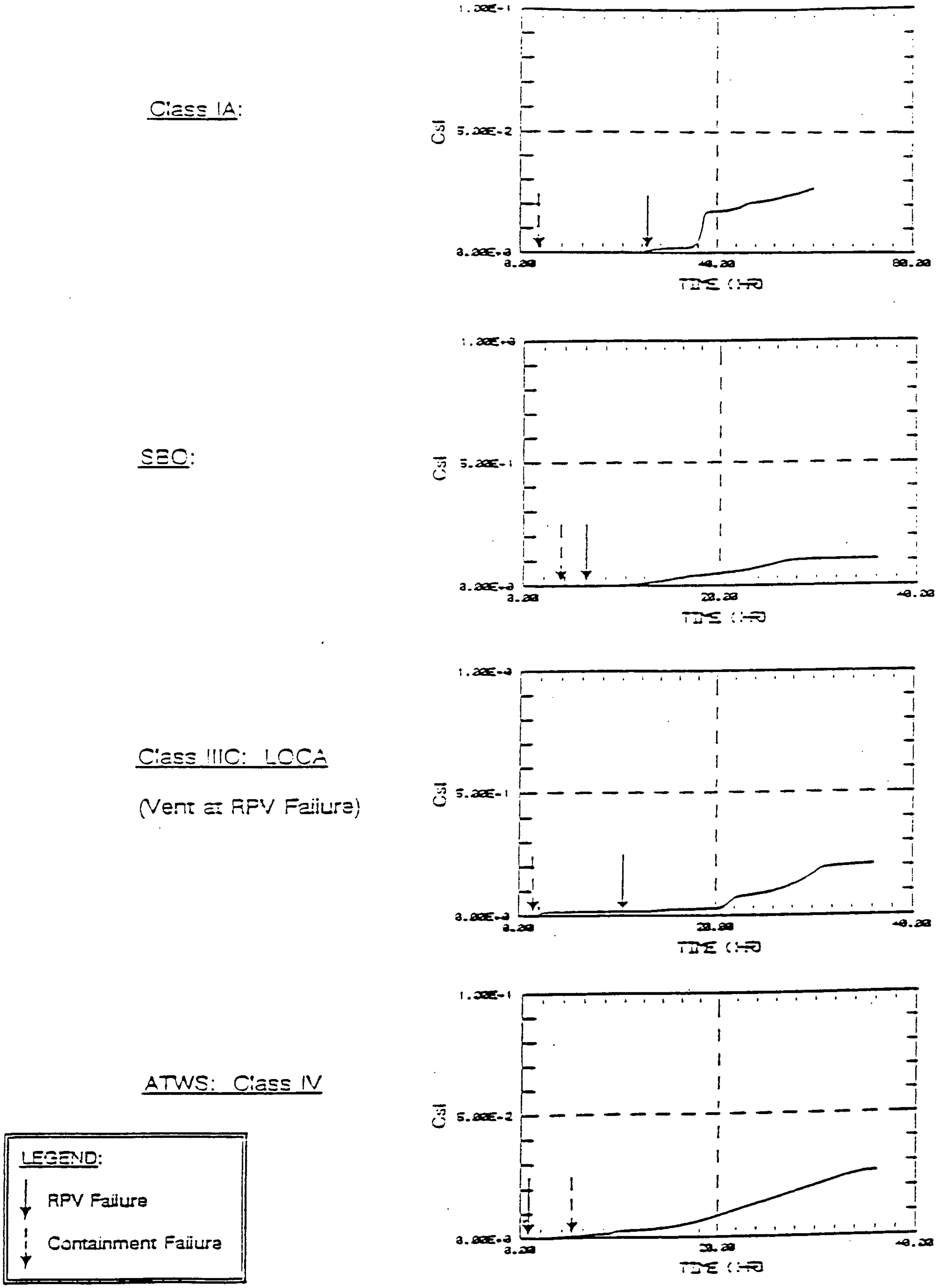

Figure 4.7-2 Comparison of the CsI Release Profile as a Function of Time Following RPV and Containment Failure

4.7.3 Radionuclide Release Categories (CET End States)

The spectrum of possible radionuclide release scenarios is represented by a discrete set of categories or bins. The end states of the containment and phenomenological event sequences may be characterized according to certain key quantitative attributes that affect offsite consequences. These attributes include two important factors:

1) Timing of release

2) Total quantity of fission products released.

Therefore, the containment event tree end states are meant to represent the source term magnitude and relative timing of the radionuclide release. The number of categories to be used in the source term characterization offers a level of discrimination similar to that included in numerous published PRAs.

The IPE process has received extensive guidance from the NRC staff to identify areas of special emphasis. There are a number of issues regarding the definition of CET end states that are summarized below:

- Timing of radionuclide release, per se, does not appear to be a parameter requested by either the Generic Letter 88-20 or NUREG-1335 (the guidance document). However, the guidance document (p.A-11) does indicate that the time of containment failure would be of interest.

- It is stated in NUREG-1335 that for accident management evaluations the timing of accident sequence events (presumably containment failure and release are key items) is important to include.

- Generic Letter 88-20 refers to a source term magnitude greater than 10% I and 10% Cs as a sequence which is to be reported (i.e., WASH-1400, BWR-3 Category or higher releases).[1,2]

- IPE guidance documents from the NRC state that the release magnitude of up to 100 sequences with frequencies above 1E-6/year should be estimated. These can be estimated using:

  - Code calculations, or
  - Past published calculations.

- The NRC staff in NUREG-1335 (p. A-12) states that:

  During the last several years, there have been extensive evaluations of fission product release (source terms) during severe accidents for a variety of reactor designs. The staff encourages the use of these existing calculations whenever they can be shown to be applicable.

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[1] BWR 3 Release Category:

Species	Release Fraction
Noble Gases (NG)	1.0
I	0.1
Cs	0.1
Te	0.3

[2] Note that in subsequent discussions, releases of this magnitude are denoted as the High (H) category in the classification scheme devised in this evaluation.

Consideration must be given to the types of sequences in the release category, however, and the timing of release characteristics for each, before selecting release characteristics to represent the category.

The description of the source term, the release timing, and the implications of each are determined using the results of MAAP calculations and past PRA evaluations. The information developed in previous studies has been used in making subjective assessments for these source term characterizations. The event sequences contributing to a radionuclide release are ranked on the basis of the product of the relative consequences (based on estimated radionuclide release fractions of noble gases, CsI, and Te) and their respective conditional probabilities, so that potentially risk-dominant scenarios are identified and adequately represented. Those that are similar in timing and release fractions are sorted into groups of release categories to reduce the number of sequences required to calculate the risk profile. The DAEC IPE includes a cross check of accident sequences for: frequency, containment bypass, containment isolation, containment system availability, and approximate source term.

The next section identifies the criteria used to define the release bins used in the DAEC IPE analysis.

4.7.4 Criteria Used in Timing and Release Magnitude Assignments

The release categories are defined based on two parameters: timing and severity. Timing of the release for each sequence is based on MAAP calculations of the sequence chronology.

4.7.4.1 Timing Bins

Three timing categories are used, as follows:

1. Early (E) — Less than 6 hours from accident initiation
2. Intermediate (I) — Greater than or equal to 6 hours, but less than 24 hours
3. Late (L) — Greater than or equal to 24 hours.

The definition of the categories is based upon past experience concerning offsite accident response:

- 0-6 hours is conservatively assumed to include cases in which minimal offsite protective measures have been observed to be performed in non-nuclear accidents.
- 6-24 hours is a time frame in which much of the offsite nuclear plant protective measures can be assured to be accomplished.
- > 24 hours are times at which the offsite measures can be assumed to be fully effective.

Figures 4.7-3 and 4.7-4 are example applications of the determination of radionuclide release time categorizations as a function of:

- The accident type

- The time of release relative to the Emergency Action Level.

The Emergency Action Level is used as the trigger for interaction and is generally considered to occur essentially at the time of initial perturbation or within 20-30 minutes.

Other studies have attempted to define characteristics of radionuclide release that would allow useful interpretation in lieu of performing a Level 3 PRA to determine the plant specific public health effects. One of the recent attempts has been as part of NUREG-1150. Figure 4.7-5 identifies the comparison of the NUREG-1150 timing categories (early and late) with those defined in this analysis.

4.7.4.2 Release Magnitude Bins

The five severity classifications associated with volatile or particulate releases[1] are defined as follows:

1) High (H) - A radionuclide release of sufficient magnitude to have the potential to cause early fatalities.

2) Moderate (M) - A radionuclide release of sufficient magnitude to cause near-term health effects.

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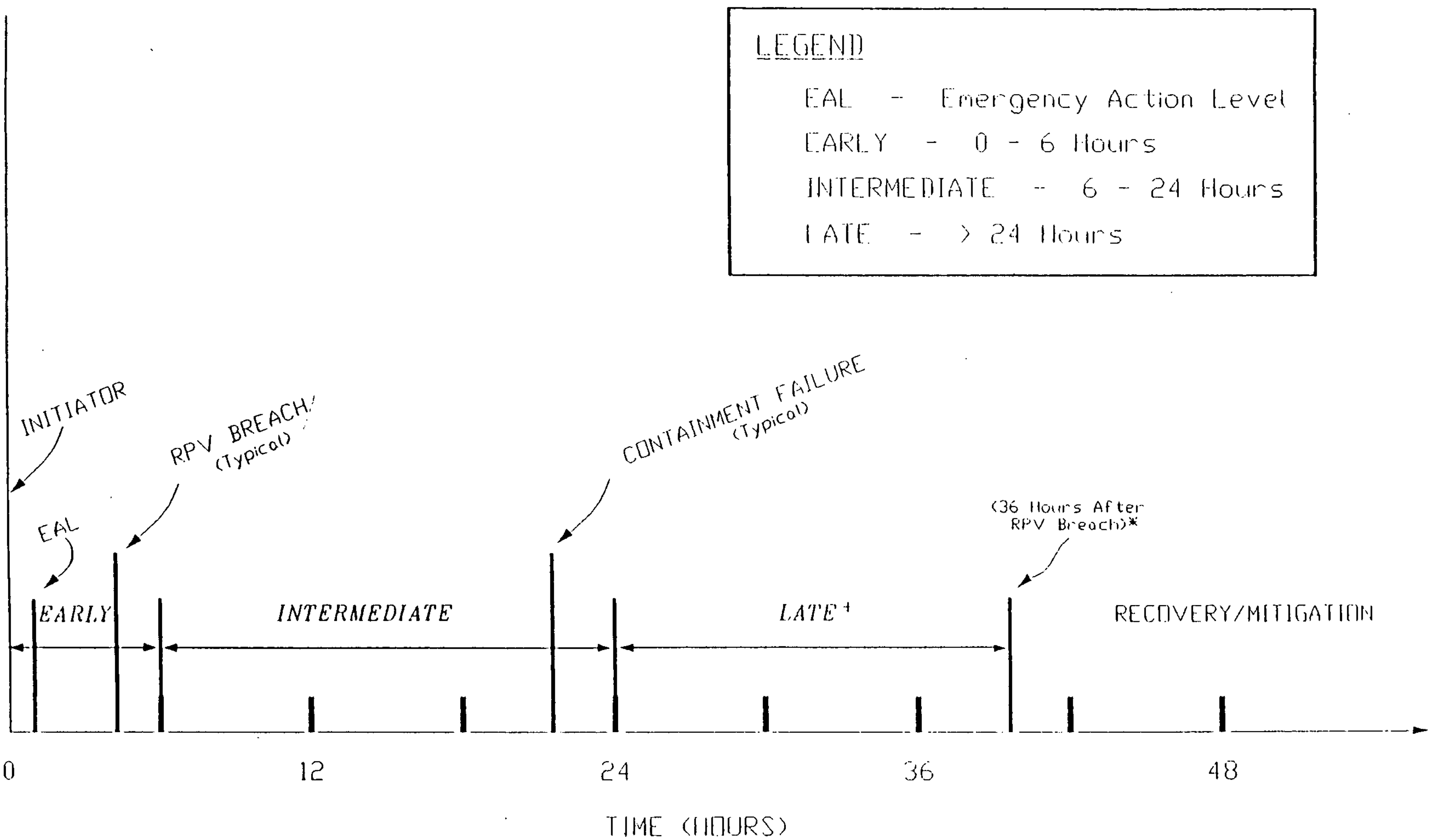
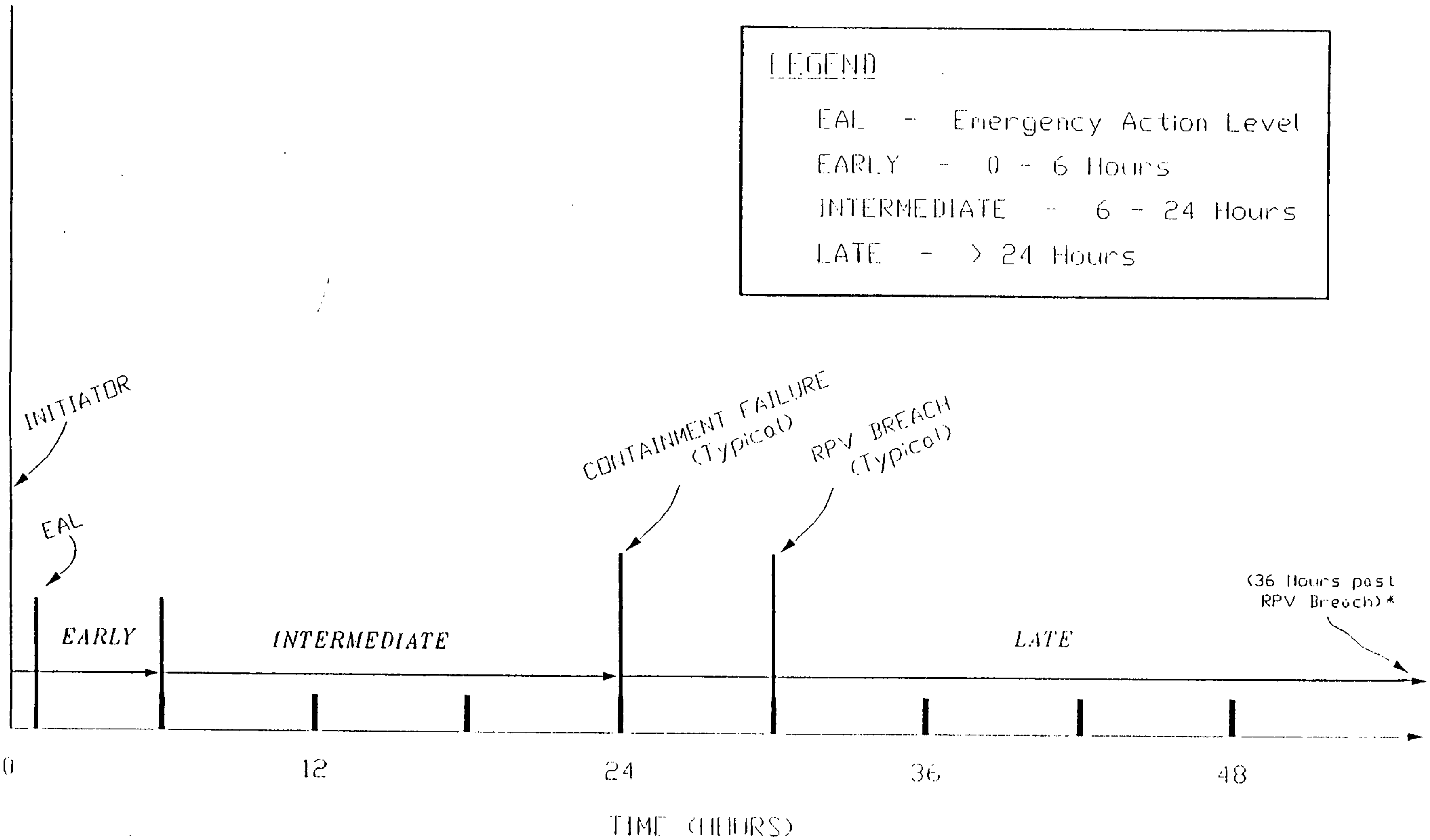
[1] The effects of noble gases may be quite dramatic, causing substantial early health effects if released early in an accident and if the associated plume is directed at an occupied location. The noble gases themselves may result in early injuries or fatalities. However, in most sequences the release of noble gases may occur over a relatively extended period of time unless an energetic failure of containment or secondary containment occurs. Therefore, the noble gases are implicitly included in the definition of release categories. There may however be situations in which noble gases alone result in early health effects, those cases are considered of low probability. The focus of the release categories is on the dominant term in cost benefit evaluations from past assessments, i.e., the latent health effects for which the above formulation adequately encompasses the effects of noble gases on the release.



* For purposes of radionuclide release calculations, 36 hours past RPV breach by molten debris is included

Figure 4.7-3

Example of the Definition of Timing Used in the in the Radionuclide Release Categorization (Typical for Class I & III Sequences)



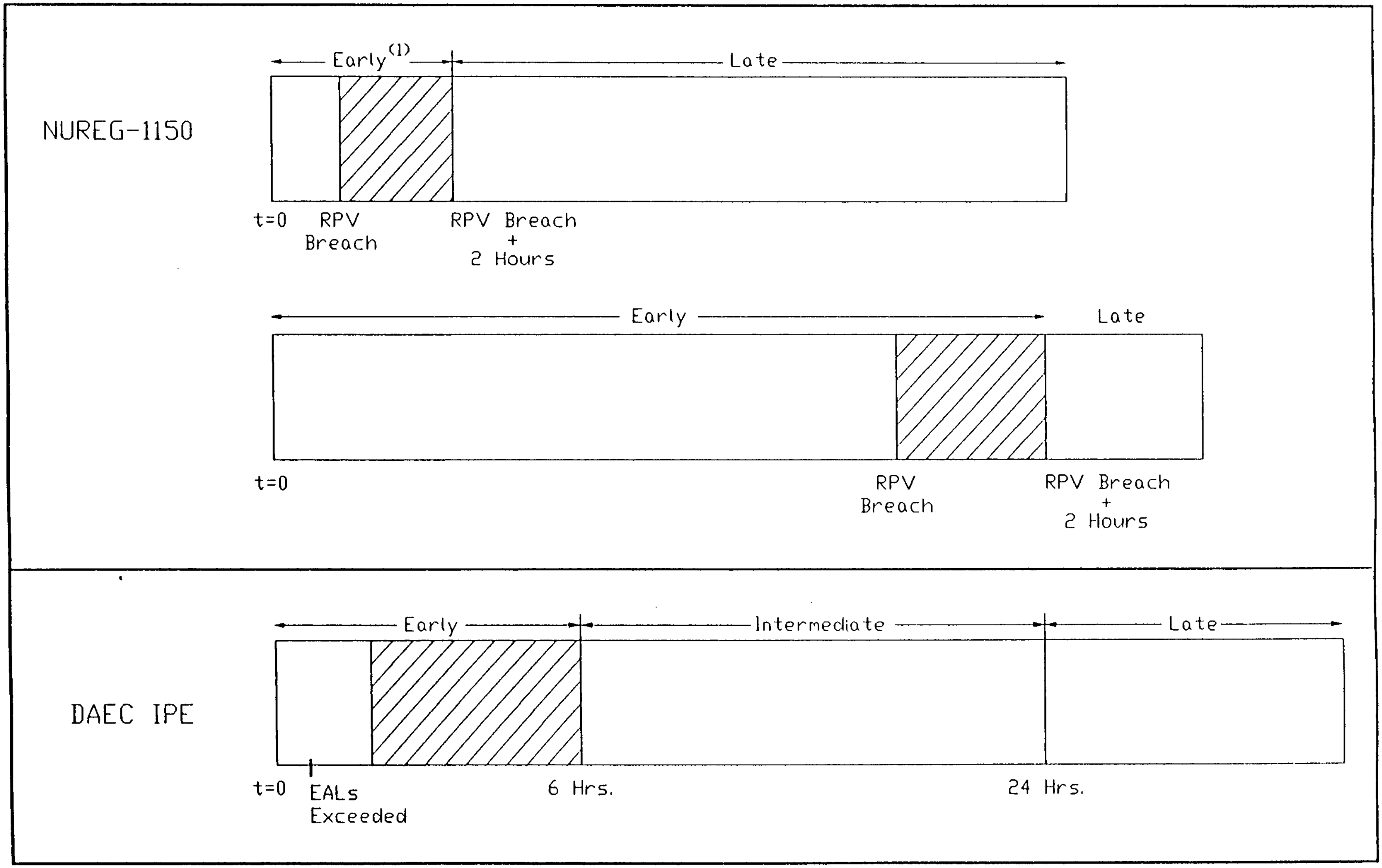
* For purposes of radionuclide release calculations, 36 hours past RPV breach by molten debris is included.

Figure 4.7-4

Example of the Definition of Timing Used
in the Radionuclide Release Categorization
(Typical for Class II Sequences)

Figure 4.7-5

Comparison of the Definition of "Early" Radionuclide Release



(1) The potential for 'early' containment failure (before or within roughly 2 hours after reactor vessel breach) is of principal conern in the NUREG-1150 risk analysis.

3) Low (L) - A radionuclide release with the potential for latent health effects.

4) Low-Low (LL) - A radionuclide release with undetectable or minor health effects.

5) Negligible (No Release) - A radionuclide release that is less than or equal to the containment design base leakage.

The quantification of the source terms associated with each of these release severity categories was accomplished through the review of existing consequence analyses performed in previous IDCOR studies, PRAs, and NRC studies containing detailed consequence modeling. To date, no single consequence analysis has evaluated all of the release paths identified in this study. Therefore, it was necessary to identify a common factor that could be used to allow the results of consequence analyses from different studies to be used in this study. The review of previous studies revealed an assumption that could be made relating release characteristics based on CsI release fraction to off-site consequences. That is, an approximate relationship exists between the fraction of CsI released and the whole-body population dose. Based on the compilation of a number of consequence analyses, one method has been developed that provides an approximate relationship between the fraction of radionuclides released and the conditional mean number of fatalities ("early" fatalities and "early" injuries occur at release fractions of the core inventory of approximately 0.1 and 0.01, respectively).

Figure 4.7-6 shows the predicted mean number of latent cancer fatalities as a function of the postulated cesium release fraction. Cesium is chosen as a measure of the source term magnitude because it delivers a substantial fraction of the total whole body population dose. A significant feature of Figure 4.7-6 is that a reduction in the source term magnitude by a given factor does not lead to a reduction in the number of latent cancer fatalities by the same factor. For low source terms, the population dose tends to be dominated by the noble gases because for the source terms considered here, the noble gas release fraction remains equal to unity even when the cesium release fraction becomes very small. This is why the curve shown in Figure 4.7-6 tends to flatten out at the left-hand end. Therefore, in the release Cs fractions of 10^{-3} to 10^{-4} the number of latent fatalities are found to be less than 1% of the latent fatalities for the highest release. In addition, the latent fatalities are dominated by the noble gas release. This grouping of releases is referred to in this analysis as the LL grouping.

Figure 4.7-7 summarizes the impacts of release magnitude on another health effects measure, i.e., the early fatalities. The line drawn through the results is a representation of where the base case results of a typical PRA might lie given "reasonable" assumptions about evacuation and the availability of medical treatment.

The wide range of uncertainties shown in Figure 4.7-7 is such that drawing conclusions about the effect of variations in the source term magnitude on public risk is not always simple. However, the most significant feature of Figure 4.7-7 is that, once the average release fraction falls below ~0.1, the conditional mean number of early fatalities is very small or zero except for a few outliers that correspond to some pessimistic assumptions.

Once the source term climbs above 0.1, however, the mean increases very rapidly because the source terms are big enough to ensure that doses above the early fatality threshold can sometimes occur among center of population a few miles (kilometers) from the site. These conclusions seem to be very robust with respect to uncertainties. Therefore, CsI fractions above 0.1 are included as the high (H) release category.

ENVIRONMENTAL EFFECTS

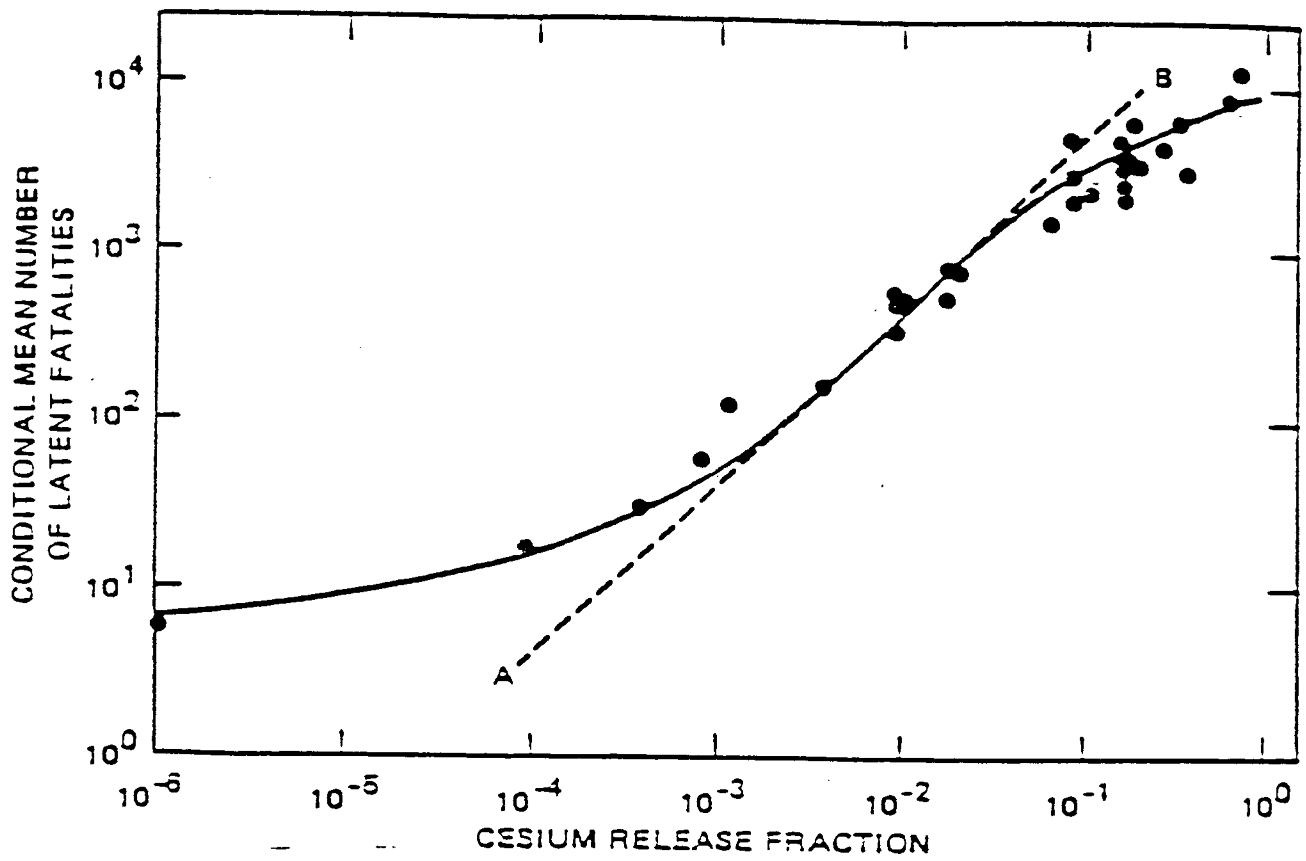


Figure 4.7-6

Sensitivity of Number of Latent Cancer Fatalities
to the Cesium Release Fraction

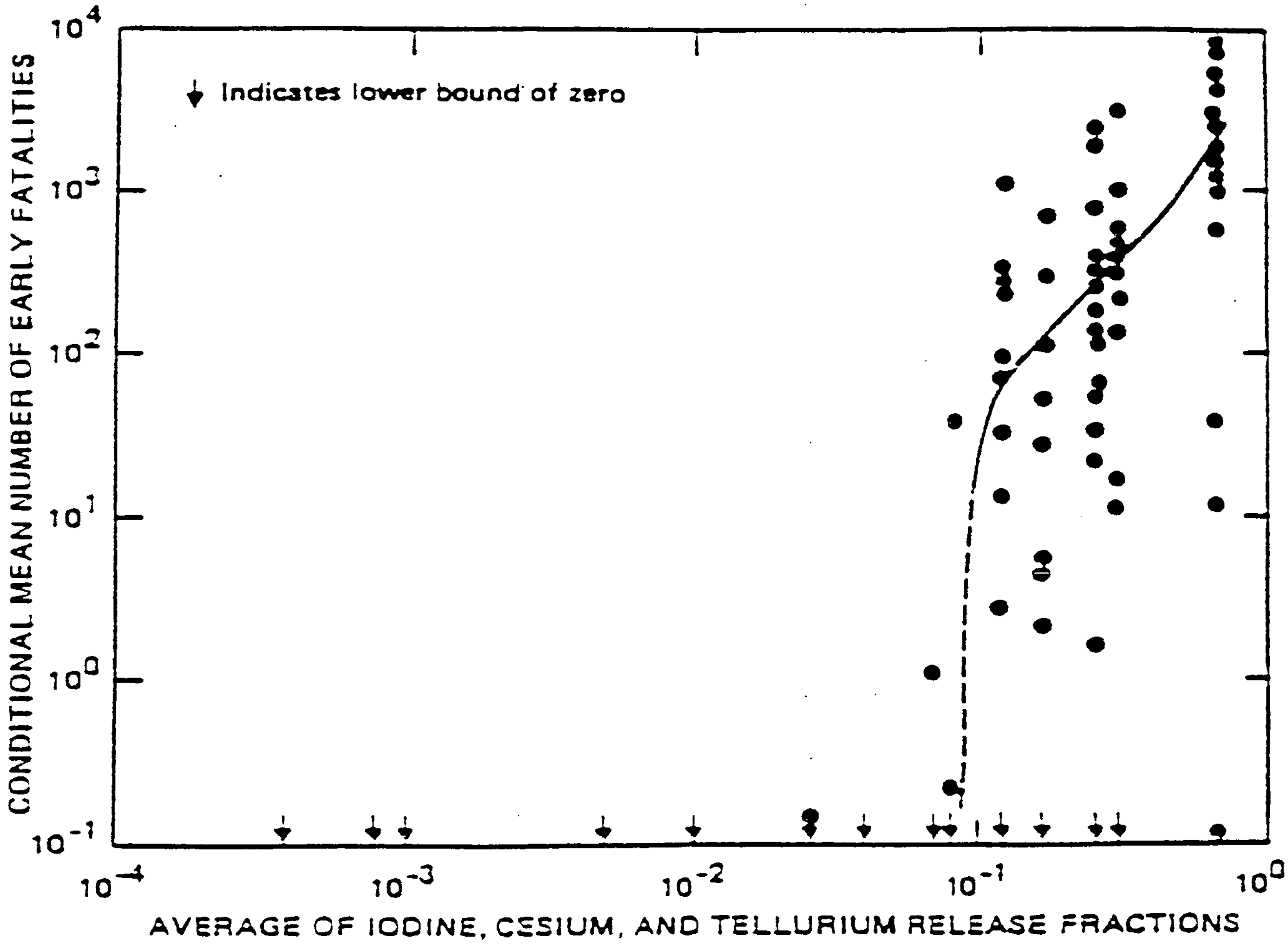


Figure 4.7-7

Sensitivity of Mean Number Early Fatalities to Source Term Magnitude

Moderate and low release categories are simple interpolations between H and LL using the approximate 1 to 1 relationship in latent health effects over this range of CsI release.

Using these insights, a numerical relationship was developed for the five release severity categories. The results of this partitioning are as follows:

Release Severity	Fraction of Release CsI Fission Products
High	greater than 10%
Moderate	1 to 10%
Low	0.1 to 1.0%
Low-Low[1]	less than 0.1%
Negligible	much less than 0.1%

This relationship allows the use of results of many consequence analyses in providing source terms from the breadth of release paths analyzed in this study. Understanding the plant specific influences on each sequence source term as affected by the various release paths allows the assignment of release severity to each of the sequences.

Plant specific deterministic calculations are also available for accident sequences that provide the other species of radionuclide releases that can cause different health effects. The purposes of the IPE include:

- Developing an awareness of margins,
- Developing an awareness of severe accident behavior,

[1] This category includes some venting sequences where only the noble gases are released.

- Identifying vulnerabilities, and
- Understanding sequences that contribute most to poor containment performance.

Thus, it is not judged necessary or desirable to provide a detailed specification on each severe accident sequence.

Because timing can be an important parameter in assessing accident management and emergency response actions, the timing of the release is included as part of the end state definition. This release timing is a surrogate for containment failure timing and is judged to be the more useful parameter.

Therefore, the containment event tree end states are characterized using a two-term matrix (i.e., severity and timing).

4.7.5 MAAP Calculational Results for Release Bins

The extensive MAAP evaluations performed for DAEC are used to enhance the knowledge of accident progression modeling expertise at IELP, to characterize the radionuclide release end states for the CETs, and to provide input on the success criteria to be applied for each CET node. Table 4.7-3 provides DAEC specific MAAP sequence calculations for radionuclide release fractions. The information contained in this table constitutes the groundwork for the classification of each DAEC CET sequence in terms of radionuclide release timing and severity.

Table 4.7-1

RELEASE SEVERITY AND TIMING CLASSIFICATION SCHEME
(SEVERITY, TIMING)

Release Severity Source Term Release Fraction		Release Timing	
Classification Category	Cs Iodide % in Release	Classification Category	Time of Release† (noble gases or CsI)
High (H)	greater than 10	Late (L)	greater than 24 hours
Moderate (M)	1 to 10	Intermediate (I)	6 to 24 hours
Low (L)	0.1 to 1	Early (E)	less than 6 hours
Low-low (LL)	less than 0.1		
No iodine (No Release)	0		

TWO TERM MATRIX

Time of Release	Magnitude of Release			
	H	M	L	LL
E	H/E	M/E	L/E	LL/E
I	H/I	M/I	L/I	LL/I
L	H/L	M/L	L/L	LL/L

†Time relative to exceeding Emergency Action Level (EAL)

Table 4.7-3 is used to characterize the containment status at the failure condition (pressure, temperature and failure size) plus the timing of events leading to vessel failure. The table includes information on the status of water injection, venting, and whether the pool has been bypassed, i.e., are there stuck open WW-DW vacuum breakers. Table 4.7-3 also summarizes the CsI distribution at the time declared as the end of the calculation (in most cases, 36 hours after the RPV breach).

From these tables many of the key insights that can be used in the severe accident evaluation can be developed. There are however subtleties associated with each computer run that make a careful scrutiny of the input data and output graphical results a prudent step. Therefore, DAEC has developed a complete set of reference books containing MAAP analyses.

Making use of these deterministic calculations, a simplified matrix can be assembled to define the end state radionuclide release magnitude. The following is a summary of that approach and provides the general guidance for disposition of the CET sequences.

Timing of Radionuclide Release

The timing of the release bins are dependent on both the Level 1 accident sequence timing and the status of the CET functional events.

Table 4.7-3

DUANE ARNOLD

MAAP SEQUENCE THERMAL HYDRAULIC/RADIONUCLIDE RELEASE SUMMARY

Rev. 15 - 10/7/92

	Sequence Timing (HRS)							Containment Conditions at Time of Failure										CsI Distribution			
Sequence Designator	RPV Depress.	Core Uncovered	Below 1/3 Core Ht.	Core Melt	Vessel Failure	Contmt. Fail/Vent	SPC On	Pressure (psia)	DW Temp (F)	Pool Temp (F)	Breach Location	Breach Size (ft^2)	Pool Bypass	Water Injection	Ped. Concrete Attack Depth (ft)	Total In-vessel H2 Prod.(Lbs)	Release Time (Hrs)	Release Duration	Mass Frac. In RB	Mass Frac. to Env.
LII-1A-1	3.38	0.69	1.06	1.28	3.38	N/A	Y	N/A	N/A	N/A	N/A	N/A	N/A	YES (CS)	0.0826	589	N/A	N/A	0.00E+00	0.00E+00
LII-1A-2	3.36	0.69	1.04	1.27	3.36	3.36(V)	N	67.7	251	135	WWV	0.1963	NO	YES (CS)	0.00	613	3.36	56.64	0.00E+00	5.93E-02
LII-1A-3	3.38	0.70	1.05	1.28	3.38	19.47	N	135	440	300	DWH	SM/Var.	NO	YES (1)	0.0614	611	19.5	40.5	1.03E-02	5.18E-02
LII-1A-3Y	3.28	0.70	1.05	1.28	3.28	19.5	N	135	418	301	DWH	SM/Var.	NO	YES (1)	0.0411	597	19.5	40.5	3.16E-03	1.64E-02
LII-1A-4	3.22	0.69	1.06	1.28	3.22	20.0	N	90.1	704	185	DWH	Var/Lg.	YES (C)	NO	2.94	619	20.0	52.0	4.04E-03	4.51E-02
LII-1A-5	N/A	0.70	1.05	1.28	3.39	Iso. failure	Y	N/A	N/A	N/A	DWH(3)	N/A	NO	YES (2)	0.551	630	3.39	56.61	9.42E-02	2.03E-01
LII-1A-6	3.34	0.70	1.05	1.28	3.44	N/A	Y	N/A	N/A	N/A	N/A	N/A	NO	YES (CS)	0.0702	614	N/A	N/A	0.00E+00	0.00E+00
LII-1A-7	3.39	0.70	1.05	1.28	3.39	Iso. failure	Y	N/A	N/A	N/A	DWH(3)	N/A	NO	YES (2)	0.506	658	3.39	56.61	7.84E-02	1.47E-01
LII-1D-1	0.59	0.62	0.64	1.39	1.92	10.14(V)	N	67.7	360	240	WWV	0.1963	NO	YES (2)	0.0811	133	10.14	59.86	0.00E+00	6.03E-03#
LII-1D-3	0.59	0.62	0.64	1.39	1.92	4.74(V)	N	67.7	325	160	WWV	0.1963	YES (D)	YES (2)	0.061	140	4.74	55.26	0.00E+00	7.45E-02
LII-1D-4	0.59	0.62	0.64	1.39	1.93	14.96 (V)	N	67.7	325	279	WWV	0.1963	NO	YES (CS)	.112	160	14.96	38.44	0.00E+00	1.30E-04#
LII-1D-5	0.59	0.62	0.64	1.39	1.91	N/A	Y	N/A	N/A	N/A	N/A	N/A	NO	YES (2)	0.0828	134	0	0	0.00E+00	0.00E+00
LII-1D-6	0.59	0.62	0.64	1.39	1.92	23.29	N	140	406	319	DWH	0.18	NO	YES (2)	0.0489	159	23.29	36.71	1.63E-02	8.42E-02
LII-1D-7	0.59	0.62	0.64	1.39	1.92	26.90	N	78.9	741	160	DWH	Var/Lg	YES (C)	NO	3.27	133	26.9	53.1	8.06E-03	2.85E-02
LII-1D-8	0.59	0.62	0.64	1.39	1.93	13.67 (V)	N	67.7	600	170	WWV	0.1963	YES (C)	NO	2.5	150	13.67	66.33	1.40E-09	6.99E-03
LII-1D-9	0.59	0.62	0.64	1.39	1.96	3.75(V)	N	67.7	300	130	WWV/DWV	0.1963	NO	HPSW(4)	0.0545	145	3.75	32.25	4.85E-03	5.98E-02
LII-1D-10	0.59	0.62	0.64	1.39	1.96	3.75(V)	N	67.7	317	137	V/DWV/RPVV(5)		NO	HPSW(4)	0.0545	145	3.75	32.25	6.26E-02	6.68E-02
LII-2A-2	N/A	44.7	46.1	46.5	51.7	41.5	N	145	398	356	DWH	Sm/Var	YES (C)	NO	0.0	768	51.7	28.3	6.03E-02	5.40E-01
LII-2A-3	5.90	39.2	40.6	41.0	44.6	42.0	N	94.7	381	325	DWH/RPVV(5)	Sm	NO	HPSW	0.0	0.0	42.9	37.1	9.71E-02	2.27E-01
LII-2T-1	5.92	37.4	38.7	39.1	45.3	40.0	N	146	360	353	DWH	0.18	YES (B)	YES	0.0	0.0	45.3	54.7	2.81E-02	1.71E-01
2T-1WW	5.92	36.6	36.0	38.5	44.1	39.4	N	144	375	353	LF/WW	2.0	YES (B)	YES	0.0	0.0	39.4	40.6	1.80E-01	8.46E-03
2T-NWW	0.592	0.615	34.2	34.6	40.2	35.1	N	42.2	367	276	DWH	0.18	NO	YES	0.0	0.0	35.1	24.9	1.65E-02	9.34E-02
2T-WWDW	0.593	0.618	34.4	34.6	40.3	35.3	N	144	371	355	DWH	0.18	NO	YES	0.0	0.0	35.3	24.7	1.45E-02	7.84E-02
LII-2T-2	5.92	37.4	38.7	39.1	45.3	40.0	N	146	380	353	DWH	0.18	NO	YES	0.0	0.0	45.3	54.7	2.84E-02	1.72E-01
2T-2WW	5.92	37.4	38.7	39.1	44.7	40.0	N	146	360	353	LF/WW*	2.0	NO	NO	0.0	687	40.0	40.0	2.23E-01	1.65E-02
LII-2T-3	0.593	0.62	22.7	23.2	27.6	18.2(V)	N	67.3	282	301	V/DWV/RPVV (5)		NO	HPSW	0.0	802	18.2	61.8	5.69E-03	1.85E-02#
LII-2L-1	LOCA	0.02	0.02	26.5	28.9	25.0	N	147	344	353	DWH	0.18	YES (A)	NO	0.0	0.0	25.0	35.0	4.69E-02	4.05E-01
2L-1WW	LOCA	25.1	0.013	27.2	29.1	25.1	N	147	343	355	LF/WW*	2.0	NO	NO	0.0	116	27.2	32.8	0.4217#	0.0284#
LII-2F-1	0.594	0.619	21.8	22.3	26.6	17 (V)	N	67.7	275	300	WWV	0.1963	YES (B)	NO	0.0	786	26.6	33.4	0.00E+00	1.55E-01

(1) - Drywell Sprays assumed lost at containment failure
(2) - Drywell Sprays used with zero fall height to simulate late injection
(3) - Failure assumed just below refuel floor (833'6" EL.)
(4) - Containment Flood Case
(5) - Deposition in condenser credited
(6) - Drywell Equipment Mass parameter input of 1 million lbs.
(7) - Suppression pool bypass is assumed after DW temp. exceeds 700 F due to vacuum breaker seal leakage.
- Release followed by a pound (#) have been adjusted to account for a DF of 10 in the pool during the time of WW vent or failure.
Y - Run names followed by a "Y" were run with Rev. 8.01 fixes in the code for comparison purposes.

LEGEND:

N/A	Not Applicable
WWV	Wetwell Vent
DWH	Drywell Head Failure
RPVV	Reactor Vessel Vent
LF	Drywell Shell Failure
DWV	Drywell Vent
DWL	Fall into torus room

(A) - Pool Bypass occurs at time = 0
(B) - Pool Bypass occurs at vessel failure
(C) - Pool Bypass occurs when DW gas Temp. > 700 F
(D) - Pool Bypass occurs during fuel melt progression

Table 4.7-3 (con't)

DUANE ARNOLD

MAAP SEQUENCE THERMAL HYDRAULIC/RADIONUCLIDE RELEASE SUMMARY
Rev. 15 - 10/7/92

Sequence Timing (HRS)								Containment Conditions at Time of Failure										CsI Distribution			
Sequence Designator	RPV Depress.	Core Uncovered	Below 1/3 Core Ht.	Core Melt	Vessel Failure	Contmt. Fail/Vent	SPC On	Pressure (psia)	DW Temp (F)	Pool Temp (F)	Breach Location	Breach Size (ft^2)	Pool Bypass	Water Injection	Ped. Concrete Attack Depth (Ft)	Total In-vessel H2 Prod.(Lbs)	Release Time (Hrs)	Release Duration	Mass Frac. in RB	Mass Frac. to Env.
LII-3A-1	LOCA	0.015	0.001	0.35	1.0	N/A	Y	N/A	N/A	N/A	N/A	N/A	NO	YES	0.0	4.44	N/A	N/A	0.00E+00	0.00E+00
LII-3A-2	LOCA	0.015	0.001	0.35	1.0	18 (V)	N	67.7	550	270	WW	0.1963	NO	YES	0.0	8.42	18	22	0.00E+00	1.71E-02#
LII-3A-3	LOCA	0.015	0.001	0.35	1.0	26.2	N	105	635	320	DWH	Sm/Var	NO	YES	0.0	1.13	26.2	13.8	3.54E-02	2.67E-01
LII-3B-1	LOCA	No Core Melt					Y													
LII-3B-1Y	LOCA	No Core Melt					Y													
LII-3B-2	LOCA	No Core Melt					Y													
LII-3C-1	LOCA	58 sec	35 sec	0.31	0.66	66 sec (V)	N	19 (8)	395 (8)	101 (8)	WWV/DWH	0.1963	YES (A)	NO	0.0	416	68 (sec)	40	2.24E-08	1.88E-01
LII-4A-1	1.66	0.118	0.993	1.39	4.20	0.964	Y	147	344	357	DWH	Sm/Var	YES (A)	YES	0.0	675	4.20	43.80	2.83E-03	1.62E-02
LII-4A-1Y	1.69	.117	1.00	1.39	4.19	0.965	Y	147	343	357	DWH	Sm/Var	YES (A)	YES	0.0	663	419	43.81	3.21E-03	1.97E-02
LII-4A1LD	0.988	0.116	1.00	1.55	3.35	0.964	Y	147	244	357	DWH	2.0	YES (A)	YES	0.0	507	3.35	44.65	2.07E-02	1.57E-01
4A1NVB	0.401	0.098	0.418	0.786	4.34	0.4(V)	Y	22	200	240	WWV	0.1963	NO	YES	0.0	466	0.40	47.6	1.41E-03	7.24E-03
LII-4A1 (6)	1.77	0.116	0.995	1.39	4.24	0.945	Y	146	356	356	DWH	Sm/Var	YES (A)	YES	0.0	633	4.24	43.76	5.51E-03	3.23E-02
LII-4ALF	1.66	0.118	0.993	1.39	4.20	0.964	Y	147	344	357	DWH/LF	0.18/2.0	YES (A)	YES	0.0	675	4.20	43.80	1.40E-02	1.27E-02
4AWWLF	4.53	0.116	0.967	1.42	4.53	0.964	Y	147	344	357	WW/LF	2.0/2.0	YES (A)	YES	0.0	675	4.53	43.47	8.32E-02#	2.78E-02#
4WWLFNB	4.49	0.098	1.01	1.43	4.49	0.98	Y	148	348	357	WW/LF	2.0/2.0	NO	YES	0.0	561	4.49	43.51	7.59E-02#	1.42E-02#
LII-V-1	LOCA	0.09	0.12	0.51	1.86	Iso. failure	Y	24.9 peak	578 peak	96.1 peak	HPCI RM.	2.18	NO	NO	0.0	398	1.86	38.14	0.497	0.249
LII-V-1LL	LOCA	0.06	0.237	0.439	1.92	Iso. failure	Y	25.5 peak	575 peak	96.2 peak	CRNR RM.	5.26	NO	NO	0.0	490	1.92	38.08	0.643	0.095
LII-V-1LLY	LOCA	0.06	0.230	0.443	1.90	Iso. failure	Y	25.2 peak	584 peak	96.1 peak	CRNR RM.	5.28	NO	NO	0.0	492	1.90	38.10	0.615	0.110
EQ-1A4	3.38	0.70	1.05	1.26	3.38	16.7	N	88.5	709	185	DWH	2.0	NO	NO	3.97	611	18.7	41.3	0.0152	0.138
EQ-1A4TR	3.38	0.70	1.05	1.28	3.38	3.81	N	70.4	563	140	LF	2.0	NO	NO	3.98	611	3.81	56.19	0.334	0.0719
EQ-1A4LF	3.38	0.70	1.05	1.28	3.38	18.7	N	88.5	709	185	DWL	2.0	NO	NO	3.97	611	18.7	41.3	0.102	0.0422
EQ-1DLF	0.594	0.619	0.639	1.39	1.92	2.04	N	20.4	212	133	LF*	2.0	NO	NO	.019	133	2.04	37.96	0.377	0.046
EQ-2TLF	0.592	0.618	29.4	29.8	33.6	29.0	N	146	366	358	DWH/LF	2.0	NO	YES	2.59	663	29.4	30.6	0.130	0.127
EQ-2T2WW	5.92	37.4	38.7	39.1	45.3	40.0	N	146	380	353	WW/LF	2.0	NO	YES	3.45	887	40.0	60.0	0.223	0.0165

(1) - Drywell Sprays assumed lost at containment failure
(2) - Drywell Sprays used with zero fall height to simulate late injection
(3) - Failure assumed just below refuel floor (833'6" EL.)
(4) - Containment Flood Case
(5) - Deposition in condenser credited
(6) - Drywell Equipment Mass parameter input of 1 million lbs.
(7) - Suppression pool bypass is assumed after DW temp. exceeds 700 F due to vacuum breaker seal leakage.
(8) - Conditions in table taken at the time of WW vent (68 sec)
- Release followed by a pound (#) have been adjusted to account for a DF of 10 in the pool during the time of WW vent or failure.
Y - Run names followed by a "Y" were run with Rev. 8.01 fixes in the code for comparison purposes.

LEGEND:

N/A	Not Applicable
WWV	Wetwell Vent
DWH	Drywell Head Failure
RPVV	Reactor Vessel Vent
LF	Drywell Shell Failure
DWV	Drywell Vent
DWL	Fail into Torus room

(A) - Pool Bypass occurs at time = 0
(B) - Pool Bypass occurs at vessel failure
(C) - Pool Bypass occurs when DW gas Temp. > 700 F
(D) - Pool Bypass occurs during fuel melt progression

Table 4.7-3 (con't)

DUANE ARNOLD

MAAP SEQUENCE THERMAL HYDRAULIC/RADIONUCLIDE RELEASE SUMMARY
Rev. 15 - 10/7/92

Sequence Timing (HRS)								Containment Conditions at Time of Failure										CsI Distribution			
Sequence Designator	RPV Depress.	Core Uncovered	Below 1/3 Core Ht.	Core Melt	Vessel Failure	Contmt. Fail/Vent	SPC On	Pressure (psia)	DW Temp (F)	Pool Temp (F)	Breach Location	Breach Size (ft^2)	Pool Bypass	Water Injection	Ped. Concrete Attack Depth (Ft)	Total In-vessel H2 Prod.(Lbs)	Release Time (Hrs)	Release Duration	Mass Frac. in RB	Mass Frac. to Env.
LII-1DLF	0.594	0.619	0.639	1.39	1.92	2.04	N	20.4	212	133	LF*	2.0	NO	NO	.019	133	2.04	37.96	0.377	0.046
LII-1DLFA	3.38	0.70	1.05	1.28	3.38	3.50	N	28.7	253	135	LF*	2.0	NO	NO	0.0143	611	3.50	36.50	0.343	0.0719
LII-1DLFB	0.594	0.619	0.639	1.39	1.92	2.04	N	20.4	212	133	LF*	2.0	NO	NO	0.021	133	2.04	37.96	0.348	0.050
LII-1A1A	3.44	0.70	1.05	1.28	3.44	N/A	Y	N/A	N/A	N/A	N/A	N/A	NO	YES	N/A	N/A	N/A	N/A	N/A	N/A
LII-1A03A	3.38	0.70	1.05	1.28	3.44	16.1	N	137	424	276	DWH	Var.	NO	YES (2)	3.67	605	16.1	53.9	2.51E-03	1.26E-02
LII-3A01A	0.0	0.016	.86E-3	0.356	0.99	18.4	N	63.5	797	166	DWH	Var.	NO	NO	4.12	249	18.4	21.6	0.0712	0.343
LII-1D11	0.594	0.619	0.639	1.39	1.92	2.03	N	20.4	212	133	LF	2.0	NO	YES (2)	0.023	133	2.03	57.97	6.40E-03	1.84E-04
LII-1D12	0.594	0.619	0.639	1.39	1.92	2.03	N	20.4	212	133	LF	0.18	NO	YES (2)	0.023	133	2.03	57.97	2.15E-03	1.72E-05
LII-1D13	0.594	0.619	0.639	1.39	1.92	2.03	N	20.4	212	133	LF*	2.0	NO	NO	0.0191	133	2.03	57.97	0.377	0.046
LII-1D13Y	0.594	0.619	0.638	1.46	1.81	1.93	N	20.4	212	133	LF*	2.0	NO	NO	0.0	130	1.93	58.07	0.249	0.0289
LII-1D14	0.594	0.619	0.639	1.39	1.92	2.03	N	20.4	212	133	LF*	0.18	NO	NO	0.0191	133	2.03	57.97	0.219	0.0156
LII-1A3LD	3.38	0.07	1.05	1.28	3.38	19.5	N	136	441	300	DWH	2.0	NO	YES (2)	3.16	611	19.5	40.5	7.59E-03	2.21E-02
LII-1DLFF	3.34	0.70	1.05	1.29	3.34	3.45	N	28.6	239	133	LF*	2.0	NO	NO	3.68	622	3.45	36.55	0.325	3.35E-02
LII-1DLFE	3.38	0.70	1.05	1.28	3.36	3.50	N	28.7	253	135	LF*	2.0	NO	NO	0.0143	611	3.50	36.50	0.343	0.072
1D-LFE2(6)	3.33	0.70	1.05	1.28	3.33	3.45	N	28.9	259	135	LF*	2.0	NO	NO	0.0134	630	3.33	36.66	0.375	0.0474
LII-1DLFD	3.38	0.70	1.05	1.28	3.36	3.50	N	28.7	253	135	LF*	2.0	NO	NO	0.0143	611	3.50	36.50	0.332	0.094
LII-1DLFC	3.38	0.70	1.05	1.28	3.38	3.38	N	28.7	253	135	LF*	2.0	NO	NO	0.0143	611	3.38	36.62	0.338	0.047
LII-1D13A	0.594	0.619	0.639	1.39	1.92	2.04	N	20.4	212	133	LF*	10.0	NO	NO	3.98	133	2.04	57.96	0.386	0.027
LII-1A3LW	3.38	0.70	1.05	1.28	3.38	19.5	N	136	441	300	WW**	2.0	NO	NO	0.015	611	19.5	40.5	1.07E-04#	5.31E-04#
3A01B (6)	0.0	0.015	8.6E-2	0.35	1.00	35.7	Y	41.8	834	149	DWH	Var.	NO	NO	0.0	4.49	35.7	4.3	1.97E-04	5.55E-04
LII-1A02C	2.25	0.70	1.05	1.28	2.25	2.25	N	67.7	233	130	DWV	0.127	NO	YES	0.591	1490	2.25	15.5	2.83E-02	0.366
1A02C-W	2.25	0.70	1.05	1.26	2.25	2.25	N	67.7	240	130	WWV	0.1963	NO	YES	0.0167	1490	2.25	57.75	0.00E+00	6.21E-02
LII-1A02B	2.64	0.70	1.05	1.29	2.64	12.36	N	67.7	360	260	DWV	0.127	NO	YES	0.024	184	12.36	11.75	1.18E-04	2.09E-03
1A02B-W	2.64	0.70	1.05	1.29	2.84	12.36	N	67.7	380	260	WWV	0.1963	NO	YES	0.023	184	12.36	11.64	0.00E+00	1.97E-03
LII-1A02A	3.38	0.70	1.05	1.28	3.38	3.38	N	67.7	253	135	DWV	0.127	NO	YES	0.043	611	3.38	38.62	5.28E-02	0.282
1A02A-W	3.38	0.70	1.05	1.28	3.36	3.38	N	67.7	253	135	WWV	0.1963	NO	YES	0.0501	611	3.38	56.62	3.73E-09	3.72E-02
1DLFG (6)	3.33	0.70	1.05	1.28	3.33	3.47	N	28.9	259	135	LF	2.0	NO	NO	0.0134	630	3.47	36.53	3.65E-01	6.06E-02
1A03LWA	3.38	0.70	1.05	1.28	3.38	19.5	N	136	441	300	WW	2.0	NO	NO	0.015	611	19.5	40.5	7.91E-04#	2.90E-05#
1A04-SRV	2.43	0.70	1.05	1.51	2.43	21.95	N	96.6	676	202	DWH	0.127	NO	NO	3.95	519	21.95	38.05	2.68E-03	4.25E-02

(1) - Drywell Sprays assumed lost at containment failure
(2) - Drywell Sprays used with zero fall height to simulate late injection
(3) - Failure assumed just below refuel floor (833'6" EL.)
(4) - Containment Flood Case
(5) - Deposition in condenser credited
(6) - Drywell Equipment Mass parameter input of 1 million lbs.
(7) - Suppression pool bypass is assumed after DW temp. exceeds 700 F due to vacuum breaker seal leakage.
* - Late drywell head failure available but did not occur
** - LII-1A03LW released FP's to the elevation below the refuel floor
# - Release followed by a pound (#) have been adjusted to account for a DF of 10 in the pool during the time of WW vent or failure.
Y - Run names followed by a "Y" were run with Rev. 8.01 fixes in the code for comparison purposes.

LEGEND:

N/A	Not Applicable
WWV	Wetwell Vent
DWH	Drywell Head Failure
RPVV	Reactor Vessel Vent
LF	Drywell Shell Failure
DWV	Drywell Vent
DWL	Fail into Torus room

(A) - Pool Bypass occurs at time = 0
(B) - Pool Bypass occurs at vessel failure
(C) - Pool Bypass occurs when DW gas Temp. > 700 F

First, the Level 1 accident sequences have the following effects on radionuclide release timing:

SEQUENCE	INFERRED TIMING
· TW (Class II)	L
· ATWS (Class IV)	E
· SBO (Late)[1]	I or L[3]
· SBO (Early)[2]	E or I or L[3]
· TQUX (Class IA)	E or I or L[3]
· TQUV (Class ID)	E or I or L[3]
· LOCA plus vapor suppression failure	E

SEQUENCE	INFERRED TIMING
· TW (Class II) with vent	L
· ATWS (Class IV) with vent	E
· SBO (Late)[4] with vent	I or L[6]
· SBO (Early)[5] with vent	E or I or L[3]
· TQUX (Class IA) with vent	E or I or L[3]
· TQUV (Class ID) with vent	I or L[3]
· LOCA plus vapor suppression failure with vent	E

Overlayed on top of the Level 1 sequence characteristic time are effects resulting from the status of the CET.

---

[1] SBO related to initial successful injection but subsequent loss of injection due to battery depletion, high pool temperature, etc.

[2] SBO related to loss of all injection in 0-2 hour time frame.

[3] Timing dependent upon subsequent CET top events.

[4] SBO related to initial successful injection but subsequent loss of injection due to battery depletion, high pool temperature, etc.

[5] SBO related to loss of all injection in 0-2 hour time frame.

[6] Timing dependent upon subsequent CET top events.

Based on the plant specific DAEC MAAP calculations, Table 4.7-4 summarizes the Level 2 CET top event and the Level 1 accident class dependencies that can alter the timings for individual CET sequences. Using these dependencies, the binning rules for radionuclide release categories at the end of every sequence can be written.

Magnitude of Radionuclide Release

The rules for assigning release magnitude categories are described below:

1. There are three fundamental variables

   - Initial containment failure mode,
   - Water availability, and
   - Reactor building effectiveness.

   An evaluation of these variables, to a large degree, determines the release magnitude. Tables 4.7-5 and 4.7-6 summarize these deterministic calculated release magnitudes in terms of these three fundamental variables.

   In addition to the containment failure modes identified in Tables 4.7-5 and 6, it is also necessary to estimate the source term for severe accidents for which the containment remains substantially intact. A recent estimate of source terms by members of NRC, AEOD and NRR staffs indicates (with the containment intact and leaking at its maximum technical specification

Table 4.7-4

SEQUENCE TIMING SUMMARY

Sequence	Sequence Timing Alone	FAILED CONTAINMENT CONDITIONS[1,2]						
		$\overline{GV}$	CZ	SI	TD	CX	HR	HR & CV
IA	---	E LII-1A-7	E LII-1A-7	E LII-1A-4	I LII-1A-4	E LII-1D-9	I LII-1A-3 LII-1A-3	I LII-1A-3D
IB Early[3]	---	E LII-1A-7	E LII-1A-7	E LII-1A-4	I LII-1A-4	E LII-ID-9	I LII-1A-3LWA LII-1A-3	I LII-1A-3D
IB Late[3]	---	I LII-1A-3	I LII-1A-7	I LII-1A-4	I LII-1A-4	E LII-ID-9	I LII-1A-3LWA LII-1D-3	L LII-1A-3
IC[3]	---	E LII-1A-7	E LII-1A-7	E LII-1A-4	I LII-1A-4	E LII-ID-9	I LII-1A-3LWA LII-1D-3	I LII-1A-3D
ID	---	E LII-1A-7	E LII-1D-8	E LII-1D-8	I LII-1D-8	E LII-ID-9	I LII-1D-6	I LII-1D-6
IIA	LATE[4] LII-2A-2, LII-2A-3							
IIV	LATE[4]							
IIT	LATE[4] LII-IIT-1, LII-IIT-2			INTERMEDIATE LII-IIT-3				
IIIA	---	E LII-IIIC-1	E LII-IIIC-1	E LII-IIIC-1	I LII-3A-3[5]	E LII-1D-9 LII-1D-10	L LII-IIIA-3	L LII-IIIA-3
IIIB	---	E LII-IIIC-1	E LII-IIIC-1	E LII-IIIC-1	I LII-3A-3[5]	E LII-1D-9 LII-1D-10	I LII-IIIA-3	I LII-IIIA-3
IIIC	---	E LII-IIIC-1	E LII-IIIC-1	E LII-IIIC-1	I LII-3A-3[5]	E LII-1D-9 LII-1D-10	I LII-IIIA-3	I LII-IIIA-3
IIID	EARLY[4]							
IVA	EARLY[4]							
IVL	EARLY[4], LII-IVA-1							
IVT	EARLY[4]							
IVV	EARLY[4]							
V	EARLY[4] LII-V-1							

Notes to Table 4.7-4

(1) GV - Combustible Gas Venting
CZ & CX - Energetic Failures
TD - Failure of debris cooling
CV & HR - Failure of containment heat removal
E - Early 0 - 6 Hours
I - Intermediate 6 - 24 Hours
L - Late > 24 Hours

(2) This table is interpreted as if each containment failure mode is treated separately; i.e., it is assumed that one, and only one, failure mode occurs. The timing then reflects this particular failure mode. No venting is included unless explicitly identified by the "failure mode".

(3) For these accident classes, no MAAP runs were performed. Timings are chosen using IA and ID runs as surrogates.

(4) The timing is set by definition of these accident sequences.

(5) Conservatively binned relative to the MAAP case.

Table 4.7-5

DAEC CET NODAL EFFECTS ON SOURCE TERM MAGNITUDE
(No Shell Failure Included)

Initial[1] Containment Failure Mode	Water Availability to the Molten Debris	Reactor Building Effectiveness	Resulting Source Term Magnitude	MAAP Reference Case
<u>Drywell</u>				
Large DW	Yes	Yes	M[3][1] - L[1]	LII-1A-3LD LII-1D-6
Large DW	Yes	No	H[3] - M[4]	LII-1A-3LD LII-1D-6
Large DW	No	Yes	See Shell Failure	Not Required
Large DW	No	No	See Shell Failure	Not Required
Small DW	Yes	Yes	M[3][1] - L[1]	LII-1A-3 LII-1D-6
Small DW	Yes	No	M[3] - M[4]	LII-3A-3 LII-1D-6
Small DW	No	Yes	See Shell Failure	Not Required
Small DW	No	No	See Shell Failure	Not Required
<u>Wetwell</u>				
Wetwell Failure w/no Suppression Pool Bypass	Yes	Yes	LL	LII-1A3LWA
Wetwell Failure w/no Suppression Pool Bypass	Yes	No	LL	LII-1A-3LW
Wetwell Failure w/no Suppression Pool Bypass	No	Yes	LL[5]	LII-1A3LWA
Wetwell Failure w/no Suppression Pool Bypass	No	No	LL[5]	LII-1A-3LW

Table 4.7-5

DAEC CET NODAL EFFECTS ON SOURCE TERM MAGNITUDE
(No Shell Failure Included)

Initial(1) Containment Failure Mode	Water Availability to the Molten Debris	Reactor Building Effectiveness	Resulting Source Term Magnitude	MAAP Reference Case
Wetwell Vent with no Suppression Pool Bypass	Yes	No(8)	L	LII-1D-1
Wetwell Vent with no Suppression Pool Bypass	No	No(8)	L	LII-1A3LWA
Wetwell Vent with Suppression Pool Bypass	Yes	No(8)	M	LII-1D-3
Wetwell Vent with Suppression Pool Bypass	No	No(8)	H	LII-IIIC-1

Table 4.7-6

DAEC CET NODAL EFFECTS ON SOURCE TERM MAGNITUDE
(Coincident Shell Failure Included)

Initial[1] Containment Failure Mode	Water Availability to the Molten Debris	Reactor Building Effectiveness	Resulting Source Term Magnitude	MAAP Reference Case
Shell Failure Only				
Large DW	Yes	Yes	LL	LII-1D-11
Large DW	Yes	No	L[6]	LII-1D-11
Large DW	No	Yes	M[3]	LII-1D-LFA LII-1D-13
Large DW	No	No	H	LII-1D-LFA
Small DW	Yes	Yes	LL	LII-1D-12
Small DW	Yes	No	L[6]	LII-1D-11
Small DW	No	Yes	M[3]	LII-1D-LFA LII-1D-14
Small DW	No	No	H	LII-1D-LFA

Table 4.7-6

DAEC CET NODAL EFFECTS ON SOURCE TERM MAGNITUDE
(Coincident Shell Failure Included)

Initial[(1)] Containment Failure Mode	Water Availability to the Molten Debris	Reactor Building Effectiveness	Resulting Source Term Magnitude	MAAP Reference Case
Drywell				
Large DW	Yes	Yes	*	Not Observed
Large DW	Yes	No	*	Not Observed
Large DW	No	Yes	M	LII-1A-4 LII-1D-14
Large DW	No	No	H	LII-1A-4 LII-1D-14
Small DW	Yes	Yes	*	Not Observed
Small DW	Yes	No	M	LII-4A-LF
Small DW	No	Yes	M	LII-1A-3 LII-1D-14
Small DW	No	No	H	LII-1A-3 LII-1D-14

Table 4.7-6

DAEC CET NODAL EFFECTS ON SOURCE TERM MAGNITUDE
(Coincident Shell Failure Included)

Initial[1] Containment Failure Mode	Water Availability to the Molten Debris	Reactor Building Effectiveness	Resulting Source Term Magnitude	MAAP Reference Case
Wetwell				
Wetwell Vent or Failure w/no Suppression Pool Bypass[7]	Yes	Yes	L	Judgement
Wetwell Vent or Failure w/no Suppression Pool Bypass[7]	Yes	No	M	Judgement
Wetwell Vent or Failure w/no Suppression Pool Bypass[7]	No	Yes	M	2T-2WW 2L-1WW
Wetwell Vent or Failure w/no Suppression Pool Bypass[7]	No	No	H[6]	2T-2WW 2L-1WW
Wetwell Vent or Failure with Suppression Pool Bypass[7]	Yes	Yes (WW Failure Only)[6]	L	LII-4A-WWLF
Wetwell Vent or Failure with Suppression Pool Bypass[7]	Yes	No	M[6]	LII-4A-WWLF
Wetwell Vent or Failure with Suppression Pool Bypass[7]	No	Yes	M	EQ-2T-2WW
Wetwell Vent or Failure with Suppression Pool Bypass[7]	No	No	H[6]	EQ-2T-2WW

Notes to Tables 4.7-5 and 4.7-6

(1) Reactor building was calculated as ineffective in the MAAP case performed. Therefore, under the low probability case of the reactor building being effective the release is approximately 1 order of magnitude lower.

(2) Reactor building is calculated as effective in the referenced MAAP case; if it is in fact ineffective the release would increase by a factor of 10.

(3) If the water injection is lost after containment failure it appears likely that a high release may occur (see LII-1D-06).

(4) For cases with makeup continuing after containment failure, it is assumed that late revaporization can be substantially delayed or prevented.

(5) These are sequences that are not probabilistically significant because of potential shell failure with no water available.

(6) This is the release magnitude for an assumed reactor building DF = 1.

(7) Class II and IV cases only can have wetwell failure or vent pressure and shell failure.

(8) Vent cases have no reactor building effectiveness.

leakage rate) that the escape fraction would be 2E-4 for the initial one hour of the release.

Therefore, with no reactor building filtration or holdup effectiveness, the leakage escape fraction could translate into a maximum CsI release fraction of 0.0048 to the environment over 24 hours (i.e., the Low (L) category) assuming the entire possible inventory of CsI was initially released to containment. If the reactor building remains effective in removing some of the radionuclides through condensation, inertial deposition, or gravitational settling, then the release fraction is estimated to be between .0001 and .001, i.e., the Low-Low (LL) category. If SGTS is operational essentially no release is expected.

There are exceptions and variations that can be important in the assessment that also create variability in the release magnitude. The remaining rules 2 through 13 are these exceptions.

2. There are energetic failures of the containment drywell at approximately the time of RPV failure. It is assumed based on a spectrum of MAAP analyses that a sufficient fraction of CsI is airborne for such cases to result in a large CsI release. Therefore, sequences involving CZ and CX functional node failures are ranked as High (H) release categories.

3. Containment isolation failure is treated conservatively in the assignment of radionuclide release end state. Sequences are assigned to a High (H) release bin in the case of IS failure even though:

   - The failures could be relatively small,

   - The failures could be from the torus airspace, or

- The failures could be into closed or filtered systems (e.g., SGTS).

Nevertheless, the IS failure, assumed to bypass the reactor building, leads to a High release.

4. For sequences in which multiple containment failures occur, there are complex interactions that may result in modifying the radionuclide release to the environment. The following are the principal multiple failures affecting the assignment in the DAEC evaluation:

   - Sequences involving wetwell vent or airspace failure followed by shell failure is judged to be dominated by shell failure when characterizing the release.

     Shell failure releases are found to be as follows:

     - Moderate (M) with reactor building effective[1]

     - High (H) with reactor building ineffective[2]

   - Any previous failures followed by drywell head failure is found to lead to a high release (conservative estimate based on LII-3C-01.)

   - The treatment of radionuclide releases for ATWS is based on the following:

[1] Observed in DAEC plant specific MAAP runs.

[2] No such cases found, but it is assumed that it is possible to have an ineffective reactor building for DF.

- For transient induced failure to scram scenarios, there are a number of DAEC MAAP deterministic evaluations that indicate that the character of the release is substantially lower than for non-ATWS cases. The ATWS sequences with initial wetwell failure are found to result in no late drywell head failure on temperature and therefore, the release is limited to a Medium or Low release depending on the reactor building effectiveness.

- MAAP runs for ATWS (with no LOCA present) result in Medium releases or lower for

  -- initial drywell head failure (LII-IVA-1)
  -- initial drywell head failure and subsequent drywell shell failure (LII-IVA-LF)
  -- initial wetwell failure and subsequent drywell shell failure (LII-IVA-WWLF)

- Because no specific MAAP runs were performed for ATWS with a large or medium LOCA present, these were conservatively binned to the High category if drywell head failure also occurred.

- Failure to scram cases with premature loss of injection (i.e., well before containment failure) lead to higher releases. Therefore, these are binned as the Class IA sequences which show a similar trend.

5. Events during which the containment flood contingency is successfully implemented and completed are found to have the possibility of direct releases from the RPV to the condenser and from the drywell through the drywell vent. Conservative estimates based on DAEC MAAP calculations are used to characterize the release categories as follows:

- Successful containment flood, with lack of reactor building, condenser, or turbine building effectiveness results in a High (H) or Moderate (M) release.

6. For DAEC, there is a strong influence of the small core power in a normal size containment.

Plant	DW Free Vol. MWe	WW Free Volume MWe
DAEC	203	175
Peach Bottom	149	124

This leads to very few MAAP cases for which the radionuclide release is calculated to be High (H). MAAP cases 1A-4 and ID7 demonstrate this point. These have a DW equipment mass of 2.7 lbm included in the calculation. This is assumed to be a major benefit of this low power core. The cases for which a High release is calculated are:

- Drywell isolation failure with no RPV or containment injection (e.g., IA-5, IA-7)

- ISLOCA (HPCI room) (LII-V-1)

- Class II sequences with a DW/H failure

- LOCA and DW/H failure (LII-3C-01, LII-3A03)

Nevertheless, all cases (other than ATWS) with no water injection to the containment are conservatively assumed to lead to both a drywell shell failure and a drywell head failure. Therefore, these scenarios are treated as resulting in drywell head failure.

However, MAAP used a drywell equipment mass of 2.7 million lbm. This figure appears to be optimistic and has a significant impact on the results. DAEC MAAP runs with shell liner failure (refer to LII-ID-13 and LII-ID-14) result in drywell temperature under 750°F and no subsequent drywell head failure. Analogous MAAP runs for Fermi 2, a Mark I reactor using an estimate of 0.5 million lbm as the drywell equipment mass, yielded temperatures above 1000°F. A sensitivity analysis for DAEC which estimated the drywell equipment mass to be 1 million lbm. (case LII-3A-01) resulted in drywell temperatures in excess of 830°F (an increase of 130°F from the base case). The extreme temperatures resulting from no water injection to ex-vessel debris is conservatively judged to be sufficient for late containment failure.

7. Scenarios involving wetwell airspace or wetwell vent failure are treated as scrubbed releases, and are assigned a severity class of Low-Low (LL). (See MAAP runs ID-1, ID-4, and 3A-2).

8. Wetwell failures below the water line result in the torus water level equilibrating inside and outside the torus to cover the breach (i.e., assumed at the ECCS suction). Therefore, the RB node for wetwell failures below the water line is used to distinguish whether the scrubbing of the source term was effective in reducing

the magnitude of the release. For the scrubbed case, the release magnitude is the same as a wetwell vent case without suppression pool bypass.

9. One of the most complex conditions is related to consequential containment failures associated with the termination of injection when containment fails. Following failure of RPV injection, more benign containment failures (e.g., wetwell failures) will eventually progress to a temperature induced drywell failure. Consequently, the release is usually relatively small during the period in which the water and saturation conditions exist inside containment. At the time of containment dryout, which tends to be late in such a sequence, the release magnitude could increase if sufficient material (located on containment structural surfaces) can be revaporized as the airspace temperature increases.

   The applicable DAEC MAAP cases are ID-11 and ID-13. For these cases involving terminating injection after containment failure, the following conclusions can be made:

   - With the reactor building effective (RB = S), the release category is a Low-Low (LL). This is treated more conservatively in the CET quantification, i.e., it is assigned to the Low (L) category.

   - With the reactor building ineffective (RB = F), the release category is assigned to a Medium release (M).

10. Use of the hard pipe vent results in bypassing the reactor building. Therefore, the reactor building node is not considered in sequences where successful wetwell venting has occurred (CV = success).

11. Suppression pool bypass is modeled in two ways in the CET:

- First, as a method of potential containment challenge during blowdown (see CZ node). Failure of this function results in a High release.

- Second, it is modeled as a failure of the vacuum breaker in the wetwell to drywell interface that allows radionuclides to bypass the suppression pool. The impact of bypassing the suppression pool is modeled as an increase of a factor of 10 in the radionuclide release.

12. Containment venting with no suppression pool bypass is calculated to result in a Low-Low release for cases with a subcooled pool. Higher releases are found possible when a saturated pool is present during the venting release process. A Low release category is used to characterize releases with venting and a saturated pool. (MAAP Case LII-ID-1)

4.8 ACCIDENT MANAGEMENT INSIGHTS: QUALITATIVE SUMMARY OF IMPACT ON RADIONUCLIDE RELEASE DUE TO FUNCTIONAL TOP EVENTS

This section summarizes the insights formulated based on the DAEC specific deterministic calculations and probabilistic modeling of severe accident progression.

As part of the deterministic assessment of containment response, MAAP thermal hydraulic calculations of a wide spectrum of postulated accident scenarios have been performed. These postulated accidents are prescribed based on past experience with other BWR PRAs and the knowledge of dominant phenomenological and system effects on containment response or radionuclide release severity or timing.

The insights that are discussed are related to the following key phenomena and functional effects:

- Containment Isolation
- RPV Depressurization
- Water Injection
  - To RPV
  - To Containment
  - Debris Cooling
- Combustible gas control
- Energetic Phenomena

- Shell Integrity

- Containment Heat Removal

    - Venting
    - RHR

- Suppression Pool Bypass

- Containment Flooding

    - DW Vent
    - RPV Vent

- Post Containment Failure Injection

    - Size

4.8.1 Containment Isolation

The MAAP calculations indicate that a drywell isolation failure that may also bypass the reactor building effectiveness can lead to both an early radionuclide release and a high magnitude release.

The DAEC reliability evaluation indicates that the containment has a high reliability. No operator actions beyond that already cited in the EOPs to backup automatic isolation were identified.

Therefore, the DAEC Mark I containment isolation assessment has shown that while the results are affected by any reduction in containment isolation reliability that there is currently no additional accident management insight regarding improvements in containment isolation.

4.8.2 RPV Depressurization

The ability to depressurize the RPV during core melt progression, i.e., prior to RPV breach by molten debris can have a major influence on the determination of the accident sequence timing, phenomenological effects, and the challenge applied to the containment. These effects are reflected in the Level 2 model in three principal ways:

- The sequence can be completely altered by modifying the conditional probability of subsequent event tree nodes dependent on the pressure status of the RPV. For example, restoration of <u>low</u> pressure injection to a damaged core could result in in-vessel recovery and no release from containment.

- The challenge to containment can cause actions or failures not otherwise implemented. For example, phenomena related to Direct Containment Heating (DCH) can be eliminated when RPV pressure is reduced.

- Radionuclide release end states may be altered as a result of the status of RPV depressurization.

Each of these effects are discussed below and grouped by beneficial impacts and adverse impacts of depressurization:

Beneficial Effects of Depressurization

- The probabilistic modification of the sequence due to the pressure status of the RPV is treated in the CET quantification and is based on previous separate effects evaluations such as the possibility of steam explosions or vapor suppression bypass. MAAP cases are used to confirm:

  -- Steam explosion likelihood based on initial conditions

  -- Vapor suppression success criteria

  -- Recovery of the injection capability (e.g., LPCI) to terminate core melt progression in-vessel.

In every case it is found that depressurization of the RPV has a substantial beneficial effect on the probability of successfully reaching a safe stable state with the lowest radionuclide release.

- The challenges to containment as a result of core melt progression without RPV depressurization are investigated in MAAP cases:

  -- With vapor suppression
  -- With degraded vapor suppression
  -- With inadequate vapor suppression.

- The containment capability is substantially improved when the RPV is depressurized at the time when vapor suppression would be challenged by a core melt progression.

- Radionuclide releases and timing are reflected by many different aspects of the accident sequence. The direct impact of the depressurization node failure on radionuclide release and timing is most adverse if RPV blowdown from high pressure occurs through an RPV breach and into an open containment or causes a containment drywell failure at the time of RPV breach.

- <u>Adverse Impacts of Complete RPV Depressurization</u>

  - Figure 4.8-1 shows that depressurizing the RPV results in reducing the time of RPV breach from 3.2 hours to 1.9 hours. This means that the RPV is breached over an hour earlier for TQUV sequences. It may be prudent to delay RPV depressurization substantially longer than is currently advocated in the EPGs when injection is not available at high or low pressure.

  - The DAEC EOPs have identified a substantial potential benefit associated with preserving both the HPCI and RCIC turbine driven system capability as long as possible given that low pressure injection systems are unavailable. As a result, RPV depressurization is halted under such conditions above an RPV pressure of 200 psig. This provides HPCI and RCIC with some minimal margin with which to work before low pressure trips cause the loss of these systems. Such actions

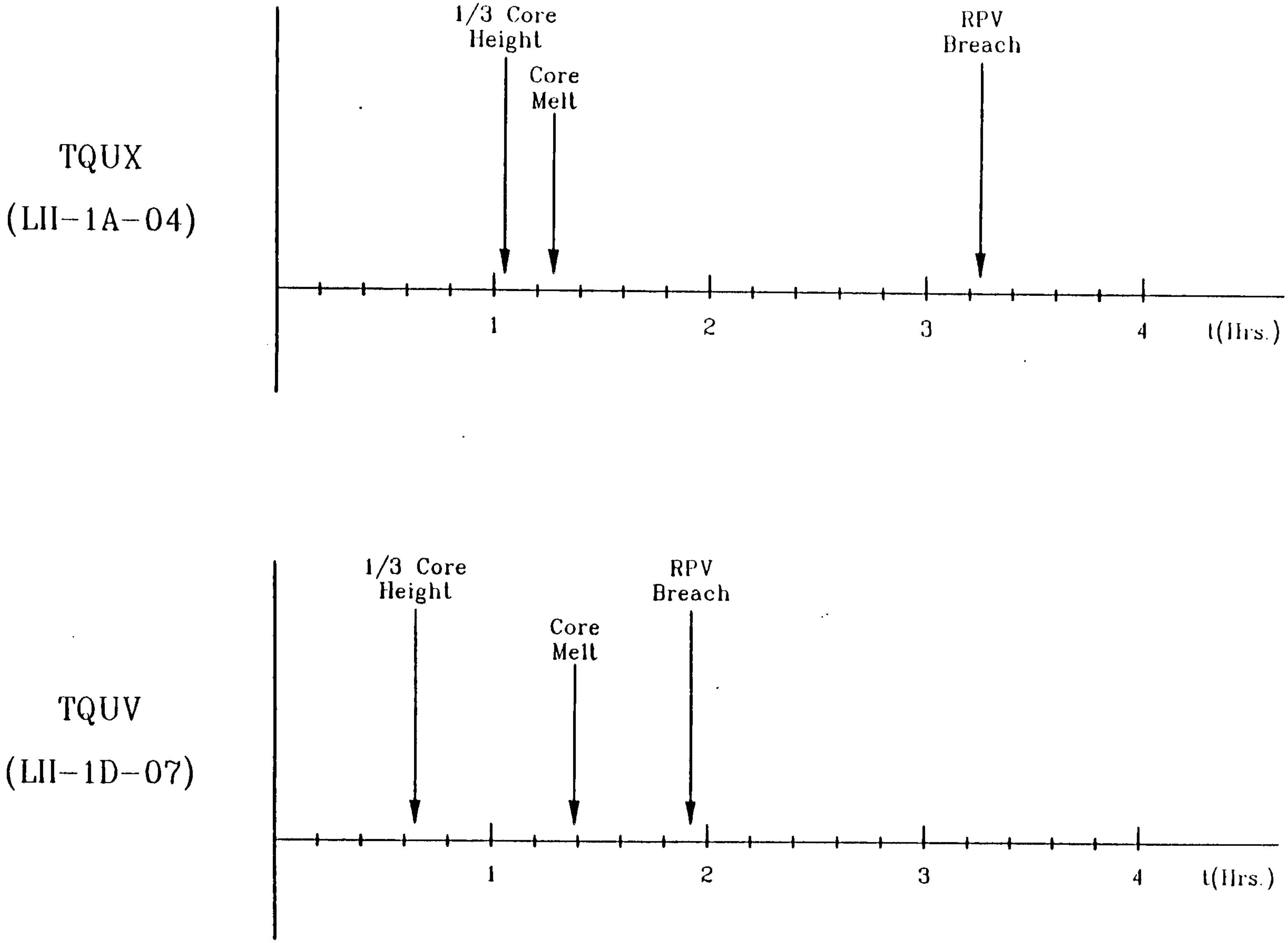


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Figure 4.8-1 Comparison of Accident Progression with RPV Depressurization Versus no Depressurization appear particularly valuable in accident sequences such as station blackout, when AC power recovery is delayed.

4.8.3 Water Injection to the RPV or Containment

One of the most important mitigating system actions that can be implemented as part of accident mitigation is the injection of water into containment or into the RPV. The adequacy of this injection for minimizing radionuclide releases can be evaluated for different combinations of other functional and phenomenological events.

Injection To Prevent Containment Failure

One of the principal benefits of water injection is that when coupled with containment pressure control (nearly equivalent to containment heat removal) successful water injection can prevent containment failure. This can prevent the following postulated containment failure modes:

- Direct contact shell failure (see NUREG/CR-5423)
- High temperature and pressure induced containment failure
- Large non-condensable gas generation.

Different methods of water injection are available from a wide variety of sources. These water sources are clearly defined in the EOPs and in training. They include the following:

- FW/Condensate

- HPCI
- RCIC
- RHR/LPCI
- CS
- RHRSW
- CRD
- Well Water
- GSW
- ESW

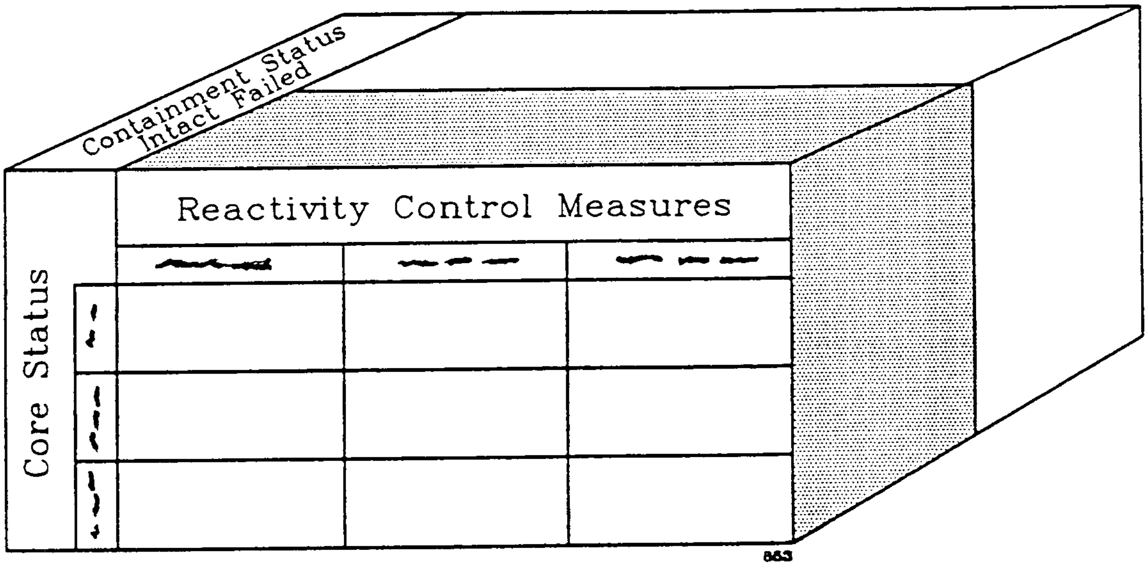
RPV Injection (Core Spray Versus LPCI Injection to RPV)

Water injection to the RPV has a number of beneficial features which include:

- Cooling residual core material in the bottom head
- Cooling fuel rods that remain intact in the core region
- Cooling the fission products that are plated out on RPV internal surfaces by steaming (dryer/separator).

The ability to provide all of these cooling benefits varies with the water source, i.e., the injection source and its flow rate. Currently there is no priority to the use of different water injection sources if a choice is to be made regarding which system(s) is (are) to be used for injection.

There are a number of different severe accident regimes under which RPV injection recovery may occur. Accident management guidance to select the optimum water source under this spectrum of conditions may be prudent. The three dimensional table (Figure 4.8-2) below summarizes some of this spectrum of cases.



For our discussions here only the containment intact slice of the table is examined (see Table 4.8-1).

Table 4.8-1

COMPARISON OF THE BASES FOR SELECTING AN OPTIMUM SYSTEM TO RECOVER A DAMAGED CORE

Assessments of RPV Injection for Containment Intact	Reactivity Control Measures		
Core Status/Location	Control Rods Intact	Control Rod Melted or Failed out with No SLC	Control Rods Melted or Failed Out with SLC and No LOCA
In-vessel Core Approximately Intact	GENE-770-38-0991 (Bottom Head Injection)*	NE	NE
In-vessel Debris	GENE-770-38-0991 (Bottom Head Injection)*	NE	NE
Ex-vessel + In-vessel Residual Debris	Judgement (Core Spray Insight)	Judgement (Core Spray Insight)	Judgement (Core Spray Insight)

* Fission product retention in the RPV not considered

NE: Not Yet Evaluated; optimum injection system still an open question.

Based on this assessment, it appears there may be marginal reasons provided over a small portion of the spectrum to choose LPCI injection. Nevertheless, based on an examination of all the possible spectrum of accidents, the following choices are considered appropriate but should be considered further during the accident management development.

The following RPV injection sources are considered viable and have the following benefits or disadvantages that may be useful in the future in establishing accident management actions:

Core Spray: This appears to be the most desirable[1] injection source for severe accident mitigation and minimizing radionuclide releases. The core spray system has a relatively high flow rate and produces a spray pattern that is most conducive to cooling material in the RPV (including residual debris) given that the RPV bottom head has been breached during core melt progression.

In summary, the MAAP sensitivity cases indicate that if core debris remains behind in the original core region, containment heat-up may occur to the point that could threaten the integrity of the drywell regardless of RPV injection and RHR availability. Accident management efforts should realize that this is a possible scenario and afford appropriate guidance to mitigate the outcome. An additional insight from DAEC MAAP runs is that there may be other water injection methods that would also usefully minimize the potential for such induced failure modes. Specifically, drywell spray or core spray would be better choices for coolant injection to eliminate this as even a <u>potential</u> failure mode, rather than injection to the recirculation loop.

Use of the core spray system can cool debris both inside and outside the RPV, and control drywell temperature rise, and thereby maximize the time to containment failure, i.e., water will follow debris out the bottom head of the vessel through the breach and fall on the debris on the drywell floor.

LPCI: This is the next most desirable injection source. It has all the advantages cited for Core Spray except that it is injected in the recirculation lines and results in the possibility of being short circuited past the core

[1] Note that conflicting conclusions may be reached using current T&H codes for sequences in which there is a failure to scram and the RPV is intact, or the control rods have melted away and the RPV is intact.

region and directly out the bottom head breach. This has the possibility of allowing revaporization from RPV surfaces in the extremely long term as one of its disadvantages. This could be most important in containment flood scenarios when venting of the RPV is directed by the EOPs. During RPV venting, the revaporization source term may escape directly to outside containment.

RHRSW: This has identical attributes to LPCI except a continuous supply of cool water is available from the RHR reservoir; LPCI recirculates water from the suppression pool.

CRD: This water source is desirable but is of limited flow rate. In addition, after RPV breach the flow path may not allow delivery to the RPV or to the drywell because of plugging of the relatively small diameter injection lines. This system is not considered here as an effective mitigating system for severe accidents that have progressed outside the RPV.

MAAP has limited modeling capability to examine the subtle differences in various in-vessel injection methods even after RPV failure. Therefore, these qualitative assessments are based on engineering judgement using MAAP guidance and inferences from the MAAP cases where appropriate.

It should also be noted that the use of drywell sprays may also be preferable to LPCI injection when some debris has moved outside the RPV. The next discussion identifies the benefits associated with the drywell sprays including the ability of the drywell sprays to limit drywell temperature increases and prevent debris attack of the drywell shell.

RPV Injection Versus Drywell Sprays

Core spray injection advantages were discussed above. Drywell sprays have many of the advantages of the core spray injection method including maintaining low drywell temperatures; however, the use of drywell sprays would be marginally effective in cooling debris that was retained in the vessel, i.e., it may still emit radionuclides.

In addition, if the operators were able to enter into containment flooding then RPV venting would be directed and the use of drywell sprays during RPV venting may also have a minimal impact on the release directly from the RPV.

The BWROG EPGs direct the use of drywell sprays based upon symptoms of containment pressure, temperature, and combustible gas concentration. Additional uses that have been visualized by investigators of severe accident conditions include the following:

- Scrub Aerosols: Scrubbing fission products from the drywell atmosphere. (Explicit symptoms to initiate sprays during radiation incidents or radionuclide releases is currently not included in the generic BWROG EPGs or the DAEC EOPs.)

- Quench molten debris: Cooling the drywell when molten debris is present on the drywell floor. However, explicit directions are given to the operator to prevent drywell spray initiation if drywell temperatures are above approximately 350°F in the drywell. For Mark I containments, the ability to prevent or mitigate drywell shell failure, is one potential benefit associated with the drywell sprays related solely to severe accident management. The time available to provide sprays prior to the RPV failure is one of the key aspects of the drywell spray investigation. This will provide water on the drywell floor when debris exits the RPV, leading to a quenched debris. See also the discussion of shell integrity (Section 4.8.6).

- Minimize both core concrete interaction and the generation of non-condensible gases: No explicit directions are provided to minimize core concrete interaction, nor does a universal symptom appear to exist that might always be available for prompting the operating crew to initiate drywell sprays.

- Prevent suppression pool bypass: Cooling the containment shell, downcomers, and the drywell to the wetwell vacuum breakers could prevent failure and consequential suppression pool bypass by the radionuclides during a postulated severe accident.

- Mitigate Combustible Gas Mixtures: The use of drywell sprays to cool the drywell can lead to improved prevention of deflagration of combustible gases. There are some adverse impacts related to deinerting (i.e., removing any steam inerting) or increased turbulence. Both of these two latter effects can have detrimental impacts on combustible gas control.

Debris Cooling

Coolant injection to the drywell via either the RPV or the drywell sprays has the benefit of providing debris cooling. This cooling will have the following beneficial effects:

- Limit temperature increase in the drywell during the core melt progression

- Limit the non-condensible gas generation in the containment and, thereby, prevent reaching the critical containment failure pressure and temperature.

EOPs currently specify coolant injection to the RPV under the symptoms that would be present for cases with the RPV in danger of being breached.

Injection During Containment Failure or Vent

As part of the evaluation of containment failure or vent and the impact on releases, it is important to assess the volume or flow rate of makeup to the debris during the melt progression. The greater the cooling (from any source) the lower the radionuclide releases.

4.8.4 Combustible Gas Control

The EOPs specify that, as part of accident mitigation, venting the containment to minimize the possibility of a combustible gas mixture should be undertaken when symptoms are met.

For the postulated severe accidents considered in the DAEC PRA, these conditions would include cases for which the containment is deinerted and radionuclides have been released from the fuel. For such cases the combustible gas venting has the following features:

- The release of radionuclides begins early in the sequence
- The vent path is assumed to be the drywell vent (This may be non-conservative, but the EOPs do not prohibit this pathway)

- The suppression pool is subcooled for the majority of the release and, therefore, radionuclide releases are found to be substantially reduced by suppression pool scrubbing. The pool is considered subcooled because of the availability of RHR pool cooling for most of the combustible gas venting cases. If drywell venting is undertaken, then the releases increase to the high category.

4.8.5 Energetic Phenomena

There are a large number of energetic phenomena that have been postulated during core melt progression accidents. These phenomena include, among others:

- Steam explosions
- Direct containment heating
- Hydrogen detonation.

While the MAAP code can provide insights regarding sufficiency of conditions to cause these phenomena, it is not believed that MAAP provides a means to calculate the results of such phenomena. Therefore, consistent with past PRA work (e.g., WASH-1400, NUREG-1150, Limerick PRA, Shoreham PRA), when these phenomena are probabilistically and deterministically considered to occur (i.e., see CET end states for CZ failed), they are assigned a high release category. The release category timing is still determined by the sequence specific core melt progression timing. No DAEC specific MAAP calculations are performed to further refine this binning.

Regarding the phenomena, it appears that the principal accident management action that can affect these phenomena is the depressurization of the RPV. The effects are:

- In-vessel steam explosion - The low probability may increase slightly.

- Direct containment heating - The already low probability would be eliminated.

- Vapor suppression failures - The low probability of containment induced failure due to vapor suppression failure would be eliminated.

The DAEC EOPs are judged to take the appropriate action to direct RPV depressurization.

4.8.6 Shell Integrity

The DAEC assessment indicates that one of the potential failure modes for a number of the accident sequences that dominate the Level I IPE is associated with direct debris contact with the Mark I drywell shell when no water is available.

The ability to supply water to the drywell before or soon after RPV breach is a high priority action to prevent the shell failure and consequential release early in the core melt progression. Drywell sprays offers one of the only means to provide water to the drywell floor prior to RPV breach in non-liquid line breaks.

However, their usefulness for severe accident conditions is impacted by the limitations in the EOPs on the use of the sprays.

Four of the DAEC EOP limitations on the use of drywell sprays that restrict the initiation or continued operation of the drywell sprays are identified as follows:

- The drywell spray initiation curve is cited in all cases of drywell spray initiation. The drywell temperature and pressure must be in the "safe" regime of the curve as shown in Figure 4.8-3 to allow initiation.

- Drywell pressure must remain greater than 2.0 psig or the sprays must be terminated.

- Suppression pool level must be lower than a height that could cover the vacuum breakers to initiate drywell sprays.

- Adequate Core Cooling Interface with Drywell Sprays
  Another important procedural limitation is that with molten debris on the drywell floor, the RPV has presumably been breached and water level would not be able to be maintained above TAF. This could be considered a condition for which adequate core cooling is not assured. For such cases, the current DAEC EOPs and Revision 4 of the BWROG EPGs may be interpreted as ambiguous regarding the use of drywell sprays to be used because adequate core cooling cannot be assured. The BWROG EPG Bases Document (OEI 8390-4B) states the following regarding drywell spray initiation:

  - For plants where water for drywell sprays is supplied by RHR pumps, operation of drywell sprays is only permitted if the pumps to be used are not required to assure adequate core cooling.

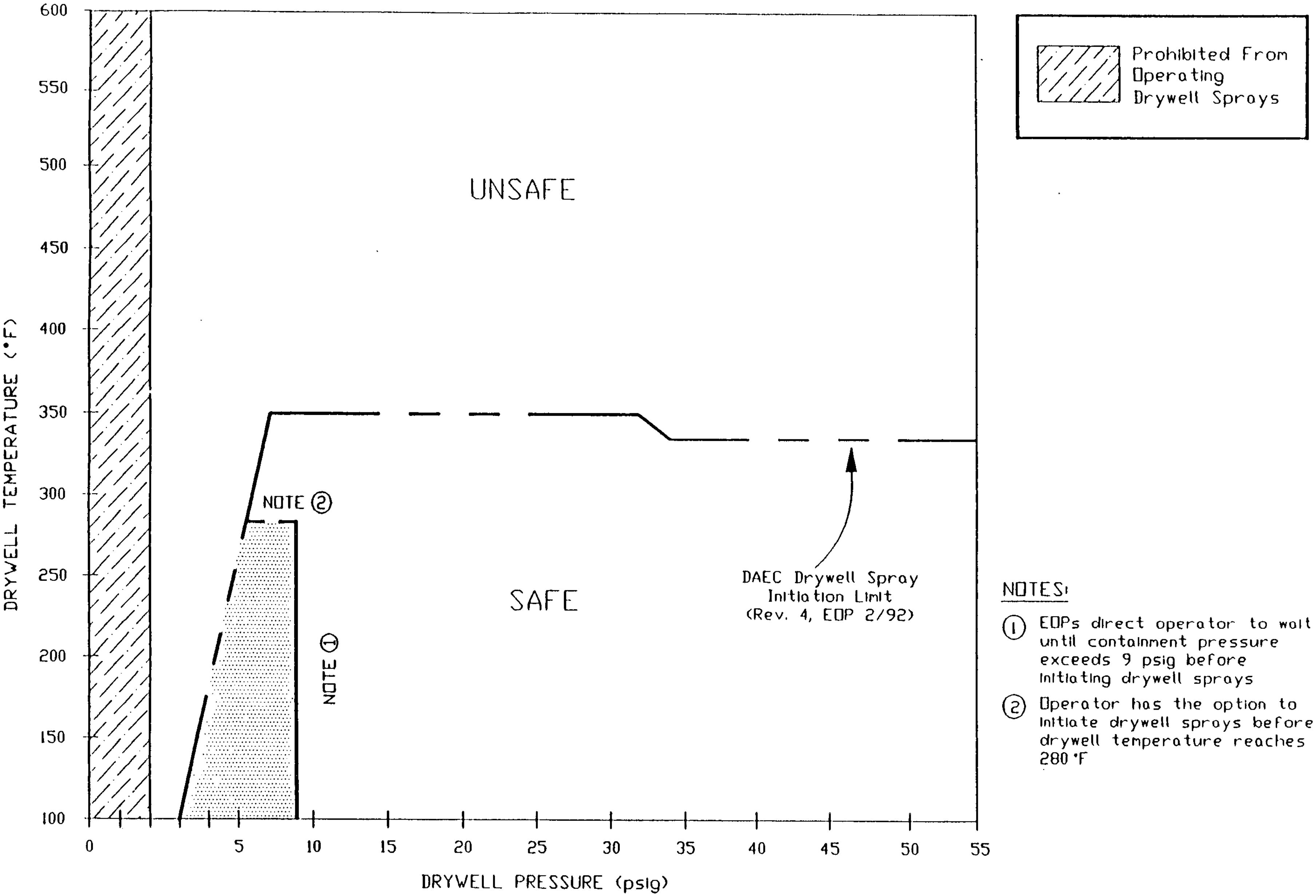


Figure 4.8-3 DAEC EOP Limits for Drywell Spray Initiation

- Maintaining adequate core cooling takes precedence over maintaining drywell temperature below design temperature, because catastrophic failure of the containment is not expected to occur at drywell design temperature.

- Accordingly, operation of RHR pumps aligned in the drywell spray mode is permitted only if continuous operation of the pumps in the LPCI mode is not required to assure adequate core cooling.

- The wording of this step does, however, permit alternating the use of RHR pumps between LPCI injection and drywell spray as the need for each occurs and so long as adequate core cooling is able to be maintained.

For DAEC, a clarification was sought to ensure that the EOPs and operator training were being properly modeled in the DAEC Level 2 analysis. The result of this clarification is the following:

- If in Class IA and cannot inject into the RPV with the RHR pumps, then operators would use RHR pumps for sprays when directed by the EOPs.

- If RPV water level cannot be restored but indication is that RHR pumps are pumping to the RPV, operators would not divert RHR pumps to spray the containment. SROs are trained to prefer injection to the core and wait for the PCPL to be reached when the EOPs say irrespective of adequate core cooling, spray the drywell.

- The current Drywell Spray Initiation Curve (DWSI) for DAEC provides restrictions on when the drywell sprays can be initiated. Given this curve, it appears that for many of the severe accidents investigated in the Level 2 analysis that the curve will not be satisfied at all or only for a short amount of time during which the sprays could be beneficial.

- Drywell sprays have the advantage that they can supply water to the drywell floor prior to RPV breach. This water would collect in the DAEC sumps and up to the level of the downcomers (~ 1 ft.). The DAEC MAAP runs have shown that debris can be effectively quenched when it is discharged from the RPV to this waiting pool of water. This would preclude the debris attack and failure of the drywell shell.

- Drywell shell failure due to debris attack can be prevented if drywell sprays are initiated before RPV breach and the drywell floor is filled with water to quench the debris. Based on the enclosed using MAAP calculations for DAEC specific severe accidents, the following sequence types would not allow or call for spray initiation before RPV breach:
    - Class IIIC
    - Class V
    - Class ID

  On the other hand, the following classes appear to satisfy the DWSI curve prior to RPV breach:
    - Class IA/IC
    - Class II

- Class IV

- It is recognized that regardless of the flexibility offered by the DWSI curve, additional changes to the EOPs may also be required to remove any ambiguity regarding the diversion of injection sources away from the RPV when adequate core cooling is not assured, i.e., low reactor water level.

Future investigations of accident management will identify whether better use of drywell sprays under severe accident conditions (can be proceduralized)

4.8.7 <u>Containment Wetwell Venting or Wetwell Breach With Continued Injection</u>

Containment venting is an extremely valuable accident management strategy that should be used as among the last resort approaches to controlling containment conditions.

Containment venting provides a useful method of containment pressure control and containment heat removal. If continued coolant injection to the containment can be maintained despite the core melt progression outside the vessel and containment venting, then radionuclide releases can be minimized. Much of this discussion also applies to situations in which the wetwell airspace may fail.

In general, the containment vent should be delayed as long as possible into an accident before it is initiated. This decision process results in a delicate balance of the optimum time to initiation of containment venting.

The factors that affect the decision to containment vent can be categorized as shown in Table 4.8-2 as follows:

- Containment Structural Capability
  - Static
  - Dynamic

- Containment Vent Valve Capability

- System Operability
  - SRV
  - EQ in Drywell
  - RCIC
  - MSIV Open
  - LPCI

- Radionuclide Activity

- Plant Availability

- Containment Leakage/Reactor Building Environment

- Deinerting

- Depletion of Non-Condensibles

- Loss of NPSH

Table 4.8-2

SUMMARY OF PLANT CONSIDERATIONS FOR OPTIMIZATION OF CONTAINMENT VENT PRESSURE

Parameter for Consideration	Less Than Design (40 PSIA)	Design (60 PSIA)	Design + Margin	Design + .5 x Design (90 PSIA)	Ultimate (120 PSIA)
Containment Structural Capability[1]					
- Static	OK	OK	OK	OK	OK
- Dynamic	OK	OK	OK	Marginal	Not Acceptable
Containment Vent Valve Capability[1]	OK	OK	OK	OK	Not Acceptable
System Operability					
- SRV	OK	OK	OK	Marginal	Not Acceptable
- EQ in Drywell	OK	L	L	L	L
- RCIC	OK	L	L	L	L
- MSIV Open	OK	L	L	L	L
- LPCI	OK	OK	L	L	L
Radionuclide Activity[1]	H	H	H	H/OK	OK
Plant Availability	OK	OK	OK	L	L
Containment Leakage/Reactor Building Environment	OK	OK	OK	L	L
Deinerting	H	H	H	H	H
Depletion of Non-Condensibles	H	H	H	H	H
Loss of NPSH	H	H	H	H	H

OK: Means that through the consideration of this parameter alone it appears acceptable to vent at pressure up to and including the value cited.

L: Means that a tentative conclusion regarding this parameter alone has been reached which would indicate it prudent to vent at lower pressures

H: Means that a tentative conclusion regarding this parameter alone has been reached which would indicate it prudent to vent at higher pressures.

[1] Heavily Weighted.

Different cases of containment venting are found to result in substantially different estimates of the radionuclide release:

	Maintain Injection to RPV or Containment	Wetwell Vented	Suppression Pool Bypass	Radionuclide Release
Case 1	YES	YES	NO	LL-L
Case 2	YES	YES	YES	M
Case 3	NO	YES	NO	L
Case 4	NO	YES	YES	H

The results of the MAAP calculations indicate that:

1) Case 1: The radionuclide releases are low (L) or very low (LL) for the case in which water injection, wetwell venting, and no suppression pool bypass are present (case LII-ID-1, LII-ID-4)

2) Case 2: Releases are approximately times larger (Moderate release) for the case in which suppression pool bypass is present. (case LII-ID-3)

3) Cases 3 and 4: Releases range from low (LII-ID-8) to high (LII-IIIC-1) for the case when injection fails at the time of containment venting and subsequent temperature induced drywell failure occurs.

The purpose of venting is to avoid containment over-pressurization and protect the containment structural integrity. Functionally, this can be accomplished by using the system designed for containment venting or combustible gas control. Additionally, the containment would be effectively vented through a breach in the structure.

The impact of venting on a potential environmental source term is dependent primarily on two factors:

1) Timing for establishing the vent pathway; and

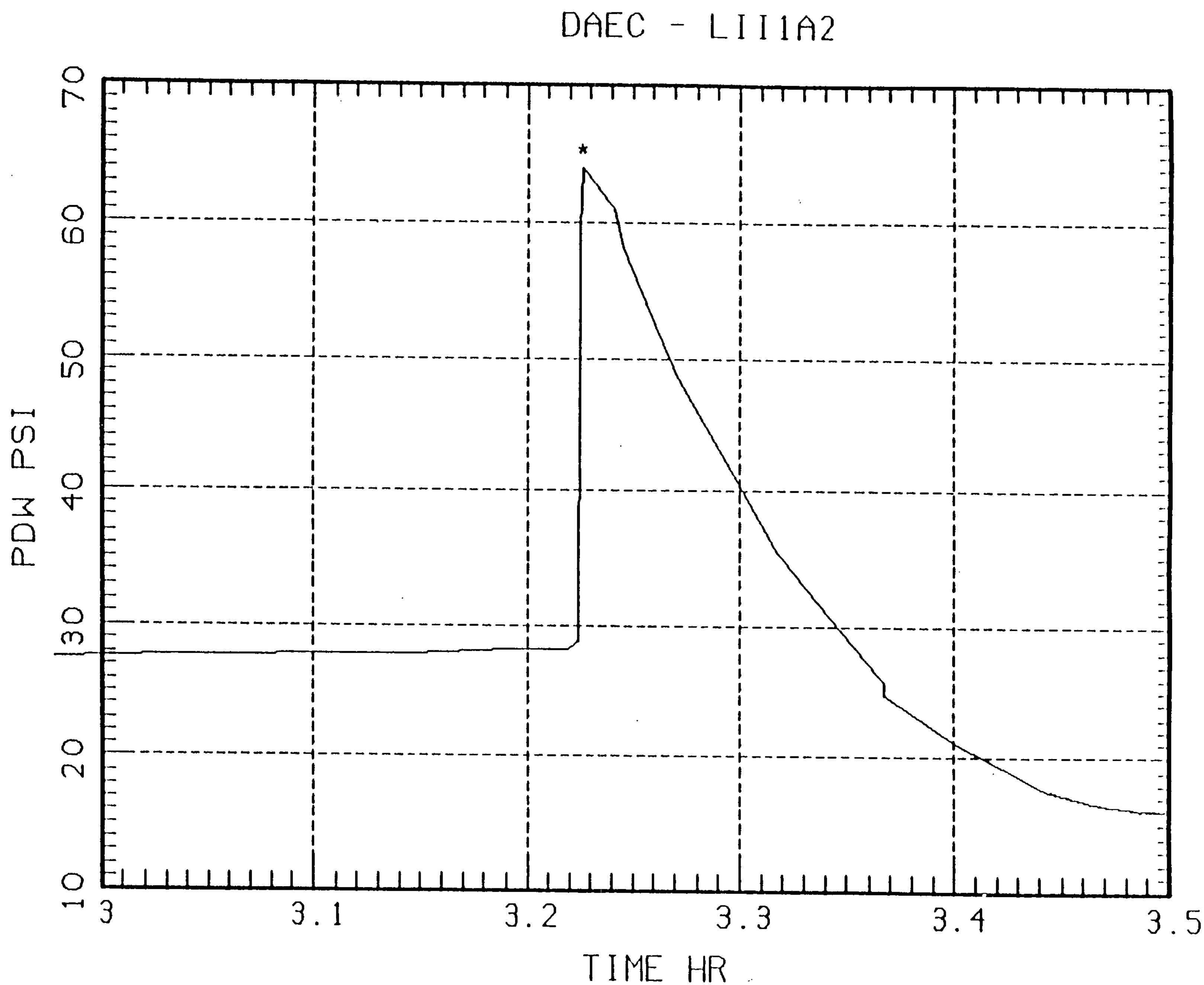
2) The suppression pool effectiveness, i.e., the availability of a pathway that routes the radionuclides through the suppression pool.

These conditions are further discussed below.

4.8.7.1 Venting: Timing of Radionuclide Release

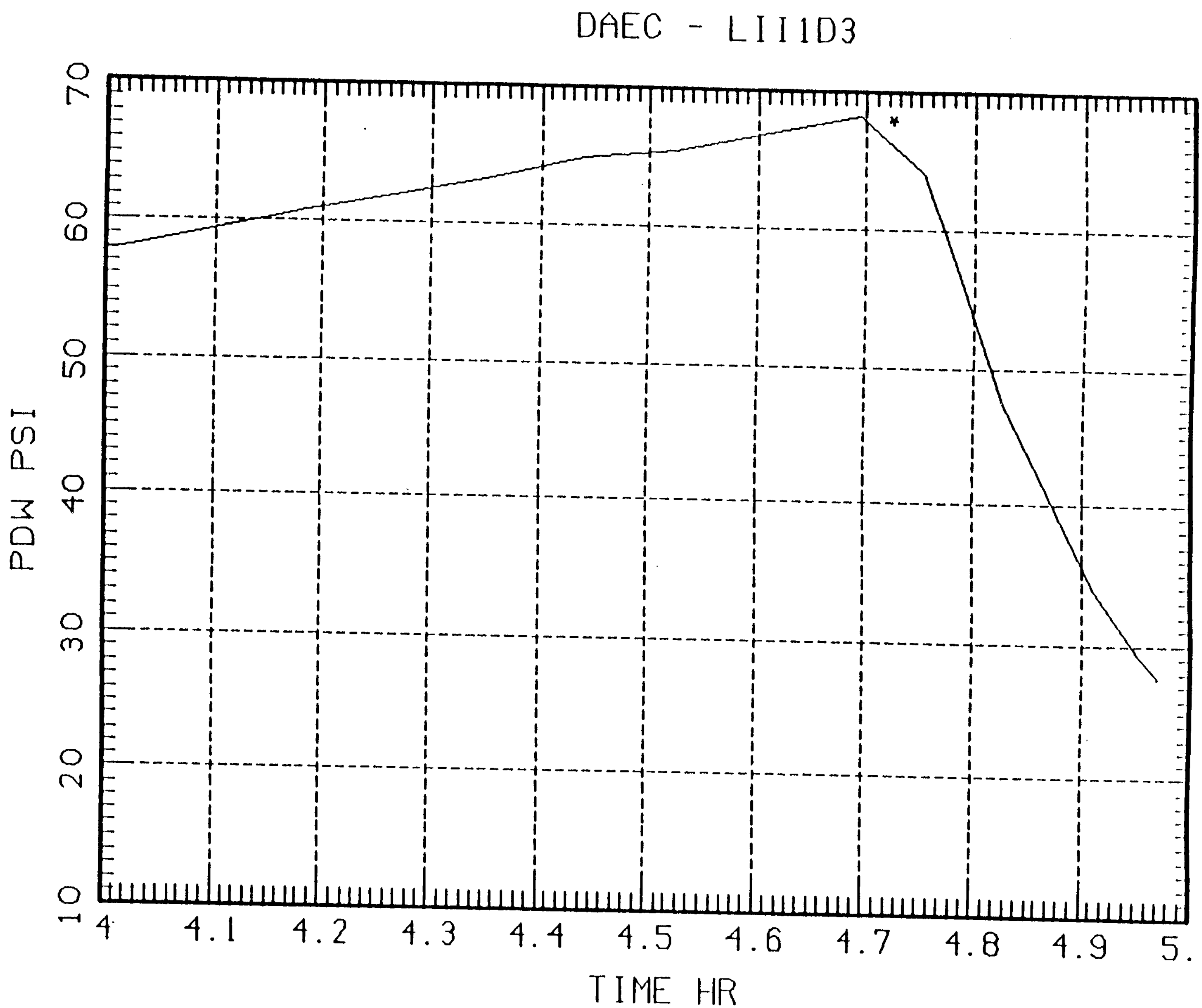
Containment venting can influence the radionuclide release by releasing material early in an accident scenario. This may be controlled by the rupture disk failing at its prescribed pressure.

Figures 4.8-4 through 4.8-6 show four postulated severe accident scenarios that have caused the containment vent rupture disk to fail.



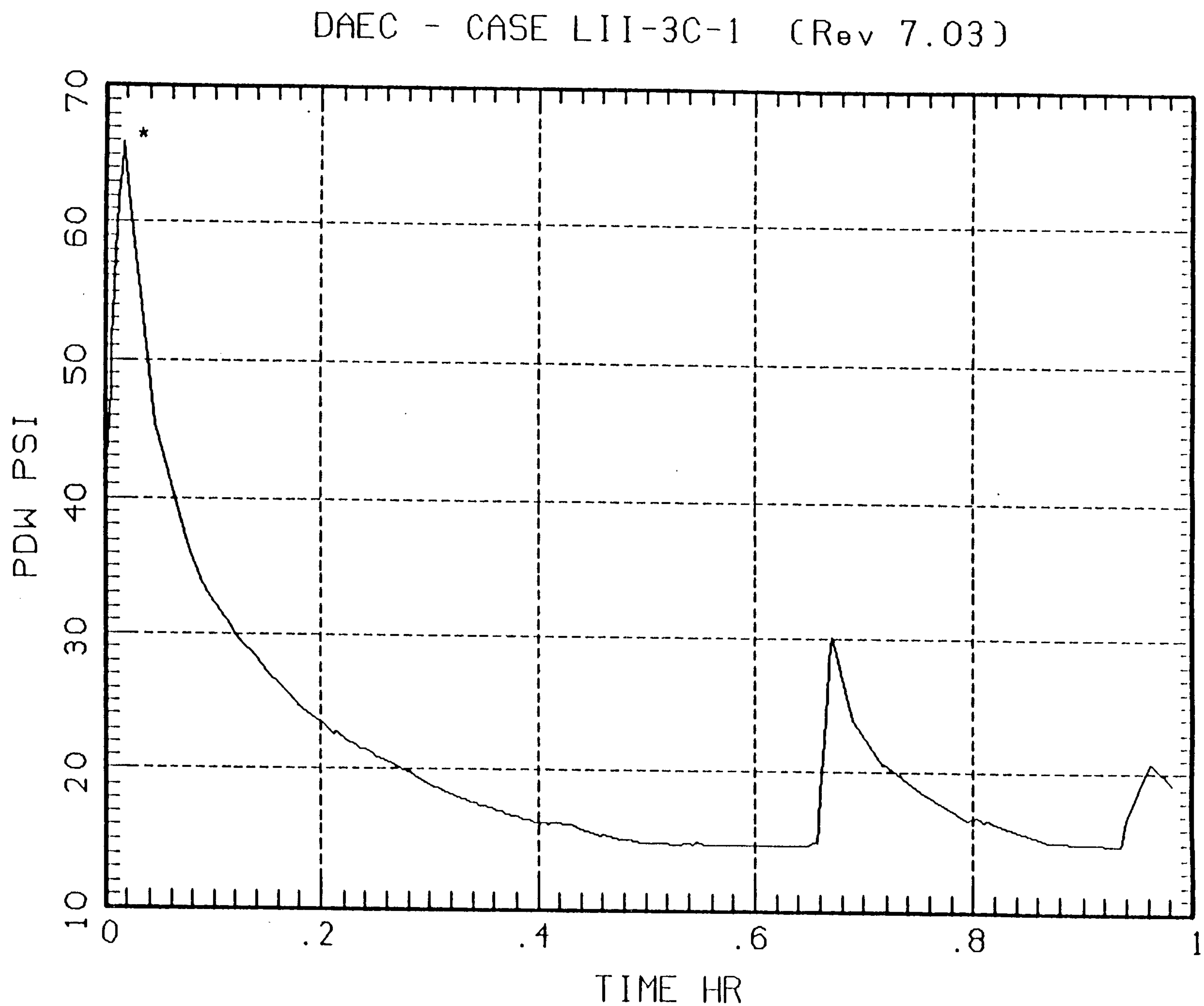
* Containment Rupture Disk Fails

Figure 4.8-4 Drywell Pressure for a Class IA (Core Melt Progression Without Injection at High RPV Pressure)



* Containment Rupture Disk Fails

Figure 4.8-5 Drywell Pressure Plot for Class ID Sequence (Core Melt Progression at Low RPV Pressure [illegible]njection Restored at RPB Breach)



* Containment Rupture Disk Fails

Figure 4.8-6 Drywell Pressure Plot for Class IIIC (LOCA with Inadequate Injection)

- Class IA Sequence (Figure 4.8-4): RPV blowdown at RPV breach causes rapid containment pressurization and the rupture disk failure at 3.2 hours into the severe accident scenario. Without containment venting, the pressure reaches a peak of approximately 70 to 80 psig and then returns to lower pressures as the steam condensers. Therefore, containment venting may not be necessary this early in the scenario.

- Class ID Sequence (Figure 4.8-5): Long slow heatup and pressurization of the drywell due to steam and non-condensibles causes the rupture disk to fail at 4.7 hours. Significantly, longer times would be available if the vent pressure were set higher.

- Class IIIC Sequence (Figure 4.8-6): A large LOCA with degraded vapor suppression system results in failing the rupture disk near the beginning of the accident. Therefore, the vent path may be opened with an existing pool bypass (i.e., vapor suppression system bypass).

Note that the RPV is breached in each of the above cases when the rupture disk fails. Therefore, there is no need to have the vent pressure set by the SRV capability.

Two examples of the extremely unlikely failure sequences for which containment venting can result in substantial releases are the following:

- <u>Sequence #1</u>

  - Large LOCA

  - Vacuum breakers stuck open during the process

- No injection available until core debris is ex-vessel (e.g., effective drywell sprays are available)

- Venting initiated at PCPL

The resulting radionuclide release as calculated by MAAP is (1) initiated early; and, (2) the consequential release is high.

This impact could be substantially mitigated if no venting occurs and the containment is allowed to absorb a substantial amount of the severe accident energy. If this occurs, the release time can be delayed substantially from several hours to nearly a day depending on whether systems can be recovered during the core melt progression.

- Sequence #2

There are accident sequences that can involve severe core damage and result in coincident, transient high containment pressure near, or slightly above, the containment design pressure. If the containment design pressure is chosen as the vent pressure, (i.e., the pressure at which rupture disk operates), then containment venting will be initiated automatically or by operator action at the time of the initial pressure rise.

Figure 4.8-7 shows a comparison of the two postulated accident scenarios - one with containment venting implemented and one without containment venting implemented.

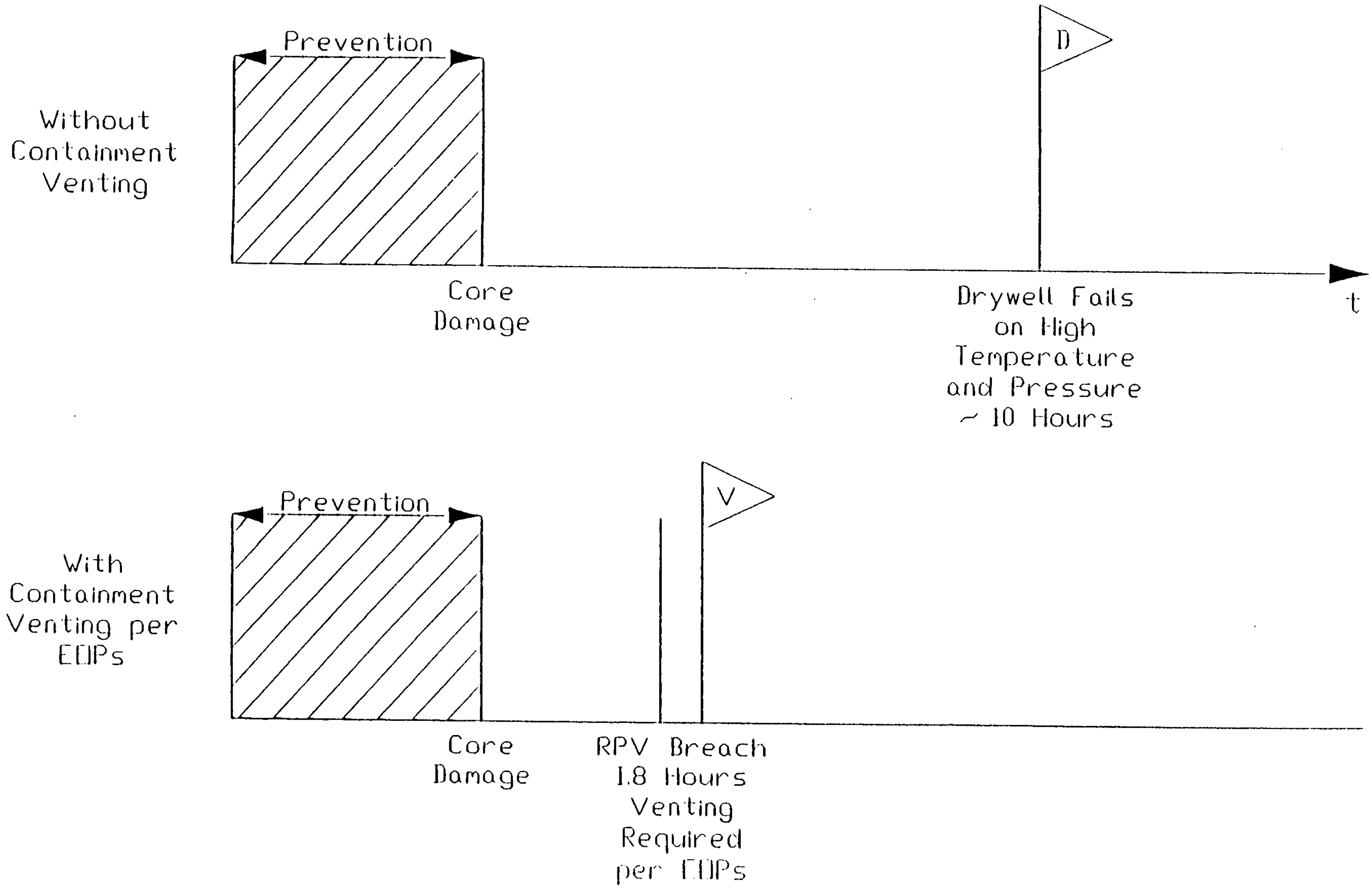


Figure 4.8-7 Comparison of the Timing of Radionuclide Release with EOP Required Vent Action Versus No Vent Action

Radionuclide Release from Either Containment Venting (V) or Drywell Failure (D)

The containment venting scenario is seen to result in a very early radionuclide release (~ 1.8 hours) compared with the no venting case (~ 10 hours). This means that the vent strategy, as implemented in the EOPs, results in releases to the environment earlier than may be necessary.

Nevertheless, venting (controlled radionuclide release) may occur earlier than an uncontrolled containment failure, but containment venting is always preferred to the potential for an uncontrolled containment failure.

4.8.7.2 Venting: Suppression Pool Scrubbing Of Effluent

Suppression pool water temperature (i.e., degree of subcooling) may affect the characteristic of the pool to retain aerosols during the vent. It is postulated that as the bulk temperature of the pool approaches saturation temperature, the effective DF of the pool decreases. In fact, current MAAP calculations [Rev. 7.01] indicate that upon approaching saturation temperature, the pool DF becomes unity (i.e., all aerosol radionuclides pass through the pool).

As alluded to in the discussion above, it appears that suppression pool DF with respect to the retention of radionuclides decreases as its temperature increases. General Electric in NEDO-25420 dated June 1981 found the following:

> Suppression pool decontamination factors appropriate for use in BWR risk assessments are presented in Table 4.8-3. Based on the data presented and the expected BWR transport conditions, suppression pool decontamination factors of at least 10^2 for elemental iodine and particulates, and 10^3 for cesium iodide are justifiable for subcooled pools. For saturated pools, decontamination factors of at least 30 for elemental iodine and 10^2 for particulates and cesium iodide are currently justifiable.

Table 4.8-3

MINIMUM SUPPORTABLE AND POTENTIALLY ATTAINABLE SUPPRESSION POOL DECONTAMINATION FACTORS FOR IODINE AND PARTICULATES

Transport Pathway and Associated Event(s)	Minimum Supportable DFs		Potentially Attainable DFs[3]
	Subcooled Pool[1]	Saturated Pool[2]	
Reactor pressure vessel to pool via safety relief valve and quencher (Transients)	10^3 CsI, I^-, HI 10^2 particulates 10^2 I_2	10^2 particulates[4] 30 I_2	10^5 - 10^6 CsI, I^-, HI 10^5 - 10^6 particulates 10^2 - 10^5 I_2
Reactor pressure vessel to pool via vents (Transients following RPV depressurization, or LOCA post blowdown period)	10^3 CsI, I^-, HI 10^2 particulates 10^2 I_2	10^2 particulates[4] 30 I_2	10^4 - 10^6 CsI, I^-, HI 10^5 - 10^6 particulates 10^2 - 10^3 I_2
Aerosol Transport to Pool Via Vents (Core-Concrete Vaporization Release)	10^2 particulates 10^2 I_2	10^2 particulates[4] 30 I_2	10^5 - 10^6 particulates 10^2 - 10^3 I_2

(1) During these conditions, complete condensation is expected when the pool is subcooled.

(2) A subcooled pool is at a temperature below the saturation temperature corresponding to the pressure in the containment, while in a saturated pool steady state boiling "steaming" is occurring.

(3) Potentially attainable by further testing (saturated-subcooled pools).

(4) Includes CsI

Natural processes such as the agglomeration of solids, plateout, deposition, washout, etc., also play an important role in limiting the quantity of fission products available for leakage to the environment.The overall attenuation factor applicable to BWR degraded core accident scenarios includes both the effects of pool scrubbing and of such natural removal processes that will occur in the various volumes of the BWR process systems and its multiple containment system.

This effect is further discussed in the following section concerning the effect on source term with the availability of RHR system heat exchangers in the suppression pool cooling mode.

Observations Regarding Containment Venting for Overpressure Protection

Observations concerning containment venting as specified in the DAEC EOPs include the following:

- Venting is a strategy to provide a defense-in-depth approach to accident management using existing BWR configurations and equipment. As such, it provides a graded response to accidents.
- Venting can be a useful part of an integrated strategy to prevent accident types that challenge the capability of the containment by overpressurization. This would allow the operating staff to maintain coolant injection makeup by avoiding coolant injection failures that may be induced by an uncontrolled containment failure at an undefined location.
- Venting can be a useful part of strategy for severe accident mitigation to preserve the multiple containment functions.
- Competing phenomena that could reduce the positive safety influence of venting have been identified, but their contribution

appears to be substantially less than the potential positive aspects for most sequences.

- Delays in venting may be justified to beyond the plant design pressure when containment temperatures are relatively low.

- Another insight derived from this evaluation is that containment failure is predicted to occur due to the combination of high pressure and high temperature - a condition for which venting has not been designed to combat. Specifically, the containment failure is predicted to occur below current EOP vent pressure when temperatures in the drywell exceed 650°F. Therefore, the second accident management insight is that for high drywell temperatures the vent pressure may need to be reduced to prevent uncontrolled releases due to drywell failures.

- Situations that direct the containment to be vented as a means to prevent containment failure by overpressurization are conditions far beyond the plant's design basis and are restricted to very specific and low frequency circumstances. Venting actions are among the last resort actions, i.e., taken only after the primary methods of performing the protective functions associated with containment heat removal and pressure control have failed. Venting is intended to prevent more serious or uncontrolled failures that are judged likely to occur should venting activities not be performed.

- Venting permits a gradual reduction of a containment pressure rise as opposed to a potentially uncontrolled depressurization associated with containment rupture.

- Venting from the wetwell maximizes suppression pool scrubbing essentially limiting releases to noble gases when the suppression pool remains in the pathway for fission products.

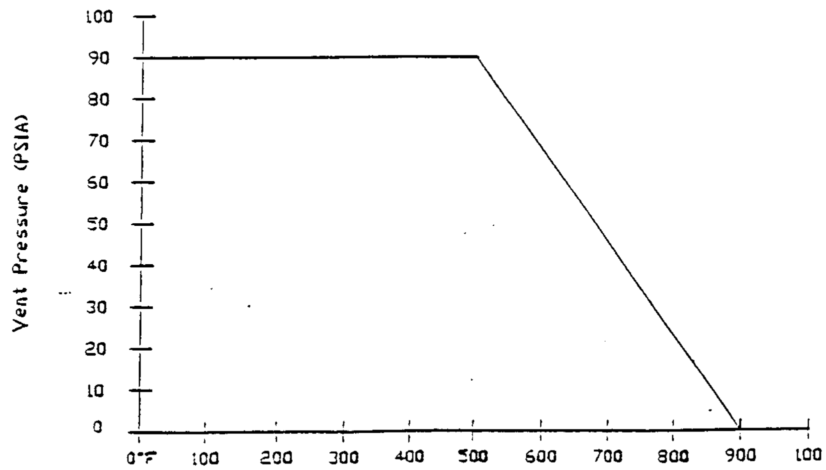
<u>Conclusion</u>

The conclusion is that optimization of containment venting under severe accident conditions would require some analyses and rethinking of the current EPG instructions, particularly related to:

- The timing of radionuclide releases from the drywell or during bypass sequences.

- Temperature dependent failures of containment.

A proposed temperature depending venting pressure is as follows:



4.8.8 Suppression Pool Cooling Mode of RHR

The RHR system heat exchangers are placed on-line by the operator to maintain the containment within specific pressure and temperature boundary conditions prescribed in the EOPs. Containment heat removal affects both the magnitude and timing of a potential source term release to the environment. Timing (and magnitude) of an impending release can be extended by controlling containment pressure below the point at which structural failure occurs. The magnitude of the release can be affected by two phenomena:

1) maintaining the suppression pool temperature less than the NPSH and vortex limits of ECCSs taking suction off the pool; and

2) controlling suppression pool water temperature below saturation.

Each of these phenomena are briefly discussed below.

4.8.8.1 Timing Of Containment Failure

It is postulated that under certain conditions the timing of containment failure after the development of a source term inside the containment can affect the magnitude of any subsequent release to the environment. This effect is further discussed in the section addressing the timing of vent initiation.

4.8.8.2 Controlling Suppression Pool Water Temperature

Maintaining suppression pool water temperature as low as possible extends the time available in which the operator can establish makeup to either the RPV or the drywell upon its breach. Plant specific MAAP calculations have shown that the availability of water to fuel debris (given that in-vessel recovery was unsuccessful) reduces both the impact to the containment, as well as the source term that accumulates inside the drywell air space. These phenomena are further discussed in the section that considers the effects of supplying water to debris ex-vessel.

The suppression pool water temperature also affects the potential for "scrubbing" aerosols if the source term is directed through the pool before egress from the containment. MAAP calculations [Rev. 7.01] indicate that there is a correlation (i.e., an inverse relationship) between the water temperature and the effective pool DF. Presently, these analyses indicate that the suppression pool is ineffective for scrubbing radionuclide aerosols once the water temperature achieves its saturation temperature. This assumption does not appear consistent with NEDO-24250 and recent experiments. In fact, due to bubble dynamics in a saturated pool, the DF may actually increase at saturation. It is the judgement of the IPE team that a DF of at least 10 for a saturated pool is reasonable. The MAAP results are adjusted accordingly based on this judgement. Of course, this adjustment will only apply to the pool scrubbing portion of the source term for events with late drywell failure.

4.8.9 Suppression Pool Bypass

Generally, assuming that the suppression pool provides a means to scrub aerosols, maintaining a positive differential pressure between the drywell and the wetwell has a beneficial effect in reducing the magnitude of a source term by directing a portion of the radionuclides into the pool. If the pool becomes bypassed (i.e., radionuclides can be transported from either the RPV or drywell to the wetwell air space), then scrubbing of aerosols cannot be accomplished. Consequently, a release following containment failure in the wetwell or containment wetwell venting will contain a larger fraction of radionuclide aerosols and particulates.

4.8.10 Containment Flooding

Given the current state of knowledge regarding severe accident phenomenology, the DAEC EOPs have established a near optimum balance among the contingency procedures which the operator can implement.

The EOPs generally define one of the following: the optimum procedural path; a procedural pathway that is close to optimum; or, a pathway for which insufficient analytical (and experimental) information is available to more precisely define the optimum pathway. Changes in the current understanding of severe accident phenomena or in the philosophy of dealing with severe accidents may impact some of the EOP steps and contingency actions. The specific issue that is addressed here is the decision regarding containment flooding versus possible alternatives.

However, it has become clear that under certain postulated severe accidents, the BWROG EPGs direct operators to perform actions such as containment flooding that could have a more adverse potential impact on the public than other mitigation strategies that could be postulated (see Containment Flooding Discussion in Section 4.9).

For Class I the representative sequence is a loss of all injection, RPV depressurization when the core water level drops to TAF, containment venting available, no suppression pool cooling, and containment flooding through the RPV initiated shortly after vessel failure.

The RPV vent case (LII-ID-10) is characterized by a CsI release starting at about 4 hours, but this release steadily continues until about 13% of the initial CsI mass has been transported to the condenser by 36 hours. The condenser/turbine building combination, however, provides for slightly higher calculated DFs than the refuel floor such that the final release to the environment is about 6.7% of the initial CsI mass. The dominant removal mechanism in the condenser/turbine building region was calculated to be gravitational sedimentation, and due to the low flow rates at the time of fission product transport from the vessel, a calculated DF of about 2 in the condenser and the turbine building appears reasonable for this case, but it is unknown whether this would be the situation in all circumstances. In any event, crediting a DF of at least 1.5 or 2 for all RPV vent cases seems reasonable.

For Class I flooding sequences, the timing and magnitude of fission product releases can be conservatively estimated as early and moderate, respectively.

In summary, the timing and magnitude of the releases can be categorized as shown in Table 4.8-4 with the notion that if anything, the drywell vent cases will experience lower releases than those reported here.

Interesting insights can be obtained by comparing these results with similar cases that do not involve containment flooding.

Table 4.8-4

SUMMARY OF MAAP RESULTS FOR DAEC 'FLOODING' SCENARIOS

Class	Sequence	Description	Time of Initial Release	CsI Fraction to Reactor/ Turbine Building	CsI Fraction to Environment	Timing of Release[1]	Magnitude of Release to Environment [3]
I	LII-ID-9	Flood, DW Vent to Refuel Floor	4 Hr.	0.5%	6.0%	E	M[3]
I	LII-ID-10	Flood, DW Vent, RPV Vent to Condenser	4 Hr.	6.3%	6.7%	E	M

(1) Time of initial release
E - Early, < 6 hours
I - Intermediate, 6 to 24 hours
L - Late, > 24 hours

(2) CsI Release Fraction Severity
H - High > 10%
M - Moderate, 1% to 10%
L - Low, 0.1% to 1%.
LL - Low-Low, < 0.1%

(3) The drywell vent case does not credit fission product scrubbing through the water pool surrounding the RPV. Therefore, the estimated release is considered to be conservative. In all likelihood, the actual release would be lower for these cases.

The accidents in question involve example core melt sequences with the containment initially intact. Various strategies were evaluated. Calculated release magnitude and timing are shown in Table 4.8-5 for the Class ID sequences. Minimum credit was given to ex-containment DF. This may bias the results as indicated in the text.

Table 4.8-5

Comparison of Class ID Sequence:

Containment Flooding Versus No Flooding

MAAP Case	Description	Release Magnitude from Containment	Time
LII-1D-9	Drywell Vent as part of containment flooding procedure	Moderate (6% CsI)	Early
LII-1D-10	RPV Vent as part of containment flooding procedure	Moderate (6.7% CsI)	Early
LII-1D-8	Wetwell Vent and No Flood	Low (0.7% CsI)	Intermediate
LII-1D-5	RHR, Sprays or Injection, and no Flood	No Release+	No Release

+ Except Leakage

A possible improved response for these types of sequences for which the EPG directions result in the highest potential consequences at the earliest time, is to provide the operators guidance on protecting containment and cooling debris using methods that do not require RPV venting and avoid using the drywell vent unless no other alternative exists. Alternate actions have been shown to produce substantially lower releases and much longer times to failure if no action is taken, i.e., even no action is better than action directed by the EPGs.

It should be noted that the drywell vent cases do not credit scrubbing as fission products are transported through the water pool that would accumulate around the vessel as flooding of the containment progression. Therefore, the calculated releases for these cases are considered conservative. For the RPV vent cases, however, the associated DF post-vessel release may not be as good in all cases. Although a lower DF may increase the actual release, in most cases this should not affect the release magnitude categorization.

More sophisticated modeling may reduce these estimates of release magnitude, but the fact remains that the actions being specified will inhibit the movement of personnel who are on-site and who are responsible for recovery, potentially early in a sequence. This would occur due to the venting and radionuclides into the turbine building area via RPV venting to the main condenser.

4.8.11 Water Injection Post Containment Failure (MU)

In the plant specific DAEC MAAP calculations, it appears that the impact of continued water injection to the RPV or drywell post containment failure (or venting) can be considered to have two possible effects:

- For cases with drywell head failures and continued water injection to the RPV, it is found that the total CsI radionuclide release to the environment is reduced by approximately a factor of 2 when compared to with drywell head failures and no continued water injection (see MAAP cases LII-1A-3 and 4). Therefore, it will be assumed that for all cases with drywell head failure that the status of top event MU will not result in a reduction in source term to the next lower magnitude. Exceptions to this are cases in which drywell sprays are available. Then, reductions of an order of magnitude are possible.

- For cases in which the containment failure is in the wetwell, the success of MU or post containment water injection to the RPV or drywell will result in minimizing the releases.

4.8.12 Containment Flood Sources

Containment flooding using RHRSW or ESW pumps requires both pumps used for injection into the RPV and the River Water pumps. Figure 4.8-8 shows that the makeup to the ESW/RHRSW pits is from the River Water Supply to the Stilling Basin and is turn to the RHRSW/ESW pits. This means that the containment flood contingency cannot be effectively carried out if only RHRSW and ESW pumps are all that are available. The AMGs could assist the operator to know that containment flooding should not be initiated if it cannot be successfully completed. Specifically, a "half" flooded containment can be a dangerous configuration.

4.8.13 Containment Injection at High Containment Pressure

The maximum primary containment water level limit (MPCWLL) has some important effects on the PRA evaluation. The specific effects discussed here are related to the impact on the frequency and magnitude of radionuclide releases.

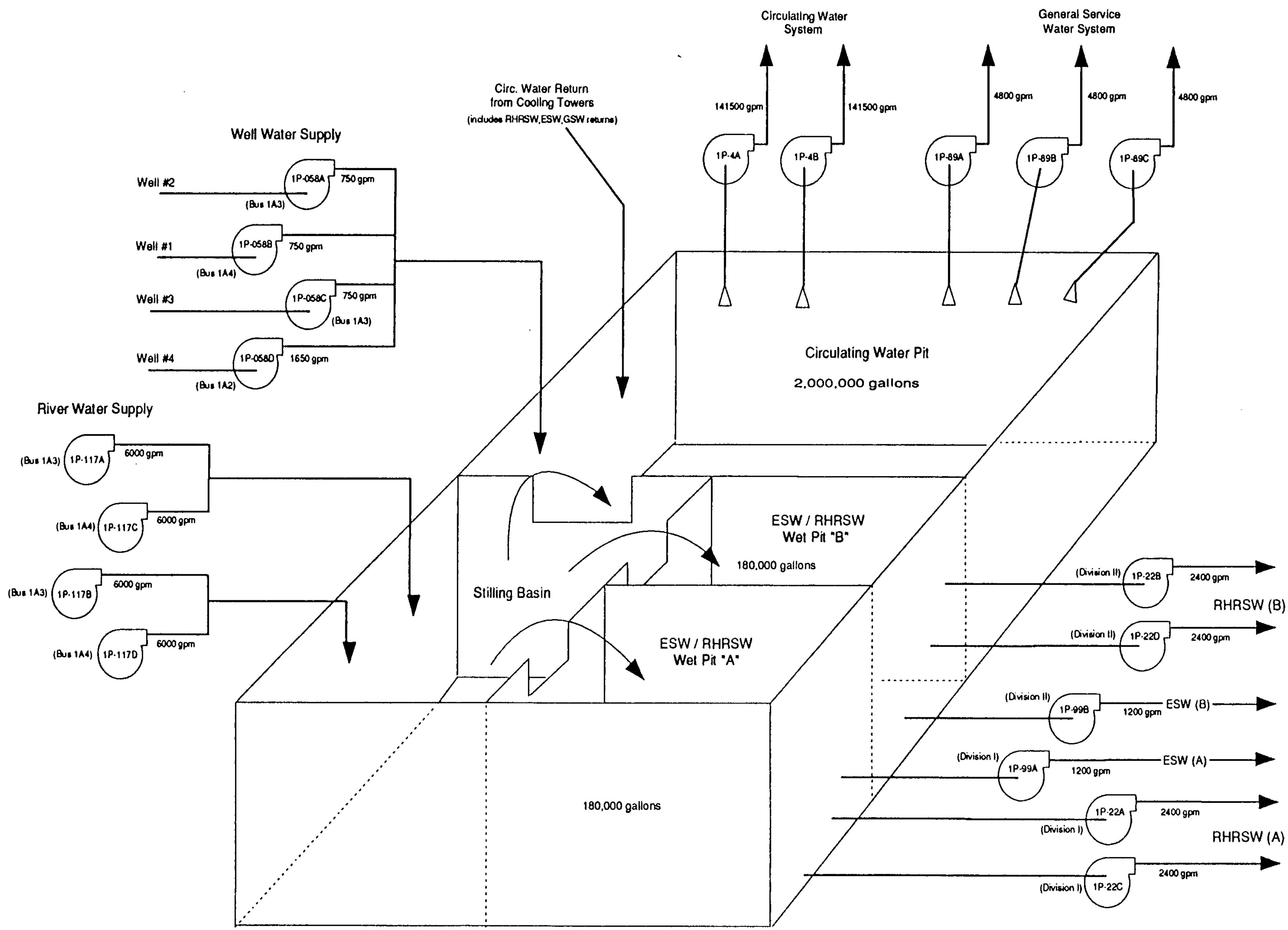


Figure 4.8-8 DAEC Water Pits: Sources and Discharge Points

Background

The EPGs were developed primarily to prevent and mitigate events prior to core damage. No calculations were performed for severe accidents to demonstrate that radionuclide releases were minimized by the actions directed in the EPGs. In fact, minimizing radionuclide releases is not even an objective of the EPGs.

Discussion

There is a set of very low frequency severe accidents for which the containment may be at elevated pressures (i.e., above the containment vent pressure) and for which the EOPs would dictate that injection to the RPV be terminated when containment pressure exceeds MPCWLL.

One of the areas of the EPGs which may have a strong impact on the IPE assessment relates to the MPCWLL treatment. The MPCWLL implications for the IPE are discussed as follows:

- The MPCWLL has associated with it directions to terminate all injection from external water sources if MPCWLL is exceeded.

- When such injection is terminated, the EOPs may therefore have eliminated the only injection source capable of preventing core damage. This can then lead to:

  a) High RPV pressure, if containment pressure exceeds the point at which the compressed gas system can maintain SRVs open (e.g. 100 psig inside containment)

b) Core damage, if all injection to the RPV is terminated or becomes unavailable due to the high containment pressure (> 90 psig) and consequential high RPV pressure (> 900 psig).

- The results of such an event have been shown (using integrated severe accident codes such as MAAP) to lead to the failure of the RPV and containment simultaneously. This is calculated to cause the energetic release of radionuclides at the time when the highest flow rates are present and result in sweeping fission products to the environment.

- Because such a strategy can lead directly to core damage and a subsequent containment challenge it is judged prudent to not terminate water injection to the containment under any circumstances for which core degradation may be aggravated by the termination of injection. This can be addressed in Accident Management investigations.

4.8.14 Reactivity Control

Because failure to scram sequences from the Level 1 analysis are postulated to challenge containment integrity early in a sequence, measures to control reactivity are extremely important for both:

- Core damage and containment failure prevention in Level 1

- Safe stable state in Level 2

During severe accidents and core melt progression, it has been found experimentally that the control rods may be the first elements to be discharged or melted away from the core region. If water is restored to the damaged core then reactivity control may be seriously jeopardized. As a result, it may be prudent to have borated the RPV during the reflood process.

The latest BWROG Rev. 4 EPGs have been implemented for level/power control to enhance the response to ATWS. For severe accidents it may be prudent to have the EOPs direct use of the SLC system whenever serious core damage has occurred.

4.8.15 Summary of Insights

A summary of Level 2 insights is given in tabular form in Table 4.8-6.

Table 4.8-6

Summary of Insights
Level 2 DAEC IPE

Functional Requirement	Insight
Containment Isolation	Containment isolation is highly reliable. The operating experience of DAEC and other BWRs indicates that containment isolation is reliable and that early release due to containment isolation failure is a negligible contribution to risk.
Depressurization	DAEC EOPs specify depressurization for most situations required. DAEC EOPs specify halting depressurization at 200 psig when turbine-driven systems are available but low pressure injection systems are not.
In-vessel Recovery	The DAEC EOPs are directed to the restoration of adequate core cooling even during degraded core states. Only minimal credit has been given in the analysis for this in-vessel recovery. This may be a conservatism in the analysis.

Table 4.8-6

Summary of Insights
Level 2 DAEC IPE

Functional Requirement	Insight
Ex-vessel Recovery	The use of CS or DW spray in lieu of LPCI appears to be most useful in response to degraded core conditions. This conclusion is based on MAAP calculations which indicate the potential for increased drywell temperatures for LPCI injection cases when debris remains in the RPV. Such conditions could lead to premature failure of containment. The prioritization of injection systems may be an action that could be included in future accident management development. The current Drywell Spray Initiation Curve (DWSI) for DAEC provides restrictions on when the drywell sprays can be initiated. Given this curve, it appears that for many of the severe accidents investigated in the Level 2 analysis that the curve will not be satisfied at all or only for a short amount of time during which the sprays could be beneficial. Drywell sprays have the advantage that they can supply water to the drywell floor prior to RPV breach. This water would collect in the DAEC sumps and up to the level of the downcomers (~ 1 ft.). The DAEC MAAP runs have shown that debris can be effectively quenched when it is discharged from the RPV to this waiting pool of water. This would preclude the debris attack and failure of the drywell shell. Drywell shell failure due to debris attack can be prevented if drywell sprays are initiated before RPV breach and the drywell floor is filled with water to quench the debris. Based on the evaluation using MAAP calculations for DAEC specific severe accidents, the following sequence types would not allow or call for spray initiation before RPV breach: · Class IIIC · Class V · Class ID

Table 4.8-6

Summary of Insights
Level 2 DAEC IPE

Functional Requirement	Insight
Ex-vessel Recovery (con't)	On the other hand, the following classes appear to satisfy the DWSI curve prior to RPV breach: • Class IA/IC • Class II • Class IV It is recognized that regardless of the flexibility offered by the DWSI curve, additional changes to the EOPs may also be required to remove any ambiguity regarding the diversion of injection sources away from the RPV when adequate core cooling is not assured, i.e., low reactor water level.
Phenomenological Effects	DCH, steam explosions, vapor suppression failure, etc. are found to have the potential to lead to relatively high releases, but the net effect is a relatively small impact on risk (i.e., frequency of large release) since the sequence frequencies (probabilities) are so low.
Shell Integrity: DW Spray Usage or Debris Cooling	In addition to core spray drywell spray offers an additional alternative to the control of drywell temperature to avoid premature containment failure. Therefore, an accident management strategy may seek the initiation of drywell sprays; this may require the relaxation of the restrictions on the use of the drywell sprays in the Drywell Spray Initiation (DWSI) curve of the EOPs, and/or relaxation of the requirement to assure adequate core cooling under certain additional conditions.

Table 4.8-6

Summary of Insights
Level 2 DAEC IPE

Functional Requirement	Insight
Containment Venting	Containment venting per the EOPs can provide a benefit in prevention of core damage and additional benefit in containment overpressure protection under severe accidents. Wetwell venting has profoundly greater potential for radionuclide scrubbing than if the drywell venting is used. There is essentially no DF for drywell venting. Therefore, drywell venting should be a last resort vent method. However, for core damage sequences, the timing of radionuclide release can be substantially affected by containment venting. In fact, releases may occur through venting when the release may otherwise be prevented. For other cases the release may occur 20 hours earlier than otherwise from a release of noble gases and scrubbed fission products during a venting operation.
Containment Flooding	Given the current state of knowledge regarding severe accident phenomenology, the DAEC EOPs have established a near optimum balance among the contingency procedures which the operator can implement. The EOPs generally define one of the following: the optimum procedural path; a procedural pathway that is close to optimum; or, a pathway for which insufficient analytical (and experimental) information is available to more precisely define the optimum pathway. Changes in the current understanding of severe accident phenomena or in the philosophy of dealing with severe accidents may impact some of the EOP steps and contingency actions. A possible improved response for current containment flood sequences for which the current EPG directions result in the highest potential consequences at the earliest time, is to provide the operators guidance on protecting containment and cooling debris using methods that do not require venting the RPV and avoid using the DW vent unless no other alternative exists. Alternate actions have been shown to produce substantially lower releases and much longer times to failure. In fact, no action is better than action directed by the EPGs.

Table 4.8-6

Summary of Insights
Level 2 DAEC IPE

Functional Requirement	Insight
Containment Injection	There is a set of very low frequency severe accidents for which the containment may be at elevated pressures (i.e., above the containment vent pressure) and for which the EOPs would dictate that injection to the RPV be terminated when containment pressure exceeds MPCWLL. Because such a strategy can lead directly to core damage and a subsequent containment challenge, it is judged prudent to not terminate water injection to the containment under any circumstances for which core degradation may be aggravated by the termination of injection. This can be addressed in Accident Management investigations.

4.9 SENSITIVITY EVALUATION FOR DAEC

As part of the containment evaluation, there are phenomenological and probabilistic issues (e.g., system reliability, operator action) that can have a large impact on the course of the events or the radionuclide release magnitude and timing. Both types of issues become candidates for sensitivity analysis. The DAEC CET provides a structure to perform sensitivity studies on issues for which a large uncertainty may exist.

Probabilistic and phenomenological uncertainties are addressed in this section to ensure that appropriate accident management actions, which may be strongly influenced by these uncertainties, are identified. These uncertainties are, in general, addressed quantitatively using either ranges of probabilities or deterministic computer calculations to simulate alternative modeling assumptions. In a few select cases, the uncertainties are discussed qualitatively to ascertain their impact on accident management actions. The accident management insights from these sensitivity evaluations are summarized in Section 4.8.

This section includes the following information:

- Approaches to sensitivity (Section 4.9.1),
- Issues for which an uncertainty or sensitivity study is desirable (Section 4.9.2), and
- Deterministic sensitivity studies (Section 4.9.3).

Table 4.9-1 (Table A.5 from NUREG-1335) identifies parameters that past studies indicate as prudent choices for sensitivity cases. From these parameters, the phenomena and assumptions used in MAAP that are subject to the most uncertainty for DAEC have been investigated.

Most of the resources of the DAEC IPE back-end analysis effort are devoted to treating uncertainties that could directly influence accident management strategies, in general, and containment failure time in particular. Stated more narrowly from the standpoint of accident management, the principal goal in performing sensitivity studies is to identify and understand physical phenomena that put a premium on specific operator actions. In addition, accident management actions have been identified to be effective for controlling or preventing postulated phenomena under certain accident sequence conditions or assuming certain modeling conditions. These phenomena may not be physically possible or may behave differently than the modeling assumptions. It is judged that it may also be prudent in the future to investigate the impact of the accident management actions over a range of postulated physical models on phenomenological assumptions.

Fewer resources should be devoted to phenomena that are to varying degrees: (1) generic rather than plant-specific; (2) being studied elsewhere on a generic basis; or (3) which do not impact accident management strategies directly, even though they could affect the source term from a given sequence. For such phenomena, only best-estimate treatments are recommended for the purpose of developing the IPE.

The results of the sensitivity cases are described in Section 4.9.3. The following 2 sections identify possible approaches to performing the sensitivity analysis and identifies the methods chosen for DAEC.

Table 4.9-1

NRC IDENTIFIED PARAMETERS FOR SENSITIVITY STUDY
(NUREG-1335)

- Performance of containment heat removal systems during core meltdown accidents

- In-vessel phenomena (primary system at high pressure)

    - H_2 production and combustion in containment
    - Induced failure of the reactor coolant system pressure boundary
    - Core relocation characteristics
    - Mode of reactor vessel melt-through

- In-vessel phenomena (primary system at low pressure)

    - H_2 production and combustion in containment
    - Core relocation characteristics
    - Fuel/coolant interactions
    - Mode of reactor vessel melt-through

- Ex-vessel phenomena (primary system at high pressure)

    - Direct containment heating concerns
    - Potential for early containment failure due to pressure load
    - Long-term disposition of core debris (coolable or not coolable)

- Ex-vessel phenomena (primary system at low pressure)

    - Potential for early containment failure due to direct contact by core debris
    - Long-term core-concrete interactions:

        -- Water availability
        -- Coolable or not coolable

4.9.1 Sensitivity Approaches

The approaches for investigating key sensitivities can take on a wide range of methods, and cover a wide spectrum of breadth and depth of investigation. This section identifies optional approaches that could be used to satisfy different objectives:

Resource Intensive Approach:

Identify all parameters or modeling assumptions that have uncertainties of larger than an error factor of 3, and include a sensitivity of varying these. In addition, identify coupled parameters that also need to be varied.

IPE Approach

- Satisfy the requirements of IPE Generic Letter 88-20 for the Level 2 portion of the IPE. Address the phenomenological issues posed by the NRC.
  - Probabilistically

    or

  - Deterministically.
- Identify a limited sample of additional containment or plant specific issues that should be addressed.

Accident Management Sensitivity Approach

This group of sensitivities would be developed to support additional investigations to attempt to optimize accident management actions or hardware use that could be implemented as part of an accident management response to severe accidents.

Conclusion

As part of this IPE report, Iowa Electric has selected the IPE Approach. Therefore, the following sections will present the results of sensitivity assessments on a group of selected issues as affected by plant specific features. The uncertainties are, in general, addressed quantitatively using probabilistic or deterministic methods. In a few select cases, the uncertainties are discussed qualitatively to ascertain their impact on accident management action. Section 4.8 includes a summary of the accident management insights derived from the plant specific evaluation and uncertainty investigation.

4.9.2 MAAP Sensitivity Runs Overview

To ensure that a broad scope of possible severe accident progression is considered in the DAEC IPE, several sensitivity analyses were performed using the MAAP code or probabilistic modeling sensitivities.

MAAP cases were selected to evaluate the key functional events for mitigating radionuclide releases associated with severe accidents at the DAEC plant. This base set of MAAP calculations represent a best estimate of how the plant will respond under severe accident conditions. However, it is recognized that considerable uncertainty exists in the modeling of the complex phenomena associated with such accidents. One should recognize that MAAP does not contain detailed models for all phenomena. Indeed, there are more mechanistic codes available such as CONTAIN and SCDAP/RELAP. These are generally used in a research setting and are not considered to be suitable for use in IPEs due to long run times and the much greater requirements they impose on the user for specialized knowledge of severe accident phenomena. An alternative code whose scope is similar to MAAP is MELCOR. However, less experience has been accumulated with the MELCOR code than with MAAP. Therefore, MAAP was chosen as the best available tool to perform the plant specific evaluation. However, where available, MELCOR results on similar plants are also utilized.

The probabilistic model sensitivity analyses were performed to examine issues that involve phenomena that are beyond the capability of MAAP (e.g., steam explosions).

Table 4.9-2 summarizes an extensive list of possible sensitivity calculations that could be performed to support a full PRA. Within Table 4.9-2 are identified those phenomena or items that are:

- Are recommended by GL 88-20 or NUREG-1335 to be addressed as part of the IPE;
- deemed sufficiently important to address; and
- useful for consideration in an accident management program.

Table 4.9-2

List of Sensitivity Items

Sensitivity Item	Specified by GL 88-20 or NUREG-1335	Examined In DAEC IPE Response	Proposed Cases for Accident Management Investigations
In-vessel Core Melt Progression			
- Hydrogen Production	X	X	
- Temperature of Melt			
- Model for control rods			
- Model for candling			
- RPV breach model and assumptions	X		
- In-vessel steam explosion		X(P)	
- Induced primary system LOCAs	X	X(P)	
- In-vessel recovery			X
- In-vessel reactivity excursion		X(P)	X

X - Identifies sensitivity cases satisfying the column heading.

(P) - Probabilistic sensitivities performed.

N/A - Not Applicable to DAEC Mark I

Table 4.9-2

List of Sensitivity Items

Sensitivity Item	Specified by GL 88-20 or NUREG-1335	Examined In DAEC IPE Response	Proposed Cases for Accident Management Investigations
Ex-vessel Core Melt Progression			
- Debris Temperature	X		
- Amount of debris discharged from vessel			
- DW sump coolability		X	
- Coolability with water present	X	X (P)	
- Effective DW floor area		X	
- Pool Bypass			
-- Vacuum Breaker		X	
-- Downcomers		N/A	
-- Other		N/A	
- Quenching Model in Pool (MKII)	X	N/A	
- DCH		X(P)	
- Amount of Material		X	
-- Retained in drywell			
-- Retained in pedestal			

Table 4.9-2

List of Sensitivity Items

Sensitivity Item	Specified by GL 88-20 or NUREG-1335	Examined In DAEC IPE Response	Proposed Cases for Accident Management Investigations
Containment Failure			
- Size	X	X	
- Location	X	X	
- Pressure (Ultimate Capability)			
- Temperature			
- ATWS induced dynamic containment failure mode			
- Containment venting		X	X
- Pool bypass		X	
- Aerosol Plugging			
- Direct contact of debris	X	X	
- Pressure Rise	X	X	
Reactor Building Effectiveness	X	X	
- Hydrogen Burn			
- Circulation Established			
- Direct Release			

Table 4.9-2

List of Sensitivity Items

Sensitivity Item	Specified by GL 88-20 or NUREG-1335	Examined In DAEC IPE Response	Proposed Cases for Accident Management Investigations
Critical Safety Functions			
- Reactivity Control			
- Pressure Control			X
- High Pressure Makeup			X
- Depressurization			X
- Low Pressure Makeup			X
- Containment Heat Removal	X	X	X
- Containment Temperature Control			X
- Containment Pressure Control			X
- Combustible Gas Control			X
- Containment Water Level Control			X
- Containment Flooding		X	X
- Drywell Spray Use		X	X

Table 4.9-2

List of Sensitivity Items

Sensitivity Item	Specified by GL 88-20 or NUREG-1335	Examined In DAEC IPE Response	Proposed Cases for Accident Management Investigations
Other Actions			
- Accident Management Actions			X
- Disregard DWSI Curve			X
- Containment Flood Always by Procedure			
- Containment Flood With no RPV Vent			X
- Containment Flood Only Late in Sequence		X	X
- Fill DW with water (MKI)		X	X
- Vent to 0 psig			X
- Vent to control 40-60 psig			X
- Vent to control 60-90 psig			X

4.9.3 Deterministic Sensitivity Studies

As part of the evaluation of uncertainties through the performance of specific sensitivity cases, Iowa Electric has reviewed the status of the NRC position on the series of so-called "issue" papers. Section 4.4.1 identifies these issues and their disposition for DAEC. Those that are carried forward to the performance of specific sensitivities include the following:

- Core Melt Progression
- In-vessel Hydrogen Generation
- RPV Pressure at Vessel Failure
- Late CsI Revaporization from the RPV
- Debris Spread in Containment
- Amount of Debris Retained in RPV
- Ex-vessel Debris Coolability
- Shell Failure
- Containment Failure Location
- Containment Failure Area
- Reactor Building Effectiveness

In addition, the MAAP model parameters generally represent inputs to phenomenological models in which significant uncertainties exist. Variations in the values of these parameters can be made to assess the impact of uncertainties in important physical models. The best estimate values used in the DAEC IPE are provided in the DAEC IPE MAAP Parameter File. These best estimate values were directly taken from the "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B," Gabor, Kenton and Associates, EPRI 1990. Sensitivity analyses were performed in accordance with the recommendations in the EPRI/GKA report as well as additional areas deemed important for DAEC.

The resulting list of deterministic sensitivities performed for DAEC is a combination of the NRC "open issues" and the GKA recommended sensitivities and includes the following:

- Core Melt Progression: Amount of Residual Debris in RPV (Section 4.9.3.1)
- Debris Coolability (Section 4.9.3.2)
  - Non-Condensible Gas Generation (Section 4.9.3.2.1)
  - Debris Cooling in the Sump (Section 4.9.3.2.2)
  - Pedestal Attack (Section 4.9.3.2.3)
  - Effective Area of Drywell Floor (Section 4.9.3.2.4)
- Aerosol Plugging (Section 4.9.3.3)
- Core Blockage (Section 4.9.3.4)
- Containment Failure Mode (Size and Location) (Section 4.9.3.5)
  - Containment Failure Area (Section 4.9.3.5.1)
  - Containment Failure Location (Section 4.9.3.5.2)
  - Drywell Shell Failure (Section 4.9.3.5.3)
- Reactor Building Modeling Assumptions (Section 4.9.3.6)
- Equipment Mass in Drywell and Effect on Drywell Temperature Post RPV Breach (Section 4.9.3.7)
- Pool Decontamination Factor (DF) (Section 4.9.3.8)
- Containment Flooding Sensitivity Evaluation (Section 4.9.3.9)
- Sensitivity of Radionuclide Release to Level 1 Sequence Type (Section 4.9.3.10)
- Drywell Spray Usage Under Severe Accident Conditions (Section 4.9.3.11)
- Drywell Spray Usage Under Severe Accident Conditions (Section 4.9.3.12)
- High Pressure Melt Injection (Section 4.9.3.13)
- Summary (Section 4.9.5)

4.9.3.1 Core Melt Progression: Amount of Residual Debris in RPV

The amount of core material remaining in the RPV is calculated by MAAP. The core begins to melt and then relocates into lower regions of the core. This continues until the lowest core node in any radial region becomes completely molten at which time all molten core material exits the core region and moves into the lower plenum.

There are a large number of bottom head penetrations to accommodate the control rod drive mechanism assembly penetrations, instrument guide tube penetrations, and a drain line penetration near the low point of the bottom head.

In past MAAP analyses, it has been observed that the amount of material molten at the onset of movement into the lower head is strongly dependent on the amount of in-vessel Zircaloy oxidation. More oxidation tends to heat up the core and results in a larger mass of molten material moving out of the core region. Due to various modeling assumptions and a general lack of detail in representing core melt progression, there is a spectrum of results. One outcome involves all material exiting at vessel failure, and the other possible scenario is observed in the cases in which some of the core material remains behind in the RPV. For the DAEC base cases, all debris is discharged near the time of RPV breach.

The BWRSAR model includes the fact that after lower plenum dryout, the debris bed temperature would increase, causing thermal attack and failure of the control rod guide tube structure in the lower plenum, which the debris would completely surround to a depth of about 10 ft. Since the control rod drive mechanism assemblies and the control rod guide tubes support the core, the remaining standing outer regions of the core would be expected to collapse into the vessel lower plenum when these support columns fail.

The BWRSAR model for core melt progression and RPV bottom head attack supports the MAAP calculation that little residual debris would be retained in the RPV.

It is important to understand the impact of each of these core melt progression scenarios. If all of the core material exits the RPV it will provide more mass for core/concrete interaction, core/water interactions, and debris attack of the shell. If core material remains behind in the vessel, it may contribute to late fission product revaporization and drywell heat-up due to radiative heat transfer from the RPV to the drywell atmosphere.

The amount of core material remaining in the reactor vessel following vessel failure will influence in-vessel revaporization and drywell heat-up. The MAAP parameter FMAXCP specifies the minimum amount of core material capable of supporting the remainder of the core. When the fractional amount of core material remaining in the core region is less than FMAXCP, the remaining core material is forced out of the core and into the lower plenum. The DAEC Base Cases were performed assuming that no residual debris remains in the core region of the RPV long term following RPV bottom head breach.

Two MAAP cases were compared to obtain the effect of residual debris in the RPV affecting the course of the severe accident. This comparison is used later to provide insights into accident management.

The 1D-LF analysis assumed that 100% of the core was discharged at the time of vessel failure. A sensitivity case (1D-LFB) was run in which this assumption was not made. The results show that for the sensitivity case, all of the core did come out of the vessel, but with a delay of about 1 hour. Table 4.9-3 shows that the results for these two cases were similar.

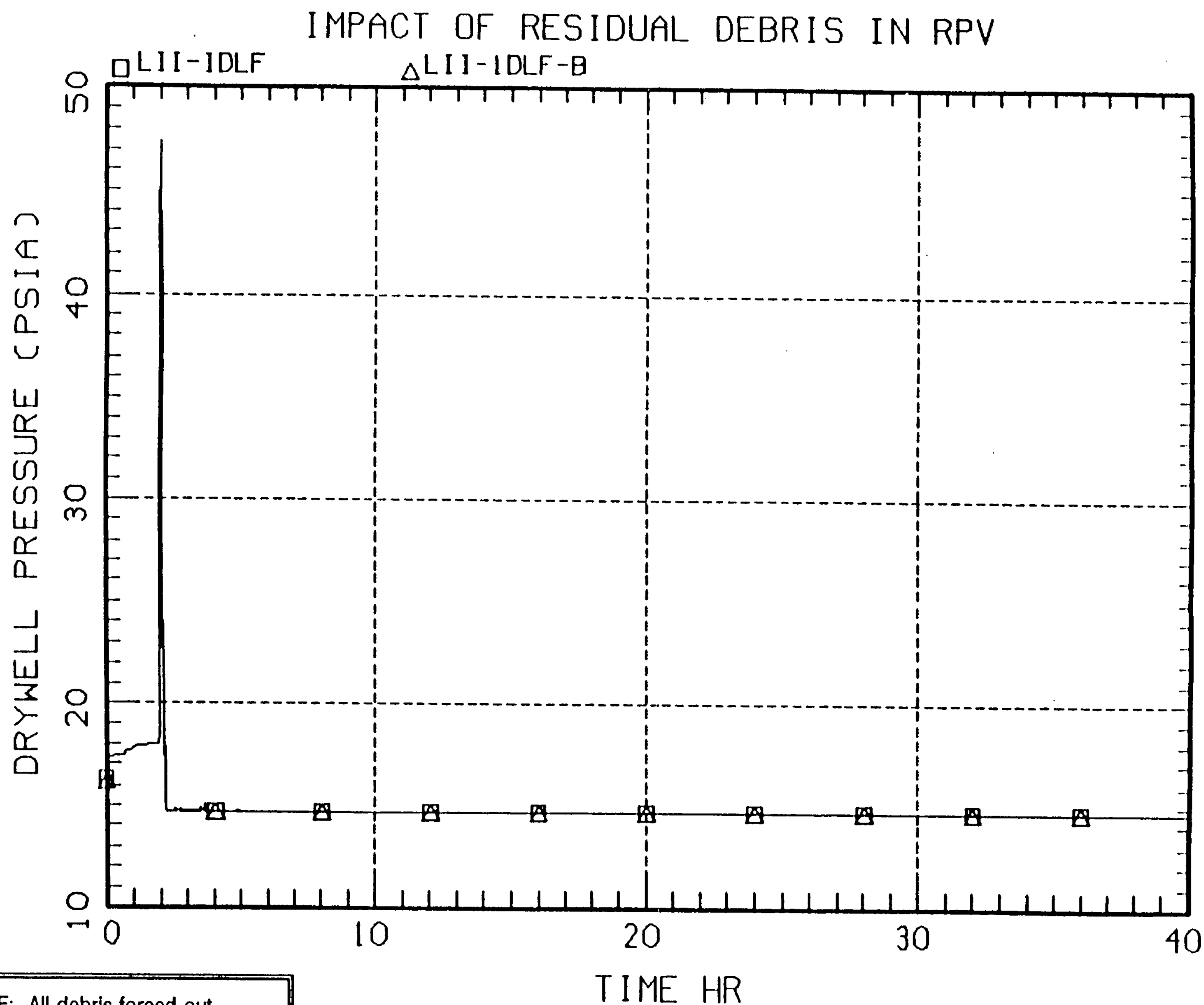
Table 4.9-3 DEBRIS RETAINED IN RPV						
MAAP Case	Accident Class	Description	Containment Failure		CsI Release	
			Time (hr)	Location	in RB	to Env.
LII-1D-LF	ID	All of core out of RPV at vessel failure	2.04	Shell (large)	37.7%	4.6%
LII-1D-LFB	ID	Delayed discharge of core (1 hr. delay)	2.04	Shell (large)	34.8%	5.0%

Figure 4.9-1 and Figure 4.9-2 provide the comparison of the containment pressures and temperatures in MAAP. As can be easily seen, the 1 hour delay in debris exit from the vessel does not appreciably affect the drywell heat-up.

Residual debris can be retained inside the RPV for rapid core melt scenarios with little clad-steam interaction, such as RPV rupture or large LOCAs.

As examples of that possibility, the following table of Class IIIA scenarios (i.e., RPV rupture) was assembled to show the potential impact of the residual debris in the RPV on containment performance.

It appears that for cases with: (1) no injection; (2) rapid core melt progression; and, (3) debris retained in the RPV that higher temperatures are possible at earlier times. Specifically, Case 3A01A shows that with an RPV rupture and little in-vessel clad water reaction and 30% of the core retained in-vessel that a drywell temperature of

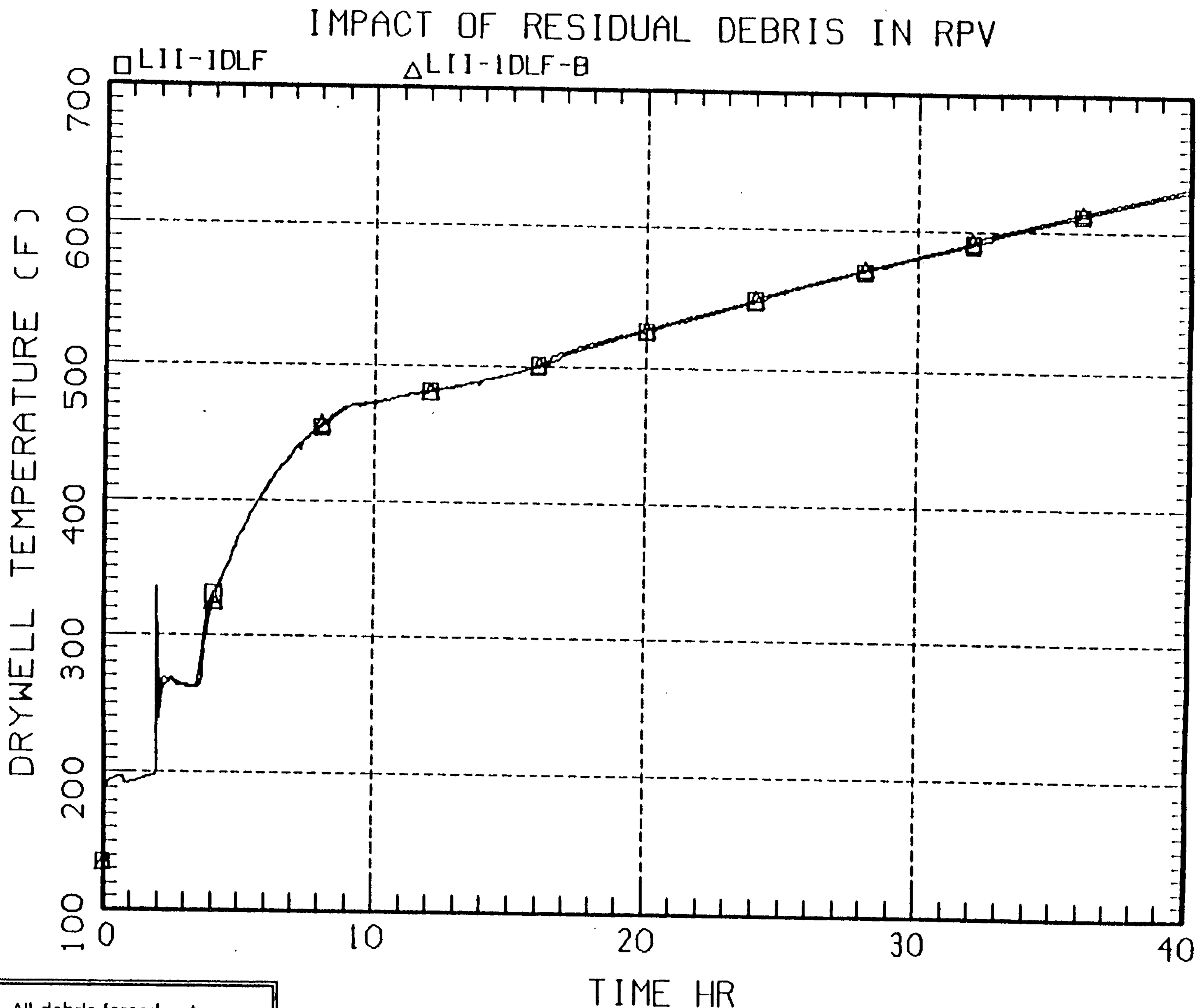


LII-1DLF: All debris forced out of RPV at time of vessel failure

LII-1D-LF-B: All debris forced out of RPV 1 hour after vessel failure

Figure 4.9-1

Drywell Pressure for Sensitivity to Residual Debris Remaining in RPV



LII-1DLF: All debris forced out of RPV at time of vessel failure

LII-1D-LF-B: All debris forced out of RPV 1 hour after vessel failure

Figure 4.9-2

Drywell Temperature for Sensitivity to Residual Debris Remaining in RPV

Comparison of Class IIIA Results									
Case #	% Core in Cont.	Water on Floor?	SPC	WWV	DWF	DW Eqpt Mass (lb)	Time of Cont. Failure	Peak DW Temp. °F	Time of Peak DW Temp (Hr)
3A01	70%	Yes	Yes	Yes(1)	No	2.68E6	N/A	723	40
3A02	70%	Yes	No	Yes	No	2.68E6	18.1	675	40
3A03	70%	Yes	No	No	Yes(2)	2.68E6	26.2	635	26.2
3A01A	70%	No	No	No	Yes(3)	2.68E6	18.4	1050	40
3A01B	70%	Yes	Yes	Yes(1)	Yes(4)	1.00E6	35.7	834	35.7

Notes:

(1) Wetwell venting not required
(2) at 26.2 hrs, 105 psia & 635°F
(3) at 20 hrs, 64 psia & 800°F
(4) at 36 hrs, 42 psia & 830°F

over 1000°F is achieved within 40 hours. This compares with the 1D cases (1D-LF and 1D-LFB), which show drywell temperatures of less than 650°F at 40 hours. This means that the drywell head failure mode on high temperature may be important for such sequences as RPV rupture and large LOCA. Therefore, QD = 1.0 for these scenarios.

In summary, these sensitivity cases indicate that core debris will usually not be retained in-vessel, except for RPV rupture or large LOCA events without injection. However, there may be unusual situations in which residual debris does remain behind in the RPV. If core debris remains behind in the original core region, containment heat-up may occur to the point that the integrity of the drywell could be threatened regardless of RPV injection and RHR availability. Accident management efforts should realize this as a possible scenario and make appropriate guidance to mitigate the outcome. Therefore, an insight from these MAAP runs is that there may be other water injection methods that could minimize the potential for such temperature induced containment failure modes. Specifically, drywell spray or core spray would be better choices for coolant injection to eliminate this as even a <u>potential</u> failure mode.

4.9.3.2 Debris Coolability

Without continued water injection after vessel breach, the core debris will dry-out and begin to heat-up. Eventually, the debris will begin to interact with the concrete basemat. There is also a possibility that core-concrete attack can occur in the presence of an overlying water pool. Prior to containment failure, any fission products that are evolved by core-concrete attack or by long term revaporization will be deposited in the drywell, entrained in an overlying water pool, or transported to the suppression pool. At containment failure, the amount of fission product release will be dictated by the airborne mass of radionuclides at failure and the subsequent rate of their revaporization from the drywell and RPV.

Three separate aspects of debris coolability will be discussed:

- Non-condensible gas generation
- Debris cooling in the sump
- Pedestal attack.

Four sensitivity cases are discussed below. The following conditions applies to all four cases.

- Water is injected to the ex-vessel debris to cover the debris
- All debris is ex-vessel.

4.9.3.2.1 Non-Condensible Gas Generation

There has been a substantial amount of disagreement on the issue of debris coolability ex-vessel. Some analyses indicate that water will ingress into the debris and provide cooling. Others have pointed to the very limited experimental data base and concluded that an impermeable crust will form that isolates the water from the debris. The EPRI document on MAAP sensitivity analysis indicates that selected cases should be run assuming that the debris-to-water heat transfer is limited to approximately 300 kw/m^2. This is done by a modification to the model parameter FCHF.

The containment drywell sumps for DAEC are located in the drywell pedestal (see Figures 4.9-3 and 4.9-4). The MAAP parameter file for DAEC has been modified to explicitly account for the sump depth in the core-concrete interaction (CCI) calculation. This has been accomplished by setting the inside pedestal floor area equal to the sump area. In addition, the pedestal floor area not accounted for by application of this modeling assumption is accounted for in the ex-pedestal floor area. Therefore, all MAAP cases for DAEC explicitly include the non-condensible gas generation associated with both the sump gas generation (i.e., 3.2' depth), and the drywell floor gas generation for CCI.

Two MAAP sensitivity cases were run to investigate the uncertainties in debris-to-water heat transfer. Base cases 1A01 and 1A03 were rerun assuming limited debris-to-water heat transfer (FCHF = .02) corresponding to an upward heat flux of about 300 Kw/m^2. The two base cases assumed a value for FCHF of .09, which results in an upward heat transfer rate of approximately 1 MW/m^2. Table 4.9-5 summarizes the results of the debris coolability sensitivity runs compared with the two base cases.

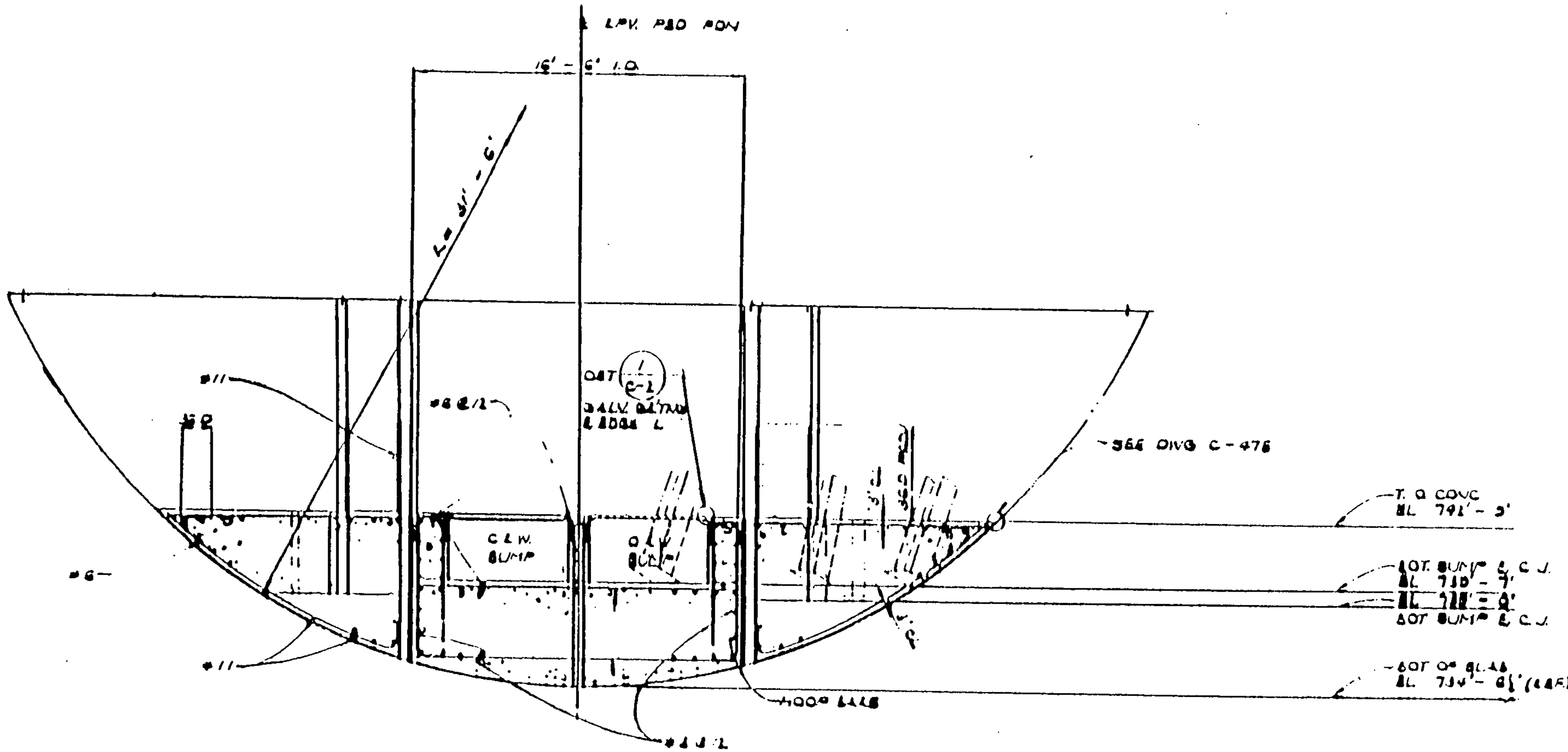
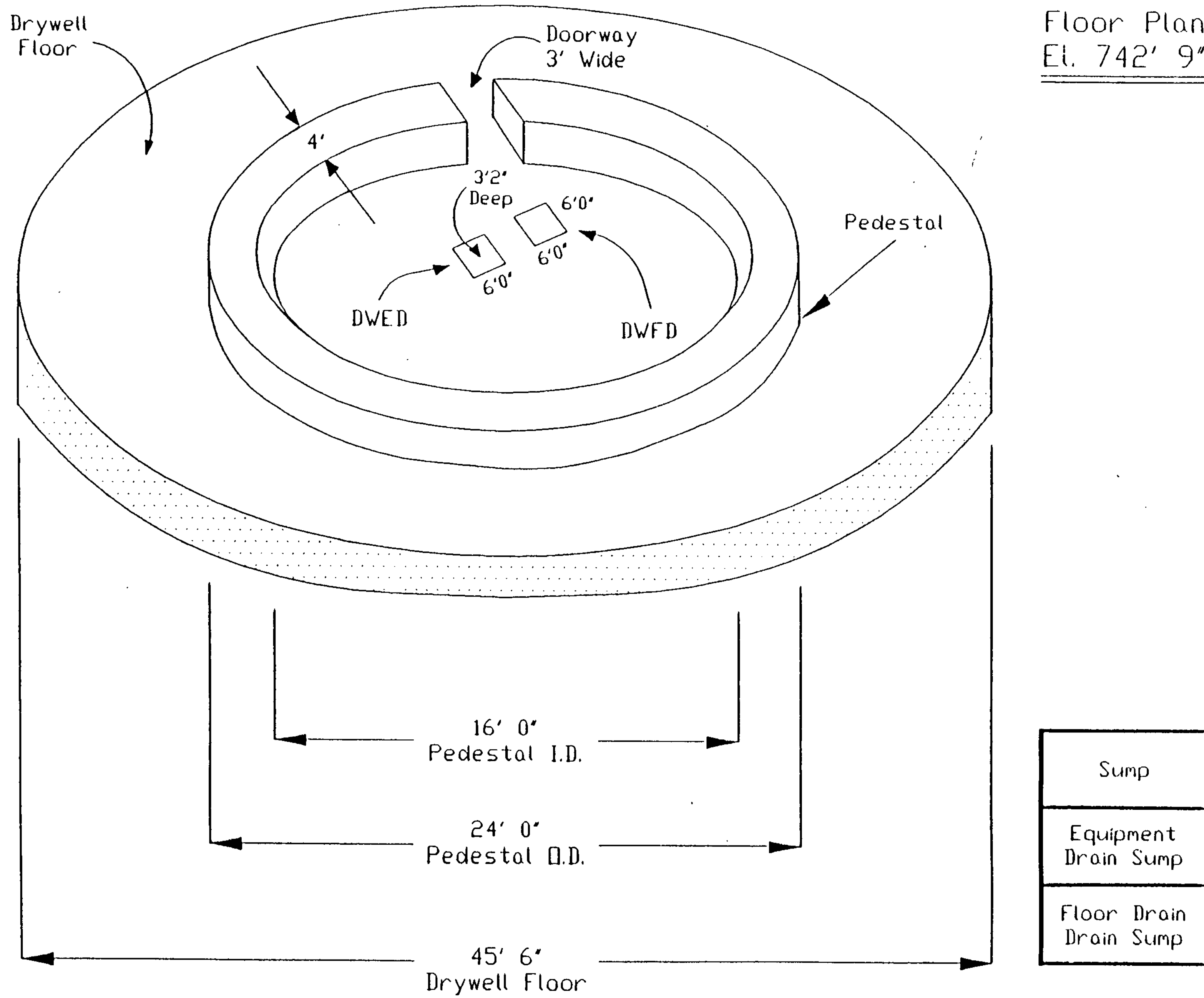


Figure 4.9-3

DAEC Drywell Floor Showing the Equipment and Floor Drain Sumps
Inside the Pedestal



Sump	Volume
Equipment Drain Sump	114 ft^3
Floor Drain Drain Sump	114 ft^3

Figure 4.9-4

Pedestal and Sump Configuration for Duane Arnold

Due to an increase in the non-condensible gas generation in case 1A03A, the containment pressurizes faster and fails 3 hours earlier than in the base case. The CsI release is found to be lower for the sensitivity case due to a decrease in the amount of airborne CsI at the time of containment failure. This is an example of a realistic estimate of core melt progression phenomena producing a higher magnitude release than an "apparent" conservative core concrete interaction model.

Figures 4.9-5 and 4.9-6 provide the comparisons between containment pressures and temperatures, respectively, for Cases LII-1A-03 and LII-1A-03A.

As can be noted from these two sensitivity cases, the choice of FCHF (heat transfer from the debris to the water), may dictate some of the details of the accident progression, but even a lower bound value does not significantly influence the results compared to the nominal value. Consequently, it is judged best to present all of the release results using the best estimate choice of FCHF (0.09) based on experimental data.

4.9.3.2.2 Debris Cooling in the Sump

DAEC has two 36 $ft.^2$ pedestal sumps that are 3.2 ft. deep. Combined, these sumps occupy 35 percent of the pedestal floor area at DAEC (modeled as occupying 100% of the MAAP pedestal floor). When vessel failure occurs, molten debris drops into the sumps with the overflow spilling into the surrounding pedestal region.

As mentioned in the previous section, the MAAP model explicitly accounts for these two drain sumps. The MAAP parameter file has been modified to model the sumps as occupying the entire pedestal floor area. In the MAAP model, the amount of floor area surrounding the sumps inside the pedestal is added to the drywell floor area outside

Table 4.9-5

EX-VESSEL DEBRIS COOLABILITY

(Core Melt Progression with Water Available to Debris (Ex-vessel)

	Base Case (1A1) (1), (4)	Limited Heat Transfer (1A01A) (1), (4)	Base Case (1A03) (5)	Limited Heat Transfer (1A03A) (5)
Containment Overpressure Failure (hrs)	N/A	N/A	19.5	16.1
Total H_2 Generated	613	815	664 (2)	855 (2)
Pedestal Concrete Attack (ft)	.082	2.24	.64 (2)	2.51 (2)
Drywell Concrete Attack (ft)(7)	.017	.042	.015 (2)	.043 (2)
CsI in Reactor Building	0.0% (6)	0.0% (6)	2.5% (2)	.33% (2)
CsI to Environment	N/A (6)	N/A (6)	5.2% (3)	1.3% (3)

Notes to Table 4.9-5

(1) at 24 hours
(2) at 30 hours
(3) at 60 hours
(4) 1AO1: Containment injection to debris on drywell floor, suppression pool cooling operating (no venting)
(5) 1AO3: Same as 1A01 except no suppression pool cooling.
(6) No release; containment remains intact.
(7) Note that the concrete attack depth is quoted for the drywell floor (i.e., outside the sumps). Section 4.9.3.2.3 summarizes the concrete attack depth in the sumps and pedestal area.

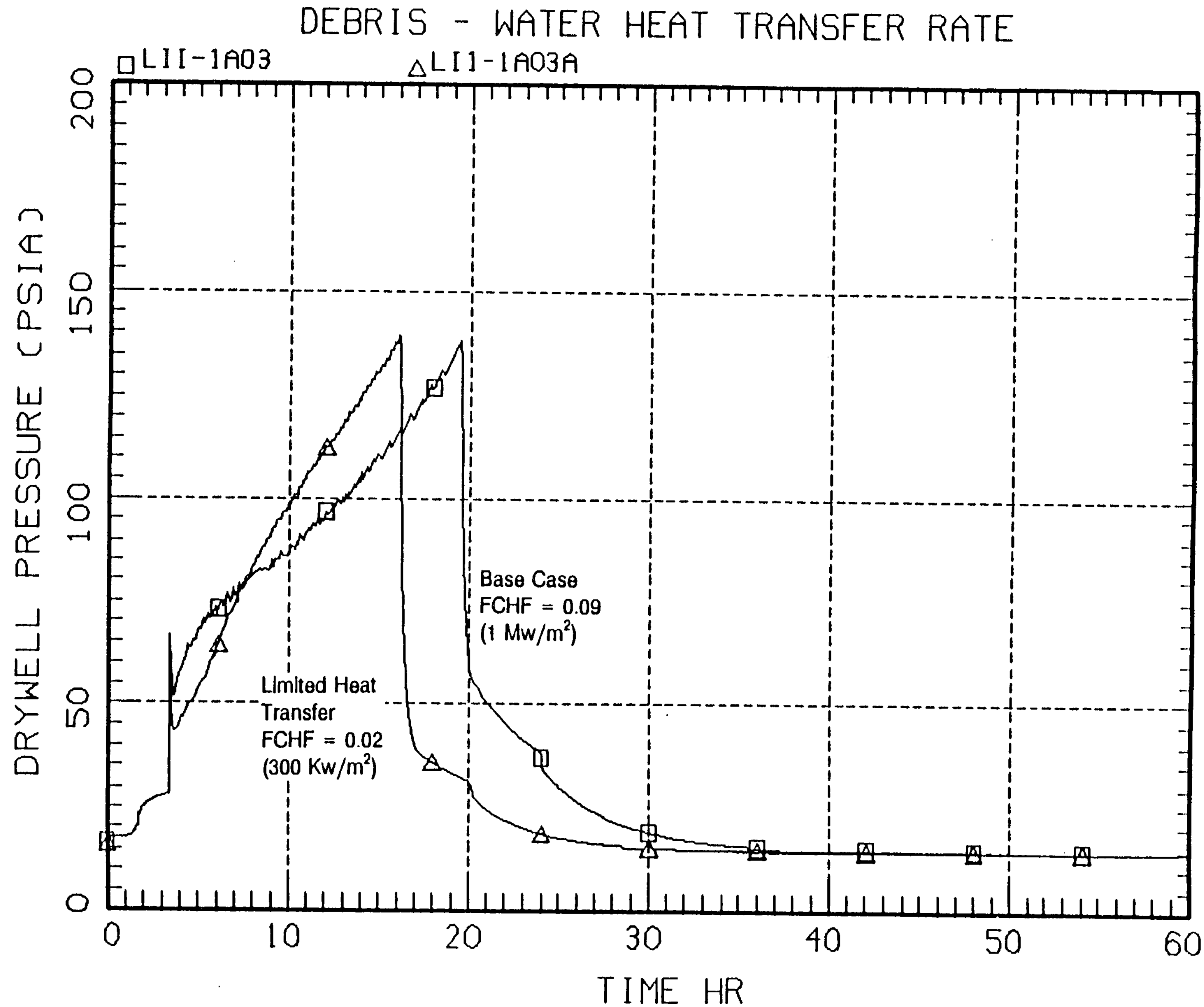


Figure 4.9-5

Drywell Pressure for Sensitivity to Water Heat Transfer for
Accidents in Which Water is Ava[illegible] to the Debris on the Drywell Floor

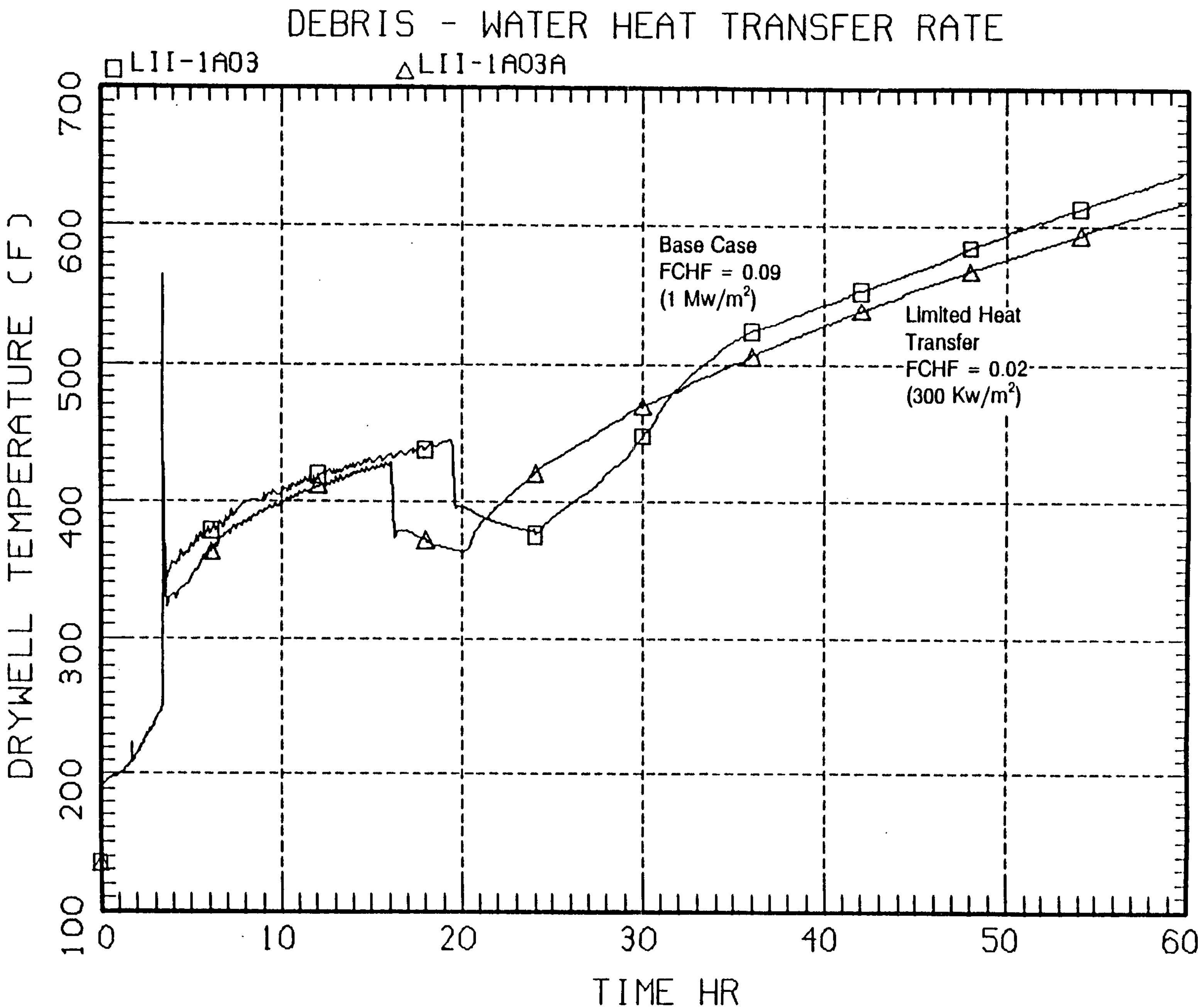


Figure 4.9-6

Drywell Temperature for Sensitivity to Water Heat Transfer for Accidents in Which Water is Available to the Debris on the Drywell Floor the pedestal. This modification allows MAAP to model debris accumulation and core-concrete interaction for the actual sump depth (i.e., inside the pedestal), before overflowing onto the pedestal/drywell floor area. Therefore, no additional sensitivity study was required to examine the modeling of debris cooling in the sumps in addition to the cases included in Table 4.9-5.

4.9.3.2.3 Pedestal Attack

There are a number of potential pedestal failure modes including:

- High pressure blowdown with inadequate relief from the pedestal region,
- High temperature of the vessel support skirt,
- Ex-vessel steam explosions causing pedestal failure, and
- Molten debris erosion of the pedestal concrete.[1]

The first three pedestal failure modes noted above have essentially been dismissed by the NUREG-1150 2nd Draft for Mark I plants; and therefore, the probability for pedestal failure due to these failure mechanisms can be estimated at approximately 1E-4 per core on-the-floor event. This leaves molten debris erosion. Table 4.9-6 summarizes the results of two MAAP cases.

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[1] Core concrete is a potential contributing failure mode to containment. For cases with no water available to cool debris, other containment failure modes are projected to occur first; well before core-concrete attack failures are induced. For cases with water present, the core concrete attack on the drywell floor can be essentially halted because of the shallow depth of the debris for the sump evaluation, the concrete attack can continue for an extended time leading to possible very long term issues relating to concrete integrity despite the presence of an overlying pool of water.

Figure 4.9-7 provides a comparison of the concrete attack depth for two of the cases in Table 4.9-6. The case with the highest concrete attack is also the one in which debris-to-water heat transfer was limited to approximately 300kw/m^2 (Case LII-1A-03A).

The MAAP cases indicate that the concrete attack is terminated within 1 to 7 hours after RPV breach for either the realistic (1 hour) or pessimistic debris cooling (7 hour) cases. Therefore, with water present MAAP predicts core-concrete interaction can be terminated without failing containment. However, because the cases being compared are without torus cooling, the containment eventually fails due to overpressure (steam plus non-condensibles). (Note the earlier containment failure time of 15 hours for the pessimistic debris cooling heat transfer coefficient case versus 20 hours for the realistic case.)

As a side light to the primary purpose of the discussion, it is noted that subsequent to containment failure that water injection is halted and core concrete interaction is initiated again (i.e., after the water on the drywell floor is evaporated). Clearly, without water core-concrete interaction continues to substantial depths with the potential for causing pedestal or basemat failure.

Figure 4.9-8 shows a generic CORCON analysis results that predict radial erosion of the pedestal wall for a "typical" BWR even with water present over the debris, of 1.25 feet in 120 minutes. Concrete erosion in the radial direction essentially ceases after the metallic and the originally heavy oxide layers invert, at about 150 minutes after vessel breach, while concrete penetration in the vertical direction continues. The initial concrete erosion rate for this CORCON case is approximately 7 inches/hour compared to 5 inches/hour for the MAAP case (1A03A). Reasons for this difference include differences in the boundary conditions (debris mass, temperature, composition) and to a lesser extent differences in the phenomenological modeling.

Table 4.9-6

SUMMARY OF CONCRETE ATTACK DUE TO MOLTEN DEBRIS WHEN WATER IS PRESENT

PEDESTAL ATTACK				
	Base Case (1A01) (1), (4)	Limited Heat Transfer (1A01A) (1), (4)	Base Case (1A03) (5)	Limited Heat Transfer (1A03A) (5)
Pedestal Concrete Attack (ft.)	.082	2.24	0.1 (2), (7)	2.2 (2), (8)
Drywell Concrete Attack (ft.)	.017	.042	.015 (3)	.043 (3)

---

(1) at 24 hours

(2) at 20 hours

(3) at 30 hours and after injection has terminated.

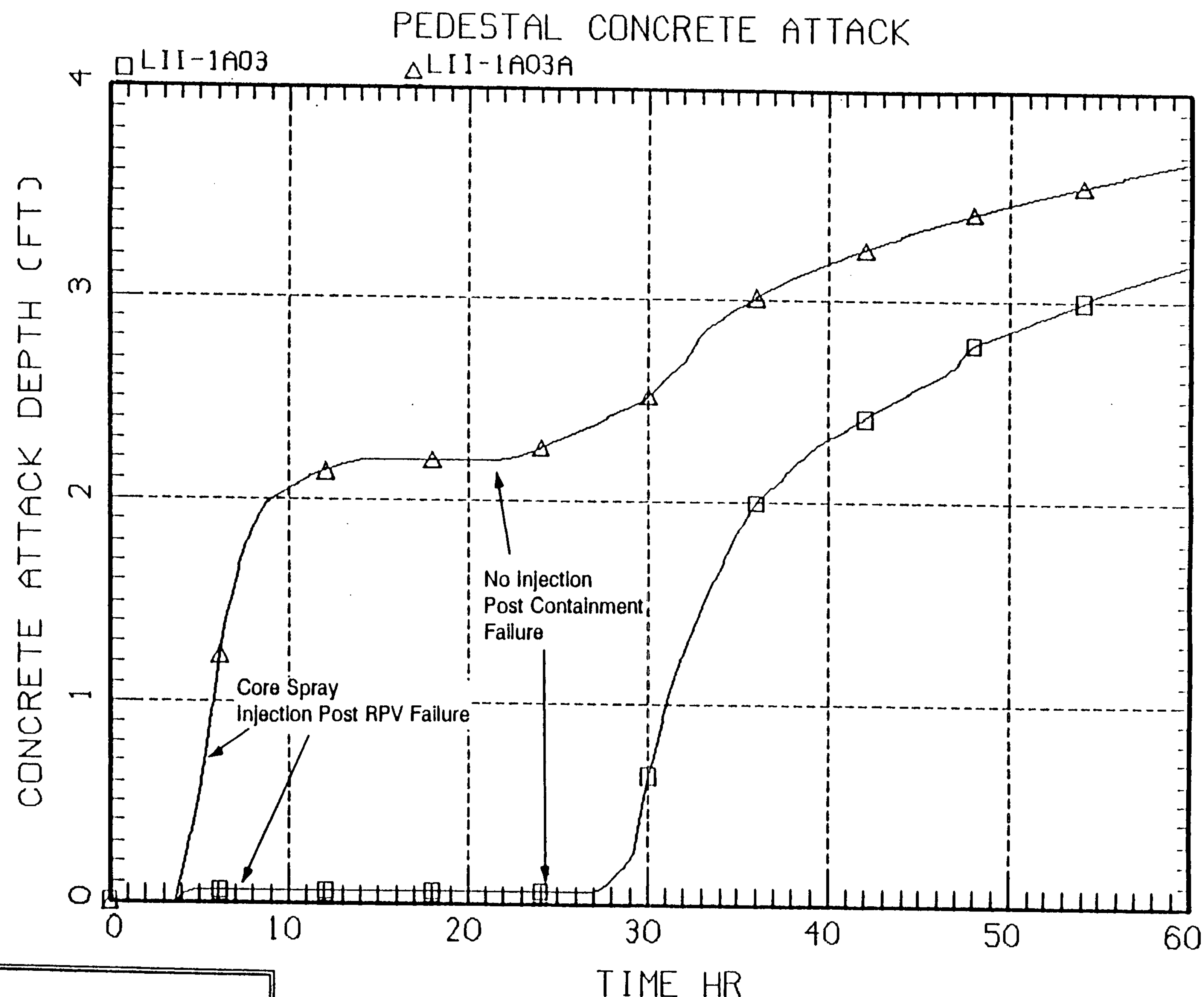
(4) 1A01: Containment injection to debris on drywell floor, suppression pool cooling operating (no venting)

(5) 1A03: Same as 1A01 except no suppression pool cooling.

(6) No release; containment remains intact

(7) With water present core-concrete interaction is terminated within 1 hour of RPV breach at approximately 0.1 ft. depth in sumps.

(8) With water present core-concrete interaction is terminated within 7 hours of RPV breach at approximately 2.2 ft. erosion depth in the sump.



Legend

▲ RPV Failure
■ Containment Failure
LII-1A03: FCHF = 0.09 (1 Mw/M^2)
LII-1A03A: FCHF = 0.02 (300 Kw/m^2)

Figure 4.9-7

Concrete Attack Depth in Pedestal for Severe Accidents With Water Available to Cool Debris on the Drywell Floor

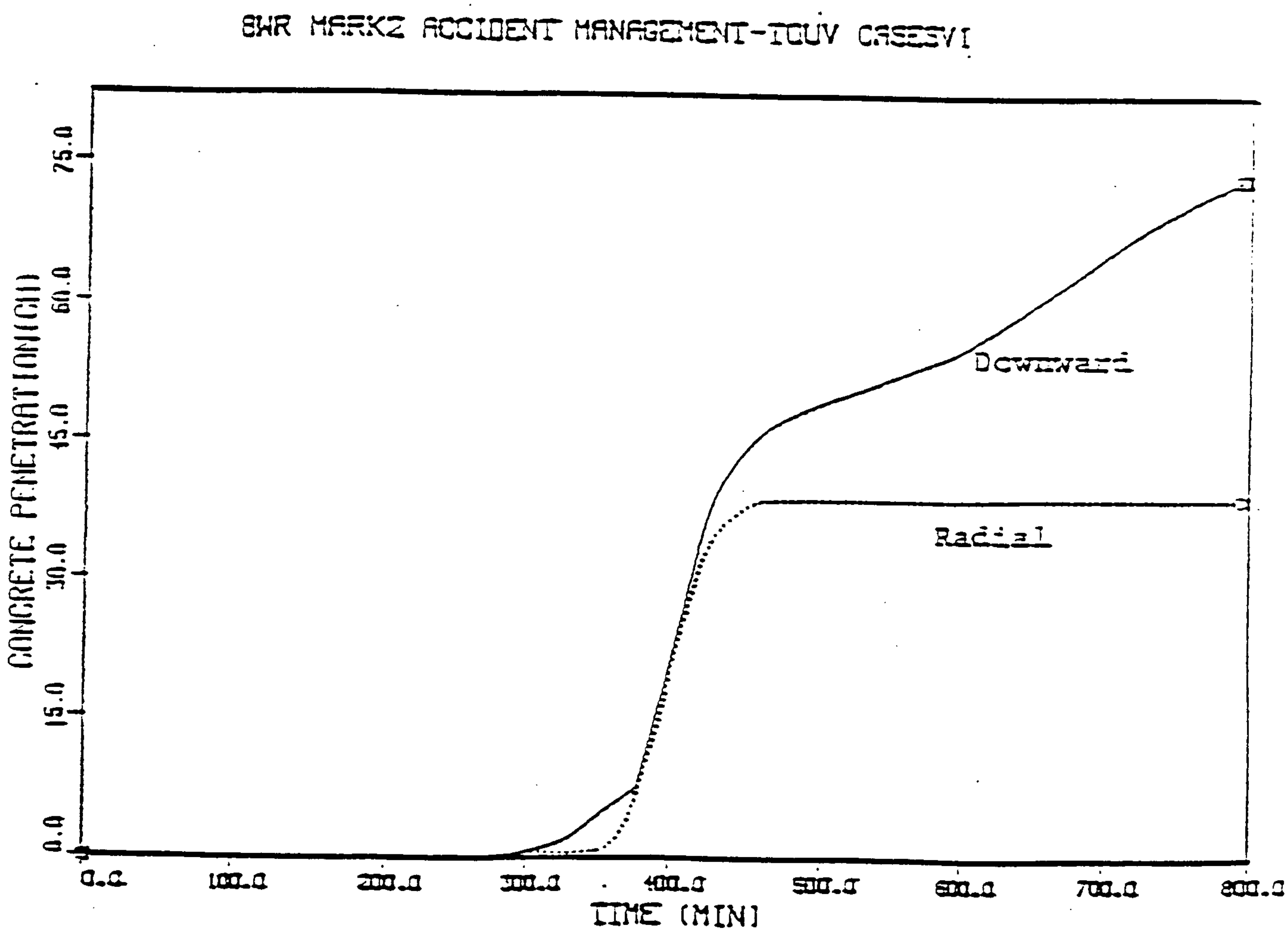


Figure 4.9-8

Drywell Floor Concrete Ablation - Case SVI

Estimated conditional pedestal failure probabilities are as follows:

a) Approximately 1E-1 for the case of no debris cooling. For cases where no water injection is available to quench debris in the pedestal (case LII-1D-07), pedestal concrete attack depth can be up to 3 ft. greater than cases with water injection (LII-1D-06). However, this should not be particularly relevant because of the other failure modes that can also lead to drywell failures with very high probabilities when no drywell water injection is available.

b) Approximately 1E-3 for the case of debris cooling available. This estimate is based upon: (1) MAAP calculations for DAEC that indicate little concrete attack if debris cooling is established; (2)

CORCON estimates of only 1.25 feet radial erosion where the pedestal is approximately 4 feet thick and over 3 feet from the edge of the sump walls.

4.9.3.2.4 Effective Area of Drywell Floor

MAAP assumes that as debris moves out of the pedestal it spreads uniformly across the entire drywell floor. This may not be a true representation due to the phenomenon of refreezing. To investigate the sensitivity to this assumption, a case was run with the drywell floor area reduced by a factor of four, i.e., the sensitivity case, LII-1A-02AW, represents a sequence with debris uniformly spread across one quadrant of the drywell. A comparison of the base case and sensitivity case is shown in Table 4.9-7.

Table 4.9-7

EXAMINATION OF EFFECTIVE DRYWELL FLOOR AREA IN CONTACT WITH DEBRIS

DEBRIS SPREAD COMPARISON

MAAP Case	Accident Class	Description	DW Concrete Attack at 36 hr.	Containment Failure		CsI Release	
				Time (hr)	Location	in RB	to Env.
LII-1A-02	IA	Base Case	.017 ft.	3.36	WWV	0.0	5.9%
LII-1A-02AW	IA	of DW floor	.051 ft.	3.38	WWV	0.0	3.7%

Due to confinement of the debris, the sensitivity case (i.e., of drywell floor covered with debris) displayed a greater depth of concrete attack than the base case, which has debris spread across the entire floor. However, the releases to the environment are essentially the same for the case with reduced available area of the drywell floor compared with the base case.

4.9.3.3 Aerosol Plugging

MAAP can assume that subsequent to a containment failure involving narrow cracks in the containment, aerosol particles will accumulate and eventually plug narrow crevice openings. As the pressure in the containment increases, the "aerosol plug" will be blown out.

The DAEC IPE conservatively does not model very narrow "crack" type leakages. Therefore, sensitivity analysis for aerosol plugging was not investigated.

4.9.3.4 Core Blockage

Uncovery of the core occurs in each of the core damage sequences investigated. As the core becomes uncovered, the clad begins to oxidize producing hydrogen as a byproduct. Eventually, melting and relocation of the core material ensues with the potential for blocking steam flow and reducing additional clad oxidation. Three blockage options are available in MAAP for treating the resulting effects from melting and relocation of core material:

1. No blockage: melting and relocation of the core will have negligible impact on the hydrogen generation, gas flows, and fission product release rates. (FCRBLK = -1)

2. Local blockage: melting and relocation of cladding away from the melting region will terminate oxidation of the Zircaloy in that node. Relocation will have a negligible impact on the gas flows or fission product release. (FCRBLK = 0.0)

3. Channel blockage: Relocation of core material will seal off and pressurize the fuel channel. The increased pressure would force the remaining water in the channel out and terminate the flow of gasses up through the affected region of the core. Without steam, oxidation of the cladding would stop. (FCRBLK = 1.0)

Considerable uncertainty and controversy has historically been associated with trying to decide which of these pictures is the most realistic.

While the actual amount of hydrogen generation may not always be of primary importance in inerted BWR containments, the increased core exit temperatures that typically occur with the no-blockage and local-blockage options will tend to result in early RPV fission products being swept into the suppression pool. If very large amounts of hydrogen are produced, then containment failure could occur even in inerted BWR containments due to the high partial pressure of the hydrogen.

The local blockage option (FCRBLK = 0.0) was selected for all of the DAEC base cases. Case LII-1A-02 was rerun with the complete channel blockage model (FCRBLK = 1.0), Case LII-1A-02B and with no blockage (FCRBLK = -1), Case LII-1A-02C, specified.

Table 4.9-8 shows a comparison of release magnitude and timing for the base case and two sensitivity runs.

Table 4.9-8

IN-VESSEL HYDROGEN GENERATION

MAAP Case	Accident Class	Description	In-vessel H_2 (lb)	Containment Failure		CsI Release[2]		
				Time (hr)	Location	in WW[1]	to Env	in RB
LII-1A-02	IA	Local Blockage (Base Case)	613	3.36	WWV	73%	5.9%	0.0%
LII-1A-02BW	IA	Complete Blockage (Channel Blockage)	184	12.4	WWV	61%	0.21 %	0.0%
LII-1A-02CW	IA	No Blockage	1490	2.25	WWV	75%	36.6 %	0.0%

[1] CsI in wetwell at vessel failure.
[2] Cumulative at 24 hours

As mentioned earlier, the case (1A-02BW) with complete core blockage (old IDCOR model) shows the smallest fraction of CsI transported to the suppression pool prior to vessel breach. The amount of hydrogen generated in-vessel also shows expected trends with the smallest amount being generated for the old IDCOR model and largest amount being associated with the "no blockage" case.

For both the base case with local blockage and the "no-blockage" case, the containment pressure at vessel failure exceeded the wetwell venting setpoint and venting was initiated. The reduced amount of hydrogen predicted in the complete blockage case resulted in a pressure after vessel failure just below the venting threshold. Therefore, the complete blockage case resulted in later venting of the wetwell. The CsI release for this case reflects the added time for fission product deposition in containment.

The earliest vessel failure occurs in the case of no blockage. Predictably, this case results in the greatest amount of H_2 generation.

The core blockage model and the subsequent core melt progression modeling can also affect the transition to the next state, i.e., the mode of RPV breach.

Without recovery of ECCS, the core will continue to melt and eventually relocate into the lower head. In the MAAP model, this relocation involves a relatively large mass of molten material. Considering that all BWRs have lower head penetrations, it is likely in this model that rapid heat-up and failure of a penetration will occur. The BWR MAAP model calculates the heat-up and failure of the lower head penetrations. However, the rate and thermodynamic state of the material entering the lower head is uncertain. Other scenarios have been postulated in which core debris remains coolable within the lower head until all of the remaining RPV water is boiled away. The debris then heats up eventually failing the lower head. These scenarios are rather controversial; in addition, due to modeling constraints in MAAP, they cannot be easily simulated.

Depending on the scenario, blockage could either increase or decrease the severity of a release. The selection of the blockage model used in the base cases was not dictated by which was the most "conservative," but rather which was considered more realistic. Therefore, FCRBLK = 0.0 (local blockage) was chosen.

4.9.3.5 Containment Failure Mode (Size and Location)

In all severe accident scenarios in which containment heat removal is not effective, some of the key uncertainties are the time, size, and location of the containment failure. A considerable amount of work has been performed to evaluate the expected failure modes of Mark I containments. The result of this work indicates that failure depends strongly on containment temperature. The same conclusion can be drawn for other containment designs.

Several cases were run to investigate the sensitivities to the assumed containment failure mode. This has been broken down into three different categories for the discussions which follow. They are: (1) the assumed containment failure area; (2) the assumed containment failure location (i.e., upper drywell region, lower drywell region, or wetwell airspace); and (3) the potential for drywell shell failure at vessel failure.

4.9.3.5.1 Containment Failure Area

Sensitivity cases were run to investigate the impact that the containment failure size has on the retention of fission products within the reactor building. The LII-1A-3 case was selected for this investigation. The base case involved successful operation of injection after vessel failure. The drywell head failed late due to overpressure with a relatively small break area. Table 4.9-9 provides a brief summary of the release for this case and for a sensitivity run performed using an increased failure size.

This table indicates that for a larger failure area, the CsI release is smaller. This is due to a slightly smaller CsI mass in the RPV for the base case at 40 hours. The large failure area results in almost immediate containment depressurization, while the smaller failure area provides for a long term purging of fission products over many hours (i.e., 10 hours for a 40 psi drop in pressure for severe accidents with a saturated pool). Figures 4.9-9 and 4.9-10 provide a comparison of drywell pressure profiles and CsI release to the environment for these two cases.

The most important effect of containment failure size was determined for a postulated ATWS scenario. For this scenario, the size effect is just the opposite of that observed above. For the ATWS case, the small drywell head failure size (Case LII-4A-01) leads to a medium release, while the large drywell head failure size (Case LII-4A-01LD) leads to a high release.

Conclusion

Based on these results, no clear generalization can be made regarding the impact of size on radionuclide release which is applicable to all sequences. The above insights have been factored into the following more focused conclusions:

- The sensitivity evaluations to examine containment failure area effects have the following insights for Class ID sequences:

  - For drywell head failures, in general, smaller failure area can lead to larger releases

Table 4.9-9

DRYWELL HEAD FAILURE AREA

MAAP Case	Accident Class	Description	Containment Failure		CsI Release		
			Time (hr)	Location	RB DF	in RB	to Env.
LII-1A-3*	IA	Small failure (0.18 ft.^{2})	3.38	DWH	1.2	1.0%	5.2%
LII-1A-3LD*	IA	Large failure (2.0 ft.^{2})	3.38	DWH	1.3	0.8%	2.2%

* Class IA loss of injection inventory accidents, but water injection available to debris ex-vessel. Drywell head failure occurs as a result of high pressure accompanied by elevated temperatures.

- For failures in lower drywell or in wetwell airspace, other phenomena may impact the release for example:

  -- Containment pressure at the time of failure will also influence the "turn-over" or "through-put" time in the reactor building.

- The containment failure mode (i.e., high temperature induced failure of the drywell head, shell failure, torus failure) is found to be much more important than the size of the containment failure. The failure mode is, therefore, used as the determining factor in the source term magnitude assessment. As such large and small failures have been found to be treated either:

  - Conservatively to include the effects of both
  - Realistically because the difference between large or small failures is minimal.

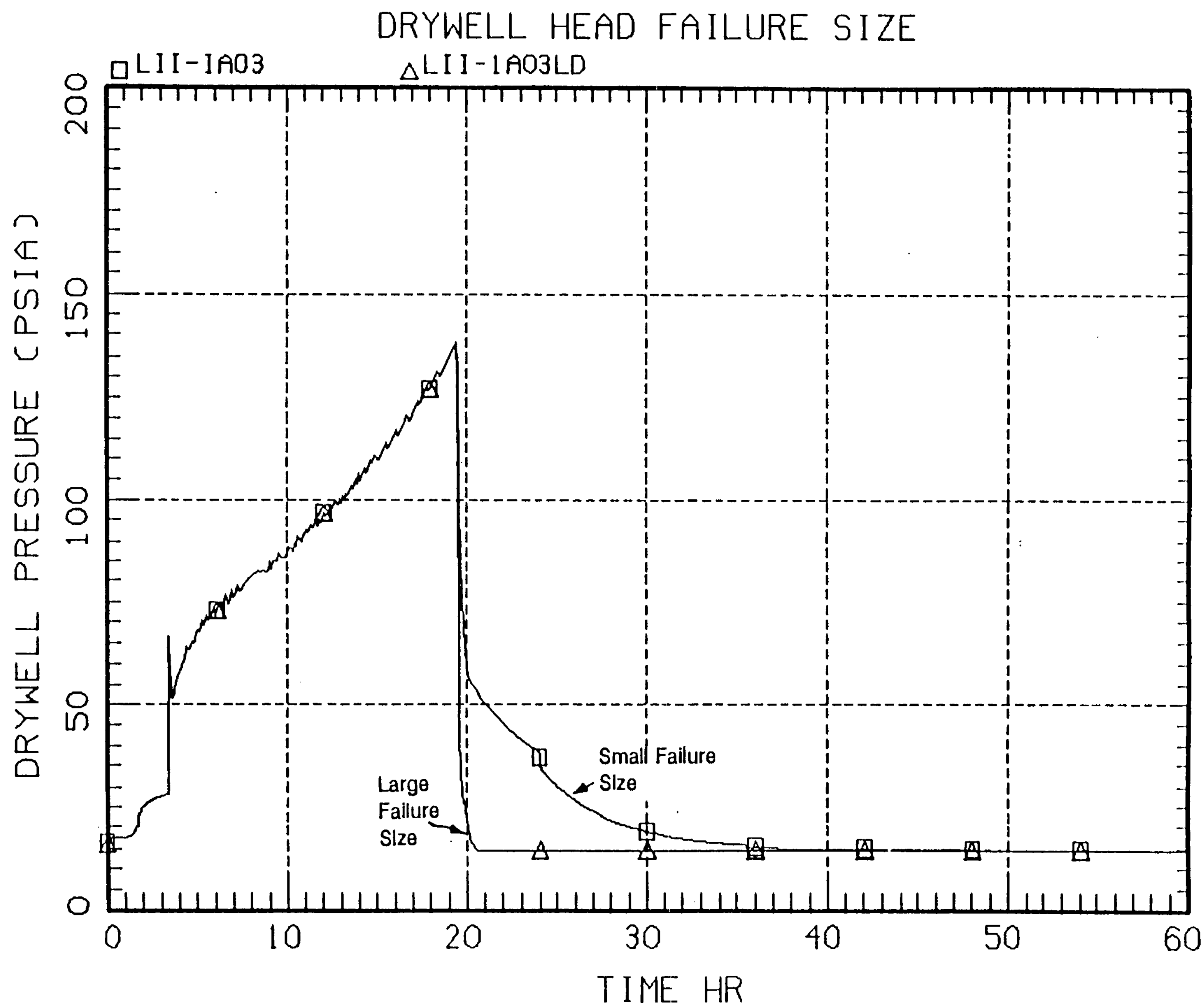


Figure 4.9-9

Drywell Pressure for Sensitivity to Containment Failure Area

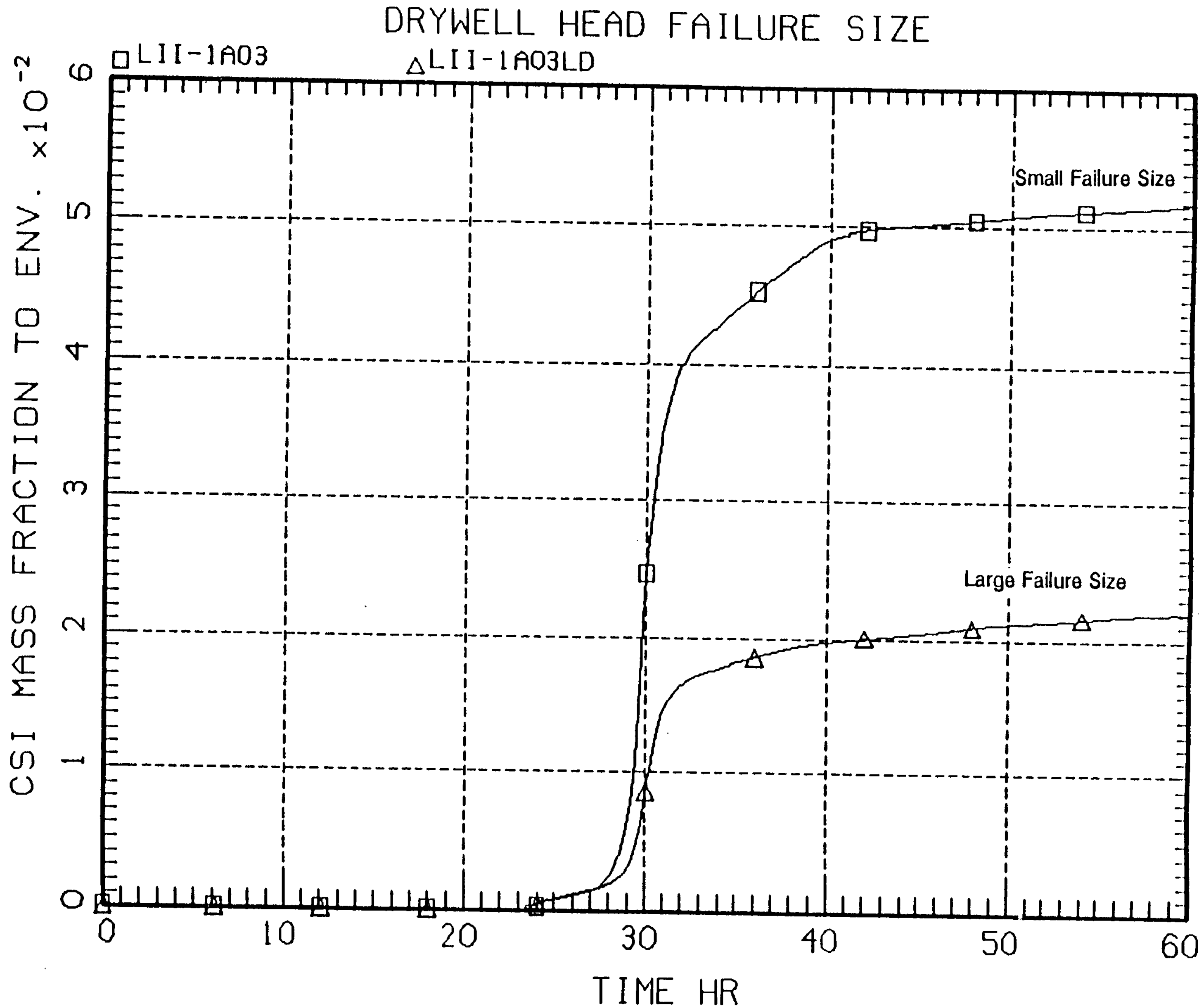


Figure 4.9-10

CsI Release for Sensitivity to Containment Failure Area

4.9.3.5.2 Containment Failure Location

As noted above, the magnitude of a radionuclide release from a severe accident is strongly influenced by the location of the containment failure. This dependence is compounded by the effects of the reactor building. For DAEC, a failure of the upper drywell head would lead to a release into the refuel floor of the reactor building. The blow-out panels on the 5th floor would quickly relieve; thus resulting in a direct release to the environment. A release to a lower elevation in the reactor building would have a more tortuous route to the environment resulting in additional opportunity for deposition mechanisms to reduce the mass of radionuclide as the effluent is transported through the reactor building.

Of course, for sequences that involve successful venting through the wetwell, and without any pool bypass or later drywell failure, the fission products would be scrubbed through the suppression pool before release to the reactor building.

Table 4.9-10 shows the difference between the various containment failure locations. The conclusions of the sensitivity evaluation to containment failure location are the following:

- Drywell head failure leads to low decontamination factors in reactor building and the potential for large releases.

- Lower drywell failures (i.e., shell failures at the drywell floor), generally lead to a factor of 2 or 3 reduction in release source term magnitude to the environment due to better reactor building retention.

- Wetwell airspace failures for similar cases generally lead to an order of magnitude reduction in release source term magnitude to the environment due to pool scrubbing and some reactor building retention.

- The failure of containment very early (e.g., shell failure) results in the possibility of releasing substantial amounts of airborne radionuclides.

- Delayed failure of containment (i.e., ~ 20 hours), results in increased residence time and the possibility of reduced airborne material at containment failure.

Table 4.9-10
CONTAINMENT FAILURE LOCATION

Case	Class	Location	Size ($ft.^2$)	Time of Cont. Failure (hr)	Cont. Pressure at Failure (psig)	Rx Bldg DF	CsI in RB	CsI to Envir.
LII-1A-3	IA	Drywell head into refuel floor	0.18	19.5	120	1.20	1.0%	5.2%
LII-1A-3LWIA	IA	Wetwell failure into torus room	2.0	19.5	121	28.3	0.08%	0.003 %
LII-1D-LFA	IA	Shell failure low in drywell into torus room	2.0	3.5	14	5.8	33.8%	7.2%

- When the containment failure is a drywell shell failure occurring shortly after RPV breach, the release to the reactor building can be quite large, and there is a strong dependence on the reactor building DF to limit the radionuclide release.

4.9.3.5.3 Drywell Shell Failure

The Mark I containment shell has been identified in past analyses as a possible containment failure location if high temperature molten core debris comes in contact with the shell.

The drywell shell postulated failure mode involves molten debris contacting the drywell shell and failing the drywell within minutes of a vessel failure. This phenomena has three primary areas of uncertainty that factor into the assessment of release source term:

1. The likelihood that core debris can reach the shell and melt-through the shell,

2. The timing at which the melt-through occurs, and

3. The release pathway and the resulting release magnitude.

With respect to the first issue, industry and NRC models suggest that without water available to cool the debris the steel shell will most assuredly melt shortly after being contacted by substantial molten debris. In contrast, earlier IDCOR theories postulated that the steel may act as a sufficient heat sink so as to prevent complete melt-through of the shell.

Even though the steel shell melts, the core debris can be considered likely to "refreeze," and thus, effectively block the newly created breach. This consideration has not been factored explicitly into the MAAP modeling of the DAEC shell failure assessment.

Each of these issues casts some uncertainty as to whether the core debris will create and maintain a large breach in the steel shell. Possible effects that may preclude the shell failure are the following:

- Water on the drywell floor before core debris is ejected. This must also be coupled with water injection in the long term.

- High pressure blowdown, which causes the dispersal of debris in non-coherent manner to locations where the debris, may be coolable.

- Sump retention of debris equivalent to approximately 30% of the original core inventory can preclude direct interaction of this initial release with the drywell shell.

- Discharge of only a portion of the core debris before water is injected, i.e., in a Class IA core melt progression. This is similar to the BWRSAR code (NUREG/CR-5565) predictions that indicate at vessel failure, only a small portion of the debris may be available for discharge to the containment.

  This initial debris is also likely to be metallic and have lower superheat than the oxide core debris that is discharged later in the sequence.

- Water in the RPV at vessel breach can be immediately available to quench the debris.

- Water injected into the RPV after its initial blowdown can act to quench the fraction of debris (e.g., metallics), that may have been initially ejected from the RPV.

Consistent with current models, the deterministic modeling of these accidents in the DAEC IPE assumes that the drywell shell is breached when the core debris is on the drywell floor and water is not present. The breach is assigned a large size (a few square feet), and is assumed to remain unobstructed for the duration of the scenario.

Drywell shell failure due to debris attack is a complex issue and a number of sensitivities have been performed to attempt to reach a consensus on the effects shell failure may have on the accident sequence. The sensitivities that are discussed in this subsection are the following:

- Reactor building release paths
- Time delay in shell failure
- Shell failure size
- Other cases involving shell failure
    - With water available
    - Drywell equipment mass
    - Torus cooling

Reactor Building Release Paths

The mitigation of a drywell shell melt-through by fission product deposition in the secondary containment can vary because of the potential exit pathways. At DAEC, three paths can be postulated:

1. Release downward to the torus room around the downcomer vent pipes.

2. Release upward through the gap to the refuel floor

3. Release upward through the gap between the shell and concrete wall and out drain lines to tanks outside the reactor building.

The downward path to the torus room results in a mitigated release because the release is to an area of the reactor building subject to deposition as the release passes through a tortuous path in the reactor building to the blowout panels located on the refuel floor.

The two upward pathways could potentially result in higher source term if the release could reach the refuel floor. A look at the basic construction of the DAEC pathways provides the answer regarding the viability of these hypothesized pathways.

The DAEC 2" gap (see Figure 4.9-11) was completely filled with compressible foam elastic sheets. The concrete of the shield wall was poured against the "insulation" Following construction of the biological shield, most of the foam was removed.

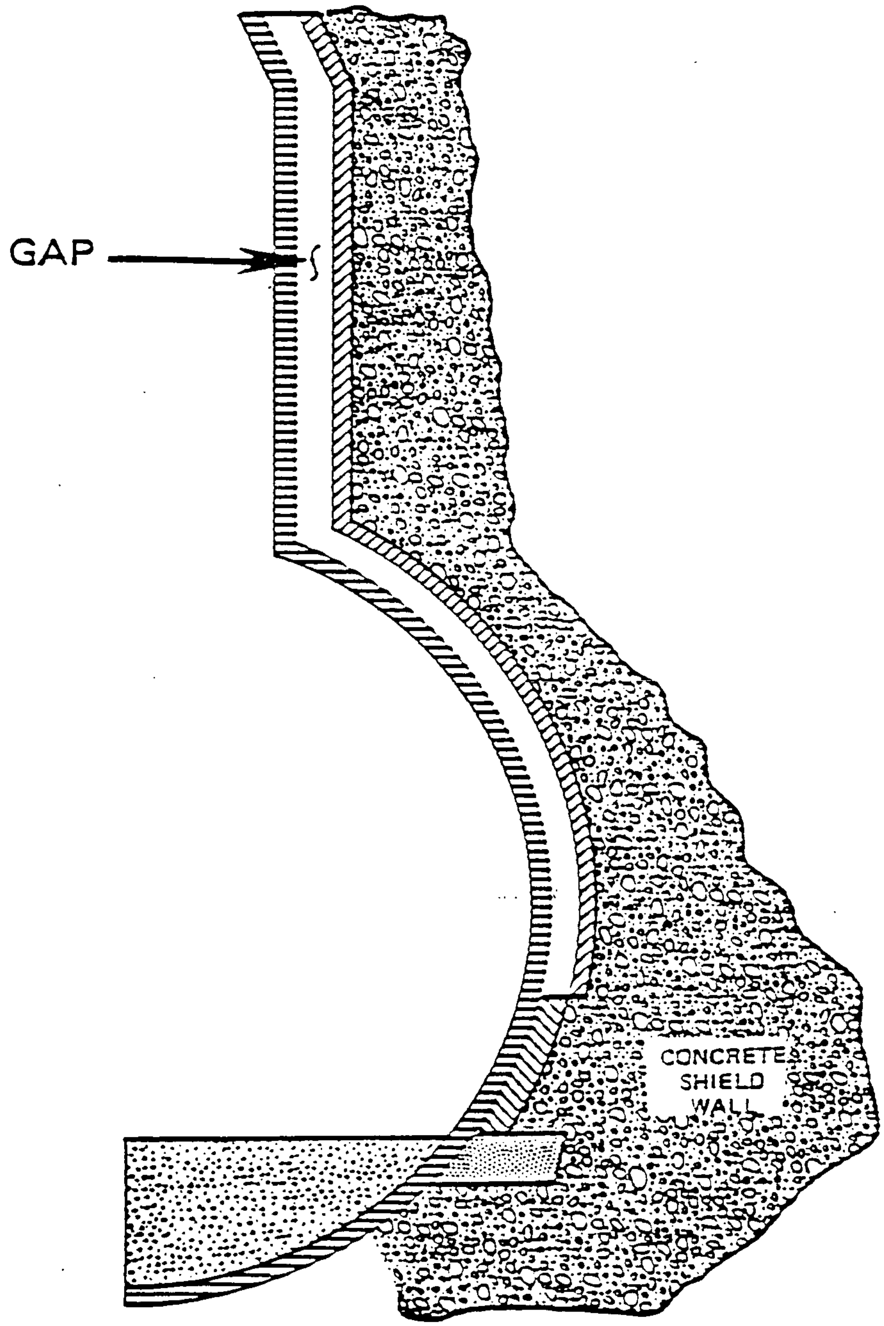


Figure 4.9-11

Typical Drywell Shell and Concrete Shield Wall Gap Construction
Showing the Approximately 2" "GAP"

There is a bellows separating the spent fuel pool area from the top of the DAEC drywell. This was installed for the practical reason of preventing water used during refueling operations leaking into the "gap." It also fortuitously prohibits any release from traveling up the gap and out the refuel floor as was identified in WASH-1400. In addition, it should be noted that if gases were to find their way to the top of the drywell gap, there are no release paths (e.g., accessible drain lines from the gap to outside containment). The only drain lines open to the 2" gap near the drywell bellows area are four 8" lines, but these pipes are capped.

The steel bellows near the drywell head prevents any release directly to the refuel floor from the 2" gap (see Figure 4.9-12). Because there are no accessible drain lines near the drywell bellows area, it is judged that any release into the 2" gap cannot bypass the secondary containment mitigation. Therefore, the deterministic modeling in the DAEC IPE assigns the release to the torus room.

The result of these investigations at DAEC is that the shell failure introduces fission products to the lower region of the reactor building, where significant deposition will occur before being released into the environment. Releases due to the shell melt-through phenomenon usually occur early release with a low or moderate severity source term.

The following table provides a compilation of drywell shell failure cases and the CsI released to the reactor building and the environment along with the reactor building DF.

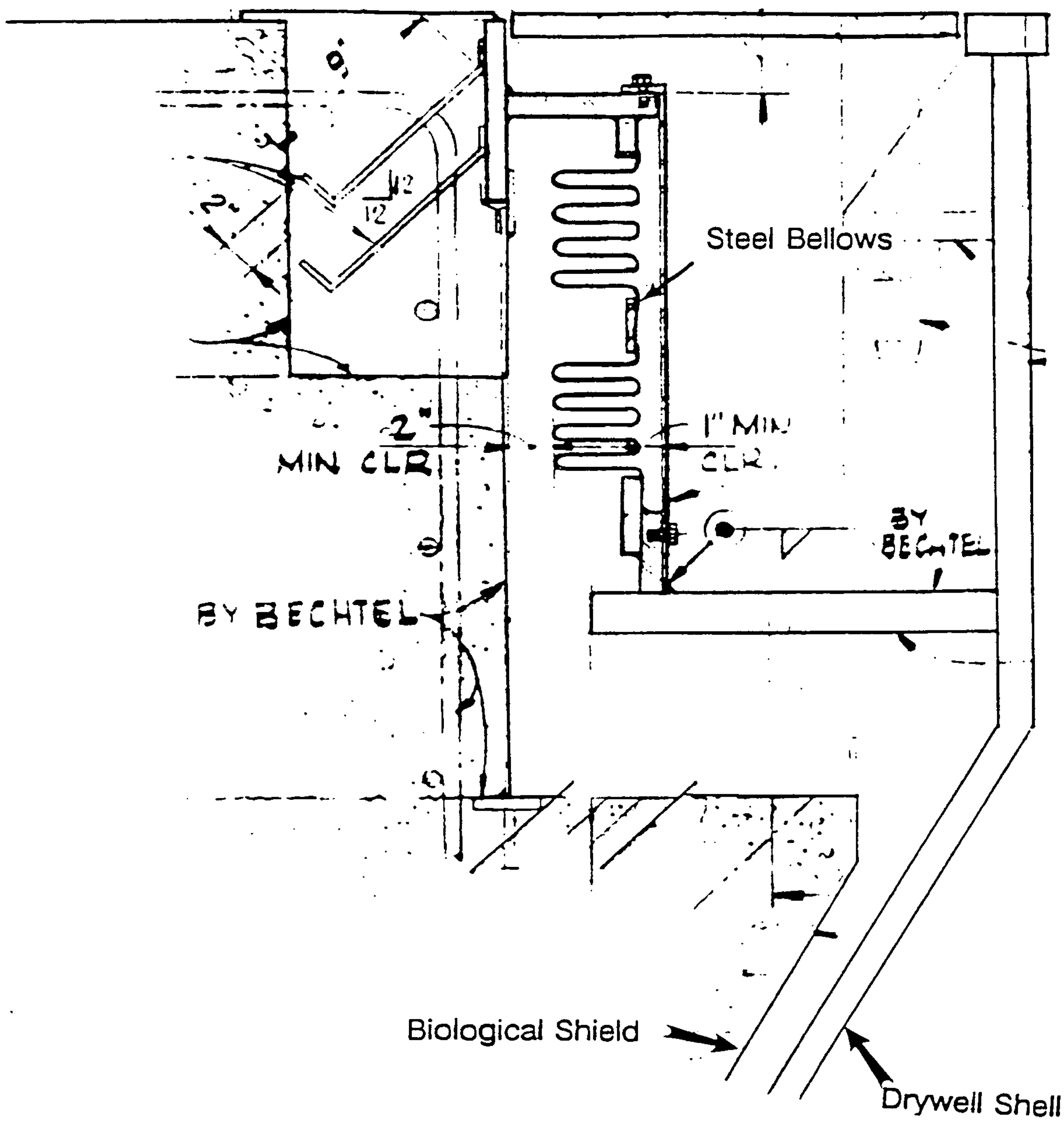


Figure 4.9-12

Drywell Head Bellows Structure

Table 4.9-11

Drywell Shell Failure Cases and CsI Released

MAAP Case	Class	Water to Debris	Containment Failure		CsI Release		DF
			Time	Location	R.B.	Envir.	
LII-1D-LF	ID	NO	2.04	Shell (large)	37.7%	4.6%	9.2
LII-1D-LFA	1A	NO	3.5	Shell (large)	34.3%	7.2%	5.8
LII-1D-LFB	ID	NO	2.04	Shell (large)	34.8%	5.0%	8.0
LII-1D-11	ID	YES	2.03	Shell (large)	0.6%	0.02%	30
LII-1D-12	ID	YES	2.03	Shell (small)	0.2%	0.002%	200100
LII-1D-13	ID	NO	2.03	Shell (large)	37.7%	4.6%	9.2
LII-1D-14	ID	NO	2.03	Shell (small)	21.9%	1.6%	15
LII-1D-LFF	IA	NO	3.34	Shell (large)	32.5%	3.4%	11
LII-1D-LFE	IA	NO	3.38	Shell (large)	34.3%	7.2%	5.8
LII-1D-LFD	1A	NO	3.38	Shell (large)	33.2%	9.4%	4.5
LII-1D-LFC	IA	NO	3.38	Shell (large)	33.8%	4.7%	8.2
LII-1D-13A	ID	NO	2.04	Shell (very large)	38.6%	2.7%	15
LII-1D-LFG	IA	NO	3.47	Shell (large)	36.5%	6.1%	7.0

The failure of the drywell shell due to direct debris attack can have a substantial influence on the radionuclide release. The observations that can be derived based on these cases are described below:

- With water available to the debris, despite the drywell shell failure, the CsI release to the environment can be minimized by the "scrubbing" effect of the water.

- The RPV pressure and RPV failure timing appear to have minimal effect on the magnitude of the source term. However, the release timing can be dramatically affected, by almost 2 hours, at a critical time in the accident progression, i.e., RPV depressurization accelerates the time to containment failure by almost 2 hours.

- Containment shell failure results in high releases of CsI to the reactor building; and therefore, high dependencies on reactor building effectiveness to demonstrate low releases.

- The early failure at the shell region rather than at other locations if there were no shell failure has the effect of reducing the net radionuclide release to the environment.

Shell Failure Time

The time at which a drywell shell failure may be induced by debris attack is another difficult parameter to bracket. Work performed by NRC contractors forms the basis of the estimates used here.

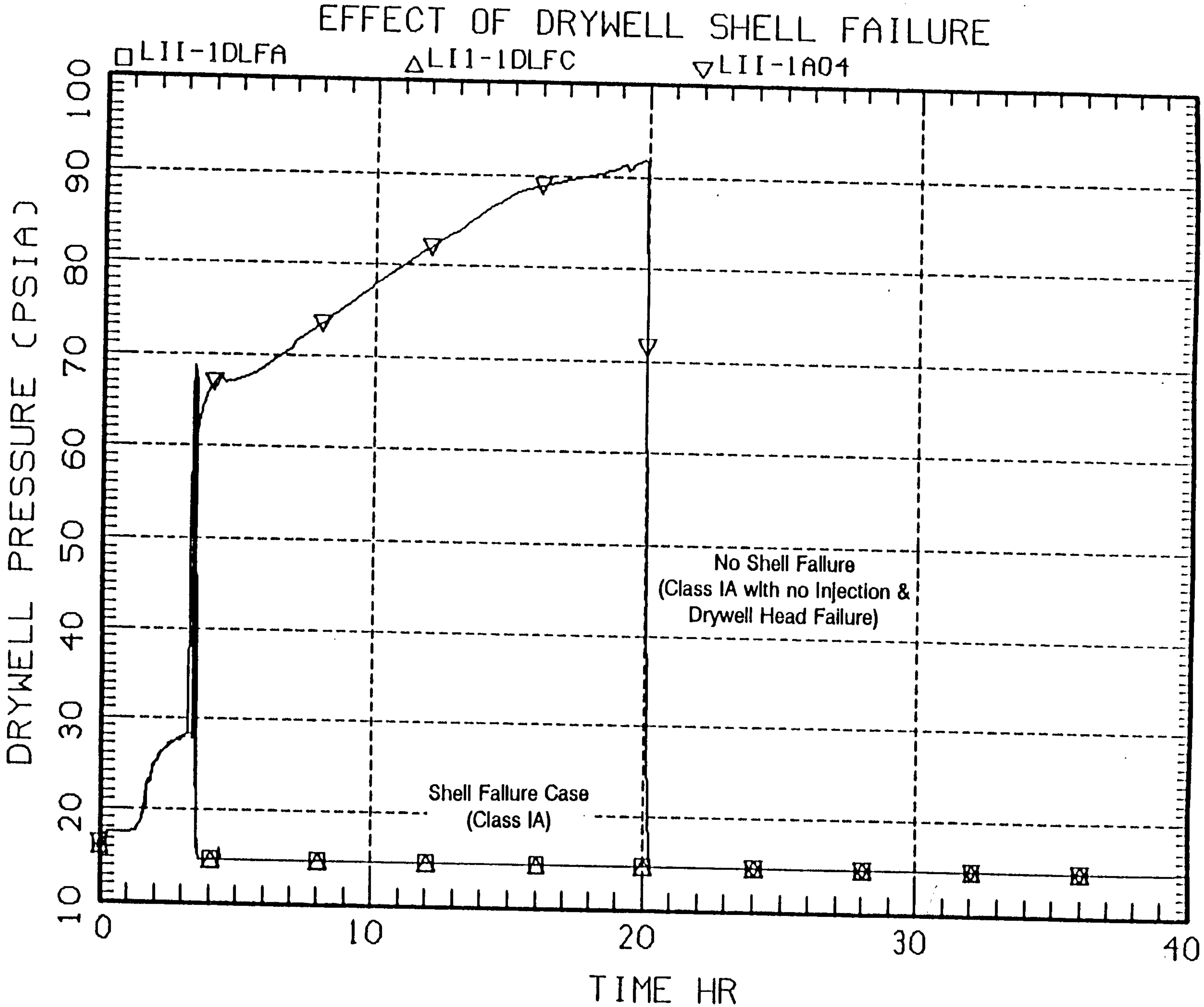


Figure 4.9-13

Drywell Pressure for Sensitivity to Shell Failure

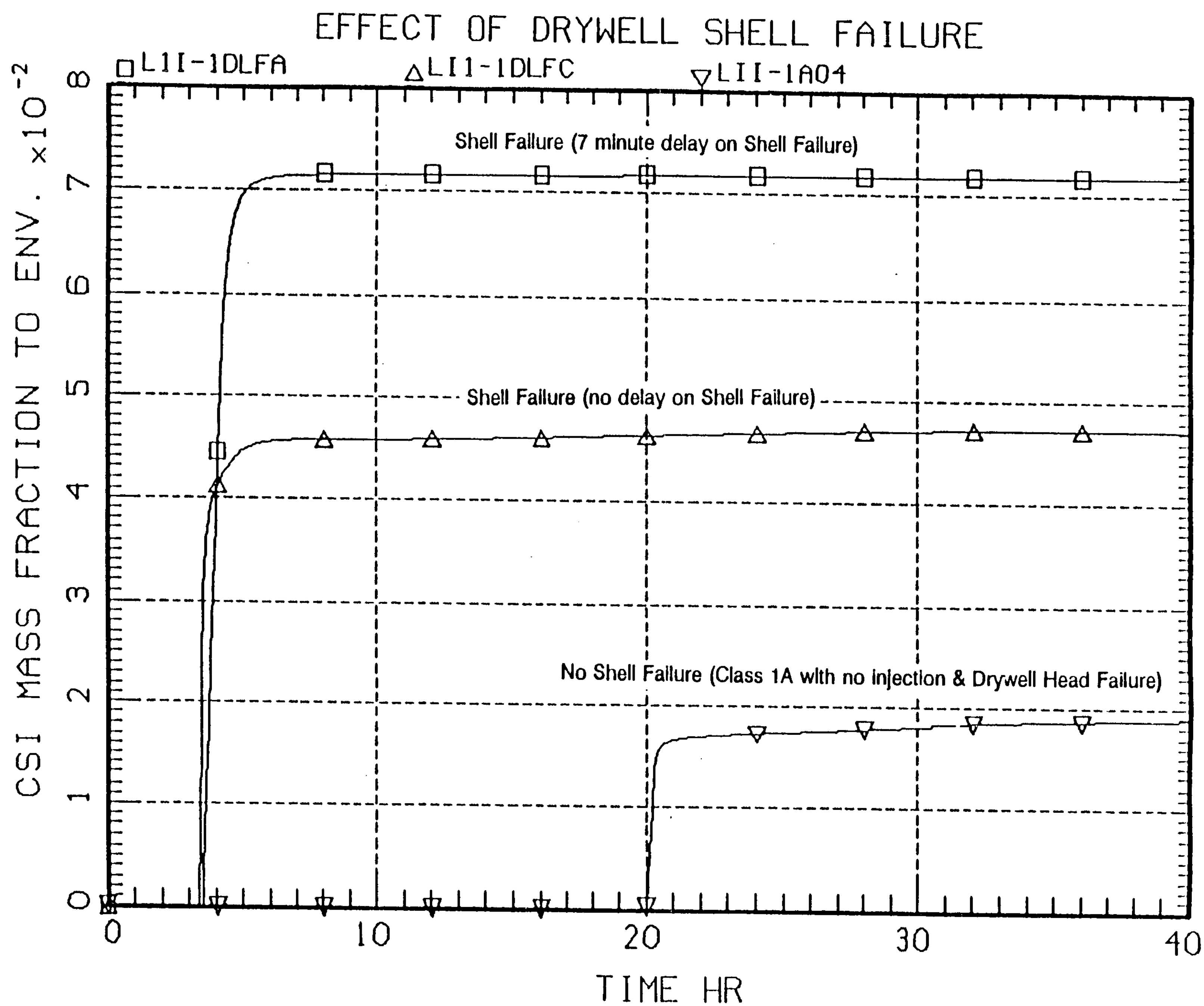


Figure 4.9-14

CsI Release for Sensitivity to Shell Failure

Figures 4.9-13 and 4.9-14 show the drywell pressure and CsI release to the environment for three cases:

- LII-1A-04 - No shell failure, but a late drywell head failure
- LII-1D-LFA - Drywell shell failure following a 7 minutes time delay
- LII-1D-LFC - Drywell shell failure following no time delay.

For both shell failure cases, drywell pressure increases sharply after vessel failure (approximately 3.5 hours), because of the RPV blowdown at RPV breach associated with Class IA. Containment blowdown due to shell failure occurs immediately for the no delay shell failure case and after seven minutes for delayed shell failure. The 7 minutes used in the MAAP calculation is an estimate for melt-through of a 0.5 inch thick plate.[1] By comparison, the late drywell head failure case does not reach a failure pressure until 17 hours after vessel failure because the shell was not assumed to fail. Each of the cases are summarized in Table 4.9-12.

Table 4.9-12 shows that a delay in the shell failure time results in a slightly higher fission product release. Alternatively, it appears that if the shell failure is not delayed, a larger fraction of the CsI is deposited in the drywell long term resulting in a smaller release from containment.

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[1] The time of 7 minutes is supported by the analyses of NRC contractors G.A. Green, R.d. Bergeron, and T.S. Kress regarding calculated shell failure times.

Table 4.9-12 DRYWELL SHELL FAILURE TIMING							
MAAP Case	Accident Class	Description	Containment Failure		CsI Release		
			Time (hr)	Location	in Wetwell	in RB	to Env.
LII-1A-04	1A	Late DW head failure	2.0	DWH	36%	0.4%	4.5%*
LII-1D-LFA	IA	Shell Failure from after RPV Failure	3.5	Shell (large)	24%	34.3%	7.2%
LII-1D-LFC	IA	Shell failure immediate after RPV failure	3.38	Shell (large)	23%	33.8%	4.7%

* These values were modified by the MAAP analysts to account for the MAAP error that sets pool DF = 1.0 for subcooling of less than 1°F.

Table 4.9-13 DRYWELL SHELL FAILURE AREA						
MAAP Case	Accident Class	Description	Containment Failure		CsI Release	
			Time (hr)	Location	in RB	to Env.
LII-1D-14	ID	0.18 ft^2 shell failure 7 min. after RPV failure	2.03	Shell (small)	21.9%	1.6%
LII-1D-13 (1)	ID	2.0 $ft.^2$ shell failure (Base) 7 min. after RPV failure	2.03	Shell (large)	37.7%	4.6%
LII-1D-13A (2)	ID	10.0 $ft.^2$ shell failure 7 min. after RPV failure	2.04	Shell (very large)	38.6%	2.7%

(1) Prolonged blowdown of containment

(2) Rapid blowdown

Shell Failure Size

Two sensitivity cases were run to investigate the effect of the assumed shell failure area to CsI release (see Table 4.9-13). The base case (LII-1D-13) assumed a large failure size of 2 $ft.^2$ while one sensitivity case (LII-1D-13A) assumed a small failure size of 0.18 $ft.^2$. The base case resulted in a release of 37.7% in the reactor building and 4.6% to the environment. The small shell failure area case reduced the release in the reactor building by nearly a factor of two (21.9%) and the release to the environment by nearly a factor of three (1.6%). The second sensitivity case assumed a very large failure size of 10 $ft.^2$. The CsI release in the Reactor Building and to the environment for this case was 38.6% and 2.7%, respectively.

- For cases with the shell intact (i.e., SI = S) and water available to the debris, the containment pressurization due to steam and non-condensibles plus the temperature increase can lead to containment failure if these pressurization effects are not mitigated (see MAAP case LII-1A-3).

- Water on the drywell floor prior to RPV breach. There are accident sequences such as water line break LOCAs, RPV bottom head failures, or "early" drywell spray initiation for which substantial amounts of water can be on the drywell floor prior to RPV breach. When the debris exits the RPV, it will be quenched by the water on the floor and it takes substantial time for the debris to reheat event with no RPV or containment injection available. For example, consider the LII-3A-01A case. In this case, the RPV has failed initially, causing a release of substantial amounts of water to the drywell. The water accumulated on the drywell floor acts to quench the debris as it is discharged from the RPV. The containment

Table 4.9-14

Accident Progression for MAAP Case LII-3A-01A

Time (Hr.)	Event	Debris Temperature[1]	Gas Temperature
0.99	RPV Breach	Low	Low
10	Ex-vessel Core Melt Progression	1200°F	621°F
18.4	Containment Failure Curve Exceeded[2]	1270°F	797°F[2]
20	Beyond Containment Failure	1320°F	> 800°F
30	Beyond Containment Failure	1420°F	> 800°F

[1] In order for drywell shell failure, temperatures of 1700°K (2600°F) may be required based on BNL (Greene) experiments. As seen, debris temperatures remain low and could not fail the shell before drywell head failure would be predicted based solely on high temperature of the drywell head seals.

[2] Containment failure due to combination of elevated temperature and pressure.

conditions that are predicted to occur during this postulated accident sequence are listed in Table 4.9-14.

Drywell Equipment Mass

The drywell equipment mass parameter is an influential parameter in the assessment of preventing late drywell head failure on high temperature. This is a difficult parameter to estimate, but it can have a strong influence on the outcome. The DAEC CET analysis assumes a nominal drywell equipment mass (1 Mlbm), i.e., the mass may be underestimated. (Note many of the MAAP runs were performed with a higher drywell equipment mass (2.7 Mlbm); and therefore, these MAAP cases show no tendency to fail the drywell on high temperatures.) (Also see Section 4.9.3.11.)

Conclusion for Shell Failure Sensitivity

The conclusion for the drywell shell failure induced by molten debris is that it has the potential to be a beneficial effect. The benefit is in creating a radionuclide release pathway through the reactor building with increased DF potential. Of course, the sensitivity case for drywell shell failure due to debris attack have shown that the shell failure can cause a relatively early radionuclide release. This reduces the amount of warning time for an organization to implement emergency response actions. If an alternative pathway through the drywell head should occur, it would have substantially greater release potential to the environment.

From the DAEC IPE results, it is found that the drywell shell melt-through is a potential early release pathway, potentially yielding a large source term. Because of this, the effectiveness of the reactor building in reducing the magnitude of the radionuclide release takes on increased importance. The containment capability for BWRs can be properly characterized only if a best estimate of both the primary containment and the secondary containment effectiveness are included.

4.9.3.6 Reactor Building Modeling Assumptions

Secondary containment configurations differ greatly among BWR plants. Generally, however, for the secondary containment to retain a significant quantity of fission products, one of two conditions must occur.

First, in many cases "active" decontamination measures may be available in the release pathway. This would include scrubbing due to the passage of fission products through deep water pools, decontamination by ventilation system filters, or scrubbing due to wide-coverage fire sprays. If such measures are functional, they will generally overwhelm the deposition mechanisms, and result in relatively small environmental releases of all fission products, except for noble gasses. A few qualifications to this statement must be offered, however. First, ventilation filters are not usually designed for large aerosol loadings, although a significant quantity of aerosols could be generated during in a severe accident[1]. Consequently, filters have been postulated to tear, overheat, or clog. Second, ventilation filters may not cover all the volume of all the affected secondary containment regions. Finally, while aerosol behavior is relatively well understood, there are significant uncertainties associated with the effectiveness of scrubbing of fission product vapors in water pools; these might impact the release when the source of fission products is at a very high temperature.

Alternatively, if no such active measures are at work, we must rely on natural settling processes. For these to be effective, the fission products must have a relatively long residence time in the secondary containment before they can be swept to the environment. This, in turn, generally requires that the ventilation systems be secured, the flowrate from the primary system or containment be relatively small, and the vigorous natural circulation between the secondary containment and the environment be avoided. The last of these requirements is often the most difficult to confirm, since vigorous natural circulation between the secondary containment and the environment can be set up if one large hole is opened (leading to large counter-current flows through the one opening), or if two holes are opened, one low in the building and one higher up. This latter configuration gives rise to a "chimney-like" flow pattern. Since it is often difficult to know the precise failure pressures and failure modes of the myriad of openings in the secondary containment, and since the pressure differentials between rooms are typically quite small, it can be difficult to establish precisely the communication pathways among areas in the secondary containment. For this reason, one must evaluate carefully any

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[1] MAAP calculations have indicated that as much as 120 lbm of aerosols could be deposited in each of two operating SGTS filter trains. This is well within their assumed structural capability.

prediction of large decontamination factors due to natural settling processes in secondary containments.

To investigate the sensitivity to the assumptions made in representing the DAEC reactor building with MAAP, two variations were made to the LII-1D-LFA drywell shell failure case. Variations in the reactor building assumptions include:

1) Operation and effectiveness of reactor building sprays, and

2) Operation and effectiveness of the Standby Gas Treatment System (SGTS).

The results indicate a calculated DF in the range of 4 to 9 depending on the modeling assumptions.

Failure of the sprays with SGTS operable (LII-1D-LF) does not change the effectiveness when compared to the base case (LII-1D-LFA), also with a single train of SGTS operating. At DAEC, the only fire sprays in the reactor building are located in the HPCI and RCIC rooms. Despite a torus room failure with fire sprays inoperable, the resulting pressure increase is not sufficient to lift the HPCI/RCIC room plugs (minimum 2.8 psid required), and create an additional pathway for fission products to reach higher elevations in the reactor building. Finally, the contribution to radionuclide reduction due to SGTS operation was investigated. The results of this analysis is summarized below:

SGTS Condition	Discharge Flow Rate	CsI To Environment
No SGTS	0	9.4%
1 TRAIN of SGTS	4000 SCFM	7.2%
2 TRAINS of SGTS	8000 SCFM	4.7%

The higher SGTS flow rate results in reduced CsI to the environment because of a more effective reactor building (i.e., particulate CsI collected on the SGTS prefilter and carbon bed filters).

These results are presented in the Table 4.9-15.

4.9.3.7 Drywell Equipment Mass and Effect on Drywell Temperature Post RPV Breach

The drywell equipment mass may impact the rate of temperature rise in the drywell.

The assumed DAEC drywell equipment mass used in the MAAP calculations is 2.7 million lbm as compared to the IDCOR estimate of Peach Bottom drywell equipment mass of 4.2 million lbm. The Peach Bottom value was obtained from discussions with General Electric and is based on the assumptions that a Mark I and Mark III drywell have about the same amount of steel. The DAEC assessment of steel in the drywell for heat sinks is considered to be non-conservative, and therefore, the MAAP estimates requiring time to containment failure are considered to be relatively long compared with cases if more realistic steel heat sink masses are used.

A comparison was made of the impact of decreasing the mass of steel by approximately a factor of 3 (see Table 4.9-16).

Table 4.9-15 REACTOR BUILDING EFFECTIVENESS							
MAAP Case	Accident Class	Description	Containment Failure		CsI Release		
			Time (hr)	Location	RB DF	in RB	to Env.
LII-1D-LFA	IA	Base Case	3.5	Shell (large)	5.8	34.3%	7.2%
LII-1D-LFE	IA	Without RB sprays with SGTS[1]	3.5	Shell (large)	5.8	34.3%	7.2%
LII-1D-LFD	IA	With RB sprays without SGTS	3.38	Shell (large)	4.5	33.2%	9.4%
LII-1D-LFE2	IA	With RB sprays and 9000 SCFM (2 x SBGTS)	3.33	Shell (large)	9.0	37.5%	4.7%

(1) No change in CsI release from base case indicates that the HPCI/RCIC plugs did not lift.

Conclusion

The heat sink masses used in the DAEC MAAP evaluation could not be absolutely verified. The drywell equipment mass can have a profound impact on drywell heat up. MAAP runs with reduced equipment mass display much higher containment drywell temperatures than runs using the higher value (2.68 million lbm). At 40 hours into the accident sequences, the difference in calculated drywell temperatures between the respective sensitivity cases can be as large as 180°F.

4.9.3.8 Pool Decontamination Factor (DF)

The pool decontamination factor has been examined because MAAP BWR 3.0B Rev. 7.01, utilized for the DAEC IPE, assumes suppression pool DF equal to 1 when the

Table 4.9-16 Effect of Equipment Mass in Drywell					
Case	Drywell Equipment Mass	Release Time (Timing)	Mass Fraction of CsI Released (into RB)	Release Magnitude to Environment	Peak DW Temperatur e (Time)
LII-1D-LFA LII-1D-LFG	2.68 million lbm 1.0 million lbm	3.38 hr (E)[1] 3.68 hr (E)[1]	41.5% 42.6%	M M	676°F (40 hr) 855°F (40 hr)
LII-3A-1 LII-3A-1B	2.68 million lbm 1.0 million lbm	---[2] 35.7 hr (L)	-- 0.075%	-- L	730°F (40 hr) 870°F (40 hr)
LII-4A-1 LII-4A-1(M)	2.68 million lbm 1.0 million lbm	0.964 hr (E) 0.945 hr (E)	1.9% 3.8%	M M	609 (48 hr) 698 (48 hr)

(1) Shell breach due to debris attack

(2) Containment did not fail

water is subcooled less than 1°C, independent of other important parameters. A DF of 1.0 is not consistent with experimental results.

Parameters affecting pool DF include the following:

- Most important:
  - Particle size
  - Particle density
  - Bubble size and shape
  - Volume fraction of steam in inlet gas

- Intermediate importance:

    - Pool depth
    - Pool temperature
    - Percent of soluble material in particles

- Least important:

    - Noncondensible gas composition
    - Pressure above pool

The low pool DF for aerosols in the 7.01 version of MAAP leads to a misrepresentation of containment capability. Battelle Columbus data (EPRI-NP-4890SP) supports DFs of at least a factor of 10. Therefore, the pool DF has been hand calculated for CsI that passes through the pool when it is saturated. The release fractions for CsI were reduced by a factor of ten for the portion of the release that passed through the suppression pool for cases with a saturated suppression pool. The potential impacts of an unadjusted MAAP calculation on the DAEC analysis are overprediction of the radionuclide source term for scenarios involving wetwell failure or venting. Applicable DAEC releases were reduced by a factor of 10 to account for this conservatism.

Table 4.9-17 shows the sequences which benefitted from such a reduction.

4.9.3.9 Containment Flooding Sensitivity Evaluation

Given the current state of knowledge regarding severe accident phenomenology, the DAEC EOPs have established a near optimum balance among the contingency procedures which the operator can implement.

Table 4.9-17 Effect of Suppression Pool Decontamination Factor		
Sequence Number	MAAP CsI Release Fraction	Corrected CsI Release Fraction
LII-1D-01	6.03E-2	6.03E-3
LII-1D-04	1.30E-3	1.30E-4
LII-3A-02	1.71E-1	1.71E-2
LII-1A-03LW	5.31E-4	5.31E-5
LII-1A-03LWA	2.90E-4	2.90E-5

The EOPs generally define one of the following: the optimum procedural path; a procedural pathway that is close to optimum; or a pathway for which insufficient analytical (and experimental) information is available to more precisely define the optimum pathway. Changes in the current understanding of severe accident phenomena or in the philosophy of dealing with severe accidents may impact some of the EOP steps and contingency actions. The specific issue that is addressed here is the decision regarding containment flooding versus other possible effective alternatives.

Postulated Scenario

The specific accident sequence investigated has the following elements:

- Core damage occurs due to the loss of injection makeup to the RPV.

- The containment is initially intact and inerted.

- The RPV water level continues to drop below 1/3 core height.

- Eventually the RPV bottom head is breached and core melt progression continues with debris released to the drywell.

- In the meantime, power and injection sources are reestablished. Several choices for debris cooling exist: water from LPCI, core spray or RHRSW to the RPV, drywell sprays.

- RHRSW, or another system using an external water source, is be used to flood the containment.

- The containment flooding process will require:

  - RPV venting when the containment water level reaches the bottom of the recirculation lines (i.e., near the bottom of the RPV and above any debris on the drywell floor).

  - Drywell venting if the containment pressure cannot be maintained below the Primary Containment Pressure Limit (PCPL) (i.e., approximately containment design pressure).

The aforementioned actions are specified in the BWROG Rev. 4 EPGs and the DAEC EOPs for scenarios requiring containment flooding. These actions are directed whenever the RPV water level is below approximately 2/3 core height and cannot be restored, <u>OR</u> when the RPV water level is indeterminant and RPV pressure cannot be maintained 50 psig greater than wetwell pressure.

Analysis

After performing detailed thermal and hydraulic analysis for various BWR IPEs (Mark I and Mark IIs), it has become clear that under certain postulated severe accidents, the BWROG EPGs direct operators to perform actions that could have a more adverse potential impact on the public than other alternative strategies that could be implemented. The actions in question involve the means to implement containment flooding, and venting the RPV and DW.

MAAP 3.0B was not designed to calculate all of the thermal-hydraulic conditions in a flooded containment. However, with the careful use of input deck changes and operator actions, the MAAP code was used to estimate the releases for each of the Class I, II, III, and IV accident categories. The results, list of assumptions, and major findings from the MAAP cases used for "flooding" are described below.

Methodology Limitations to Model "Flooding" with MAAP

There are five major limitations in MAAP which have prevented it from being used for "flooding" scenarios in the past; namely: (1) there are no provisions to estimate the heat transfer from the vessel to a surrounding water pool; (2) water from the containment, will not reenter the vessel as the level rises in the containment; (3) gas and fission product transport through a failed vessel does not take into account the possible existence of a water pool surrounding the vessel; (4) the model of the pedestal region cannot tolerate going "solid" with water; and (5) the model of the suppression pool/wetwell region cannot go "solid."

The first three limitations would require extensive coding changes to resolve, but the last two can be accommodated with input changes. Therefore, by making the appropriate changes to the input deck to overcome the "going solid" limitations, one can analyze the resulting output keeping in mind the first three limitations.

To prevent the pedestal region from going solid, 1000 m^3 of free volume was "borrowed" from the drywell region. That is, for these runs, the pedestal free volume was increased by 1000 m^3, and correspondingly, the drywell free volume was decreased by 1000 m^3 from their initial values. This allows the pedestal region to collect more water, while maintaining the appropriate total volume and elevations in the now-combined drywell/pedestal region.

Preventing the suppression pool from going solid requires a slightly different course of action. After flooding is initiated, one can isolate the suppression pool from the calculation if pool cooling is not on, and the pool is not acting as the source of water for an injection or spray system, if vacuum breakers are not stuck open, and if the vessel SRVs are no longer opening. In the example flooding scenario, the suppression pool was effectively "isolated" from the drywell by setting the curb height between the drywell and wetwell to a large value, and by setting the communication paths (the vacuum breaker area and downcomer area) to a minuscule value (~1.E-6 $ft.^2$). This is deemed acceptable since minimal interactions between the drywell and wetwell gases are anticipated once flooding has begun. The shortcoming to this approach is that once these artificial changes are made, the calculated wetwell level, temperature, and pressure are no longer reliable. Basically, the approach taken is to admit that at some point shortly after initiating flooding, the interactions between the wetwell and other containment volumes are no longer important.

Class I

For Class I the representative sequence is a loss of all injection, RPV depressurization when the core water level drops to TAF, containment venting available, no suppression pool cooling, and containment flooding through the RPV initiated shortly after vessel failure. This sequence leads to rapid heatup and vessel failure before 2 hours. In the wetwell venting case, the containment pressure is reduced to, and maintained at, approximately 2 psig by leaving the vent valves open after actuation. In the RPV vent case, containment flood is initiated when vessel breach occurs. The wetwell is artificially "isolated" in MAAP on high torus level at roughly 4 hours, at which time the wetwell vent closes and the drywell vent opens. Flooding of the drywell continues until the level reaches TAF. The open drywell vent drops the pressure until the water in the drywell heats and saturates, at which point the pressure equilibrates at 6 psig.

The drywell vent case is characterized by a small CsI release when the vent is first opened at about 4 hours. After 36 hours, 6.5% of the initial CsI mass is calculated to be transported to the reactor building, and with the low DFs available on the refuel floor, 6% of the CsI mass goes to the environment. For these calculated releases, MAAP takes no credit for scrubbing of fission products even though they are transported from the vessel to the drywell through a water pool and out through the drywell vents.

The RPV vent case is also characterized by a CsI release starting at about 4 hours, but this release steadily continues until about 13% of the initial CsI mass has been transported to the condenser by 36 hours. The condenser/turbine building combination, however, provides for slightly higher calculated DFs than the refuel floor such that the final release to the environment is about 6.7% of the initial CsI mass. The dominant removal mechanism in the condenser/turbine building region was calculated to be gravitational sedimentation, and due to the low flow rates at the time of fission product transport from the vessel, a calculated DF of about 2 in the condenser and the turbine building appears reasonable for this case, but it is unknown whether this would be the situation in all circumstances. In any event, crediting a DF of at least 1.5 or 2 for all RPV vent cases seems reasonable.

Class II

For this accident class, the representative sequences are a loss of all decay heat removal. LPCI injection is lost, and after vessel failure, containment flooding is initiated.

For the vent case (LII-2T-3), about 2.5% of the CsI was released to the reactor building as the wetwell and drywell vents were used to maintain the containment pressure below 53 psig. Of this release, about 1.9% of the initial CsI mass ends up in the environment. The release is lower than the previous flooding cases because a decontamination factor in the pool is credited during the time of wetwell venting.

Conclusions

For Class I flooding sequences, the timing and magnitude of fission product releases can be conservatively estimated as early and moderate, respectively. Class III flooding sequences are considered similar to the Class I results, and therefore, the fission product releases can be conservatively estimated as early and moderate.

In summary, the timing and magnitude of the releases can be categorized as shown in Table 4.9-18 with the notion that if anything, the drywell vent cases will experience lower releases than those reported here.

Interesting insights can be obtained by comparing these results with similar cases that do not involve containment flooding.

Table 4.9-18

SUMMARY OF MAAP RESULTS FOR DAEC 'FLOODING' SCENARIOS

Class	Sequence	Description	Time of Initial Release	CsI Fraction to Reactor/ Turbine Building	CsI Fraction to Environment	Timing of Release[1]	Magnitude of Release to Environment [3]
I	LII-ID-9	Flood, DW Vent to Refuel Floor	4 Hr.	0.5%	6.0%	E	M[3]
	LII-ID-10	Flood, DW Vent, RPV Vent to Condenser	4 Hr.	6.3%	6.7%	E	M

(1) Time of initial release
E - Early, < 6 hours
I - Intermediate, 6 to 24 hours
L - Late, > 24 hours

(2) CsI Release Fraction Severity
H - High > 10%
M - Moderate, 1% to 10%
L - Low, 0.1% to 1%
LL - Low-Low, < 0.1%

(3) The drywell vent case does not credit fission product scrubbing through the water pool surrounding the RPV. Therefore, the estimated release is considered to be conservative. In all likelihood, the actual release would be lower for these cases.

Table 4.9-19 Comparison of Class ID Sequence: Containment Flooding Versus No Flooding			
MAAP Case	Description	Release Magnitude from Containment	Time
LII-1D-9	Drywell Vent as part of containment flooding procedure	Moderate (6% CsI)	Early
LII-1D-10	RPV Vent as part of containment flooding procedure	Moderate (6.7% CsI)	Early
LII-1D-8	Wetwell Vent and No Flood	Low (0.7% CsI)	Intermediate
LII-1D-5	RHR, Sprays or Injection, and no Flood	No Release+	No Release

+ Except Leakage

The accidents in question involve example core melt sequences with the containment initially intact. Various strategies were evaluated. Calculated release magnitude and timing are shown in Table 4.9-19 for the Class ID sequences. Minimum credit was given to ex-containment DF. This may bias the results as indicated in the text.

These results can be applied to other groups of accidents which have different boundary conditions.

A possible improved response for these types of sequences for which the EPG directions result in the highest potential consequences at the earliest time, is to provide the operators guidance on protecting containment and cooling debris using methods that do not require venting the RPV and avoid using the DW vent unless no other alternative exists. No action has been shown to produce substantially lower releases and much longer times to failure, i.e., even no action is better than action directed by the EPGs.

It should be noted that the drywell vent cases do not credit scrubbing as fission products are transported through the water pool that would accumulate around the vessel as flooding of the containment progression. Therefore, the calculated releases for these cases are considered conservative. For the RPV vent cases, however, the associated DF post-vessel release may not be as good in all cases. Although a lower DF may increase the actual release, in most cases this should not affect the release magnitude categorization.

More sophisticated modeling may reduce these estimates of release magnitude, but the fact remains that the actions being specified will inhibit the movement of personnel who are on-site and who are responsible for recovery, potentially early in a sequence. This would occur due to the venting and radionuclides into the turbine building area via RPV venting to the main condenser.

4.9.3.10 Sensitivity of Radionuclide Release to Level 1 Sequence Type

There are a number of subtle effects that can influence the radionuclide release magnitude. As has been shown, there are a number of parameters that influence the radionuclide release to create high releases. There appears to be one overwhelming effect that results in a medium or lower release for ATWS related sequences (Class IV) involving a wetwell failure. The variation in radionuclide release to the type of accident sequence can be seen by examining two sequences:

- Loss of Makeup (Class IA): LII-1A-04

- ATWS (Class IV): 4A-01

These results are important in characterizing the radionuclide releases for representative loss of makeup sequences compared with ATWS.

Figure 4.9-15 supports the premise that under ATWS conditions a substantial quantity of CsI is transferred to the wetwell as compared with the Class I, loss of makeup cases. This substantial difference in the CsI location then makes much less of the CsI available for release in the ATWS scenarios.

Important results are shown in Table 4.9-20 for steam flow mass through the SRVs. Table 4.9-21 shows the CsI release to the environment for representative accident classes. The results for ATWS are a function of the larger steam flow rate to the suppression pool during the time of radionuclide release. This larger steam flow rate tends to flush fission products out of the RPV and to the suppression pool. This phenomenon results in reducing the fission products available for subsequent release due to revaporization (i.e., up to four times as much CsI is scrubbed through the suppression pool prior to vessel failure compared to the loss of makeup sequence). The result is that the CsI available to be released from the RPV later in the sequence is reduced, resulting the potential for a lower radionuclide release for the ATWS case as opposed to a loss of makeup sequence.

4.9.3.11 Drywell Spray Usage Under Severe Accident Conditions

The DAEC EOPs make extensive use of the drywell sprays based upon symptoms of containment pressure, temperature, and combustible gas concentration. In accident management investigations, it is also of interest to identify whether the drywell sprays may provide even more capability to reduce the potential source term under severe accident conditions. The particular uses that have been considered include:

Table 4.9-20

COMPARISON OF THE CAUSE AND EFFECT OF SEQUENCE VARIATIONS

Total Integrated Steam Mass Through SRV at:	CASE	
	4A-01 (ATWS)	1A04 (Boiloff)
Core Uncovered	2.45E+5 Kg	1.25E+5 Kg
1/3 Ht.	1.44E+6 Kg	1.58E+5 Kg
Onset Melt	1.45E+6 Kg	1.65E+5 Kg
Vessel Failure	1.50E+6 Kg	1.95E+5 Kg
CAUSE		
Incremental Steam Mass Through SRVS between:	CASE	
	4A01 (ATWS)	1A04 (Boiloff)
Between 1/3 Ht. → melt	10,000 Kg	7,000 Kg
Between 1/3 Ht. → vessel failure	60,000 Kg	37,000 Kg
EFFECT		
CsI to Reactor Building	1.6%	13%

The result is about a factor of 2 increase in SRV flow for the Class 4 ATWS case. Therefore, there is more transport of fission products to the suppression pool where the subsequent re-evolution is considered extremely low.

- Scrub Aerosols: Scrubbing fission products from the drywell atmosphere. (Explicit symptoms to initiate sprays during radiation incidents or radionuclide releases is currently not included in the generic BWROG EPGs or the DAEC EOPs).

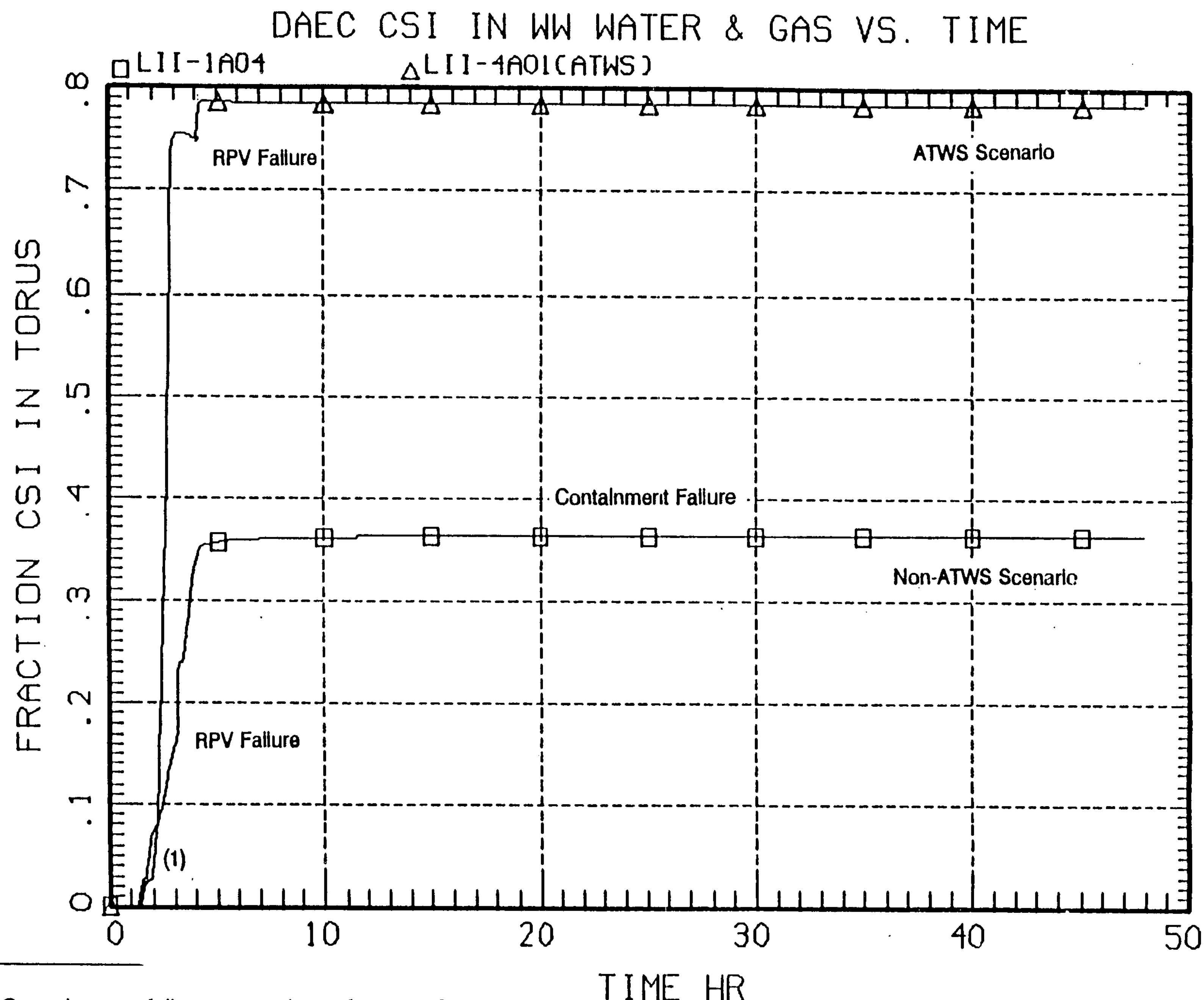
- Quench molten debris: Cooling the drywell when molten debris is present on the drywell floor. (Explicit directions are given to the operator to prevent drywell spray initiation if drywell temperatures are above approximately 350°F in the drywell.)

Table 4.9-21

RADIONUCLIDE RELEASE TO THE ENVIRONMENT FOR DRYWELL SHELL OR HEAD FAILURES:[1]

Case	Class	Release	Release Category
LII-1A-4	Class IA	4.51E-2	M
LII-1D-LF	Class ID	4.6E-2	M
LII-2T-1	Class II	1.71E-1	H
LII-4A-01	Class IV	1.62E-2	M

[1] Without Reactor building effectiveness



(1) Containment failure at 1.0 hour for ATWS

Figure 4.9-15

Comparison of CsI in Suppression Pool for Class 1A and ATWS Scenarios

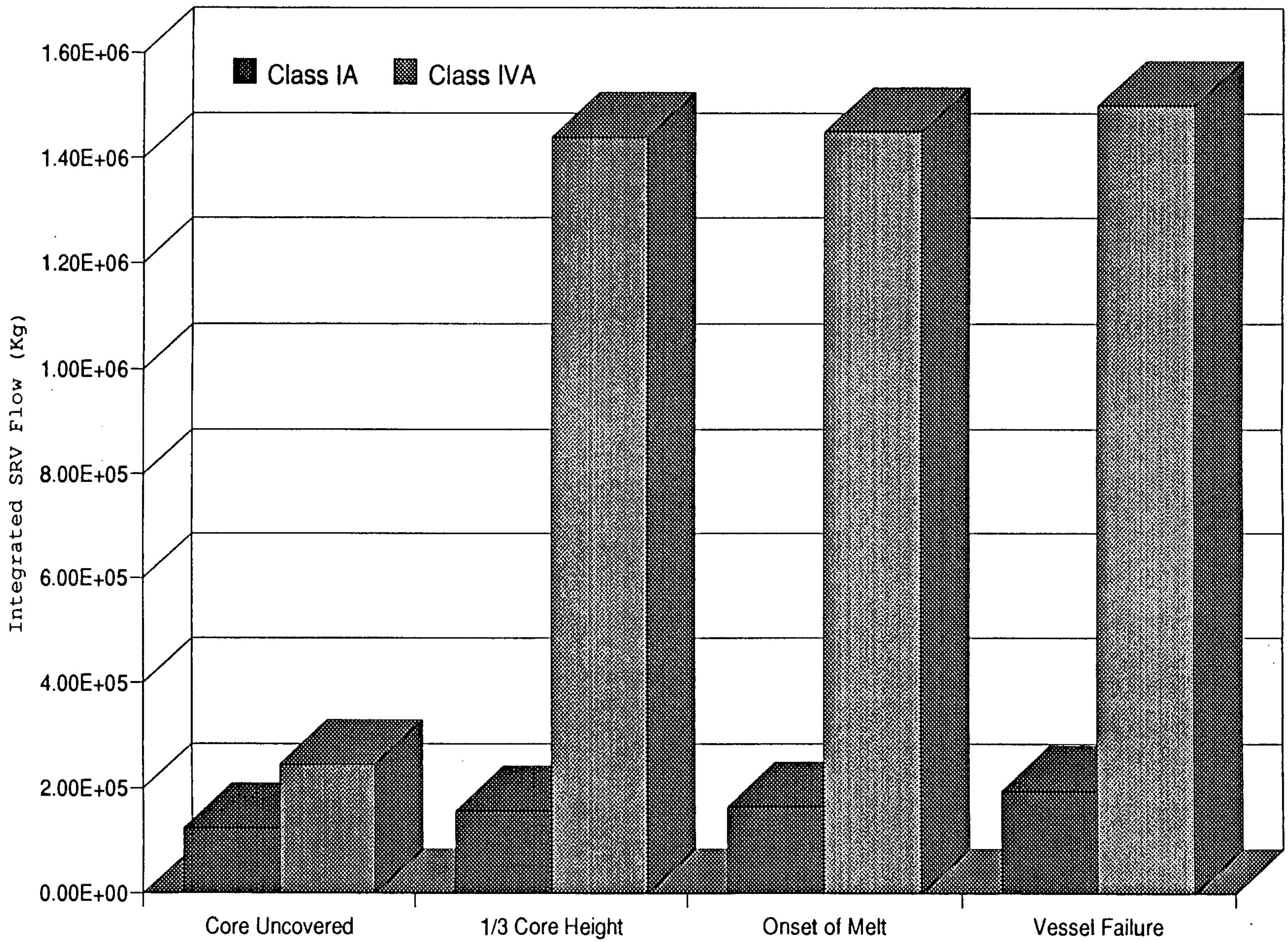


Figure 4.9-16
Comparison of Total Steam Mass Through SRVs

- Minimize both core concrete interaction and the generation of non-condensible gases: No explicit directions are provided to minimize core concrete interaction, nor does a universal symptom appear to exist which might always be available. (See the drywell spray initiation curve given in the EOPs.)

- Prevent suppression pool bypass: Cooling the containment shell, downcomers and vacuum breakers from the drywell to the wetwell to prevent failure and consequential suppression pool bypass by the radionuclides during a postulated severe accident.

- Mitigate Combustible Gas Mixtures: During the development of the DAEC EOPs, it was recognized that the drywell sprays had a wide variety of uses in postulated accidents that are not part of the licensing design basis. Therefore, the EOPs incorporated this existing plant capability within the EOPs to allow the operating staff to use drywell sprays for the following types of events as specified in the DAEC EOPs:

  - Reduce the drywell temperature to avoid exceeding the containment design temperature (e.g., small LOCAs, loss of drywell coolers);

  - Reduce the drywell pressure when primary containment pressure exceeds 9 psig;

  - Reduce the drywell pressure regardless of adequate core cooling when the containment pressure exceeds the Primary Containment Pressure Limit (PCPL); and

- Suppress hydrogen detonation at high hydrogen and oxygen concentrations.

However, their usefulness for severe accident conditions is affected by the limitations in the EOPs on the use of the sprays.

Four of the DAEC EOP limitations on the use of drywell sprays which restrict the initiation or continued operation of the drywell sprays are as follows:

- Drywell spray initiation curve is cited in all cases of drywell spray initiation. The drywell temperature and pressure must be in the "safe" regime to allow initiation as shown in Figure 4.9-17.
- Drywell pressure must remain greater than 1.68 psig or the sprays must be terminated.

- Suppression pool level must be lower than a height that could cover the vacuum breakers to initiate drywell sprays.

- The restriction on diversion of water from RPV injection if adequate core cooling is not assured.

Therefore, it is the purpose of this sensitivity to present a summary comparison between the deterministic containment calculations for postulated severe accidents and the drywell spray initiation curve for DAEC, i.e., the first item above.

The second bullet is also addressed by this sensitivity. The third bullet is not considered to be a limitation for the purposes of this examination.

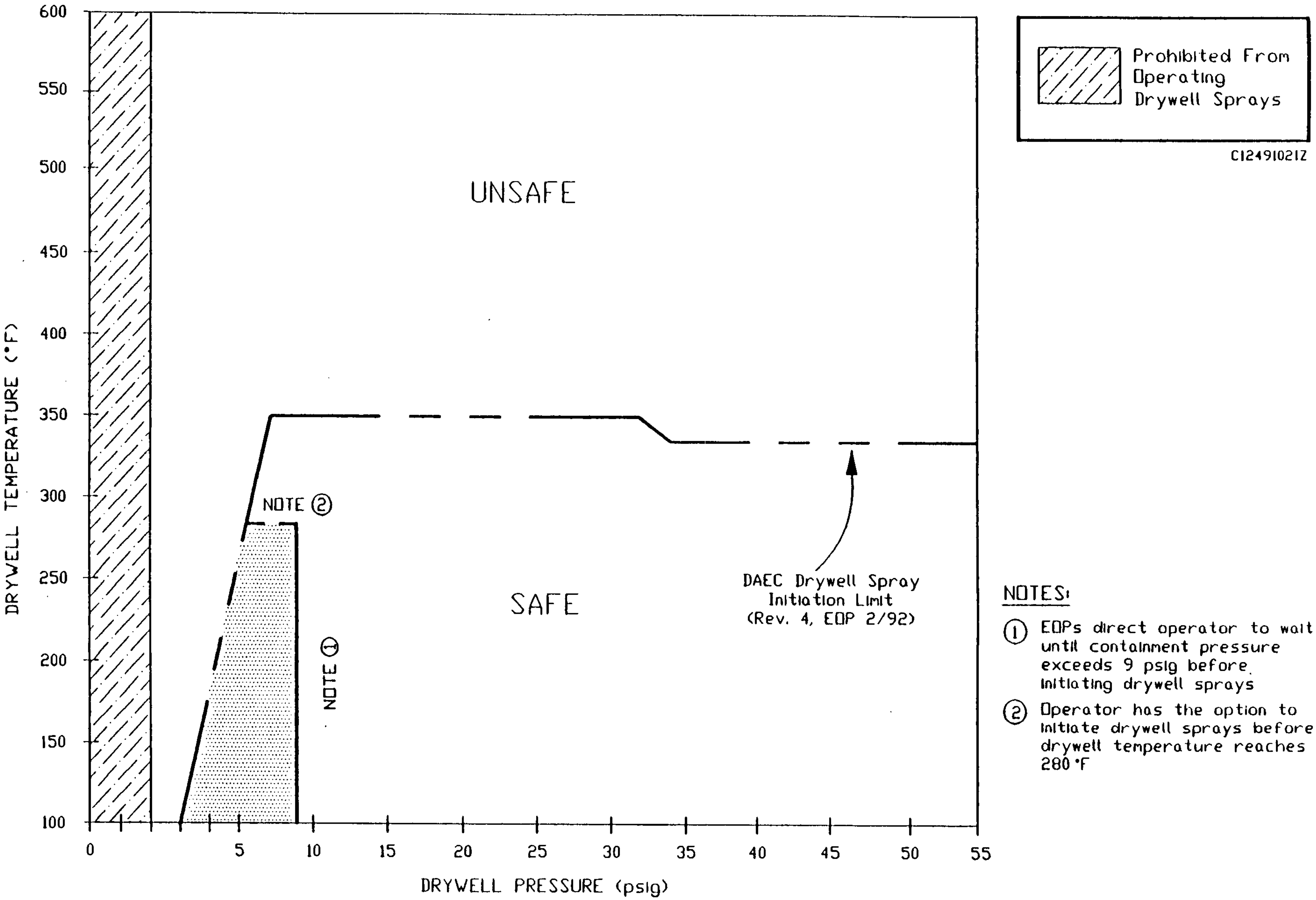


Figure 4.9-17 DAEC EOP Limits for Drywell Spray Initiation

The fourth bullet is another important procedural limitation. The RPV has presumably been breached and water level would not be able to be maintained above TAF. This could be considered a condition for which adequate core cooling is not assured. For such cases, the current EOPs and Revision 4 of the BWROG EPGs may be interpreted as ambiguous regarding the use of drywell sprays to be used because adequate core cooling cannot be assured.

- For plants where water for drywell sprays is supplied by RHR pumps, operation of drywell sprays is only permitted if the pumps to be used are not required to assure adequate core cooling.

- Maintaining adequate core cooling takes precedence over maintaining drywell temperature below design temperature, because catastrophic failure of the containment is not expected to occur at drywell design temperature.

- Accordingly, operation of RHR pumps aligned in the drywell spray mode is permitted only if continuous operation of the pumps in the LPCI mode is not required to assure adequate core cooling.

- The wording of this step does, however, permit alternating the use of RHR pumps between LPCI injection and drywell spray as the need for each occurs and so long as adequate core cooling is able to be maintained.

For DAEC, a clarification was sought to ensure that the EOPs and operator training were being properly modeled in the DAEC Level 2 analysis. The result of this clarification is the following:

- If in Class IA and cannot inject into the RPV with the RHR pumps, then operators would use RHR pumps for sprays when directed by the EOPs.

- If RPV water level cannot be restored, but indication is that RHR pumps are pumping to the RPV, operators would not divert RHR pumps to spray the containment. SROs are trained to prefer injection to the core and wait for PCPL to be reached when the EOPs say spray irrespective of adequate core cooling, spray the drywell.

Data Presentation and Summary

The technique used in plotting the curves is to plot the MAAP calculated drywell temperature and pressure point for a severe accident on the same axis as the allowable limit from the drywell spray initiation (DWSI) curve. The MAAP calculated temperatures and pressures must be less than the "allowable" in order for the operator to "safely" initiate drywell sprays.

The accidents investigated are shown in Table 4.9-22.

An example of the plotting technique is shown in Figure 4.9-18 for a Class ID sequence, Loss of Makeup Inventory at Low RPV Pressure (TQUV). The loss of inventory makeup at low pressure has been found to be an important accident sequence contributor to high releases. A typical TQUV pressure and temperature trace and the drywell spray initiation curve are shown together on Figure 4.9-18. The combination of pressure and temperature are within the "safe" region for only for approximately 7 minutes past RPV failure, i.e., the time between breach and drywell shell attack by debris. In addition, the

EOP procedure to maintain adequate core cooling is judged to prevent the diversion of water flow from the RPV to containment sprays.

A second example is Class IA type sequences.

The DAEC EOPs appear to allow drywell sprays to be initiated during in-vessel core melt progression for accident Class IA. In other words, the calculated deterministic conditions for postulated Class IA/IB accidents are such that both the symptoms to initiate and the parameters to allow initiation are all satisfied prior to RPV breach. This is judged to be an important severe accident consideration because of the need to assure cooling of debris on the drywell floor and prevent shell failure. Nevertheless, small errors in instruments or instrument failures can lead the operating staff to delay or prevent initiation of the drywell sprays until it is too late. Therefore, additional margin or incentive to spraying the drywell during core melt progression is considered desirable as part of accident management and an outcome of the IPE.

In summary, the current Drywell Spray Initiation Curve (DWSI) and EOP procedures for DAEC provides restrictions on when the drywell sprays can be initiated. Given this curve and current EOP guidelines, it appears that for some of the severe accidents investigated in the Level 2 analysis, little or no time window exists during which the sprays could be initiated.

It is also recognized that regardless of the flexibility offered by the DWSI curve, additional changes to the EOPs may also be required to allow diversion of injection sources away from the RPV when adequate core cooling was apparently not assured, i.e., low reactor water level.

Table 4.9-22 Drywell Spray Usage		
Accident Type	Class[1]	Time Window for DW Spray Actuation (Hrs)
Loss of Makeup Injection to RPV at High Pressure (TQUX)	IA	4[2]
Unavailability of Containment Decay Heat Removal	II	20[3]
Large LOCA	IIIC	~0
LOCA Outside of Containment	V	0
Loss of Makeup Injection to RPV at Low Pressure (TQUV)	ID	6[2]
ATWS	IV	1[4]

[1] Definition of classes included in the DAEC PRA.

[2] The need for adequate core cooling may prevent use of drywell sprays.

[3] If containment heat removal equipment is unavailable, DW spray equipment is likely also unavailable.

[4] Operation of drywell sprays prior to containment failure would not have a major impact on the containment response.

Nevertheless, the drywell sprays are not currently considered to be effective methods of cooling debris in the containment when:

(1) Drywell pressures are below the PCPL (~53 psig)

OR

(2) H_2/O_2 are not in a combustible mixture.

OR

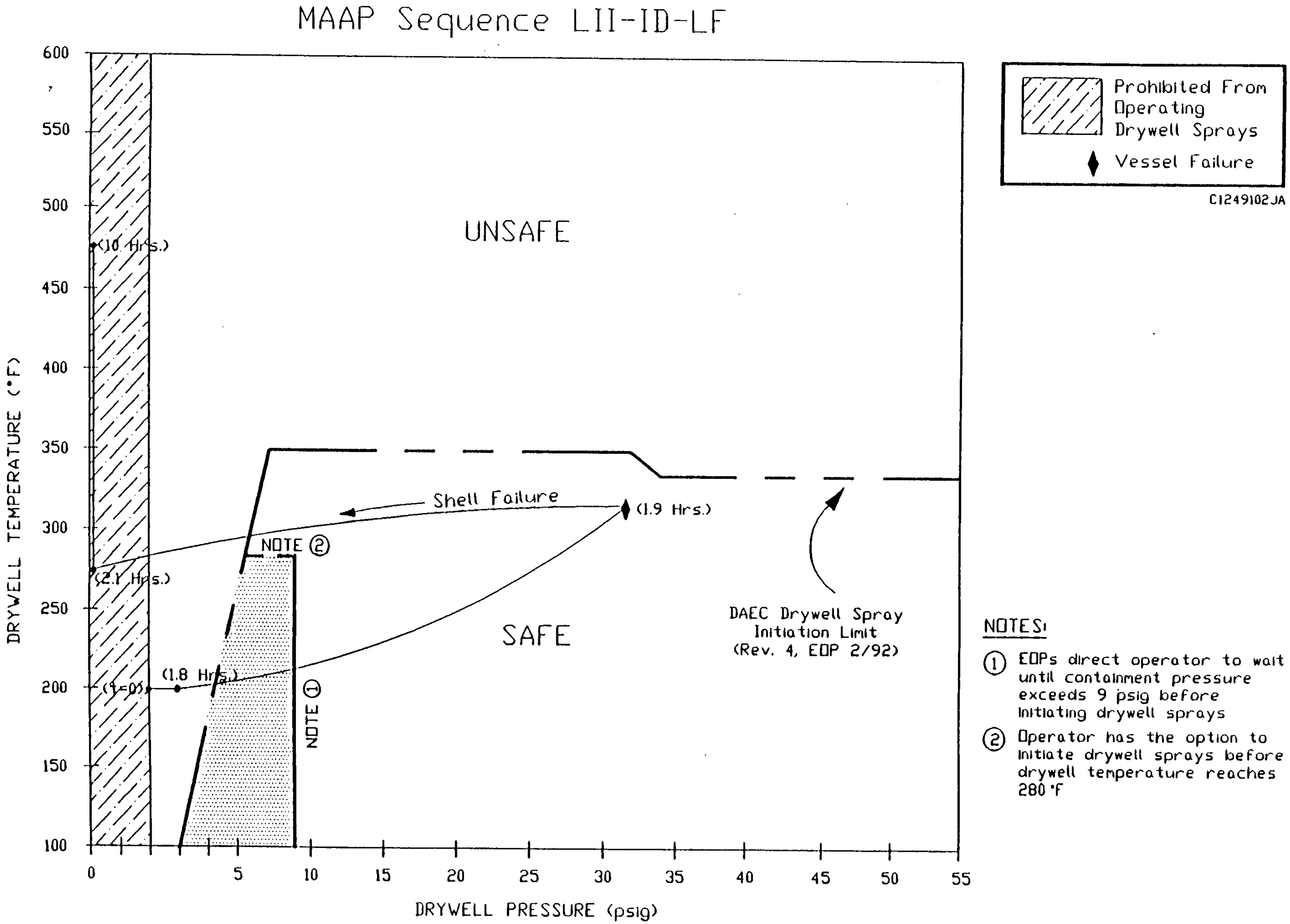


Figure 4.9-18 DAEC EOP Limits for Drywell Spray Initiation for Loss of Makeup at Low Pressure with Shell Failure

(3) When little or no time duration exists for the operating staff to assess plant symptoms and implement the sprays.

These apply to Classes IIIC, V, and IV for DAEC. For other accident classes drywell spray initiation appears viable.

4.9.3.12 Revaporization of Deposited Fission Products

There have been questions raised concerning possible chemical reactions taking place between the steel surface and deposited CsI. It has been hypothesized that this chemical reaction would effectively delay the revaporization of the CsI. In an effort to simulate this behavior within the MAAP code, the mass of the steam dryers in the RPV was increased by about a factor of twenty. This was a way to slow down the heat up and eventual revaporization of the CsI. (See Table 4.9-23)

4.9.3.13 High Pressure Melt Ejection

The RPV pressure at vessel failure, in conjunction with the assumed drywell shell failure, may have some impact on the release of fission products from containment. The base case assumed that the operator would depressurize when the vessel water level reached the top of active fuel. In the sensitivity case the vessel remained at high pressure. Drywell shell failure at 7 minutes after vessel failure was assumed for these cases. Table 4.9-24 compares the results for the two cases.

Table 4.9-23 DELAYED CsI REVAPORIZATION								
MAAP Case	Accident Class	Description	Containment Failure		CsI Release			
			Time (hr)	Location	in RB at 30 hrs	to Env. at 30 hrs	in RPV at vessel failure	in RPV at 20 hrs
LII-1D-LFA	IA	Base Case DW Eqpt. mass = 2.7m lbm	3.50	Shell (large)	34.3%	7.2%	78%	12%
LII-1D-LFG	IA	Decreased DW mass DW eqpt. mass = 1m lbm	3.47	Shell (large)	36.5%	6.1%	80%	11%
LII-1D-LFF	IA	Delayed Fission product Revap. In-vessel Steam Dryer Mass[1]	3.45	Shell (large	32.5%	3.4%	85%	18%

[1] Delay in fission product revaporization was in the MAAP code by artificially increasing the dryer mass in the RPV.

Due to a delay in the CsI revaporization rate (IDLFF), the amount of CsI released to the environment is slightly lower than the base case. A decrease in the drywell equipment steel mass (IDLFG) resulted in an increase in the fission product release due to the increase drywell gas temperature. No major impact on conclusions was found based on these sensitivities.

Table 4.9-24 RPV PRESSURE AT VESSEL FAILURE								
MAAP Case	Accident Class	Description	Containment Failure		CsI Release			
			Time (hr)	Location	in RB	to Env	in RPV at vesse l failure	R.B. DF
LII-1D-LF	ID	RPV fails at low pressure Shell failure at RPV failure + 7 minutes	1.92	Shell (large)	37.7%	4.6%	95/5	9.2
LII-1D-LFA	IA	RPV Fails at high pressure Shell failure at RPV failure + 7 minutes	3.38	Shell (large)	34.3%	7.2%	78%	5.8[1]

[1] With the vessel failing at high pressure, there is less deposition in the reactor building.

4.9.5 Summary

The DAEC sensitivity evaluations have identified the IPE variations that can be introduced by modeling uncertainties associated with core melt progression physical phenomena. Table 4.9-25 summarizes the key results of these evaluations.

Table 4.9-25

SUMMARY TABLE OF DETERMINISTIC CALCULATIONS (MAAP) SENSITIVITIES

Sensitivity Variable Investigated	Phenomena Affected	Accident Management Result/Conclusion
	Core Melt Progression: Amount of Residual Debris in RPV (Section 4.9.3.1)	In the unlikely situation that core debris remains behind in the RPV (examples include RPV rupture (examples include RPV rupture or large LOCA without makeup injection), then accident management actions should focus on trying to keep core debris cool in the RPV. The core spray system would be the preferred RPV injection source under such degraded conditions.

Table 4.9-25

SUMMARY TABLE OF DETERMINISTIC CALCULATIONS (MAAP) SENSITIVITIES

Sensitivity Variable Investigated	Phenomena Affected	Accident Management Result/Conclusion
	Debris Coolability (Section 4.9.3.2)	Modeling differences can result in differences in the non-condensible gas generation rate for cases in which water is available to the debris when it is on the drywell floor. The modeling differences can include reduced heat transfer coefficients from the debris to the water. This can also be affected by sump depth.
	- Non-Condensible Gas Generation (Section 4.9.3.2.1)	If water can be supplied to the debris then core-concrete interaction of the drywell floor can be minimized. This control of core-concrete interaction and the non-condensible gas generation results in a safe stable state for containment if torus cooling can also be achieved.
	- Debris Cooling in the Sump (Section 4.9.3.2.2)	The sump depth at DAEC is explicitly included in the baseline calculations. therefore, the above insight is applicable.
	- Pedestal Attack (Section 4.9.3.2.3)	Substantial core concrete interaction in the deep sumps (2.2 ft. deep) can occur if the worst case heat transfer coefficient between debris and water is used. The vertical depth of core concrete interaction is not sufficient to threaten containment and the horizontal erosion is considered to be substantially less than the vertical. No additional accident management actions appear desirable.
	- Effective Area of Drywell Floor (Section 4.9.3.2.4)	The area of the drywell floor which is covered with debris can also influence the amount of non-condensible gases. However, this sensitivity investigation determined only insignificant differences in the concrete depth of attack and time to release. No additional accident management actions deemed appropriate based on this potential modeling difference.
	Aerosol Plugging (Section 4.9.3.3)	Not an accident management issue.

Table 4.9-25

SUMMARY TABLE OF DETERMINISTIC CALCULATIONS (MAAP) SENSITIVITIES

Sensitivity Variable Investigated	Phenomena Affected	Accident Management Result/Conclusion
	Core Blockage (Section 4.9.3.4)	

Table 4.9-25

SUMMARY TABLE OF DETERMINISTIC CALCULATIONS (MAAP) SENSITIVITIES

Sensitivity Variable Investigated	Phenomena Affected	Accident Management Result/Conclusion
	Containment Failure Mode (Size and Location) (Section 4.9.3.5)	Containment failure location has a significant influence on the time and magnitude of radionuclide releases. Therefore, accident management actions that may be available to alter the pathways can be important in protecting public health and safety.
	- Containment Failure Area (Section 4.9.3.5.1)	Containment failure size does not appear to be either a controllable parameter through accident management actions or one that has a clear impact on release magnitude. therefore, no accident management actions are derived form the potential sensitivity to failure size.
	- Containment Failure Location (Section 4.9.3.5.2)	Accident management actions should address the following key issues: · Water injection to the RPV (preferentially) or to the drywell can inhibit severe containment failure locations. For example, water injection has the following beneficial effects: - Water will either prevent or cause a refreezing of debris attach at the shell-drywell floor interface. The first situations results in no release, the second situation should result in a substantial reduction in the release magnitude. The "best" method of supplying the water to the debris is to provide the water in the drywell before RPV breach. the supply after RPV breach could be through the RPV or drywell sprays. Even though containment flooding may have been invoked, it is important that the debris cooling be a main focus of the accident management actions to continue to preclude adverse containment failure locations.

Table 4.9-25

SUMMARY TABLE OF DETERMINISTIC CALCULATIONS (MAAP) SENSITIVITIES

Sensitivity Variable Investigated	Phenomena Affected	Accident Management Result/Conclusion
	- Containment Failure Location (Section 4.9.3.5.2) (con't)	- Water injection can prevent high drywell temperatures that may result in a challenge to the drywell head as a failure location. the water injection is preferentially desired to the RPV to ensure that any residual debris retained in the RPV is cooled and DW temperature increases are minimized - Containment venting will reduce pressure in containment to avoid catastrophic containment failure and an uncontrolled release. In addition, if pool bypass can be avoided, the use of the wetwell vent will provide, a scrubbed release path.
	- Drywell Shell Failure (Section 4.9.3.5.3)	SEE ABOVE

Table 4.9-25

SUMMARY TABLE OF DETERMINISTIC CALCULATIONS (MAAP) SENSITIVITIES

Sensitivity Variable Investigated	Phenomena Affected	Accident Management Result/Conclusion
	Reactor Building Modeling Assumptions (Section 4.9.3.6)	The ability to minimize releases to the environment can be dramatically affected by the effectiveness of secondary containment, i.e., the reactor building. The first goal of the secondary containment accident management should be to retain integrity as long as possible. From our analyses, this appears possible when containment integrity can be maintained and only "leakage" is occurring. SGTS can be effective mitigating system under these conditions. For the severe accidents that may fail containment, the reactor building integrity is found to be compromised due to opening of the protective "blowout" panes in the refuel floor. Therefore, accident management actions that are formulated to mitigate severe accidents with a breached containment should include provisions to enhance reactor building effectiveness recognizing the potential for failures in: · Refuel floor blowout panels · HPCI/RCIC room plugs · Railroad doors · Steam tunnel to turbine building blowout panels. The accident management actions that appear most fruitful include: · Operation of SGTS (both trains if it all possible) · Operation of fire protection sprays (i.e., do not terminate the fire protection spray once if it has been automatically initiated)
	Equipment Mass in Drywell and Effect on Drywell Temperature Post RPV Breach (Section 4.9.3.7)	No accident management actions are derived directly from this sensitivity. Rather it is clear that the containment's robustness to deal with temperature excursions may fall in a broad range, such that the uncertainty in this range argues for as much drywell cooling, e.g., sprays as feasible.

Table 4.9-25

SUMMARY TABLE OF DETERMINISTIC CALCULATIONS (MAAP) SENSITIVITIES

Sensitivity Variable Investigated	Phenomena Affected	Accident Management Result/Conclusion
	Pool Decontamination Factor (DF) (Section 4.9.3.8)	The use of the suppression pool as a method of reducing the radionuclide release, i.e., using the wetwell vent may be adversely impacted by high suppression pool temperature. Therefore, accident management actions must acknowledge the possibility that high pool temperatures may result in elevated releases. The accident management guidance may include the following: · Maintain pool cooling · Add mass to the pool · Minimizing venting.
	Containment Flooding Sensitivity Evaluation (Section 4.9.3.9)	A possible improved response for these types of sequences for which the EPG directions result in the highest potential consequences at the earliest time, is to provide the operators guidance on protecting containment and cooling debris using methods that do not require venting the RPV and avoid using the DW vent unless no other alternative exists. No action has been shown to produce substantially lower release and much longer times to failure, i.e., even no action is better than action directed by the EPGs. The actions being specified by the containment flood contingency will inhibit the movement of personnel who are on-site and who are responsible for recovery, potentially early in a sequence. This would occur due to the venting and radionuclides into the turbine building area via RPV venting to the main condenser.
	Sensitivity of Radionuclide Release to Level 1 Sequence Type (Section 4.9.3.10)	The types of sequence can strongly affect accident management actions. These will be studied more in the future.

Table 4.9-25

SUMMARY TABLE OF DETERMINISTIC CALCULATIONS (MAAP) SENSITIVITIES

Sensitivity Variable Investigated	Phenomena Affected	Accident Management Result/Conclusion
	Drywell Spray Usage Under Severe Accident Conditions (Section 4.9.3.11)	The current Drywell Spray Initiation Curve (DWSI) for DAEC provides restrictions on when the drywell sprays can be initiated. Given this curve, it appears that for some of the severe accidents investigated in the Level 2 analysis that the curve will not be satisfied at all or only for a short amount of time during which the sprays could be beneficial. Accident management guidance should recognize that drywell shell failure due to debris attack can be prevented if drywell sprays are initiated before RPV breach and the drywell floor is filled with water to quench the debris. Based on the sensitivity evaluation using MAAP calculations for DAEC specific severe accidents, the following sequence types would not allow or call for spray initiation before RPV breach: · Class ID · Class IIIC · Class V On the other hand, the following classes appear to marginally satisfy the DWSI curve prior to RPV breach: · Class IA/IC · Class IV

Table 4.9-25

SUMMARY TABLE OF DETERMINISTIC CALCULATIONS (MAAP) SENSITIVITIES

Sensitivity Variable Investigated	Phenomena Affected	Accident Management Result/Conclusion
	Drywell Spray Usage Under Severe Accident Conditions (Section 4.9.3.11) (con't)	For example, the DAEC EOPs appear to allow drywell sprays to be initiated during in-vessel core melt progression for accident Class IA. In other words, the calculated deterministic conditions for postulated Class IA/IB accidents are such that both the symptoms to initiate and the parameters to allow initiation are all satisfied prior to RPV breach. This is judged to be important severe accident consideration because of the need to assure cooing of debris on the drywell floor and prevent shell failure. Nevertheless, small errors in instruments or instrument failures can lead the operating staff to delay or prevent initiation of the drywell sprays until it is too late. Therefore, additional margin or incentive to spraying the drywell during core melt progression is considered desirable as part of accident management and an outcome of the IPE. It is recognized that regardless of the flexibility offered by the DWSI curve, additional changes to the EOPs may also be required to remove any ambiguity regarding the diversion of injection sources away from the RPV when adequate core cooling is not assured, i.e., low reactor water level.
	Containment Venting	Additional investigations as part of the DAEC accident management program could include: · Treatment of containment venting: - How to control containment pressure after vent initiation (i.e., is there a vent pressure control band?) - What influence drywell temperature has on vent pressure control band.

5. UTILITY PARTICIPATION AND INTERNAL REVIEW TEAM

5.1 IPE PROGRAM ORGANIZATION

The organizational structure for the IPE program is given by Figure 5.1-1. The personnel that represent the core of the project are contained in the Safety Analysis Group. Personnel in the Safety Analysis Group have given the project its guidance and coordination, as well as providing a great deal of the analysis. This group also supplied the framework and manpower to integrate the study results for the report preparation and submittal. The project was structured to accomplish two major objectives. The first objective has been to produce an IPE that accurately reflected the as-built plant and its operation. The second objective has been to involve the remainder of the plant staff, as much as feasible, in the process and to introduce them to the concepts of PRA methodology and severe accident issues. To accomplish these objectives the following concepts/approaches were used.

1. The system notebook and fault trees were prepared by personnel in the Systems Engineering Group who had been assigned responsibility for individual systems.

2. An independent internal review committee was established. Members of this committee were selected from various departments throughout the plant.

3. Training for PRA methods and Fault tree analysis was given to the Systems Engineers and the independent review committee.

4. An external review process was performed to insure that our methods were reasonable and within the bounds of acceptable industry practices.

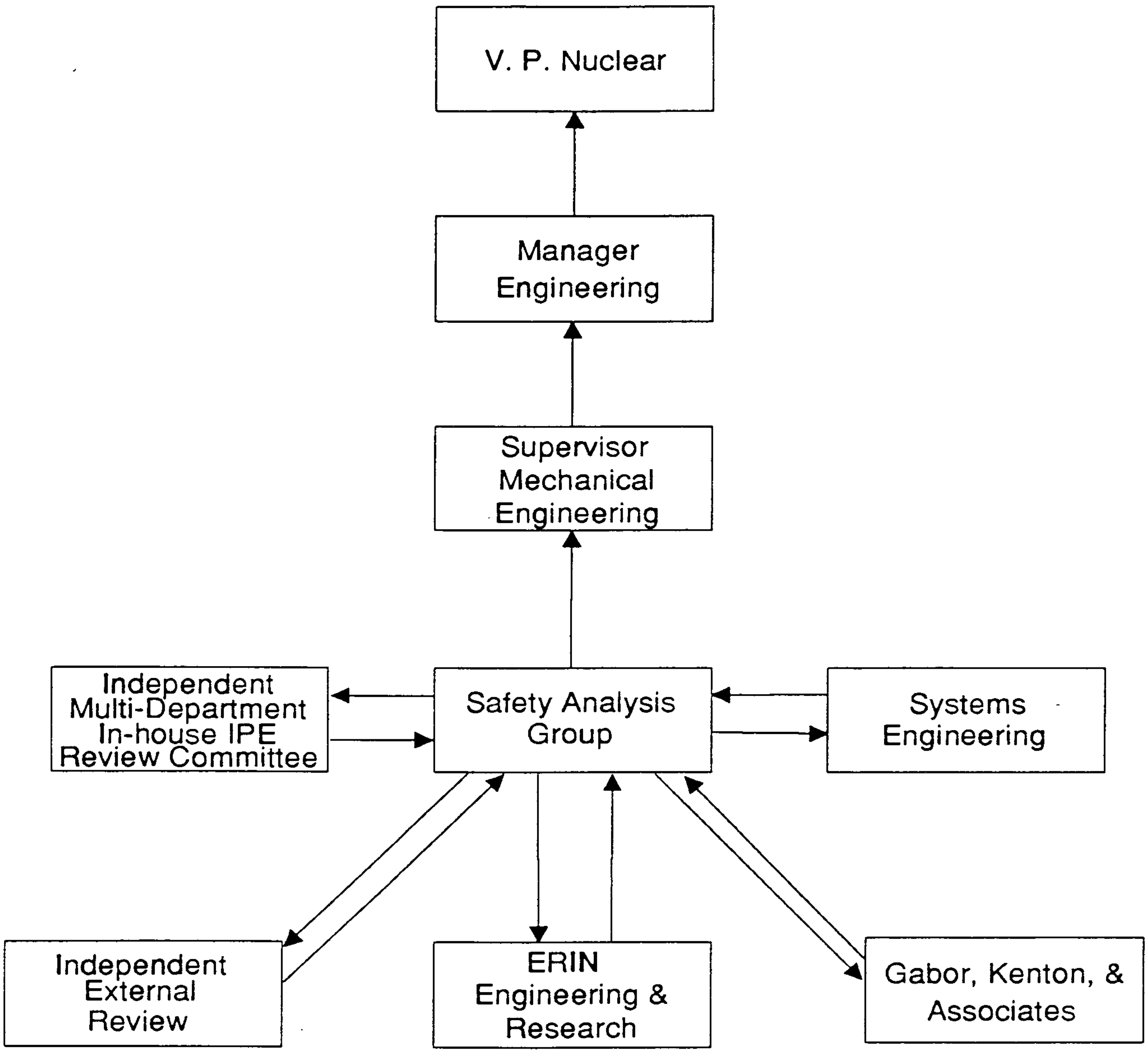


Figure 5.1-1

IPE Organizational Structure

5. Work performed by consultants was closely controlled and was done in such a way that technology transfer to DAEC was maximized.

6. The PRA was quantified completely by DAEC staff.

As previously indicated the Safety Analysis group supplied the core manpower for coordination and continuity of the project as well as for the integration of the analyses. Qualification and experience level of this group are represented in brief by the following items:

1. One person (who had worked in the group during a major portion of the production of the analysis) has an SRO license on DAEC. One other person has an SRO certification on a similar BWR. Another person has multiple licenses and operating experience from PWR plants.

2. All personnel in the group have had previous experience at other companies. This experience has included work at INPO, NRC, National Labs, International Labs, Consultants, NSSS Vendors (including G.E.), other power plants and the U.S. Nuclear Navy.

3. Years of experience of personnel in the group range from three to twenty with an average of about 13. All personnel presently in the group are degreed engineers with three people having advanced engineering degrees and one person having a P.E. license. The present manning level in the group is four engineers.

4. Personnel in the group have worked on, and performed formal reviews of, other major PRAs as well as having prepared several publications on subjects related to PRA topics.

5.2 COMPOSITION OF INDEPENDENT REVIEW TEAM

5.2.1 In-House Review

An independent in-house review committee was created to review the information contained in the initial preparation of the system notebooks. Each system notebook and the associated fault trees were prepared by the cognizant System's Engineer for that system. When the system notebooks were completed, personnel in the Safety Analysis group insured that the notebooks and modeling were done in a consistent fashion. Meetings of the Review Committee were then convened and each System's Engineer made presentations of his system(s) to the Committee. Comments were generated, recorded and resolved. In this fashion the project insured that the plant design and operation were realistically accounted for at the earliest stages of the project. This also meant that a wide cross section of plant personnel were exposed to the PRA process and severe accident considerations.

The In-House Review Committee consisted of approximately 15 people from the following DAEC organizations:

1. Engineering (Safety Analysis Group, Systems Engineering, and Component Engineering)

2. Technical Support

3. Emergency Planning

4. Training Center / Simulator Instructor Personnel

5. Operations

6. Licensing

A continuous in-house review was being performed by the Safety Analysis Group as various parts of the project were performed. These reviews were performed, as much as was feasible, by personnel not directly associated with production of the part they were responsible for reviewing. Finally, a review of the completed IPE report was performed by personnel from the same organizations listed above.

5.2.2 External Review

Primarily two consultant companies (see Figure 5.1-1) were used to provide specific expertise and advice to the project. The consultants provided work that was, as much as was feasible, reviewed independently within their organizations and was then reviewed and accepted by the Safety Analysis Group. These companies, in twin, provided review of the DAEC IPE project and injected their expertise and knowledge gained from other projects they had been or were working on.

In addition to the two primary consultant companies, two completely independent consultants were hired to perform a review of the DAEC IPE final project submittal. James H. Moody of Moody Consulting and Dr. G. W. Parry of Halliburton NUS performed this review. The purpose of this review was to help insure that the DAEC IPE approach and methods were consistent with industry standards and that the IPE submittal met the intent of Generic Letter 88-20.

5.3 AREAS OF REVIEW AND MAJOR COMMENTS

5.3.1 Reviews During Project Development

To insure that the actual plant configuration was reflected in the IPE modeling process two key approaches were taken. The first was to utilize personnel from Systems Engineering to perform the systems' notebooks and fault tree models for the systems they had cognizance over. The second was to create a multidisciplined review committee to perform an independent review.

The systems engineers and review committee members were instructed in PRA methodology and IPE requirements. When a systems notebook was completed the cognizant systems engineer would make a presentation to the review committee. Many of the questions generated could be resolved during the presentations. For the most part these questions or comments dealt with the models and success criteria etc.. An example of these comments is "Although the FSAR says you need 2 RHR pumps for shutdown cooling, in reality you only need one". In all cases the comments were resolved and models were changed as needed. In the process of performing the IPE the cognizant systems engineers updated and changed the systems notebooks and modeling as the plant underwent significant changes.

This combined with the ongoing reviews by the project team and their consultants insured the models to be current through the last outage completed in April of 1992. By performing the modeling and the reviews in this way the project was able to obtain realistic and accurate models very early in the effort as well as to involve a wide cross section of the plant staff in the process.

This discussion was intended to give an idea as to how the ongoing review was performed and the nature of the comments that were generated . The next section will discuss the specific comments that were generated during the final formal review that was conducted at the completion of the project. Either personnel who were directly involved with the ongoing review process or personnel from the same organizations were given a draft of the final IPE report to review. In addition, two independent consultants from outside the company as well as project team members were given the document to review and comment on. These comments have been collected into general groups and are listed in section 5.3.2.

5.3.2 Comments Made During Final Review

As indicated in the previous section, all comments during the final review have been collected into groups based on some aspects they had in common. The review comments are presented in the left hand column, and are addressed in the right hand column. (It should be noted that page numbers identified in the comments referred to the draft document, and may not reflect the final submittal.)

5.3.2.1 Editorial

Many of the comments generated were editorial in nature. These have all been corrected in this document, and are not listed individually.

5.3.2.2 IPE Philosophy

This section itemizes the comments that concerned the manner in which the IPE analysis was conducted. These types of observations were included to facilitate the understanding of the IPE methods used by IELP.

1. Section 3.2.1 should be less formal in the description of the process applied. Key ideas that need to be reflected are as follows: Initial generic BWR trees were created to provide a starting point for modeling. Analysis group took a first cut at incorporating plant unique features. System Engineers were trained on PRA and Fault Tree techniques. System Engineers were directed to review and modify the trees to explicitly model the DAEC system. These revised models were presented to IPE review boards and validated. The selection of this process accomplished several objectives. First, the creation of basis models gave basis consistency to the structure of Fault trees. Secondly, the plant knowledge of the PRA group, and the System Engineers, as well as the review board, could be factored directly into the system models.	This is a good summary of the methods and organizations used to accomplish the steps listed in Section 3.2.1.

2.	Many pages were observed to have "engineering judgement" as selection criteria, where actual criteria should be listed. (Engineering judgement may select the criteria, but screening for other features must use the criteria.)	In this submittal, the term "engineering judgement" is defined to indicate those decisions that were based on experience with past and ongoing Probabilistic Risk Assessments.
3.	At the recent ANS Executive Conference on IPEs, one of the NRC staff praised the use by some utilities of importance measures to analyze the results. We normally do this and find it very informative, particularly when supplemented by sensitivity analyses.	Because of the restrictive schedule and resources, the DAEC PRA project has been limited as to the amount of peripheral analysis that can be done. The project has been very careful to try and insure that all GL 88-20 requirements have been addressed and has restricted itself to that endeavor. In general, therefore, importance measures studies were not used as they were seen as beyond the scope of this effort. An exception to this is the HRA. Experience and judgment were utilized initially to determine important human errors. Importance studies performed later confirmed the judgement that was used.

4.	We have a known problem (with plan to correct) with the EDG PSA system model. What if any, impact does this problem have on the IPE? As we develop the model enhancements, is there a link to the IPE we need to cover? If our model shows we can't carry sufficient loads, does this affect your fault tree results or is it modeled in the IPE by some other means?	Any plant modifications that have a potential to change this Risk Analysis should be reviewed to determine those changes. However, there are no programatic requirements or procedures in place at DAEC that direct this review. It is the intent that the Systems Engineers who are responsible for IPE System Notebooks and are cognizant of all possible modifications to their systems will alert the Safety Analysis group of any need to change the fault tree models. Members of the Safety Analysis group actively participate in the EOP Working Group, and should be aware of changes to procedures that will affect the models.
5.	AC Power reliability is twice as good as our company goal of 0.010. One must change. Also RHR unavailability is listed as 0.006 while the goal is 0.01. RHRSW is probably over - conservative. RWS appears extremely low if the number is taken with respect to 4 trains. If it is two trains (I believe this is the case) it is	The unavailabilities listed in Table 3.3-4 represent the observed values at DAEC. These may be either above or below the company goal, as this is a "best estimate" rather than a "worst case" analysis. The fault trees explicitly model the train unavailabilities

	probably accurate, but the model should not take credit for 4 pumps if that is the case.	for systems that have more than one component (i.e. pump) in each train.
6.	Are the System Notebooks or this submittal going to be MDL Documents? Suggest at least have a copy in the library for reference.	This submittal is not scheduled to become part of the MDL. A copy will, however, be located in the library.
7.	How were transients listed in UFSAR excluded from list. For Example, I don't see any of the Reactivity Increasing transients listed as initiators?	Part of the determination of the sequences to analyze is to group all of the transients listed in the UFSAR by their plant response. The initiating event frequencies were developed with respect to these groups. Reactivity Increasing transients would respond very similar to a Turbine Trip with Bypass. On high reactor power (120%), an automatic SCRAM signal would be generated and the transient would proceed along the same event paths.
8.	The mission time is not addressed in the summary section and probably should be.	The mission time for the accidents analyzed is 24 hours.

9.	If we are going to use the Brunswick PRA as the source for our large LOCA initiator frequency, the IPE needs a statement on similarity and NRC acceptance of the Brunswick PRA.	The Initiating Event Analysis (a Tier II document) states the basis for the Large LOCA event frequency. This document states, "Numerous studies and PRAs have analyzed the potential for LOCAs. The frequency estimates from each study have, for the most part, remained comparatively similar (i.e., within the same order of magnitude)...." The value that we used was higher than the NUREG-1150 studies (due to the inclusion of inadvertent ADS as a possible contributor), and similar to other studies.
10.	Suggestion: Add a table at the end of the Acronyms with a list of all the functional event codes. (eg. C, M, P, D, Q, U, X, V, etc.)	These designators are used only in the quantification of the event trees, and are defined in each of the event tree descriptions.

5.3.2.3 Clarifications

This section includes the comments concerning the presentation or wording of the information in the submittal.

	Comment	Response
1.	Page 2-8, the words "with rare exceptions, no passive component failure (e.g. pipe failure) were included". Include the words LOW ENERGY before pipe failure. Is there a better example in that we did consider the high energy? Perhaps "In general" would be better? I think we tried to consider passive device failures in terms of things that would have high consequences or effect multiple systems or safety functions.	The DAEC PRA did consider some pipe breaks for LOCA and ISLOCA. These were high energy breaks either from high energy pipes or from low energy pipes that were pressurized by high energy lines. Feed and Steam line breaks (also high energy) were also considered. The only low energy line break to be considered was SLC suction. Changes will be made to the wording in Section 2 to clarify what was modeled.
2.	Page 3-228 clarify the statement that CS break detection is not modeled. Is this saying that loop select is not modeled, or that the ability to detect and mitigate CS breaks (ISLOCA type events) is not credited?	These are handled explicitly in the ISLOCA analysis.
3.	On pages 2-7 and 8, the report speaks about "major components" in fault trees and appears to speak about equipment. Are the Human Actions included?	Human Actions are explicitly included in the fault trees models.

4. Why was 3.0e-11 selected as the cutoff? The criteria needs to be stated.

This truncation limit was selected because it is the lowest value that would allow an efficient solution of the models. It should be noted that this value is approximately 5 orders of magnitude below the total CDF for DAEC and no truncation of significant terms is expected at this level. Historically, Nuclear Plant PRAs used truncation values of about 10^{-9}; and the guidance in GL 88-20 requests that a value of lower than 10^{-7} be used.

5. Using the numbers in the report,

   LOOP = .117/yr

   λ(battery) = 2×10^{-6}

   CCF(battery) = 0.04

   gives a contribution to the long term (non-recoverable) station blackout of 3.3×10^{-6}/yr, if the battery failure rate is interpreted as a standby failure rate with an effective test interval of one month. Have you treated the battery failure rate as an operating rate?

In this case the battery failure is treated as an operating failure rate. The standby rate is included in the Loss of Division II DC initiating event, and its value is 1×10^{-6}. It would be double counting to also include this failure here since there is no difference between losing all DC with or without AC available.

6.	Page 2-11: Station Blackout SER approves DAEC battery coping based on 1D2 only for 4 hours. Reference to 1D1 > 4 hours or 1D2 > 12 hours is not valid for SBO. Where did these times come from? We have taken the position that 1D1 is also valid for 4 hours, but NRC has not accepted that position.	The intent of the SBO analysis is to prove that DAEC can survive a station blackout for at least 4 hours. It applies all known conservatisms to the battery systems to make that determination. The IPE, however, is supposed to represent the best estimate of the progression of events, including operator actions to prolong the life of the station batteries by removing loads.
7.	On page 3-31, 0-1 hour time phase includes flow from feedwater, condensate, and low pressure ECCS. How is this possible in a Blackout.	It was determined that core damage would not occur prior to the end of the 1 hour time phase. It is therefore possible to prevent core damage with these systems once power has been restored. The wording in this section was adjusted to clarify this point.
8.	Based on the importance to CDF of offsite power, the offsite power system should be described in section 1.2.1.	It is true that the loss of offsite power is significant. However, the offsite power system is not explicitly modeled as a fault tree. Methodology supplied by NUMARC 87-00 is utilized to develop a loss of offsite power (LOOP) frequency. This is the same

methodology used to generate the LOOP frequency in the DAEC Station Blackout (SBO).

9. The ATWS trees are not structured to make the dependencies between Human Interactions explicit. As examples, it's not clear to me, why, on the ATWS trees, you need three branch points to discuss level control, unless they are to do with different time phases. If this is so I would have ordered the events chronologically, i.e. L(Controlled) first, to make sure any probabilistic dependencies could be handled. Also with successful HPCI, the need to inhibit ADS is dependent on whether they lower level. I would suggest a more detailed description of ATWS scenarios to help understand the trees.

This section was intended to give a brief description of the event trees. The Tier II documents contain the detailed descriptions of these sequences.

The Level/Power control events are handled chronologically. Level must be prevented from going too high in the short term to prevent power excursions. This is an immediate operator action on entering the ATWS scenario ("dial feedwater controller down to its minimum..."). Next, water must be lowered to control power, but it must not be lowered too far (below -30") to prevent core damage. These dependencies are handled in the node sequencing and the split fraction quantifications.

10.	Page 3-380, ATWS model: Is there any distinction between a partial SCRAM failure (with return to criticality) vs. a full blown ATWS in the model?	There is no distinction made in the model. The operator response times for the full ATWS is much more limiting than the partial ATWS, especially for the initiation of SLC. The failure of early SLC initiation is a major contributor the CDF due to loss of reactivity control.
11.	On page 3-38, how can ARI be successful only if the Recirculation Pump Trip is successful? There is a RPT signal generated by ARI.	The response time necessary for ATWS-RPT in a full ATWS is very short (less than 1 minute), while the operators will not perform the ARI initiation until 60 seconds after the ATWS is detected. The ATSW-RPT signal in this event tree is generated by the RPV High Pressure (1140 psig) signal which occurs almost immediately after the turbine trip.
12.	On page 3-39, does "Feedwater Runback" mean "reducing feedwater for power/level control."	Yes. This is accomplished by the operators dialing the feedwater controller to its minimum setting (+158").
13.	Reference to failure rates DAEC/NPRDS may be weak.	The NPRDS failure rates were not explicitly used in the DAEC IPE. These values were used to compare our reported equipment reliability with that reported by other

		plants. This information comparison was used to justify the use of generic failure rates for components in the DAEC specific fault tree analysis.
14.	Table 3.3.2-1 may require additional explanation. If this is saying that the HPCI pump and turbine unit, and same for RCIC has a zero failure rate at DAEC and 10E-5 rate in the industry I find it very hard to believe. DAEC has had failures such as the overspeed trip tappet on RCIC, and HPCI controller problems, as I recall. These systems are designed to be about 1.0E-2 reliable.	See above. Table 3.3-4 contains the DAEC specific unavailabilities observed for HPCI/RCIC.
15.	I have some other concerns about the HRA. For example, all the HEPs (Type C) have been treated as if they were time limited. It's not clear to me that all of the human interactions necessarily are, e.g., inhibiting ADS for successful HPI, depressurizing on HCTL if boron is injected and level is lowered. The HEPs don't seem to be out of line with many other IPEs, although the EPRI	To a greater or lesser extent all Type C HEPs are time limited. This includes the examples cited. It is the intent of the study to use RMIEP methodology as a first choice where appropriate. As explained in Section 3.3.3.2 certain DAEC human actions did not meet the same conditions that would allow for using RMIEP values and other more appropriate methods were

	method appears in most cases to give lower HEPs than the RMIEP method, particularly as the justification given is not a particularly strong one. When using the EPRI method, it's necessary to say where the median response times in Table 3.3.2 come from. It would also help to briefly describe the methods rather than incorporate them by reference only.	chosen. The reasons for not choosing RMIEP methods in certain cases are considered compelling and justifiable. The details of the median response times used in the EPRI method are recorded in the Tier 2 documents. Our interpretation of the submittal requirements indicated no need for reporting these details. As with the rest of the report, it is a judgmental decision as to the extent of what should be included in the submittal. We have made every effort to comply with the submittal guidance and feel we have made a reasonable attempt to supply all that is required. Although it would probably be helpful to describe the methods, to limit the size of the submittal no additional description is planned.
16.	The HRA is mentioned on page 1-15, but no explanation of how it factors into the IPE is given. Some explanation of the HRA impact on the Level 1 analysis is necessary.	Additional analysis was performed and the effects of important human actions are described in Section 3.4.1. The information in 3.4.1 is followed up in Section 6.

17. A supporting statement for "importance" level is needed in 3.3.3. What criteria was used to define "important."

Initially, the importance was determined from results of PRAs of similar BWRs. This was then confirmed by performing an importance analysis on the DAEC Level I results. This analysis showed that all screened HRA values had low "importance" (Fussel-Vessely, Birnbaum, Risk Achievement, etc.).

18. Were simulator weekly critiques reviewed to identify failure rates? Do failure rates of 1.0 mean that it always occurs?

Simulator weekly critiques were not reviewed as a part of this analysis. This may be a good source of human reliability data. A sample of the operators, trainers, and EOP writers were interviewed, and a select group of simulator scenarios were observed by the HRA development team.

A value of 1.0 indicates that the specified action is not performed within the time frame assumed in the model.

19. Loss of Instrument Air initiator of 8E-3 does not seem realistic. My expectation would be more equivalent to 1 per year.

The Loss of Instrument Air initiator frequency was initially developed using plant data for the 1974-1990 time frame. Since then, the Feedwater/Condensate systems have been modified to remove the dependence on a short term loss of air. The frequency used in this report reflects all industry Loss of Air events that lasted longer than 1 hour.

20. On failure rate Table 3.3.2-1, why is MSIV = 0? SLC -- We had a valve mispositioning event in mid-80's for complete loss of SLC. MOV failure rate seems too low.

These values came from a CFAR report from NPRDS. They were used only as a basis for using generic failure rates, and were not used for the specific values used in the analysis.

21. ESW planned LCO appears optimistic. There were over 100 hours in planned LCOs in '90 and '91.

Your observation seems consistent with the information in Table 3.3-4, System Unavailabilities. For this number we used 16 hours of PM unavailability and 72 hours of CM unavailability, for a total of 88 hours, over the same 1990 - 1991 period.

22. SCRAM rate is high. It represents an anomalous period of performance.

The frequency used in the analysis (3.6 per year) is the average number of SCRAMs per year for the operating life of DAEC (through 1990), with the exception of the first year. 20 SCRAMs that occurred in the first year of

		operation were excluded from the data. The table entry that you are referring to was removed from this report.
23.	Well Water reliability seems extremely optimistic depending on how train is defined.	There are no trains, in the classic sense, for the Well Water system. This value is based on the average observed unavailability of a well water pump.
24.	Page 3-325 - LPCI injection initiation on triple low should use the 18.6 inch number. The 64.5 number is the conditions under which the device is calibrated and includes offset for accident conditions to assure operation given drywell heating, etc.	The 64.5 number is correct for this analysis since it is the value that the instrument is calibrated to. This setpoint is used so that in the worst case scenario, triple low will occur above 18.6". This report evaluates "best estimate" scenarios.
25.	Consider an early section in 3.2.1, or under the ECCS system discussions, that identifies that initiation logic and circuitry was modeled separately in 3.2.1.10, and that failure modes are not (generally) included in the system dependency matrices.	The automatic actuation signal dependencies are included in each of the system dependency matrices, however these are not propagated through to the dependencies of the instruments. This is discussed, as you noted, separately in Section 3.2.1.10 on Instrumentation.

26.	It appears from the list on page 1-16 that instrumentation/logic circuit failure beta factors were not used for the front end analysis.	The instrumentation/logic common cause failures are dominated by mis-calibration errors. These errors are modeled in the fault trees.
27.	In Section 3.4.3 consider pointing out that DAEC is a LPCI loop select plant and that normal shutdown cooling mode of the RHR system does not meet later plant requirements for safety grade redundancy in this function. Certain components, such as the shutdown cooling recirc loop suction valves, are single point vulnerabilities. Other normal and emergency means of decay heat removal are available. One such mode, evaluated under Appendix R, is alternate shutdown cooling. In this mode, SRVs are maintained open with low pressure systems providing flow to the vessel. At reactor pressures below 400 psig, the open SRVs provide subcooled or saturated water flow to the suppression pool which in turn is cooled by either loop of RHR.	This discussion would be more appropriate in the discussion of the RHR system, Section 3.2.13. The Decay Heat Removal analysis in Section 3.4.3 does not credit the use of Shutdown Cooling mode of RHR (nor does the CDF analysis).

28.	Section 3.2.1.4.c - (page 3-221) states that each core spray loop is designed for full capacity. This is misleading and should not be included. No analysis exists (appendix K and 50.46 assumptions) that can demonstrate one core spray pump is capable of maintaining clad temp under 2200. The analysis and design basis is for Core Spray system, independent of LPCI to provide core cooling, or, for the case of one electrical division failure, for two RHR and 1 core spray pump to meet the ECCS criteria. If MAAP can demonstrate 1 pump is enough it can be so noted here. However, it should not be so noted as a design basis of a Core Spray subsystem (since it wasn't for our product line).	The confusion about the design capability of the Core Spray system stems from the various documents available at DAEC (UFSAR, System Descriptions, Training Manuals). While it is true that there is no licensing safety analysis for DAEC that indicates that a single CS pump is capable of providing adequate core cooling for the DBA LOCA, this IPE analysis does use that assumption. This decision is based on MAAP runs and heat balance evaluations. In addition, the scenarios evaluated here do not have as severe of a level transient as the DBA LOCA analyzed in the UFSAR.
29.	Section 3.2.1.4.4 - delete or clarify the statement that each CS pump can provide adequate core cooling flow - i.e., if this is based on MAAP runs, so indicate. No Appendix K analysis I know of can support this.	See above.

30. Page 3-9. FEEDWATER CONDENSATE - paragraph 2 - "if the operators do not take manual control of feedwater level controller in time AND THE FEEDWATER CONTROL VALVE RESPONSE IS UNSUCCESSFUL IN PROMPTLY REDUCING FEEDWATER FLOW" add the text in upper case noted here.

The design of the Feedwater Control system is such that the operators must dial down the feedwater controller to its minimum normal setting immediately following a reactor trip in order to prevent level from reaching the 211" Feedwater trip setting. The trees are modeled so that the failure of this action requires the operators to restart the feedwater system.

31. Page 3-43 - How did we treat feedwater runback? With recirc pumps tripped, and normal level control in effect the power level (above 211") would require the failure of the high level trip, as well as the feedwater controller failure, and failure of operator action. We need to make sure that the split fraction here causes high pressure vessel failure only when level control AND high level trip functions have failed. If you consider the likely chugging of level and power, it is actually less likely, in my opinion, that feedwater can be maintained because the high level trip is likely to occur - again this is judgement based without any supporting models.

This node is considered in ATWS cases in which the MSIVs have isolated. In these situations, we conservatively assume that RPV level must be maintained below 158" in order to adequately control the pressure increase at the onset of the accident.

32. Page 3-24, Section 3.1.2.7.1 - "long term loss of air causes feed reg valve lock". May wish to note in this section that 1992 modifications provided dedicated accumulators and eliminated the condensate demineralizer design features that would cause condensate trip on loss of air - probably doesn't change the split fraction, but may want to verify that the new configuration is reflected in the tree.

This is reflected in the models. The Loss of Air transient models only those that last longer than the Feedwater Regulating Accumulators can provide backup (about 1 hour). In these cases, the loss of Feedwater causes the loss of Condensate.

33. On page 3-21, why do we lose the main condenser on a loss of RWS?

The loss of makeup to the circ water pits has a high likelihood of causing a loss of the circulating water system function. This, in turn, causes a loss of condenser vacuum. Note that this is recoverable by following AOP.

34. In section 3.1.2.10.2 for the Main Condenser, if the LOCA is within the capability of HPCI, the MSIVs should be open.

While this is essentially correct, it is difficult to model this phenomena explicitly within our event tree structure. The way that it is modeled is definitely conservative.

35.	Page 3-228 - under assumptions we say that room cooling may not be required - we should be specific on which dependencies we modeled - did we include it or not as a failure mode. (3-247 says it is not required).	"Room cooling is not necessary for successful CS system operation." This statement will be made in lieu of the one that exists.
36.	Page 3-288 - Note for clarity that the Auxiliary Heating Boiler was not modeled (credited) as a source of HPCI and RCIC motive steam.	This is correct.
37.	Page 3-13: High steam flow signals (from how many steam lines) lead to loss of PCS?	The analysis is independent of the number of steam line high flow signals.
38.	Page 3-5: In LOCA, coolant injected into the vessel from Suppression Pool only returns to the Torus after filling the bottom of the containment (including sumps) to the lip of the downcomers. Thus it is not correct that all water returns to the Torus unless that level has been reached using water from other injection sources.	All water that does not remain in the Drywell returns to the Torus. The amount retained in the Drywell is small compared to the total.

39.	CRDs are listed in Table 3.1.2-3 as a system used to control RPV inventory at high pressure, but no verbiage is provided, as is for the rest of the items.	No detailed description is given since this analysis does not credit the use of CRD as a water source for core damage prevention.
40.	In section 3.2.1.5 there is no credit taken for the diesel fire pump as an external injection source.	Due to the uncertainties involved in the lineup time and system pressure-flow characteristics of the lineup, this system was not credited as a water source for core damage prevention. Further analysis of this assumption is warranted, however the approach in this submittal is conservative.
41.	I thought total relief capacity was supposed to be 110%. On page 3-195, it totals 87.1%.	The total flow capacity of the Safety Relief Valves and Code Safeties, at normal operating pressures, is correctly indicated in this section. You may be adding the 25% relief capacity of the Turbine Bypass Valves in your estimation.
42.	Page 3-28: Why is the base case used for venting (during LBLOCA). Division II DC is required to open CV-4357.	This page is in the LBLOCA section. Your copy may have been missing the page with the section heading on it.

43.	Page 3-8: Vapor suppression. SRV tailpipe break. Is this one of the rare exceptions (p2-8) where a pipe failure is analyzed? If not, shouldn't be referenced, (or should be included as a form of LOCA)? How does containment overpressurization failure cause unavailability of low pressure injection systems? If loss of suppression pool volume from failure in Torus occurs before a drywell failure this is valid. Torus design Pressure < DW design pressure?	SRV tailpipe break is included in this model because it affects the mode of operation for vapor suppression. Containment overpressurization affects all injection systems. If there is a break in the drywell, or torus, the environmental conditions in the reactor building is uncertain. Also, if the break is energetic enough, the status of the containment penetrations is uncertain. To account for these uncertainties, no injection is credited in this analysis.
44.	Page 2-13: If containment failure pressure assumption is conservative, should state basis for this pressure and relationship to design pressure.	The containment failure pressure analysis is contained in Section 4.
45.	Page 4-59: Won't the block walls in the heater bays give before 0.25 psig? Also, Bechtel has estimated in calcs. that doors give (conservatively) at 1 psig.	Any break into the reactor building will result in pressures greater than 0.25 psid, therefore considering lower failure pressures would not impact the analysis.

	Comment	Response
46.	In section 3.1.2.6, SORV does not cause an automatic SCRAM. Operators are directed to manually SCRAM on high torus temperature of 110 F.	This is correct. The text has been changed to reflect this observation.
47.	On page 4-55, explain the leakage path through the railroad door? It doesn't seem to be included in the MAAP nodalization figures.	The doors that contain air locks are not modeled. These are assumed to have a higher failure pressure differential than the blowout panels.
48.	On Table 3.4.3-2, why is the ESW cross tie included as a success path? The spool piece is bolted to the wall and will take a long time to install.	In scenarios in which one diesel generator is unavailable and the opposite train of ESW is unavailable, it is reasonable to assume that the operators will make maximum use of the equipment available - i.e. the cross tie. In the quantification of the CDF, however, no credit was given for this action.
49.	The internal flooding analysis needs: a: a comparison to other core damage frequencies. b: some verbiage to explain in layman's terms why the frequency is so low (no pipes in rooms, etc.).	The internal flooding analysis was done by means of a screening process. In the screening process, walkdowns were done and an evaluation of the various areas of the plant were completed to determine the most

Is the 1E-7 CDF or initiator frequency? If other items at the same frequency are included in the IPE, flooding should be also for consistency.

likely and/or the highest consequence areas of the plant. By this means, only those portions of the plant that were believed to have the highest likelihood and/or consequences of flooding were chosen for more detailed evaluation. At this stage, the frequencies of initiation and/or propagation of the flood to a point where redundant trains of safety related equipment could be affected were evaluated. It was concluded that frequencies were already on the order of 1E-7 and that no further evaluation was needed. Analysis was therefore terminated at this point, since a core damage scenario resulting from flooding would be much less than 1E-7. Although there are other items included in the report that are of the same low magnitude, they are the artifacts of how the rest of the analysis was done and like the flooding scenarios they do not contribute significantly to the CDF. Because of the differences in the methodologies a direct comparison is not possible. As a point of interest, the internal flooding was done much the same as is intended for the majority of the rest of the external events due to be completed by the IPEEE project at a future date.

50.	Page 3-4: Accident sequence functional event for Containment Pressure Control/Vapor Suppression includes 3 CSFs for Containment Pressure Control, Suppression Pool Temp. Control, and Suppression Pool Level Control. This is not clear from Table 3.1.2-2 or form the discussions of CSFs on page 3-4 and page 3-5.	The table has been changed to make this relationship more explicit.
51.	Page 3-7: Where is the discussion that explains why Containment Temperature Control is not modeled?	The Level II analysis includes failures of the containment due to high temperatures.
52.	On page 1-13 of Section 1, there is no criteria listed for the selection of the 5 critical safety functions. You need a sentence to explain where they came from.	These critical safety functions are typical for any Probabilistic Risk Assessment of a nuclear power plant (NUREG/CR-2300 PRA Procedures Guide). They were tailored to the functions addressed in the DAEC EOPs.

53. Maybe I misunderstand the meaning of "insights relating to Peach Bottom." How are the seven items listed insights between DAEC and Peach Bottom. Are they similarities or differences?	Peach Bottom was the NUREG-1150 plant that most closely resembled DAEC. The list of insights from Peach Bottom were listed for comparative purposes.
54. Page 3-8: 1st paragraph, last sentence is confusing. "In the event tree quantification (for the SORV case?) the PCS reliability is the same as in the non-SORV cases, while the turbine bypass valves are assigned comparatively higher unavailabilities (than what?, in which case?)"	The TBVs are assigned higher unavailabilities than in the non-SORV case.

5.3.2.4 Comments on Specific Models

This section reports comments that deal with specific modeling assumptions and methods.

	Comment	Response
1.	Section 3.1.3.0: Reactor Pressure Control - revise (and verify as correct) to reflect both SRV and Code Safety Valves. Note that steam continuously flowing to the suppression pool OR THE CONTAINMENT DRYWELL are potential consequences.	This is correct, and is modeled explicitly in the vapor suppression fault tree.
2.	This is a general comment on Event Sequence Analysis. In a number of places, manually initiated operator actions are described which can lead to a different result. In calculating success/failure, are the operator, HFE, and <u>failure of the appropriate instruments</u> considered? For example: Operators rely on the Neutron Monitoring System, and Rod Position Information System to make a number of key path decisions in the EOPs (ATWS). Section 3.2.1.10, only addresses instrumentation which generates automatic	With the exception of hydrogen and oxygen concentration instrument failures, containment water level erratic, and RPV water level erratic failures (all in the Level II portion of the study), no instrument indication failures were explicitly modeled. For the most part these failures are embedded in the human reliability analyses, as for example, "crew fails to diagnose" etc.. In these cases the number of parameters that could have a potential affect on the outcome are so numerous that empirical models resulting from research by the

safety system actions.

NRC and EPRI are used. In such methodologies explicit failures are not modeled.

3. The HEP section does not discuss dependency between human interactions. This is particularly important for the ATWS case. The actions to initiate SLC, inhibit ADS, and control level are likely to be dependent , in my opinion, because of the way the procedures (if they follow the BWROG EPGs) are set up. It can be important to consider this dependency because it can give a different perspective on what is important. For example, in TTC I would have expected sequence 89 to be higher frequency than 87 since I think it highly unlikely that they don't inject Boron in 40 minutes, but don't control level, (dependency between successes decreases frequency) whereas the joint failure probabilities to initiate SLC and control level will be increased by dependencies.

The dependencies between various operator actions have been considered, and different values included where appropriate. In this particular case, the late SLC initiation and level/power control actions are explicitly coupled by the event tree structure - i.e. if level/power control is not successful, no credit is given for late SLC injection. Early SLC injection is independent, since it is directed by a different logical path along the EOP flow charts; and ADS lockout is an immediate action upon the entry of the ATWS EOP.

4.	A complete summary table of the HEPs used is not included.	It may be helpful to include a summary table, however all of that information is already included in the tables provided. This summary is part of the fault tree database, and is maintained on the PC.
5.	A list of references is needed for section 3.3.3 "Human Failure Data." Why wasn't EPRI-NP-6560-L used?	Other than showing in the text of the report, references, per se, have not been given in any of the sections. However, Table 3.3.3-1 and succeeding tables effectively give the references which were used in the generation of this part of the report. The one omission is EPRI-NP-6560-L which was referred to in the report as "EPRI Method" on Table 3.3.3-4c. Information will be added to clarify that EPRI-NP-6560-L was used.
6.	The size of the statistical sample should be included in 3.3.3.	The human action basic event failures were derived from methodology provided by the NRC and EPRI. Although some of the data derived in these reports came from statistical samples, the DAEC PRA essentially applied the methods without modification. As such, the human action basic events are the result of empirical

		models/equations and not the result of a statistically derived data base.
7.	Table 4.5-7, GV node description. DAEC has no H2 recombiners in the SGTS system that I know of.	This is a correct observation. The model does not take any credit for this equipment, so therefore does not need to be modified.
8.	On table 3.3.4-1, the Chillers would have the highest β factor at DAEC in my opinion. Why would RHRSW pumps have lower β than RHR? They have the additional common factor of poor water quality, are less well separated, and operated by less experienced operators, are located in areas more susceptible to cold temperature, and are less frequently observed. β for HPCI/RCIC seems high to me. While availability is worse than say Core Spray, they seem more diverse than A and B Core Spray. Why don't we have β for RWS, Electrical transformers, and electrical breakers?	Each of the values on the table were derived from generic common cause data sources. As such, plant specific environmental conditions and modes of operation may not be reflected. If DAEC specific common cause failure data were available, it would replace the values used in this analysis. The ß factor for RWS pumps was included in the system models; its value is 0.0375. Transformer and breaker common cause failures were not explicitly modeled, but assumed to be part of the major component failure.

9.	Table 3.2.1.1-1 - Review the DC bus dependencies for the ADS logic trains. As I recall, logic A is not power seeking to 1D23, although Bus B is power seeking to 1D13. (System is designed as a division 1 system since HPCI is Division 2, therefore both trains should be capable of Division 1 power. We deviate from NEDO 10139 in this design feature. The individual SRV pilot valves are correctly shown as power seeking between the two divisions). Should also note that PSV 4401 and 4407 are Low-Low set auto initiated in the column under auto actuation signal. Note: Table 3.2.3.1 appears to show this as I recall it.	ADS logic train A is not power seeking, this is correct. This does not change the quantification of CDF in this report, however, since the operators are always directed to prevent automatic initiation of ADS.
10.	Section 3.1.2.6.1: May need to revise description and event tree to reflect a Safety valve (vs. a safety relief) with the separate consequence of relief to drywell (vs. suppression pool).	The vapor suppression node model handles this difference, whether the difference is due to a broken SRV tailpipe or an inadvertent Code Safety operation.

11.	In section 3.1.2.7.2 for the SRVs, the drywell nitrogen system is an independent system.	This is a correct observation. Loss of Instrument Air leads to a Group 3 Isolation, which affects containment nitrogen.
12.	On page 3-42, why is feedwater runback addressed when feedwater is unavailable.	This is included because it is different than the base ATWS case.
13.	Table 3.2.2-1 - Condenser Dependency Matrix - Auto actuation signal of low turbine pressure is probably meaning the low steam line pressure in run signal. Check status of modeling and the plant configuration on high steam line radiation group 1 isolation. Earlier text (associated with recovering MSLs says it may not be recoverable given high rad - this is true whether it is administratively prohibited by EOPs or by this interlock, so I wouldn't change that language. Under loss of ISA, need to note that Nitrogen is the supply in question for the inboard valves air for the outboard.	These are good observations. The models correctly address each of these points.

14.	In section 3.1.2.7.2 for condensate, there is a procedure for manual positioning of the startup reg valve.	The current model does not take credit for this manual action.
15.	In section 3.2.1.7, why are the Condensate Demineralizer block valves not addressed as a failure mode for this system.	Flow blockage by the Condensate Demineralizer has been included in the model as an undeveloped event. It may be recoverable, so blockage does not always indicate failure of the condensate system.
16.	ESW (Table 3.2.1.6-1) should be reviewed for DC dependencies. Specifically, as I recall the start logic takes a signal from the EGD start logic which is essential DC. I don't recall whether the breakers are completely AC on their control loop out of these pumps.	The EDG start signal has been explicitly modeled in the ESW fault tree.
17.	In section 3.1.2.7.2, how does instrument air affect Diesel water supply?	There is no affect. The valve failure is in the desired position. This was not apparent at the initial quantification, so it was modeled as a separate case.
18.	Page 3-25, HPCI or RCIC - Don't understand the statement that the HPCI and RCIC turbine auxiliaries	There are several CV's in the HPCI and RCIC steam supply systems that rely on N_2 to reposition. These,

	lose back up gas and that valves need to rely on nitrogen alone. I don't believe that these systems have CV dependencies.	however are not required to operate for HPCI to perform its function during an accident, so there is no impact on CDF.
19.	Table 3.2.1.9-1 What about the loss of DC that can cause steam leak detection to activate? With the panalarm units with burnout protection now, don't we have a failure mode that isolates MOV 2238 and 2239? Consider also loss of power to the CST level system that causes the auto transfer to torus for HPCI and RCIC on loss of the bubbler.	These dependencies have not been included. My impression is that the loss of DC itself is more significant than the activations of these systems. Auto transfer of HPCI and RCIC to the Torus would not be considered a failure. Either source is acceptable.
20.	Page 3-289: I don't like this assumption since the pump can start on level, but the inject valve will not open unless the low level is present about 15 seconds into the start sequence. Therefore, most NORMAL transients (given momentary start signals with feedwater still on, of which we have had several) depend on this valve opening to protect HPCI.	The event trees do not consider success/failure of a system unless it is actually needed. If the feedwater has recovered level, then HPCI will not be needed to prevent core damage. Since it is possible for a run-time failure of feedwater to subsequently require the use of HPCI later in the sequence, the failure of the min-flow valve could have an impact. My impression is that an operator failure to secure the un-needed HPCI system

		prior to system damage would be required.
21.	Table 3.2.11-1: Consider the 1D2 steam leak detection dependencies for the RCIC system.	See #19
22.	In section 3.1.2.7.2, HPCI and RCIC have no nitrogen dependence.	See #18
23.	The discussion on the cooling of the essential switchgear rooms does not address the consequences of a control building isolation, either as designed or spuriously.	The need for control building cooling during the accident has been considered in the analysis.. Hot days are explicitly modeled in the AC power system. During the progression of the accident, Control Building isolation has also been considered. There must be a radiation release in order for isolation to be necessary, but the Level I analysis ends prior to core damage or radiation release. Control Building isolation during the accident is therefore not considered. It should be noted that EOPs handle loss of Control Building Ventilation events.

24.	Page 2-11 states that the vent is independent of AC power. Is this really true?	In the short term this is true. There is an accumulator on the system that provides for loss of air scenarios. As a backup, a gas bottle hookup is provided on the accumulator to allow repressurization in long term station blackout situations.
25.	Section 2.4.1: The statement "containment venting is independent of AC power" is too strong (instrument and service air is needed for normal venting, as is the AC solenoids both of which fail the valves closed). Suggest "Containment venting can be accomplished independent of AC power with the enhancements to the plant being installed in 1992. This would cover the Hard Vent.	See above
26.	In Section 3.4.3.1, should there be a discussion on Hard Vent specific design criteria? Our response to GL 89-16 requires us to base our specific design criteria for Hard Vent on the IPE. Does our Hard Vent design meet the required design criteria? Has there been specific criteria developed?	Several considerations from the IPE were included in the hard piped vent design. It was designed to be DC power controlled for operation during SBO events. Provisions were made to allow recharging of the pneumatic motive power to the valves during long term blackouts. It was also designed to be remotely operated

in case of adverse reactor building environmental conditions.

While the design meets the GL-89-16 requirements, other design specifications would have been helpful in reducing CDF, such as operation without any electrical power, but the reliability of the system as it exists meets the design intent. One additional note, the rupture disk pressure of 50 psid seems excessive to protect containment integrity. The DBA LOCA safety analysis shows peak Torus pressure to be much lower (~26 psig). A rupture disk designed for 30 psid would allow use of the vent over a greater range of containment conditions.

27.	In section 3.1.2.7.2 for containment vent, the accumulators are not "oversized", they are properly sized.	This statement was meant to indicate the long term air supply available for venting.

5.3.2.5 Comments on Reporting Requirements

This section contains the comments on the presentation of results, particularly the screening of sequences. The resolution of these comments required modification of several sections of the report, in addition to some re-analysis of the human failure rate importance.

1. Accident sequences that fall below the screening criteria due to the consideration of HEPs less than 1E-1 have not been reported.

By the definition (see Section 3.3.3.1 "Type E"- recovery actions) used in this report no human recovery actions were explicitly modeled. If, however, a broader definition is used for recovery actions to include all post accident human actions (as implied by the answers given to questions at the Ft. Worth meeting on GL 88-20), then several sequences must be considered. Section 3.4.1 now addresses this.

2. The discussion of the dominant sequence, TDC-27, needs to be clarified. In paragraph 6.2.1.1 the cause is stated to be common cause failure of the diesels. In paragraph 3.1.2.8.1 the loss of a DC bus is attributed to transformer failures, breaker trips, bus failures, etc., with

The cause stated is the common cause failure of the batteries. Due to common maintenance practices on both division batteries, the batteries are determined to be the leading contributor to CCF of both divisions.

an event frequency of 10^{-3}/yr. This suggests a CCF factor of 10^{-3} also, which is not discussed in Section 3.3.4. Since this is the dominant sequence, I strongly suggest that a detailed discussion be given, highlighting the contributors (cutsets).	A discussion of the CCF is added to Section 3.3.4. There is only one cutset in this sequence - Loss of Div I * CCF Loss of Div II.

6. PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

The purpose of this section is to present unique safety features and identify potential improvements and strategies. In Section 6.1 a list of descriptions of unique safety features is given. This list represents those features of DAEC that contribute to the overall plant safety. In Section 6.2 a discussion of significant sequences is given. Sequences that were identified by the application of GL 88-20 screening criteria (see Section 3.4.1) are included in this discussion. As part of this discussion, those factors that most adversely influence the event sequence are given. This section then identifies potential improvements and strategies.

6.1 UNIQUE SAFETY FEATURES

This section identifies significant and unique safety features at DAEC which help to minimize the risk due to severe accidents. These features limit the potential for challenges to core cooling and containment systems, and assure the capability of these systems to cope with transients and accidents in general. A list of safety features follows:

1. The feedwater pumps are motor driven instead of steam driven. This provides reliable operation even independent of the status of the MSIVs. In addition, the feedwater regulating valves have large accumulators installed. This requires an extended loss of instrument air prior to the failure (undetermined state) of these valves.

2. The offsite power switchyard has a highly reliable and diverse dual ring bus arrangement, minimizing the chance of a loss of offsite power. Essential loads are normally operated from the Startup transformer. This precludes the need for a "fast transfer" on loss of the main generator.

3. DAEC has a variety of redundant and independent service water systems, minimizing the impact of loss of any single system. In addition, the RHRSW, ESW, GSW, and Fire systems can be crosstied to the RHR inject lines in the event that alternate emergency low pressure systems are needed.

4. Most equipment located in the Reactor Building does not require HVAC for extended periods. Plant analyses have been performed to 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.

5. DAEC EOPs specify halting depressurization at 200 psig when turbine driven systems are available but low pressure injection systems are not.

6.2 POTENTIAL IMPROVEMENTS AND STRATEGIES

It is concluded that DAEC is a plant with a low risk of core damage and fission product release. It has only one sequence that meets the 1E-6 screening criteria and even this sequence is just at 1E-6. There are, therefore, no sequences or phenomena that are identified in this study that would make DAEC an outlier plant. As a result, no further changes would appear to be necessary.

By applying the screening criteria given by Generic Letter 88-20, five sequences were identified as contributing at least five percent to the core damage frequency (see Section 3.4.1). Of a total core damage frequency of 7.84×10^{-6} per year, these five sequences contribute 3.4×10^{-6} per year, or 47% of the total. The next five subsections describe those factors which most influence these event sequences and possible improvements and strategies for reducing their contribution. In the sixth subsection, Level II insights and possible improvements / strategies are discussed. In the final subsection, a discussion of significant human actions is given with insights and possible improvements / strategies.

6.2.1 <u>Loss of All 125V DC (TDC-27)</u>

6.2.1.1 Description

This sequence represents the loss of all 125 Volt DC power due to the common cause failure of both station batteries. While Abnormal Operating Procedures (AOP)s exist for the loss of either division of DC power individually, there are no AOPs that cover the situation for simultaneous loss of both divisions. Therefore, this sequence is conservatively assumed to lead directly to core melt.

This is the only sequence in the DAEC IPE study that meets more than one of the GL 88-20 screening criteria (functional sequence > 10^{-6} per year and contributes 5% or more to total CDF). The frequency of this sequence is 1.0×10^{-6}; 13% of the total CDF.

6.2.1.2 Factors that Adversely Influence this Sequence

The major factor that influences this sequence is that there are no procedures for the operators to follow that specifically cover this situation. There are AOPs that deal with events in which only one DC division is unavailable. Many of the strategies identified in these procedures can be adapted to cover the total loss cf DC case.

6.2.1.3 Potential Improvements and Strategies

The best solution to this sequence would be to develop an AOP or EOP to cover the total loss of 125V DC. Many of the control breakers in the plant that require DC control power are stored energy breakers. These can be locally closed in by operators. Other strategies would include using EHC panel power to manually jack open the TBVs in order to depressurize, taking local manual control of the RCIC system, and using portable generators to power essential DC loads.

6.2.2 Loss of Decay Heat Removal (TC-3)

6.2.2.1 Description

This sequence is initiated by a non-recoverable main condenser transient. All injection systems are available, however containment heat removal is not. This results in the Maximum Primary Containment Water Level Limit (MPCWLL) being reached several hours into the transient. Duane Arnold EOPs (EPGs, rev 4, also) instruct operators to terminate injection to the containment from sources outside of the containment irrespective of core coolability.

When this action is taken, primary containment temperatures and pressures will increase rapidly. The water in the suppression pool will heat to the point where HPCI and RCIC are no longer operable. Even though Core Spray and LPCI can pump saturated water from the pool, containment pressures will quickly rise to the point where the back pressure on the pilot valves of the SRVs will cause the relief valves to close, and render low pressure injection ineffective.

This sequence has a frequency of 7.9×10^{-7} per year, or a contribution of 10% to the total DAEC core damage frequency. It should be noted that the total contribution to CDF of sequences that are a result of a loss of containment heat removal is 24%.

6.2.2.2 Factors that Adversely Influence this Sequence

The major factor that causes this sequence to lead to core damage is the EOP "Defeat" that occurs on a drywell pressure of 53 psia and directs operators to terminate injection to the RPV from sources external to the drywell, irrespective of core cooling. It is likely that in this slow containment pressurization transient that the Emergency Response Organization will recommend against the termination of injection to the RPV, however this analysis does not take credit for any actions, or lack of actions, that are not present in existing procedures.

6.2.2.3 Potential Improvements and Strategies

While the EOP "Defeat" dealing with containment pressure reaching the MPCWLL is based on sound technical evaluations for the non-severe accident scenarios that were considered in EOP development, the long term containment heatup sequences evaluated in this study are adversely impacted by this instruction. It is prudent that these types of scenarios be evaluated for future inclusion in Accident Management Guidelines or the Emergency Operating Procedures.

6.2.3 ATWS with Failure of SLC (TTC-87)

6.2.3.1 Description

This sequence is initiated by a turbine trip with the main condenser initially available. There is successful recirculation pump trip and cycle of the SRVs to prevent an overpressurization of the RPV. Injection is successfully maintained by the feedwater system, and the operators control level within the proper limits.

The significant failure in this sequence is that Standby Liquid Control is not initiated in time to prevent the overpressurization of Primary Containment. This can be caused by either hardware failures in the SLC system, or operators delaying the injection of SLC for longer than 40 minutes. The quantification of this operator action is discussed in Section 3.3.2 Human Reliability Data.

The frequency of this sequence is 7.4×10^{-7} per year, or 9% of the total CDF.

6.2.3.2 Factors that Adversely Influence this Sequence

The main factors that influence this sequence are the failure to SCRAM and the failure to inject SLC in a timely manner. The SCRAM failures are dominated by common cause failures. The SLC hardware failures are also dominated by common cause. The key factor that can be affected by the results of this study is the failure of the operators to inject SLC in time to prevent containment, and subsequent core, failure.

6.2.3.3 Potential Improvements and Strategies

The most effective strategy against this type of situation is to maintain the heightened awareness of the operations staff of the importance of timely injection of Standby Liquid Control in ATWS scenarios.

6.2.4 Station Blackout for Greater than 15 Hours (TE-123)

6.2.4.1 Description

This sequence is initiated by a loss of offsite power and a failure of both emergency diesel generators. Recovery of AC power, offsite or emergency, is not accomplished within the time frame that HPCI injection can be maintained.

The frequency of this sequence is 5.7×10^{-7} per year, or 7% of the total CDF.

6.2.4.2 Factors that Adversely Influence this Sequence

This sequence demonstrates the importance of electric power for the safe shutdown of nuclear power plants. In this case, all measures are taken to provide core cooling, but the lack of power ultimately leads to core damage.

There is a possibility that core injection using the diesel fire pump crosstie through the RHRSW system into the reactor may provide additional makeup capability. If power is not recovered, however, containment pressure will continue to increase during the event due to the loss of decay heat removal. The hard piped vent relies on DC power, as does HPCI. When the batteries have been depleted by HPCI cperation, the containment vent will close on loss of DC. Also, the SRVs require DC power to remain open. Due to the high RPV pressure compared to the discharge of this lineup, the fire system is not considered as a credible form of core cooling.

6.2.4.3 Potential Improvements and Strategies

The recovery from the total loss of AC scenarios requires a reliable injection source that is not dependent on electric power. At DAEC, this would be the diesel fire pump (DFP). The lineup that would allow the DFP to inject into the reactor has never been tested. A strategy would be to test this lineup, without actually injecting fire water into the reactor, and to train the operators in arranging this alignment.

Additionally, there needs to be consideration for the depletion of the 125 VDC station batteries. DC reserve must be maintained in order to keep the containment and RPV at a low enough pressure for the firewater to RPV lineup to be successful.

6.2.5 Loss of Offsite Power with Early HPCI/RCIC Failure (TE-34)

6.2.5.1 Description

This sequence represents a scenario in which there is a loss of offsite power, subsequent failure of both diesel generators, and a failure of the steam driven high pressure injection systems. In this situation, the operators properly depressurize the RPV so that there is no high pressure ejection of the core from the vessel.

This sequence has a frequency of 3.8×10^{-7} per year, or 5% of the total CDF.

6.2.5.2 Factors that Adversely Influence this Sequence

This sequence demonstrates the importance of electric power for the safe shutdown of nuclear power plants. This case shows the importance of the steam driven injection systems; it is dominated by the common cause failure of both steam driven high pressure injection systems.

6.2.5.3 Potential Improvements and Strategies

There are no procedures or strategies associated with reducing the likelihood of this sequence. Improvements of the reliability of the Emergency Diesel Generators and the HPCI/RCIC systems would impact this sequence, however minimally. This type of sequence represents the limit of risk of core damage to the current generation of BWRs.

6.2.6 Level II Considerations

In this section, insights gained from the Level II portion of the study are addressed, and possible improvements / strategies are given. More detailed information is given in Table 4.8-6.

6.2.6.1 Containment Injection

Insight- There is a set of very low frequency severe accidents for which the containment may be at elevated pressures (i.e. above the containment vent pressure) and for which the EOPs would dictate that injection to the RPV be terminated when containment pressure exceeds MPCWLL. This can lead directly to core damage and a subsequent containment challenge.

Possible improvement / strategy- The prudency of terminating water injection to the containment under any circumstances for which core degradation may be aggravated should be evaluated.

6.2.6.2 Ex-vessel Recovery

Insight- The use of CS or DW spray in lieu of LPCI appears to be most useful in response to degraded core conditions.

Possible improvement / strategy- The prioritization of injection systems may be an action that could be included in future accident management development.

Insight- Initiation of DW sprays prior to RPV breach would preclude the debris attack and failure of the drywell shell. However, Class IIIC, Class V, and Class ID sequence types would not allow or call for spray initiation before RPV breach.

Possible improvement / strategy- Consideration of changes to EOPs to allow the use of Drywell Spray Initiation (DWSI) as well as removing any ambiguity regarding the diversion of injection sources away from the RPV when adequate core cooling is not assured (i.e. low reactor water level) could be included as part of future accident management development.

6.2.6.3 Shell Integrity: DW Spray Usage or Debris Cooling

Insight- DW sprays offer an additional alternative to the control of the drywell temperature to avoid premature containment failure.

Possible improvement / strategy- Relaxation of the restrictions on the use of the DW sprays in the DWSI curve of the EOPs may be a possible future accident management item to develop.

6.2.6.4 Containment Flooding

Insight- Current EPG directions with regard to containment flooding sequences can result in the highest potential consequences at the earliest time.

Possible improvement / strategy- Future accident management strategies should provide guidance to the operator on protecting containment and cooling debris using methods that do not require the venting of the RPV and avoid using the DW vent unless no other alternative exists.

6.2.7 Other Considerations

In this section a brief discussion regarding the significant human reliability actions is given. Additional information regarding these items is given in Section 3.4.1.

6.2.7.1 Operators Fail to Manually Depressurize

The basic event designator for this event is OOPAF-MANUAL-DEP.

> Insight- As indicated in Section 3.4.1 this is one of three significant human actions modeled in the PRA. It exists only because of the policy to immediately lockout the automatic initiation of the ADS actuation. The possibility of this failure can be eliminated if the practice of immediately locking out the automatic actuation of ADS is eliminated (for example, instead of locking ADS out, reset the timer). However, the act of manually initiating the ADS, once the automatic initiation is locked out, is a proceduralized and practiced action that is carried out from the control room, and there is approximately an hour to accomplish this relatively simple action.
>
> Possible improvement / strategy- Evaluate the benefits of resetting the timer instead of immediately locking out the automatic initiation of ADS.

6.2.7.2 Operators Fail to Initiate Containment Heat Removal

Containment heat removal can be accomplished by either of two actions. The operators failing to initiate torus cooling mode of RHR is designated as LOPAF-TORUS-COOL. The operators failing to vent containment via torus vent is designated as VOPAF-TORUS-VENT.

Insight- These are significantly important human actions that can affect a number of sequences. These actions are proceduralized and practiced. They can also be carried out from the control room and are reasonably straight forward and simple actions. The operating staff is well trained and very familiar with these actions.

Possible improvements / strategy- Due to the high reliability of accomplishing these actions, no further improvements are proposed. Continued recognition of the importance of these actions, with appropriate training is being done as it has been done in the past.

7. SUMMARY OF CONCLUSIONS

7.1 SUMMARY OF RESULTS

Core damage frequency for DAEC is estimated at 7.84E-6. Core damage frequency by initiator is shown on Figures 7.1-1 and 7.1-2. Core damage frequency by initiator type is shown on Figure 7.1-3. Core damage frequency by damage class is given by Figures 7.1-4 and 7.1-5.

The results of the Level II are expressed by the following figures:

Figure 7.1-6, which shows internal events by release magnitude and timing; and

Figures 7.1-9 through 7.1-20, which show containment performance by accident class.

7.2 ASSESSMENT OF RESULTS PER IPE PURPOSES

GL 88-20 stated several objectives the expected to be accomplished by the performance of an IPE. The DAEC IPE has been completed to meet the following objectives:

	Objective	Method of Achievement
1.	To develop an appreciation for severe accident behavior at DAEC.	A Level II PRA was conducted with a significant participation by the utility staff in its direction, production, and review.
2.	To understand the most likely severe accident sequences that could occur at DAEC.	Utilizing the NRC criteria for screening given in Appendix 2 of GL 88-20, five sequences from the DAEC PRA were identified that contributed 5% or more to the core damage frequency. One of these also was equal to or greater than 1E-6/yr. These results are shown in Figure 7.2-1.
3.	To gain a more quantitative understanding of the overall probabilities of core damage and fission product releases.	The DAEC Safety Analysis Group was involved with all aspects of the preparation of the PRA. A large and diverse group of personnel, internal to the utility, were

4. If necessary, reduce the overall probabilities of core damage and fission product releases at DAEC by modifying, where appropriate, hardware, procedures, or training that would help or mitigate severe accidents.

used to review the PRA at various stages of its development. All quantification was done in-house.

It is concluded that DAEC is a plant with a low risk of core damage and fission product release. It has only one sequence that meets the 1E-6 screening criteria and even this sequence is just at 1E-6. There are, therefore, no sequences or phenomena that are identified in this study that would make DAEC an outlier plant. As a result, no further changes would appear to be necessary.

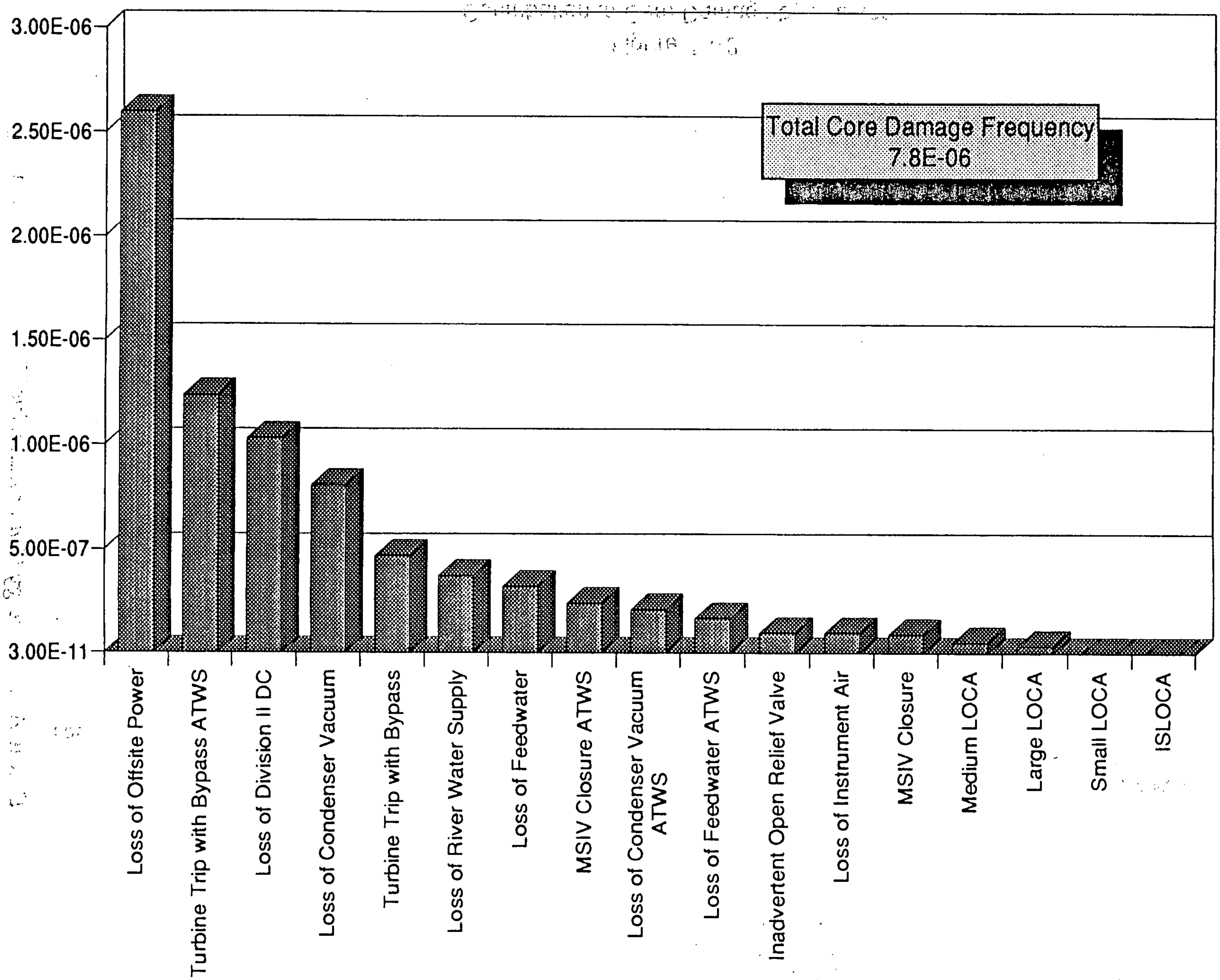


Figure 7.1-1
Core Damage by Initiator

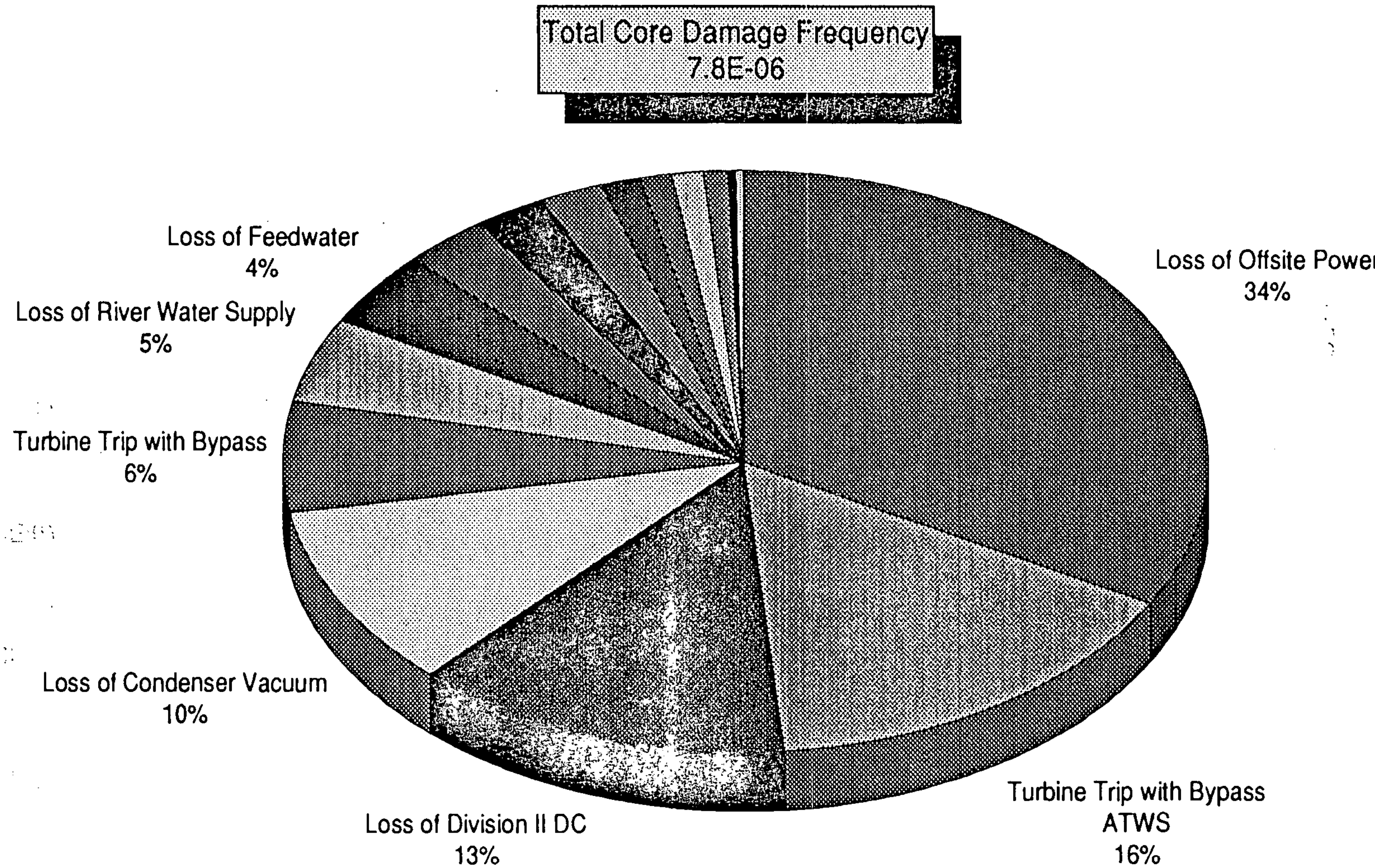


Figure 7.1-2
Contribution to Core Damage by Initiator

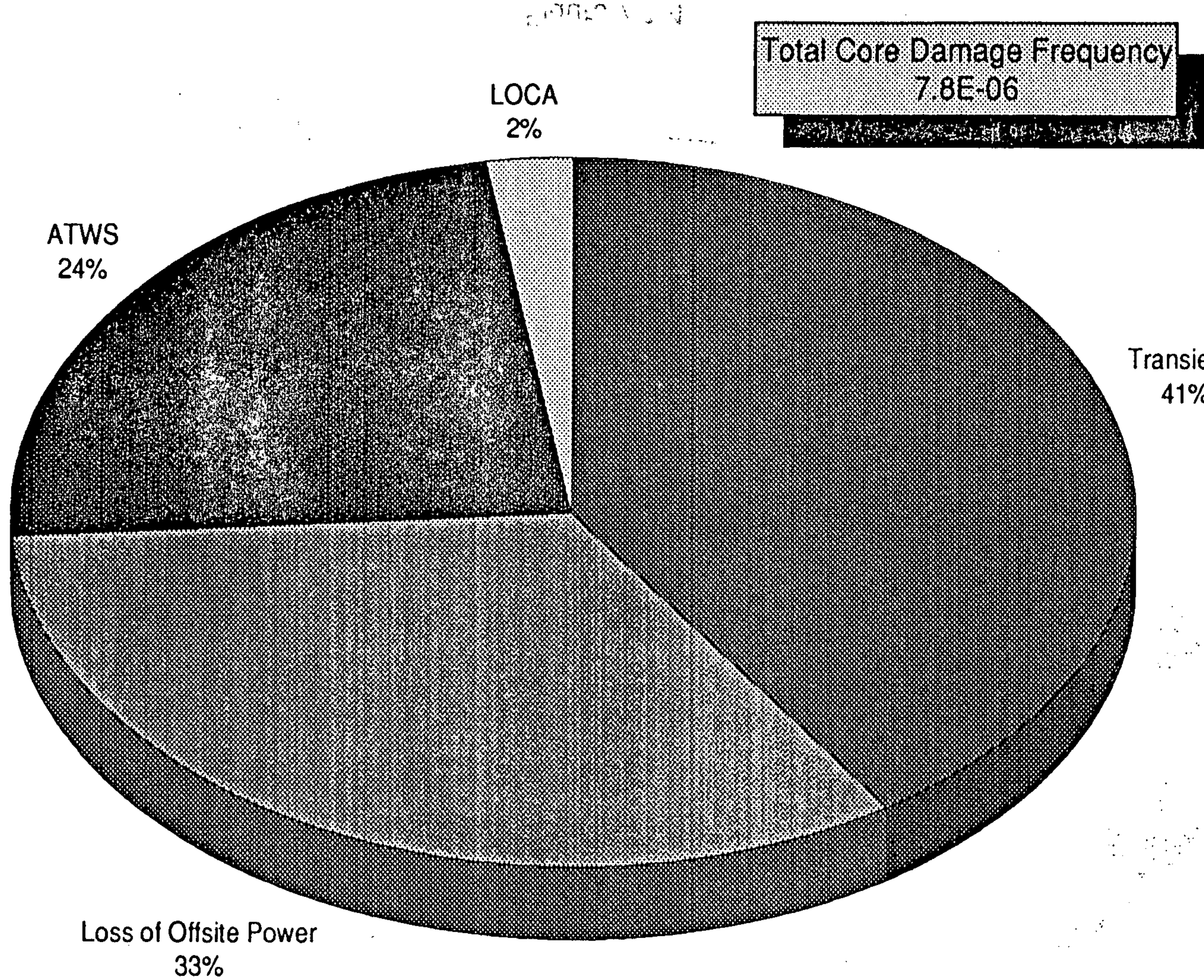


Figure 7.1-3
Contribution to Core Damage

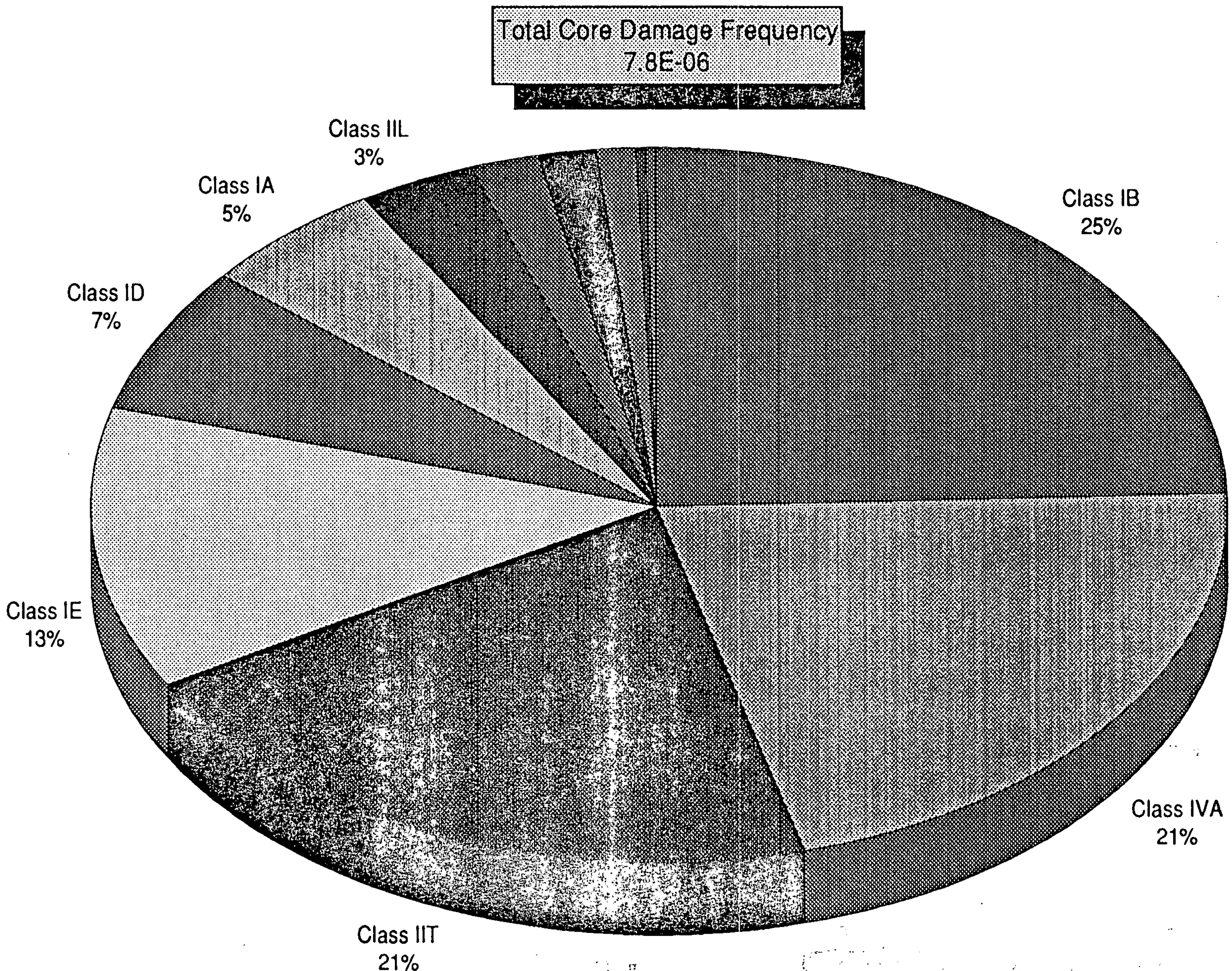


Figure 7.1-4
Core Damage Contribution by Class

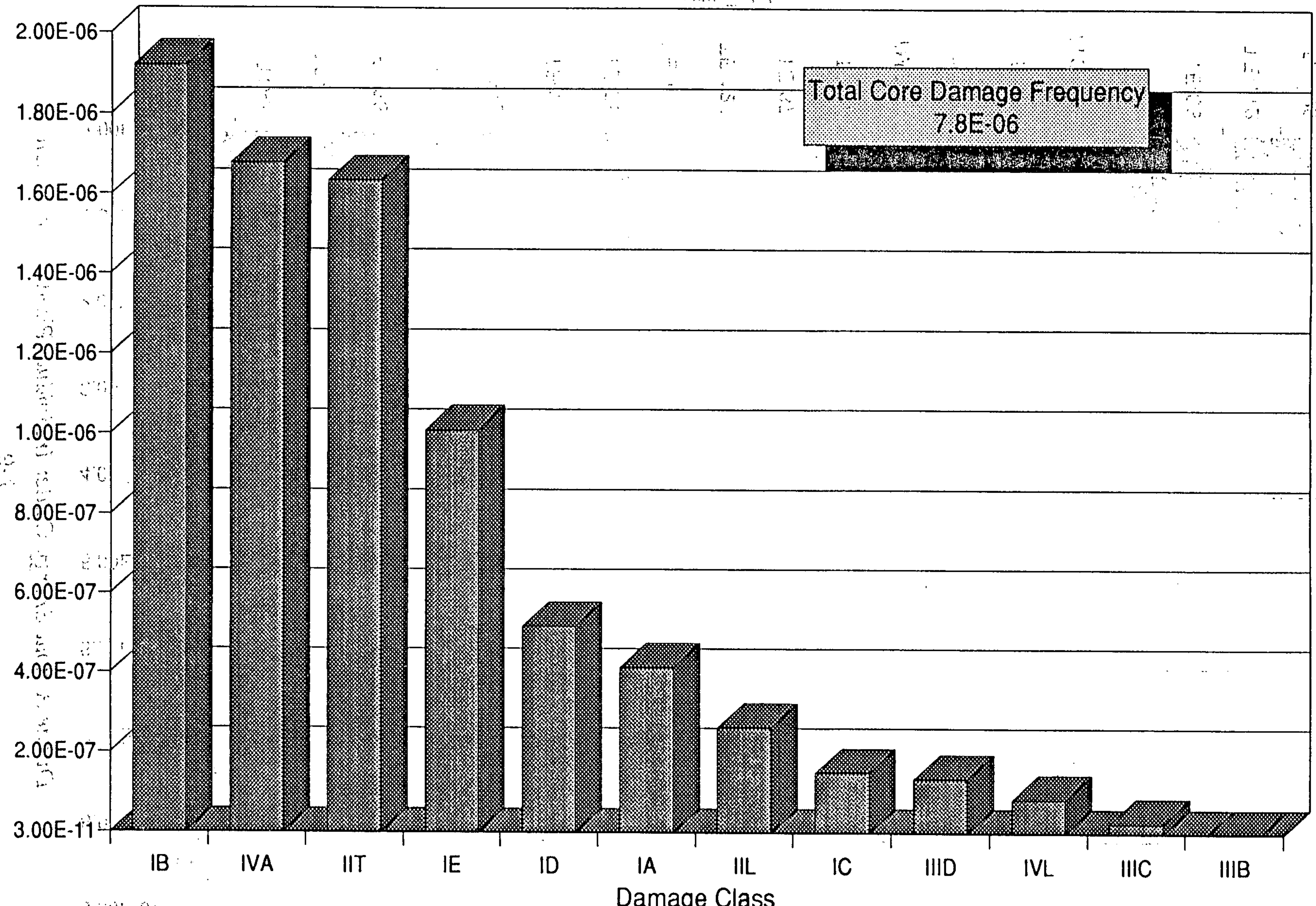


Figure 7.1-5
Core Damage by Class

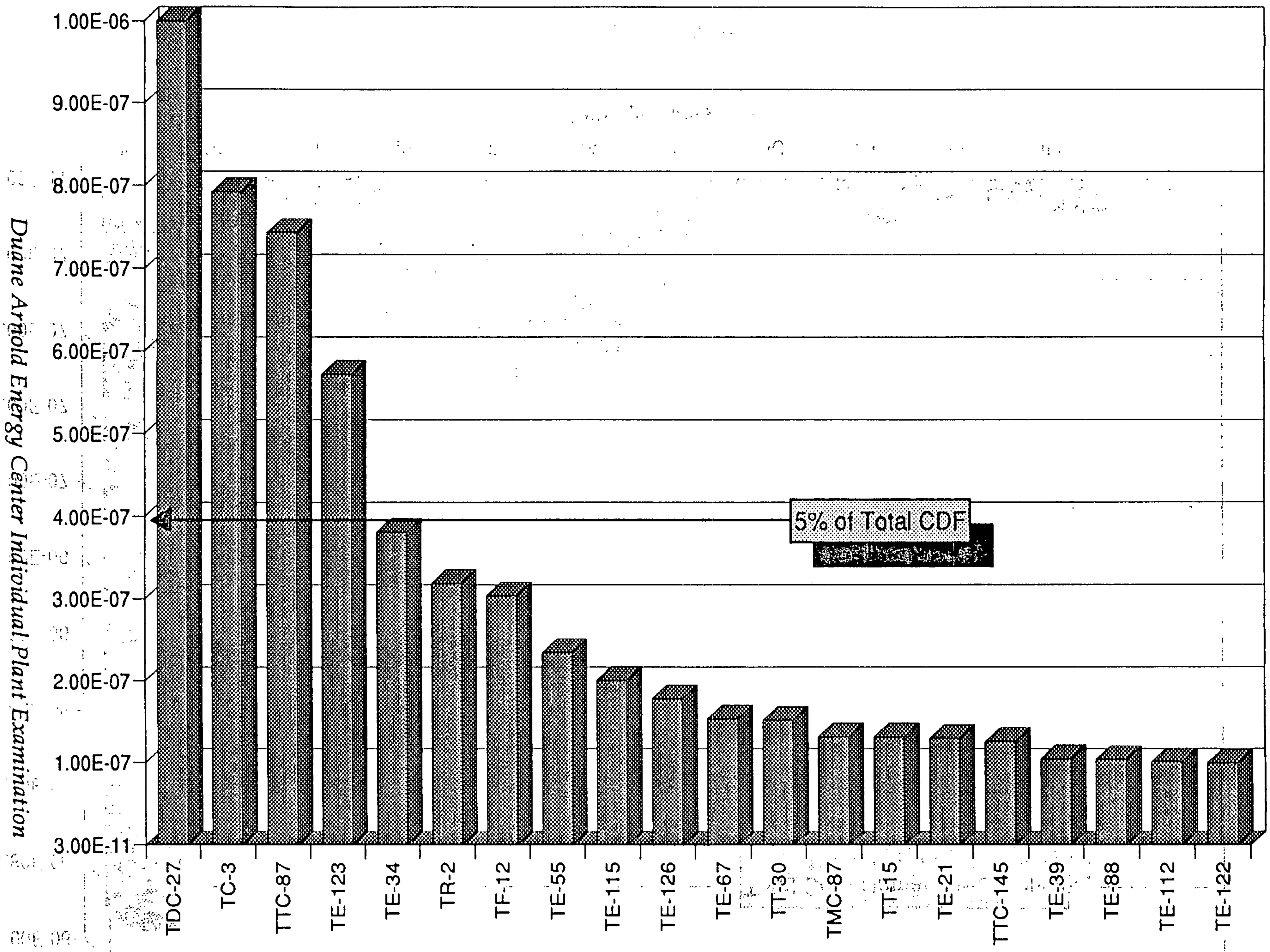


Figure 7.1-6
Most Significant Sequences

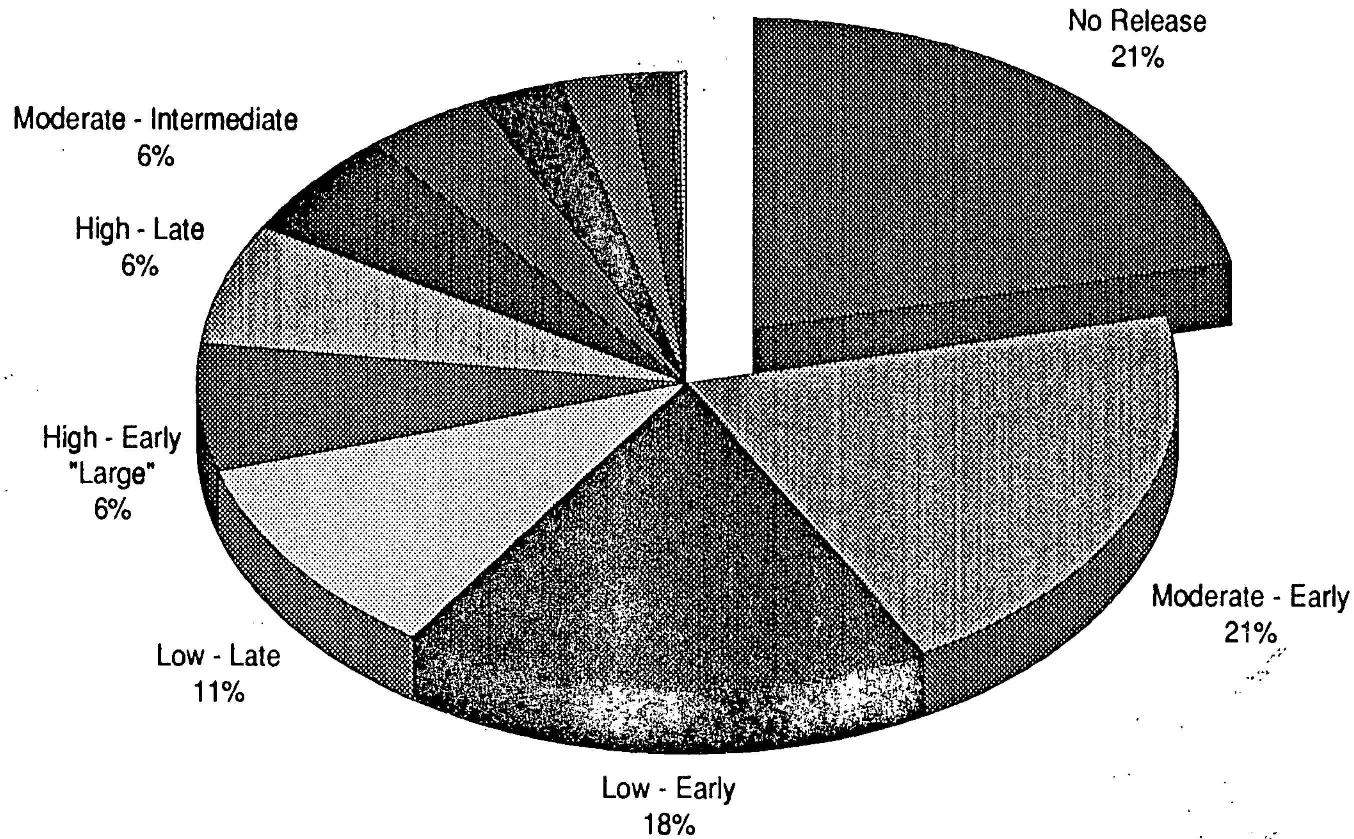


Figure 7.1-7
Containment Release Magnitude and Timing

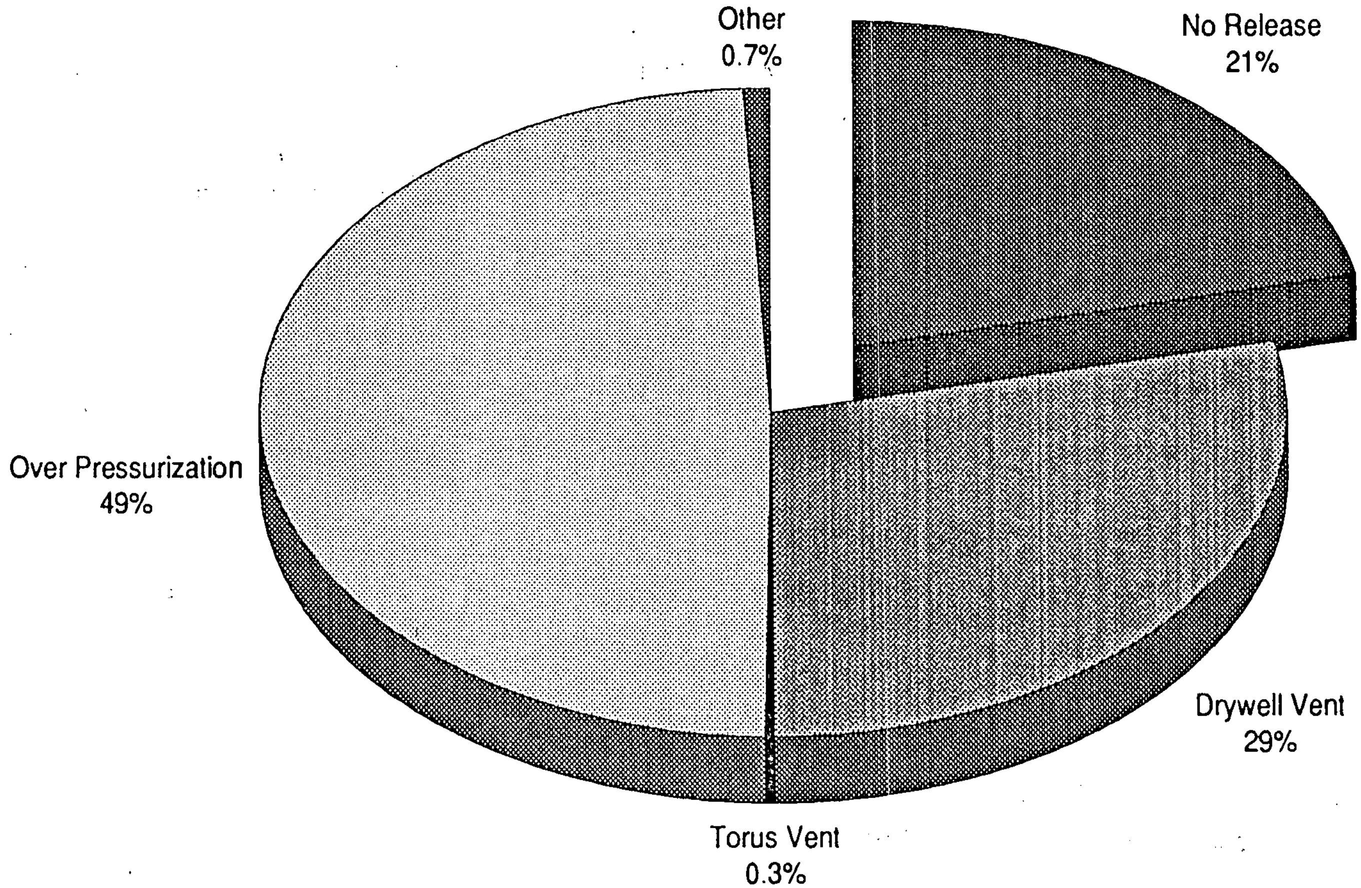


Figure 7.1-8
Containment Failure Modes

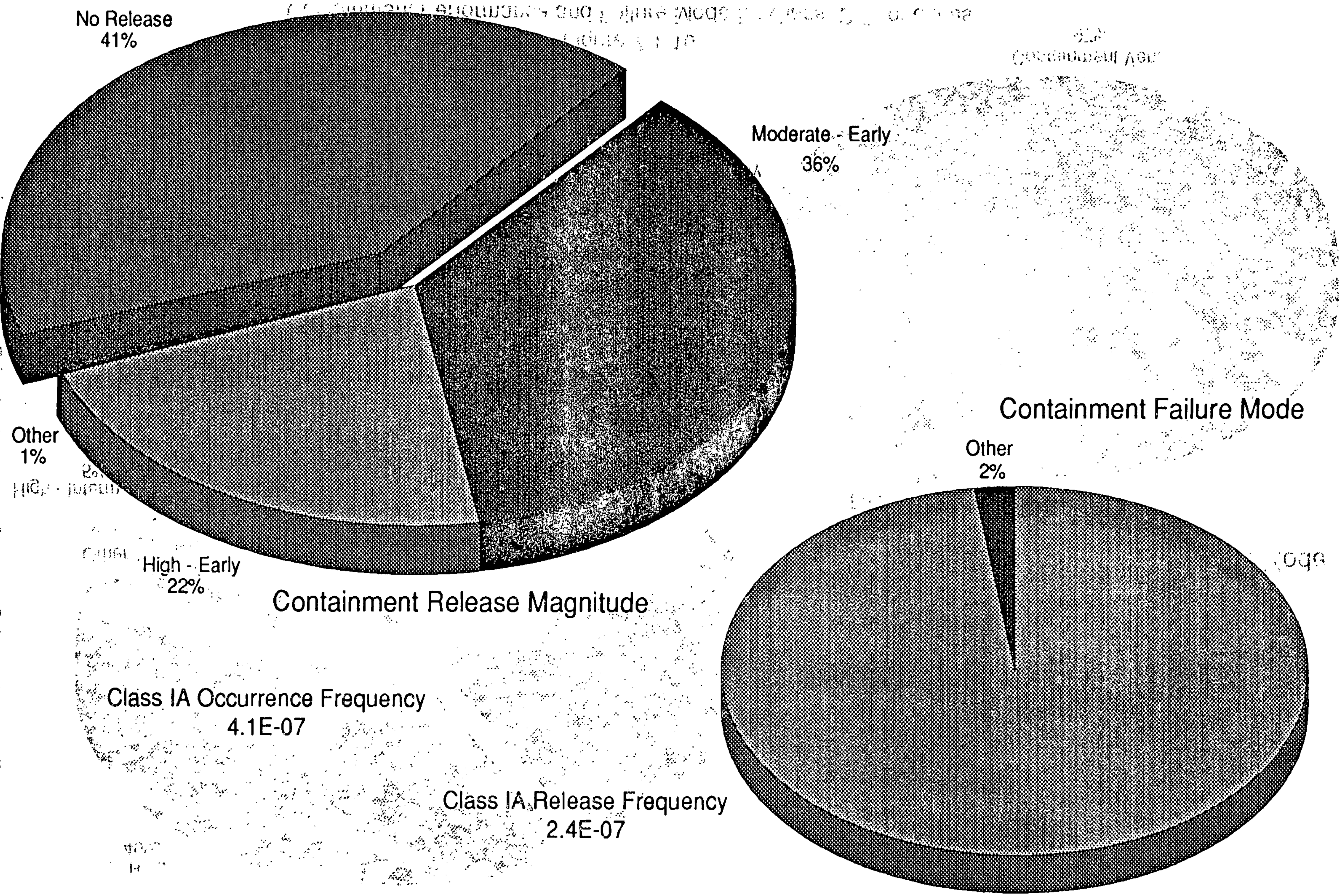


Figure 7.1-9
Containment Performance and Failure Mode for Class IA Sequences

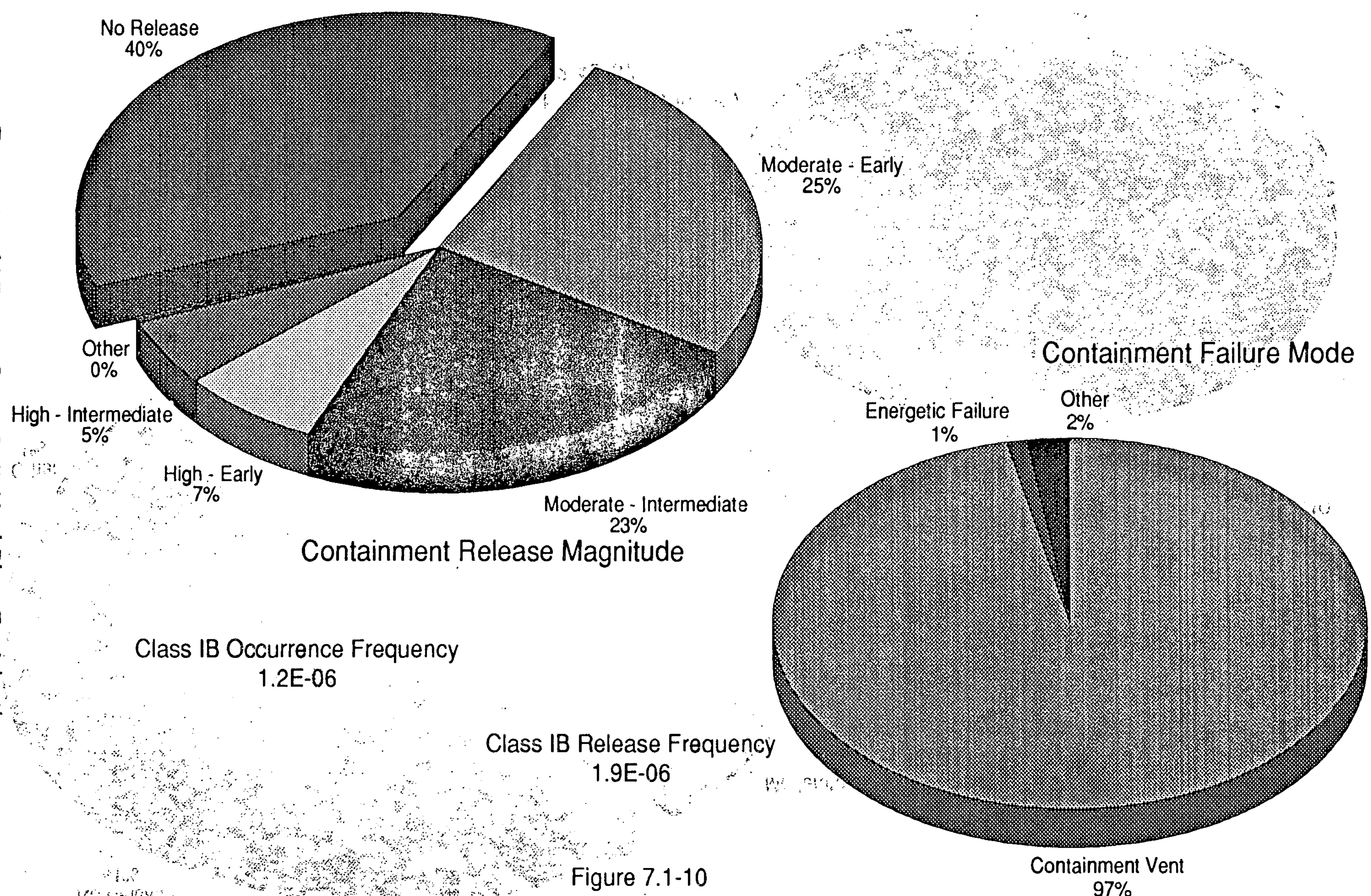


Figure 7.1-10
Containment Performance and Failure Mode for Class IB Sequences

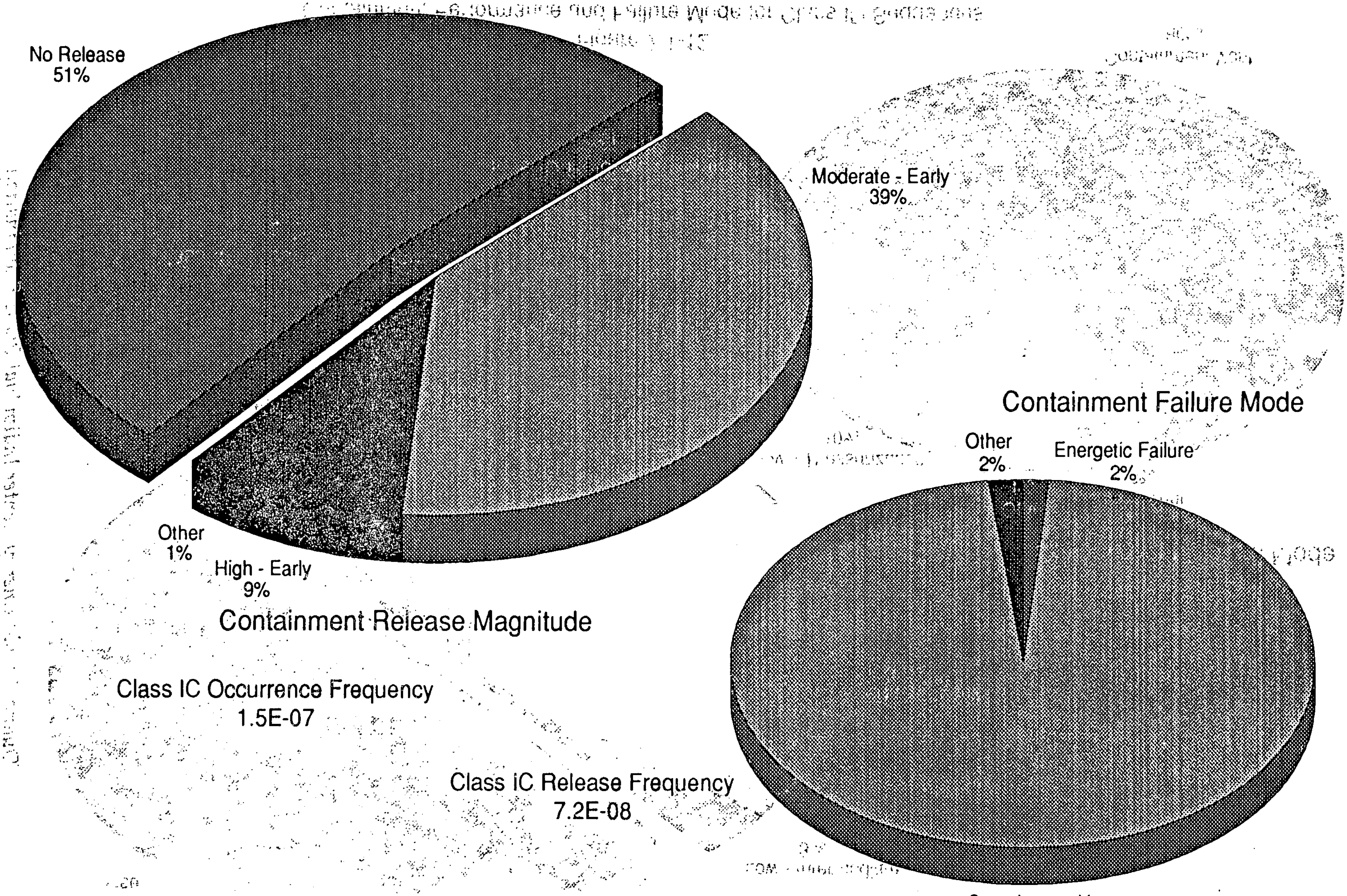


Figure 7.1-11
Containment Performance and Failure Mode for Class IC Sequences

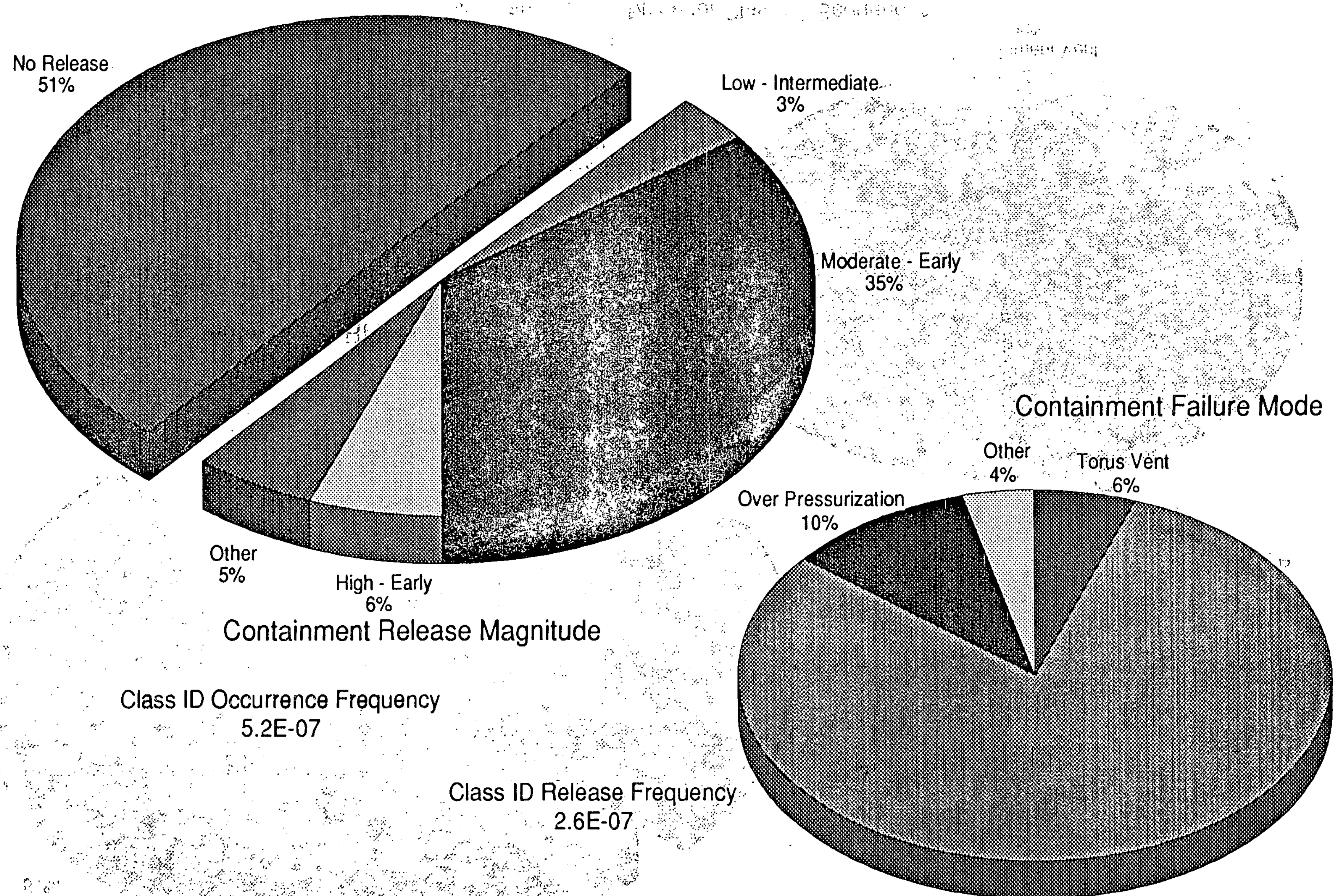


Figure 7.1-12
Containment Performance and Failure Mode for Class ID Sequences

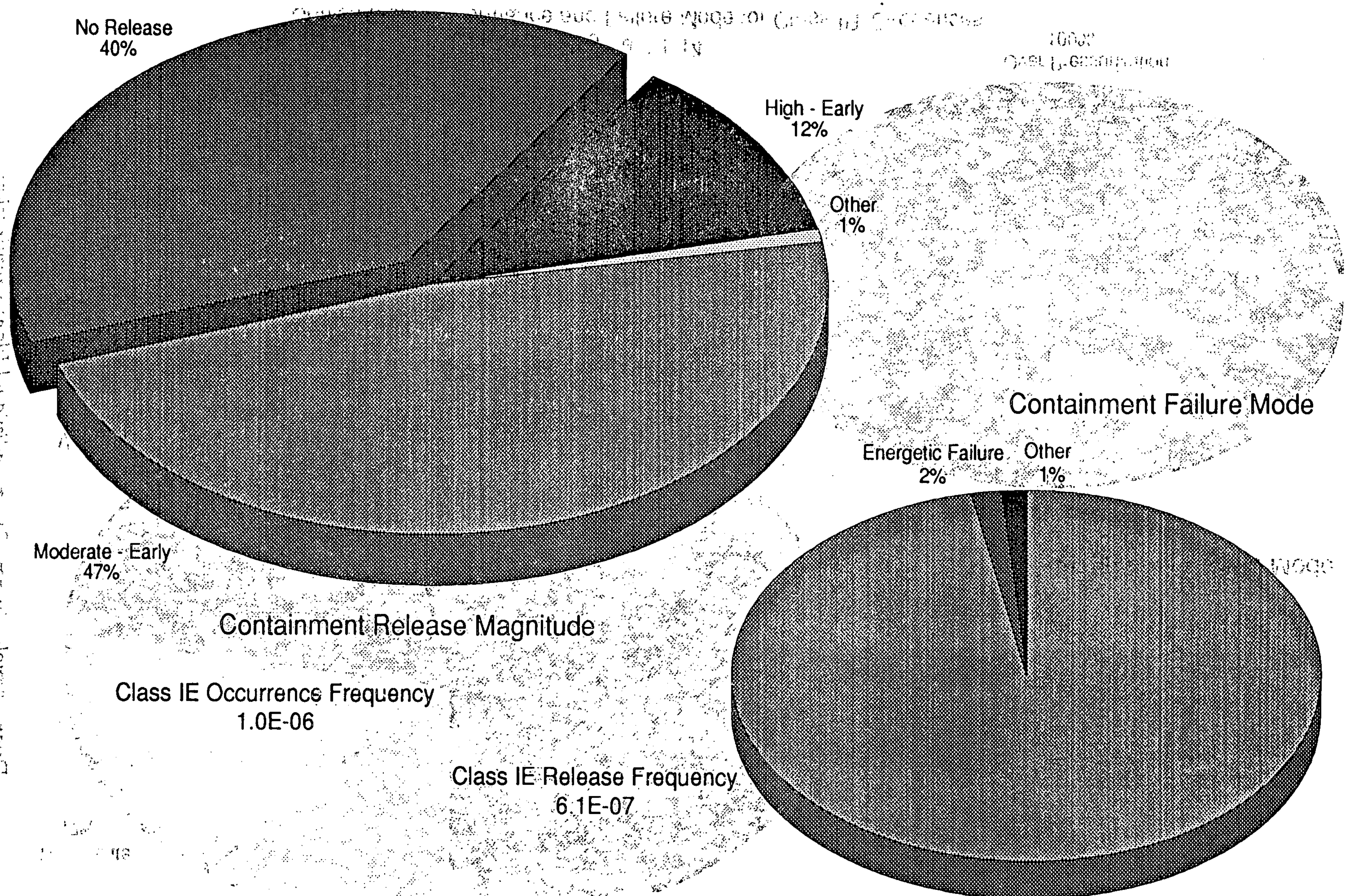
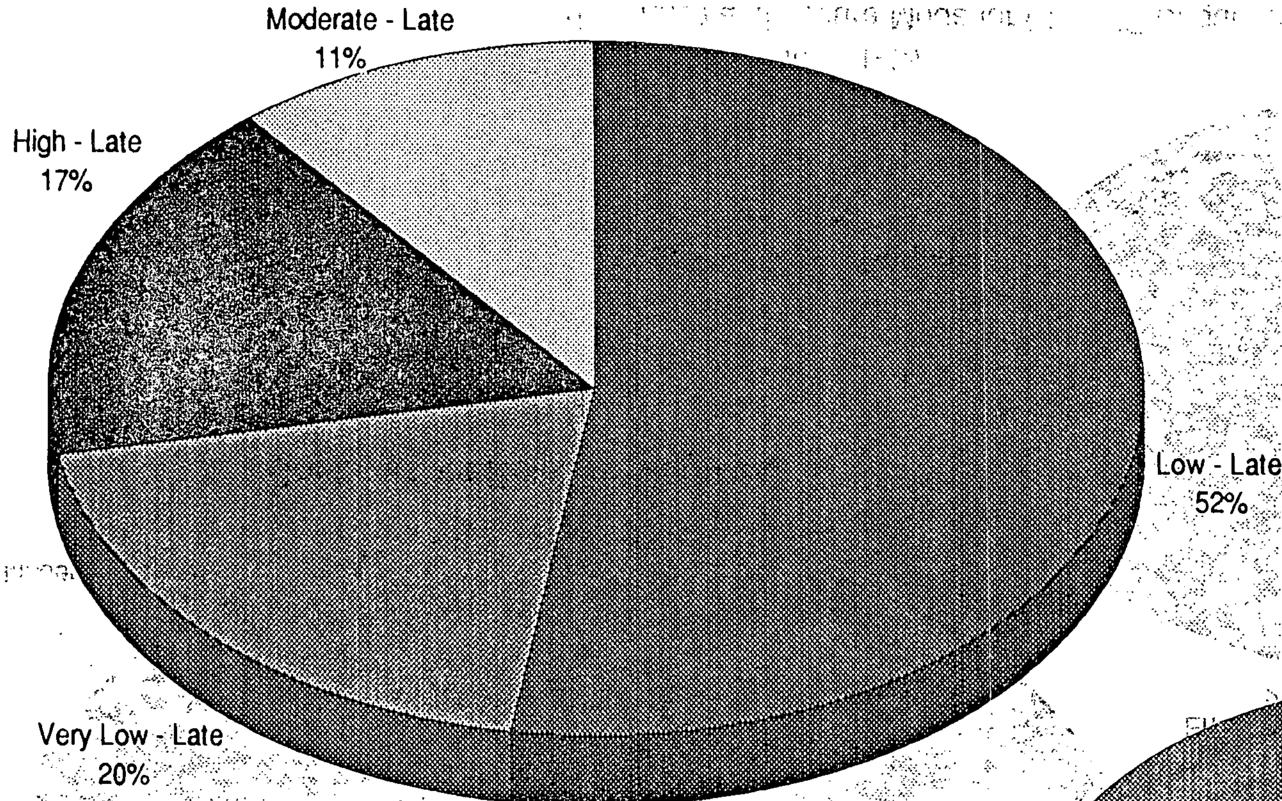
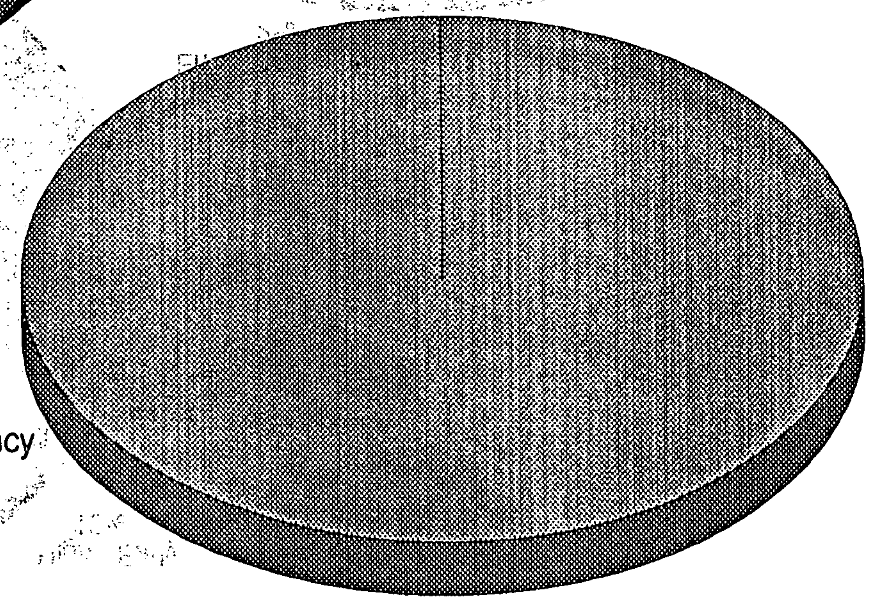


Figure 7.1-13
Containment Performance and Failure Mode for Class IE Sequences



Containment Release Magnitude

Containment Failure Mode



Class IIT Occurrence Frequency
1.6E-06

Class IIT Release Frequency
1.6E-06

Figure 7.1-14
Containment Performance and Failure Mode for Class IIT Sequences

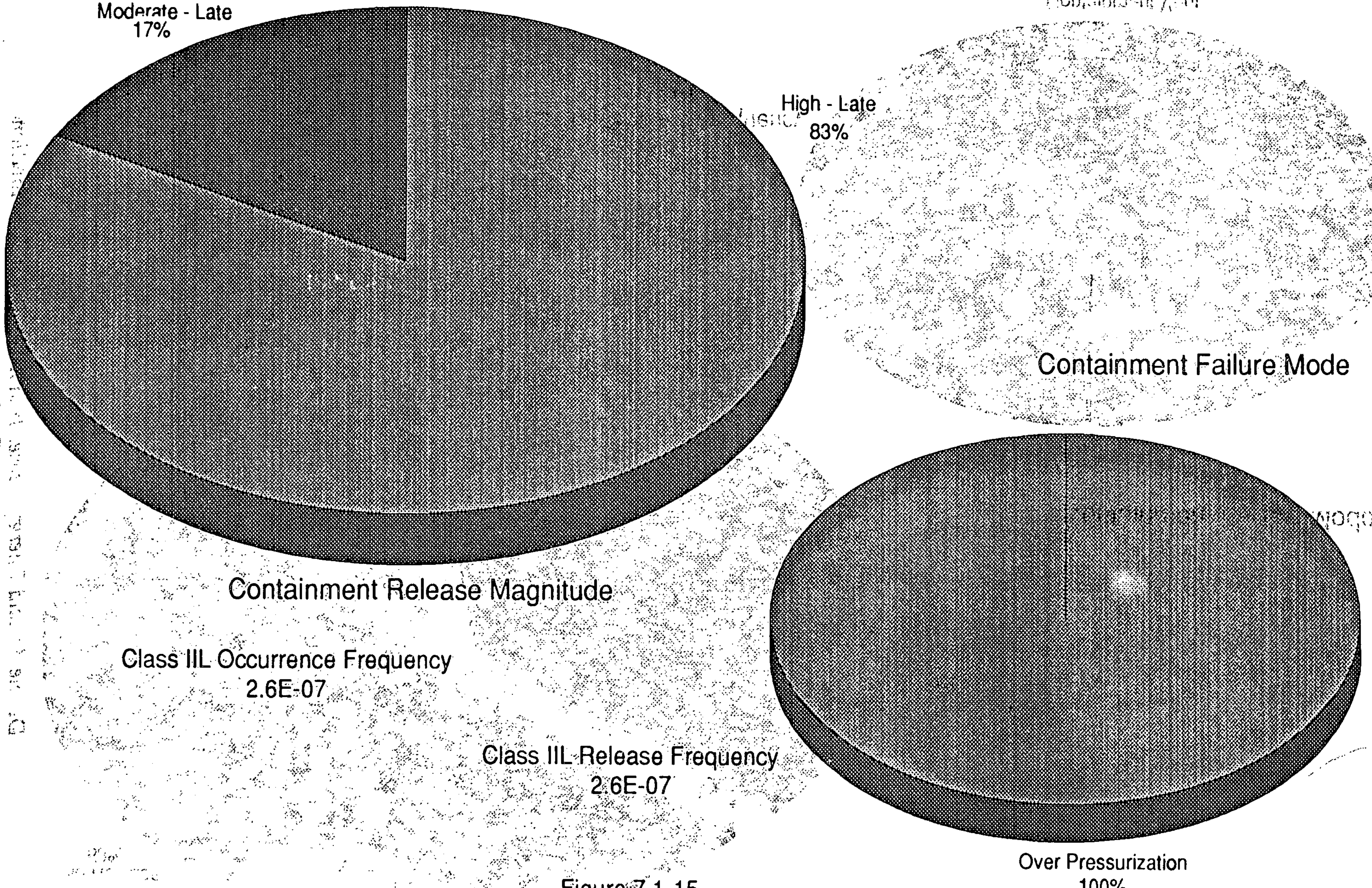


Figure 7.1-15
Containment Performance and Failure Mode for Class IIL Sequences

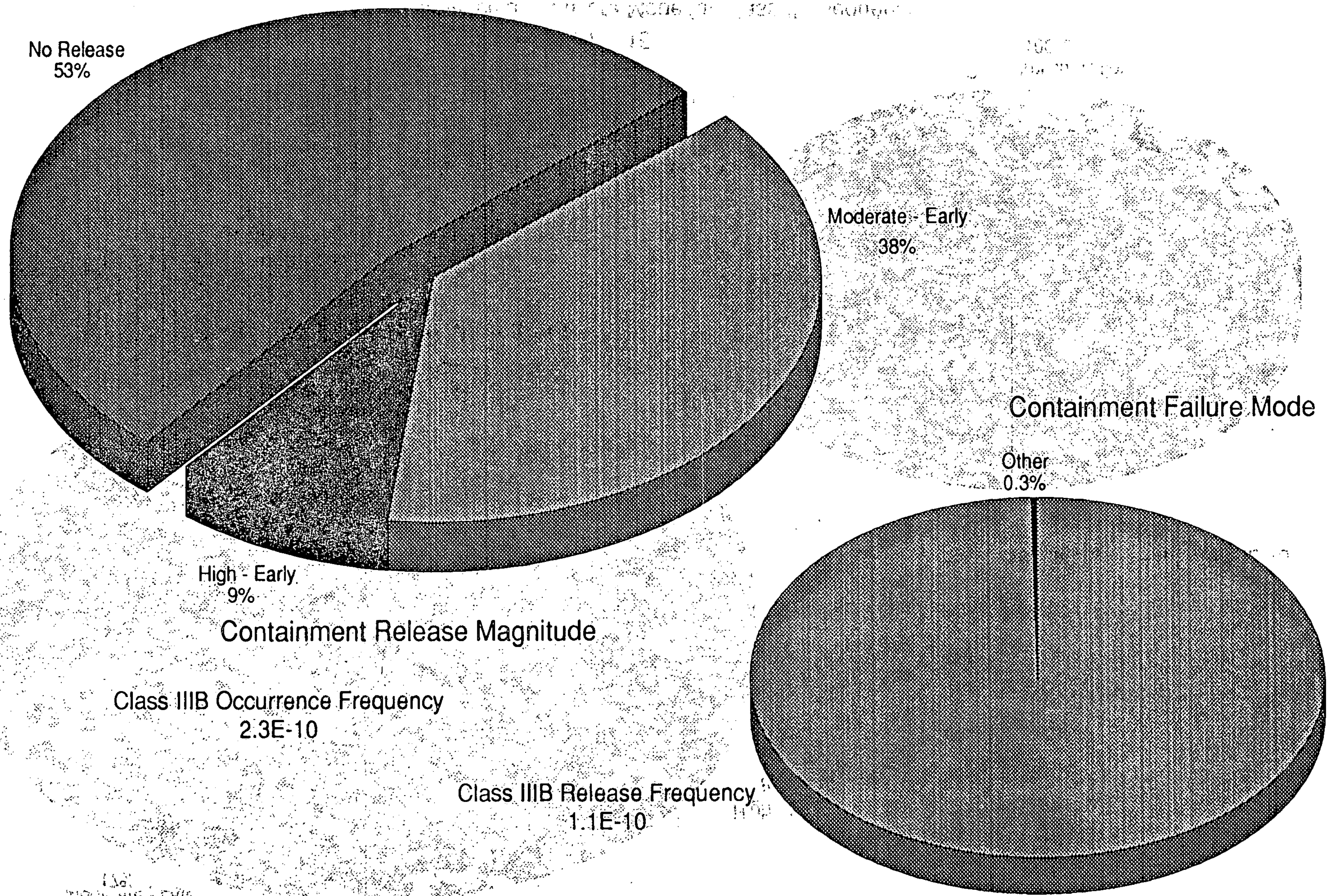


Figure 7.1-16
Containment Performance and Failure Mode for Class IIIB Sequences

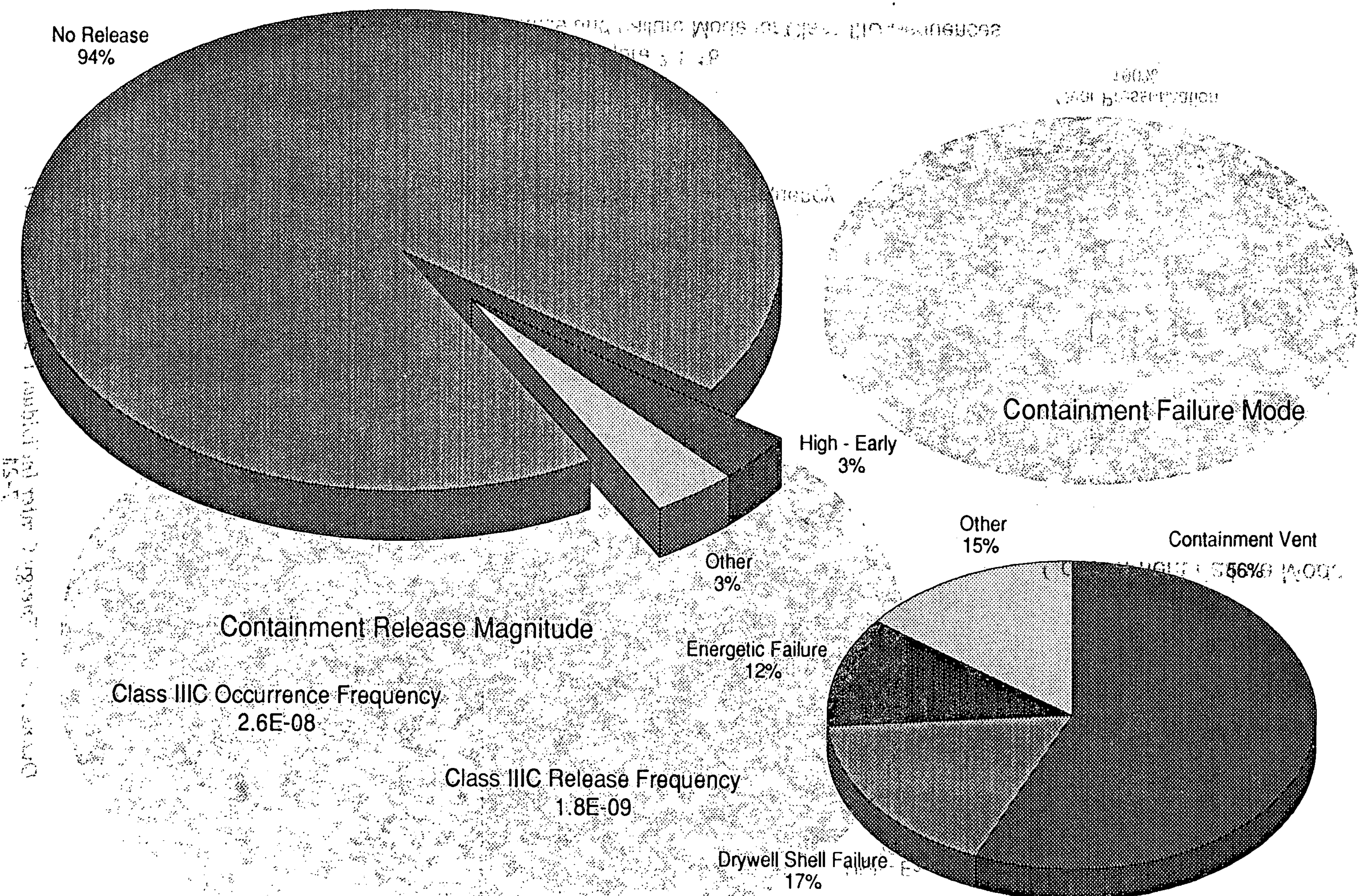


Figure 7.1-17
Containment Performance and Failure Mode for Class IIIC Sequences

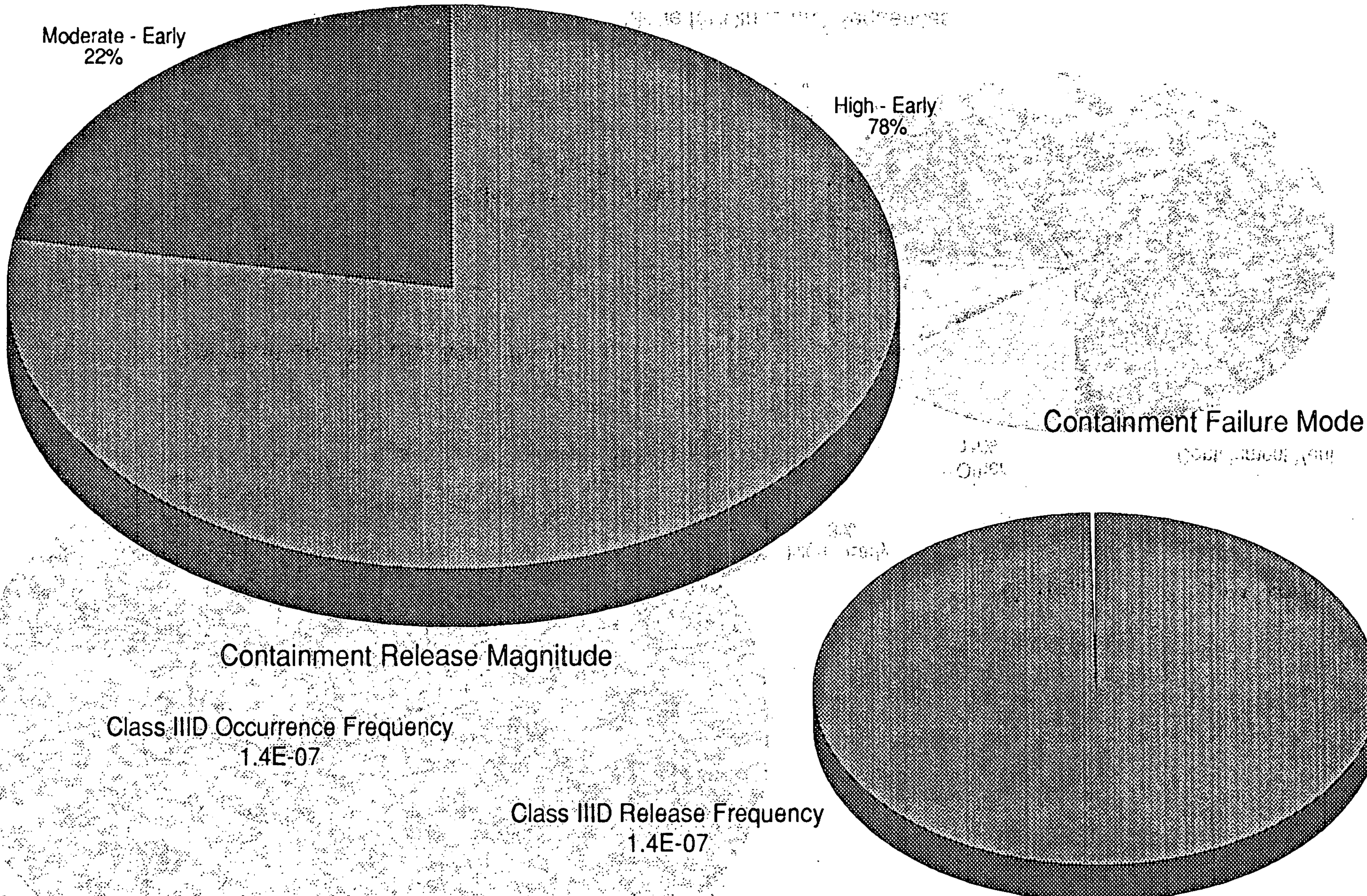


Figure 7.1-18
Containment Performance and Failure Mode for Class IIID Sequences

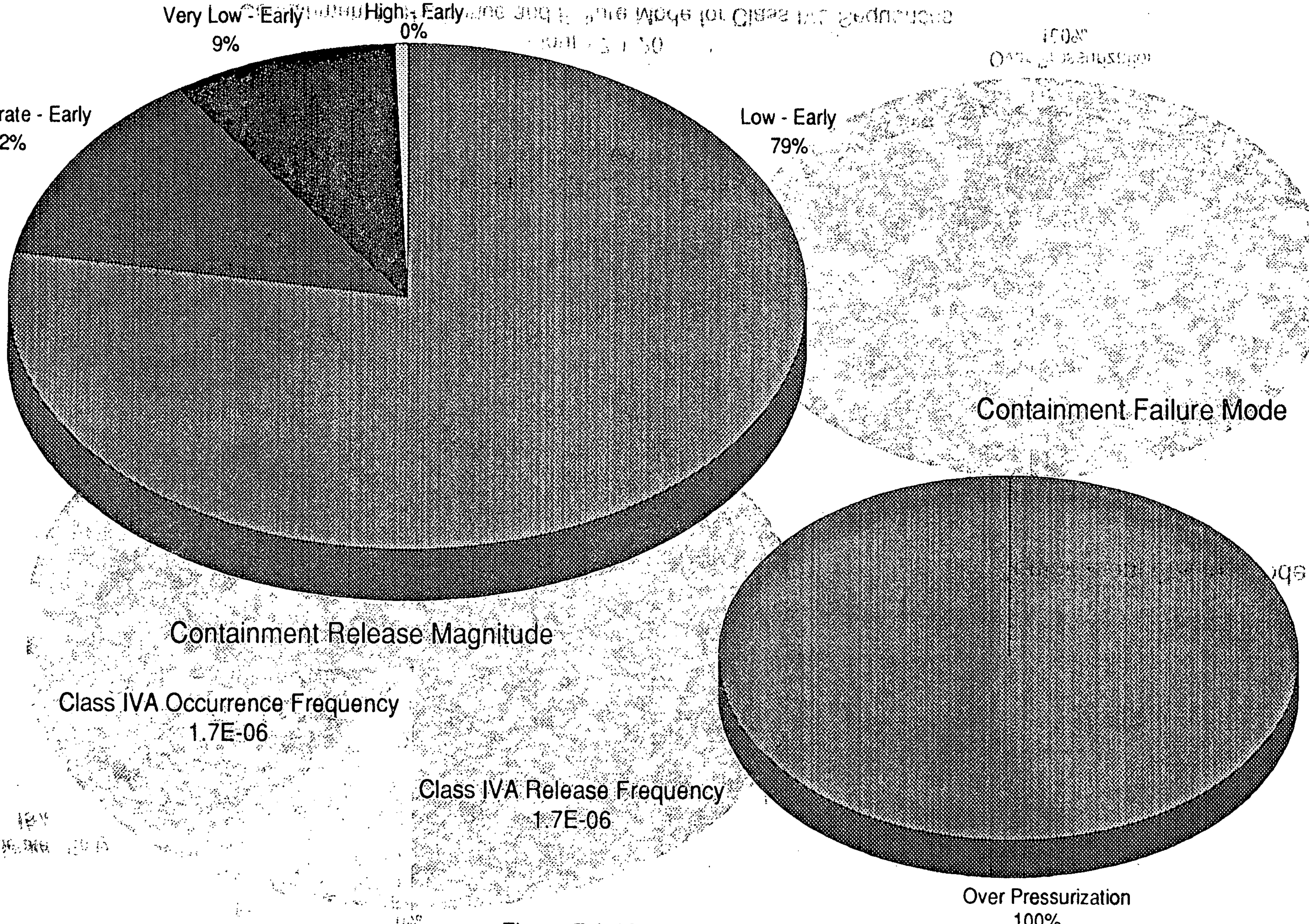
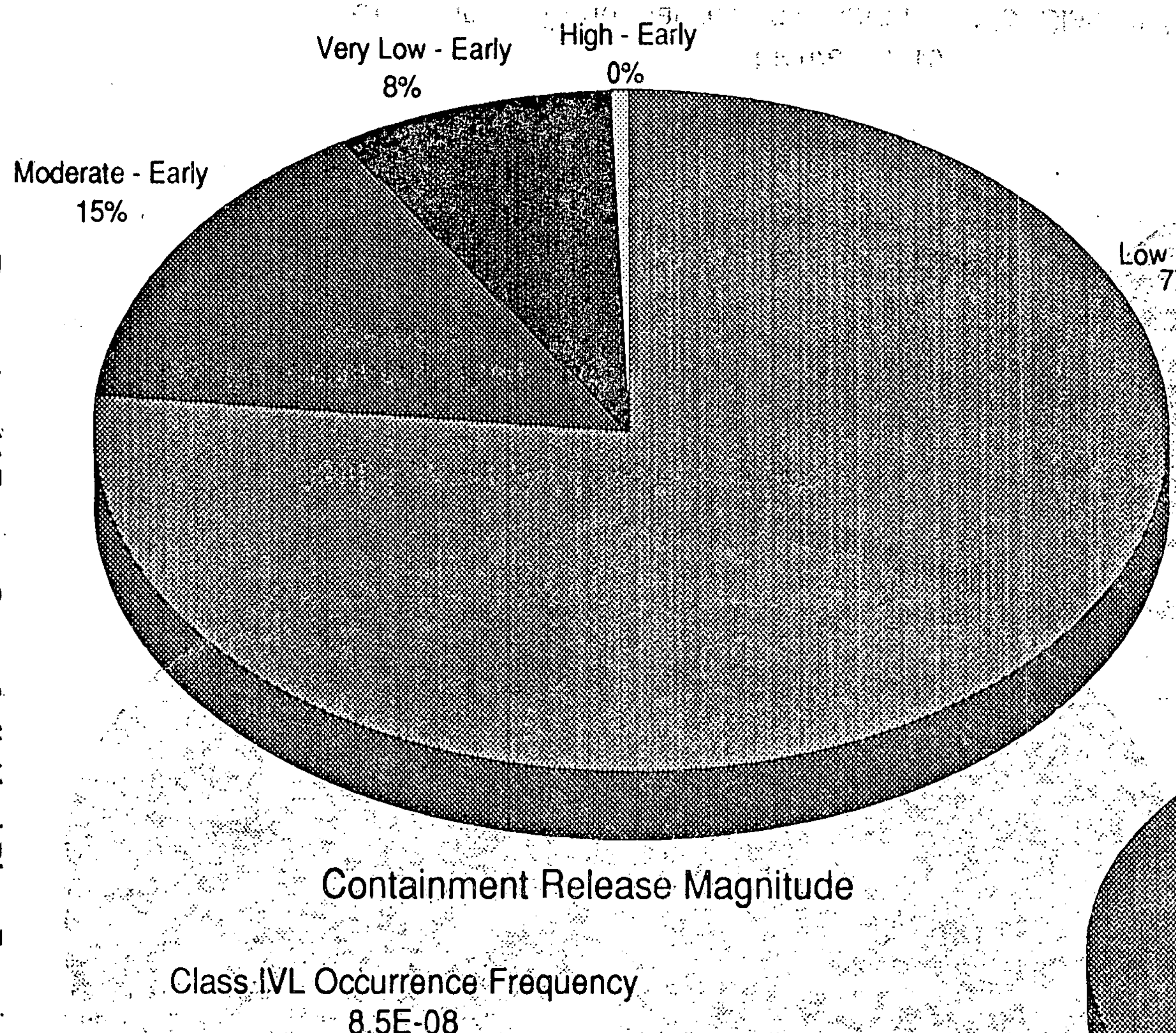


Figure 7.1-19
Containment Performance and Failure Mode for Class IVA Sequences



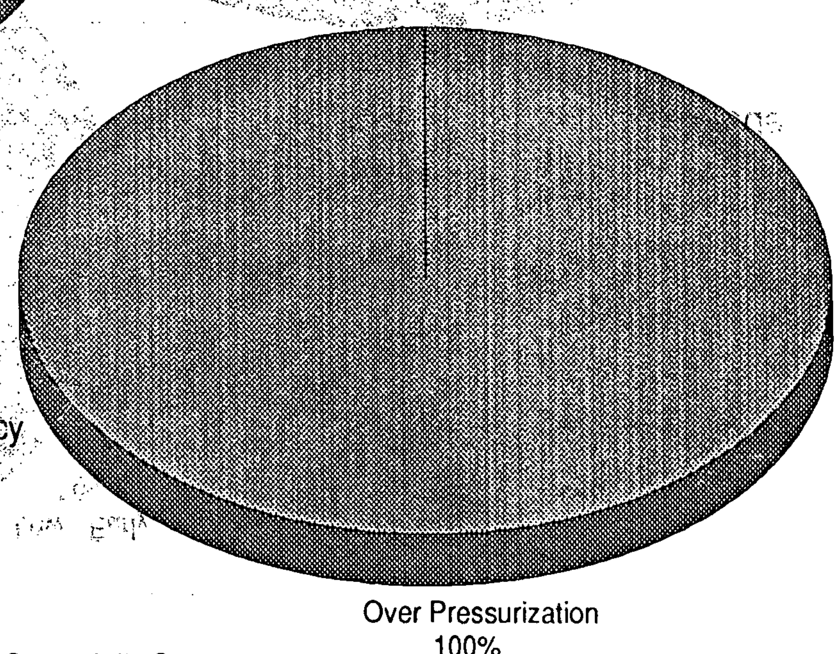


Figure 7.1-20
Containment Performance and Failure Mode for Class IVL Sequences

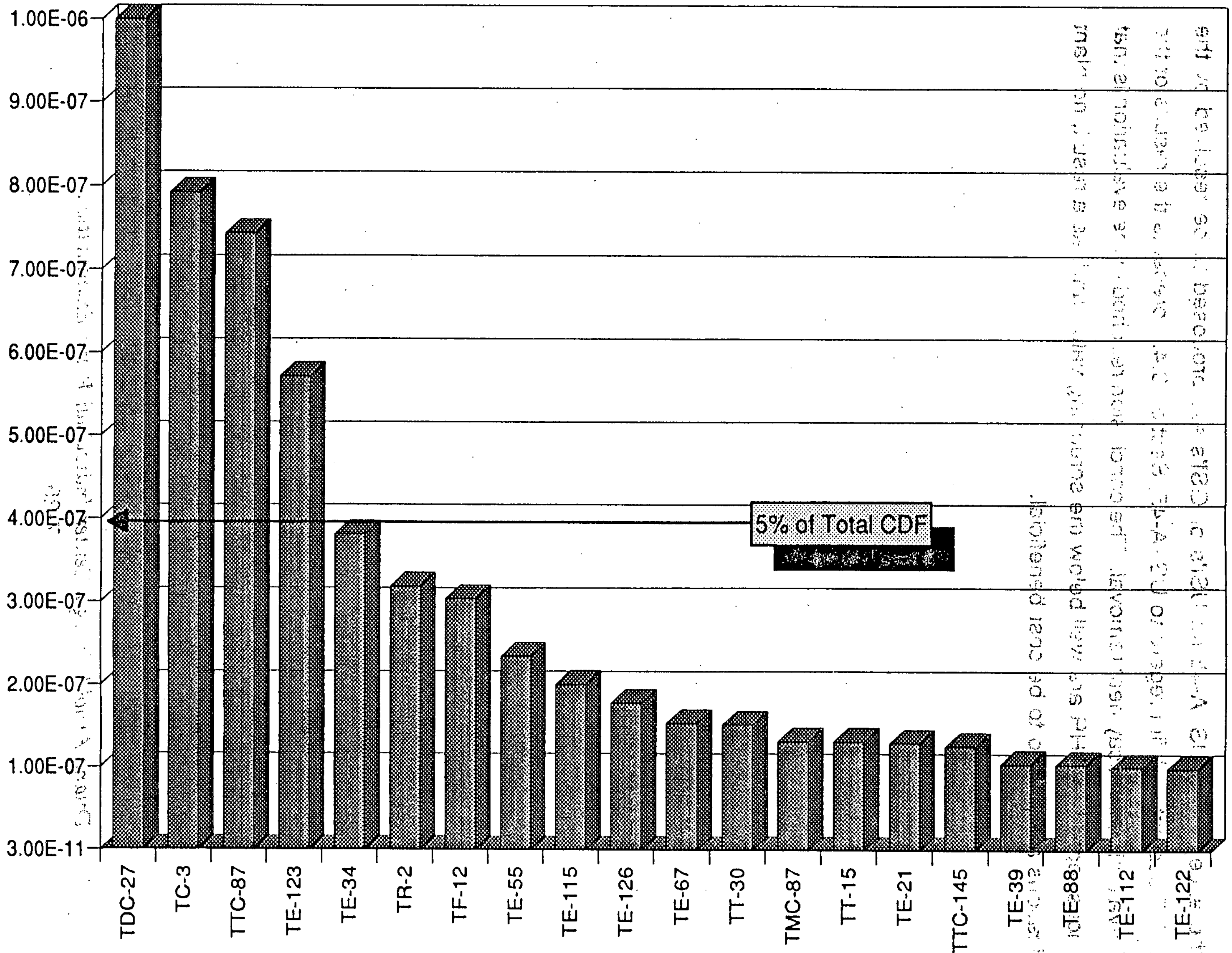


Figure 7.2-1
Most Significant Sequences

7.3 PROPOSED RESOLUTION OF USIs AND GSIs

With the exception of USI-A-45 no USI's or GSI's are proposed to be resolved by the DAEC IPE submittal. With regard to USI-A-45, Section 3.4.3 presents the results of the DAEC evaluation of decay heat removal. The conclusion reached in the evaluation is that sequences regarding DHR are well below the screening value and, as a result, no plant modifications are judged to be cost beneficial.