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An Estimate of Release Frequencies for CRBRP

Potential Core Disruptive Accidents

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ABBREVIATIONS AND ACRONYMNS

AC	Annulus Cooling (Containment System)
CDA	Core Disruptive Accidents
CI	Containment Isolation
CRBRP	Clinch River Breeder Reactor Plant
CU	Clean-Up (System, Scrubbers and Filters)
DHRS	Direct Heat Removal System
EDV	Early Dirty Venting
EPS	Electric Power System
LDV	Late Dirty Venting
LOCA	Loss of Coolant Accident
LOHS	Loss of Heat Sink
MHTS	Main Heat Transfer System
OP	(Containment) Overpressure
PACC	Protected Air-Cooled Condenser
PPS	Plant Protection System
PRA	Probabilistic Risk Assessment
PT	Pump Trip
PWST	Protected Water Storage Tank
RCB	Reactor Containment Building
SCRAM	(Shutdown System)
SHRS	Shutdown Heat Removal System
TBS	Turbine Bypass System
ULOF	Unprotected Loss of Flow
UTOP	Unprotected Transient Overpower
VPS	(Containment) Vent and Purge System

Section 1.0
INTRODUCTION

The purpose of the report is to describe the results of an analysis which evaluates various types of releases for potential core disruptive accidents at the Clinch River Breeder Reactor Plant (CRBRP). Event tree methodology is used to define potential internally and externally initiated accident sequences which cause release of radionuclides from the CRBRP primary system and containment. These accident sequences are quantified by simplified fault tree models of the safety systems and best estimate values of the phenomenological events.

Results indicating dominant accident sequences and their relative contributions to potential Core Disruptive Accidents (CDA's) and associated types of containment release modes are provided. These containment release (or response) modes are defined in a form suitable for use in radiological risk analysis. Once the containment leak rates and source terms are identified and calculated for each containment release mode, the radiological health effects can be assessed by combining the frequency and radiological source term for each containment release mode.

Five separate event trees were constructed to analyze the progression of the accident in an orderly and systematic manner using both phenomenological uncertainties and functional frequencies (or system unavailabilities).

The remainder of the report is organized as follows: Section 2 provides a summary of the limitations and the results of this study. Section 3.0 provides an overview of the methodology, and Section 4.0 describes the procedure for defining the accident sequences. Section 5.0 details the quantification of defined accident sequences and the CRBRP containment response. Assessment of the effect of external events on the plant behavior is presented in Section 6.0. Section 7.0 contains a more detailed presentation of the results and conclusion of the study. Finally Section 8.0 contains a brief analysis of sensitivity of the results and a discussion of the uncertainties.

Section 2

SUMMARY OF THE LIMITATIONS AND RESULTS

This section briefly discusses the limitations and the results obtained for the CRBRP in this accident analysis study.

2.1 SCOPE AND LIMITATIONS

This study is a limited effort to identify and evaluate contributors to core disruptive accidents and containment releases for the CRBRP. In most instances system and event data were selected from available literature and are deemed to represent conservative values for initiator frequencies and system function unavailabilities. No detailed fault tree analysis has been performed and potential modifications to enhance the reliability of specific systems are not considered in the baseline analysis. (A sensitivity study is included in Section 8 which indicates that increased redundancy of the DHRS would reduce the frequency of protected core disruptive events.)

While numerical values are presented for various core disruptive accident categories and containment release modes, they should be considered as rough, figure of merit type indicators. The major effort of this work was to qualitatively structure the plant safety logic into a format suitable for presentation of predominate core disruption and containment release sequences. This work also attempted to include both internal and external initiators to provide a more comprehensive picture of the total spectrum of contributions to core disruption and containment release. Thus quantification of the core disruptive and containment release frequences was performed, primarily, to gain perspective on the relative importance of the different accident sequences which were constructed.

A logical next follow-on step would be to critically review the quantification process to increase our understanding and confidence in the calculated estimates.

The sources of the uncertainties in the analysis are generally recognized, but the scope of the effort has permitted only a modest effort in estimating the effect of these uncertainties on the results of this study. Some of the sources of uncertainties and limitations are:

- Uncertainties in the data used for system unavailabilities and phenomenological uncertainties
- Uncertaintites in the modeling regarding the physical behavior of the core, vessel, cavity, and the containment under accident conditions; e.g., how the dispersion of the fuel debris inside the cavity can affect upper RCB overpressurization and does it lead to early (<24 hours) or late (>24 hours) venting to prevent a threat to containment integrity.
- Accidents occuring originally within the core at power are analyzed. Other sources and conditions of accidents are not assessed. However, they are judged not to be dominant accident sequences.
- The basemat penetration mode of containment response was not investigated to identify distance of penetration of the molten debris and material into the basemat.
- Loss of Flow (LOF) driven Transient Overpower (TOP) accidents are not analyzed as part of this study. They are judged not to be dominant sequences, however.
- This study estimates the frequency of the containment release caused by low-probability beyond design basics (Class 9) accidents, and is not intended to analyze the Design Basis Accidents (DBA's). However, it is believed that the design basis accidents will not have significant risk implications compared to Class 9 accidents (limitation).

2.2 RESULTS SUMMARY

The reader is directed to Section 7.0 for detailed results of this analysis. Table 2-1 shows the estimated frequency of the Core Disruptive Accidents (CDA's) for CRBRP. Table 2-1 shows that more than 58 percent of the frequency of core disruptive accidents are caused by protected Loss of Heat Sink (LOHS) accidents. More than 90 percent of the LOHS frequency is contributed by simultaneous failure of all three rupture disks in the Intermediate Heat Transport (IHT) loops accompanied by activation of the Sodium-Water Reaction Pressure Relief System (SWRPRS) which dumps the IHTS Sodium into a dump tank. Decay Heat Removal System (DHRS) would be the only means of post-accident decay heat removal in this situation.

Following LOHS accidents the most dominant CDA is an Unprotected Loss of Flow (ULOF) accident, which comprises 24% of all CDAs. Two of every three ULOFs is initiated by a spurious Plant Protection System (PPS) signal followed by the trip of the primary sodium pumps and failure of both primary and secondary shutdown systems.

The third most frequent core disruptive accident is a Transient Overpower accident (TOP) and represent 8 percent of the fequency of the core disruptive accidents. Almost all of the TOP Core disruptive accidents are initiated by earthquakes, more than 70 percent of which are those earthquake greater than Safe Shutdown Eathquake (>SSE).

TABLE 2-1

ESTIMATED FREQUENCY OF CORE DISRUPTIVE ACCIDENTS (CDA'S) FOR CRBRP

Accident Category[a]	Estimated CDA Reccurance[b] Frequency, Initiated by Internal (Random) Events Mean per Year (Percent of Total)	Estimated CDA Recurrance Frequency by External Events, Mean Per Year (Percent of the Total)
ULOF	5.5×10^{-5} (24)	1.0×10^{-7} (<1)
UTOP	1.7×10^{-6} (1)	3.9×10^{-8} (<1)
UTOP & ULOF	2.2×10^{-6} (1)	1.1×10^{-6} (1)
ULOHS	1.1×10^{-5} (5)	1.6×10^{-9} (<1)
ULOCA	4.3×10^{-11} (<1)	8.9×10^{-8} (<1)
Total Unprotected	7.0×10^{-5} (31)	1.3×10^{-6} (1)
LOHS	1.3×10^{-4} (58)	7.0×10^{-7} (<1)
LOCA	1.0×10^{-6} (<1)	4.9×10^{-6} (2)
TOP	1.0×10^{-7} (<1)	1.8×10^{-5} (8)
Total Protected	1.3×10^{-4} (58)	2.4×10^{-5} (10)
Total	2.0×10^{-4} (89)	2.5×10^{-5} (11)

a) See definitions on page 4-13

b) Total number of challenges = 23 mean frequency/year

These three dominant sequences all together cause almost 75 percent of the frequency of core disruptive accidents, and close to 60% of the frequency of unfiltered releases from the containment.

Additional results of this analysis are frequency estimates of the CRBRP containment releases. Table 2-2 shows the containment release frequencies.

For this analysis nine different containment response modes have been defined; some represent containment failure and some imply successful operation of the containment as designed. The most frequent responses of the containment are early (before 24 hours) or late (after 24 hours) venting of the containment atmosphere through the clean-up system (scrubbers and filters). This occurs at a mean frequency of 2.4×10^{-4} per year or nearly 95 percent of the time a CDA occurs. A considerable uncertainty is associated with the timing of different physical phenomena, and, therefore, the distribution of the filtered releases betweeen early and late release is subject to great uncertainty until further investigations and/or studies are carried-out.

Seven modes of unfiltered containment releases have been identified for the CRBRP containment. These modes vary over a wide range of release characteristic, from a slowly leaking containment after a core disruptive accident and vessel failure (low consequence), to a gross instantaneous failure of the containment due to a CDA initiated missile or sodium spray fire (high consequence).

The mean frequency of the unfiltered release from CRBRP containment is estimated to be 1.3×10^{-5} per year or 5 out of every one hundred CDA events. The most frequent of these release modes are either an overpressure failure caused by total loss of AC power or leakage accross the confinement/containment if a venting cannot be established when necessary and the containment maintains its integrity. These two release modes comprise 73% of the frequency of the unfiltered releases.

Table 2.2

SUMMARY OF CRBRP CONTAINMENT RELEASE FREQUENCIES

Containment[a)] Response Mode	Description of Containment Response Model		Mean Frequency Per Year Caused by Internal Accidents (Percent of Total Release)	Mean Frequency Per Year Caused by External Events (Percent of the Total Release)
R1 (MIS)	Early Containment Failure due to CDA Initiated Missile		1.2×10^{-8} (<1)	2.0×10^{-9} (<1)
R2 (EOP)	Early Containment Overpressure Failure due to CDA Initiated Sodium Fire		2.4×10^{-8} (<1)	3.9×10^{-9} (<1)
R3 (CI)	Containment Isolation Failure		5.1×10^{-7} (<1)	1.0×10^{-7} (<1)
R4 (EDV)	Early Dirty Vent, Unfiltered Release		1.7×10^{-7} (<1)	1.3×10^{-6} (0.5)
R5 (OP)	Containment Overpressure Failure		1.2×10^{-6} (0.5)	3.3×10^{-6} (1.3)
R6 (LDV)	Late Dirty Vent, Unfiltered Release		1.7×10^{-7} (<1)	1.3×10^{-6} (0.5)
R7 (LKG)	Leakage Accross the Steel Shell		3.4×10^{-7} (<1)	4.2×10^{-6} (1.6)
R8 (ECV)	Early Clean Vent Through the Scrubbers and Filters		1.0×10^{-4} (38.5)	2.4×10^{-5} (9)
R9 (LCV)	Late Clean Vent Through the Scrubbers and Filters		1.0×10^{-4} (38.5)	2.4×10^{-5} (9)
Total Frequency of Release		Filtered	2.0×10^{-4} (77)	4.8×10^{-5} (18)
		Unfiltered	2.5×10^{-6} (1)	1.0×10^{-5} (4)

a) See definitions on page 4-26

The mean frequency of hypothetically based ultra-high energetic CDAs causing early releases by missile ejections is estimated to be about 1.4×10^{-8}/yr or one per 20,000 CDAs.

Additionally, early containment failures could possibly occur because of a large sodium spray fire. The ejection of sufficient sodium spray into the upper reactor containment building would, as in the missile case, require an ultra-high energetic CDA. In this sequence, however, there is also the consideration of how much oxygen would be available to react with the sodium spray. Some investigators (2) have limited the oxygen supply to that in the head access area, and thus, limited the extent of potential spray fires. This analysis presumes that a spray fire might not be terminated early because of oxygen starvation.

Within the assumption of this analysis there is a small possibility of containment of radionuclides within the primary system given a CDA. The mean frequency of a CDA and core retention within the primary system is about 3.4×10^{-8} per year or once out of every seven thousand CDAs. This is primarily due to estimates of little or no core retention capability of the reactor vessel following a CDA.

Section 3
METHODOLOGY OVERVIEW

In this study probabilistic methods are employed to identify potential accident sequences and to evaluate the expected frequency of the CRBRP containment responses. The methodology involves the use of event trees to define the possible accident sequences. These sequences are then quantified using simplified system fault models to identify potentially significant sequences.

As a first step, potential accident initiators were derived from appropriate safety and reliability studies. Possibility of omitting significant sequences was minimized by a thorough review of their documents by those experienced with PRAs for special classes of reactors as well as for LMFBR's.

Once the accident initiators were identified, five separate event trees were constructed for different phases of accidents in order to follow their progression in a systematic manner. These event trees are:

- Accident Initiator Event Tree
- Shutdown Heat Removal Event Tree
- Reactor Vessel Event Tree
- Reactor Cavity Event Tree
- Upper Reactor Containment Building Event Tree

These event trees identify potential scenarios which can cause containment failure. The frequency of potential Core Disruptive Accidents (CDAs) were evaluated with the accident initiator event tree. This event tree was evaluated for each one of the identified accident initiator groups to estimate the frequency of different CDAs.

Once the frequency of CDAs are know, this information is converted into different initial core damage category frequencies. Four categories of core damage were defined ranging from benign core melt to an extremely high energetic core distruptive accident with potential for failing the vessel and containment building directly. These core damage categories were then used as initiating events in the reactor vessel event tree. The phenomenological events inside the reactor vessel produce the outcomes which are grouped into four different modes of vessel response.

The four vessel response modes are initiating events in the reactor cavity event tree. Different phenomenological events in the cavity result in four modes of cavity response.

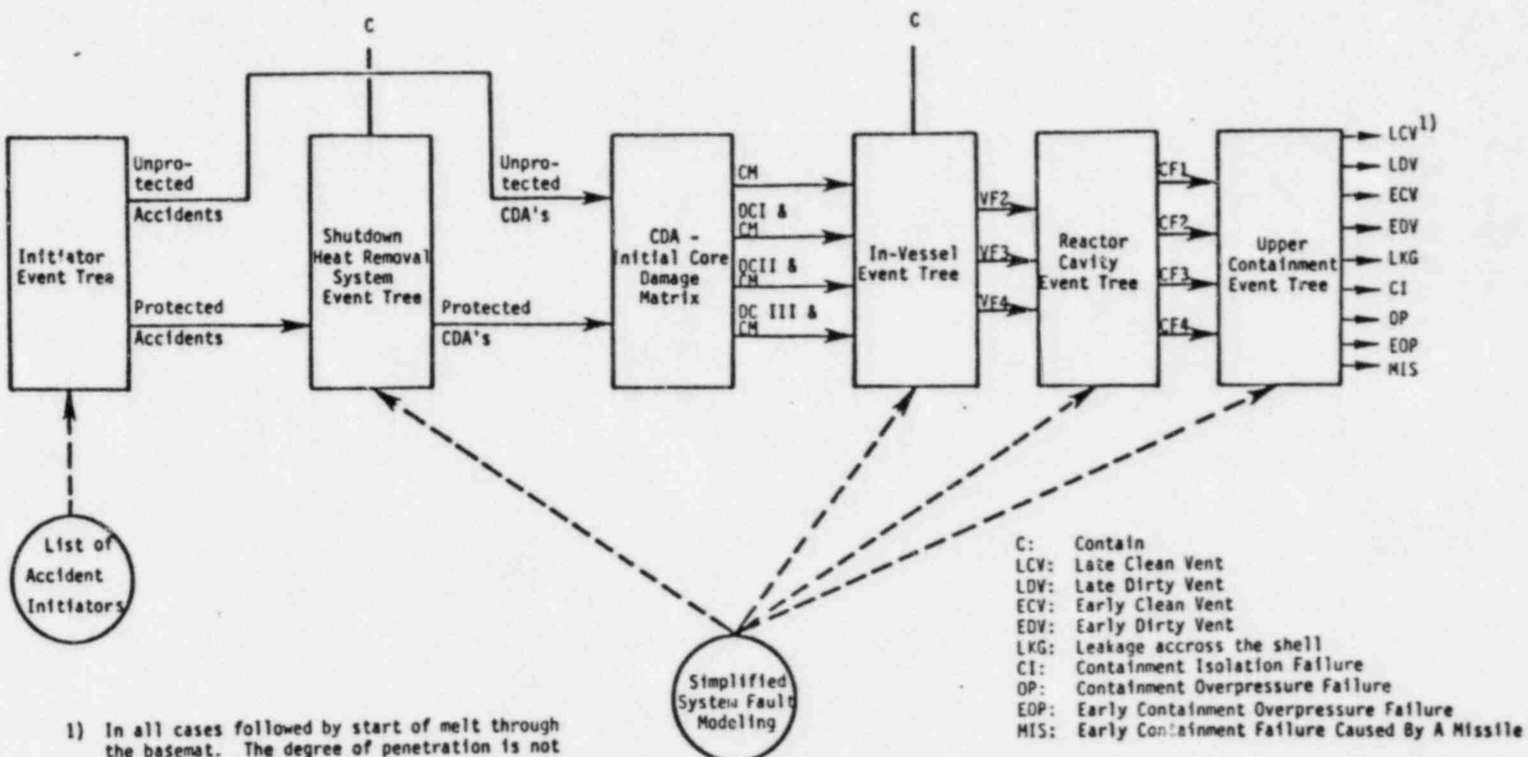
The four cavity response modes then initiate events in the upper RCB which are then analyzed using the upper RCB event tree. The outcome of the upper RCB event trees are the containment response modes. Nine different modes of containment response are identified and evaluated in this study.

Once the accident sequences are defined by means of event trees, simple fault tree models were constructed to represent each safety function (i.e., to define event tree branch points). The event trees togeter with the simplified fault trees form the basis of a safety logic model, which evaluates the frequency of these sequences and containment responses.

Figure 3.1 shows the interrlationship among the steps of this study to model the CRBRP accidents.

Figure 3.1

CRBRP ACCIDENT SEQUENCE MODELING FLOW DIAGRAM



1) In all cases followed by start of melt through the basemat. The degree of penetration is not analyzed.

Section 4.0
ACCIDENT SEQUENCE MODELING

This section describes the systematic approach to define and model the potential accident sequences which can result in release of radionuclide from the containment.

4.1 DESCRIPTION OF KEY PLANT SAFTEY FUNCTIONS

The accident analysis of the CRBRP requires the investigation of the following systems or functions to assess the frequency of their failure:

- Turbine Bypass System (TBS)
- Plant Protection System (PPS)
- Emergency Shutdown System (SCRAM)
- Pump Trip (PT)
- Shutdown Heat Removal System (SHRS)
- Containment Isolation (CI)
- Containment Annulus Cooling System (AC)
- Containment Vent and Purge System (VPS)
- Clean-Up System, Scrubbers and Filters (CU)
- Electric Power System (EPS)

This section is intended to provide a brief description of these systems and their functions. Further information can be obtained in various documents published concerning the CRBRP (References, 1, 2, and 3).

4.1.1 Turbine Bypass System (TBS)

The TBS provides electrical power to the plant auxiliary loads during loss of load events, by keeping the turbine generator online. Upon load rejections, the TBS is designed to initiate steam dumping by bypassing steam to the condenser and throttling down on the steam flows to the turbine generator to reduce the power at 3%/min. from 100% to 15% in order to supply plant auxiliary electric loads. The present design of CRBRP accomplishes this objective except when loss of off-site power results from emergency or faulted events.

4.1.2 Plant Protection System (PPS)

For the purposes of this study the PPS is defined as part of the reactor shutdown system which detects any condition in the plant which may affect the generation or orderly transport of heat. This system also initiates the appropriate response to these abnormalities in the plant to mitigate their consequences. In most cases this response is emergency shutdown by activating the pump trip and SCRAM systems. This system consists of two separate logic trains which provide functional redundancy and partial diversity.

4.1.3 Shutdown System (SCRAM)

SCRAM refers to the mechanical subsystem of the reactor shutdown system, and includes the primary and secondary control rod systems and their actuators. The primary SCRAM system has 9 rod assemblies and their associated actuator mechanisms. SCRAM is initiated by removing power from the stator winding of a stepping motor for each rod. The primary rods drop into the core by gravity, assisted by springs. The secondary SCRAM system has 6 rods and their associated actuator mechanisms. SCRAM is initiated by removing power from two of three solenoid valves for each of the six assemblies. In addition, the secondary rods are aided by coolant flow for insertion. The secondary SCRAM system is not designed for a safe-shutdown earthquake. Primary control rods are used for both Control and SCRAM, but the secondary rods are only used foor SCRAM.

4.1.4 Pump Trip (PT)

The purpose of this function is to shift the primary and intermediate sodium pump drives from the main motors to the pony motors when a PPS signal shuts down the reactor. Each pump circuit has two breakers in series which receive signals from the PPS. The two redundant trains of PPS provide separate signals to the breakers which removes the power from the motor of the breaker and, therefore, trips the pump.

4.1.5 Shutdown Heat Removal System (SHRS)

This system is designed to remove the sensible and decay heat from the reactor following a reactor shutdown. This heat can be transferred to the ultimate heat sink via two different paths, each operating in forced or natural circulation mode. One path is via the primary and intermediate transport loops by using the steam-water subsystem as the ultimate heat sink. The other path is the Direct Heat Removal System (DHRS) and deposits the heat into the atmosphere through air blast heat exchangers.

4.1.5.1 Shutdown Heat Removal Via Main Heat Transport System (MHTS)

The MHTS subsystem can transfer the decay heat to normal or emergency heat sinks in the steam-water subsystem via three primary and intermediate sodium loops. Decay heat can be removed via MHTS with either forced or natural circulation in the primary and intermediate sodium loops. Heat from each intermediate loop is then transferred to the steam-water subsystem through a steam generator system consisting of two evaporator modules and one steam superheater module on each loop. The heat then can be removed from the steam generators by either main feedwater or auxiliary feedwater system. The main feedwater system uses three motor-driven feedwater pumps and three motor-driven condensate pumps and takes water from the condenser hot well or condensate water storage tank. The auxiliary feedwater system has two 50% motor-driven pumps and one 100% steam turbine-driven pump, and they take suction from the Protected Water Storage Tank (PWST) with the condensate storage tank providing an unprotected source of make-up. The steam is then vented directly into the atmosphere and it is also condensed through the three Protected Air-Cooled Condensers (PACCs). The venting, used for short-term heat removal, is provided through two power-operated relief valves on each steam drum. One PACC is associated with each steam generator. Saturated steam is supplied to each PACC from its related steam drum and is returned as saturated water to the steam drum, which is at a lower elevation, by gravity flow. Two fans, each on a separate class IE power source, force the air across the PACC tubes.

For decay heat removal via MHTS in the natural circulation mode, all primary and intermediate loops should remain intact and active operation of either main feedwater or auxiliary feedwater is required to remove the heat from the steam generators.

However, studies are being conducted which show that the decay heat can be removed adequately from the steam drums using the PACCs in the natural draft operation mode. In this mode natural circulation is established between each steam drum and its associated PACC, with the help of natural draft across the PACC tubes, and higher elevation of PACC with respect to its associated steam drum.

4.1.5.2 Shutdown Heat Removal via Direct Heat Removal System (DHRS)

If the process of decay heat removal through MHTS is unavailable, the DHRS can remove the decay heat from the reactor vessel provided that at least one of the primary loop pony motors is operational to provide coolant mixing inside the reactor vessel, and all primary loops are capable of providing a flow path. This system draws spill-over sodium from an overflow vessel and circulates it through the overflow heat exchanger by two Electro Magnetic (EM) pumps. The heat is then extracted from the overflow heat exchanger via two NaK loops and is transferred to the atmosphere using two air blast heat exchangers. Both NaK loops are necessary to provide adequate heat removal which makes DHRS a single train non-redundant system. The DHRS must be initiated manually.

4.1.6 Containment Systems

The reactor confinement/containment system is designed as a last barrier to prevent the release of radioactivity to the environment. Several auxiliary systems and engineered safety systems are provided to mitigate the consequences of an accident which may threaten containment integrity.

4.1.6.1 Containment Isolation System

The containment isolation system is designed to seal-off all reactor containment building penetrations in the event of an accident in order to prevent any escape of radioactive material from within the containment building to the environment. The system is comprised of isolation valves with their control and actuating equipments. The control system includes both automatic and manual operation in most cases. The isolation valves and their associated actuators close on loss of air or electric power. All lines, except those that lead into closed Class II Systems (i.e., the IHTS), which penetrates the containment have redundant isolation valves in series, with one located within and one located outside the RCB. This ensures operation following either internal or external accidents.

4.1.6.2 RCB Annulus Air Cooling System

This system is provided to maintain the temperature of the steel containment shell and confinement concrete structure so as to prevent the structural failure of the confinement/containment system.

Six redundant 133,000 CFM (cubic feet per minute) fans are provided (3 on each train) which push the outside air into the annulus between the steel containment and concrete confinement. The annulus is partioned to provide a spiral air flow path discharing at the top of the confinement dome. The intake and the exhaust opening are protected against missiles, and screens are used at the intake to protect the fans from the debris. The fans use redundant power sources from either off-site or emergency power. The Reactor Containment Building (RCB) annulus air cooling system must be manually initiated.

4.1.6.3 Reactor Containment Building Vent and Purge System

This system is designed to relieve the containment pressure build-up within the upper RCB. The system consists of two vent lines with redundant fans to provide forced venting of the containment atmosphere. This system is designed to remain functional with sodium aerosols entering the system. The RCB vent and purge system in combination with clean-up system maintain a 1/4-inch of water gauge (or 0.622 m bars) negative pressure inside the containment after the initial ventdown. The vent system is connected to the clean-up system through redundant pipes. The purge system is operated by opening redundant isolation valves after containment is at negative pressure. Check valves and narrow range pressure instrumentation interlocks on the purge lines prevent backflow from the containment. Both vent and purge requires remote manual actuation from the control room.

4.1.6.4 Reactor Containment Building Clean-Up System

The RCB clean-up system is comprised of three filtration stages located on the vent lines before venting through the discharge at the top of the confinement dome. There are two 100% redundant filter units consisting of a heating coil, demister, prefilter bank (jet scrubber) and HEPA (wet scrubber) filter bank. The system is required to remove 99% of solids and/or liquid radioactive material and 97% of the vapors. The system is designed to remain functional with predictable sodium aerosol ingestion and contained radioactivity and heat generation from fission products.

4.2 ACCIDENT INITIATING EVENTS

The starting point for modeling of the accident sequences which result in releases of radionuclides from the containment is to identify the initiating events which may start such a sequence of events.

There are two basic causes for initiating an accident:

- An increase in the reactor power beyond the design capacity of the heat transport system (overpower transients).
- Imbalance between the heat produced in the core and the heat being removed from the core due to inadequate (or loss of) heat removal (undercooling transients).

either of these two conditions require the shutdown of the reactor and remova of the decay heat.

In order to account for all the possible initiating events in a systematic fashion six different accident initiating event categories were defined. These categories are defined for each subsystem which is part of the plant normal operation for generation and transport of heat in a controlled manner.

These categories are:

I - Operator Subsystem

II - Reactor Subsystem

III - Heat Transport Subsystem

IV - Steam/Water Subsystem

V - Electrical/Control Subsystem

VI - External Events

The first category is the accidents which do not require fast automatic shutdown of the reactor, but the reactor must be shutdown manually for repair. The secondary category of accidents are the ones happening in the reactor system such as fuel failure or other reactivity related accidents. The accidents initiated in the heat transport system (primary or secondary sodium loops) are grouped in category three. Category four includes the accidents involving steam, feedwater or condensate systems. The accidents which start in the electrical or control systems are in category five. And the last category includes the accidents which are initiated due to external causes. Table 4.1 shows a list of these accident initiators and their estimated mean recurrence frequency.

Table 4.1

LIST OF ACCIDENT INITIATING EVENTS FOR CRBRP

Accident Initiator	Estimated Frequency Mean Per Year	Reference
I. Operator System		
1. Normal Shutdown with SHRS Available	3.3	CRBRB Safety Study
2. Normal Shutdown with one HT loop unavailable	2.2	CRBRP Safety Study
3. Normal Shutdown with DHRS unavailable	6.3×10^{-1}	CRBRP Safety Study
II. Reactor Safety		
4. Local Fault Propagation, Subassembly Faults	2.7×10^{-4}	CRBRP Safety Study
5. Fuel Pin Failure, Local Radial Motion	2.7×10^{-5}	CRBRP Safety Study
6. Core Support Structure Failure	5.2×10^{-7}	CRBRP Safety Study
7. Large Scale Core Motion	6.2×10^{-6}	CRBRP Safety Study
8. Loss of Hydraulic Holdown	2.7×10^{-4}	CRBRP Safety Study
9. Single CR Assembly Withdrawal, Low Speed	3.8×10^{-3}	CRBRP Safety Study
10. Single CR Assembly Withdrawal, High Speed	2.7×10^{-6}	CRBRP Safety Study
11. Control Assembly Group Withdrawal, Low Speed	3.8×10^{-2}	CRBRP Safety Study
12. Control Assembly Group Withdrawal, High Speed	8.0×10^{-7}	CRBRP Safety Study
13. Voiding or Gas Bubble in the Core	2.7×10^{-5}	CRBRP Safety Study
14. Moderator in the Coolant	2.7×10^{-5}	CRBRP Safety Study
15. Spurious DHRS Injection, Valve Faults	2.9×10^{-3}	NUREG/CR-2681
16. Uncontrolled CR Assembly Drop, CRDM Faults	6.5×10^{-1}	EPRI NP-2230

Table 4.1

LIST OF ACCIDENT INITIATING EVENTS FOR CRBRP (Continued)

Accident Initiator	Estimated Frequency Mean Per Year	Reference
III. Heat Transport System		
17. Primary Pipe Rupture	5.2×10^{-6}	CRBRP Safety Study
18. Reactor Vessel Rupture	5.2×10^{-6}	CRBRP Safety Study
19. Loss of Flow in 1 Primary Loop	7.5×10^{-1}	NUREG/CR-2681
20. Loss of Flow in 2 Primary Loops	2.5×10^{-2}	NUREG/CR-2681
21. Loss of Flow in all 3 Primary Loops	1.3×10^{-2}	NUREG/CR-2681
22. Spurious PPS Signal	8.8	CRBRP Safety Study
23. Intermediate Pipe Rupture	5.2×10^{-6}	CRBRP Safety Study
24. Intermediate Heat Exchanger Rupture	5.2×10^{-6}	CRBRP Safety Study
25. Loss of Flow in 1 Intermediate Loop	7.5×10^{-1}	NUREG/CR-2681
26. Loss of Flow in 2 Intermediate Loops	2.5×10^{-2}	NUREG/CR-2681
27. Loss of Flow in All 3 Intermediate Loops	1.3×10^{-2}	NUREG/CR-2681
28. Failure of the Rupture Disk in 1 Int Loop	1.0	WARD-D-0118
29. Failure of the Rupture Disks in 2 Int Loops	3.8×10^{-2}	WARD-D-0118
30. Failure of the Rupture Disks in all 3 Int Loops	2.5×10^{-2}	WARD-D-0118
31. Drain Valve Failure Dumping Na into the IHTS	2.7×10^{-5}	Failure Data

Table 4.1

LIST OF ACCIDENT INITIATING EVENTS FOR CRBRP (Continued)

Accident Initiator	Estimated Frequency Mean Per Year	Reference
IV. Steam/Water System		
32. Loss of One Steam Generator Loop	1.3×10^{-1}	CRBRP Safety Study
33. Loss of All 3 Steam Generator Loops	3.8×10^{-5}	CRBRP Safety Study
34. Steam Generator Tube Rupture (Leak)	1.3×10^{-1}	CRBRP Safety Study
35. Steam Pipe Rupture	2.0×10^{-3}	Zion Prob. Safety Study
36. Loss of Main Feedwater (incl FW Pipe Rupture)	2.6	NUREG/CR-2681
37. Loss of Main Condenser	3.3×10^{-1}	CRBRP Safety Study
38. Turbine-Generator Trip	1.1	CRBRP Safety Study
V. Electrical/Control System		
39. Loss of DC Power	5.2×10^{-7}	CRBRP Safety Study
40. Loss of I & C to Vital Plant Components (due to fire)	6.2×10^{-5}	CRBRP Safety Study
VI. External Events		
41. Loss of Offsite Power	1.4×10^{-1}	EPRI NP-2230
42. Operating Basis Eathquake (OBE)	2.3×10^{-3}	CRBRP PSAR
43. Safe Shutdown Eathquake (SSE)	2.4×10^{-4}	CRBRP PSAR
44. Greater than SSE (BFE)	5.5×10^{-5}	CRBRP PSAR
TOTAL	23.1	

4.3 EVENT TREE DEVELOPMENT

This section describes the event trees developed to define the possible accident scenarios for CRBRP. The following five event trees were constructed:

- Accident Initiator Event tree
- Shutdown Heat Removal Event tree
- Reactor Vessel Event tree
- Reactor Cavity Event Tree
- Upper Reactor Containment Building Event tree

4.3.1 Accident Initiator Event Tree

This event tree defines the relationship between each accident initiator and the accident category (CDA) resulting from it. A generalized accident initiator event tree is shown in Figure 4.1. The event heading descriptions and their success criteria are defined in Table 4.2. The outcome of this event tree will be different types of accident categories (or CDAs), such as protected Loss Of Heat Sink (LOHS), Unprotected Loss Of Flow (ULOF), etc.

4.3.2 Shutdown Heat Removal Event Tree

This event tree defines the modes of failure of shutdown heat removal system. The starting event is a protected accident, i.e., an accident with successful shutdown of the reactor, and the outcoming events are either termination of the accident, Cold Shutdown Available (CSA) or a Protected Core Disruptive Accident (PCDA). The Unprotected Core Disruptive Accidents (UCDA's) will be the outcome of the initiator event tree.

Figure 4.1

Initiator Event Tree for CRBRP

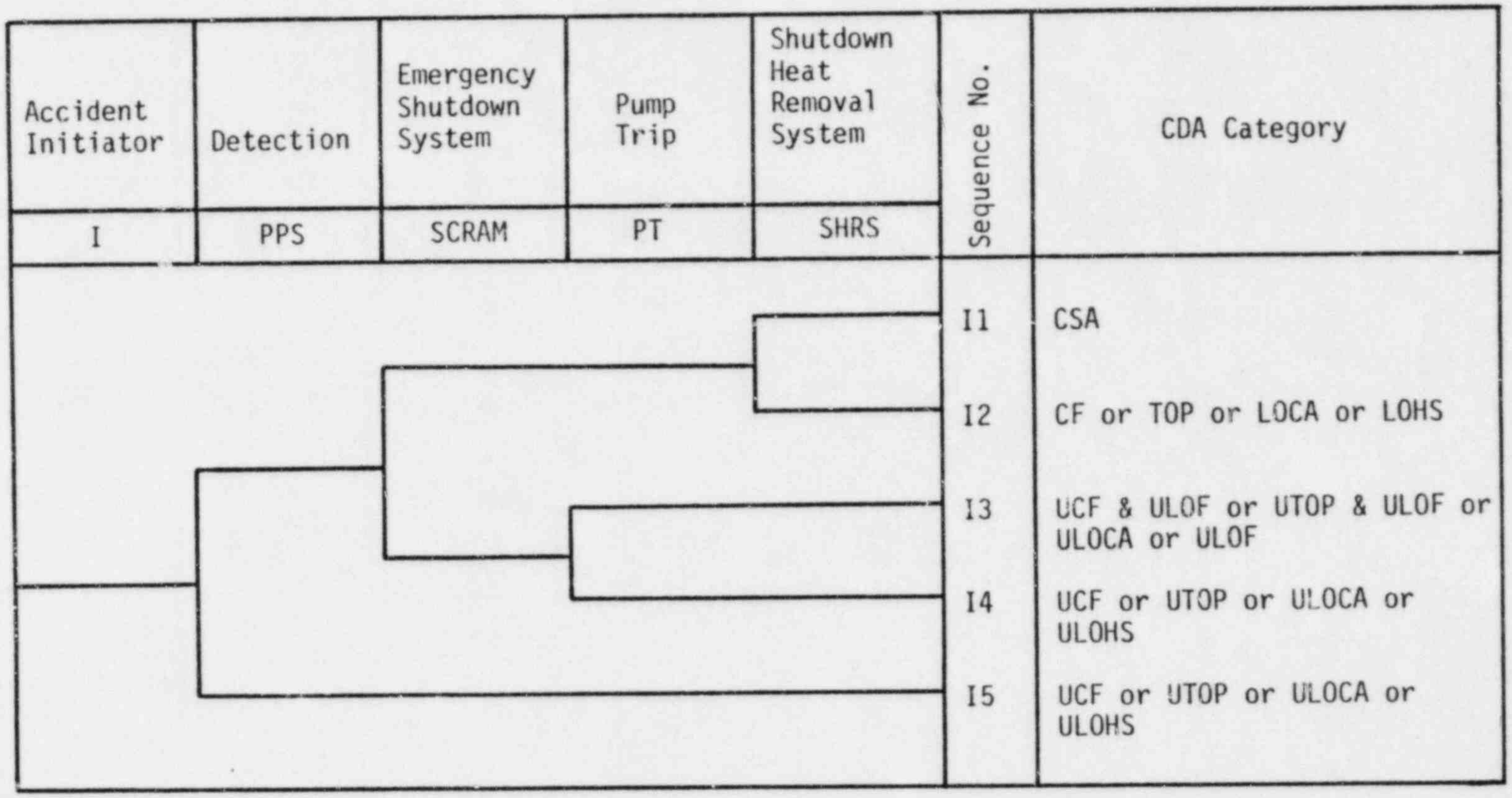


Table 4.2

ACCIDENT INITIATOR EVENT TREE HEADINGS DESCRIPTION AND SUCCESS CRITERIA

Heading	Description	Success Criteria
PPS	Detection - Plant Protection Signal	To detect and provide signal at least on one of the two redundant channels.
SCRAM	Emergency Reactor Shutdown	Opening of at least 2 out of 3 SCRAM breakers to release the primary control rods or opening of at least 2 out of 3 solenoid operated valves to vent the argon pressure, actuating the SCRAM latch in each secondary rod. The number of rods required will depend on the power level and the type of incident.
PT	Primary Sodium Pump Trip	Trip of all three primary sodium pumps given an emergency shutdown signal.
SHRS	Shutdown Heat Removal System	Success criteria for the SHRS is defined in Section 4.3.2

Four protected and seven unprotected core disruptive accidents are the outcome of these two event trees. The protected accidents are:

- CF - Core Fault, the accidents which are initiated within the fuel such as a fuel pin failure and failure of SHRS after successful SCRAM.
- TOP - Transient Overpower, a reactivity insertion transient followed by successful SCRAM and failure of SHRS.
- LOCA - Loss of Coolant Accident, primary system ruptures spilling the primary sodium into the cavity, drops the sodium level in the tank. The reactor is shutdown but SHRS fails.
- LOHS - Loss of Heat Sink, the accident starts in the heat sink (secondary sodium loops or steam/water system or electrical/control system). The fuel is initally intact, no overpower transient occurs and the primary system boundary remains intact maintaining the level of sodium. The reactor is shutdown and the SHRS fails.

The seven unprotected accidents are:[1)]

- UCF - Unprotected Core Fault, starts the same way as CF, but no SCRAM and no primary pump trip.
- UCF & ULOF - Unprotected Core Fault and Loss of Flow, same as UCF with trip of primary sodium pumps.
- UTOP - Unprotected Transient Overpower, a reactivity insertion transient and failure to SCRAM the reactor and trip the primary pumps.
- UTOP & ULOF - Unprotected Transient Overpower and Loss of Flow, same as UTOP with trip of primary sodium pumps
- ULOCA - Unprotected Loss of Cooling Accident, same as LOCA except the failure to shutdown the reactor
- ULOF - Unprotected Loss of Flow, an undercooling accident with failure to shutdown the reactor and trip the primary pumps
- ULOHS - Unprotected Loss of Heat Sink, same as ULOF except that the primary sodium pumps do trip

1) The LOF driven TOP is not considered in this study

Figure 4.2 shows the SHRS event tree. The description of the event tree headings and their success criteria is presented in Table 4.3.

4.3.3 CDA-Initial Core Damage Matrix

The purpose of this matrix is to define the distribution of the initial energetics of core disruptive accidents. The energy of a core disruption is important in determining structural and mechanical integrity of the primary system (i.e., reactor vessel head) and the possiblity of early containment failure due to extermely high initial energetics, or the timing of thermal vessel failure.

The element A_{ij} of this matrix is the conditional probability that the core disruptive accident CDA_i will have an energetic of the magnitude ICD_j.

The outcome of the initiator and shutdown heat removal event trees, i.e., CDA Vector is multiplied by the CDA-Initial Core Damage (CDA-ICD) matrix to obtain the initial core damage vector. This vector defines the core disruptive accidents in terms of four energy categories:

- CM - Melting of the core, no energetic disassembly (like LWR core melt)
- DCI - Benign core disruption, (partial) fuel dispersal, no vessel head damage
- DCII - Moderate core disruption, fuel dispersal in the vessel, moderate head seal leakage
- DCIII - Large core disruption, extensive head damage, permitting free communication of gases and liquids between vessel and upper RCB

These categories are the input events for the reactor vessel event tree. Figure 4.3 shows the CDA-ICD matrix.

The initial core damage categories are the responses of the core in the initiation phase of the core disassembly and do not reflect the following energetic disassemblies which may happen due to energetic recriticality and/or Fuel Coolant Interaction (FCI). These phenomena are considered in the reactor vessel event tree. In other words the transition and termination phases of the disassembly and their effect on the primary system integrity is reflected in the reactor vessel event tree.

Figure 4.2

SHUTDOWN HEAT REMOVAL SYSTEM EVENT TREE FOR CRBRP

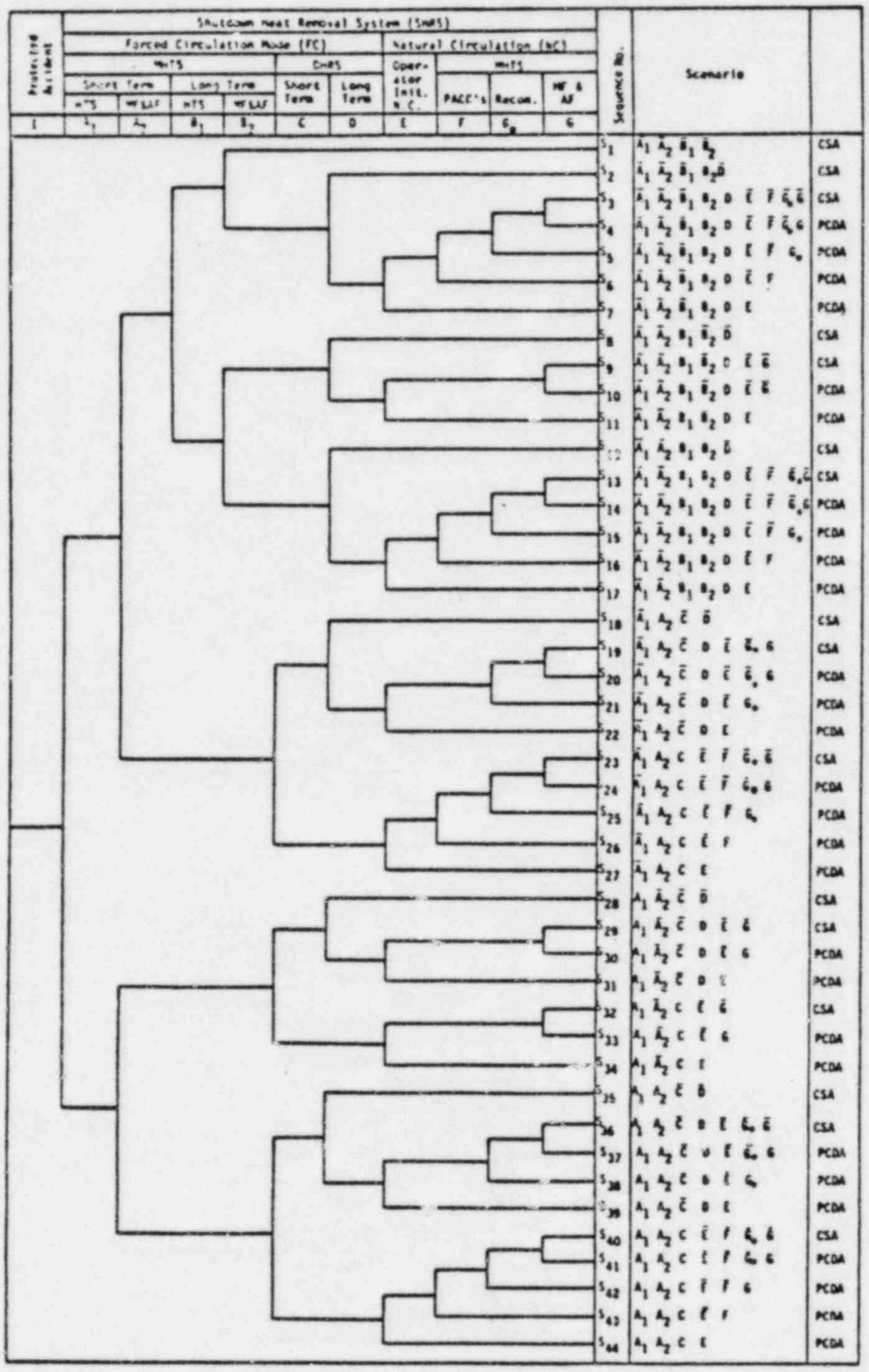


Table 4.3

SHRS EVENT TREE HEADINGS DESCRIPTION AND SUCCESS CRITERIA

Heading	Description	Success Criteria
HTS (S)	Short Term forced decay heat removal through main heat transport loops	Forced operation of one heat transport loop (one pony on of the primary loops and the pony on its associated secondary loop) for four hours
MF&AF (S)	Short term forced decay heat removal by main or auxiliary feedwater system	Either main feedwater system or one turbine driven AFW pump or two motor driven AFW pumps help remove the decay heat for four hours
HTS (L)	Long term forced decay heat removal through main heat transport loops	Same as HTS (S) for after 4 hours up to 24 hours
MF&AF (L)	Long term forced decay heat dissipation in the main or auxiliary feedwater system	Either main feedwater system or one of the AFW pumps are available to dissipate the decay heat
DHRS (S)	Short term forced operation of decay Heat Removal System (DHRS)	One EM pump in the sodium loop, both NaK loops and airblast heat exchangers, and all three primary ponys are needed for successful removal of the decay heat in the first four hours

Table 4.3

SHRS EVENT TREE HEADINGS DESCRIPTION AND SUCCESS CRITERIA (Continued)

Heading	Description	Success Criteria
DHRS (L)	Long term forced operation of decay Heat Removal System (DHRS)	One EM pump in the sodium loop, both NaK loops and airblast heat exchangers, and at least one primary pony motor is needed for successful removal of the decay heat beyond the first four hours
OP (NC)	Initiation of natural circulation by the operator	Success in this event requires the operator intervention, shutting off all the primary ponys, if decay heat can not be removed by forced circulation
PACC	Dissipation of the decay heat through the Protected Air Cooled Condensers (PACC's)	The success requires operation of one of the PACC Systems in the forced or natural draft mode in the same heat transport loop. If feedwater is not available the cooldown of the system with complete natural circulation is possible for two hours before the dry-out of the steam system. However, it would take several hours to heat the system to a temperature at which core damage can occur.

Table 4.3

SHRS EVENT TREE HEADING DESCRIPTION AND SUCCESS CRITERIA (Continued)

Heading	Description	Success Criteria
REC	Recovery of main or auxiliary feedwater system	The success implies the recovery of the main or auxiliary feedwater system within 2 hours after the start of natural draft in the PACC's
MF & AF	Operation of the main auxiliary feedwater system after either is recovered given they had failed previously	The success of this event requires operation of the main feedwater pump after recovery for a period of 24 hours

Figure 4.3

CORE DISTRUTIVE ACCIDENT - INITIAL CORE DAMAGE MATRIX

Accident Classes \ Initial Core Damage Category		Thermal	Mechanical		
		Core Melt (CM)	Benign Core Disruption (DCI)	Moderate Core Disruption (DCII)	Large Core Disruption (DCIII)
Protected	CF	1.0	ε	ε	ε
	TOP	1.0	ε	ε	ε
	LOCA	1.0	ε	ε	ε
	LOHS	1.0	ε	ε	ε
Unprotected	UCF	ε	0.8	0.2	0.05
	UCF & ULOF	ε	0.8	0.2	0.05
	UTOP				
	Slow Ramp	ε	0.9	0.1	0.02
	Med. Ramp	ε	0.9	0.1	0.02
	Fast Ramp	ε	0.9	0.1	0.02
	Step	ε	0.9	0.1	0.02
	UTOP & ULOF	ε	0.8	0.2	0.05
	ULOCA	ε	0.8	0.2	0.05
	ULOF	ε	0.9	0.1	0.01
	ULOHS	ε	0.8	0.2	0.05

In a protected accident the control rods successfully insert and the core power (and energy) will drop. During the iniation phase of the protected accidents the core is disrupted by melting until possible recriticality in the later phases of the accident occurs (in most cases such a recriticality is expected to occur, Ref. 7).

On the other hand if the control rods fail to terminate the reaction, the high reactivity insertion rates will cause super-prompt criticality and extremely high power levels. If this high power level and pressure is maintained for a long enough period of time (few milliseconds) an adequate amount of energy will be created causing fuel vaporization and subsequent energetic disassembly (Ref. 3, 14, 15). It is conservatively assumed that all unprotected accidents result in core disassembly. The energetics of the disassembly, however, vary from benign to very high energies which may cause damage to the core support structures, core barrel, vessel head or even indirectly to the containment. The distribution of the initial energetics of the unprotected accidents were estimated after thorough investigation of the relevent literature (references 2, 3, 14, 15). Wherever necessary conservative assumptions were made to produce a defensible upper bound considering present knowledge of the behavior of the LMFBR core under transient conditions.

4.3.4 Reactor Vessel Event Tree

This event tree uses the initial core damage categories as initiating events and the outcome of this event tree consists of four different vessel response modes. These modes are as follows:

- VF1 - No vessel head failure, no melt-though the bottom of the vessel, fuel is retained inside the vessel
- VF2 - No vessel head failure, melt-through the bottom of the vessel
- VF3 - Moderate vessel head failure, melt-through the bottom of the vessel
- VF4 - Large vessel head failure caused by either the energetic of the initial CDA or energetic recriticality of FCI in the vessel. A spray fire or missile may be generated by ultra-high energetics in this category and directly fail the containment/confinement builiding. Melt-through of the bottom of the vessel follows.

Figure 4.4 shows the reactor vessel event tree. A brief description of the reactor vessel event tree headings and their success criteria is presented in Table 4.4.

4.3.5 Reactor Cavity Event Tree

The reactor vessel response modes are input to the cavity event tree. Because of the simplicity of the cavity system, only one question is asked once the vessel fails and the debris and other materials are transferred into the cavity. The only question of concern seems to be the distribution of the hot debris on the liner which may affect the timing of the liner failure (faster liner failure in the case of localized accumulation of the debris), and therefore, heat and gas generation due to sodium and concrete interaction. The faster the liner breaks, the earlier the venting of the upper RCB must be initiated.

Figure 4.4

REACTOR VESSEL EVENT TREE FOR CRBRP

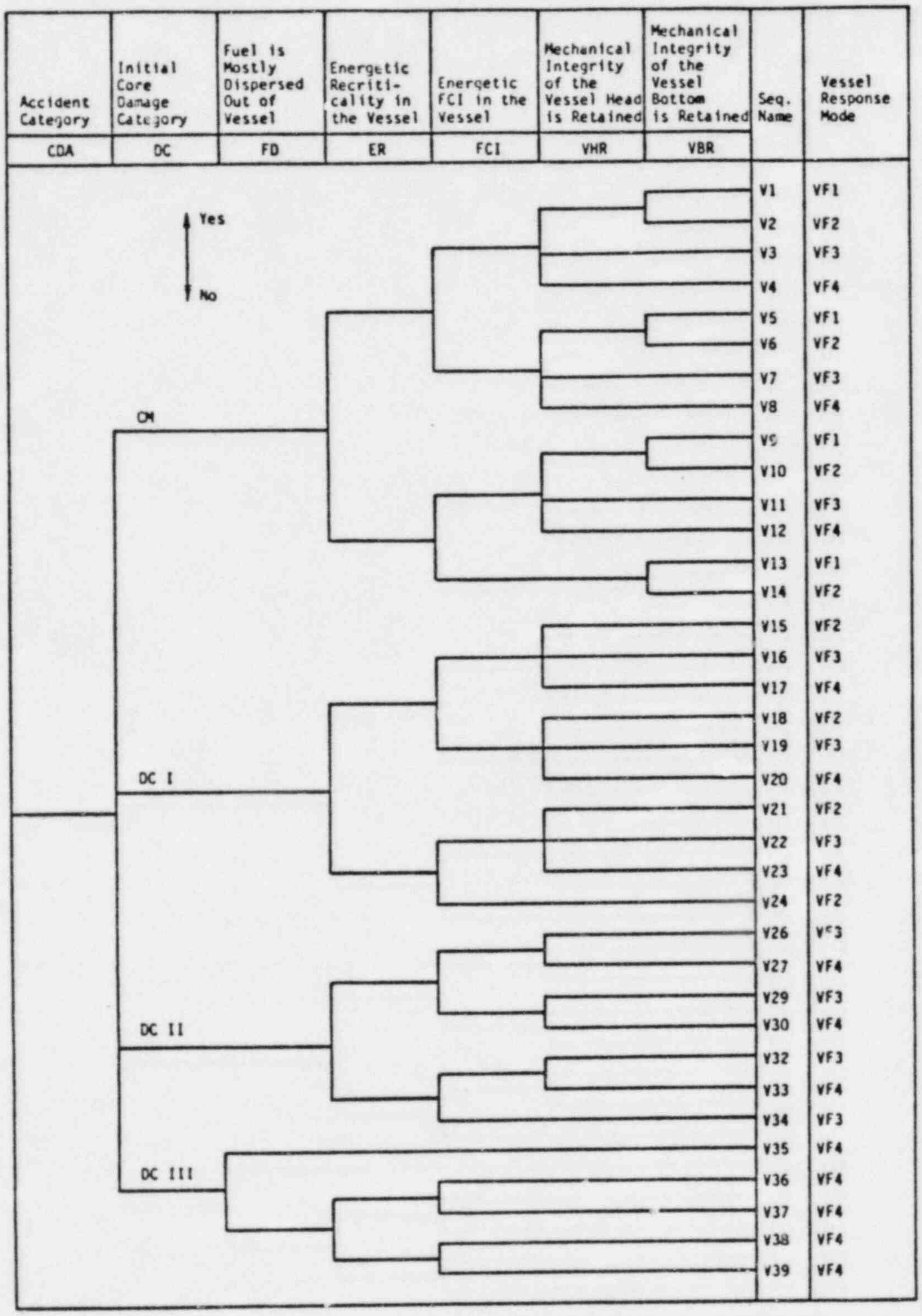


Table 4.4

REACTOR VESSEL EVENT TREE HEADING DESCRIPTION AND SUCCESS CRITERIA

Heading	Description	Success Criteria
FD	Fuel dispersion after a core distruptive accident	Success implies that the fuel is mostly dispersed out of the vessel such that there is significantly less heat-producing material left inside.
ER	In-Vessel energetic recriticality	Top branch implies that the fuel and clad motion following core disruption results in a critical mass which in turn causes an energetic reaction which m*y be sufficient to inflict damage on the system. However, recriticality of high enough energy to cause severe structural damage is usually considered very unlikely.
FCI	In-Vessel energetic fuel coolant interaction	Top branch implies that an energetic FCI occurs either as a consequence of subcooled liquid sodium entry into the mostly destroyed core area or upon contact of the melt draining downward into the lower plenum. However this question is included for completeness and an FCI of significant energy is very unlikely.

Table 4.4 (Continued)

REACTOR VESSEL EVENT TREE HEADING DESCRIPTION AND SUCCESS CRITERIA

Heading	Description	Success Criteria
VHR	Mechanical integrity of the vessel head	This heading has three branches. The top branch represents no head damage given an energetic recriticality and/or FCI. The middle branch is a moderate damage to the vessel head caused by energetic recriticality and/or FCI. And the lower branch shows large head damage and release caused by an energetic recriticality and/or FCI.
VBR	Mechanical integrity of the vessel bottom	Success implies that no melt through the bottom of the vessel occurs.

The following four reactor cavity response modes have been identified:

- CF1 - Late/moderate basemat penetration. The fuel is reasonably uniformly dispersed within the reactor cavity, no potential for coincident containment failure

- CF2 - Early/severe basemat penetration. More localized debris concentration within the reactor cavity e.g., fuel jet, no potential for coincident containment failure

- CF3 - Same as CF1, except that coincident containment failure by missile ejection or sodium spray fire due to sodiuim pool boiling is possible

- CF4 - Same as CF2, except that coincident containment failure by missile ejection or sodium spray fire due to sodium pool boiling is possible

Figure 4-5 shows the event tree for the reactor cavity.

4.3.6 Upper Reactor Containment Building Event Tree

The upper RCB event tree analyzes the physical relationship between the reactor cavity response modes and the responses of the upper RCB. The following nine modes of upper RCB responses were defined for the CRBRP containment:

- LCV - Late (after 24 hours) clean (filtered) venting of the containment

- ECV - Early (before 24 hours) clean (filtered) venting of the containment

- LDV - Late dirty (unfiltered) venting of the containment

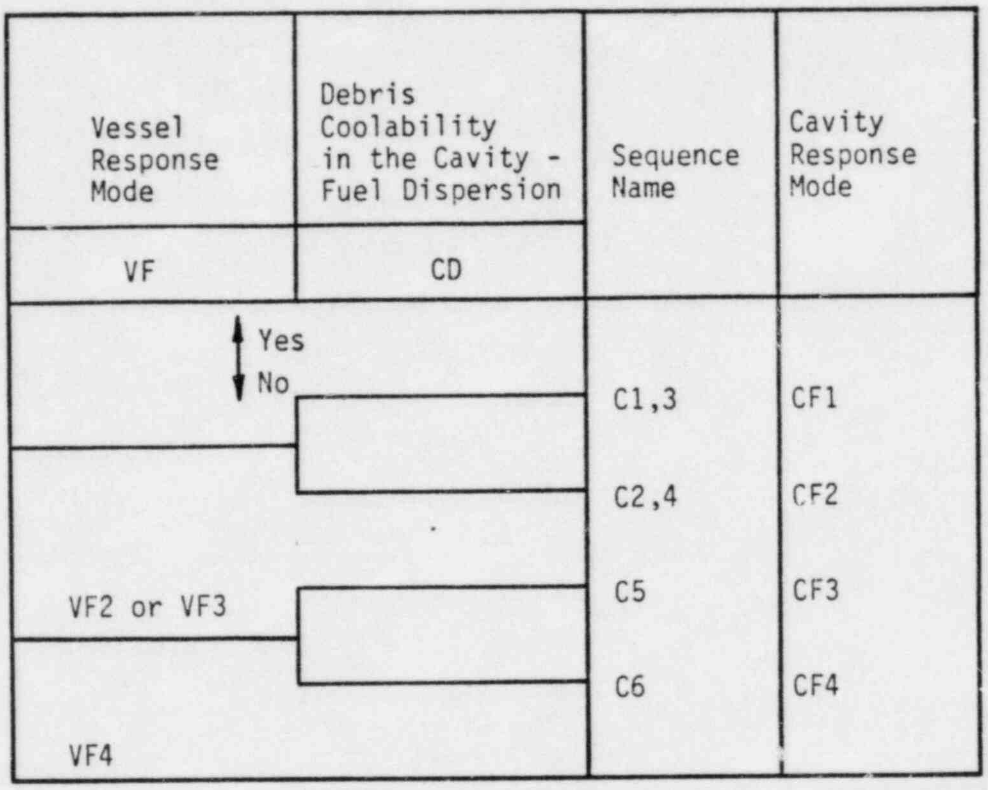
- EDV - Early dirty (unfiltered) venting of the containment

- CI - Containment isolation failure

- LKG - Leakage beyond the design basis accross the reactor containment building, containment building intact

Figure 4.5

REACTOR CAVITY EVENT TREE FOR CRBRP



- OP - Containment overpressure failure late in the accident

- MIS - Early failure of the containment caused by CDA generated missile

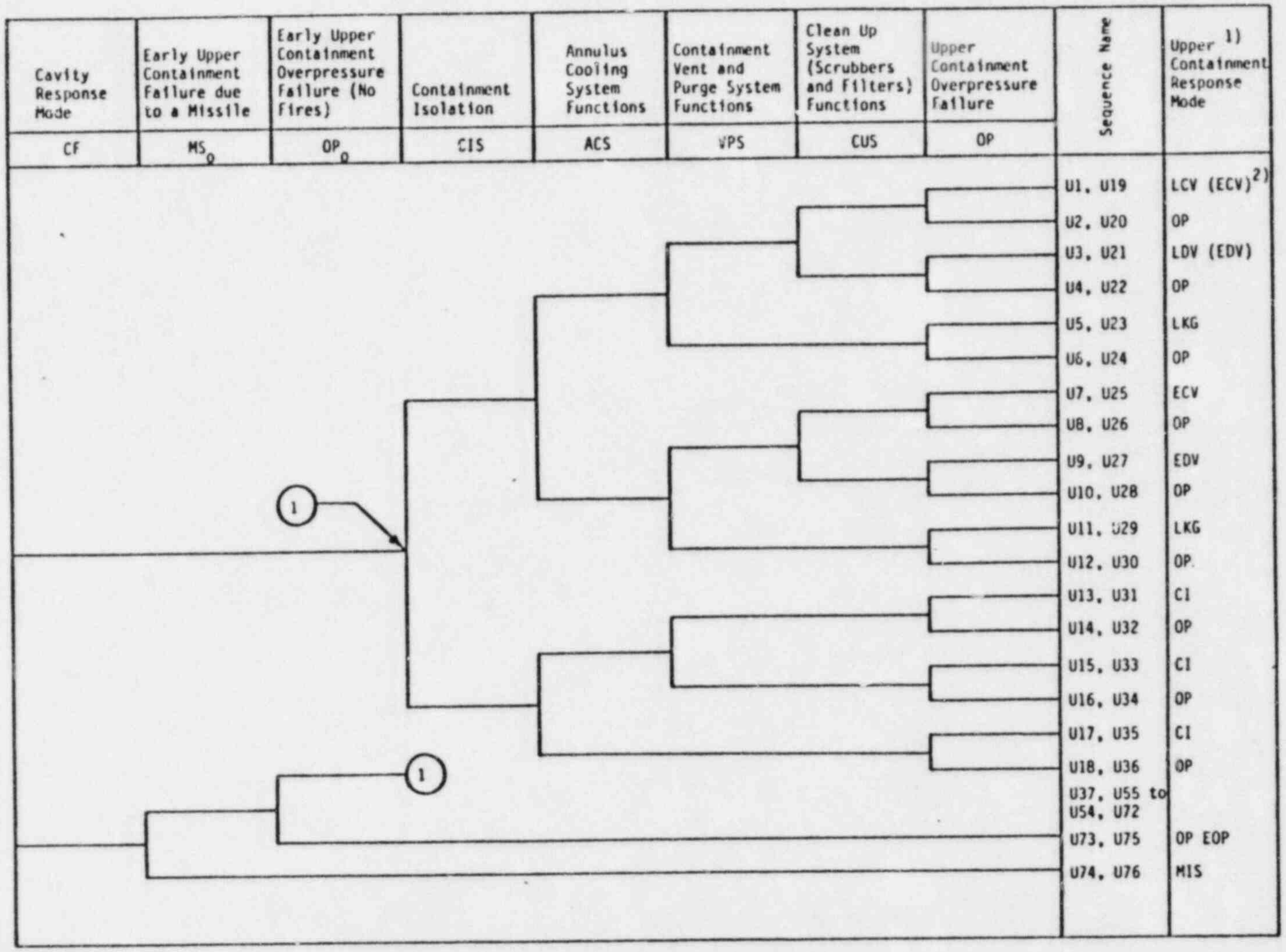
- EOP - Early containment overpressure failure due to CDA, mainly caused by sodium spray fire after large vessel head damage and transfer of large quantities of sodium sodium into the upper RCB

All of these containment release modes are followed by the debris attacking the concrete basemat. However, the degree of concrete basemat pentration was not analyzed in this study.

Figure 4.6 shows the event tree for the upper RCB. The headings for this event tree are briefly described and the success criteria for each event is given in Table 4.5.

Figure 4.6

UPPER REACTOR CONTAINMENT BUILDING EVENT TREE FOR CRBRP



1) In all cases followed by start of melt through the basemat. The degree of penetration is not investigated

2) In the event of CF2, CF3, or CF4

Table 4.5

UPPER REACTOR CONTAINMENT BUILDING EVENT TREE HEADINGS DESCRIPTION AND SUCCESS CRITERIA

Heading	Description	Success Criteria
MS_o	Early containment failure due to a CDA-generated missile	The success implies that the initial energetics of CDA or following energetic recrictica-lity and/or FCI are not capable of creating a large enough missile to fail the containment early in the accident.
OP_o	Early containment over-pressure failure	The success of this event implies that the initial energetics of CDA or following energetic re-criticality and/or FCI do not in-ject sufficient sodium vapor into the upper RCB or the condition for large sodium spray fire to fail the containment (e.g.suffi-cient oxygen to burn all the sodium which enters the upper RCB) does not exist.
CI	Containment isolation	Containment is successfully isolated to insure minimal release of radionuclides to the environment.
AC	Annulus air cooling	At least three out of the six fans are required to adequately cool the annulus.

Table 4.5

UPPER REACTOR CONTAINMENT BUILDING EVENT TREE HEADINGS DESCRIPTION AND SUCCESS CRITERIA (Continued)

Heading	Description	Success Criteria
VPS	Vent and purge system	At least one of the vent lines to provide the path and one of the two blowers to provide forced vent is required.
CU	Clean up System	At least one of the two redundant scrubber/filter systems is operational.

4.4 SYSTEM FAULT MODELING

For quantification of the accident sequences defined by the event trees, frequency of the events which are the headings of the event trees have to be estimated. The accident sequences are defined in terms of both the system unavailabilities and phenomenological uncertainties. Simplified fault trees are used to estimate the unavailability of some of the systems used to mitigate the progression of the accident. These systems are:

- Main Heat Transport Systems, including the primary and the secondary sodium loops
- Auxiliary feedwater system for short and long term operation
- Decay Heat Removal System for short and long term operation
- Protected Air Cooled Condenser
- Annulus Air Cooling System
- Containment Vent and Purge System
- Clean up System
- Electric Power System

The simplified fault models constructed for these systems is shown in Appendix A as part of the plant safety logic model.

The remaining systems were not modeled using fault trees and frequency estimates were used. These systems are:

- Plant Protection system
- Emergency Shutdown System (SCRAM)
- Pump Trip
- Steam Generator System

- Main Feedwater and Condensate System
- Containment Isolation System

Estimates were also used for the phenomenological uncertainties.

4.5 ANALYSIS OF DEPENDENT EVENTS

In designing the safety systems great care has been taken to provide several independent ways in which a safety function can be performed. Generally this concern has been expressed in terms of redundancy and diversity so that, ideally, several independent system failures are necessary to fail a safety function. Yet, events occur which may affect several functions simultaneously and jeopardize the redundancy of the mitigative functions. In general, two categories of equipment failure can be identified:

- Independent or random failures
- Dependent failures

Due to diversity achieved through single failure criteria, the loss of a mitigating function caused by independent random events have very low frequencies. The dependent falures (or common cause failures) can play a major role in the operability of the safety functions and overall safety of the plant.

The causes of the dependency between several events is classified in five categories:

- Initiating event dependencies
- Functional dependency - intersystem or intercomponent
- Common component - intersystem
- Physical dependencies - intersystem or intercomponent
- Human interaction dependencies - intersystem or intercomponent

All of these types of dependencies have been accounted for in this analysis and Table 4.6 describes each category and explains the method used to treat that type of dependency.

Table 4.6

DEPENDENT EVENTS, DESCRIPTION AND METHOD OF TREATMENT IN THIS STUDY

Dependent Event Causal Category		Description	Method of Treatment In This Analysis
Initiating events		Such as Siesmic, fire, flood or etc. initiated either as an external event or in some cases initiated inside the plant.	No detailed assessment of these initiators (mostly referred to as external events) is preformed as part of this study. However, the frequency of Siesmic and fire initiators causing core damage and release of radionuclide has been estimated and is presented in Section 6.0
Functional Dependencies	Inter-System	If the function of one System is precluded by success or failure of another System.	These dependencies are accounted for during the construction of the event trees.
	Inter-Component	The effective functioning of one component is precluded due to success or failure of another component. e.g. failure of one of the air blast heat exchangers in the DHRS System precludes the requirement for the other one.	These types of dependencies have been accounted for in the Simplified System fault models developed for each safety function.
Common Component, Intersystem		The failure of a single component affects two or more safety functions at the same time. e.g. loss of AC power to both SHRS and Annulus Cooling System simultaneously.	This is accounted for by construction of the plant safety logic model explained in Section 5.1.
Physical Dependencies	Inter-System	This happens when failure of one function puts more stress on the operation of another function, mostly in terms of more severe environment. e.g. failure of HVAC will affect the operation of all equipments which require cooler environment to operate.	This type of dependencies were accounted for in the quantification of plant safety logic model. e.g. the numbers used for fans and filters in the vent and clean up system reflect the extreme environmental condition such as temperature, aerosols, and etc.
	Inter-Component	The same as intersystem physical dependencies except it happens between components of the same system. e.g. failure of one of the two parallel pump puts more stress on the second pump.	This is accounted for during the construction of the simplified fault trees as part of a common cause event introduced wherever effects such as this and/or others were judged to exist.
Human Interaction Dependencies	Inter-System	Two or more systems failing to perform as designed due to human error in any stages of the man-machine interface; design, manufacturing, installation, test, maintenance, or operation.	Part of this dependency caused during the emergency operation is accounted for by introducing common cause events in the plant safety logic model wherever human action is required. However the determination of the dependencies between the safety function caused by human interaction during other stages of his involvement needs extensive information which is not available at this stage of the project.
	Inter-Component	Failure of two or more components caused by a single human error during any stage of machine interface which fails a system.	This has been accounted for in the simplified system fault trees as part of a common cause event for the human failures committed during design, manufacturing, installation and operation. To estimate the degree of damage done by human errors committed during test and maintenance leading to common cause failures more detailed information about the test and maintenance procedures is required.

Section 5.0
ACCIDENT SEQUENCE QUANTIFICATION

This section presents the method and the data utilized to quantify the accident sequences, and a description of the dominant accident sequences for internal (random) initiators.

5.1 PLANT SAFETY LOGIC MODEL

An overall safety logic model was constructed for the purpose of quantifying the accident sequences. The logic of this model involves representation of the event trees as a fault tree by boolean intersections including both failure and success states. This model is being used to quantify the frequency of the system unavailabilities, core disruptive accident, and release frequencies. If desired in the same calculations other intermediate results such as frequencies of the different shutdown heat removal system failure modes can be obtained.

This approach makes it possible to keep track of all the interdependencies due to common component, common human actions, or etc. which may affect operability of several systems through the same cause. This is evident especially due to support systems such as electric power and ultimate heat sinks.

This logic model also provides a workable tool for the analysis of sensitivities of the CDA or release frequenncies to may variables, such as reliability of the emergency shutdown system. Section 8.0 discusses the sensitivity of the results to some of the input variables. A listing of this logic model is presented in Appendix A.

Figure 5.1 shows a simplified structure of the plant safety logic model.

Figure 5.1

EXAMPLE OF PLANT SAFETY LOGIC MODEL STRUCTURE

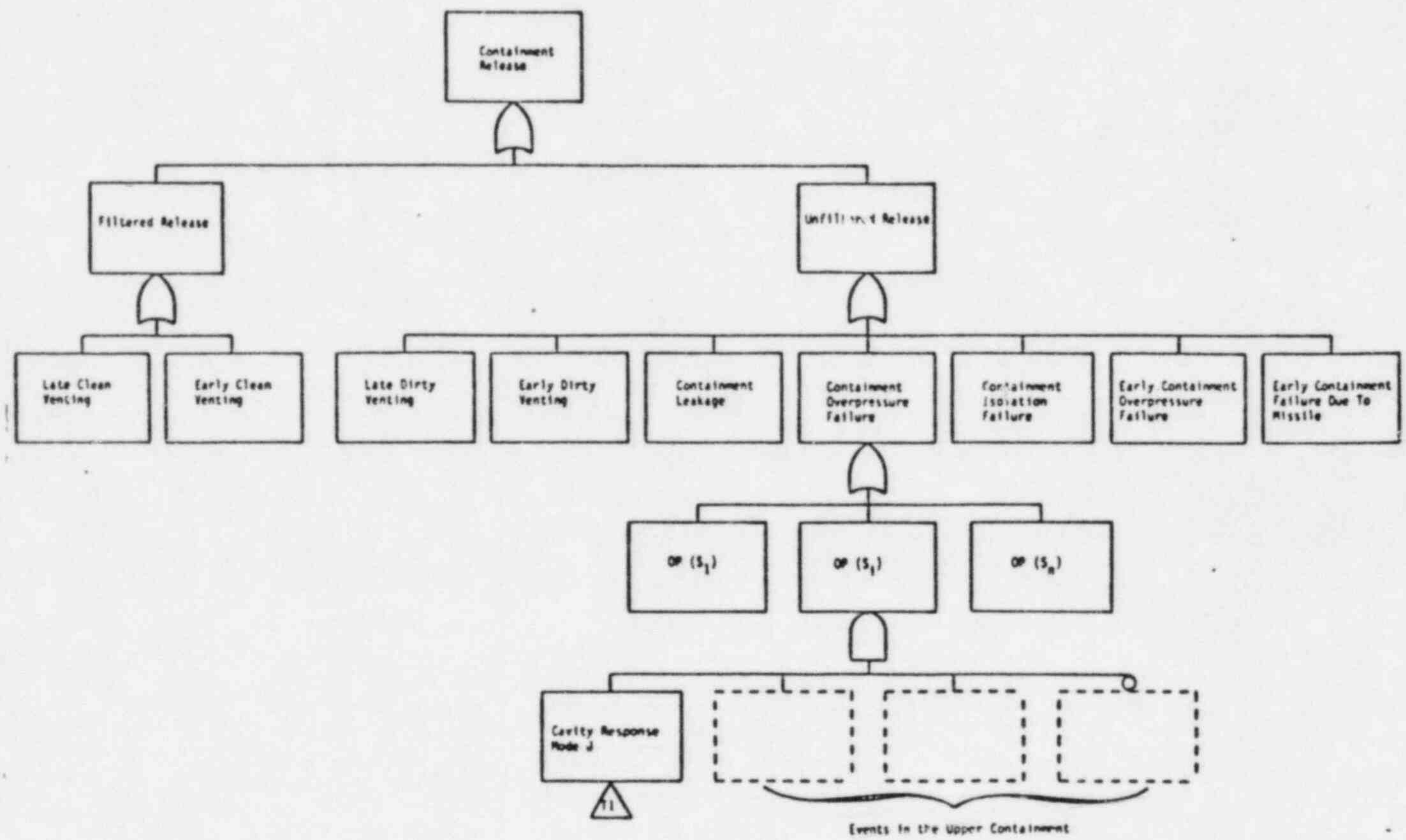


Figure 5.1

EXAMPLE OF PLANT SAFETY LOGIC MODEL STRUCTURE (Continued)

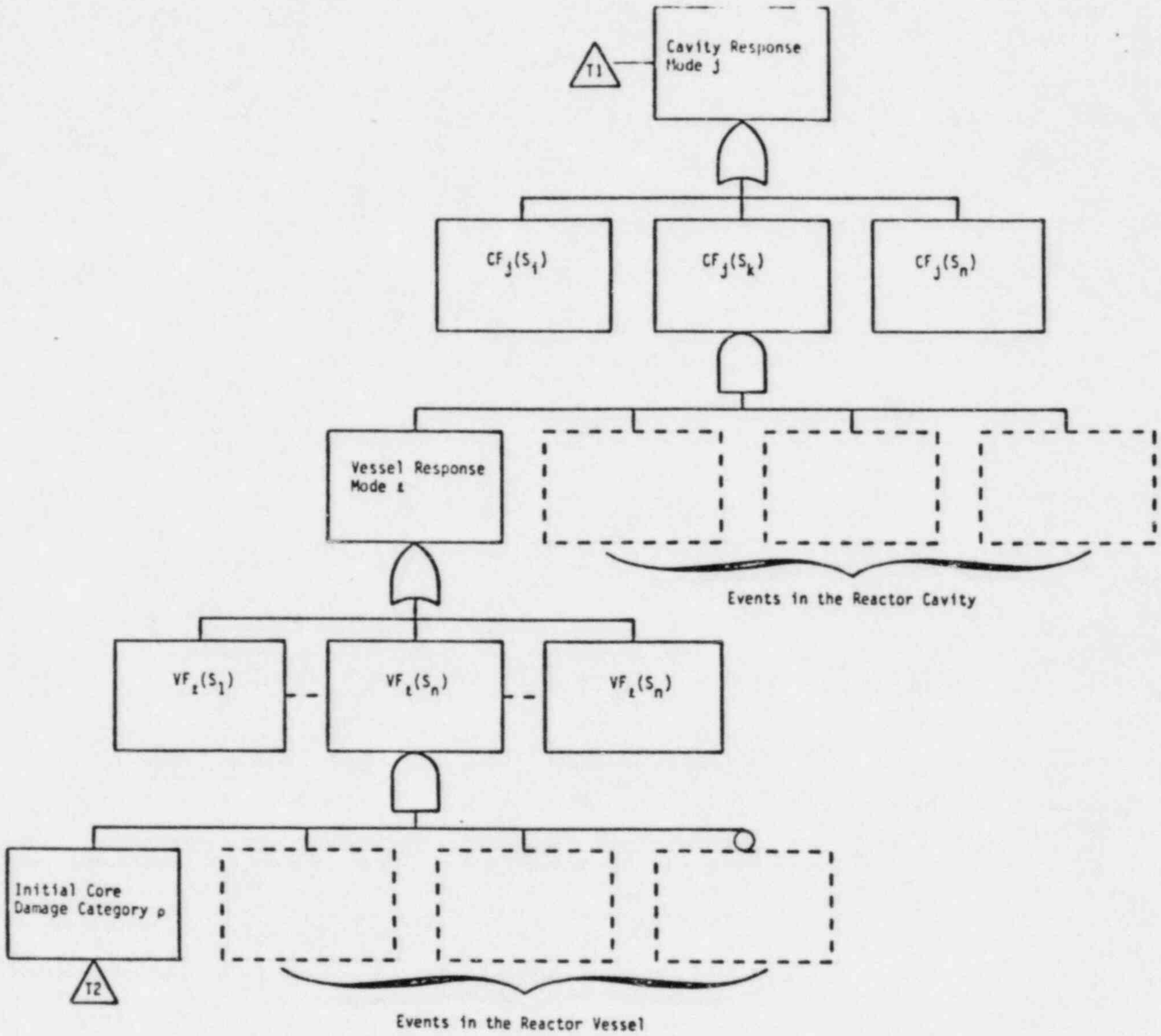
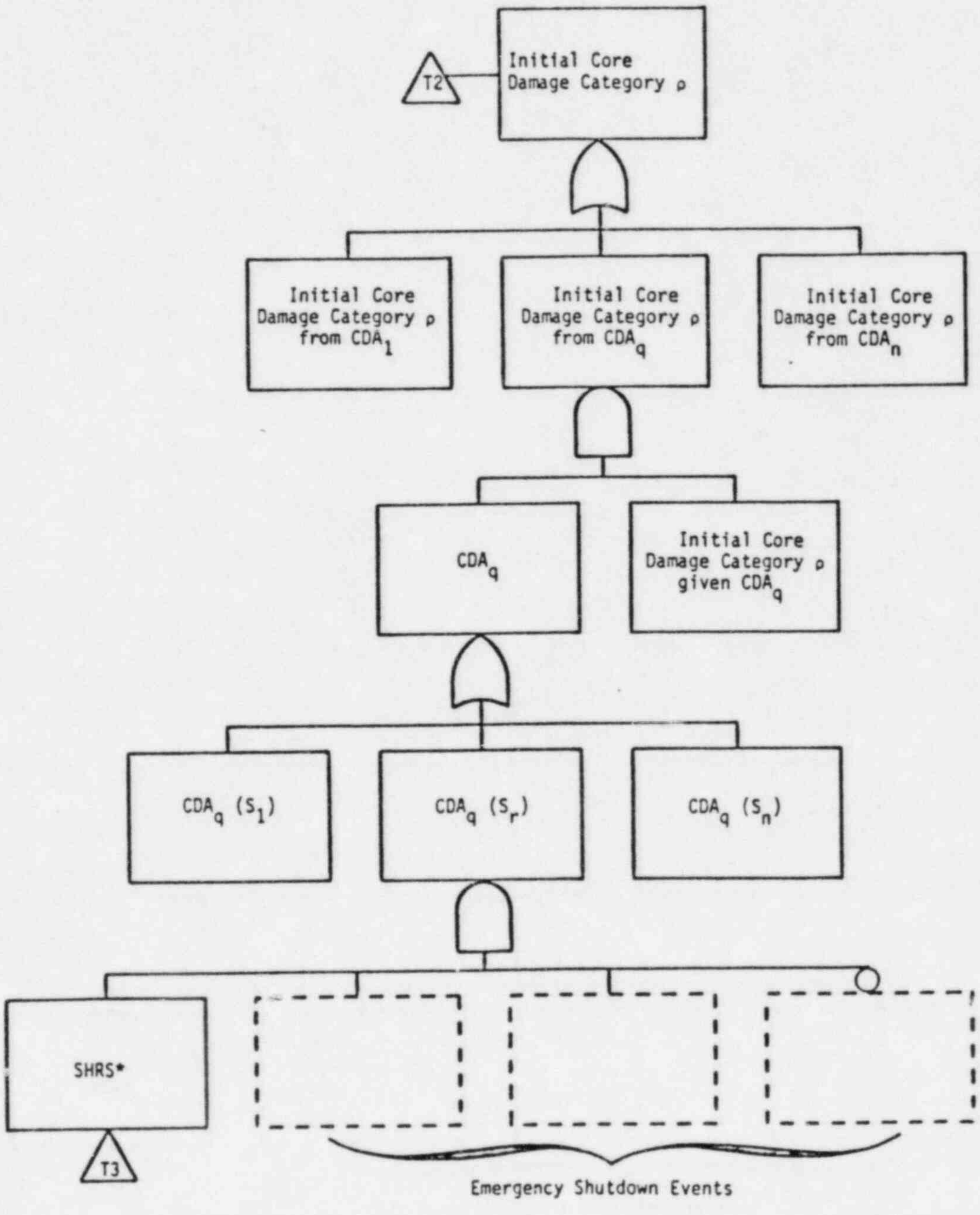


Figure 5.1

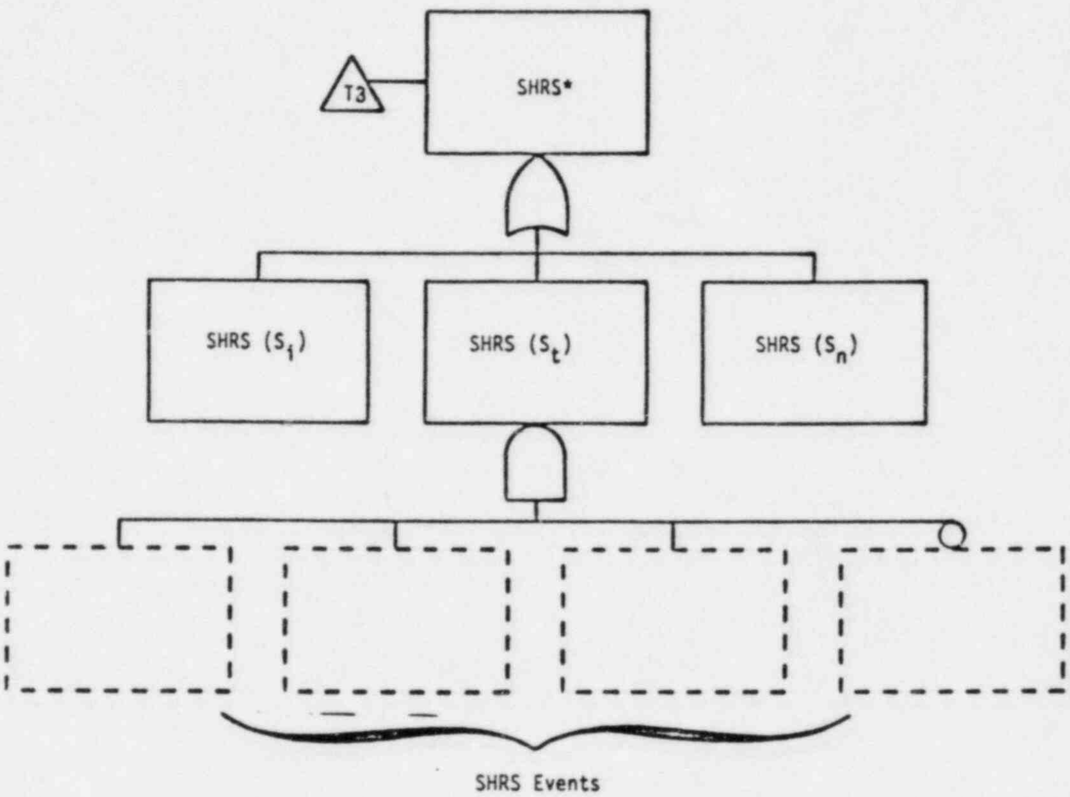
EXAMPLE OF PLANT SAFETY LOGIC MODEL STRUCTURE (Continued)



* Only for Protected Core Disruptive Accidents

Figure 5.1

EXAMPLE OF PLANT SAFETY LOGIC MODEL STRUCTURE (Continued)



* Only for Protected Core Disruptive Accidents

5.2 DATA BASE

Two types of input parameter are necessary for the quantification of the accident sequences, 1) the equipment or human failure frequency and 2) the phenomenological uncertainties. Appendix B contains the data used for quantification of these sequences.

5.3 DOMINANT ACCIDENT SEQUENCES

This section describes the frequency of the core disruptive accidents and containment release caused by internal (random) accident initiators and the dominant accident sequences. Thirteen dominant CDA sequences were identified with mean frequencies of greater than 10^{-6} per year.

The two most dominant sequences are:

- A LOHS caused by failure of all three intermediate loop rupture disks, LOHS (RD), and
- A ULOF caused by a spurious pps signal, ULOF (PPS)

These two sequences cause more than 78% of the frequency of CDA by random initiators. The total mean frequency of internally initiated CDA's are 2.0×10^{-4} per year. The frequency of the containment release modes are presented in Table 5.2, the total mean frequency of unfiltered release from internally initiated accident is 2.5×10^{-6} per year.

Table 5.1

LIST OF DOMINANT CDA SEQUENCES FOR INTERNALLY INITIATED ACCIDENTS

Rank	CDA Sequence	Mean Frequency of CDA Per Year	Percent
1	LOHS (RD)[1)]	1.2×10^{-4}	60%
2	ULOF (PPS)	3.6×10^{-5}	18%
3	ULOF (FW)	7.7×10^{-6}	4%
4	ULOF (1_HTS)	7.2×10^{-6}	4%
5	ULOHS (FW)	4.7×10^{-6}	2%
6	LOHS (FW)	4.6×10^{-6}	2%
7	ULOHS (1_HTS)	4.3×10^{-6}	2%
8	ULOF (T/G)	2.8×10^{-6}	1%
9	ULOHS (T/G)	1.7×10^{-6}	<1%
10	LOHS (LOSP)	1.3×10^{-6}	<1%
11	LOHS (3_HTS)	1.3×10^{-6}	<1%
12	LOHS (NSD/DHRS)	1.1×10^{-6}	<1%
13	LOCA (RPB)	1.0×10^{-6}	<1%

1) The characters in the parenthesis are the accident initiators causing the CDA:

RD - Simultaneous rupture of all three intermediate loop tupture disks

PPS - Inadvertent plant protection signal

FW - Malfunctions in the feedwater system

1_HTS - Loss of one heat transport loop

T/G - Turbine/generator trip

LOSP - Loss of off site power

3_HTS - Loss of all three heat transport loops

NSD/DHRS - Normal shutdown due to failures in DHRS, DHRS unavailable

RPB - Rupture of the primary boundary

Table 5.2

CONTAINMENT RELEASE FREQUENCIES FOR INTERNALLY INITIATED ACCIDENTS

Containment Response Mode	Description of Containment Response Mode	Mean Frequency Per Year
LKG	Leakage Accross the Steel Shell	3.4×10^{-7}
LCV	Late Clean Vent through the Scrubbers and Filters	1.0×10^{-4}
ECV	Early Clean Vent Through the Scrubbers and Filters	1.0×10^{-4}
LDV	Late Dirty Vent, Unfiltered Release	1.7×10^{-7}
EDV	Early Dirty Vent, Unfiltered Release	1.7×10^{-7}
CI	Containment Isolation Failure	5.1×10^{-7}
OP	Containment Overpressure Failure	1.2×10^{-8}
MS	Early Containment Failure due to CDA Initiated Missile	1.2×10^{-8}
OP_o	Early Containment Overpressure Failure due to CDA Initiated Sodium Fire	2.4×10^{-8}
Total Frequency of Release	Filtered	2.0×10^{-4}
	Unfiltered	2.5×10^{-6}

Section 6.0
EXTERNAL EVENTS

In this section external events which impact the frequency of the core disruptive accident and containment release modes are briefly discussed and their contribution in the frequency of the containment release is estimated.

These events are basically additional accident initiating events, and are treated with the same approach as for internal (random) initiators. However, the effect of these accidents on the plant safety functions are more severe in terms of both additional stress and common cause. Therefore these accidents are analyzed separately and will be integrated in Section 7.0.

6.1 SIESMIC ACCIDENTS

The analysis of siesmic accidents was divided into three sizes of earthquake, Operating Basis Earthquakes (OBE), Safe Shutdown Eathquakes (SSE), and earthquakes greater than SSE (>SSE). Table 6-1 shows the range of magnitutde and frequency of these earthquakes.

First events trees were used to define the sequences leading to CDA and containment release under siesmic load and if the plant response is affected by adding new sequences due to earthquakes. Second the conditional frequency of plant safety function failures and events were estimated. Third using the events trees and conditional frequencies the frequencies of CDA's and containment releases were estimated for OBE, SSE, and greater than SSE accidents.

Table 6.1

Categories of Earthquakes and Their Frequencies

Earthquake	Ground Acceleration g	Richter Magnitude	Point Estimate Frequency per Year
Operating Basis Earthquake (OBE)	$0.05\ g < a < 0.15\ g$	$5 < a < 5.7$	1.4×10^{-3}
Safe Shutdown Earthquake (SSE)	$0.15 < a < 0.35\ g$	$5.7 < a < 7$	1.5×10^{-4}
Greater than Safe Shutdown Earthquake (>SSE)	$a > 0.35\ g$	$a > 7$	3.4×10^{-5}

The event trees used in this analysis are consistent with the ones previously used except in the beginning of each event tree, the question was asked with regards to the status of the system under consideration after earthquake, e.g. in the cavity event tree the first question is whether the liner has survived the earthquake or is failed under siesmic stress. However, in order to to identify the type of accident initiator which is caused by the earthquake the pre-initiator event tree of Figure 6.1 is defined. The type of accidents are either a reactivity insertion, or a rupture of the Primary Boundary (PB) or combination of both in a core which may or may not already be damaged by the earthquake.

The conditional frequency estimates of Table 6.2 were employed as best estimates to qualify the siesmic sequences defined. These frequencies are used from NUREG/CR-2681 (Ref. 4) and are purely based on engineering judgement. Some of these numbers, however, were changed which reflects our jedgement of these events under siesmic conditions.

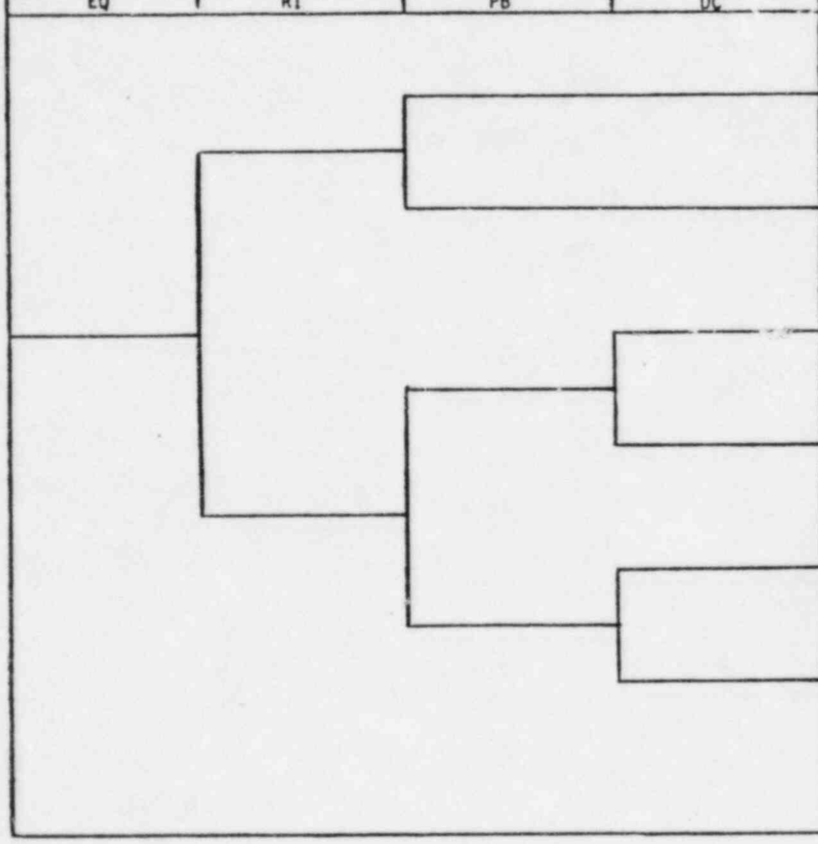
Ten CDA sequences with frequencies greater than 10^{-7} were identified for siesmic initiator as shown in Table 6.3. The most dominant sequence is a Transient Overpower (TOP) CDA caused by an earthquake greater than SSE which comprise 52% of the frequency of CDA's caused by earthquakes. Total frequency of siesmically initiated core disruptive accident is 2.5×10^{-5} mean frequency per year.

Table 6.4 shows the containment release frequencies due to earthquakes. Total frequency of unfiltered releases from the containment due to earthquake is 1.1×10^{-5} mean per year.

Figure 6.1

PRE-INITIATOR EVENT TREE (SIESMIC) FOR CRBRP

Earthquake	Reactivity Insertion Due to Fuel Movements Caused by Earthquake	Rupture of the Primary Boundary Caused by Earthquake (Siesmic Loading)	Core Damage Caused by Earthquake Under Siesmic Loading
EQ	RI	PB	DC



	Description	Frequency Mean Per Year OBE SSE >SSE
EQ_1	No Reactivity Insertion, No Rupture of the Primary Boundary, and No Core Damage	2.8×10^{-4} 1.4×10^{-5} -
EQ_2	No Reactivity Insertion, The Primary Boundary Ruptures, but No Significant Damage	2.8×10^{-7} 6.0×10^{-7} -
EQ_3	Reactivity Insertion, the Primary System Boundary Intact and No Significant Core Damage	1.1×10^{-3} 1.3×10^{-4} 1.8×10^{-5}
EQ_4	Reactivity Insertion, the Primary System Boundary Stays Intact, but Significant Damage to the Core Occurs	1.1×10^{-6} 5.3×10^{-6} 2.0×10^{-6}
EQ_5	Reactivity Insertion, the Primary Boundary Ruptures, but No Significant Core Damage Occurs as a Result of EQ	1.1×10^{-6} 5.3×10^{-6} 2.0×10^{-6}
EQ_6	Reactivity Insertion, the Primary System Boundary Ruptures and Significant Core Damage Occurs as a Result of the EQ	1.1×10^{-7} 1.1×10^{-7} 1.4×10^{-6}

Table 6.2

CONDITIONAL FREQUENCIES EVENTS UNDER THREE SIZES OF EARTHQUAKES

Function	Mean Conditional Frequency of Failure Per Demand		
	OBE	SSE	BFE
Containment Rupture	ϵ	ϵ	10^{-4}
CIS	2×10^{-3}	2×10^{-3}	2×10^{-3}
ACS	3×10^{-4}	10^{-2}	10^{-1}
VPS	3×10^{-4}	10^{-2}	10^{-1}
CUS	8×10^{-3}	10^{-2}	10^{-1}
Loss of Offsite Power	10^{-2}	1	1
Failure of all 3 Diesels	10^{-4}	10^{-2}	10^{-1}
Total Loss of AC	10^{-6}	10^{-2}	10^{-1}
PPS No PB	3.2×10^{-6}	4.0×10^{-5}	9.7×10^{-4}
PPS PB	2.4×10^{-7}	1.3×10^{-5}	9.7×10^{-4}
SCRAM No PB	2.3×10^{-4}	2.6×10^{-3}	1.7×10^{-2}
SCRAM PB	3.3×10^{-4}	3.3×10^{-3}	1.9×10^{-2}
PT No PB	8.7×10^{-4}	7.5×10^{-4}	3.2×10^{-4}
PT PB	5.8×10^{-4}	6.5×10^{-4}	2.1×10^{-4}
SHRS No PB	1.2×10^{-3}	2.6×10^{-2}	4.3×10^{-1}
SHRS PB	6.0×10^{-2}	2.8×10^{-1}	9.0×10^{-1}
Reactivity Insertion Caused by CQ	8.0×10^{-1}	9.0×10^{-1}	1.0
Rupture of the PB Caused by CQ	1.0×10^{-1}	4.0×10^{-1}	8.0×10^{-1}
Damage to the Core Caused by CQ	1.0×10^{-1}	2.0×10^{-1}	4.0×10^{-1}

Table 6.3

LIST OF DOMINANT CDA SEQUENCES FOR SIESMICALLY INITIATED ACCIDENTS

Rank	Sequence	Mean Frequency Per Year	Percent Contribution
1	TOP (>SSE)[1]	1.2×10^{-5}	52
2	TOP (SSE)	3.5×10^{-6}	14
3	LOCA (>SSE)	3.1×10^{-6}	12
4	LOCA (SSE)	1.7×10^{-6}	7
5	TOP (OBE)	1.3×10^{-6}	5
6	UTOP & ULOF (SSE)	5.1×10^{-7}	2
7	LOHS (SSE)	3.6 x 10-7	1
8	UTOP & ULOF (SSE)	3.5×10^{-7}	1
9	LOHS (OBE)	3.1×10^{-7}	1
10	UTOP & ULOF (OBE)	2.5×10^{-7}	1

1) The characters represent the size of earthquake which causes the CDA:

OBE - Operating Basis Earthquake

SSE - Safe Shutdown Earthquake

>SSE - Greater than Safe Shutdown Earthquake

Table 6.4

CONTAINMENT RELEASE FREQUENCIES INITIATED BY THREE MAGNITUDES OF EARTHQUAKES

Cont. Res. Mode	Description	Mean Frequency Per Year (Percent of Total Release)			
		OBE	SSE	>SSE	All EQ's
LCV	Late Clean Vent ($\geq$24h)	1.5×10^{-6}	7.8×10^{-6}	1.5×10^{-5}	2.4×10^{-5}
ECV	Early Clean Vent ($\leq$24h)	1.5×10^{-6}	7.8×10^{-6}	1.5×10^{-5}	2.4×10^{-5}
LDV	Late Dirty Vent	1.2×10^{-8}	6.2×10^{-8}	1.2×10^{-6}	1.3×10^{-6}
EDV	Early Dirty Vent	1.2×10^{-8}	6.2×10^{-8}	1.2×10^{-6}	1.3×10^{-6}
LKG	Leakge Across the Shell	2.9×10^{-7}	1.5×10^{-6}	3.1×10^{-6}	4.9×10^{-6}
CI	Containment Isolation	6.0×10^{-9}	3.2×10^{-8}	6.6×10^{-8}	1.0×10^{-7}
OP	Containment OP Failure	1.5×10^{-9}	2.3×10^{-7}	3.1×10^{-6}	3.3×10^{-6}
EOP	Early Cont. OP (Na Fire)	9.6×10^{-11}	3.3×10^{-10}	3.5×10^{-9}	3.9×10^{-9}
MIS	Early Cont. Fail (Missile)	4.8×10^{-11}	1.7×10^{-10}	1.8×10^{-9}	2.0×10^{-9}
	Filtered	3.0×10^{-6}	1.6×10^{-5}	3.0×10^{-5}	4.9×10^{-5}
	Unfiltered	3.2×10^{-7}	1.9×10^{-6}	8.7×10^{-6}	1.1×10^{-5}

6.2 FIRE ACCIDENTS

The evaluation of containment releases due to fires is estimated in the following steps:

- Identification of critical areas where fire can be initiated leading to simulaneous failure of one or more safety functions
- Estimate of the frequency of such fires and likelihood of growth such that it fails one or more safety functions
- Estimate of the frequency of core disruption given fire initiated in a critical area
- Estimate of the frequency of containment release given fire in a critical are which leads to CDA

Three types of fire may result in damage in CRBRP, cable or oil fire, sodium fire, or fires due to sodium-water interaction.

The cable or oil fires may happen in five critical ares which will affect one or more safety functions. These areas are:

- Fire in the cable spreading rooms
- Cable or oil fire in the intermediate bay adjacent to cable spreading room
- Diesel generator cells
- AC power switchgear cells located in the diesel generator building
- Fire in the DC switchgear cells

Sodium fires can be started in three areas:

- Sodium fire in the head access area
- Sodium fire in the primary heat transport cells
- Sodium fire in the intermediate bay

The fire due to sodium-water interaction may take place in the steam generator cells. Estimates of the frequency of the fire initiation in these nine areas are shown in Table 6.5.

The fire in the CSR propagates to the other CSR and fails both cable spreading rooms with probability of 1.5×10^{-4} [Ref. 8]. Failure of both CSR's causes loss of instrumentation and control and all means of decay heat removal except by natural circulation in the primary and intermediate loops and forced circulation in the steam/water loop with turbine driven auxiliary feedwater pump. The mean frequency of turbine driven AFW pump is estimated to be 5×10^{-2} per demand. Therefore the frequency of total loss of shutdown heat removal system and therefore a LOHS core disruptive accident due to fire in CSR will be 3×10^{-8}. Given such a CDA it is conservatively assumed that since the instrumentation and control is lost and all the containment systems (except for containment isolation) rely on operation intervention therefore the containment will fail due to overpressure under this scenario.

Fire in the intermediate bay adjacent to cable spreading room will have similiar frequency and consequences as a fire in the one of the cable spreading rooms.

A fire in one of the diesel generator cells has to propagate to other two cells to fail all three emergency diesel generators. It can be shown that failure of all three diesel generators in such a mode is negligible to the other modes of failure of three diesels already considered in Section 6.0.

A fire in one of the AC switchgear cells needs to propagate to the other two switchgear ceels to cause significant consequences. This will happen with the frequency of 5×10^{-6} [Ref. 8]. This causes total loss of AC and if the frequency of turbine driven AFW pump is assumed to be 5×10^{-2} then the frequency of failure of SHRS and therefore LOHS core disruptive accident will be 10^{-9} per year. Due to loss of AC the containment will fail due to overpressure.

Table 6.5

FREQUENCY ESTIMATE OF FIRE INITIATION IN CRITICAL AREAS OF CRBRP

Fire Location	Mean Frequency Per Year
Fire in Cable Spreading Room	4×10^{-3}
Fire in Intermediate Bay Adjacent to Cable Spreading Room	4×10^{-3}
Fire in Diesel Generator Cell	7×10^{-3}
Fire in AC Switchgear Cell	4×10^{-3}
Fire in DC Switchgear Cell	4×10^{-3}
Sodium Fire in the Head Access Area	10^{-4}
Sodium Fire in the PHTS	10^{-5}
Sodium Fire in the IMB	10^{-7}
Large Na-Water Reaction in the Steam Generator Cell	8×10^{-7}

Loss of all DC requires initiation of the fire in one DC switchgear cell and propagation to the other two cells. It is estimated that this will occur with frequency of 5×10^{-6} [Ref. 8] per deand. Once all DC is lost then turbine driven AFW pump will be the only mean of decay heat removal. Therefore the estimated mean frequency of CDA for this scenario is estimated to be 10^{-9} per year. it is assumed at this point that loss of DC disables the control and instrumentation and therefore activation of containment systems is not possible and therefore the containment fails due to overpressure.

Sodium fire may be initiated in one primary heat transport cell and propagated into other two cells. This event which incapacitates the PHTS totally happens with the mean conditional frequency of 10^{-4} [Ref. 8]. This scenario will fail the flow of natural circulation and DHRS because of failure of all ponys. Therefore a LOHS will occur with frequency of 10^{-9} per year.

This scenario will affect the containment performance due to generation of a great deal of sodium aerosols which increases the common mode failure of the containment systems. Considering this common mode failure the containment will fail with mean frequency of 5×10^{-2} per demand or 5×10^{-11} per year.

A sodium fire in an IMB cell requires rupture of a intermediate sodium pipe and propagation of the fire to the upper levels of the IMB which may cause loss of all safety related control and power cable resulting in the ability to remove decay heat. This scenario occurrs with a mean frequency of 10^{-10} per year causing a LOHS accident. For the same reason as previous scenario the containment will fail with condition frequency of 5×10^{-2}/d or absolute mean frequencyof 5×10^{-12} per year due to overpressure failure.

A sodium fire in the Head Access Area (HAA) may spread into the CDM area and result in a common cause failure of all CDM's which precludes control rod insertion before operators action can be taken to scram the reactor. The conditional frequency of such scenario is estimated to be 10^{-3} per demand and therefore a fir in the HAA can result in a ULOF accident with the mean frequency of 10^{-7} per year. Considering the common cause due to higher concentration of aerosols the containment will fail with a mean frequency of 5×10^{-8} per year.

There are two dominant initiators for a large sodium-water reaction. First failure of the steam generator water header. The frequency of such initiator is similar to that of a pressure vessel and is therefore estimated to be 10^{-7} per year. This initiator even though may cause severe consequences, however, because of very low likelihood of initiation does not contribute to the frequency of release. The second initiator is a failure of steam generator tubes (beyond design basis accident of 7 tubes). Failure of the Sodium Water Reactor Pressure Relief System (SWRPRS), with conditional frequency of 8×10^{-6} per demand [Ref. 8] and common cause failure of all three HTS loop (5×10^{-1} per demand) and failure of DHRS due to contamination of air blast heat exchangers will result in a LOHS accident with mean frequency of 4×10^{-7} per year. Similar to other sodium fire accidents the containment failure will occur with conditional frequency of 5×10^{-2}/d or absolute mean frequency of 2×10^{-8} per year.

This concludes that the most dominant fire related scenario in terms of core disruption is a sodium water reaction due to steam generator tube rupture (4×10^{-7} per year). but the most dominant fire related sequence to cause unfiltered release is the fire in the CSR or in the IMB adjacent to CSR failing instrumentation and control for all the vital safety systems (6.0×10^{-8} per year). Nevertheless, no fire related accident is identified which significantly affects the frequency of CDA's or unfiltered release from the containment as shown in Table 6.6.

Table 6.6

FREQUENCIES OF CDA AND CONTAINMENT UNFILTERED RELEASE FOR INTERNALLY AND EXTERNALLY INITIATED ACCIDENT

Accident Initiator (I)	Mean Freq. of Initiator Per Year (I)	Mean Conditional Frequency			Mean Freq. of CDA Per Year (I.CDA)	Mean Freq. of Unfiltered Release Per Year (I.CDA.UFR)
		CDA Given I (CDA/I)	Unfiltered Release Given CDA	Unfiltered Release Given Initiator		
Random Initiators	23.1	8.7×10^{-6}	1.2×10^{-2}	1.1×10^{-7}	2.0×10^{-4} (89%)	2.5×10^{-6}
Siesmic, Operating Basis Earthquake	1.4×10^{-3}	1.4×10^{-3}	1.6×10^{-1}	2.3×10^{-4}	2.0×10^{-6} (1%)	3.2×10^{-7}
Siesmic, Safe Shutdown Earthquake	1.5×10^{-4}	4.0×10^{-2}	3.2×10^{-1}	1.3×10^{-2}	6.0×10^{-6} (1%)	1.9×10^{-6}
Siesmic, Greater than SSE	3.4×10^{-5}	5.0×10^{-1}	5.1×10^{-1}	2.6×10^{-1}	1.7×10^{-5}	8.7×10^{-6}
Fire Initiators	2.2×10^{-2}	2.6×10^{-5}	1.6×10^{-1}	4.0×10^{-6}	5.6×10^{-7}	8.7×10^{-8}
Sum	23.1	1.0×10^{-5}	6.1×10^{-2}	6.1×10^{-7}	2.3×10^{-4}	1.4×10^{-5}

6.3 OTHER EXTERNAL INITIATORS

A discussion of the several other external events is presented in this section.

- Floods (internal or external)

The siesmic Category 1 systems and equipment which are located in the intermediate bay of the steam generator building, reactor service building, diesel generator building and control building require flood protection. These systems and components provide most the unfractions necessary for the prevention of core damage.

The PSAR examines the probable maximum external flood potential from an Operating basis Earthquake (OBE) causing postulated failure of Norris Dam. This condition will produce the maximum plant flood level as stipulated by the regulatory guide 1.59. It was calculated that maximum wave forces exerted on the plant structures are relatively insignificant and will not cause damage to the plant structures.

In order for internal floods to fail one or several safety functions they dould have to be caused by rupture of a large tank or a large pipe. The mean frequency of a massive rupture of a large tank or large pipe is about 10^{-7} per year. If the probability of failure of one or more safety functions given the flood and the probability of CDA given failure of those functions are combined then the frequency of these scenarios will be insignificant to the sequences considered.

• Tornados

Several studies were performed in recent years which estimate the frequencyof such events. Reference 9 estimated that the annual mean probability of a tornado with velocity of 360 mph or greater to be 2.8×10^{-5}. If this frequency is combined by failure of the structure and damage to the safety related equipment and CDA given that failures, the frequency of CDA's caused by this initiators will become less significant than the sequences already considered.

Reference 10 estimates that the annual probability that any tornado generated missile events hits a safety related structure at NEC region 1 is 7×10^{-5}, the probability that there is a hit sufficient to cause backface scabbing if all safety related stuructres has 6 inch walls is 3×10^{-5} and if they have 18 inch walls is 2×10^{-6}. Even though the results are for a particular sample plant configuration, however, these estimates show a tornado missile to fail a safety structure with a frequency of 2×10^{-9} per year or less. This scenario therefore will not significantly impact the frequency of CDA or containment release.

6.4 CONCLUSION

The analysis of the external events show that siesmic events are the major contributor to the frequency of containment release. The fires contribute less than 1% to the frequency of containment failure and other external events have no signifiant impact of the frequency of core disruptive accidents or containment release, as shown in Table 7-6.

Section 7
RESULTS AND CONCLUSIONS

This section presents the results of this study and discusses these findings. The results of this analysis are being presented in the following ways:

- Dominant Core Disruptive Accident Sequences
- Frequency of Different Core Disruptive Accidents
- Dominant Containment Accident Sequences for each Release Mode

The first results of the study on the dominant CDA sequences is shown in Table 7.1. The expression in the parenthesis represents the initiating accident of each particular CDA.

Eighteen dominant CDA sequences with frequencies higher than 10^{-6} per year are identified. The most dominant sequences is a Loss of Heat Sink (LOHS) accident caused by common cause failure of all three rupture disks in the intermediate loops and dumping of the intermediate sodium. This will leave the DHRS as the only mean of decay heat removal and a LOHS occurs upon failure of DHRS or primary ponys. This sequence constitutes 52% of the frequency of core disruptive accidents. The second most dominant sequence is an Unprotected Loss of Flow (ULOF) accident caused by a spurious Plant Protection Signal (PPS). A spurious PPS signal is generated (with a mean frequencyof 8.8 per year), the primary sodium pumps trip to ponys but both primary and secondary shutdown systems fail to stop the reaction. This sequence is 16% of frequency of all CDA's. A Transient Overpower (TOP) core disruptive accident caused by an earthquake greater than Safe Shutdown Earthquake (>SSE) is the third most dominant sequence. This sequence is 6% of the frequency of all CDA's.

Table 7.1

LIST OF DOMINANT CDA SEQUENCES

Rank	CDA Sequence	Mean Frequency of CDA Per Year	Percent
1	LOHS (RD)[1]	1.2×10^{-4}	52%
2	ULOF (PPS)	3.6×10^{-5}	16%
3	TOP (>SSE)	1.3×10^{-5}	6%
4	ULOF (FW)	7.7×10^{-6}	3%
5	ULOF (1-HTS)	7.2×10^{-6}	3%
6	ULOHS (FW)	4.7×10^{-6}	2%
7	LOHS (FW)	4.6×10^{-6}	2%
8	ULOHS (1-HTS)	4.3×10^{-6}	2%
9	TOP (SSE)	3.5×10^{-6}	2%
10	LOCA (>SSE)	3.1×10^{-6}	1%
11	ULOF (T/G)	2.8×10^{-6}	1%
12	ULOHS (T/G)	1.7×10^{-6}	<1%
13	LOCA (SSE)	1.7×10^{-6}	<1%
14	LOHS (LOSP)	1.3×10^{-6}	<1%
15	LOHS (3_HTS)	1.3×10^{-6}	<1%
16	TOP (OBE)	1.3×10^{-6}	<1%
17	LOHS (NSD/DHRS)	1.1×10^{-6}	<1%
18	LOCA (RPB)	1.0×10^{-6}	<1%

1) Refer to Tables 5.3 and 6.3

The frequencies of different types of core disruptive accidents are shown in Table 7.2. For comparison the results of three other previous studies are included. Even though a great deal of differences exists between the assumptions, limitations and objectives of each one of these studies, nevertheless the CDA frequencies seem to be compatible within the uncertainties of each study. However, if the frequency of containment failure is compared (except for Sandia Study which does nore acover the analysis of the containment) the result varies within one order of magnitude from CRBRP-1 study which estimates 2.6×10^{-6} mean frequency per year to the GE study estimate of 2.6×10^{-5} mean frequency per year.

Table 7.3 shows the dominant containment release sequence for the nine release modes defined in Section 4.3.6. The release modes decline in their radiological consequences from left to right with R1 (missile failure) or R2 (early overpressure failure due to sodium spray fire) being the highest consequence release modes and two filtered release modes R8 (early filtered release before 24 hours) and R9 (late filtered release after 24 hours) being the least, especially benign release mode of R9.

Each containment sequence in Table 7.3 shows the type of core disruptive accident and the mode of containment response.

Table 7.2

COMPARISON OF ACCIDENT CATEGORY FREQUENCY ESTIMATES FROM FOUR DIFFERENT ACCIDENT ANALYSIS STUDIES FOR CRBRP

Accident Category	Estimated CDA Recurrents Frequency Per Year**			
	GEFR [Ref. 5] (Mean)	CRBRP [Ref. 2] (Mean)	Sandia [Ref. 4] (Mean)	SAI [This Study] (Mean)
ULOF	1.2×10^{-6}	3.7×10^{-5}	4.3×10^{-6}	5.5×10^{-5}
UTOP	1.9×10^{-5}	8.6×10^{-8}	3.3×10^{-6}	2.1×10^{-6}
ULOF & UTOP	6.0×10^{-6}	1.4×10^{-6}	6.3×10^{-6}	3.3×10^{-6}
ULOHS	2.5×10^{-5}	1.1×10^{-5}	7.4×10^{-7}	1.1×10^{-5}
Total Unprotected	5.1×10^{-5}	4.9×10^{-5}	1.5×10^{-5}	7.1×10^{-5}
LOHS	---	9.2×10^{-5}*	2.1×10^{-4}*	1.3×10^{-4}
LOCA	---	6.9×10^{-7}	2.3×10^{-5}	5.9×10^{-6}
TOP	---	---	---	1.8×10^{-5}
Total Protected	5.1×10^{-6}	9.3×10^{-5}	2.5×10^{-4}	1.6×10^{-4}
TOTAL	5.6×10^{-5}†	1.4×10^{-4}	2.7×10^{-4}	2.3×10^{-4}

* This also includes TOP Frequency

† The main difference of GEFR Study is due to lower SHRS Failure probability used in GEFT Study

** The initiators used in these studies resulted in the following number of transients per year: GEFR = 16 mean frequency per year, Sandia = 17 mean frequency per year, CRBRP = 22 mean frequency per year, and SAI = 23 mean frequency per year

Table 7.3

DOMINANT ACCIDENT SEQUENCES FOR EACH RELEASE MODE

R1 (MIS)	R2 (EOP)	R3 (CI)	R4 (EDV)	R5 (OP)	R6 (LDV)	R7 (LKG)	R8 (ECV)	R9 (LCV)
LOHS-MIS 7.3×10^{-9}	LOHS-EOP 1.5×10^{-8}	LOHS-CI 3.1×10^{-7}	TOP-EDV 9.4×10^{-7}	TOP-OP 2.8×10^{-6}	TOP-LDV 9.4×10^{-7}	TOP-LKG 3.0×10^{-6}	LOHS-ECV 6.4×10^{-5}	LOHS-LCV 6.4×10^{-5}
ULOF-MIS 2.9×10^{-9}	ULOF-EOP 5.9×10^{-9}	ULOF-CI 1.2×10^{-7}	LOCA-EDV 2.4×10^{-7}	LOHS-OP 8.4×10^{-7}	LOCA-LDV 2.4×10^{-7}	LOCA-LKG 9.1×10^{-7}	ULOF-ECV 2.7×10^{-5}	ULOF-LCV 2.7×10^{-5}
TOP-MIS 1.5×10^{-9}	TOP-EOP 2.9×10^{-9}	TOP-CI 7.7×10^{-8}	LOHS-EDV 1.1×10^{-7}	LOCA-OP 6.5×10^{-7}	LOHS-LDV 1.1×10^{-7}	LOHS-LKG 2.1×10^{-7}		
	ULOHS-EOP 1.3×10^{-9}			ULOF-OP 3.0×10^{-7}				
1.4×10^{-8}	2.8×10^{-8}	6.1×10^{-7}	1.5×10^{-6}	4.6×10^{-6}	5.2×10^{-6}	5.2×10^{-6}	1.2×10^{-4}	1.2×10^{-4}

Section 8

SENSITIVITY ANALYSIS AND UNCERTAINTY

This section discusses the sensitivity analysis made and assesses the uncertainty of the results due to uncertainty in the variable of the sensitivity analysis cases. However, this section will not attempt to address all the sensitivities which may impact the frequency of the core disruption and containment release and it is believed that more cases should be investigated.

This study investigates the sensitivity of the results to three parameters or assumptions in the study.

Case 1: Sensitivity of the CDA and Release Mode Frequencies to Frequency of Failure of SCRAM System.

The sensitivity of the results to the frequency of SCRAM failure was assessed by reducing its failure probability from 2.6×10^{-6} mean per demand to 10^{-7} mean per demand, this is the failure frequency used for SNR-300, LMFBR plant in West Germany [Refs. 10, 12] which has relatively similar design [Ref. 1].

The result in Table 8.1 shows that the frequency of unprotected CDA's is reduced by more than a factor of 2 and the total frequency of core disruption is reduced by less than 20%. The reduction in frequency of unfiltered releases is about 7%. The reason for small sensitivity of the containment release frequencies to the SCRAM system failure probability is that they are dominated by siesmic accidents which is insensitive to this kind of variations in SCRAM system failure probability. Table 8.2 shows the sensitivity of the CDA's and release frequencies caused only by internal events to the SCRAM failure probability.

Another observation is that by reducing the SCRAM failure probability most of the reduction will be in the high consequence release modes R1 and R2 with each reduced by a factor of 2. Release modes R3 and R5 are reduced by 16% and 15% respectively.

Case 2: Sensitivity of the CDA and Release Mode Frequencies to the Frequency Failure of DHRS

In this case the sensitivity of the results is investigated with respect to a more redundant DHRS. It was assumed that the DHRS achieves redundancy with existing configuration, i.e., no equipment is added and the configuration is retained but the capacity of the components are increased as necessary to improve the redundancy.

The results in Table 8.1 show that the frequency of protected CDA's are reduced by about a factor of 2. The total frequency of core disruption is reduced by 35% by improving the DHRS redundancy. The frequency of unfiltered release from the containment is reduced by 10%. The containment release modes which are affected the most are release modes R3 and R5 which are reduced 30% and 17% respectively.

Case 3: Sensitivity of the CDA and Release Mode Frequencies to CDA-Initial Core Damage Matrix

This case investigates that how much the results are sensitive to the CDA-ICD matrix. This matrix defines the likelihood that each core disruptive accident results in a certain initial energetics. To obtain the sensitivity of the results to this matrix a new matrix was defined which shifts the core disruptive accident towards higher energetics. For example, if the case case assumes that the likelihood of a very large energy core disruption (which may cause generation of a missile or sodium spray fire early in the accident) given a ULOF accident is about 1% and 99% of the energy is either benign or moderate. The new matrix was defined with great conservatism to provide an upper bound for the frequency of CDA and containment release.

The results in Table 8.1 show that the frequency of unfiltered release from the containment increases by 12%. However, the largest increase is in high consequence modes R1 and R2 which increase by a factor of 2. Release mode R5 is also increased by close to 25%. It should be recognized that even through the frequency of containment failure is not significantly affected, the risk may be affected since the frequency increase mostly happens in high consequence modes.

At the end it is possible to conclude that the results of this analysis as shown here are within a factor of 2 or less sensitive to the uncertainties in failure probability of the SCRAM or DHRS systems and the uncertainty in the CDA-ICD matrix.

Table 8.1

RESULTS OF SENSITIVITY ANALYSIS OF THE CDA AND CONTAINMENT RELEASE FREQUENCIES

Case Study	CDA Mean Frequency Per Year			Containment Release, Mean Frequency Per Year									Unfiltered Release Mean Freq. Per Year
	Protected	Unprotected	Total	R1 (MIS)	R2 (E_OP)	R3 (CI)	R4 (EDV)	R5 (OP)	R6 (LDV)	R7 (LKG)	R8 (ECV)	R9 (LCV)	
Baseline	1.6x10-4	7.1×10^{-5}	2.3×10^{-4}	1.4×10^{-8}	2.8×10^{-8}	6.1×10^{-7}	1.5×10^{-6}	4.6×10^{-6}	1.5×10^{-6}	5.2×10^{-6}	1.2×10^{-4}	1.2×10^{-4}	1.3×10^{-5}
Sensitivity Case I (SCRAM)	1.6×10^{-4}	3.1×10^{-5}	1.9×10^{-4}	6.5×10^{-9}	1.3×10^{-8}	5.1×10^{-7}	1.4×10^{-6}	3.9×10^{-6}	1.4×10^{-6}	5.0×10^{-6}	1.1×10^{-4}	1.1×10^{-4}	1.2×10^{-5}
Sensitivity Case II (DHRS)	8.3×10^{-5}	7.1×10^{-5}	1.5×10^{-4}	1.4×10^{-8}	2.7×10^{-8}	4.3×10^{-7}	1.4×10^{-6}	3.8×10^{-6}	1.4×10^{-6}	5.0×10^{-6}	8.8×10^{-5}	8.8×10^{-5}	1.2×10^{-5}
Sensitivity Case III (CDA_CD Matrix)	1.6×10^{-4}	7.1×10^{-5}	2.3×10^{-4}	2.8×10^{-8}	5.6×10^{-8}	6.1×10^{-7}	1.6×10^{-6}	5.7×10^{-6}	1.6×10^{-6}	5.5×10^{-6}	1.2×10^{-4}	1.2×10^{-4}	1.5×10^{-5}

Table 8.2

RESULTS OF SENSITIVITY ANALYSIS OF THE CDA AND CONTAINMENT RELEASE FREQUENCIES FOR INTERNALLY INITIATED ACCIDENTS

Case Study	CDA Mean Frequency Per Year			Containment Release, Mean Frequency Per Year									Unfiltered Release Mean Freq. Per Year
	Pro-tected	Unpro-tected	Total	R1 (MIS)	R2 (E_OP)	R3 (CI)	R4 (EDV)	R5 (OP)	R6 (LDV)	R7 (LKG)	R8 (ECV)	R9 (LCV)	
Baseline	$1.3x10^{-4}$	$7.0x10^{-5}$	$2.0x10^{-4}$	$1.2x10^{-8}$	$2.4x10^{-8}$	$5.1x10^{-7}$	$1.7x10^{-7}$	$1.2x10^{-6}$	$1.7x10^{-7}$	$3.4x10^{-7}$	$1.0x10^{-4}$	$1.0x10^{-4}$	$2.5x10^{-6}$
Sensitivity Case I (SCRAM)	$1.3x10_{-4}$	$3.0x10^{-5}$	$1.6x10^{-4}$	$4.5x10^{-9}$	$9.0x10^{-9}$	$4.1x10^{-7}$	$7.4x10^{-8}$	$5.2x10^{-7}$	$7.4x10^{-8}$	$1.4x10^{-7}$	$8.1x10^{-5}$	$8.1x10^{-5}$	$1.2x10^{-6}$
Sensitivity Case II (DHRS)	$8.3x10^{-5}$	$7.0x10^{-5}$	$1.5x10^{-4}$	$1.2x10^{-8}$	$2.3x10^{-8}$	$3.3x10^{-7}$	$5.5x10^{-8}$	$3.9x10^{-7}$	$5.5x10^{-8}$	$1.1x10^{-7}$	$6.4x10^{-5}$	$6.4x10^{-5}$	$9.7x10^{-7}$
Sensitivity Case III (CDA_CD Matrix)	$1.3x10^{-4}$	$7.0x10^{-5}$	$2.0x10^{-4}$	$2.6x10^{-8}$	$5.2x10^{-8}$	$5.1x10^{-7}$	$3.2x10^{-7}$	$3.2x10^{-7}$	$3.2x10^{-7}$	$6.3x10^{-7}$	$9.9x10^{-5}$	$9.9x10^{-5}$	$4.2x10^{-5}$

REFERENCES

1. CRBRP PSAR, "CRBRP Preliminary Safety Analysis Report", April 1975

2. CRBRP-1,, "CRBRP Safety Study - An Assessment of Accident Risk from CRBRP, March 1977.

3. NUREG/CR-1507, SAND80-1267, "LMBFR Accident Delineation Study - Phase I", November 1980.

4. NUREG/CR-2681, SAND82-0720, "Estimated Recurrence Frequencies for Initiating Accident Categories Associated with the Clinch River Breeder Reactor Plant", April 1982.

5. CEFR-13023, "Risk Analysis Methods Development", April 1980.

6. SAND81-0260.

7. NUREG/CR-0427, "Accident Progression for a Loss of Heat Sink with SCRAM in a LMFBR", R.A. Bari, H. Ludewig, W.T. Pratt and Y.H. Sun, October 1978.

8. SAI-035-76-PA, "Evaluation of the Risk Due to Fires at CRBRP", January 1977.

9 "Tornado Risk Evaluation Using Wind Speed Profiles", R.G. Garson, J. Morla-Catalan, and C. Allin Cornel, Journal of the Structural Division, ASCE, Vol. 101, September 1975 SAI-320-82-PA.

10. "Tornado Missile Simulation and Risk Analysis", L.A. Twisdale, W.L. Dunn, and J. Cho, Meeting on Probabilistic Analysis of Nuclear Safety, ANS, Newport Beach, May 1978.

11. "Conditional Risk Assessment of SNR-300 LMFBR Plant", E.T. Rumble, M. Schikkor, B. Najafi, August 1982.

12. GRA-A-700 "Risikoorientierte Analyse ZUM SNR-300, Berlicht Der GRS" April 1982

13. EPRI NP-217, "User's Guide for WAM-BAM Computer Code, "F.L. Leverenz, H. Kirch, January 1975.

14. NUREG/CR-0224 Multiphase Transients with Cooland and Core Materials in LMFBR Core Disruptive Accidents Energetics Evaluations," T.G. Theofanous, et. al., July 1978.

15. "CRBR CDA Energetics," T.G. Theofanous and C.R. Bell, to be publichsed February 1983.

Appendix A
LISTING OF THE SAFETY LOGIC MODEL

A listing of the plant Safety Logic model is presented in this Appendix. The listing follows the format of the WAM_Series fault tree codes [Ref. 13] which were used to quuantify the model.

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INPUT FAULT TREE DESCRIPTION

(1) GATE NUMBER
(2) GATE NAME
(3) GATE TYPE
(4) NUMBER OF GATES INPUT
(5) NUMBER OF COMPONENTS INPUT
(6) NUMBER OF EVENTS IN COM GATE TO BE CONSIDERED AT ONE TIME
(7)-(14) NAMES OF THE INPUTS

(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)
1	CRBR-CONT	OR	2	0	-0	NSR	SR						
2	NSR	OR	4	0	-0	LKG	LCV	ECV	CONTAIN(V)				
3	SR	OR	7	0	-0	MIS	OP	CI	LDV	EDV	MT	E-OP	
4	MTS	AND	1	1	-0	VF4	MS/VF4						
5	E-OP	AND	2	1	-0	VF4	NMS/VF4	OPO/VF4					
6	MT	AND	1	1	-0	VF	MT/VF						
7	VF	OR	3	0	-0	VF2	VF3	VF4					
8	LCV	OR	2	0	-0	U2	U40						
9	LDV	OR	2	0	-0	U4	U42						
10	U2	AND	6	0	-0	CF1	NCI	NAC	NVPS	NCU	NOP/CF1		
11	U4	AND	6	0	-0	CF1	NCI	NAC	NVPS	CU	NOP/CF1.CU		
12	ECV	OR	6	0	-0	U8	U20	U26	U46	U58	U64		
13	EDV	OR	6	0	-0	U10	U22	U28	U48	U60	U66		
14	CI	OR	7	0	-0	U14	U16	U18	U32	U34	U36	CONTINUE1	
15	CONTINUE1	OR	6	0	-0	U52	U54	U56	U70	U72	U74		
16	LKG	OR	8	0	-0	U6	U24	U12	U30	U44	U50	U62	U68
17	OP	OR	8	0	-0	U1	U3	U5	U7	U9	U11	U13	CONTINUE2
18	CONTINUE2	OR	8	0	-0	U15	U17	U19	U21	U23	U25	U27 ?	CONTINUE3
19	CONTINUE3	OR	8	0	-0	U29	U31	U33	U35	U39	U41	U43	CONTINUE4
20	CONTINUE4	OR	8	0	-0	U45	U47	U49	U51	U53	U55	U57	CONTINUE5
21	CONTINUE5	OR	8	0	-0	U59	U61	U63	U65	U67	U69	U71	U73
22	U1	AND	5	1	-0	CF1	NCI	NAC	NVPS	NCU	OP/CF1		
23	U3	AND	5	1	-0	CF1	NCI	NAC	NVPS	CU	OP/CF1.CU		
24	U5	AND	4	1	-0	CF1	NCI	NAC	VPS	OP/CF1.VPS			
25	U6	AND	5	0	-0	CF1	NCI	AC	VPS	NOP/CF1.VP			
26	U7	AND	5	1	-0	CF1	NCI	AC	NVPS	NCU	OP/CF1.AC		
27	U8	AND	6	0	-0	CF1	NCI	AC	NVPS	NCU	NOP/CF1.AC		
28	U9	AND	5	1	-0	CF1	NCI	AC	NVPS	CU	OP/1.AC.CU		
29	U10	AND	6	0	-0	CF1	NCI	AC	NVPS	CU	NOP/1.AC.C		
30	U11	AND	4	1	-0	CF1	NCI	AC	VPS	OP/1.AC.VP			
31	U12	AND	5	0	-0	CF1	NCI	AC	VPS	NOP/1.AC.V			
32	U13	AND	4	1	-0	CF1	NAC	NVPS	CIS	OP/CF1.CI			
33	U14	AND	5	0	-0	CF1	NAC	NVPS	NOP/CF1.CI	CIS			
34	U1?	AND	4	1	-0	CF1	NAC	VPS	CIS	OP/1.CI.VP			
35	U16	AND	5	0	-0	CF1	NAC	VPS	NOP/1.CI.V	CIS			
36	U17	AND	3	1	-0	CF1	AC	CIS	OP/1.CI.AC				
37	U18	AND	4	0	-0	CF1	AC	NOP/1.CI.A	CIS				
38	U19	AND	5	1	-0	CF2	NCI	NAC	NVPS	NCU	OP/CF2		
39	U20	AND	6	0	-0	CF2	NCI	NAC	NVPS	NCU	NOP/CF2		
40	U21	AND	5	1	-0	CF2	NCI	NAC	NVPS	CU	OP/CF2.CU		
41	U22	AND	6	0	-0	CF2	NCI	NAC	NVPS	CU	NOP/CF2.CU		
42	U23	AND	4	1	-0	CF2	NCI	NAC	VPS	OP/CF2.VPS			
43	U24	AND	5	0	-0	CF2	NCI	NAC	VPS	NOP/CF2.VP			
44	U25	AND	5	1	-0	CF2	NCI	AC	NVPS	NCU	OP/CF2.AC		
45	U26	AND	6	0	-0	CF2	NCI	AC	NVPS	NCU	NOP/CF2.AC		
46	U27	AND	5	1	-0	CF2	NCI	AC	NVPS	CU	OP/2.AC.CU		

47	U28	AND	6	0	-0	CF2	NCI	AC	NVPS	CU	NOP/2.AC.C	
48	U29	AND	4	0	-0	CF2	NCI	AC	VPS			
49	U30	AND	5	0	-0	CF2	NCI	AC	VPS	NOP/2.AC.V		
50	U31	AND	4	1	-0	CF3	NAC	NVPS	CIS	OP/CF2.CI		
51	U32	AND	5	0	-0	CF3	NAC	NVPS	NOP/CF2.CI	CIS		
52	U33	AND	4	1	-0	CF2	NAC	VPS	CIS	OP/2.CI.VP		
53	U34	AND	5	0	-0	CF2	NAC	VPS	NOP/2.CI.V	CIS		
54	U35	AND	3	1	-0	CF2	AC	CIS	OP/2.CI.AC			
55	U36	AND	4	0	-0	CF2	AC	NOP/2.CI.A	CIS			
56	U39	AND	7	1	-0	CF3	NCI	NAC	NVPS	NCU	NMS/VF4	NOPO/VF4	OP/CF3
57	U40	AND	8	0	-0	CF3	NCI	NAC	NVPS	NCU	NMS/VF4	NOPO/VF4	NOP/CF3
58	U41	AND	7	1	-0	CF3	NCI	NAC	NVPS	CU	NMS/VF4	NOPO/VF4	OP/CF3.CU
59	U42	AND	8	0	-0	CF3	NCI	NAC	NVPS	CU	NMS/VF4	NOPO/VF4	NOP/CF3.CU
60	U43	AND	6	1	-0	CF3	NCI	NAC	VPS	NMS/VF4	NOPO/VF4	OP/CF3.VPS	
61	U44	AND	7	0	-0	CF3	NCI	NAC	VPS	NMS/VF4	NOPO/VF4	NOP/CF3.VP	
62	U45	AND	7	1	-0	CF3	NCI	NVPS	NCU	AC	NMS/VF4	NOPO/VF4	OP/CF3.AC
63	U46	AND	8	0	-0	CF3	NCI	NVPS	NCU	AC	NMS/VF4	NOPO/VF4	NOP/CF3.AC
64	U47	AND	7	1	-0	CF3	NCI	NVPS	AC	CU	NMS/VF4	NOPO/VF4	OP/3.AC.CU
65	U48	AND	8	0	-0	CF3	NCI	NVPS	AC	CU	NMS/VF4	NOPO/VF4	NOP/3.AC.C
66	U49	AND	6	1	-0	CF3	NCI	AC	VPS	NMS/VF4	NOPO/VF4	OP/3.AC.VP	
67	U50	AND	7	0	-0	CF3	NCI	AC	VPS	NMS/VF4	NOPO/VF4	NOP/3.AC.V	
68	U51	AND	6	1	-0	CF3	NAC	NVPS	NMS/VF4	NOPO/VF4	CIS	OP/CF3.CI	
69	U52	AND	7	0	-0	CF3	NAC	NVPS	NMS/VF4	NOPO/VF4	NOP/CF3.CI	CIS	
70	U53	AND	6	1	-0	CF3	NAC	VPS	NMS/VF4	NOPO/VF4	CIS	OP/3.CI.VP	
71	U54	AND	7	0	-0	CF3	NAC	VPS	NMS/VF4	NOPO/VF4	NOP/3.CI.V	CIS	
72	U55	AND	5	1	-0	CF3	AC	NMS/VF4	NOPO/VF4	CIS	OP/3.CI.AC		
73	U56	AND	6	0	-0	CF3	AC	NMS/VF4	NOPO/VF4	NOP/3.CI.A	CIS		
74	U57	AND	7	1	-0	CF4	NCI	NAC	NVPS	NCU	NMS/VF4	NOPO/VF4	OP/CF4
75	U58	AND	8	0	-0	CF4	NCI	NAC	NVPS	NCU	NMS/VF4	NOPO/VF4	NOP/CF4
76	U59	AND	7	1	-0	CF4	NCI	NAC	NVPS	CU	NMS/VF4	NOPO/VF4	OP/CF4.CU
77	U60	AND	8	0	-0	CF4	NCI	NAC	NVPS	CU	NMS/VF4	NOPO/VF4	NOP/CF4.CU
78	U61	AND	6	1	-0	CF4	NCI	NAC	VPS	NMS/VF4	NOPO/VF4	OP/CF4.VPS	
79	U62	AND	7	0	-0	CF4	NCI	NAC	VPS	NMS/VF4	NOPO/VF4	NOP/CF4.VP	
80	U63	AND	7	1	-0	CF4	NCI	NVPS	AC	NCU	NMS/VF4	NOPO/VF4	OP/CF4.AC
81	U64	AND	8	0	-0	CF4	NCI	NVPS	NCU	AC	NMS/VF4	NOPO/VF4	NOP/CF4.AC
82	U65	AND	7	1	-0	CF4	NCI	NVPS	AC	CU	NMS/VF4	NOPO/VF4	OP/4.AC.CU
83	U66	AND	8	0	-0	CF4	NCI	NVPS	AC	CU	NMS/VF4	NOPO/VF4	NOP/4.AC.C
84	U67	AND	6	1	-0	CF4	NCI	AC	VPS	NMS/VF4	NOPO/VF4	OP/4.AC.VP	
85	U68	AND	7	0	-0	CF4	NCI	AC	VPS	NMS/VF4	NOPO/VF4	NOP/4.AC.V	
86	U69	AND	6	1	-0	CF4	NAC	NVPS	NMS/VF4	NOPO/VF4	CIS	OP/CF4.CI	
87	U70	AND	7	0	-0	CF4	NAC	NVPS	NMS/VF4	NOPO/VF4	NOP/CF4.CI	CIS	
88	U71	AND	6	1	-0	CF4	NAC	VPS	NMS/VF4	NOPO/VF4	CIS	OP/4.CI.VP	
89	U72	AND	7	0	-0	CF4	NAC	VPS	NMS/VF4	NOPO/VF4	NOP/4.CI.V	CIS	
90	U73	AND	5	1	-0	CF4	AC	NMS/VF4	NOPO/VF4	CIS	OP/4.CI.AC		
91	U74	AND	6	0	-0	CF4	AC	NMS/VF4	NOPO/VF4	NOP/4.CI.A	CIS		
92	CIS	OR	0	2	-0	CISS	CISS						
93	NCI	NOT	1	0	-0	CIS							
94	NAC	NOT	1	0	-0	AC							
95	NVPS	NOT	1	0	-0	VPS							
96	NCU	NOT	1	0	-0	CU							
97	NMS/VF4	NOT	0	1	-0	MS/VF4							
98	NOPO/VF4	NOT	0	1	-0	OPO/VF4							
99	NOP/CF1	NOT	0	1	-0	OP/CF1							
100	NOP/CF1.CU	NOT	0	1	-0	OP/CF1.CU							
101	NOP/CF1.VP	NOT	0	1	-0	OP/CF1.VP							
102	NOP/CF1.AC	NOT	0	1	-0	OP/CF1.AC							
103	NOP/1.AC.C	NOT	0	1	-0	OP/1.AC.CU							
104	NOP/1.AC.V	NOT	0	1	-0	OP/1.AC.VP							
105	NOP/CF1.CI	NOT	0	1	-0	OP/CF1.CI							
106	NOP/1.CI.V	NOT	0	1	-0	OP/1.CI.VP							

107	NOP/1.CI.A	NOT	0	1	-0	OP/1.CI.AC							
108	NOP/CF2	NOT	0	1	-0	OP/CF2							
109	NOP/CF2.CU	NOT	0	1	-0	OP/CF2.CU							
110	NOP/CF2.VP	NOT	0	1	-0	OP/CF2.VPS							
111	NOP/CF2.AC	NOT	0	1	-0	OP/CF2.AC							
112	NOP/2.AC.C	NOT	0	1	-0	OP/2.AC.CU							
113	NOP/2.AC.V	NOT	0	1	-0	OP/2.AC.VP							
114	NOP/CF2.CI	NOT	0	1	-0	OP/CF2.CI							
115	NOP/2.CI.V	NOT	0	1	-0	OP/2.CI.VP							
116	NOP/2.CI.A	NOT	0	1	-0	OP/2.CI.AC							
117	NOP/CF3	NOT	0	1	-0	OP/CF3							
118	NOP/CF3.CU	NOT	0	1	-0	OP/CF3.CU							
119	NOP/CF3.VP	NOT	0	1	-0	OP/CF3.VPS							
120	NOP/CF3.AC	NOT	0	1	-0	OP/CF3.AC							
121	NOP/3.AC.C	NOT	0	1	-0	OP/3.AC.CU							
122	NOP/3.AC.V	NOT	0	1	-0	OP/3.AC.VP							
123	NOP/CF3.CI	NOT	0	1	-0	OP/CF3.CI							
124	NOP/3.CI.V	NOT	0	1	-0	OP/3.CI.VP							
125	NOP/3.CI.A	NOT	0	1	-0	OP/3.CI.AC							
126	NOP/CF4	NOT	0	1	-0	OP/CF4							
127	NOP/CF4.CU	NOT	0	1	-0	OP/CF4.CU							
128	NOP/CF4.VP	NOT	0	1	-0	OP/CF4.VP							
129	NOP/CF4.AC	NOT	0	1	-0	OP/CF4.AC							
130	NOP/4.AC.C	NOT	0	1	-0	OP/4.AC.CU							
131	NOP/4.AC.V	NOT	0	1	-0	OP/4.AC.VP							
132	NOP/CF4.CI	NOT	0	1	-0	OP/CF4.CI							
133	NOP/4.CI.V	NOT	0	1	-0	OP/4.CI.VP							
134	NOP/4.CI.A	NOT	0	1	-0	OP/4.CI.AC							
135	CF1	AND	2	0	-0	NCD23	VF2+VF3						
136	CF2	AND	1	1	-0	VF2+VF3	CD/VF23						
137	CF3	AND	2	0	-0	NCD4	VF4						
138	CF4	AND	1	1	-0	VF4	CD/VF4						
139	VF2+VF3	OR	2	0	-0	VF2	VF3						
140	NCD23	NOT	0	1	-0	CD/VF23							
141	NCD4	NOT	0	1	-0	CD/VF4							
142	CONTAIN(V.)	OR	4	0	-0	V1	V5	V9	V13				
143	VF2	OR	8	0	-0	V2	V6	V10	V14	V15	V24	V21	V18
144	VF3	OR	9	0	-0	V3	V7	V11	V16	V19	V22	V26	CONTINUE6
145	CONTINUE6	OR	3	0	-0	V29	V32	V34					
146	VF4	OR	8	0	-0	V4	V8	V12	V17	V20	V23	V27	CONTINUE7
147	CONTINUE7	OR	7	0	-0	V30	V33	V35	V36	V37	V38	V39	
148	V1	AND	1	4	-0	CM	ER/CM	FCI/CM1	VHR11	VBR1			
149	V2	AND	2	3	-0	CM	NVBR1	ER/CM	FCI/CM1	VHR11			
150	V3	AND	1	3	-0	CM	VHR12	ER/CM	FCI/CM1				
151	V4	AND	1	3	-0	CM	ER/CM	FCI/CM1	VHR13				
152	V5	AND	2	3	-0	CM	NFCI/CM1	ER/CM	VHR21	VBR2			
153	V6	AND	3	2	-0	CM	NFCI/CM1	NVBR2	ER/CM	VHR21			
154	V7	AND	2	2	-0	CM	NFCI/CM1	VHR22	ER/CM				
155	V8	AND	2	2	-0	CM	NFCI/CM1	ER/CM	VHR23				
156	V9	AND	2	3	-0	CM	NER/CM	FCI/CM2	VHR31	VBR3			
157	V10	AND	3	2	-0	CM	NER/CM	NVBR3	FCI/CM2	VHR31			
158	V11	AND	2	2	-0	CM	NER/CM	VHR32	FCI/CM2				
159	V12	AND	2	2	-0	CM	NER/CM	FCI/CM2	VHR33				
160	V13	AND	3	1	-0	CM	NER/CM	NFCI/CM2	VBR4				
161	V14	AND	4	0	-0	CM	NER/CM	NFCI/CM2	NVBR4				
162	V15	AND	1	3	-0	DC1	ER/DC1	FCI/DC11	VHR41				
163	V16	AND	1	3	-0	DC1	VHR42	ER/DC1	FCI/DC11				
164	V17	AND	1	3	-0	DC1	ER/DC1	FCI/DC11	VHR43				
165	V18	AND	2	2	-0	DC1	NFCI/DC11	ER/DC1	VHR51				
166	V19	AND	2	2	-0	DC1	NFCI/DC11	VHR52	ER/DC1				

167	V20	AND	2	2	-0	DC1	NFCI/DC11	ER/DC1	VHR53			
168	V21	AND	2	2	-0	DC1	NER/DC1	FCI/DC12	VHR61			
169	V22	AND	2	2	-0	DC1	NER/DC1	VHR62	FCI/DC12			
170	V23	AND	2	2	-0	DC1	NER/DC1	FCI/DC12	VHR63			
171	V24	AND	3	0	-0	DC1	NER/DC1	NFCI/DC12				
172	V26	AND	1	3	-0	DC2	VHR72	ER/DC2	FCI/DC21			
173	V27	AND	1	3	-0	DC2	ER/DC2	FCI/DC21	VHR73			
174	V29	AND	2	2	-0	DC2	NFCI/DC21	VHR82	ER/DC2			
175	V30	AND	2	2	-0	DC2	NFCI/DC21	ER/DC2	VHR83			
176	V32	AND	2	2	-0	DC2	NER/DC2	VHR92	FCI/DC22			
177	V33	AND	2	2	-0	DC2	NER/DC2	FCI/DC22	VHR93			
178	V34	AND	3	0	-0	DC2	NER/DC2	NFCI/DC22				
179	V35	AND	1	1	-0	DC3	FD					
180	V36	AND	2	2	-0	DC3	NFD	ER/DC3	FCI/DC31			
181	V37	AND	3	1	-0	DC3	NFD	NFCI/DC31	ER/DC3			
182	V38	AND	3	1	-0	DC3	NFD	NER/DC3	FCI/DC32			
183	V39	AND	4	0	-0	DC3	NFD	NER/DC3	NFCI/DC32			
184	NVBR1	NOT	0	1	-0	VBR1						
185	NVBR2	NOT	0	1	-0	VBR2						
186	NVBR3	NOT	0	1	-0	VBR3						
187	NVBR4	NOT	0	1	-0	VBR4						
188	NFCI/CM1	NOT	0	1	-0	FCI/CM1						
189	NFCI/CM2	NOT	0	1	-0	FCI/CM2						
190	NER/CM	NOT	0	1	-0	ER/CM						
191	NER/DC1	NOT	0	1	-0	ER/DC1						
192	NER/DC2	NOT	0	1	-0	ER/DC2						
193	NER/DC3	NOT	0	1	-0	ER/DC3						
194	NFCI/DC11	NOT	0	1	-0	FCI/DC11						
195	NFCI/DC12	NOT	0	1	-0	FCI/DC12						
196	NFCI/DC21	NOT	0	1	-0	FCI/DC21						
197	NFCI/DC22	NOT	0	1	-0	FCI/DC22						
198	NFCI/DC31	NOT	0	1	-0	FCI/DC31						
199	NFCI/DC32	NOT	0	1	-0	FCI/DC32						
200	NFD	NOT	0	1	-0	FD						
201	CM	OR	2	0	-0	CM.PCDA	CM.UCDA					
202	CM.PCDA	OR	4	0	-0	CM.CF	CM.TOP	CM.LOCA	CM.LOHS			
203	CM.UCDA	OR	7	0	-0	CM.UCF	CM.UTOP	CM.UTO&LOF	CM.ULOCA	CM.ULOF	CM.ULOHS	CM.UCF&LOF
204	CM.CF	AND	1	1	-0	CF	CM/CF					
205	CM.TOP	AND	1	1	-0	TOP	CM/TOP					
206	CM.LOCA	AND	1	1	-0	LOCA	CM/LOCA					
207	CM.LOHS	AND	1	1	-0	LOHS	CM/LOHS					
208	CM.UCF	AND	1	1	-0	UCF	CM/UCF					
209	CM.UTOP	AND	1	1	-0	UTOP	CM/UTOP					
210	CM.UTO&LOF	AND	1	1	-0	UTOP&ULOF	CM/UT&LOF					
211	CM.ULOCA	AND	1	1	-0	ULOCA	CM/ULOCA					
212	CM.ULOF	AND	1	1	-0	ULOF	CM/ULOF					
213	CM.ULOHS	AND	1	1	-0	ULOHS	CM/ULOHS					
214	CM.UCF&LOF	AND	1	1	-0	UCF&ULOF	CM/UC&LOF					
215	DC1	OR	2	0	-0	DC1.PCDA	DC1.UCDA					
216	DC1.PCDA	OR	4	0	-0	DC1.CF	DC1.TOP	DC1.LOCA	DC1.LOHS			
217	DC1.UCDA	OR	7	0	-0	DC1.UCF	DC1.UTOP	DC1.UT&LOF	DC1.ULOCA	DC1.ULOF	DC1.ULOHS	DC1.UC&LOF
218	DC1.CF	AND	1	1	-0	CF	DC1/CF					
219	DC1.TOP	AND	1	1	-0	TOP	DC1/TOP					
220	DC1.LOCA	AND	1	1	-0	LOCA	DC1/LOCA					
221	DC1.LOHS	AND	1	1	-0	LOHS	DC1/LOHS					
222	DC1.UCF	AND	1	1	-0	UCF	DC1/UCF					
223	DC1.UTOP	AND	1	1	-0	UTOP	DC1/UTOP					
224	DC1.UT&LOF	AND	1	1	-0	UTOP&ULOF	DC1/UT&LOF					
225	DC1.ULOCA	AND	1	1	-0	ULOCA	DC1/ULOCA					
226	DC1.ULOF	AND	1	1	-0	ULOF	DC1/ULOF					

227	DC1.ULOHS	AND	1	1	-0	ULOHS	DC1/ULOHS						
228	DC1.UC&LOF	AND	1	1	-0	UCF&ULOF	DC1/UC&LOF						
229	DC2	OR	2	0	-0	DC2.PCDA	DC2.UCDA						
230	DC2.PCDA	OR	4	0	-0	DC2.CF	DC2.TOP	DC2.LOCA	DC2.LOHS				
231	DC2.UCDA	OR	7	0	-0	DC2.UCF	DC2.UTOP	DC2.UT&LOF	DC2.ULOCA	DC2.ULOF	DC2.ULOHS	DC2.UC&LOF	
232	DC2.CF	AND	1	1	-0	CF	DC2/CF						
233	DC2.TOP	AND	1	1	-0	TOP	DC2/TOP						
234	DC2.LOCA	AND	1	1	-0	LOCA	DC2/LOCA						
235	DC2.LOHS	AND	1	1	-0	LOHS	DC2/LOHS						
236	DC2.UCF	AND	1	1	-0	UCF	DC2/UCF						
237	DC2.UTOP	AND	1	1	-0	UTOP	DC2/UTOP						
238	DC2.UT&LOF	AND	1	1	-0	UTOP&ULOF	DC2/UT&LOF						
239	DC2.ULOCA	AND	1	1	-0	ULOCA	DC2/ULOCA						
240	DC2.ULOF	AND	1	1	-0	ULOF	DC2/ULOF						
241	DC2.ULOHS	AND	1	1	-0	ULOHS	DC2/ULOHS						
242	DC2.UC&LOF	AND	1	1	-0	UCF&ULOF	DC2/UC&LOF						
243	DC3	OR	2	0	-0	DC3.PCDA	DC3.UCDA						
244	DC3.PCDA	OR	4	0	-0	DC3.CF	DC3.TOP	DC3.LOCA	DC3.LOHS				
245	DC3.UCDA	OR	7	0	-0	DC3.UCF	DC3.UTOP	DC3.UT&LOF	DC3.ULOCA	DC3.ULOF	DC3.ULOHS	DC3.UC&LOF	
246	DC3.CF	AND	1	1	-0	CF	DC3/CF						
247	DC3.TOP	AND	1	1	-0	TOP	DC3/TOP						
248	DC3.LOCA	AND	1	1	-0	LOCA	DC3/LOCA						
249	DC3.LOHS	AND	1	1	-0	LOHS	DC3/LOHS						
250	DC3.UCF	AND	1	1	-0	UCF	DC3/UCF						
251	DC3.UTOP	AND	1	1	-0	UTOP	DC3/UTOP						
252	DC3.UT&LOF	AND	1	1	-0	UTOP&ULOF	DC3/UT&LOF						
253	DC3.ULOCA	AND	1	1	-0	ULOCA	DC3/ULOCA						
254	DC3.ULOF	AND	1	1	-0	ULOF	DC3/ULOF						
255	DC3.ULOHS	AND	1	1	-0	ULOHS	DC3/ULOHS						
256	DC3.UC&LOF	AND	1	1	-0	UCF&ULOF	DC3/UC&LOF						
257	AC	OR	1	5	-0	AC1	OPR-ACS	ACS-IN	EXHAUST	P-SBTCB	CC-CFANS		
258	AC1	COM	6	0	4	AC2	AC3	AC4	AC5	AC6	AC7		

COMBINATION GATE 6 EVENTS 4 AT A TIME

258	AC1	OR	1	0	ADDCOM2							
259	COMBO1	AND	4	0	AC2	AC3	AC4	AC5				
260	COMBO2	AND	4	0	AC2	AC3	AC4	AC6				
261	COMBO3	AND	4	0	AC2	AC3	AC4	AC7				
262	COMBO4	AND	4	0	AC2	AC3	AC5	AC6				
263	COMBO5	AND	4	0	AC2	AC3	AC5	AC7				
264	COMBO6	AND	4	0	AC2	AC3	AC6	AC7				
265	COMBO7	AND	4	0	AC2	AC4	AC5	AC6				
266	COMBO8	AND	4	0	AC2	AC4	AC5	AC7				
267	COMBO9	AND	4	0	AC2	AC4	AC6	AC7				
268	COMBO10	AND	4	0	AC2	AC5	AC6	AC7				
269	COMBO11	AND	4	0	AC3	AC4	AC5	AC6				
270	COMBO12	AND	4	0	AC3	AC4	AC5	AC7				
271	COMBO13	AND	4	0	AC3	AC4	AC6	AC7				
272	COMBO14	AND	4	0	AC3	AC5	AC6	AC7				
273	COMBO15	AND	4	0	AC4	AC5	AC6	AC7				
274	ADDCOM1	OR	8	0	COMBO8	COMBO9	COMBO10	COMBO11	COMBO12	COMBO13	COMBO14	COMBO15
275	ADDCOM2	OR	8	0	ADDCOM1	COMBO1	COMBO2	COMBO3	COMBO4	COMBO5	COMBO6	COMBO7

276	AC2	OR	1	2	-0	SG11	D-1A-PL	CF-1A
277	AC3	OR	1	2	-0	SG14	D-2A-PL	CF-2A
278	AC4	OR	1	2	-0	SG14C	D-3A-PL	CF-3A
279	AC5	OR	1	2	-0	SG11	D-1B-PL	CF-1B
280	AC6	OR	1	2	-0	SG14	D-2B-PL	CF-2B
281	AC7	OR	1	2	-0	SG14C	D-3B-PL	CF-3B

282	VPS	OR	1	5	-0	VPS1	OPR-VPS	VPS-IN	EXHAUST	V-PA-F	CC-VALVS		
283	VPS1	OR	2	0	-0	VPS2	VPS6						
284	VPS2	OR	1	1	-0	VPS3	CC-VFANS						
285	VPS3	AND	2	0	-0	VPS4	VPS5						
286	VPS4	OR	1	1	-0	SGI1	VF-1A						
287	VPS5	OR	1	1	-0	SGI4	VF-1B						
288	VPS6	AND	2	0	-0	VPS7	VPS8						
289	VPS7	OR	1	2	-0	SGI1	V-3A-F	V-4A-F					
290	VPS8	OR	1	2	-0	SGI4	V-3B-F	V-4B-F					
291	CU	OR	1	2	-0	CU1	OPR-CUS	RUP-CUS					
292	CU1	OR	1	3	-0	CU2	M-SC	J-SC	A-MS				
293	CU2	OR	1	1	-0	CU3	CC-CUPPS						
294	CU3	AND	2	0	-0	CU4	CU5						
295	CU4	OR	1	1	-0	SGI1	CP-1A						
296	CU5	OR	1	1	-0	SGI4	CP-1B						
297	LOHS	OR	6	0	-0	LOHS1	LOHS2	LOHS6	LOHS7	LOHS8	LOHSCONT		
298	LOHSCONT	OR	5	0	-0	LOHS9	LOHS10	LOHS11	LOHS12	LOHS13			
299	LOHS1	AND	1	1	-0	SHRS/	I(1)						
300	SHRS/	AND	3	0	-0	SHRS	NPPS	NSCRAM					
301	LOHS2	AND	1	1	-0	SHRS/	I(2)						
302	LOHS6	AND	1	1	-0	SHRS/	I(6)						
303	LOHS7	AND	1	1	-0	SHRS/	I(7)						
304	LOHS8	AND	1	1	-0	SHRS/	I(8)						
305	LOHS9	AND	1	1	-0	SHRS/	I(9)						
306	LOHS10	AND	1	1	-0	SHRS/	I(10)						
307	LOHS11	AND	1	1	-0	SHRS/	I(11)						
308	LOHS12	AND	0	2	-0	SHRS-	I(12)						
309	LOHS13	AND	1	1	-0	SHRS/	I(13)						
310	CF	AND	1	1	-0	SHRS/	I(3)						
311	TOP	AND	1	1	-0	SHRS/	I(4)						
312	UCF&ULOF	AND	1	1	-0	ULOF/	I(3)						
313	ULOF/	AND	2	1	-0	NPPS	NPT	SCRAM					
314	UTOP&ULOF	AND	1	1	-0	ULOF/	I(4)						
315	LOCA	AND	2	2	-0	NPPS	NSCRAM	SHRS-	I(5)				
316	ULOCA	AND	1	1	-0	ULOCA/	I(5)						
317	ULOCA/	OR	0	2	-0	PPS	SCRAM						
318	UCF	AND	1	1	-0	UCF/	I(3)						
319	UCF/	OR	1	1	-0	UCF//	PPS						
320	UCF//	AND	0	2	-0	SCRAM	PT						
321	UTOP	AND	1	1	-0	UCF/	I(4)						
322	ULOF	OR	8	0	-0	ULOF6	ULOF7	ULOF8	ULOF9	ULOF10	ULOF11	ULOF12	ULOF13
323	ULOF6	AND	1	1	-0	ULOF/	I(6)						
324	ULOF7	AND	1	1	-0	ULOF/	I(7)						
325	ULOF8	AND	1	1	-0	ULOF/	I(8)						
326	ULOF9	AND	1	1	-0	ULOF/	I(9)						
327	ULOF10	AND	1	1	-0	ULOCA/	I(10)						
328	ULOF11	AND	1	1	-0	ULOF/	I(11)						
329	ULOF12	AND	1	1	-0	ULOF/	I(12)						
330	ULOF13	AND	1	1	-0	ULOF/	I(13)						
331	ULOHS	OR	7	0	-0	ULOHS6	ULOHS7	ULOHS8	ULOHS9	ULOHS11	ULOHS12	ULOHS13	
332	ULOHS6	AND	1	1	-0	UCF/	I(6)						
333	ULOHS7	AND	1	1	-0	UCF/	I(7)						
334	ULOHS8	AND	1	1	-0	UCF/	I(8)						
335	ULOHS9	AND	1	1	-0	UCF/	I(9)						
336	ULOHS11	AND	1	1	-0	UCF/	I(11)						
337	ULOHS12	AND	1	1	-0	UCF/	I(12)						
338	ULOHS13	AND	1	1	-0	UCF/	I(13)						
339	NPPS	NOT	0	1	-0	PPS							
340	NSCRAM	NOT	0	1	-0	SCRAM							
341	NPT	NOT	0	1	-0	PT							

342	SHRS	OR	8	0	-0	S4	S5	S6	S7	S10	S11	S14	CONT.11
343	CONT.11	OR	8	0	-0	S15	S16	S17	S20	S21	S22	S24	CONT.12
344	CONT.12	OR	8	0	-0	S25	S26	S27	S30	S31	S33	S34	CONT.13
345	CONT.13	OR	7	0	-0	S37	S38	S39	S41	S42	S43	S44	
346	S4	AND	8	0	-0	NA1	NA2	NB1	NE	NGO	NF	B2	CONT.14
347	CONT.14	AND	2	0	-0	D	G						
348	S5	AND	7	1	-0	NA1	NA2	NB1	NE	NF	B2	D	GO
349	S6	AND	7	0	-0	NA1	NA2	NB1	NE	B2	D	F	
350	S7	AND	6	0	-0	NA1	NA2	NB1	B2	D	F		
351	S10	AND	7	0	-0	NA1	NA2	NB2	NE	B1	D	G	
352	S11	AND	6	0	-0	NA1	NA2	NB2	B1	D	E		
353	S14	AND	5	0	-0	NA1	NA2	NF	NGO	CONT.15			
354	CONT.15	AND	5	0	-0	NF	B1	B2	D	G			
355	S15	AND	7	1	-0	NA1	NA2	NE	NF	B1	B2	D	GO
356	S16	AND	7	0	-0	NA1	NA2	NE	B1	B2	D	F	
357	S17	AND	6	0	-0	NA1	NA2	B1	B2	D	E		
358	S20	AND	7	0	-0	NA1	NC	NE	NGO	A2	D	G	
359	S21	AND	5	1	-0	NA1	NC	NE	A2	D	GO		
360	S22	AND	5	0	-0	NA1	NC	A2	D	E			
361	S24	AND	7	0	-0	NA1	NE	NF	NGO	A2	C	G	
362	S25	AND	5	1	-0	NA1	NE	NF	A2	C	GO		
363	S26	AND	5	0	-0	NA1	NE	A2	C	F			
364	S27	AND	4	0	-0	NA1	A2	C	E				
365	S30	AND	6	0	-0	NA2	NC	NE	A1	D	G		
366	S31	AND	5	0	-0	NA2	NC	A1	D	E			
367	S33	AND	5	0	-0	NA2	NE	A1	C	G			
368	S34	AND	4	0	-0	NA2	A1	C	E				
369	S37	AND	7	0	-0	NC	NE	NGO	A1	A2	D	G	
370	S38	AND	5	1	-0	NC	NE	A1	A2	D	GO		
371	S39	AND	5	0	-0	NC	A1	A2	D	E			
372	S41	AND	7	0	-0	NE	NF	NGO	A1	A2	C	G	
373	S42	AND	5	1	-0	NE	NF	A1	A2	C	GO		
374	S43	AND	5	0	-0	NE	A1	A2	C	F			
375	S44	AND	4	0	-0	A1	A2	C	E				
376	NA1	NOT	1	0	-0	A1							
377	NA2	NOT	1	0	-0	A2							
378	NB1	NOT	1	0	-0	B1							
379	NB2	NOT	1	0	-0	B2							
380	NC	NOT	1	0	-0	C							
381	NE	NOT	1	0	-0	E							
382	NF	NOT	1	0	-0	F							
383	NGO	NOT	0	1	-0	GO							
384	E	OR	0	2	-0	OP-RSHRS	OP-RSHRS						
385	A1	OR	2	0	-0	SG6	SG7						
386	SG6	OR	0	2	-0	RVESSELR	PLOOPSOR						
387	SG7	AND	3	0	-0	SG8	SG9	SG10					
388	SG8	OR	3	3	-0	SG11	SG8P	SG8I	MHXI001F	MSGDR01F	MLOOP01R		
389	SG11	AND	2	0	-0	SG12X	SG13						
390	SG12X	OR	1	1	-0	SG12	MNET						
391	SG12	AND	0	2	-0	LOSP	NRLOSP						
392	SG13	OR	1	2	-0	SG13A	DGA	DGABC					
393	SG13A	OR	0	2	-0	DGAB	DGCA						
394	SG9	OR	3	3	-0	SG14	SG9P	SG9I	MHXI002F	MSGDR02F	MLOOP02R		
395	SG14	AND	2	0	-0	SG12X	SG15						
396	SG15	OR	1	2	-0	SG15A	DGB	DGABC					
397	SG15A	OR	0	2	-0	DGAB	DGABC						
398	SG10	OR	3	3	-0	SG14C	SG10P	SG10I	MHXI003F	MSGDR03F	MLOOP03R		
399	SG14C	AND	2	0	-0	SG12X	SG15C						
400	SG15C	OR	1	2	-0	SG15CA	DGC	DGABC					
401	SG15CA	OR	0	2	-0	DGBC	DGCA						

402	SG8P	OR	1	2	-0	SG8PA	PPM01	PPM0123			
403	SG8PA	OR	0	2	-0	PPM012	PPM031				
404	SG8I	OR	1	2	-0	SG8IA	IPM01	IPM0123			
405	SG8IA	OR	0	2	-0	IPM012	IPM031				
406	SG9P	OR	1	2	-0	SG9PA	PPM02	PPM0123			
407	SG9PA	OR	0	2	-0	PPM012	PPM023				
408	SG9I	OR	1	2	-0	SG9IA	IPM02	IPM0123			
409	SG9IA	OR	0	2	-0	IPM012	IPM023				
410	SG10P	OR	1	2	-0	SG10PA	PPM03	PPM0123			
411	SG10PA	OR	0	2	-0	PPM023	PPM031				
412	SG10I	OR	1	2	-0	SG10IA	IPM03	IPM0123			
413	SG10IA	OR	0	2	-0	IPM023	IPM031				
414	A2	AND	2	0	-0	MF	AF				
415	MF	OR	1	2	-0	SG12X	MAINFEED	MAINCOND			
416	AF	OR	1	3	-0	SG44	NSIGAFWS	AVALVES0	NOWATER0		
417	SG44	AND	2	0	-0	SG45	SG44A				
418	SG44A	OR	0	2	-0	APMTURB0	NOSTEAM0				
419	SG45	OR	2	0	-0	SG46	SG47				
420	SG46	OR	1	1	-0	SG11	APMA000A				
421	SG47	OR	1	1	-0	SG14	APMB000A				
422	C	OR	3	2	-0	SG6	SG6A	SG39	DOPERR	UNADHRS	
423	SG6A	OR	3	0	-0	SG6A1	SG6A2	SG6A3			
424	SG6A1	OR	2	0	-0	SG11	SG8P				
425	SG6A2	OR	2	0	-0	SG14	SG9P				
426	SG6A3	OR	2	0	-0	SG14C	SG10P				
427	SG39	OR	2	3	-0	SG40	SG41X	DVALVES0	DGVESSLR	DHXOVFLF	
428	SG40	OR	2	0	-0	SG42X	SG43X				
429	SG42X	OR	2	0	-0	SG11	SG42				
430	SG42	OR	0	3	-0	DPMNAK1F	DBL0001F	DHX0001F			
431	SG43X	OR	2	0	-0	SG14	SG43				
432	SG43	OR	0	3	-0	DPMNAK2F	DBL0002F	DHX0002F			
433	SG41X	OR	2	0	-0	SG41X1	SG41X2				
434	SG41X1	OR	1	1	-0	SG11	DPMNA01F				
435	SG41X2	OR	1	1	-0	SG14	DPMNA02F				
436	F	OR	2	0	-0	SG6	SG50				
437	SG50	AND	3	0	-0	SG51	SG52	SG53			
438	SG51	OR	1	4	-0	SG51A	MLOOP01R	M[illegible]001F	MSGDR01F	MCV0001P	
439	SG52	OR	1	4	-0	SG52A	MLOOP02R	MHXI002F	MSGDR02F	MCV0002P	
440	SG53	OR	1	4	-0	SG53A	MLOOP03R	MHXI003F	MSGDR03F	MCV0003P	
441	SG51A	OR	0	3	-0	PHXPAC1P	PMV0001C	PRV0001D			
442	SG52A	OR	0	3	-0	PHXPAC2P	PMV0002C	PRV0002D			
443	SG53A	OR	0	3	-0	PHXPAC3P	PMV0003C	PRV0003D			
444	B1	OR	2	0	-0	SG6L	SG7L				
445	SG6L	OR	0	2	-0	RVESSELRL	PLOOPSORL				
446	SG7L	AND	3	0	-0	SG8L	SG9L	SG10L			
447	SG8L	OR	3	3	-0	SG11L	SG8PL	SG8IL	MHXI001FL	MSGDR01FL	MLOOP01RL
448	SG11L	AND	2	0	-0	SG12XL	SG13L				
449	SG12XL	OR	1	1	-0	SG12L	MNETL				
450	SG12L	AND	0	2	-0	LOSP	NRLOSPL				
451	SG13L	OR	1	2	-0	SG13AL	DGAL	SGABCL			
452	SG13AL	OR	0	2	-0	DGABL	DGCAL				
453	SG9L	OR	3	3	-0	SG14L	SG9PL	SG9IL	MHXI002FL	MSGDR02FL	MLOOP02RL
454	SG14L	AND	2	0	-0	SG12XL	SG15L				
455	SG15L	OR	1	2	-0	SG15AL	DGBL	DGABCL			
456	SG15AL	OR	0	2	-0	DGABL	DGABCL				
457	SG10L	OR	3	3	-0	SG14CL	SG10PL	SG10IL	MHXI003FL	MSGDR03FL	MLOOP03RL
458	SG14CL	AND	2	0	-0	SG12XL	SG15CL				
459	SG15CL	OR	1	2	-0	SG15CAL	DGCL	DGABCL			
460	SG15CAL	OR	0	2	-0	DGBCL	DGCAL				
461	SG8PL	OR	1	2	-0	SG8PAL	PPM01L	PPM0123L			

462	SG8PAL	OR	0	2	-0	PPM012L	PPM031L			
463	SG8IL	OR	1	2	-0	SG8IAL	IPM01L	IPM0123L		
464	SG8IAL	OR	0	2	-0	IPM012L	IPM031L			
465	SG9PL	OR	1	2	-0	SG9PAL	PPM02L	PPM0123L		
466	SG9PAL	OR	0	2	-0	PPM012L	PPM023L			
467	SG9IL	OR	1	2	-0	SG9IAL	IPM02L	IPM0123L		
468	SG9IAL	OR	0	2	-0	IPM012L	IPM023L			
469	SG10PL	OR	1	2	-0	SG10PAL	PPM03L	PPM0123L		
470	SG10PAL	OR	0	2	-0	PPM023L	PPM031L			
471	SG10IL	OR	1	2	-0	SG10IAL	IPM03L	IPM0123L		
472	SG10IAL	OR	0	2	-0	IPM023L	IPM031L			
473	B2	AND	2	0	-0	MFL	AFL			
474	MFL	OR	1	2	-0	SG12XL	MAINFEEDL	MAINCONDL		
475	AFL	OR	1	2	-0	SG44L	AVALVESOL	NOWATEROL		
476	SG44L	AND	2	0	-0	SG45L	SG44AL			
477	SG44AL	OR	0	2	-0	APMTURBOL	NOSTEAMOL			
478	SG45L	AND	2	0	-0	SG46L	SG47L			
479	SG46L	OR	1	1	-0	SG11L	APMA000AL			
480	SG47L	OR	1	1	-0	SG14L	APMB000AL			
481	D	OR	3	1	-0	SG6L	SG6AL	SG39L	UNADHRS	
482	SG6AL	AND	3	0	-0	SG6A1L	SG6A2L	SG6A3L		
483	SG6A1L	OR	2	0	-0	SG11L	SG8PL			
484	SG6A2L	OR	2	0	-0	SG14L	SG9PL			
485	SG6A3L	OR	2	0	-0	SG14CL	SG10PL			
486	SG39L	OR	2	3	-0	SG40L	SG41XL	DVALVESOL	DOVESSLRL	DHXOVFLFL
487	SG40L	OR	2	0	-0	SG42XL	SG43XL			
488	SG42XL	OR	2	0	-0	SG11L	SG42L			
489	SG42L	OR	0	3	-0	DPMNAK1FL	DBL0001FL	DHX0001FL		
490	SG43XL	OR	2	0	-0	SG14L	SG43L			
491	SG43L	OR	0	3	-0	DPMNAK2FL	DBL0002FL	DHX0002FL		
492	SG41XL	OR	2	0	-0	SG41X1L	SG41X2L			
493	SG41X1L	OR	1	1	-0	SG11L	DPMNA01FL			
494	SG41X2L	OR	1	1	-0	SG14L	DPMNA02FL			
495	G	AND	2	0	-0	MFN	AFN			
496	MFN	OR	1	2	-0	SG12XL	MAINFEEDN	MAINCONDN		
497	AFN	OR	1	2	-0	SG44N	AVALVESON	NOWATERON		
498	SG44N	AND	2	0	-0	SG45N	SG44AN			
499	SG44AN	OR	0	2	-0	APMTURBON	NOSTEAMON			
500	SG45N	AND	2	0	-0	SG46N	SG47N			
501	SG46N	OR	1	1	-0	SG11L	APMA000AN			
502	SG47N	OR	1	1	-0	SG14L	APMB000AN			

Appendix B
DATA BASE

Failure data and data source employed in this study are presented here. Table B-1 shows the input data for the component or human failures, and Table B-2 lists the frequences estimates for the phenomenological uncertainties.

Table B.1

EQUIPMENT AND HUMAN FAILURE DATA

Component	Failure Mode	Event Code	Failure Frequency (Mean)	Reference
Reactor Vessel	Rupture	RVESSELR	2.7×10^{-7}/hr	SAND81-0260
Primary Loop	Rupture	PLOOPSØR	2.7×10^{-8}/hr	SAND81-0260
Intermediate Loop	Rupture	MLOOPØ1R 2 3	2.1×10^{-7}/hr	SAND81-0260
Intermediate Heat Exchanger (IHX)	Rupture & Plugging	MHXIØØ1F 2 3	9.1×10^{-7}/hr	SNR300
Steam Generator System	Rupture & Plugging	MSGDRØ1F 2 3	5.7×10^{-5}/hr	SAND81-0260
Electric Power	Distribution Faults	MNET	4.2×10^{-6}/hr	SNR-300
Loss of Offsite Power	HCDA	LOSP	1×10^{-3}/d	SNR-300
Non-recovery of LOSP		NRLOSP	0.1/d in 2 hours 0.01/d in 10 hours	SNR-300 SNR-300
1 Diesel	Fail to Start or Run	DGA DGB DGC	1×10^{-2}/d + 3×10^{-3}/hr	SNR-300
2 Diesels	Fail to Start or Run	DGAB DGBC DGCA	1×10^{-3}/d	SNR-300
3 Diesels	Fail to Start or Run	DGABC	1×10^{-4}/d	SNR-300
1 Pony Motor	Fails to Start or Run	PPM01 PPM02 PPM03	3.8×10^{-4}/d + 1.3×10^{-5}/hr	WASH-1400
		IPM01 IPM02 IPM03		
2 Pony Motors	Fail to Start or Run	PPM012 PPM023 PPM031	7.4×10^{-6}/d	WASH-1400
3 Pony Motors	Fail to Start	PPM0123 IPM0123	1.5×10^{-7}/d	WASH-1400

Table B.1

EQUIPMENT AND HUMAN FAILURE DATA (Continued)

Component	Failure Mode	Event Code	Failure Frequency (Mean)	Reference
Main Feedwater System	Fails to Operate	MAINFEED	7.7×10^{-4}	
Main Condenser	Fails to Operate	MAINCOND	1.6×10^{-5}/hr	SAND81-0260
PPS	No Signal to Start AFWS	NSIGAFWS	1×10^{-6}/d	SNR-300
Failure of AFWS	Control Valves	AVALVESØ	2.3×10^{-3}/d	SAND81-0260
Protected Water Storage Tank	Water Unavailable	NOWATERØ	2.7×10^{-8}/hr	SAND81-0260
Turbine driven pump	Fails to Operate	APMTURBØ	5.4×10^{-5}/hr	SAND81-0260
No steam to turbine driven pump		NOSTEAMØ	1.2×10^{-4}	
Motor driven pump	Fails to Start, Fails to Run	APMAØØØA APMBØØØA	1×10^{-3}/d + 5.8×10^{-8}/hr	SNR-300
Operator	Fails to Initiate DHRS	DOPERR	1.1×10^{-3}/d	SAND81-0260
Failure of DHRS Valves		DVALVESØ	2.4×10^{-4}/d	SAND81-0260
No Overflow Vessel	Rupture	DOVFSSLR	1.6×10^{-6}/hr	SAND81-0260
No Overflow Heat Exchangers	Rupture or Plugging	DHXOVFLF	9.1×10^{-7}/hr	SNR-300
No Pump	Fails to Run	DPMNAØ1F SPMNAØ2F	1×10^{-6}/hr	SNR-300

Table B.1

EQUIPMENT AND HUMAN FAILURE DATA (Continued)

Component	Failure Mode	Event Code	Failure Frequency (Mean)	Reference
NaK Pump	Fails to Run	DPMNAK1F DPMNAK2F	1×10^{-6}/hr	SNR-300
Fan	Fails to Start or Run	DBLØØØ1F DLBØØØ2F	2.3×10^{-3}/d + 6.3×10^{-6}/hr	SNR-300
Airblast Heat Exchanger	Rupture or Plugging	DHXØØØ1F DHXØØØ2F	9.1×10^{-7}/hr	SNR-300
Check Valve	Fails Plugged	MCVØØØ1P MCVØØØ2P MCVØØØ3P	8×10^{-7}/hr	SNR-300
PACC Heat Exchanger	Plugged or Rupture	PHXPAC1P PHXPAC2P PHXPAC3P	2.3×10^{-4}/d	SAND81-0260
Motor Operated Isolation	Fails Closed	PMVØØØ1C PMVØØØ2C PMVØØØ3C	1.2×10^{-7}/hr	SNR-300
Venting Valve (Relief	Fails to Open	PRVØØØ1D PRVØØØ2D PRVØØØ3D	6×10^{-3}/d	LER
Pump Trip	Fails to Operate	PT	8×10^{-4}/d	CRBRP-1
Plant Protection System	Fails to Provide Proper SCRAM and Trip Signal	PPS	1.6×10^{-6}/d	CRBRP-1
Emergency Shutdown System	Fails to Operate	SCRAM	2.6×10^{-6}/d	CRBRP-1
Operator	Fails to Start Natural Circulation	OP-RSHRS	10^{-2}/d	SAND81-0260
Recovery	Failure to Recover NAFW or AFW Within 2 Hours	GØ	10^{-1}/d	

Table B.2

PHENOMENOLOGICAL UNCERTAINTIES MEAN ESTIMATES

Description	Event Case	Phenomenological Uncertainty (Mean)	Reference
Fuel is mostly dispersed out of vessel given high energy core disruption	FD	1.0×10^{-4}	Emergency Judgement
Energetic recriticality given a core melt	ER/CM	9.9×10^{-1}	NUREG/CR-0427
Energetic recriticality given an energetic disruption	ER/DC	1.0×10^{-2}	Engineering Judgement
Energetic FCI given non-energetic initial disruption and energetic recriticality	FCI/CM1	1.0×10^{-3}	Engineering Judgement
Energetic FCI given non-energetic initial disruption and no energetic recriticality	FCI/CM2	1.0×10^{-4}	Engineering Judgement
Energetic FCI given energetic initial disruption	FCI/DC	1.0×10^{-2}	Engineering Judgement
No vessel head seal damage given non-energetic or benign initial disruption and energetic recriticality and FCI	VHR11 VHR41	8.0×10^{-1}	Engineering Judgement
No vessel head seal damage given non-energetic or benign initial disruption and energetic recriticality	VHR21 VHR51	9.0×10^{-1}	Engineering Judgement
No vessel head seal damage given non-energetic or benign initial disruption and energetic FCI	VHR31 VHR61	9.0×10^{-1}	Engineering Judgement

Table B.2

PHENOMENOLOGICAL UNCERTAINTIES MEAN ESTIMATES (Continued)

Event Description	Event Case	Phenomenological Uncertainty (Mean)	Reference
Moderate head seal damage given nonenergetic or benign initial disruption andd energetic recriticality and FCI	VHR12 VHR42	2.0×10^{-1}	Engineering Judgement
Moderate head seal damage given nonenergetic or benign initial disruption and energetic recriticality	VHR22 VHR52	1.0×10^{-1}	Engineering Judgement
Moderate head seal damage given nonenergetic or benign and energetic FCI	VHR32 VHR62	1.0×10^{-1}	Engineering Judgement
Large head seal damage given nonenergetic or benign initial disruption and energetic recriticality and FCI	VHR13 VHR43	1.0×10^{-2}	Engineering Judgement
Large head seal damage given nonenergetic or benign initial disruption and energetic recriticality	VHR23 VHR53	1.0×10^{-3}	Engineering Judgement
Large head seal damage given nonenergetic or benign initial disruption and energetic FCI	VHR33 VHR63	1.0×10^{-3}	Engineering Judgement
Moderate head seal damage given moderate initial disruption and energetic recriticality or FCI or both	VHR72 VHR82 VHR92	9.9×10^{-1}	Engineering Judgement
Large head seal damage given moderate initial disruption and energetic recriticality or FCI or both	VHR73 VHR83 VHR93	1.0×10^{-2}	Engineering Judgement

Table B.2

PHENOMENOLGICAL UNCERTAINTIES MEAN ESTIMATES (Continued)

Event Description	Event Case	Phenomenolgical Uncertainty (Mean)	Reference
Retention of the debris in the bottom of the vessel given dispersed fuel (energetic disruption)	VBR	1.0×10^{-4}	Engineering Judgement
Retention of the debris in the bottom of the vessel given fuel is not dispersed in non-coolable geometry	VBR4	1.0×10^{-2}	Engineering Judgement
Coolable debris in the reactor cavity given vessel failure	CD/VF	5.0×10^{-1}	Engineering Judgement