4.0 BACK-END ANALYSIS

4.1 Plant Data and Plant Description

4.1.1 Introduction

This section documents the features of the containment building and safeguards systems which strongly influence the progression of severe accident sequences and their potential consequences. The section also tabulates various critical design parameters of the reactor coolant system (RCS) and emergency core cooling system (ECCS). This information, presented as Table 4–1, includes the containment design parameters and both active and passive systems which maintain core cooling and mitigate pressure/thermal loads that challenge containment integrity.

4.1.2 Containment Configuration/Structural Design

General

The containment building houses the reactor pressure vessel (RPV), the RCS, and supporting systems. The containment boundary forms a low-leakage barrier against the release of radioactivity to the environment and is the last of several barriers which provide defense-in-depth protection from radionuclide release. It is designed to perform this function under the post-accident environmental conditions resulting from a postulated double-ended pipe rupture of a large RCS pipe (LOCA). The containment also provides physical protection to enclosed components and systems from environmental conditions resulting from severe natural phenomena (e.g., tornadoes).

The SHNPP containment building is a steel-lined reinforced concrete shell in the form of a vertical right cylinder with a hemispherical dome and a flat base. The outer structure height measures 160 feet from the base to the springline of the dome. The inside diameter is 130 feet. The containment free volume is approximately 2,100,000 ft^3. The side walls of the cylinder and dome are 4.5 ft and 2.5 ft thick, respectively. The inside radius of the dome is equal to the inside radius of the cylinder, i.e., the discontinuity at the springline due to the change in thickness is on the outer surface. The base consists of a 12 ft thick structural reinforced concrete slab. The base liner is installed on top of the structural slab and covered with an additional five feet of concrete. The reinforced concrete utilizes a quartz based aggregate.

Table 4–1
SHNPP Design Information

Reactor Core	
Nominal Power, MW_{th}	2775
Number of Fuel Assemblies	157
Total UO_2 mass, lb	171,200
Zircaloy mass, lb	40,200
RPV	
Inside Diameter, in.	157.0
Water Capacity w/Core & Internals in place, ft^3	5,590
Reactor Coolant System	
Total Volume inc. Pressurizer, ft^3	9,375
Number of Reactor Coolant Pumps	3
Total Reactor Flow, gpm	292,800
Average Coolant Temp in vessel, ° F	588.8
Operating Pressure, psia	2,250
Design Pressure, psia	2,500
Design Temperature, ° F	650
Number of PORVs	3
Lowest PORV setpoint, psia	2,330
Mass Flow rate per PORV, lb/hr	210,000
Number of safety relief valves (SRVs)	3
Lowest SRV Setpoint, psia	2,475
Mass Flow Rate per SRV, lb/hr	380,000
Steam Generators	
Number of steam generators	3
Heat Transfer Rate, Btu/hr	3168×10^6
Feedwater Temperature, ° F	435
Steam Flow Rate, lb/hr	4.06×10^6
Number of PORVs per steam generator	1
PORV Setpoint, psia	1,120

Table 4–1 (cont.)
SHNPP Design Information

Number of Safety Valves per steam generator	5
Lowest SV Setpoint, psia	1,185
High Pressure Injection System	
Number of Pumps	3
Capacity, gpm @ psia	650 @ 1340 (150 @ 2500)
Shutoff Head, psia	>2700
Refueling Water Storage Tank (minimum), gal	432,000
Low Pressure Injection System (RHR)	
Number of Pumps	2
Capacity, gpm @ psia	3,750 @ 104
Shutoff Head, psia	147
Accumulators	
Number of Accumulators	3
Pressure, psia	680
Water Capacity (Nominal), ft^3	1,012
Auxiliary Feedwater System	
Type Drive	Motor
Number of Pumps	2
Capacity, gpm	450
Type Drive	Turbine
Number of Pumps	1
Capacity, gpm	900
Containment (Large Dry)	
Inside Diameter, ft	130
Maximum Inside Height, ft	220
Free volume, ft^3	2.1×10^6
Design Leak Rate, %volume/day	0.1

Table 4–1 (cont.)
SHNPP Design Information

Design Pressure, psia	60
Operating Pressure, psia	15.7
Operating Temperature, °F	120
Construction	Reinforced Concrete
Wall Thickness, ft	4.5
Dome Thickness, ft	2.5
Basemat Thickness, ft	12
Liner Thickness (Bottom Floor), in.	0.25
Liner Thickness (Cylinder Wall), in.	0.375
Liner Thickness (Dome), in.	0.50
Cavity Floor Area, ft^2	588
Cavity Floor Thickness, ft	12
Cavity Volume, ft^3	8,827
Concrete Type	Quartz-based
Containment Spray System	
Number of Pumps	2
Spray set point, psia	25
Capacity, gpm	1830
Containment Air Cooling System	
Number of Units	4
System Setpoint, psia	18
Flow Rate (normal op), ft^3/minute	125,000
Design Heat Removal(normal op), Btu/minute	2.28×10^6
Flow Rate (post-accident), ft^3/minute	62,500
Design Heat Removal (post-accident), Btu/minute	67×10^6

The containment internal structures consist of the reactor cavity, three steam generator compartments, and a fuel transfer canal located above the reactor cavity. The steam generator compartments house the steam generators, reactor coolant pumps, and the pressurizer. The primary function of the steam generator compartment walls is to serve as secondary shield walls and to resist jet loads due to pipe rupture. The cavity walls are designed to withstand jet forces coincident with the pressure load resulting in the pipe rupture. The reactor cavity is heavily reinforced to support the RPV and the primary shield wall.

The containment system principal performance objectives are to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

The containment boundary is designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions: 1) its ferric materials behave in a non-brittle manner, and 2) the probability of rapidly propagating fracture is minimized. The design reflects consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions and the uncertainties in determining material properties, residual, steady-state and transient stresses, and size of flaws.

It is designed to accommodate, without exceeding the design leakage rate (0.1% volume/day) and with sufficient margin, the calculated pressure and temperature conditions resulting from any design basis LOCA. This margin reflects consideration of:

- The effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded ECCS functioning

- The limited experience and experimental data available for defining accident phenomena and containment responses

- The conservatism of the calculation model and input parameters

The containment is designed to withstand loads from the following:

- Design internal pressure
- Design external pressure
- Design internal temperature (accident)
- Design internal temperature (normal)

Post-LOCA fission product cleanup is accomplished by the containment spray system which sprays water with sodium hydroxide (NaOH) into the containment free volume to scrub iodine from the containment atmosphere.

During power operations the containment atmosphere is maintained by continuous operation of the containment normal ventilation system. Temperature is controlled manually by the control room operator through the use of the containment fan coolers.

Base Slab

The base slab or external foundation mat is a 144 ft. diameter, circular reinforced concrete slab with a minimum thickness of 12 ft. The top of the basemat is at elevation 211'5" and the slab dips down to elevation 207'2" in the cavity region. A 5 ft. thick foundation mat rests on the basemat and serves to support the internal steel and concrete structures. A 1/4 in. liner plate covered on top by 1 in. of ethafoam separates the 5 ft. thick slab from the basemat. No anchorages of the internal structures and internal mat penetrate through the containment liner plate into the external mat. The basemat is supported by solid rock.

Cylinder

The cylinder is 130 ft inside diameter, 160 ft from the top of the base slab to the springline of the dome and 4.5 ft thick. The concrete thickness of the wall is increased to 6.5 ft. around the major penetrations such as the equipment hatch, personnel lock, emergency air lock, main steam penetrations, and feedwater penetrations.

The reinforced concrete cylindrical wall is designed to withstand the loadings and stresses anticipated during the operating life of the plant. Hoop tension in the cylindrical concrete wall is resisted by horizontal reinforcing bars near both the outer and inner surfaces of the wall. Longitudinal tension in the cylindrical wall is resisted by rows of vertical reinforcing bars placed near the interior and exterior faces of the wall.

Asymmetric loads, such as wind, tornado, and seismic excitations, induce tangential shears into the cylindrical concrete wall and concrete dome. Although the liner plate in the cylindrical wall and dome has shear capacity available to resist tangential shear, no credit was taken for this capacity. The excess tangential shear is taken by diagonal seismic reinforcing bars.

Containment Liner

The containment liner is designed to serve as a leak-proof membrane and is not relied upon for the structural integrity of the containment. It is anchored to the concrete by means of Nelson studs 5/8" diameter by 4" long. The liner is not anchored to the concrete base slab hence does not act compositely with it.

In order to minimize liner stresses, strains and deformations under the design loading conditions, the cylindrical wall liner plate connection with the foundation mat lower plate is an unanchored embedded 90 degree free-standing welded connection. In order to allow free deformation of the liner plate during test pressure conditions, an inch of ethafoam is provided on the inside face of the liner plate facing the concrete of the internal mat. Ethafoam is also provided against the back up plate and the end of the horizontal liner plate, to allow vertical movement at the concrete connection during the same test procedure.

The dimensions of liner thickness are as follows:

- 0.375 inch thick on cylinder wall
- 0.50 inch thick on dome wall
- 0.25 inch thick on basemat floor

Dome

The dome is hemispherical, 130 ft. inside diameter, of 2.5 ft. thick reinforced concrete. A continuous welded steel liner plate, one-half inch thick, is provided on the inside face of the dome.

The reinforced concrete dome is designed to withstand the loads anticipated during the operating life of the plant and postulated accidents and events. Meridional and circumferential reinforcing bars are provided to resist the resulting tensile forces and bending moments.

Cavity

The significance of the reactor cavity is related to the determination of water availability and the ultimate disposition of the molten core after vessel breach. The plant-specific details of the cavity configuration were evaluated to determine whether water can accumulate during core damage. The presence (or lack) of water may affect pressure loads at vessel breach and debris coolability in the longer term. The absence of water on the cavity floor combined with the debris remaining on the cavity floor and forming a pool more than a few inches deep may result in concrete attack and the potential for basemat penetration. If water is available on the cavity floor (and replenished continuously), debris cooling may be possible and concrete attack can be mitigated. Prior studies have identified that for high RCS pressure scenarios, involving high-pressure melt ejection (HPME) of the molten material, the debris can be dispersed to the containment atmosphere, producing pressure loads that can threaten containment integrity. This effect is called direct containment heating (DCH). The communication paths betweent he

cavity and the upper compartment are important in causing the potential for DCH. The SHNPP cavity configuration is shown in Figures 4–1 and 4–2.

The cavity is composed of a cylindrical region directly under the RPV and a connected rectangular tunnel for passage of the in-core instrument tubes. The cavity floor is 12 ft thick and covered with a 0.25 inch steel liner and an additional 1.5 ft of concrete (for a total concrete depth of 13.5 ft). The SHNPP cavity is made of a concrete containing a quartz aggregate. The use of quartz, as opposed to limestone which is present in many containment designs, reduces noncondensable gas generation for situations where core-concrete interaction (CCI) is possible. The quartz aggregate behaves more like basalt.

One important consideration related to the cavity design is whether the bottom of the RPV would be covered by water for cases in which the refueling water storage tank (RWST) is discharged prior to RPV failure. Some recent work has postulated that if the RPV were partially submerged, it is possible to cool the debris in the vessel through conduction through the vessel walls to the surrounding water. During scenarios in which the RWST is injected, the water level would be approximately 13.5 ft. above the bottom of the cavity. The height of the vessel bottom head relative to the cavity floor is 15 ft. 5 in. Thus the bottom head would not be submerged as a result of discharging the normal RWST inventory into containment, and ex-vessel cooling is not likely to happen at SHNPP. The SHNPP cavity is surrounded by a 1.5 ft. concrete curb on the containment floor. Thus, only in cases with the RWST injected would there be water present in the cavity prior to RPV failure.

The cavity area below the vessel bottom head can communicate directly with the lower compartment via the instrument tunnel (approximate cross section 186 ft^2). The outer end of the instrument tunnel away from the RPV is not sloped as in some designs (e.g., Zion). This should result in a geometry somewhat less favorable for debris dispersion out of the cavity following vessel failure. For sequences in which debris is dispersed from the cavity tunnel and into the area surrounding the mouth of the cavity (at elevation 221'), there is the possibility that debris can flow (or be expelled) through the drain holes at the bottom of the secondary shield wall, enter the annular compartment, and then directly contact the containment liner. The drain holes are each 2.25 ft square.

Figure 4–1
Cavity Plan and Elevation

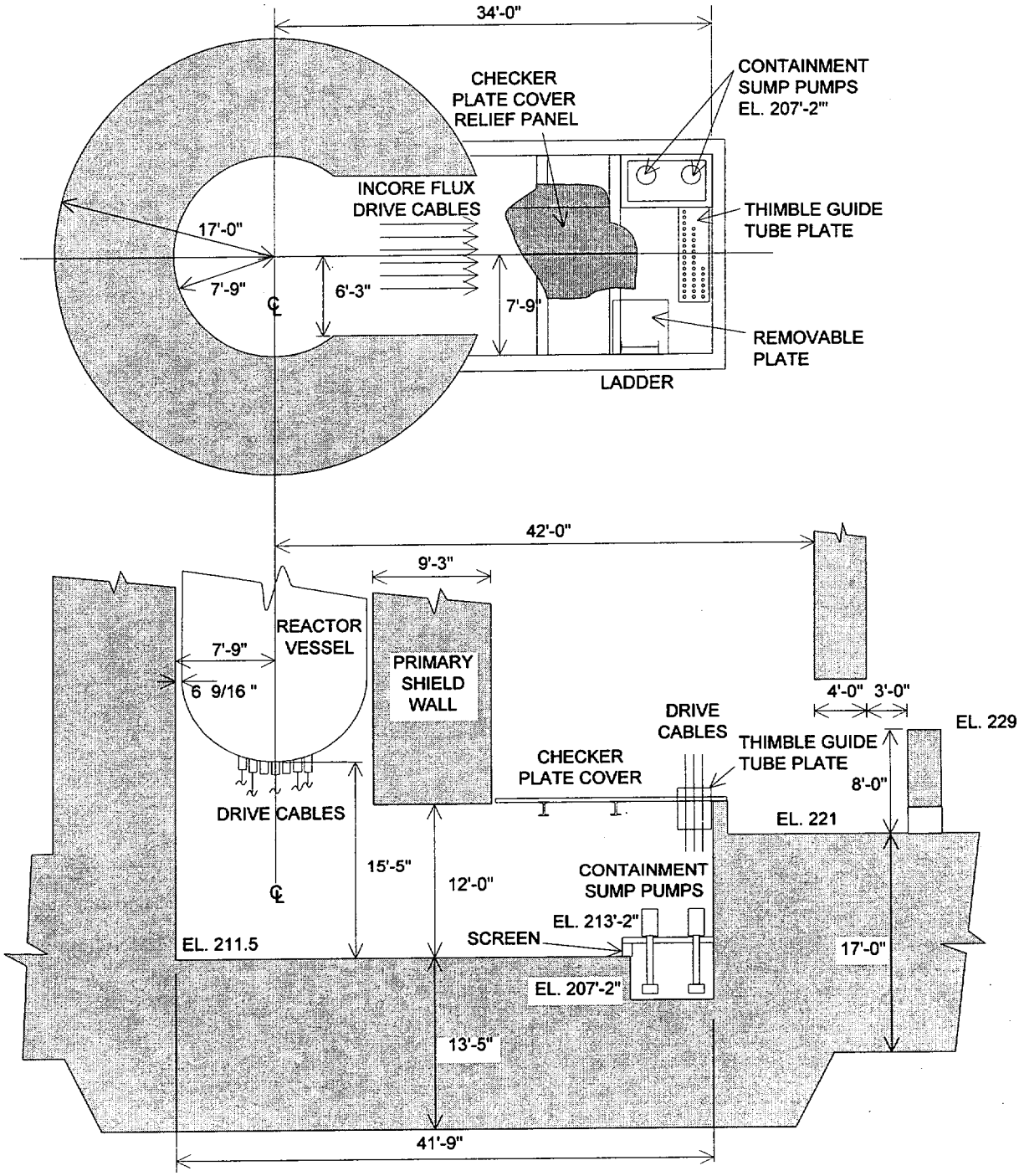
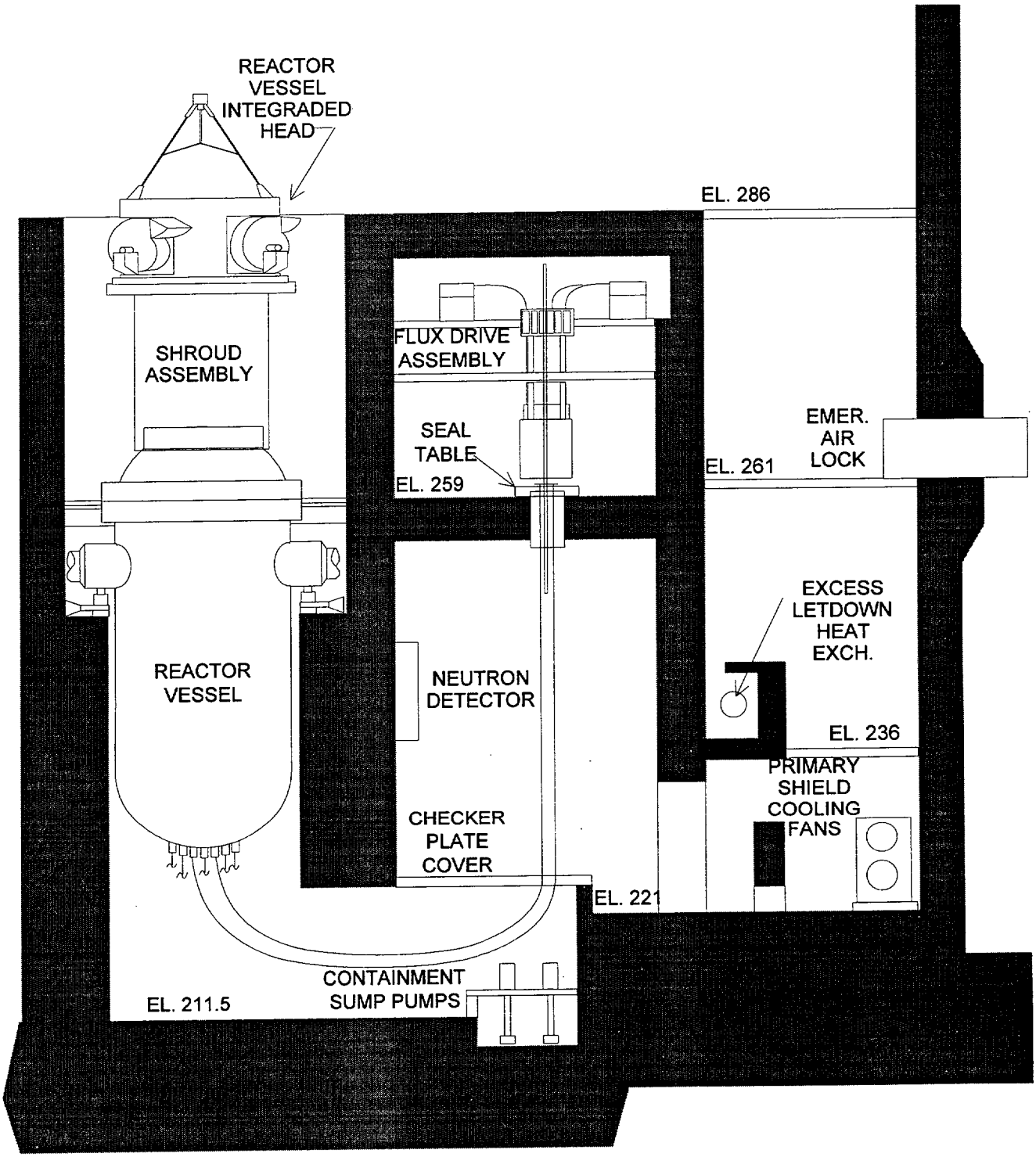


Figure 4–2
Cavity General Configuration



Interior Structures

Within the containment are a number of reinforced concrete walls, all of which perform three major functions:

1. equipment structural support
2. biological shielding, to permit personnel entry during reactor operations
3. missile shielding, to protect the steel liner from puncture by internally generated missiles

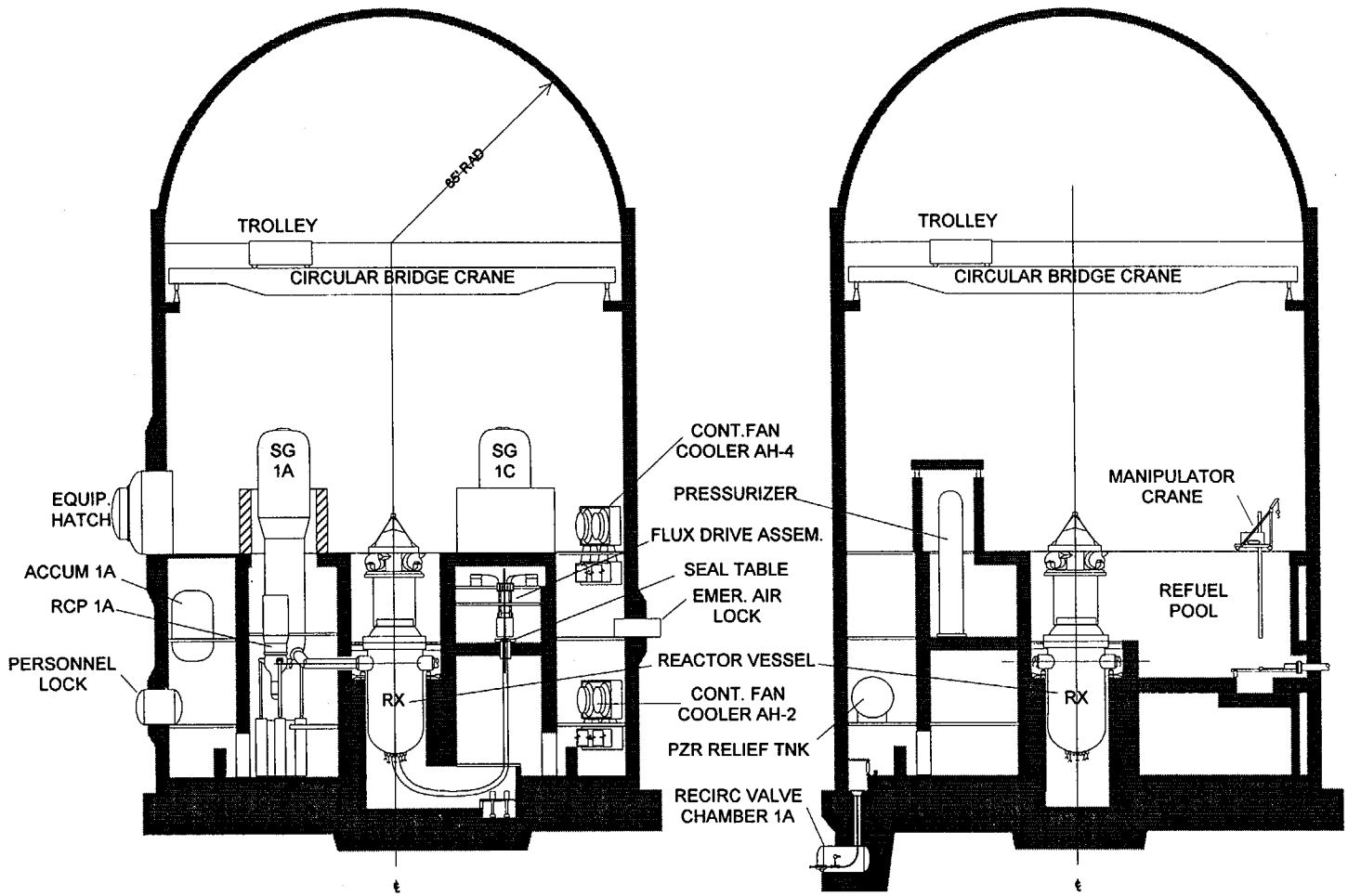
The first wall, 19 ft. inside of the containment outer wall, is the secondary shield wall. This wall is 4 ft. thick and extends from elevation 221' (foundation mat) to the operating floor (elevation 286'). The annulus space between the secondary shield wall and the containment cylinder wall houses cable trays, piping, and ducting as well as major equipment (pressure relief tank, accumulators). The annulus includes several levels of grating that are accessible by ladders. The grating at the operating floor supports the four containment air cooling units. Walls interior to the secondary shield wall are arranged to form equipment support cubicles for major components: three steam generator/reactor coolant pump cubicles and the pressurizer cubicle. Common walls of these cubicles enclose the RPV and form part of the refueling cavity.

The intermediate floors in the containment building are largely grated and good mixing paths appear to exist for all regions of the containment. Thus, the existence of localized high hydrogen concentrations in the containment is considered unlikely.

To shield the containment dome from missiles generated by the failure of a control rod drive mechanism a concrete shield plug with a steel plate on the lower side covers the RPV when the plant is in operation.

The primary shield wall surrounds the RPV and is approximately 9 ft. thick. The underside is open to the cavity to allow passage of the in-core instrumentation. The top of the shield wall has six penetrations for reactor coolant piping to pass through to the Steam Generator/Reactor Coolant Pump cubicles. The general containment configuration is shown in Figure 4–3.

Figure 4–3
General Containment Configuration



NOTE: VIEWS REPRESENT MULTI-ANGLE CROSS SECTIONS

Mechanical and Electrical Penetrations

The reinforced cylindrical concrete wall has penetrations for piping and wiring. There are three general types:

- type I: high pressure, high temperature piping (above 200 ^{o}F)
- type II: general piping (penetrations subject to relatively small pipe rupture forces and temperatures up to 200 ^{o}F)
- type III: electrical

In type I penetrations the process pipe is connected to a containment penetration sleeve (which is partially embedded in the concrete wall) by a forged flued head fitting. The flued head fittings are designed to carry the forces and moments due to the normal operating conditions and due to the postulated pipe rupture loads by transferring these forces to the containment penetration sleeves and further into the concrete containment wall.

In type II penetrations the process pipe passes through a containment penetration sleeve which is partially embedded and anchored into the concrete wall. The annular gap between the process pipe and the sleeve is sealed on both the inside and outside faces of the concrete wall. The inside plate is designed to withstand the internal pressure and to transfer all of the normal loads and/or the postulated accident piping rupture loads from the piping system to the penetration sleeve and then into the concrete wall. The outside seal is flexible to accommodate thermal expansion movements. This type penetrations include HVAC penetrations and groups of small diameter lines (instrument, sampling lines) which incorporate socket weld couplings welded to closure plates.

Modular type penetrations are used for all electrical (type III penetration) conductors passing through the containment wall. Each penetration assembly consists of a stainless steel header plate attached to a carbon steel welded ring which is in turn welded to the pipe sleeve. The header plate accepts either three or six modules depending on the penetration diameter and voltage classification. The modules are held in the header plates by retaining clamps . Each module is a hollow cylinder through which conductors pass. The conductors are hermetically sealed into the module with an epoxy compound. The header plates are attached to penetration sleeves located in the wall of the containment and welded to the containment liner.

Equipment and Personnel Access Hatches

Access into the concrete containment building is provided by an equipment hatch, a personnel air lock, and an emergency air lock.

The equipment hatch is a welded steel assembly having an inside diameter of 24' with a weld-on cover. There is sufficient material to allow for six removals and rewelding. A 15' inside diameter bolted cover is provided in the equipment hatch cover for passage of smaller equipment.

The breech-type personnel air lock has a 9' inside diameter with full diameter breech doors to open outwardly from each end of the lock. Doors for the lock are hydraulically sealed and electrically interlocked. During plant shutdown, it will be necessary to open both doors at the same time: therefore, a key-operated electrical interlock defeat system which is under strict administrative control is provided. Opening of the doors after unsealing will be done with a hydraulic motor, as will closing before sealing. Manual hand pump operation of the sealing ring and door swing mechanism is provided in case of a power failure.

The personnel emergency air lock has an outside diameter of 5' with a 2.5' diameter door located at each end of the lock. The doors of the lock are in series and are mechanically interlocked to ensure that one door cannot be opened until the second door is sealed. Violation of the interlock can only be made by use of special tools and procedures under strict administrative control.

Fuel Transfer Penetration

A fuel transfer penetration is provided to transport fuel assemblies between the refueling canal in the containment and the fuel transfer canal in the fuel handling building. The penetration consists of a 20 inch stainless steel pipe installed inside a 26 inch pipe. The inner pipe acts as the transfer tube and is fitted with a pressurized double-gasketed blind flange in the refueling canal and a standard gate valve in the fuel transfer canal. This arrangement prevents leakage through the transfer tube in the event of an accident. The penetration sleeve is welded to the refueling canal steel liner and anchored to the concrete wall. Bellows expansion joints are provided on the pipes to compensate for any differential movement between the two pipes or other structures.

Containment Safeguard Systems

Active containment heat removal is accomplished by the containment spray system (only during injection phase) and the containment fan coolers. These systems are capable of maintaining and reducing containment temperature and pressure within design limits following a LOCA or a main steam line break occurring inside containment.

The operation of the containment is supported by a variety of containment-related auxiliary systems that perform the following functions:

- containment isolation
- containment heat removal
- containment air purification and clean-up
- combustible gas control

The containment isolation system (CI) automatically isolates non-essential lines penetrating containment if containment pressure increases to the set point or if power is lost to the containment isolation actuation system logic. This system, along with the containment heat removal systems (containment spray and containment fan coolers), are described in Section 3.2.

4.2 Plant Models and Methods for Physical Processes

4.2.1 Modifications To MAAP/PWR Revision 17.02 For Use in SHNPP Level 2 Calculations

The MAAP code was in a stable mode at the time the back-end analysis commenced. The most recent code version available, MAAP/PWR revision 19.00, had corrected all identified errors in revision 18.00 and included few new models. Generally, the code performed well for all the sequences studied in the PRA. The only exception was due to an error is subroutine SPRAY (reported in MAAP RAAP 1/2/93, Trouble Report #262) that yielded physically impossible results in some sequences. The problem was solved by using the corresponding subroutine from revision 18.00 of the code. All affected sequences were re-run with the corrected code.

4.2.2 MAAP Parameter File

The MAAP code was the principal tool used to analyze postulated severe accidents at SHNPP. The plant specific characteristics enter the MAAP code via its parameter file.

There are two important aspects to the creation of this file. The first is calculating and documenting the specific parameters. This step included independent review to assure parameter descriptions were correctly interpreted, and assumptions made were reasonable, so that the models would correctly model plant response. The second consists of deciding what values to use for the variable parameters, most of which are in the model parameter section. Variable parameters are used to deal with the uncertainties in the underlying physical processes. The following two subsections discuss these aspects.

Parameter File Development

The MAAP parameter file for SHNPP was developed over a period of several months. The starting point was a sample parameter file based on the Zion plant, which is a PWR with a large, dry containment building. A data sheet showing calculations, any assumptions, and references used, was generated for each parameter. The MAAP User's Guide and the code manuals were used for the interpretation of most parameters. Additional guidance and expertise was obtained from Marc Kenton and his staff at Gabor, Kenton & Associates (GKA). GKA also reviewed the complete parameter file to determine if the correct interpretation of the parameter definition had been used and if the values obtained were reasonable based on their past experience.

For cases where an appropriate plant specific value could not be found after considerable effort, and models affected were not particularly sensitive to the parameter in question, the MAAP default values (those used in the sample file) were used.

Once most of the parameters related to the RCS had been input, a steady state run was performed. The results of this run compared favorably with plant data. MAAP results for success criteria development were compared to similar studies done earlier with the RELAP–5 code, and similar results were obtained. Further fine tuning and confirmation of code resulted from critical inspection of results generated by the code. Noted deviations from expected results led to further refinements of the parameter file.

Another aspect of the parameter file was deciding on a set of nominal model parameter values to be used in base case calculations. It is important to understand the significance of the key phenomenological assumptions involved in this decision.

Phenomenological Assumptions Used In MAAP Calculations

Deterministic calculations of severe accident progression must necessarily deal with the many uncertainties in the underlying physical processes. In the case of MAAP calculations, the treatment of physical phenomena is controlled by the values which are input for so-called "model" parameters.

Two general types of MAAP calculations were made in the SHNPP PRA. "Base-case" calculations utilized a set of nominal model parameter values. These values were selected to be consistent with those recommended in the EPRI Guidance Document (Ref 4–1). Base-case calculations were performed for each of the important plant damage states (PDSs) to provide insights into the timing and likely primary system and containment response. Guided in part by the base-case results, sensitivity calculations were also performed with MAAP. Along with hand calculations and the use of other computer codes, this latter set of runs were used to assist in assessing the impact of uncertainties in

the phenomena had on the accident progression and the fission product source terms. Such sensitivity calculations thus provided important inputs to the process of quantifying the containment event tree (CET) end states. Usually, sensitivity calculations involved varying one of the key model parameters from its nominal value, leaving the other parameters fixed. For example, in the base case calculations, a relatively small fraction of the debris was assumed to participate in DCH. In several sensitivity calculations, this assumption was changed to a more pessimistic view, which allowed for larger fractions of the debris to participate.

The important sensitivity calculations used in quantifying the CET are discussed in section 4.2.3. The key phenomenological assumptions made in the base-case calculations are as follows:

1. Water lying on top of debris in containment was assumed to ingress into the debris, i.e., the upper debris crust was assumed to crack and thus not prevent water from penetrating. Parameters input to the critical heat flux model were adjusted so that a heat flux of about 83.7 kW/m^2 were obtained at atmospheric pressure from unquenched debris.

2. The vessel was assumed to fail soon (nominally 1 minute) after debris relocation to the lower head. This is consistent with the typical MAAP calculation of large debris relocation rates to the lower head. No credit was taken for cooling through the lower head. The failure was assumed to initiate at an in-core instrument tube penetration, and ablation of this hole was calculated.

3. The median containment failure pressure was estimated to be 150 psig based on the analyses summarized in section 4.4. A relatively large failure area (0.5 m^2) was assumed to develop at that time.

4. Most calculations involving sequences with very high primary system pressure were performed two ways, both with and without an induced rupture of the hot leg. This was done since the occurrence of an induced rupture will have a large effect on the subsequent accident progression, and thus, in the team members' judgment, cannot be dealt with adequately with one or two sensitivity calculations. Based on a review of both SHNPP-specific MAAP calculations as well as studies with more detailed codes, it was felt that the case for induced rupture, while strong, was not overwhelming.

5. For sequences in which the primary system pressure at vessel failure remained sufficiently high to result in high pressure melt ejection of the debris, a relatively small fraction (3 percent) of the dispersed debris was

assumed to fragment finely and so participate in direct containment heating.

6. The so-called MAAP PWR "core blockage" model, which assumes that melting in a region of the core will quickly terminate hydrogen production in that region, was not credited. As a result, calculated hydrogen source terms were generally consistent with the NUREG–1150 median values (see section 4.2.3).

7. All fission product tellurium was assumed to chemically bind with unoxidized zirconium. As such, it was released from the debris only if the zirconium was oxidized in subsequent ex-vessel concrete attack.

As noted above, these assumptions are the same as those documented in the EPRI Guidance Document (Ref 4–1).

4.2.3 Treatment of Phenomenological Uncertainties in the Quantification of the CET

The construction and quantification of the CET generates the overall results for the level 2 PRA. To perform these tasks, it is necessary to define the important phenomenological events that drive the containment performance and to assign probabilities ("split fractions") to the occurrence of the various events. For many of the events, there are substantial uncertainties, either in whether the event would occur or in how the containment would respond if it did. To provide guidance on the importance of these uncertainties, sensitivity calculations were performed, mainly using the MAAP code.

This section discusses the key areas of uncertainty in the response of the SHNPP primary system and containment to a severe accident, the calculations that were performed to assess the importance of these uncertainties, and the conclusions that were drawn from these calculations.

The general areas to be addressed are:

a. The magnitude of the expected hydrogen source term and the threat to containment integrity posed by hydrogen detonations and deflagrations.

b. The likelihood that vessel failure could be prevented by flooding the reactor cavity and cooling the molten debris in the lower head of the RPV (RPV).

c. Events which could affect the RCS pressure at the time of vessel breach, notably temperature-induced rupture of the hot leg, surge line, or steam generator tubes. This is important because the RCS pressure at vessel

breach has a large impact on the containment loading and thus on the probability that the containment will survive the immediate post-vessel breach period.

d. For similar reasons, the mode (size and timing) of vessel breach, which can have an important effect on containment loads.

e. HPME, direct containment heating (DCH), and the potential for debris/liner contact in sequences in which the RPV fails at high pressure.

f. The ability to cool core debris in the containment, whether it is relocated to the floor of the containment by the blowdown of the RPV or remains in the reactor cavity.

g. The magnitude of fission product revaporization from the RCS. This can contribute significantly to the release of fission products from the containment, especially if the containment fails long after vessel failure.

The selection of the key uncertainties that should be considered, and the specific ways in which they were treated, was based on several sources. First, surveys of severe accident phenomenology were reviewed. This review included consideration of the recommendations in NUREG–1335 (Ref 4–25): Table A.5 from the NUREG (Table 1) indicates key areas where sensitivity studies have been most important in the past. Also consulted were the NUREG–1150 (Ref 4–2) analyses of Zion and Surry together with their supporting documentation (Ref 4–41 and 4–44), the EPRI "Guidance Document" on performing IPEs with MAAP (Ref 4–46), and a recent review of severe accident phenomenology published by the Swedish RAMA project (Ref 4–4). Next, to relate the findings of the literature to the specific features of SHNPP, a set of integrated thermal hydraulic/fission product calculations were performed with the MAAP 3.0B code (Ref 4–47). These allowed the sensitivity of the containment response to the various issues to be directly assessed. Some hand calculations were also performed.

Selection of MAAP Sensitivity Calculations

Based on the literature review and the consideration of plant-specific features, the phenomena shown in Table 4–2 were found to be of particular importance for SHNPP. All of these phenomena were included in the CET or the associated fault trees except when a plant-specific evaluation ruled out consideration (e.g. cooling of debris in the lower head). It should be noted that these phenomena include all those identified in NUREG–1335.

Table 4–2
Phenomenological Issues Studied

Phenomenological Issue	Uncertainty	Method-of-Treatment
Hydrogen Production and Combustion	Magnitude of hydrogen production during core degradation	MAAP sensitivity calculations (eutectic temperature, effects of core blockage, simple changes to core melt progression model)
	Magnitude of hydrogen production during DCH	MAAP sensitivity calculations (debris fragmentation and entrainment rate)
	Mode of hydrogen burning, i.e. the possibility of local burns	Conservatively neglected based on low calculated cavity gas temperatures
	Magnitude of hydrogen production during CCI	MAAP sensitivity calculations (debris coolability and debris distribution in containment)
	Potential for detonations	Review of containment configuration; hand calculations
	Possibility of cooling in lower head	Neglected based on calculated cavity water level
Timing and Mode of Lower Head Failure	Size (area) of vessel breach	Literature survey, MAAP calculations of hole ablation
	Time required to fail vessel	MAAP sensitivity calculations
Performance of Containment Heat Removal Systems During Severe Accidents	Failure of containment fan coolers due to harsh conditions in the containment	Literature survey
In- and Ex-Vessel Steam Explosions	Possibility of induced containment failure	Literature survey

Table 4–2 (cont.)
Phenomenological Issues Studied

Phenomenological Issue	Uncertainty	Method-of-Treatment
Debris Entrainment from Cavity	Pressure in RCS at vessel breach	MAAP sensitivity calculations (time of vessel failure, possibility of induced ruputres, effect of cooldown)
	Area of vessel breach	Literature survey, MAAP calculations of ablation
	Possibility of contact of debris with containment liner and failure of liner	Stand-alone computer calculations
Debris Coolability in Containment	Whether debris in cavity and on containment floor can be cooled	Sensitivity calculations (MAAP parameter FCHF), hand calculations
Revaporization of fission products from the RCS	Effect of chemical reactions between cesium and steel surfaces	MAAP sensitivity calculations (vapor pressure multiplier)

To support the quantification of the CET, these phenomena were studied in several ways. In a small number of cases, namely in-vessel and ex-vessel steam explosions and degradation of containment heat removal systems during severe accidents, the phenomena were quantified in a manner consistent with the severe accident literature. This was done when it was judged that the treatment found in the literature was applicable to SHNPP, when it appeared that the phenomena would not have a major influence on the containment response, or when it was decided that the discussion in the literature could not be significantly improved upon within the scope of the IPE.

For the other issues, it was found both useful and necessary to perform plant-specific calculations. As mentioned previously, for most of the phenomena one or more MAAP sensitivity calculations were performed. In this regard, the selection of sensitivity calculations meets or exceeds the recommendations in the EPRI Guidance Document. In several areas, it was found more appropriate to conduct hand calculations.

A list of the MAAP sensitivity calculations is shown in Table 4–3; not shown are the base case calculations which were performed for each of the dominant PDSs and which are described elsewhere in this report. The results of the key sensitivity calculations are

described briefly in the following sections, organized by the phenomena being studied and arranged roughly in chronological order from the standpoint of accident progression.

Table 4–3
Sensitivity Calculations Made with MAAP

1. Station Blackout (SBO) Cases

   a. SBO00 – Base Case
   b. SBO01 – Induced hot leg rupture
   c. SBO02 – Delayed RPV head failure (1/2 hour)
   d. SBO04 – Recover at 10 hours
   e. SBO05, 05A – Increased fuel melting temperature
   f. SBO00E – Blockage model used
   g. SBO06A – Reduced revaporization rate
   h. SBO03, 00A, 00B, 00C, 00D – Various DCH cases

2. Small Break LOCA (SBLOCA) Cases

   a. SLOCA00 – Base case 2" LOCA with failure of injection
   b. SLOCA01 – No entrainment
   c. SLOCA02 – No entrainment, low debris coolability
   d. SLOCA03 – Delayed RPV head failure
   e. SLOCA03A – Delayed vessel failure with steam generator cooldown
   f. SLOCA00A – Increased fuel melting temperature
   g. SLOCA00B, SLOCA00C – Effect of sequence definition on hydrogen generation

Hydrogen-related Phenomena

A. Hydrogen Source Terms

In a severe accident, the major source of hydrogen is oxidation of zirconium and steel by steam within the original core boundary, during HPME from the reactor cavity, and during CCIs. Each of these sources will be discussed briefly below.

(1) In-core oxidation

Base case calculations of core melt progression in the SHNPP PRA were performed with the MAAP "blockage" model turned off as recommended in reference 4–46; in practice, this means that the analyst is taking relatively little credit for the effects of geometry degradation or zirconium relocation on hydrogen production, and the results obtained have historically corresponded fairly well to results from more detailed NRC codes such as MELCOR and MELPROG. The calculations performed with MAAP for SHNPP generally support this conclusion, and the following lists the hydrogen source terms that were computed (results are

expressed in terms of the fraction of the total in-core Zircalloy mass that is oxidized). Also shown in parenthesis are the corresponding median hydrogen source terms (where available) quoted in the Surry NUREG–1150 expert· ellicitations (Ref 4–41).

- Base case station blackout (SBO): 34 percent (44 percent)
- SBO with a large induced rupture of the hot leg: 37 percent (50 percent)
- SBO with increased eutectic melting temperature: 42 percent
- Small LOCA (2") with failure of injection: 20 percent (48 percent)
- Small LOCA (2") with failure of recirculation: 27 percent
- Small LOCA (4"): 27 percent

In addition, one SBO calculation was made in which induced rupture did not occur and the MAAP "blockage" model was activated; this resulted in 17 percent clad reacted, which should be considered as a reasonable estimate for the lower bound of hydrogen production.

As can be seen, the MAAP calculations, while generally lower, are reasonably consistent with those estimated in NUREG–1150 when the blockage model is not employed. The MAAP values quoted are the hydrogen mass generated at the time of vessel failure; in many cases, a considerable additional mass was generated from fuel remaining in the RPV after vessel failure. The biggest discrepancy is seen in the small LOCA calculations and indeed the MAAP-calculated hydrogen production values are smaller than usual. Since small LOCA core degradation calculations are especially complicated by intermittent accumulator discharge into an overheated core, the results of such sequences are expected to vary and this particular finding is not judged to be of concern. The hydrogen source terms developed in NUREG–1150 were also considered during CET quantification, and, in any event, the SHNPP containment is not vulnerable to even very large hydrogen source terms as discussed later.

(2) High Pressure Melt Ejection

The SHNPP cavity configuration is qualitatively similar to that of Surry, in that the wall that defines the end of the cavity farthest from the RPV meets the floor at a right angle. One noteworthy difference is that the total upward flow area out of the cavity at that end is quite large compared to Surry if we assume that the metal

cover (through which the instrument tubes penetrate) is displaced by the RPV blowdown.

If the RCS should fail at high pressure, e.g. in excess of a few hundred psi, the results of hand calculations and limited simulant fluid experiments in the Surry configuration (Ref 4–24) indicate that gas velocities will be sufficiently high to disperse the debris upwards through the cavity cover. The degree to which the debris could be oxidized by the steam and air present in the containment or released from the RCS during this process is a function of several factors:

(a) the extent to which the debris is fragmented into small particles

The debris could be dispersed as a film "flooded" up the cavity walls if fragmentation does not occur quickly enough (Ref 4–28). In such a case, one would expect relatively inefficient heat transfer between the metals in the debris and the gas exiting the RPV, and only limited zirconium oxidation. Conversely, if the relative velocity between the gas and the debris is high enough to cause droplets to be rapidly stripped from the film, perhaps the entire melt could be converted into small particles before the film is expelled. Typical Weber number criteria predict droplet sizes on the order of a few mm, in agreement with experimental data (Ref 4–48), so debris droplets (to the extent that they are formed) can be expected to oxidize if sufficient steam and oxygen are available. Note that oxidation of the debris by oxygen will release energy but will not produce hydrogen; oxidation by steam produces both.

(b) whether the debris is confined to a small volume outside the cavity

An extensive set of integral DCH experiments (Ref 4–49) as well as the Limited Flight Path Test Series at Sandia (Ref 4–50) indicate that debris expelled from the reactor cavity through the in-core instrument tunnel would not be expected to be dispersed beyond the confines of the lower compartment volume immediately adjacent to the area where the in-core instrument tubes leave the cavity. At SHNPP, this implies that the debris should not pass beyond the concrete floors at the 259' and 286' elevations. This would limit the time available for interactions and would also prevent utilization of the bulk of the steam already present in the containment for oxidizing the debris and receiving energy from the debris.

Some debris could also be expelled from the reactor cavity through the annulus around the RPV. Based on very limited test data (Ref 4–51), the fraction of the debris which would flow upwards in this fashion rather than horizontally through the instrument tunnel should correspond roughly to the flow area through the annulus relative to the total. Since there is a much larger flow area associated with the instrument tunnel, upward dispersal should be restricted to a relatively

small mass of material and is not expected to make a major impact on the accident progression.

(c) the availability of steam from the RPV blowdown to react with the debris

Debris entrainment could occur over a short period (e.g. a few seconds or less). This is suggested by simple scoping models of debris dispersal (Ref 4–48). In this case, insufficient steam may be discharged from the RCS to oxidize a large fraction of the zirconium and steel in the melt (unless the RPV failure size is very large). For example, if the initial RPV breach size is relatively small, it will ablate to a size on the order of 1 foot in diameter (Ref 4–17) by the time the debris has been expelled from the RPV; in one second, such a hole will release enough steam to produce at most about 300 lb_m of hydrogen. More steam would be released in this time period if the vessel breach size is larger, but in this case the debris entrainment time would probably also be shorter (Ref 4–28).

MAAP contains a relatively simple model for high pressure melt ejection that allows the user to explore these issues in a simplified way. Some sensitivity analyses were conducted with this model as described in Section 4.6.

(3) Core-concrete interactions

Finally, hydrogen will be produced if sustained CCIs take place after vessel failure. This occurs since degradation of the concrete aggregate causes steam to be released into the debris pool where it interacts with unoxidized steel and zirconium. The rate of hydrogen generation by this process is believed to be calculated fairly accurately by MAAP (Ref 4–52), and the production rate in the long term is limited primarily by the overall rate of attack (dependent mainly on the decay heat directed into the concrete and the chemical composition of the aggregate) and the density of reinforcing steel in the concrete. It was judged that this process is not subject to uncertainties that need to be investigated as part of the PRA. However, as discussed below, uncertainties in the cooling of the debris by an overlying water pool were investigated.

B. Modes of Hydrogen Combustion

It is convenient to discuss separately the various ways in which hydrogen can burn in the containment. In principle, the combustion of carbon monoxide would also be addressed; since the SHNPP concrete aggregate is composed of igneous rock, however, very little carbon monoxide is expected to be produced in severe accidents that proceed past the point of RPV failure. Burning of hydrogen can be either global (consuming a large mass in a short period of time) or local (in which hydrogen is burned in a more-or-less continuous fashion).

In accident calculations in which the cavity dries out, relatively low temperatures are predicted in the SHNPP reactor cavity. For example, in low pressure sequences (i.e. no debris dispersal), typical cavity gas temperatures are about 500°F.

The primary reason for this is that vigorous natural circulation is predicted to occur through the instrument tunnel and up the RPV annulus, driven by the density difference between the hot concrete decomposition gasses and the colder gasses in the rest of containment. Natural circulation is promoted in SHNPP by the relatively large cavity flow areas available. This implies that ample quantities of oxygen would be available to react with hydrogen in the reactor cavity if burns occur. The MAAP model allows for continuous "recombination" of hydrogen and oxygen only if the bulk temperature of the cavity exceeds a user-input auto-ignition temperature (default value 1310°F). This was not the case at SHNPP, so auto-ignition was not credited. In reality, temperatures near the debris surface would probably be sufficient to promote recombination. This would prevent the long term buildup of combustible gasses caused by CCI. Neglect of recombination in the PRA represents a substantial conservatism especially when considering long term recovery from SBO sequences (see below).

(1) Detonations

Based on previous analyses (Ref 4–53, 4–54), it is not expected that direct initiation of detonations is very likely, given the substantial amount of energy required to initiate such a detonation, especially in the presence of steam. Of somewhat greater concern is the possibility of deflagration-to-detonation transitions (DDT). A detailed assessment of the likelihood of such events has not been performed for SHNPP and is considered beyond the scope of the PRA. However, the potential for high local hydrogen concentrations to build up in the SHNPP containment has been assessed semi-quantitatively.

Hydrogen can be introduced into the containment through the quench tank, i.e. because of flow through the pressurizer power-operated relief valves (PORVs) and safety valves, from a primary system LOCA, from core-concrete attack occurring in the containment, or via a breach in the bottom of the RPV. The last of these is not normally of concern from the standpoint of causing local hydrogen concentrations since vessel failure is accompanied by rapid discharge of steam for all cases in which the primary system fails at pressure; if the RCS pressure is low, there will be little hydrogen release at the time of RPV failure. We will treat each of the other sources in turn.

The quench tank is located outside the secondary shield wall. It sits in a well ventilated area which should not be prone to the buildup of high hydrogen concentrations. Hydrogen released into this area in transient-initiated sequences would be expected to be swept into the lower compartment by a global natural circulation pattern which is expected to develop (see below). In addition, for any sequence in which the rupture disk

has failed, the water remaining in the quench tank will be saturated. Flashing of the water will dilute the hydrogen so that the issuance of a mixture with very high hydrogen concentration is not expected.

RCS LOCAs or core-concrete attack occurring in the lower compartment would also result in the introduction of hydrogen into the areas adjacent to the steam generators. Mixing of these areas with the rest of containment is expected to be efficient, primarily because there is ample flow area to couple this region to the rest of containment. The limiting flow area for global natural circulation around the SHNPP containment is the area which couples the steam generator compartments to the upper containment. This is composed of the annular space around the steam generators and pressurizer and a vent area over each reactor coolant pump and is quite large in SHNPP (~800 ft^2).

The simple hand calculation presented in reference 4–55 indicates that the time constant for containment mixing at SHNPP is on the order of 10 minutes when the only driving force for fluid motion is the convective heat losses expected from the reactor coolant system. Under actual accident conditions, prior to vessel failure the issuance of steam from RCS breaks would enhance mixing further (Ref 4–54). After vessel failure, convective heating of the containment atmosphere from the debris (Ref 4–54) or steaming of the debris, depending on the debris location and the availability of water, would contribute further to mixing.

The conclusion that the SHNPP containment should be well mixed is borne out by MAAP calculations. Figures 4–4 and 4–5 show hydrogen concentrations in the various MAAP containment nodes for a high pressure SBO and a small LOCA sequence, respectively. While it is understood that a coarsely nodalized, lumped parameter code such as MAAP tends to overpredict mixing, these results together with the qualitative assessment lead to the conclusion that hydrogen pocketing is not expected.

Given that mixing will be efficient, it is relatively easy to qualitatively bound the potential for detonations. If we conservatively assume that all the steam is condensed, the achievement of a 14 percent average hydrogen concentration (more-or-less a practical lower bound for detonations to occur (Ref 4–13) requires the reaction of an equivalent of approximately 96 percent of the Zircalloy in the core (Ref 4–14) and the release of all this hydrogen to the containment before the occurrence of a burn. A larger amount would likely be necessary in reality, in view of the fact that steam concentrations will always be nonzero and considering that some hydrogen will remain in the RCS prior to vessel failure. Since this much hydrogen will not be produced in-vessel, the conditions for a detonation would arise only if core-concrete attack proceeds in the presence of containment heat removal and deflagrations are somehow prevented until a high hydrogen concentration is reached. This is judged to be very unlikely.

Figure 4–4
Containment Hydrogen Concentrations for SBO Sequence

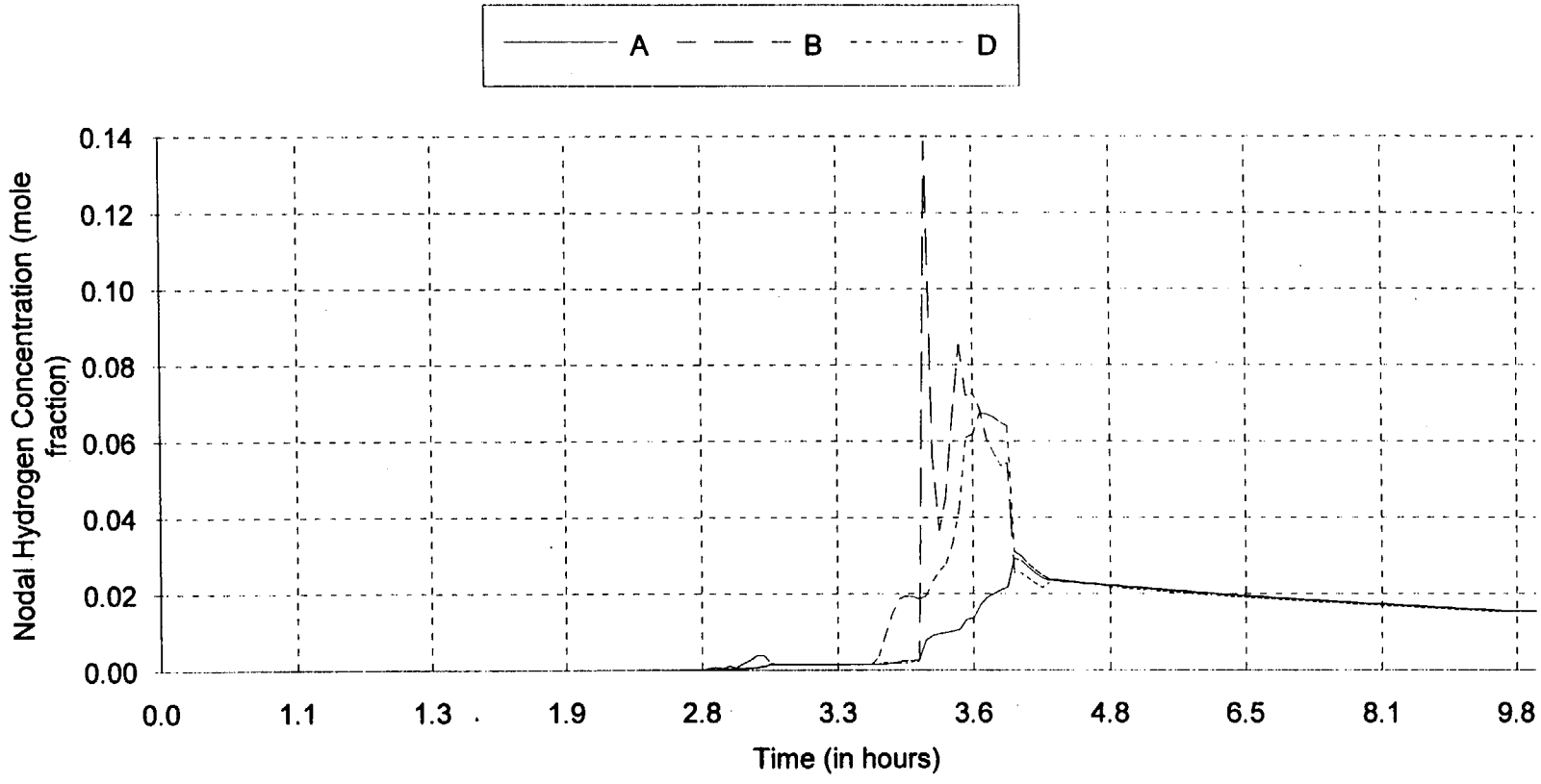
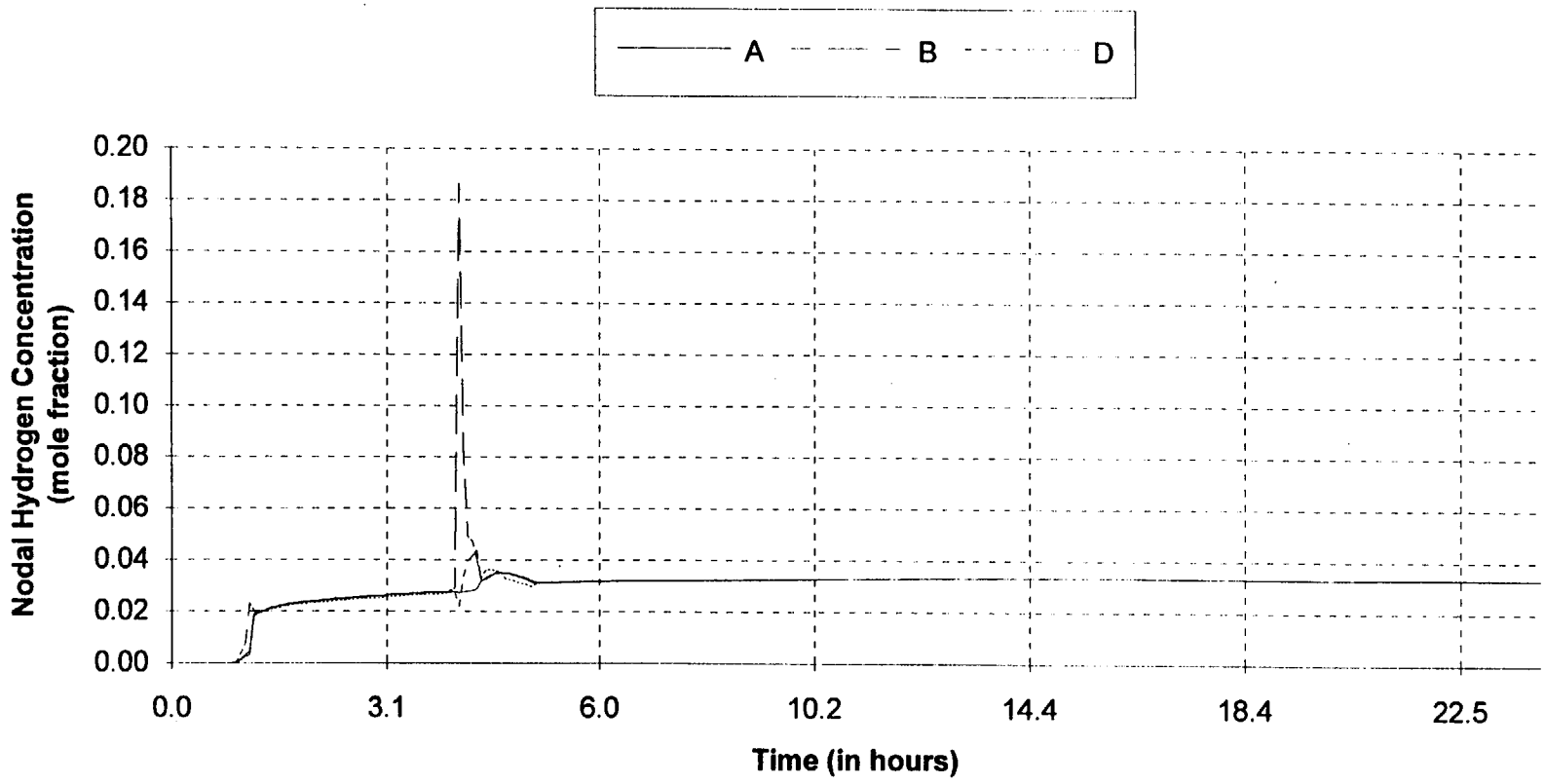


Figure 4–5
Containment Hydrogen Concentrations for Small LOCA Sequence



Thus the potential for hydrogen detonations is qualitatively assessed to be very small, and as in the NUREG 1150 analyses for large, dry containments, this is not expected to be a significant contributor to risk at SHNPP.

(2) Hydrogen Deflagrations

Absent detonations and localized burning sufficient to prevent the buildup of combustible gasses, hydrogen deflagrations present a potential threat to containment integrity. As in other large, dry containments, this threat is greatly reduced by the usual preponderance of steam in the containment for sequences in which no containment heat removal exists. For example, in MAAP calculations performed for small LOCA and SBO initiators with no containment heat removal, the steam mole fraction in the containment was above the 55 percent steam inerting limit before the core even became uncovered. This is illustrated in Figure 4–6 and 4–7 for both an SBO and a small LOCA sequence with failure of containment heat removal.

Stated another way, a containment pressure in excess of 3 atmospheres in a large, dry containment essentially guarantees steam inerting for any sequence in which the containment gas temperature is not extremely high (Ref 4–53). Such values for the containment pressures are a necessary consequence of discharging the mass of water necessary for uncovering the core in the first place. Thus, for deflagrations, we are concerned with sequences in which containment heat removal is or has been operational, and especially those sequences in which containment heat removal is recovered after an extended period of hydrogen production.

Sequences with containment heat removal sufficient to prevent steam inerting will, as stated above, be at pressures less than about 45 psia. For all but transient initiators with no subsequent LOCAs, hydrogen will be introduced gradually into the containment through the break(s) in the primary system. Since containment heat removal presupposes availability of AC power, it is expected that ignition sources will also exist in the containment during this period, and the probability that very high hydrogen concentrations will exist before a deflagration occurs is considered quite small. In support of this conclusion, Figure 4–8 indicates that initiation of burns takes a monotonically lower energy deposition after hydrogen concentrations have increased past the value required for globally combustible conditions.

Figure 4–6
Steam Concentrations in High Pressure SBO Sequence

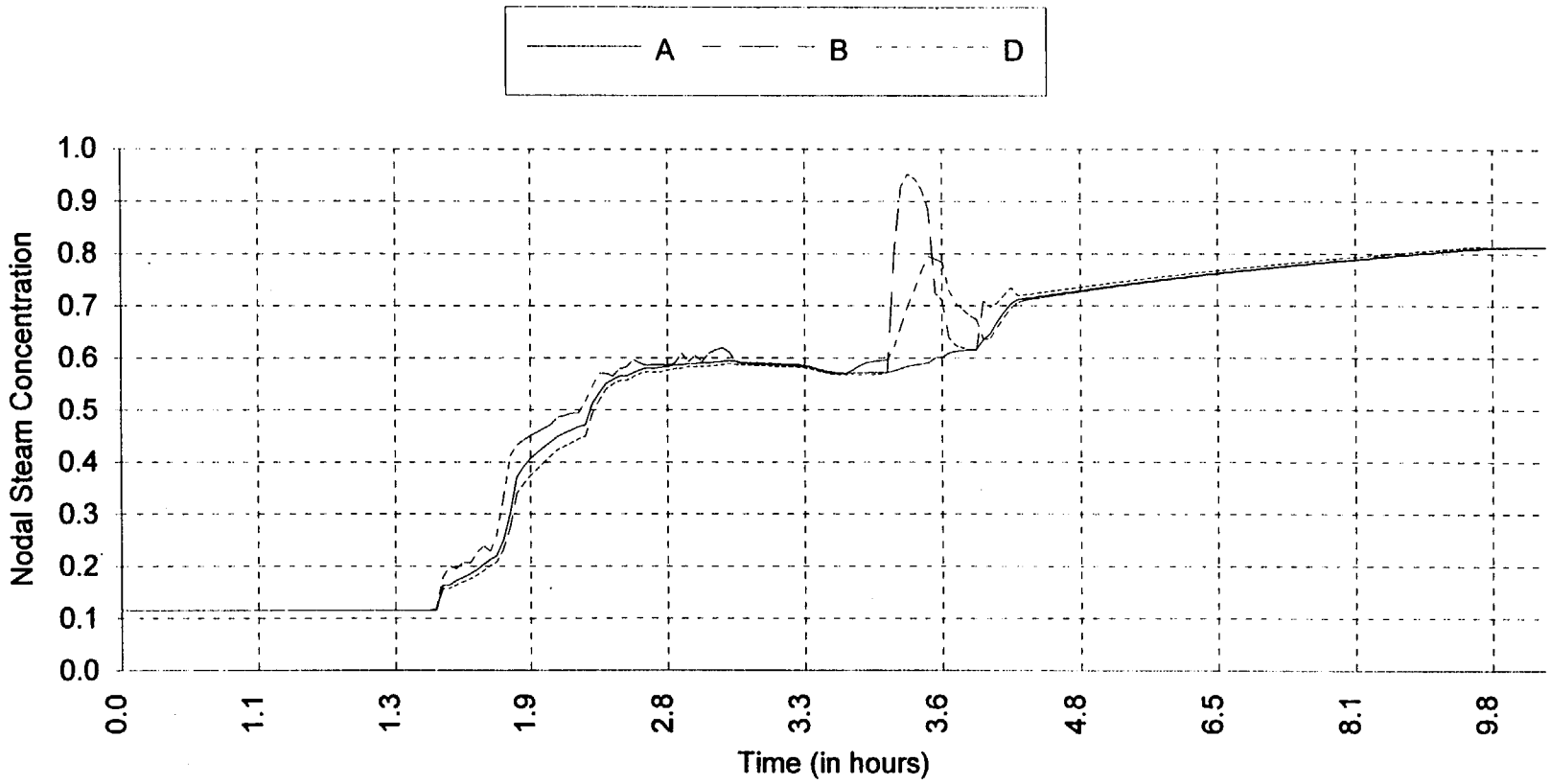


Figure 4–7
Steam Concentrations in Small LOCA Sequences without Containment Heat Removal

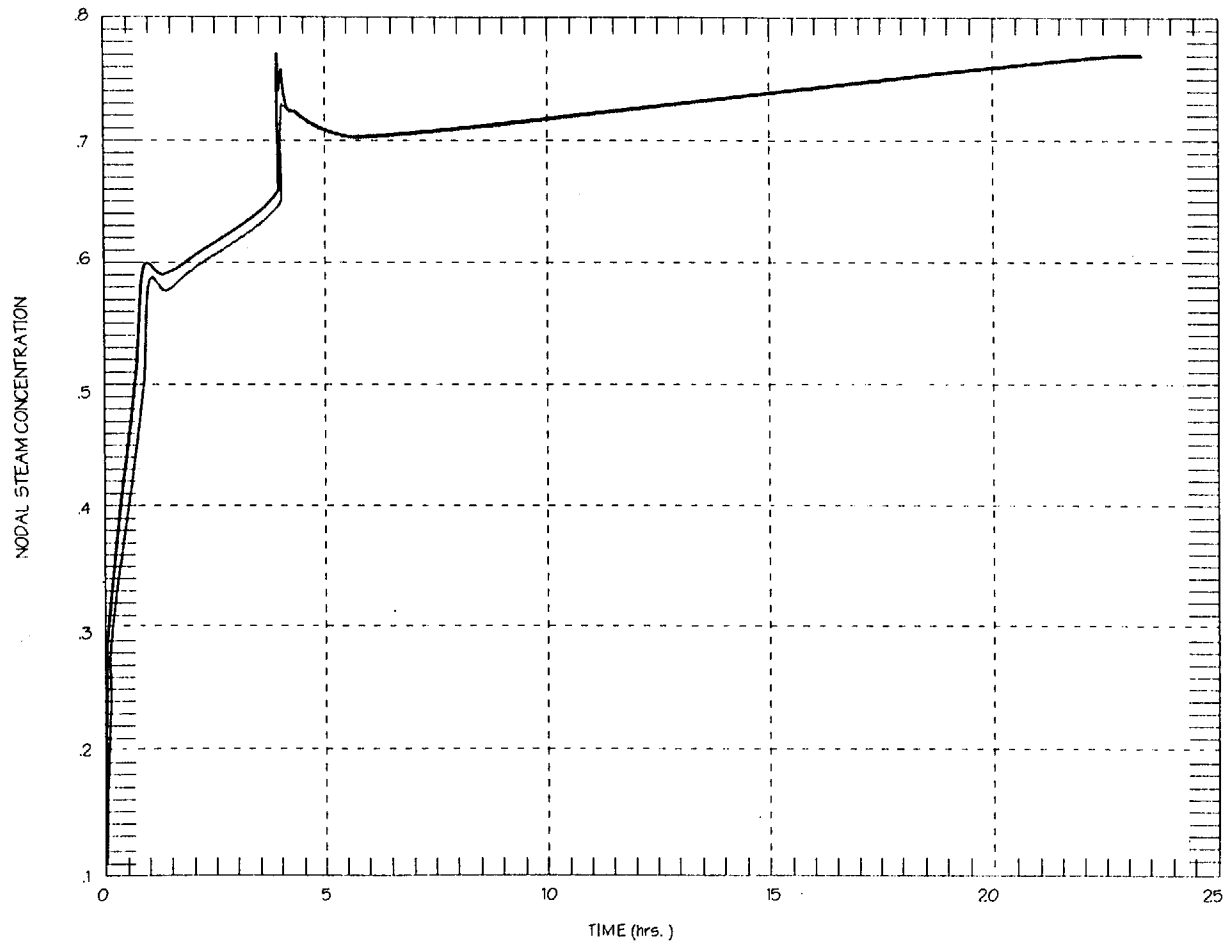
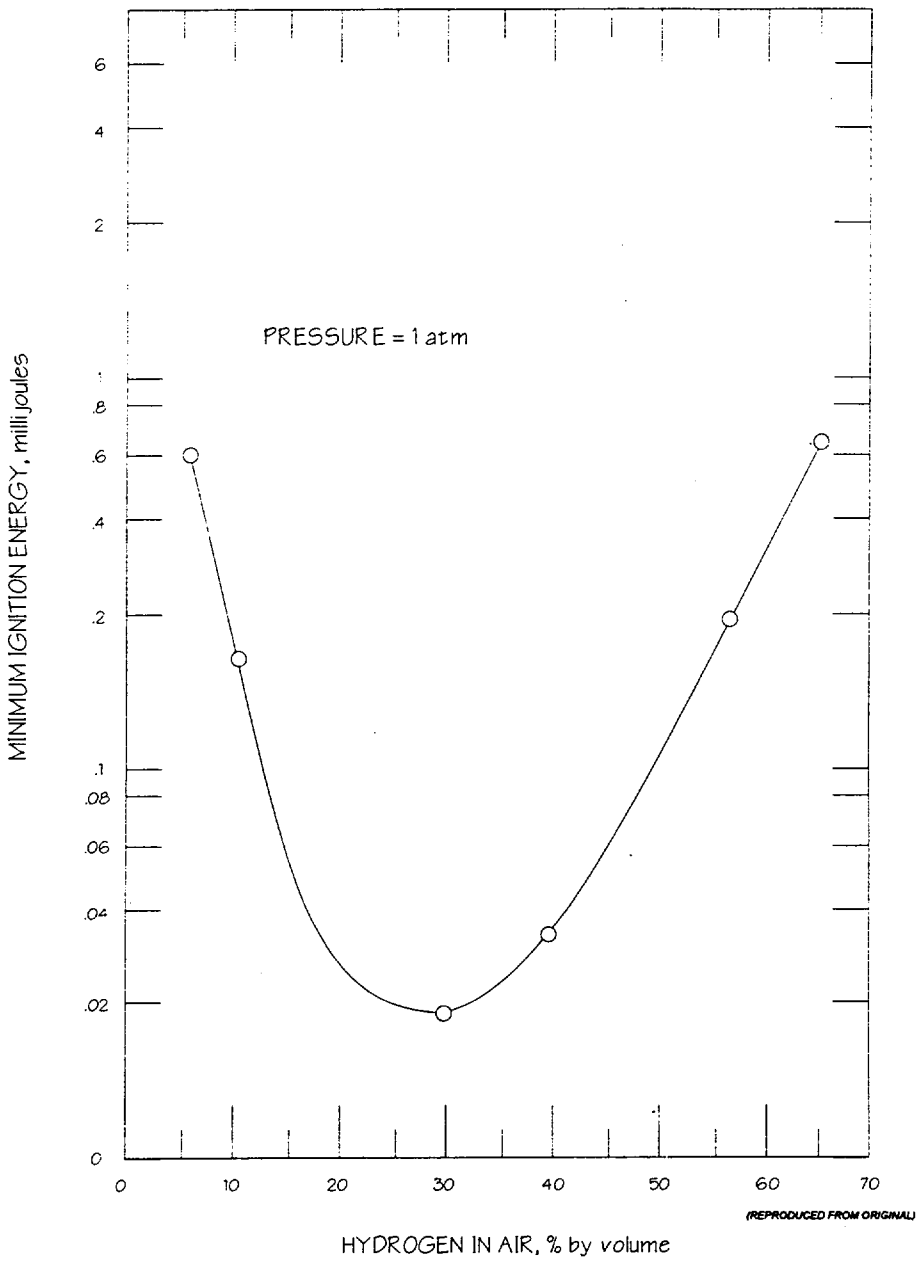


Figure 4–8
Energy Depostion Required to Initiate a Hydrogen Burn



Similarly, recent calculations (Ref 4–56) indicate that with a constant rate of energy addition to a mixture bearing hydrogen, the "induction time" required for a burn to occur decreases monotonically with concentration. Finally, it is noteworthy that the global burn in the Three Mile Island accident occurred at about 8 percent hydrogen concentration, i.e. the minimum concentration required for a complete burn under nearly dry conditions (Ref 4–13). In fact, this is the default MAAP model.

For cases with a large global burn, the post-burn pressure can be computed acceptably (albeit somewhat conservatively) using a simple energy balance method (Ref 4–14). For example, if we assume that 75 percent of the clad reacts (a high value based on both MAAP and detailed code calculations performed for other PWRs), this amounts to 1460 lb_m of hydrogen produced in SHNPP. A global burn requires a minimum hydrogen concentration of 8 percent. Because of the large containment volume relative to the zirconium mass in SHNPP, the steam fraction can be no higher than 30 percent if the hydrogen concentration is to be large enough to allow a global burn (Ref 4–57). If we then burn all of this hydrogen, the reference procedure for calculating the peak pressure indicates that the final pressure will be about 80 psia, for which there is a negligible chance of containment failure.

This value for the final pressure is actually somewhat high, since it neglects loss of energy during the burn to heat sinks or sprays, since it assumes that the burn is 100 percent complete, and because the hydrogen source term was large and the steam fraction was made as high as possible, i.e. just below the inerting limit.

It is concluded that there is a negligible chance of containment failure caused by the combustion of in-vessel generated hydrogen.

Cases in which containment heat removal is recovered late in an accident sequence, i.e. after a long period of hydrogen buildup from CCIs, are next considered. In such cases, deinerting of the hydrogen-rich atmosphere can occur over time as the steam is condensed. It is of interest to know the potential for large hydrogen burns. To explore this, a MAAP calculation, case SBO04, was made in which two trains of containment sprays and the fan coolers were recovered at 10 hours into a SBO sequence. This was about 5.3 hours after vessel failure, and a total hydrogen mass of about 1300 lb_m had collected in the containment at this time. The sprays deinerted the containment, and a burn was calculated to occur at just past 13 hours with a peak pressure of 75 psia. The delay time of 13 hours provides enough time for radionuclide deposition. Two conclusions can be drawn from this calculation:

(1) Containment failure is generally unlikely, since the steam fraction and thus the pre-burn pressure must be reduced greatly for the mixture to be flammable.

(2) In the unlikely event that containment failure would occur, an extensive period of spray operation leading up to the burn should scrub the containment atmosphere and result in relatively low fission products releases (Ref 4–58).

(3) Conclusions Regarding Hydrogen Behavior

The potential for DDT at SHNPP is judged to be small as in most other large dry containments because of the well-mixed character of the containment and the moderate hydrogen concentrations expected; detonations are essentially impossible without very effective containment heat removal and extended periods of CCI. Practically speaking, hydrogen deflagrations of any magnitude are also only possible if containment heat removal is available and either substantial DCH or a long period of CCI occurs. For cases with containment heat removal, hydrogen burns, if they occur, are not expected to result in containment failure given the containment volume, ultimate pressure, the expected magnitude of the hydrogen source terms, and the difficulty of accumulating large quantities of hydrogen in the presence of containment heat removal without getting a smaller burn first.

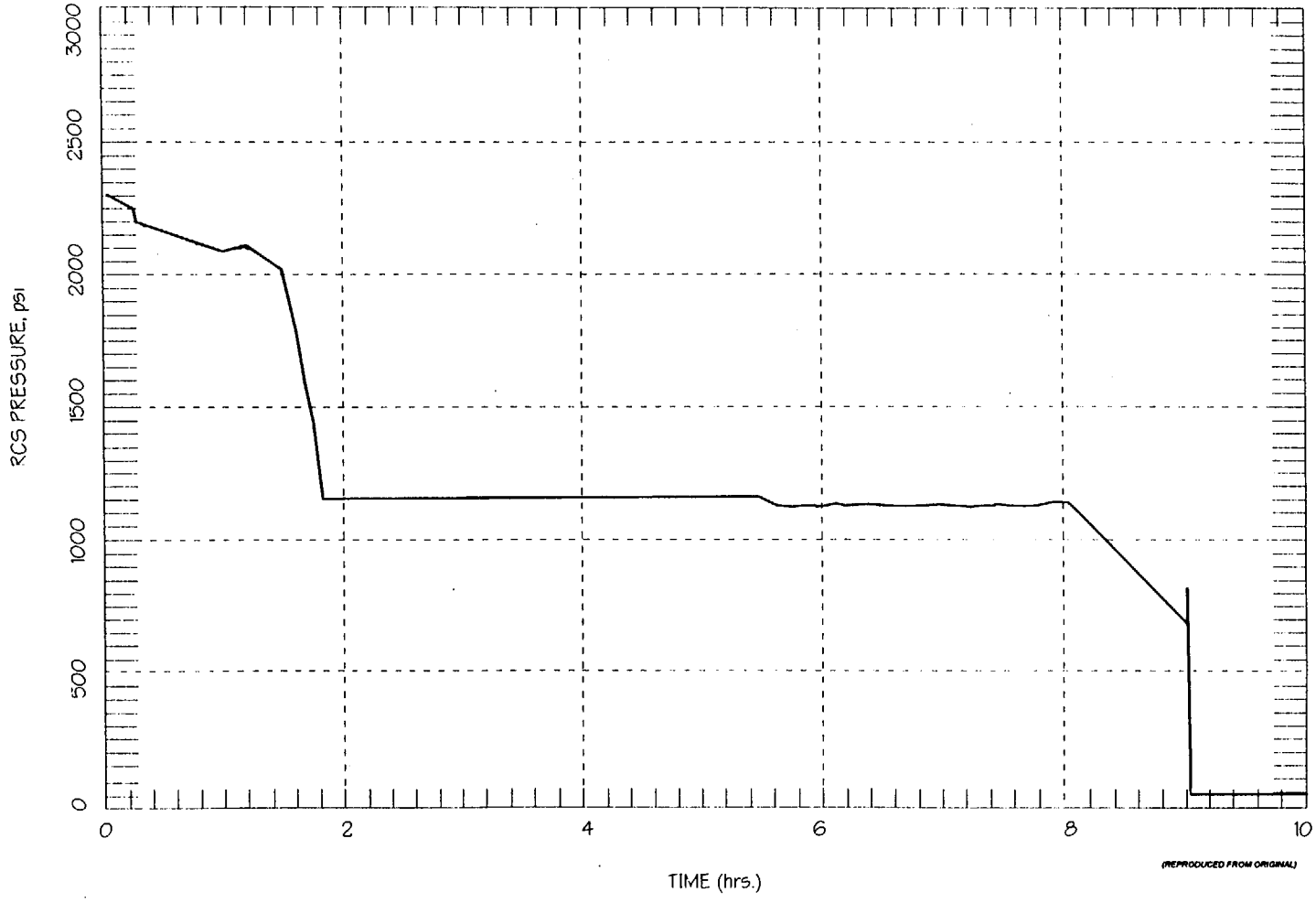
Prevention of Vessel Failure by Debris Cooling in the Lower Head

There has been great interest in the possibility that flooding the outside of the RPV could prevent vessel failure, even in cases in which no source of water is available to the vessel. This is not favored at SHNPP, in the sense that the typical water level in the containment if the RWST is injected would be approximately at the bottom of the RPV. Thus, continued injection of water e.g. by refilling the RWST, would be necessary for this event to occur. For this reason, no credit was taken for this phenomenon in the PRA.

Induced Ruptures of the Primary System

As will be discussed further below, the accident progression is extremely sensitive to the pressure in the RCS when melt-through of the RPV head occurs. For sequences involving loss of all feedwater, the primary system pressure will ultimately rise to a value corresponding to the lowest setpoint of an available pressurizer PORV or, if these are not available, a safety valve. The RCS may become depressurized if relatively large seal LOCAs occur, if a pressurizer PORV or safety valve sticks open, or if a PORV is manually opened by the operators. On the other hand, typical expected seal LOCA sizes, e.g., 250 gpm per pump at nominal RCS conditions as assumed in NUREG–1150, would only be expected to partially depressurize the RCS as shown in Figure 4–9 from case SH1P–2.

Figure 4–9
RCS Pressure During SBO Sequence



For cases with very small seal LOCAs, e.g., 21 gpm/pump, it is considered relatively likely that a pressurizer S/RV will stick open, since these cycle many times prior to core uncovering and for much of this time are discharging water rather than steam, but it is not known whether such a failure would be better characterized as a fully stuck-open valve or a weeping-type leak.

One last possibility exists. Code calculations and scale model experiments support the conclusion that the hot legs and surge line will be substantially heated by natural circulation of hot gasses from the core to the upper plenum and thence into the hot legs (Ref 4–59). Calculations by both MAAP and the SCDAP/RELAP code indicate that the steam generator tubes will not see a great increase in their temperature. (Ref 4–60). It is of great interest to assess whether the hot legs or surge line are heated enough to cause failure and depressurization of the RCS prior to RPV melt-through, since this would prevent phenomena which depend on an energetic blowdown of the RCS.

In MAAP, the surge line is not heated to the same degree as the hot leg for SBO sequences not involving a stuck open relief valve for the simple reason that flow into the surge line is reduced dramatically once the water level nears the bottom of the core and steaming diminishes. For this reason, higher temperatures are predicted in MAAP in the hot leg than in the surge line; SCDAP predicts the opposite for reasons that are not currently understood.

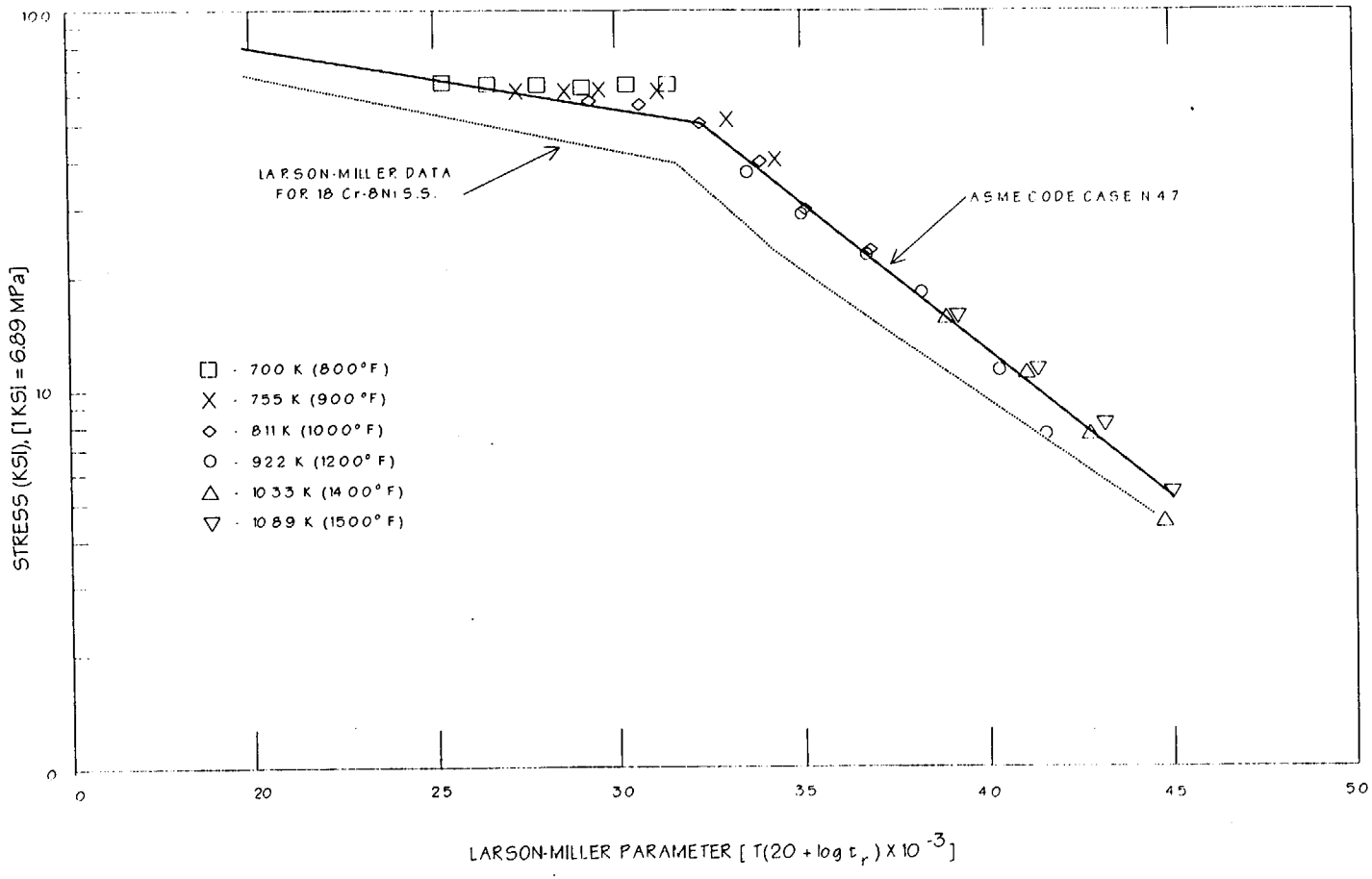
Whichever code is used, the temperature of the hot leg as a function of time is calculated. Two methods have been proposed to assess whether a given temperature profile results in hot leg failure. It has been recently proposed that an ultimate stress criterion is most appropriate. On the other hand, most analyses have used a creep rupture criterion, and this was done here.

For isothermal conditions, the appropriate way to assess the failure of a structure due to creep rupture is to consult a master rupture curve for the material in question such as that shown in Figure 4–10. For a given stress, the Larson-Miller parameter defines the time of rupture in terms of the temperature. One procedure that has been proposed for dealing with time-dependent (rather than isothermal) temperature profiles is to assume that the time of hot leg rupture, $t_{rupture}$ is given by the expression

$$1 = \int_0^{t_{rupture}} \frac{dt}{t_{rupture}(T,P)}$$

where T is the temperature of the hot leg as a function of time, and $t_{rupture}$ is the time required to fail the hot leg at a given temperature T and pressure P. This can be regarded as the analog of Miner's law applied to creep rupture.

Figure 4–10
Larson-Miller Parameter[T(20 + log t_r) x 10^{-3}]



(REPRODUCED FROM ORIGINAL)

This method gives results consistent with the results of a large hot leg rupture experiment run at MPA Stuttgart (Ref 4–61). A stand-alone program (Ref 4–61) was used to post-process MAAP-calculated hot leg (or surge line or steam generator tube) temperatures to assess the likelihood of creep rupture. Interestingly, hot leg rupture was not predicted in the nominal SHNPP MAAP 3.0B cases. This is in major part due to the fact that the hot leg piping itself is made of stainless steel which is less prone to creep rupture. It is quite possible that the nozzles, which are composed of low alloy steel, would be calculated to fail, but this would require a more in-depth study since both the stress field and the temperature distribution would be affected by the shape of the nozzle as well as the effects of the adjacent RPV wall.

When assessing these results, it is important to note that simplifications in the MAAP core melt progression model are believed to reduce the calculated hot leg temperatures and thus under-predict the likelihood of induced hot leg failure. In MAAP 3.0B, all core constituents (i.e. zirconium, uranium dioxide, and zirconium dioxide) are assumed to melt at a single "eutectic" melting temperature. Core heatup is relatively isothermal in high-pressure sequences due to the efforts of vigorous natural circulation between the core and the upper plenum. These factors have the effect of causing rapid gross melting of the core to occur once the eutectic temperature has been reached, and the disruption of core-upper plenum natural circulation follows immediately thereafter. This terminates heatup of the hot leg. Such a treatment is not considered particularly realistic. Based on small scale experiments, it is expected that the zirconium, along with some dissolved uranium, will relocate first, leaving behind the oxide materials in a relatively rod-like geometry. This would lead to an extended but slower rate of heatup of the hotlegs, which should ultimately lead to higher hot leg temperatures.

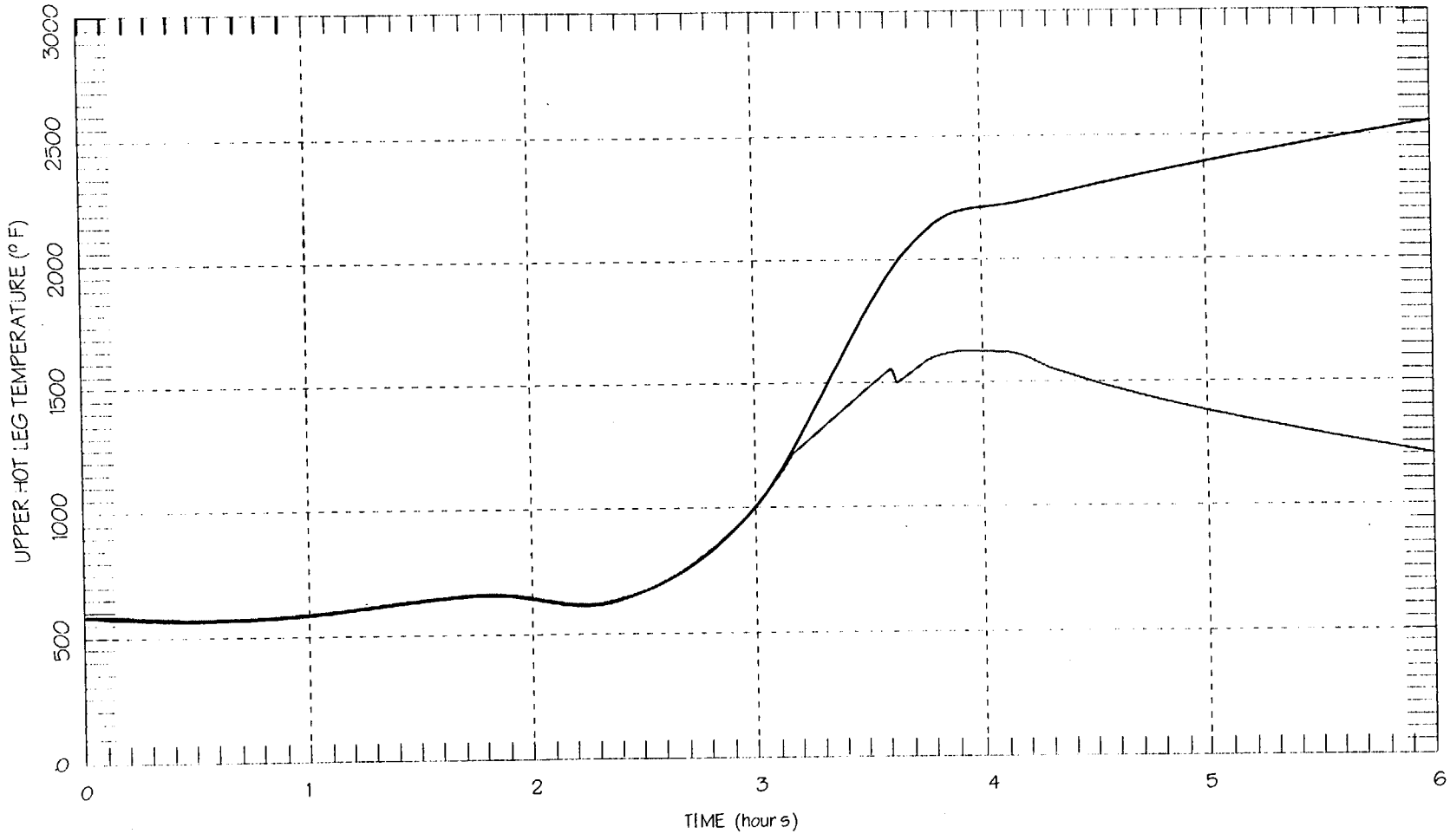
To investigate this possibility in a simplified way, the MAAP code version used was modified slightly to perform a sensitivity calculation. The modification assumed that:

a. zirconium oxidation ceased locally (in a core node) when the temperature reached a relocation temperature (oxide film "break out") of 2500 K, and

b. actual fuel melting does not occur until temperatures of 3000 K are reached.

This code modification is intended to capture the essence of the behavior observed in experiments in a simple way. The effects of these modifications were dramatic. Peak hot leg temperatures increased from about 1040K in the base case calculation to over 1700K, as shown in Figure 4–11. In addition, the time of vessel failure was greatly delayed.

Figure 4–11
Effect of Changes in Fuel Melting Model on Hot Leg Temperature for SBO



These results indicate that induced rupture could happen even for intermediate pressure cases. However, lacking a more detailed analysis of such cases, it was concluded that it would be most appropriate to adopt split fractions similar to those used in NUREG–4551, rather than rely quantitatively on the MAAP results. Thus, the following probabilities were assigned to the likelihood of induced rupture:

(1) cases with RCS pressures of about 2500 psia near the time of vessel failure – 0.9

(2) cases with RCS pressures of between 2500 psia and 1000 psia – 0.1

(3) cases with RCS pressures less than 1000 psia – 0.0

Based on the modified MAAP results and on preliminary MAAP 4 calculations (Ref 4–62), it is believed that these assignments are conservative in the sense that induced rupture would actually be more likely.

Timing and Mode of RPV Failure

SHNPP, like most PWRs, has in-core instrument tubes which penetrate the lower head. After molten debris relocates to the lower head, most previous analysts have assumed that either an instrument tube or the partial penetration weld which attaches the tube to the head would fail (Ref 4–4, 4–17). Ablation of the initially small opening would result in a final hole size on the order of 1 foot in diameter (Ref 4–17).

In the NUREG–4551 analysis for Surry, a low but non-negligible chance (13 percent) of gross lower head failure was assessed to be possible for medium to high RCS pressures; however, it is noteworthy that only 1 of 4 experts considered this a possible failure mode. The occurrence of gross failure in the study gave rise to an increased post-RPV failure peak containment pressure but was not associated to any significant degree with any other failure modes such as "rocket" induced containment failure or cavity collapse.

Recent work at EG&G (Ref 4–63) has indicated that global lower head rupture may be favored over instrument tube failures at high ($\geq$ 1700 psia) RCS pressures. To date, however, these analyses have apparently not considered jet impingement on the penetration tube weld, which was the assumed IDCOR local failure mode (Ref 4–17). The size of such a global rupture, should it occur, has also not been determined.

A user can input an arbitrary initial RPV failure size into MAAP. It was decided not to do this here, i.e. to confine attention to the default model which assumes an instrument tube failure.

a. The majority of accident sequences are not expected to result in RCS pressures higher than 1700 psia at debris relocation.

b. Creep failures of the lower head would not necessarily result in gross head failure due to expected non-uniformities in head heatup (Ref 4–46) and the difficulty in establishing a convecting pool (Ref 4–64). (A convecting pool would tend to result in peak heat fluxes being directed at the equator of the head and would thus promote gross failure.)

Two sensitivity cases were run to investigate the effect of vessel breach timing. Case SLOCA03 was a 2" LOCA in which the time required to fail the lower head was increased from the default value of 1 minute to 30 minutes. The extra time available for steaming in the lower head caused the RCS to repressurize to 1050 psia, i.e. about the pressure on the secondary side of the steam generators (see Figure 4–12). This occurs since the steaming rate is larger than can be relieved through the break. The increased RCS pressure somewhat increases the likelihood of HPME and related phenomena, although the RCS very quickly depressurizes after the lower head dries out. Thus, unless vessel failure occurs in a narrow window, the repressurization effect may not be very important.

Case SLOCA03A was similar except that it was assumed that the operators depressurized the steam generators in order to cool down the RCS. In this case, the repressurization was very minor. We conclude that repressurization can result in RCS pressures up to the secondary side pressure, which may be anywhere from ~200 psia to ~1000 psia. Even in the latter cases, there is a relatively small "window" in which vessel failure could occur at a much higher pressure than existed at the time just prior to core slump.

<u>High Pressure Melt Ejection, Direct Containment Heating</u>

As stated previously, the SHNPP cavity configuration is qualitatively similar to that of Surry. Scale model experiments conducted at Brookhaven National Laboratory (Ref 4–24) confirm that essentially complete debris dispersal would be expected from this type of cavity if the primary system fails at medium to high pressures. This is called HPME in the literature. The cutoff pressure at which HPME is not expected is generally believed to be in the range of a few hundred psia as illustrated in Figure 4–13 (Ref 4–28).

MAAP contains a very simple one-dimensional model for predicting whether debris dispersal occurs or not, and it is believed that use of this model is reasonable.

Figure 4–12
Effect of Delay in Vessel Head Failure on RCS Repressurization

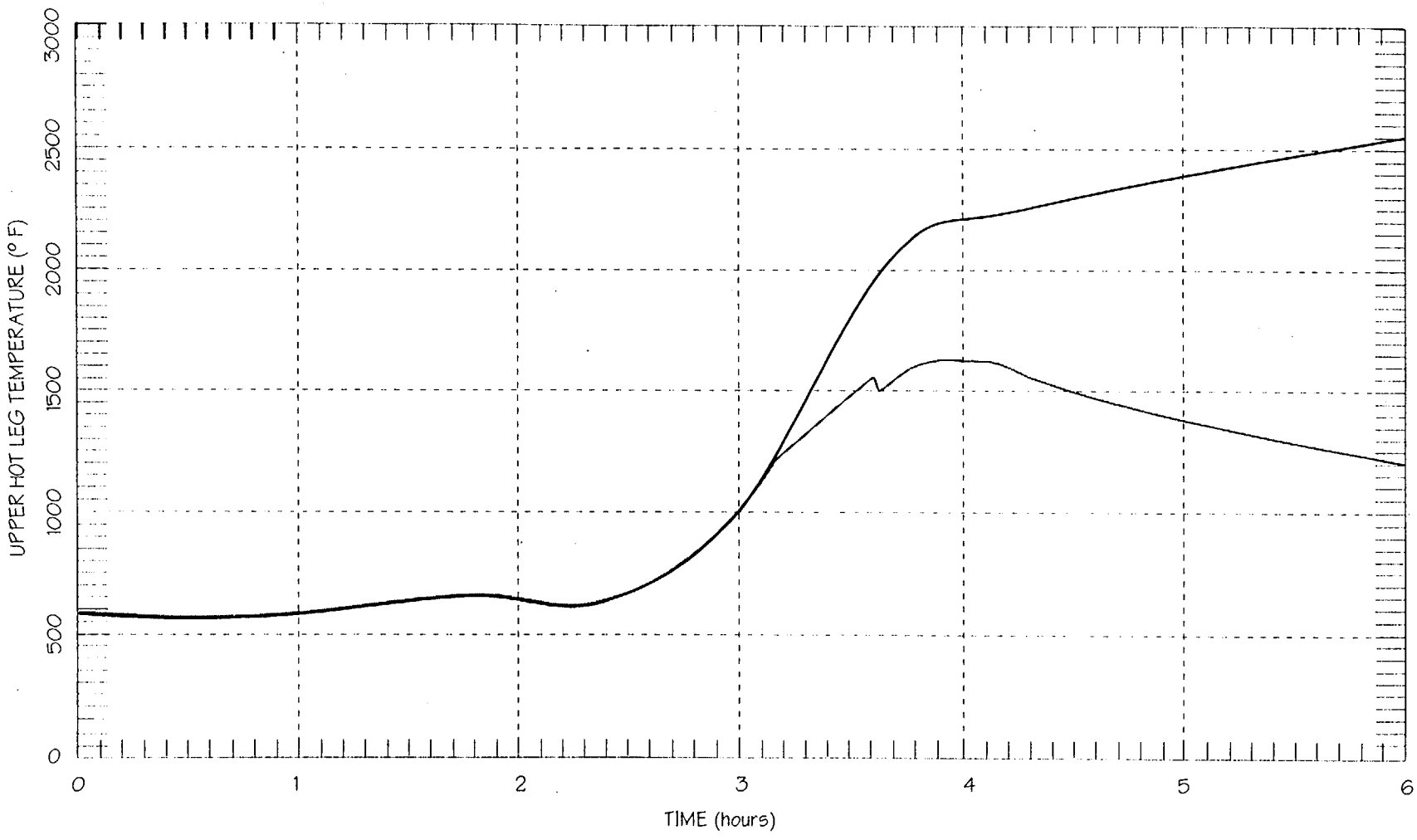
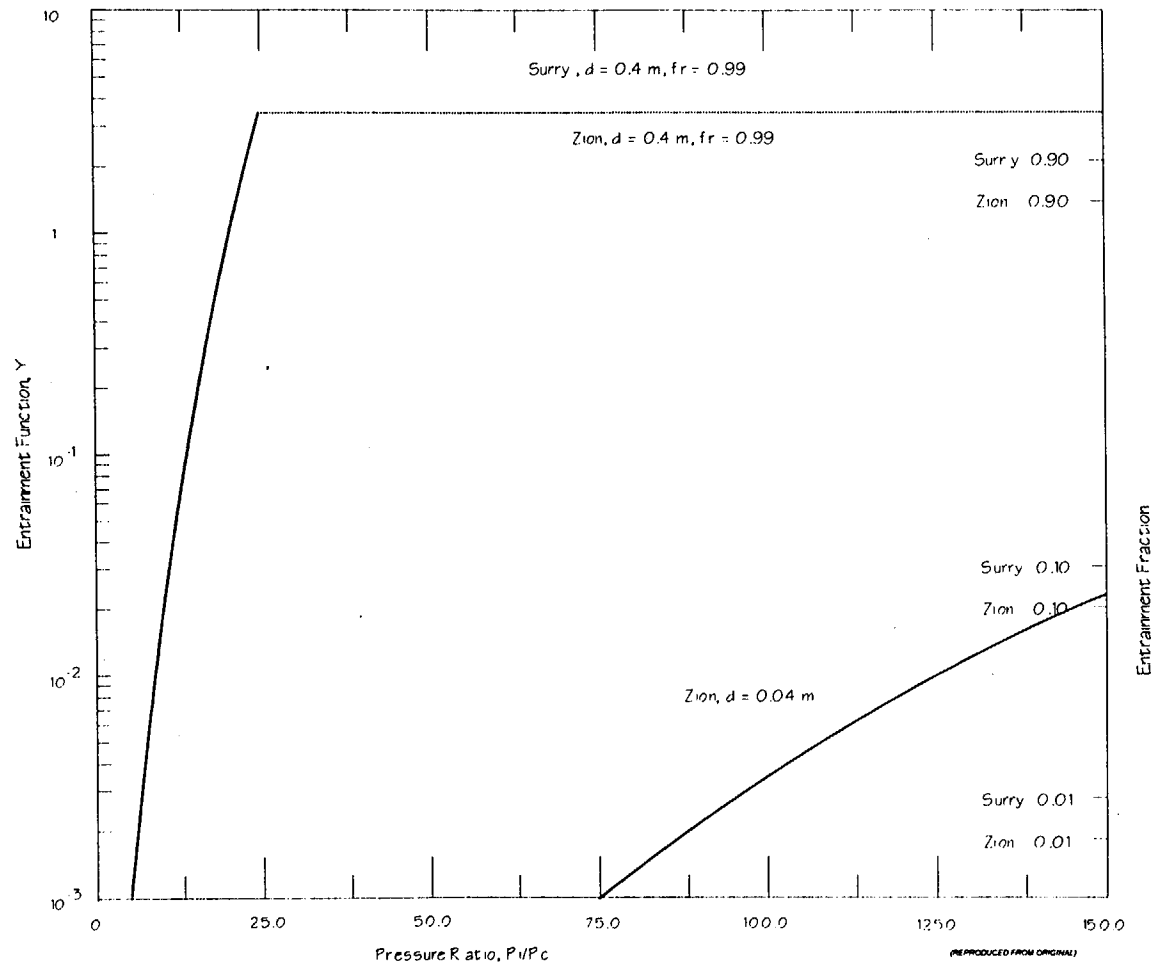


Figure 4–13
Effect of RCS Pressure P_i on Debris Entrainment



What are the implications of HPME? First, the fact that the debris is dispersed from the cavity would appear to greatly reduce the likelihood that it cannot be cooled by an overlying water pool. Based on scale model experiments, the debris will be dispersed over a relatively wide area adjoining the mouth of the instrument tunnel, and it does not appear very likely that continuous crusts could be formed on the dispersed debris that would effectively insulate the debris from the water (see Section 4.8).

What is not clear is the effect of the debris dispersal process itself on the containment pressure loading, i.e. whether DCH of the containment can threaten its integrity.

Recent integral experiments conducted at Sandia National Laboratories and at Argonne National Laboratory indicate that the "heating" aspect of direct containment heating has a relatively small impact on containment pressurization. This is the case because the bulk of the debris is confined to a relatively small volume, and the energy which can be extracted from the debris is then limited by the small heat capacity of the recipient gas. This effect has been exploited in a simple scoping model for DCH developed by Pilch (Ref 4–65) which has been relatively successful in explaining a wide variety of data. The same experiments imply that hydrogen production during the event is a more significant potential contribution to containment pressurization. Relatively complete hydrogen production has been observed in experiments, though as discussed in Section 4.2 there is still the possibility that the reaction could be steam limited if large quantities of debris are involved. Just as significant, however, is the observation that combustion of this hydrogen is incomplete and occurs over an extended time frame (Ref 4–49). In fact, only a small fraction of the hydrogen burned in an experiment with a high initial inertant concentration typical of SBO sequences. Thus, combustion will not contribute the maximum possible pressure increment to the total load imposed on the containment at vessel failure. This is interpreted to be a result of the lower containment regions being oxygen starved and the upper containment regions are at relatively low temperature so that global burns are prevented there by steam inerting. Only incomplete burns are seen as hot, hydrogen rich jets pass through the operating deck and enter the oxygen-bearing upper region of the containment.

MAAP contains a scoping model for DCH which requires one to input the fraction of the debris which is finely fragmented (i.e. to a size so small that completely efficient oxidation and heat transfer are assumed to occur), the fraction of the dispersed debris which is expelled through the reactor annulus (rather than through the instrument tunnel), and the time scale over which dispersal occurs. Increasing all three parameters will tend to increase the total amount of energy transferred to the containment atmosphere. One other input to the model concerns how burns are treated. By default, MAAP uses typical "maps" of hydrogen versus steam concentration to determine if a mixture is flammable. As discussed above, experiments to date have indicated that only incomplete burns would be expected. However, one also has the option of forcing MAAP to always burn the hydrogen which is released from the RPV or created during the DCH process itself.

Informal comparisons of the MAAP model to more detailed, yet still parametric CONTAIN results (Ref 4–66) indicate that the models predict similar results if the CONTAIN debris "trapping" fractions are input in such a way as to prevent substantial debris dispersal into the upper volume of the containment. As stated before, dispersal of large quantities of debris into the upper compartment would not be expected at SHNPP, and even allowing the debris to mix with the entire MAAP "lower compartment" would appear to be unrealistic (though conservative).

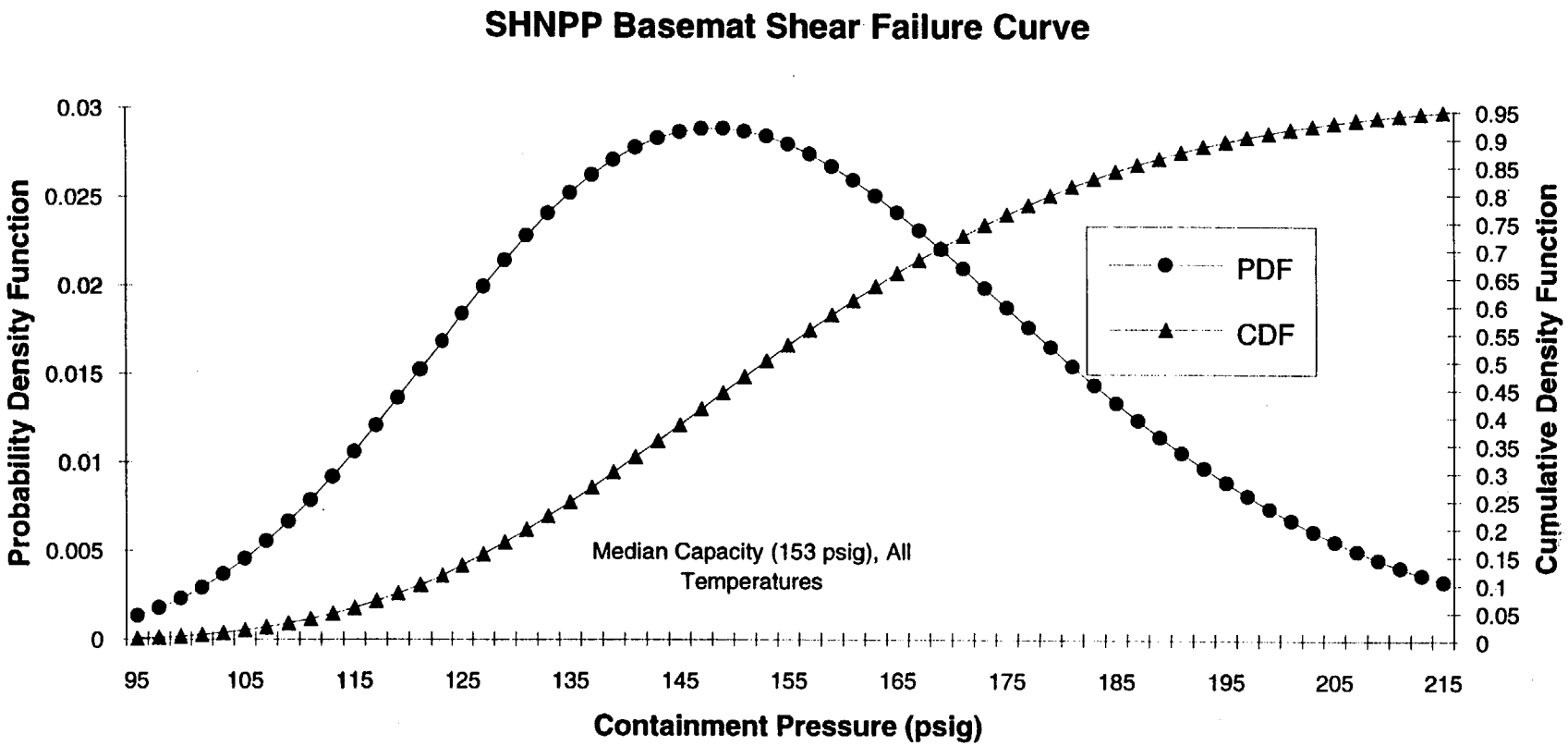
Several scoping calculations were run with the MAAP model, and the peak pressures obtained are listed below. In all cases, approximately the entire core mass was discharged at vessel failure, which is quite conservative based on NUREG–1150 assessments.

Case	Fraction Fragmented	Time Constant for Dispersal (sec)	Burns Forced to Occur?	Peak Pressure (psia)
SBOOOA	1	0.5	Yes	100
SBOOD	1	2	Yes	110
SBOOOC	5	.5	No	75
SBOOB	.5	.5	Yes	85

These results show a clear effect of forcing hydrogen burns to occur and assuming a smaller amount of debris is finally fragmented. Based on the IET results, the best estimate results would lie near case SBOOOC i.e. a final pressure of about 75 psia.

The containment fragility curve for SHNPP is shown in Figure 4–14. Considering the curve, even under the most conservative assumptions employed here, it is concluded that DCH is very unlikely to cause containment failure.

Figure 4–14
Containment Fragility Curve



Contact of Dispersed Debris with the Containment Liner

If a severe accident should occur at SHNPP and progress to the point of vessel failure, the subsequent behavior depends greatly on the pressure in the primary system at the time of failure. If the pressure is high, as discussed previously simulant fluid experiments as well as calculations indicate that debris would be forced from the cavity, mainly through the instrument tunnel. A key uncertainty is the form that the debris would take. If the debris was highly fragmented, substantial cooling of the debris could be expected as it interacted with the water which lies on the containment floor at that time. On the other hand, some evidence indicates that the debris could be expelled from the cavity mainly as a coherent film; this would reduce the magnitude of debris/water interactions. The available analytical investigations (Ref 4–28) indicate that a significant fraction of the debris could exist in both states, at least in very high pressure sequences.

Consider a sequence in which a relatively large quantity of debris is expelled from the reactor cavity as a coherent mass. The debris would enter the area below the seal table that lies inside the secondary shield wall. There is a possibility that the debris will then flow (or be pushed through) the 13 drain holes (either 18" or 36" wide) which lie at the bottom of the northern half of the secondary shield wall and contact the containment liner. This scenario is reminiscent of the Mark I liner melt-through issue (Ref 4–67), but a major difference is that the Mark I debate has focused nearly entirely on low pressure melt spreading transients, which are not relevant here.

The subsequent heatup of the liner is greatly affected by the depth of debris in contact with it. The debris that was pushed into the annular region between the secondary shield wall and the containment wall would either be already cooled sufficiently that it would pose little risk to the liner, or, if still molten, it would be expected to spread in the azimuthal direction. The available floor area in the annular compartment which is immediately adjacent to the drain holes in question is approximately 1200 ft^2. For reference, the floor area inside the secondary shield wall in this area is about 500 ft^2.

Based on NUREG–1150, a reasonable best estimate for the mass of core debris which would be released at vessel failure is about 50 percent of the total, or a volume of approximately 200 ft^3. If this mass was spread over the sum of the two areas just identified, the depth would be approximately 0.1 ft (3 cm). Porosity or additional mass discharged at vessel failure could make the debris deeper; discharging less debris from the vessel, the cavity, or the region inside the secondary shield wall would make the depth smaller as would additional azimuthal spreading inside or outside the secondary shield wall. An upper bound estimate of debris depth can be obtained by discharging the entire 400 ft^3 core volume to the 1200 ft^2 floor area which lies in the 90° sector between the drain holes and the containment liner; this would result in a depth of about 0.3 ft (10 cm).

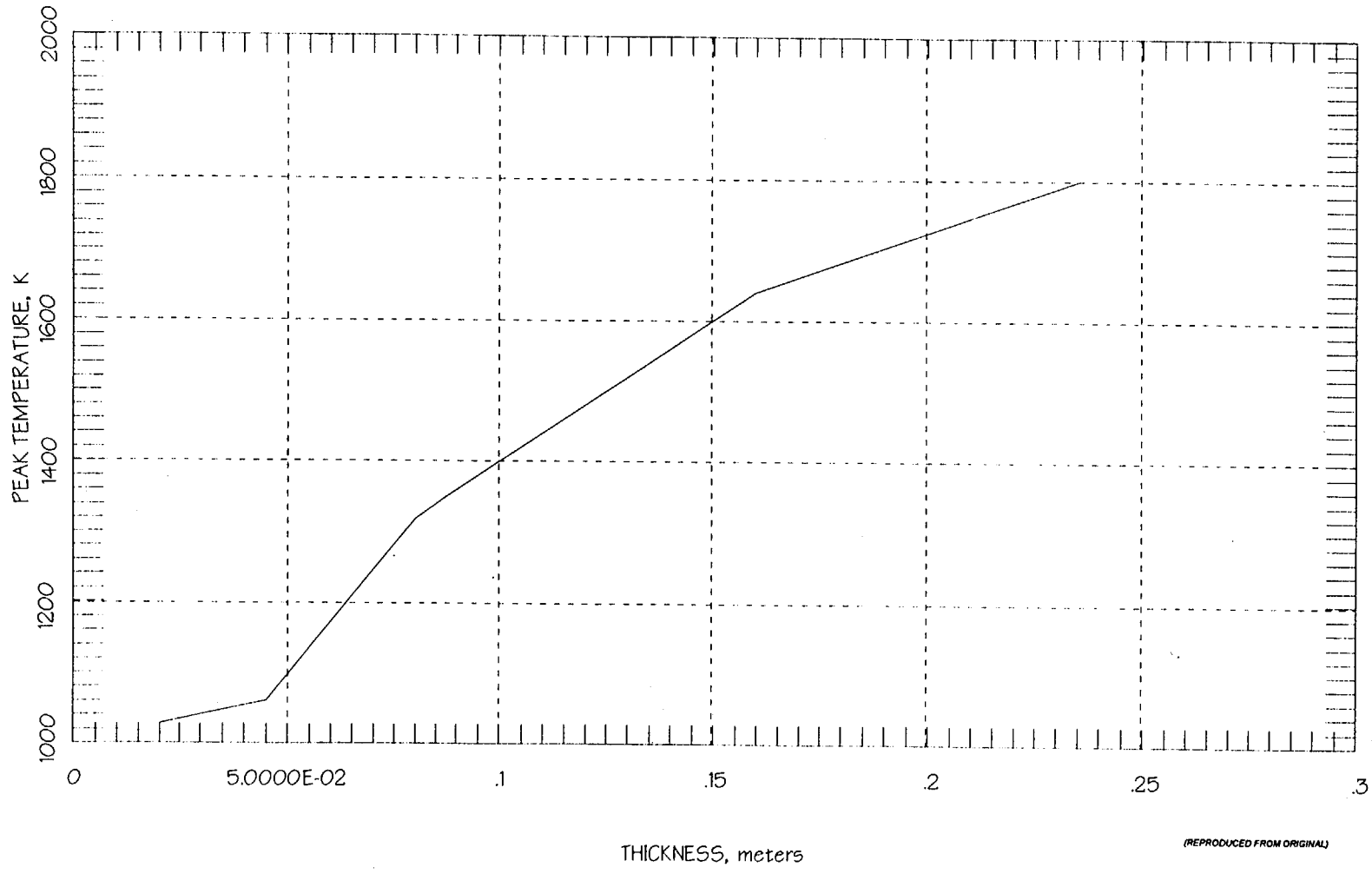
A considerable amount of water would be expelled with the debris and soon afterwards when the accumulators discharge. This water, along with the water that would already be present on the annular compartment floor at the time of vessel failure (about 6" based on MAAP calculations in SBO sequences and 6 feet or more in sequences in which the RWST has been injected) would substantially cool the debris. Based on experiments performed at Fauske and Associates to study the Mark I issue, a very high heat flux can be expected for the transient period when the debris is moving towards the wall and its surface is very hot, i.e. on the order of 20 MW/m^2 of floor area (Ref 4–68) corresponding to the hydrodynamic "flooding" rate. As noted by Theofanous and his co-workers in the reference (Ref 4–67) study on the Mark I issue, such debris/water interactions could prevent or disrupt intimate debris contact with the liner altogether.

If we assume that this does not occur, for the small debris thicknesses considered most likely an inherently two-dimensional problem is set up. The liner is heated by contact with the debris. It is cooled by conduction in the radial direction into the concrete which lies behind the liner, and by axial conduction into the concrete floor and the overlying water pool. To estimate the liner's response, a 2-dimensional conduction model was developed to solve for the temperatures in the debris, liner, and concrete as a function of time.

Separate calculations were made with assumed debris depths of 2, 4, 8, 16, and 24 cm. In each case, it was assumed that an overlying water pool existed for at least a time sufficient for the debris to quench. This is guaranteed for sequences in which the RWST has been injected, but is less certain for SBO sequences (recall that the water is only about 6" or 15 cm deep in this case). Short-term quenching rates from the debris surface were conservatively assumed to be only 1 MW/m^2 (a typical debris bed dryout heat flux and only one twentieth of the flooding limit). This was translated into a velocity for the quench front that would move downward into the debris over the time frame of interest. For additional conservatism, the core material was assumed to be fully molten at an effective melting temperature of 2500 K; decay heating was modeled in the debris, but it was assumed that there would be little contribution of chemical heating (zircalloy oxidation) in the area immediately adjacent to the liner due to rapid cooling.

The peak liner temperatures obtained with the model are shown in Figure 4–15.

Figure 4–15
Peak Containment Liner Temperature



(REPRODUCED FROM ORIGINAL)

As shown, liner temperatures exceed the concrete melting temperature for debris thicknesses greater than about 15 cm and approach the steel melting temperature for debris thicknesses greater than about 20 cm. It should be noted that there is a substantial gradient through the liner in both the radial and axial directions so the peak liner temperature somewhat overstates the threat to the liner. Also, the conduction calculation neglected debris porosity, which is conservative (for a fixed debris depth) from a quenching, heat conduction and internal heat generation standpoint.

When evaluating the significance of these calculated temperatures, it should also be noted that the SHNPP containment liner, unlike that in Mark I BWRs, is backed up by the containment wall itself. The lack of an intervening gap between the liner and the wall implies that:

a. The liner serves essentially as a membrane for leak-tightness. Since the cracks expected in the concrete wall as the containment pressurizes are narrow, a rather small intact thickness would suffice for preventing liner failure, even at elevated temperatures.

b. If liner failure did occur, the failure would tend to be self-sealing as debris was pushed into the resulting gaps (Ref 4–67).

We conclude that the likelihood of liner failure is negligible for low pressure sequences in which debris dispersal is minimal, and quite small for sequences in which the RWST has been injected and the debris would be forced energetically through a relatively deep pool.

For unrecovered SBO sequences which fail at high RCS pressure, the water mass is relatively small in the lower compartment, and we would expect liner integrity to be challenged in cases in which relatively large amounts of debris are forced into the annular compartment region due to the shallow water depth. Even if prompt failure was avoided, the liner would again be threatened after the water was boiled away. On the other hand, MAAP calculations imply that whereas only a 6" layer of water exists on the containment floor in an SBO sequence, continued condensation of water steamed from the debris keeps the debris covered for more than a day. Given the relatively thin dimensions of this layer of water, these results may be somewhat optimistic. The conclusion that liner failure could be expected in cases in which there was no water on the drywell floor was also reached in the Mark I study. Failures could be self-sealing, however.

Finally, it should be noted that the containment recirculation sump lies in the annular region outside the secondary shield wall near the area where the debris is expected to flow. Nevertheless, the debris is not expected to enter the sump because a 1–1/2' concrete wall, slightly higher than the openings in the secondary shield wall and deeper than the expected debris layer thickness, protects the entire perimeter of the sump.

Coolability of Debris Covered by Water in the Cavity or Lower Containment Floor

In sequences which have an operational containment heat removal system, if the containment survives the loadings imposed on it by hydrogen combustion and phenomena occurring around the time of vessel failure, its fate depends in large measure on whether the debris expelled from the RPV can be cooled sufficiently to avoid concrete attack.

If water covers the debris, the ability of the debris to be cooled must then be addressed. As demonstrated in reference 4–69, if there are no cracks in the crust and it is desired to prevent concrete ablation, conduction limits the heat flux which can be obtained from a solid debris layer to a value on the order of 150 kW/m^2. For reference, if we assume that all the core debris is spread uniformly over the SHNPP cavity floor, at a decay power of 1 percent of which 80 percent is still in the debris (the remainder having been released in the form of volatile fission products and noble gasses), the required heat flux for steady-state is 400 kW/m^2; this neglects any heat load from chemical reactions which would eventually cease. If all the debris is assumed to be dispersed over the 180° sector inside the secondary shield wall, the required heat flux is about 100 kW/m^2; this observation reinforces the notion that dispersed debris should be coolable. The actual degree of debris spreading is not known, but it is thought reasonable that spreading will be extensive (Ref 4–70).

Experiments performed at Sandia National Laboratory and Fauske and Associates have produced asymptotic heat fluxes on the order of 800 kW/m^2, but considering the implications of the conduction calculation, such heat fluxes cannot be maintained unless one of the following three conditions occurs:

a. Molten conditions are maintained inside the pool so that energy can be convected to the debris surface rather than relying solely on conduction. Such convection would nearly certainly result in downward heat fluxes sufficient to cause concrete ablation and thus eventual containment failure (albeit after several days).

b. The surface of the debris allows water to ingress into the debris so that conduction limits are unimportant.

c. The debris surface is highly irregular, so that fin effects reduce the conduction constraints on the heat flux. Based on visual observations of the debris simulant surface configuration, it is not expected that this could result in augmentation sufficient to explain the observed heat flux.

There is no definitive data available that pertains to the issue of water ingression. Several recent experiments in the MACE test series at Argonne National Laboratory exhibited

non-coolable or at most partially-coolable debris. However, in all cases a crust tended to form which was supported by the walls of the experimental apparatus; for this reason, these results are not really relevant to the reactor-scale case in which the distance to the walls would be too large to support an integral crust. The Fauske and Associates experiments mentioned previously did consistently exhibit a heat flux of about 800 kw/m^2 which was sustained until the debris began to quench completely. One may criticize the scale of these experiments also and the non-prototypical materials used for debris simulants, however. Some experiments conducted with lava in Iceland did show unambiguous ingression of water by many feet (Ref 4–71).

The IPE Generic Letter states that the possibility that debris may not be coolable should be considered for debris layers deeper than 25 cm. If the entire core volume is relocated to the SHNPP cavity and spread uniformly, the resulting debris thickness is about 22 cm.

To explore the effects of varying debris coolability, two small break LOCA sequences with early vessel failure were run, SLOCA01 and SLOCA02. In both cases, the debris was confined to the reactor cavity, injection and sprays were lost when the RWST dried out, but fan coolers were available throughout. The heat flux from unquenched debris was controlled to be about 800 kw/m^2 in the first calculation and about 150 kw/m^2 in the second at atmospheric pressure; the model-calculated heat flux increases somewhat as the pressure changes to about 200 kw/m^2.

In case SLOCA01, the debris quenched quickly in the reactor cavity, and only about 0.1 ft. of concrete ablation and 50 lb_m of hydrogen were produced ex-vessel. In the second case, concrete ablation occurred over an 11-hour period until the decay heat dropped sufficiently for the debris to freeze and end the attack. The presence of water over the interval reduced the rate of ablation so that only about 200 kg of hydrogen was produced. The final erosion depth was only 0.25m, and no hydrogen burning resulted.

Revaporization of Fission Products

One of the potential long term sources of fission products in severe accidents results from previously settled aerosols which revaporize from overheated primary system structures.

The MAAP model for revaporization, while mechanistic, neglects the possibility that fission products could chemically react with steel structures in the RCS. This has been seen in experiments (Ref 4–72), and the NRC staff have expressed the concern (Ref 4–73) that such reactions could delay revaporization in such a way as to increase the source term in sequences (i.e. SBO) sensitive to this phenomenon.

To explore this possibility, the base case SBO sequence was repeated with the vapor pressures of cesium iodide and cesium hydroxide which drive the revaporization rate each reduced by a factor of 10. The result was a slight increase in the fission product mass

retained in the RCS (from 89 to 91 percent) and a small drop in the environmental release (from 1 percent to 0.7 percent). This is in agreement with a more extensive set of sensitivity calculations recently performed at Fauske & Associates (Ref 4–74). Based on these results, no explicit consideration was given to the retardation of revaporization caused by chemical reactions in the PRA.

4.3 Bins and Plant Damage States

In order to develop a complete accident sequence definition for transfer to the containment assessment, the core damage bin information must be combined with the status of the containment systems. The combination of these two characteristics results in a plant damage state. This section addresses the development of core damage bins, a model to address containment systems (the containment safeguards event tree), and how the two are then combined to form the plant damage states for the SHNPP PRA.

The solution of the numerous event trees results in the generation of a large number of accident sequences. Once developed, these accident sequences must be propagated through the containment safeguards assessment and then the containment event tree in order to develop release categories. To reduce the burden of the analyst the accident sequences can be grouped, commonly referred to as binning, into accident sequence categories.

The containment safeguards event tree (CSET) provides a means for interfacing the core damage model with the containment safeguards functions. The event tree addresses the status of the containment systems in order to complete the system-level information needed by the level 2 PRA analyst. Additionally, the use of a CSET which incorporates fault tree and event tree models allows the core damage sequence cutsets to be linked directly to the CSET. The direct linking of the models results in containment and core safety system dependencies being identified and explicitly addressed. This section describes the CSET and the top events which makeup the model.

4.3.1 Accident Sequence Binning Attributes

Methodology for Developing Core Damage Bins

The method of binning the accident sequences is much like that used to categorize the transient initiating events. A set of parameters is identified which can be used to define unique accident sequence classes. These parameters are typically defined based on needs of the level 2 analysis. For example, one parameter commonly used in the binning process is RPV failure timing. The timing of vessel failure is important to the level 2 analysis in determining the timing of containment pressure challenges, in-vessel hydrogen generation, and the radionuclide retention within the RCS. This parameter, therefore, is typically chosen for binning accident sequences. Once the important parameters are identified the next step is to determine the physically possible combinations of the parameters. Each combination is considered a core damage bin definition.

Due to the physical processes involved, some combinations of events are not possible. For example, RCS leakage rate may be chosen as one binning parameter with a large

LOCA leakage rate applied to accident sequences in which a large RCS break is present. For these sequences, the RCS pressure would be near the containment pressure. A second parameter could be RCS pressure. It is illogical to combine a large LOCA leakage rate with an RCS pressure condition of high RCS pressure. This type of assessment is performed for all potential core damage bins to ensure that logical bins are developed.

Once the core damage bins are finalized, the different event tree accident sequences are assigned to the core damage bins by comparing the core damage bin parameters and the cutsets which comprise the specific accident sequence. It is important to have a basic understanding of the accident sequences and the component failures which comprise the accident sequences in order to properly bin the sequences. In some cases, cutsets from one accident sequence are placed into more than one bin to accurately account for the accident sequence progression identified by the specific cutset.

Identification of Core Damage Bin Parameters

The core damage bin parameters are based, for the most part, on a review of prior analyses, plant-specific analyses, and engineering judgment. The status of various safety systems is important in determining accident progression and is also taken into account.

The following sources were reviewed and used as reference material during the development of the core damage bin parameters:

- Zion PSA
- Seabrook PSA
- Millstone 3 PRA
- Oconee PRA (NSAC/60)
- NUREG 1150
- NUREG/CR 2300
- NUREG/CR 2815
- NUREG/CR 4550 (Surry, Zion, and Sequoyah)
- HBR2 PRA

Based on a review of the information provided in these reports, a list of core damage bins is developed. Additionally, the SHNPP safety systems were reviewed to determine if their operation or failure would significantly impact accident progression. For example, the operation of auxiliary feedwater (AFW) following an S1 class LOCA will result in the timing of RPV failure being extended due to the decay heat removed via the steam generators. If this decay heat removal mechanism is not present, the energy will be retained in the RCS and core heatup will occur sooner. Thus, the status of AFW is important in assessing containment response.

In choosing binning parameters it is important to minimize the number chosen, to the extent possible, to reduce the number of potential core damage bins. If too many bin parameters are chosen, the benefits gained by binning the accident sequences are lost. A general rule is that each parameter should differentiate a unique feature or event occurrence necessary to identify differences in the accident sequences during the performance of the containment assessment. The parameters identified for the SHNPP PRA are presented in Table 4–4 and their selection is described in the following text.

Table 4–4
Core Damage Bin Parameters

Parameters
RCS pressure at RPV breach
RCS leakage prior to RPV failure
Presence of water in the reactor cavity and lower compartment at RPV failure
Availability of heat removal using the steam generators
Timing of RPV failure

RCS Pressure at RPV Breach

The RCS pressure at RPV breach is important in assessing several containment phenomena, such as the likelihood of high pressure melt ejection and subsequent direct containment heating, location of debris following RPV failure, and the potential for RCS pipe failure due to creep rupture prior to RPV failure. Based on analyses performed as a part of the NUREG 1150 study and other sources, three different pressure regimes are identified which result in different containment responses. These pressure regimes (Table 4–5) are adopted for SHNPP. For convenience, these pressure ranges are qualitatively described as high, medium, and low RCS pressures.

Table 4–5
RCS Pressure Ranges

RCS Pressure	Classification
Operating pressure to 600 psig	High
Between 200 and 600 psig	Medium
Less than 200 psig	Low

These three regimes represent the different possibilities for this core damage bin parameter. Each of the accident sequences is binned into one of three bins based on predicted plant pressures at the time of RPV failure. The binning process includes consideration of RCS depressurization as a result of operator action, stuck open relief valves, or breaks in the RCS.

RCS Leakage Rate Prior to RPV Failure

The RCS leakage rate is an input in evaluating event timing, fission product retention in the RCS, and hydrogen release rate into containment. It is also important in addressing containment event tree issues such as the source term in the containment and RCS pressure since there is a direct relationship between the RCS pressure and RCS leakage rate. In a few instances, however, the leakage rate may change during the course of the event and not greatly impact RCS pressure.

Based on a review of prior PRAs and the potential leakage rates identified by the initiating events, four leakage rates are defined. These leakage rates and example accident sequences are listed in Table 4–6.

Table 4–6
RCS Leakage Rates

RCS Leakage Rate Description	Example Accident Sequence
Cycling relief release rate	Station blackout without a reactor coolant pump (RCP) seal LOCA and no decay heat removal
Small LOCA release rate	Accident resulting in a RCS (PORV stuck open, RCP seal LOCA, or either small LOCA class
Medium LOCA release rate	Medium LOCA accident sequences
Large LOCA release rate	Large LOCA accident sequences

Other phenomena associated with core melt progression may also impact the RCS pressure. NUREG 1150, along with other recent analyses (Ref 4–34) have postulated that (for accidents sequences involving high RCS pressure and temperature i.e., cycling relief valve leakage rate) the potential exists for creep rupture of RCS piping due to pressure and temperature loading. The failure locations most often postulated have been the hot legs, pressurizer surge line, or steam generator tubes. The failure of the RCS results in RCS depressurization. For these sequences, the RCS release rate changes with time and may have characteristics of both cycling relief and LOCA release rates. It is difficult to address these phenomenological issues in the core damage model. To simplify the binning process and the event tree analysis, the potential for creep rupture is not addressed in the core damage models but is treated in the CET.

The possibility of creep rupture is addressed in the CET for those sequences which are not at low pressure. In evaluating the phenomena associated with both high and medium pressure sequences, the containment event tree provides a model for creep rupture and adjusts the bin as appropriate.

Reactor Cavity / Lower Compartment Status

RCS inventory and water injected from the RWST into containment is collected first in the recirculation sump and then on the lower compartment floor. For some accident sequences, sufficient water can be discharged from the containments sprays and/or the safety injection system to flood the reactor cavity. The presence of water in the reactor cavity and the lower compartment can have an impact on the progression of the accident sequence once the RPV is failed and on important containment phenomena.

The presence of water and the ability to continuously flood the rector cavity is important with respect to debris coolability, containment pressurization rate, and radionuclide scrubbing in the post-RPV failure time frame. If water can be supplied to the cavity, the heat generated from the debris can be carried from the cavity as steam and transferred to the fan coolers or other heat removal mechanisms. In addition, any radionuclides which are released from the debris are scrubbed by the overlying pool of water and the resultant source term in the containment is reduced.

Based on analysis, if the RWST inventory is injected into the containment, there is sufficient inventory to almost assure that a pool of water will cover the debris for a wide range of temperatures and pressures and accounting for steam generation inside containment. Thus if safety injection or the containment sprays function, the cavity is expected to be flooded. The identified outcomes for this parameter can be collapsed to only two cases:

- wet (i.e., RWST emptied into containment)
- dry (i.e., minimal water present in the cavity and lower compartment)

Availability of Heat Removal Using the Steam Generators

For sequences in which RCS makeup is required but not available, i.e., LOCA sequences without injection, the ability to cool the RCS using the steam generators may be important. Success of steam generator cooling during a small break LOCA without injection will increase the time until core damage. If secondary-side heat removal is available the decay heat is removed and RCS pressure decreases which reduces the rate of inventory loss from the RCS and extends the time until the core is uncovered. Larger break sizes empty the hot legs and short circuit natural circulation which renders secondary-side heat removal ineffectual.

In addition to affecting core damage timing, the presence of water in the steam generators also has an influence on radionuclide releases. As radionuclides are released from the fuel into the RCS it is possible that they will pass through the steam generator tubes. If steam generator cooling is present the radionuclides will be cooled and a fraction of the radionuclides will plate out on the internal walls of the steam generator tubes. This will reduce the quantity of radionuclides released to the containment and thereby reduce the containment source term.

The effectiveness of this plateout mechanism is somewhat dependent on the size of the break. The larger the break, the less time that radionuclides will be held up in the RCS. Conversely, small LOCAs will maintain circulation in the RCS longer and more radionuclides plate out will occur. Prior analyses for similar designs indicate that break

ranges in excess of 5 inches will not be affected by secondary-side heat removal availability.

Another radionuclide issue affected by the availability of secondary-side heat removal is the source term following a SGTR which results in core damage. If the tube break is covered by water the release will be scrubbed and the resultant release to the environment reduced. SHNPP procedures require that, unless the ruptured steam generator is also faulted (i.e., open relief valve), a water level be maintained within the ruptured steam generator sufficient to cover the tubes. The water present in the steam generator will allow for radionuclide scrubbing.

Two states are provided for this core damage bin parameter:

- steam generator cooling functioning
- steam generator cooling failed

Timing of RPV Failure

Timing is important to both the containment analysis and in assessing public risk. The timing of RPV failure impacts the containment response and the potential for mitigation. For example, if a large LOCA accident sequence occurs with no injection, the timing of core damage and vessel failure will be much earlier than a small LOCA accident sequence in which injection has been successful and a failure occurs at the point of recirculation.

The time available between the start of the event and RPV failure provides an estimate of the time available for public evacuation. The onset of core damage or the failure of the RPV may be the point at which public evacuation is initiated.

Timing of evacuation is a major input to consequence analysis since it, in combination with the rate of evacuation, determines the fraction of the public removed from the impacted area and not exposed to any release plume. Based on accident analyses performed on similar designs, two time regimes have been shown to be important:

- early RPV failure (< 4 hours)
- late RPV failure (> 4 hours)

The two regimes can be considered to reflect the success or failure of safety injection and secondary-side heat removal which can be directly related to the accident sequence initiator.

Development of Core Damage States from Bin Parameters

The defined bin parameters and their associated states represent the important parameters for use in accident sequence binning. However, not all of the possible combinations of these parameters are logically possible. Therefore, it is important to examine the potential bin combinations and to eliminate those which are not physically possible.

Based on the possible combinations of the various bin parameters, a total of 96 possible core damage bins can be generated. Certain conditions can be applied to these potential bins which eliminate illogical core damage bins. For the SHNPP PRA, the combinations contained in Table 4–7 are considered to be illogical.

Table 4–7
Examples of Illogical Core Damage Bin Parameter Combinations

Exclusion Condition	Example Core Damage Bin
High RCS pressure and large LOCA condition	Large LOCA leakage rate and high RCS pressure
Cycling relief valve and low pressure condition	Cycling relief valve leakage rate and low RCS pressure
Steam generator cooling and RCS relief valve challenge	Cycling relief valve leakage rate and Successful steam generator cooling
If no cooling available, then the sequence cannot be considered late.	Transient without SSHR or safety injection.
High RCS pressure and medium LOCA release rate.	Medium LOCA and high RCS pressure
Medium or low RCS pressure and a cycling relief valve	Transient with low RCS pressure

Using these exclusion rules, the impossible combinations are eliminated and the plausible core damage bins determined. This process resulted in the generation of 17 potential core damage bins. These bins are presented in Table 4–8.

Table 4–8
Core Damage Bin Definitions

Bin	SG[1] cooling available	RCS leakage rate[2]	Timing of RPV failure	RCS pressure at RPV failure	Cavity status at RPV failure
1	No	SL	Early	High	Dry
2	No	CR	Early	High	Dry
3	Yes	SL	Late	High	Dry
4	No	SL	Early	High	Wet
5	Yes	SL	Late	High	Wet
6	No	SL	Late	High	Wet
7	N/A[3]	LL	Early	Low	Dry
8	Yes	SL	Late	Low	Dry
9	N/A[3]	ML	Early	Medium	Dry
10	Yes	SL	Late	Low	Dry
11	N/A[3]	LL	Early	Low	Dry
12	N/A[3]	ML	Early	Medium	Wet
13	N/A[3]	SL	Late	Low	Wet
14	No	CR	Late	High	Dry
15	N/A[3]	SL	Early	Medium	Wet
16	Yes	SL	Late	Medium	Wet
17	Yes	SL	Late	Medium	Wet

1 SG – steam generator
2 CR – cycling relief rate, SL – small LOCA relief rate, ML – medium LOCA relief rate, LL – large LOCA relief rate.
3 N/A – not applicable

4.3.2 Core Damage Functional Sequence Binning

The bin parameters developed in Table 4–8 were applied to the event tree end states to identify what sequences comprised each bin. The results of this process are summarized in Tables 4–9 through 4–16.

Table 4–9
Transient Event Tree Core Damage Bins

Accident Sequence	Core Damage Bin	RCS leakage rate	Timing of RV failure	SSHR available	RCS pressure at RV failure	Cavity status at RV failure
TBSX	6	Small LOCA	Late	No	High	Wet
TBX	4	Small LOCA	Early	No	High	Wet
TBU	2	Cycling safety valve	Early	No	High	Dry
TBUW	1	Small LOCA	Early	No	High	Dry
TBH	2	Cycling safety valve	Early	No	High	Dry
TBHW	4	Small LOCA	Early	No	High	Wet

Table 4–10
S1 LOCA Event Tree Core Damage Bins

Accident Sequence	Core Damage Bin	RCS leakage rate	Timing of RV failure	SSHR available	RCS pressure at RV failure	Cavity status at RV failure
S1BH	4	Small LOCA	Early	No	High	Wet
S1BSX	6	Small LOCA	Late	No	High	Wet
S1BX	4	Small LOCA	Early	No	High	Wet
S1GXD	17	Small LOCA	Late	Yes	Medium	Wet
S1PSX	5	Small LOCA	Late	Yes	High	Wet
S1PX	5	Small LOCA	Late	Yes	High	Wet
S1UB	1	Small LOCA	Late	No	High	Dry
S1UGL	10	Small LOCA	Late	Yes	Low	Dry
S1UGSD	8	Small LOCA	Late	Yes	Low	Wet
S1UGD	8	Small LOCA	Late	Yes	Low	Wet
S1UP	3	Small LOCA	Late	Yes	High	Dry

Table 4–11
S2 LOCA Event Tree Core Damage Bins

Accident Sequence	Core Damage Bin	RCS leakage rate	Timing of RV failure	SSHR available	RCS pressure at RV failure	Cavity status at RV failure
S2U	15	Small LOCA	Early	N/A	Medium	Dry
S2X	13	Small LOCA	Late	N/A	Low	Wet

Table 4–12
Medium LOCA Event Tree Core Damage Bins

Accident Sequence	Core Damage Bin	RCS leakage rate	Timing of RV failure	SSHR available	RCS pressure at RV failure	Cavity status at RV failure
MU	9	Medium LOCA	Early	N/A	Low	Dry
MX	12	Medium LOCA	Early	N/A	Low	Wet

Table 4–13
Large LOCA Event Tree Core Damage Bins

Accident Sequence	Core Damage Bin	RCS leakage rate	Timing of RV failure	SSHR available	RCS pressure at RV failure	Cavity status at RV failure
AU	7	Large LOCA	Early	N/A	Low	Dry
AX	11	Large LOCA	Early	N/A	Low	Wet

Table 4–14
Steam Generator Tube Rupture Event Tree Core Damage Bins

Accident Sequence	Core Damage Bin	RCS leakage rate	Timing of RV failure	SSHR available	RCS pressure at RV failure	Cavity status at RV failure
RBH	B[1]–1	Small LOCA	Late	No	High	Dry
RBX	B–6	Small LOCA	Late	No	High	Wet
RBWY	X[2]–1	Small LOCA	Late	No	High	Dry
RBSX	B–6	Small LOCA	Late	No	High	Wet
RPY	B–3	Small LOCA	Late	Yes	High	Dry
RUB	B–1	Small LOCA	Late	No	High	Dry
RUBW	X–1	Small LOCA	Late	No	High	Dry
RUG	B–16	Small LOCA	Late	Yes	Medium	Dry
RUP	B–3	Small LOCA	Late	Yes	High	Dry
RUW	X–16	Small LOCA	Late	Yes	Medium	Dry
RWGY	X–16	Small LOCA	Late	Yes	Medium	Dry
RWPY	X–16	Small LOCA	Late	Yes	Medium	Dry

[1] B – Containment bypassed, isolated steam generator
[2] X – Containment bypassed, unisolated steam generator

Table 4–15
ATWS Event Tree Core Damage Bins

Accident Sequence	Core Damage Bin	RCS leakage rate	Timing of RV failure	SSHR available	RCS pressure at RV failure	Cavity status at RV failure
ATWS1[1]	14[2]	Cycling Safety Relief	Late	Yes	High	Wet
ATWS2	3	Small LOCA	Late	Yes	High	Dry
ATWS3	7	Large LOCA	Early	N/A	Low	Dry
ATWS4	7	Large LOCA	Early	N/A	Low	Dry
ATWS5	14	Cycling Safety Relief	Early	No	High	Dry
ATWS6	3	Small LOCA	Late	Yes	High	Dry
ATWS7	7	Large LOCA	Early	N/A	Low	Dry
ATWS8	7	Large LOCA	Early	N/A	Low	Dry
ATWS9	7	Large LOCA	Early	N/A	Low	Dry
ATWS10	7	Large LOCA	Early	N/A	Low	Dry

1 Due to the long character strings involved, the ATWS core damage sequences are listed numerically from the top of the event tree to the bottom (i.e., ATWS3 defines sequence IEMT1MATP).

2 This sequence is does not result in core damage. The reactor, however, is not shutdown and the plant is not considered to be in a safe, stable state. Therefore, this sequence is assumed to result in a long term core damage sequence.

Interfacing Systems LOCA

ISLOCA sequences result in a loss of RCS inventory outside the containment. As a result, the cavity condition will be dry at RPV failure. The dominant sequences are associated with the residual heat removal system. The timing is early and the leakage rate is large, thus, the RCS pressure is expected to be low. ISLOCA sequences are placed into core damage bin 7.

Internal Flooding

The internal flooding sequences are of a similar nature as the transient event sequences discussed previously. The dominant flooding sequence deals with the failure of the service water system which results in a loss of component heat sinks and eventual core damage. This sequence is placed into core damage bin 1.

Containment Safeguards Event Tree Model

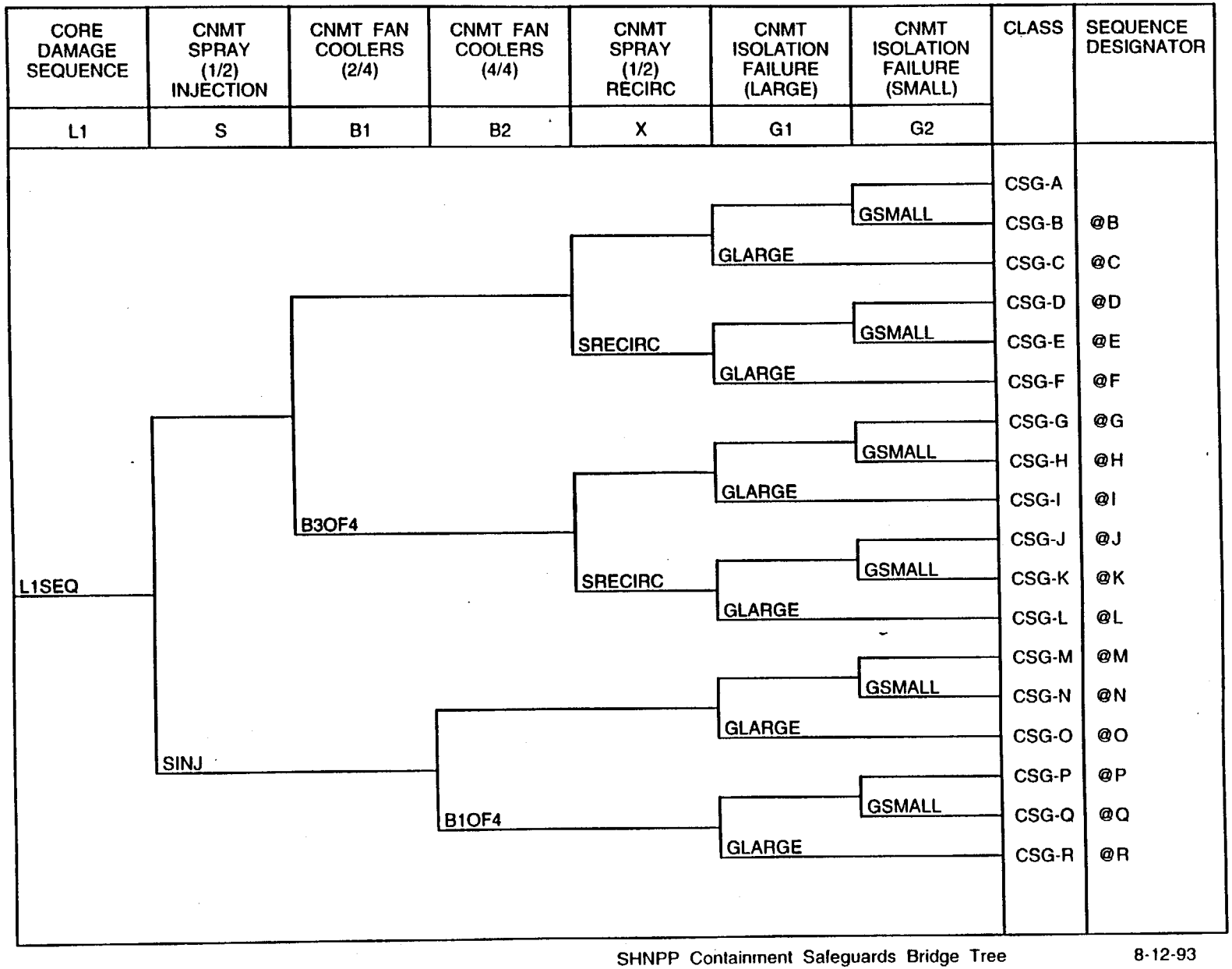
The containment safeguards event tree (CSET) is comprised of six top events (Figure 4–16) which result in a total of 18 possible end states. The CSET is different from the core damage event trees since it does not have end states that represent the termination of the event. The end states of the CSET represent the possible states of the systems associated with the containment which are of interest in the PRA. The status of the containment systems is important in determining containment pressure challenges, source term composition, and other physical parameters associated with the level 2 PRA.

CSET Top Events

CSS – Containment Sprays Function (Injection)

This event addresses the status of the containment sprays during the injection phase of the event. The availability of containment sprays is important in determining whether RWST water was transferred to the containment, the initial containment pressurization prior to vessel failure, and the potential for the reactor cavity being flooded at the time of vessel failure.

Figure 4–16
Containment Safeguards Event Tree



SHNPP Containment Safeguards Bridge Tree 8-12-93

The operators are instructed to terminate containment sprays as soon as the containment pressure is less than 10 psig. In addressing the containment aspect, however, it is important to determine whether the containment sprays are capable of operation. As the accident progresses, the operators will need to re-initiate containment spray to address source term scrubbing and to provide additional containment heat removal. Successful containment heat removal is dependent on the number of fan coolers and containment spray trains functioning and both system success criteria are dependent on the other system. Thus, the success or failure of the containment spray trains impacts the success criterion for the containment fan cooler system. The exact relationship is described in the respective fan cooler top events. The model includes the failure of the containment spray system due to system or support system failures. The success criterion for this event is that one of two containment spray trains function.

CFC1 – Containment Fan Coolers (2/4 Required)

If the containment sprays succeed, the number of fan coolers needed to maintain successful containment cooling is reduced. The success of the fan coolers is defined as both the successful operation of the fan and adequate cooling water to the fan coolers. As such, one dependency which impacts this event is the status of the service water system which is propagated, as needed, through the core damage model. The number of fan coolers needed presented in Table 4–16.

Table 4–16
Containment Heat Removal Success Criteria

Number of Spray Trains Functioning	Number of Fan Coolers Required
2/2	None
1/2	2
0/2	4

For this event, the containment sprays are assumed to be operating. Since the success criterion for the containment sprays is that at least one of two trains function, it is not possible to decide how many trains of containment sprays are functioning. Conservatively, and to keep the model somewhat simple, the assumption is made that one of two sprays are running. Therefore, two containment fan coolers are needed and the event logic reflects this.

CFC3 – Containment Fan Coolers (4/4 Required)

This event is similar to the prior event with the exception that all fan coolers are needed. This event in used for cases in which both containment spray trains are failed. If containment sprays are not available, all fan coolers will be needed to provide adequate heat removal.

CSSR – Containment Sprays Available (Recirculation)

The containment sprays may be operated in recirculation mode if they are aligned to the containment sumps. This provides a long term containment spray mechanism. The event addresses the failure of the containment sprays during recirculation and includes the faults associated with injection and recirculation in order to account for the failure of one train in injection and the other train failure during recirculation.

The ability of the containment sprays to function during recirculation is directly related to the success of containment sprays during injection. Therefore, the recirculation event is not asked if containment sprays failed in injection.

CIL – Large Containment Isolation Failure

This event addresses the status of the containment isolation system and its ability to maintain adequate containment integrity. It addresses the isolation system function and does not include any overpressure conditions due to loading. The particular event addresses those isolation failures greater than 4 inches in diameter. The distinction is made between isolation failures greater and less than 4 inches to account for changes in the ability to pressurize containment and radionuclide holdup time. If a large isolation failure is present, the relief capacity of the leak should be sufficient to preclude significant pressurization. The retention time of the radionuclides may be small in the containment and may result in an increase in the source term.

The model addresses the failure of the equipment hatch and other mechanical penetrations. An event for ISLOCAs is provided in the model to address these sequences. SGTR events with a stuck open secondary valve are also classified as a large isolation failure due to similar source terms.

CIS – Small Containment Isolation Failure

The last event in the CSET addresses the possibility of a small isolation failure.

The model identifies the important small isolation lines (less than 4 in.). SGTR events without a stuck open steam generator SRV are also included under this event. A SGTR event may result in a small LOCA leakage rate which bypasses containment.

Containment Safeguards Event Tree Sequences

The 16 end states represent different configurations of the containment safeguards systems. An alphabetic code is used to distinguish each end state and the end states are described in Table 4–17.

4.3.3 Plant Damage State Frequencies and Dominant Sequences

With the development of the CSET states and the core damage bins, the plant damage states can be developed. The plant damage state represents the combination of the two parameters. Given 17 different core damage bins and 18 different containment safeguards event tree end states, a total of 306 plant damage states are possible. In order to manage this process, the PDSs were screened based on a truncation value of 1.0×10^{-7}/year. This value was chosen for two reasons. First, other events which are not addressed by the PRA begin to contribute (meteorites, etc.) and the insights to be gained are limited by whose events not addressed. Second, this value is nearly three orders of magnitude below the core damage frequency. Even considering uncertainties it is doubtful that sequences which occur at this frequency will significantly impact the overall results and findings.

In an approach similar to the level 1 model, the CSET is solved using a direct fault tree linking approach. The different core damage sequences are combined with the solution of each CSET end state cutsets to develop the plant damage state cutsets and frequency.

The results of the CSET solution yields a total of 17 plant damage states which exceed the truncation value. The plant damage states define both probabilistic and deterministic criteria which are needed to solve the CET. Although each plant damage state represents many different combinations of failures, the basic accident progression is similar and one or two sequences can be selected to represent all sequences. Table 4–18 provides the plant damage states which have frequencies above the cutoff value, and the associated representative sequence. The representative sequence is chosen based on its contribution to the core damage bin frequency, severity of containment challenge, and the potential for significant release.

Table 4–17
Containment Safeguards Event Tree End States

Containment Safeguards Event Tree End State	Description
A	Successful isolation, sprays and fans functioning
B	Small isolation failure, sprays and fans functioning
C	Large isolation failure, sprays and fans functioning
D	Successful isolation, sprays fail in recirculation and fans functioning
E	Small isolation failure, sprays fail in recirculation and fans functioning
F	Large isolation failure, sprays fail in recirculation and fans functioning
G	Successful isolation, sprays fail in recirculation and fans fail
H	Small isolation failure, sprays functioning and fans failed
I	Large isolation failure, sprays functioning and fans failed
J	Successful isolation, sprays fail in recirculation and fans fail
K	Small isolation failure, sprays fail in recirculation and fans fail
L	Large isolation failure, sprays fail in recirculation and fans fail
M	Successful isolation, sprays fail in injection and fans functioning
N	Small isolation failure, sprays fail in injection and fans functioning

Table 4–17 (cont.)
Containment Safeguards Event Tree End States

Containment Safeguards Event Tree End State	Description
O	Large isolation failure, sprays fail in injection and fans functioning
P	Successful isolation, sprays fail in injection and fans fail
Q	Small isolation failure, sprays fail in injection and fans fail
R	Large isolation failure, sprays fail in injection and fans fail

Table 4–18
Plant Damage State Results

Plant Damage State	Frequency (/year)	Representative Sequence
1P	2.3×10^{-5}	Loss of offsite power, failure of AFW after battery depletion, RCP seal LOCA at 1.5 hours, no containment safeguards, containment isolated
15A	1.2×10^{-5}	S2 LOCA with failure of safety injection, all containment safeguards functioning, containment isolated
13A	1.2×10^{-5}	S2 LOCA with failure of recirculation, all containment safeguards functioning, containment isolated
2A	7.1×10^{-6}	Loss of feedwater, failure of AFW, and operator fails to accomplish feed-and-bleed cooling, all containment safeguards functioning, containment isolated
7A	6.6×10^{-6}	ATWS with overpressurization due to insufficient moderator feedback, all containment safeguards functioning, containment isolated
12A	2.3×10^{-6}	Medium LOCA with failure of recirculation, all containment safeguards functioning, containment isolated
16B	1.9×10^{-6}	Large LOCA with failure of recirculation, all containment safeguards functioning, small containment isolation failure
9A	1.2×10^{-6}	Medium LOCA with failure of low pressure injection, all containment safeguards functioning, containment isolated
2D	9.8×10^{-7}	Reactor trip with a total loss of instrumentation busses, containment sprays fail at recirculation, containment isolated
2G	9.7×10^{-7}	Same as 2A with the insufficient containment heat removal
7C	5.0×10^{-7}	ISLOCA in the RHR system, large containment bypass, containment safeguards functioning
5A	2.1×10^{-7}	S1 LOCA with failure of recirculation, all containment safeguards functioning, containment isolated
3B	1.5×10^{-7}	SGTR with failure of RCS depressurization and loss of safety injection source, small containment bypass, all containment safeguards functioning
15B	1.4×10^{-7}	Same as 15A with an RHR relief valve stuck open, resulting in a small containment isolation failure

Table 4–18 (cont.)
Plant Damage State Results

Plant Damage State	Frequency (/year)	Representative Sequence
10A	1.3×10^{-7}	S1 LOCA with failure of injection, successful RCS cooldown/depressurization and failure of low pressure injection, all containment safeguards functioning, containment isolated
3A	1.2×10^{-7}	Excessive LOCA, all containment safeguards functioning, containment isolated

4.4 Containment Failure Characterization

4.4.1 Containment Failure Curve Development

A containment failure assessment was performed for SHNPP. It identified four different containment failure modes which are possible for the SHNPP containment. Other possible leakage paths were identified for the hatches and penetrations but do not result in more limiting releases and are not addressed.

The analysis provided both the median capacity for each failure and the associated uncertainty for temperatures of 300°F, 500°F, and 800°F. The four failure modes and their median failure pressure are presented in Table 4–19. Review of the capacities indicates that the failures identified are only weakly linked to temperature, with the capacity somewhat reduced at higher temperatures. Based on the analysis, the decreased capacity is not significant and the 300°F assessment can be used in the analysis.

Table 4–19
Postulated Containment Failures

Containment Failure Mode	Median Capacity (psig)
Basemat Shear	153
Wall-Basemat Junction Shear	205
Cylinder Membrane	210
Dome Membrane	284

As shown in the table, failure of the basemat of the containment (at the outer radius of the mat where it joins the containment wall) is the most likely failure mode. This failure mode, therefore, is adopted as the characteristic response for the SHNPP containment. A discussion of this location is provided in Section 4.1. Using the uncertainties provided in the report, the log normal distribution of capacities can be plotted. Figures for each failure mode are provided as Figures 4–17 through 4–20. The probability of containment failure for a given failure mode and pressure can be determined by reading the cumulative density value for a particular containment pressure.

With the individual curves identified, the curves were convoluted to form a bounding curve. This curve bounds all points such that it can be used in place of the individual curves and greatly simplify the analysis. As can be seen from the individual failure mode curves, the limiting faiure mode is basemat shear. Therefore, for the level 2 analysis, this is the failure curve which was used.

Figure 4–17
Containment Failure Capacity for Basemat Shear

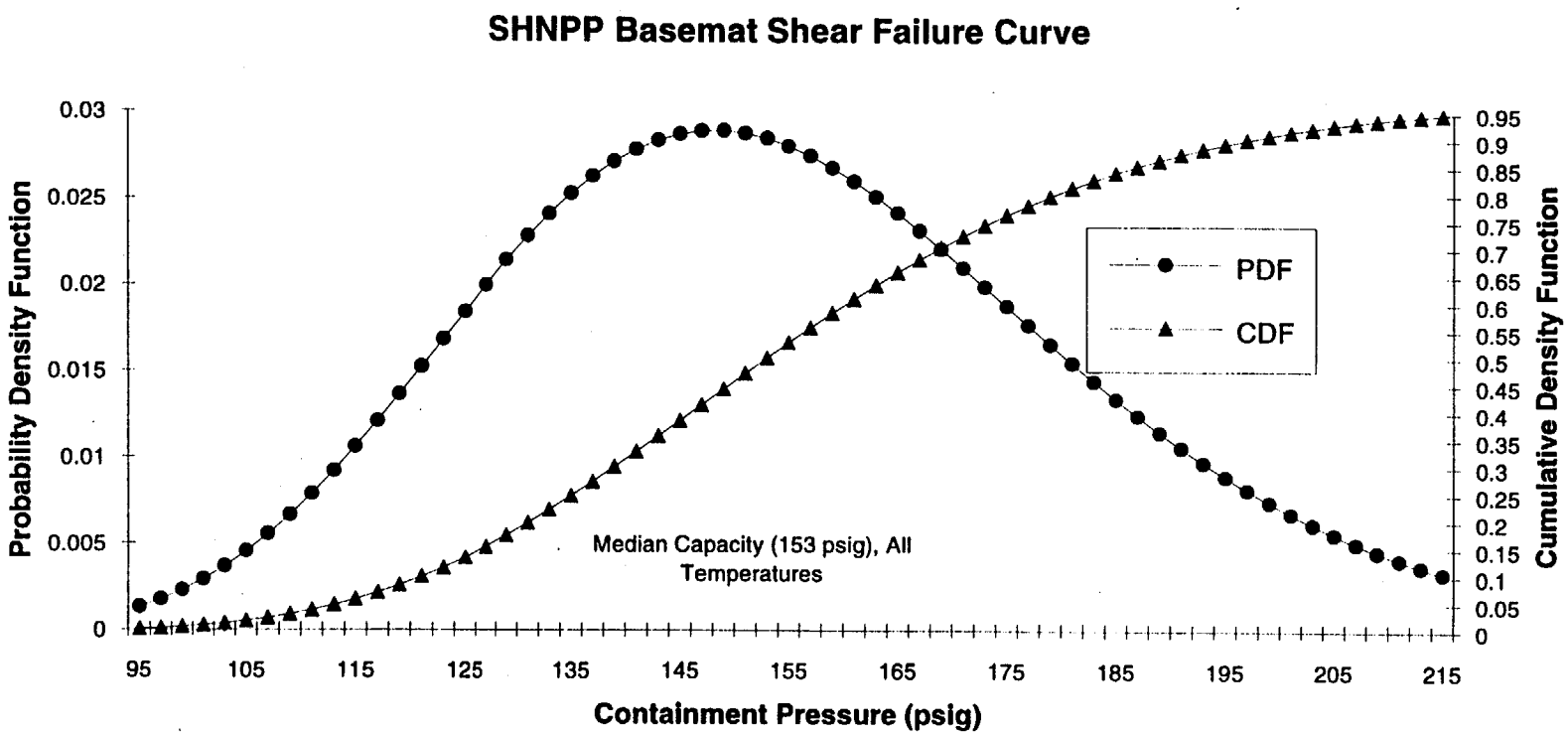


Figure 4–18
Containment Failure Capacity for Wall-Basemat Junction Shear

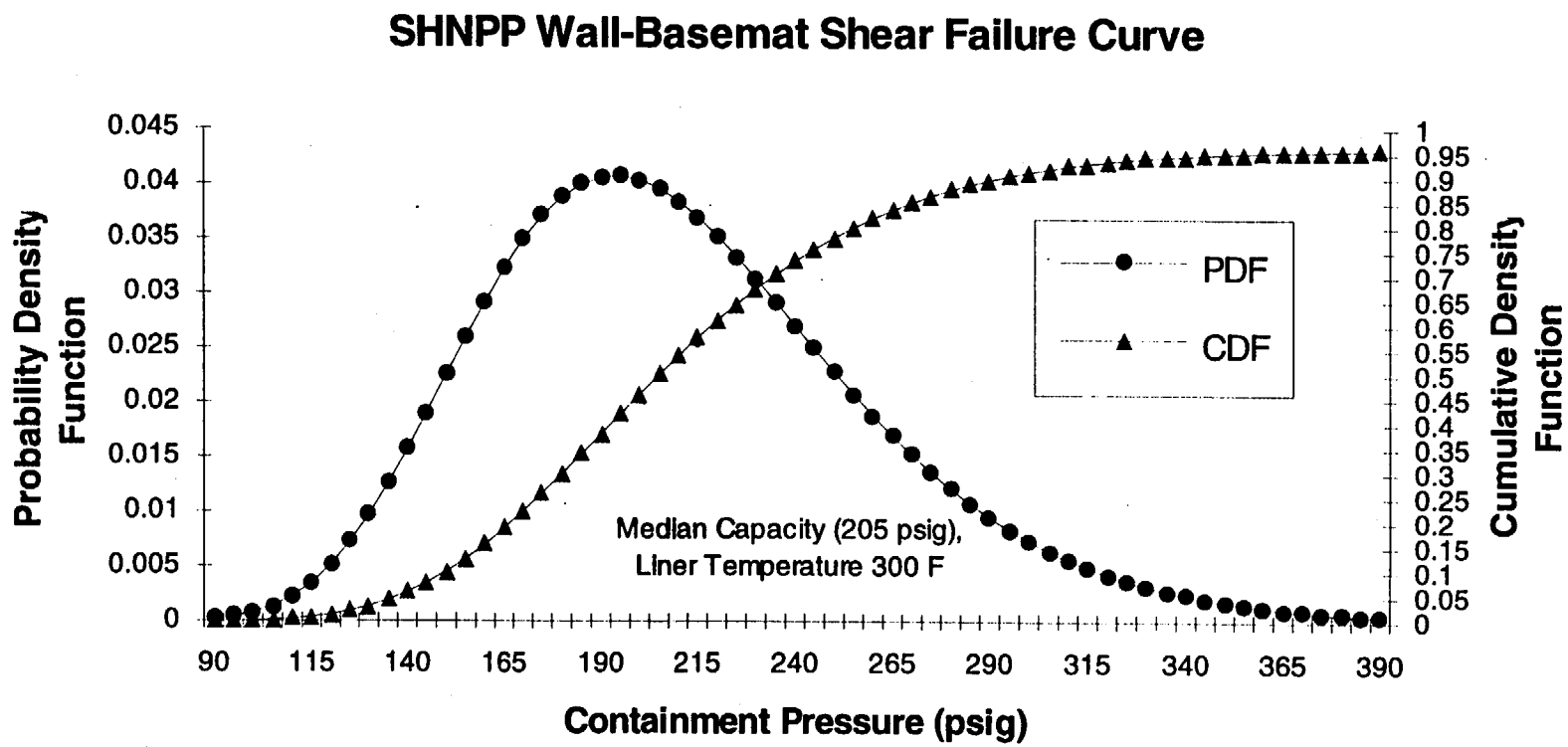


Figure 4–19
Containment Failure Capacity for Cylinder Membrane Failure

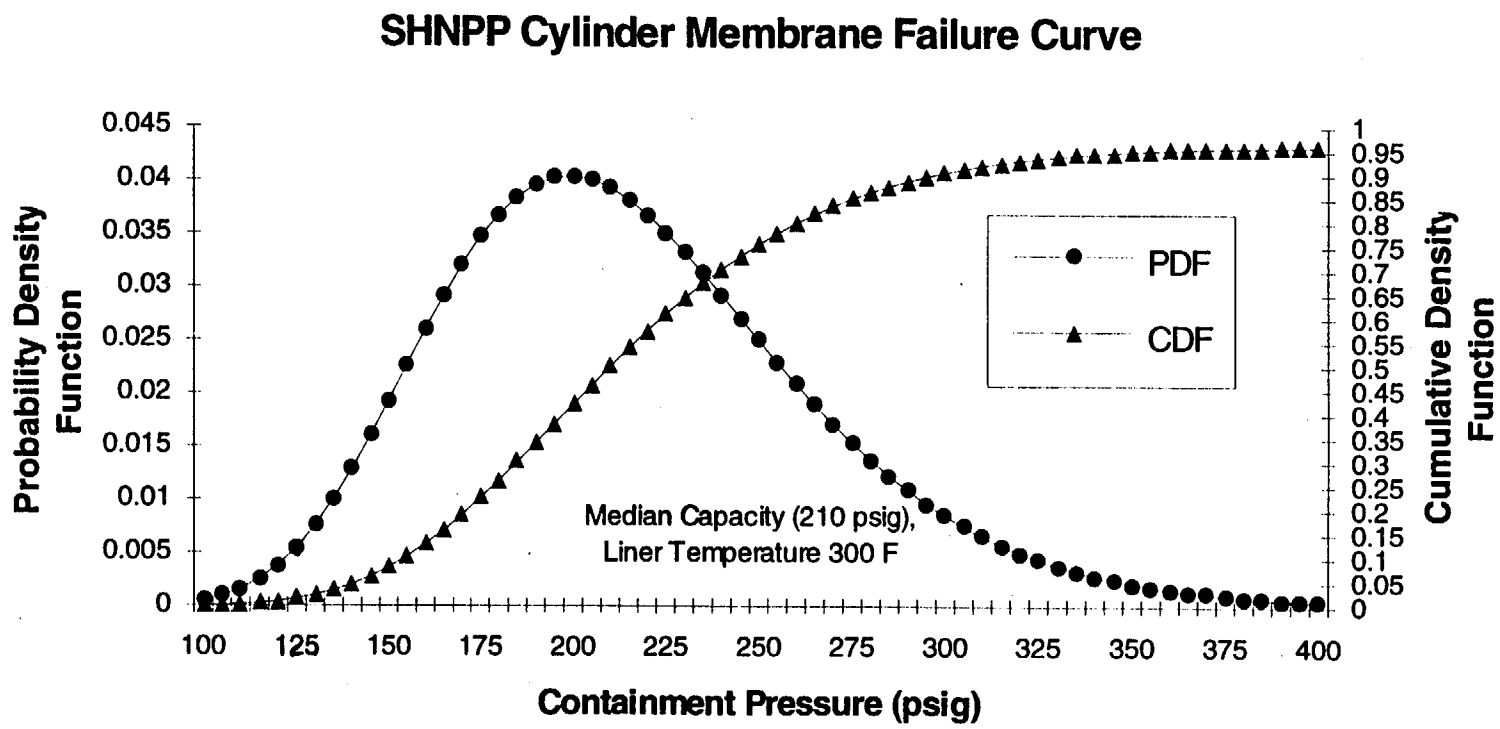
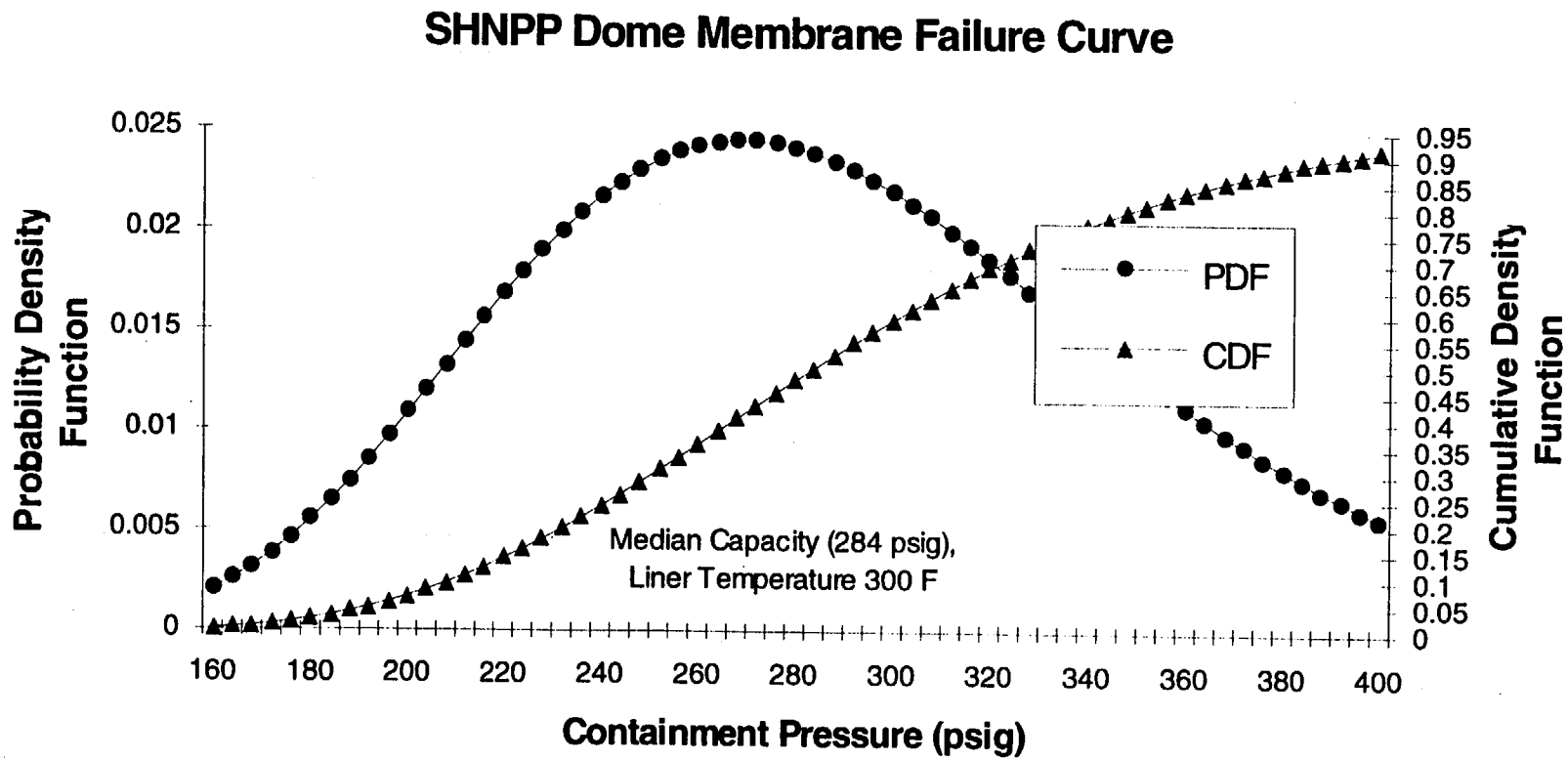


Figure 4–20
Containment Failure Capacity for Dome Membrane Failure



4.5 Containment Event Tree

4.5.1 Development Approach

This section provides a description and basis for the SHNPP CET. The overall structure of the CET is described along with the top events that comprise the important containment issues. The supporting logic used to customize the CET for specific plant damage states is also described. The CET is supported by information contained in Section 4.2. The quantification of the events that comprise the CET and the CET quantification results are found in Section 4.6.

The CET (Figure 4–21) is composed of 15 top events that are associated with key phenomenological inputs to the containment assessment. These events can be grouped into seven major categories described below.

In-vessel Recovery: The potential exists for restoration of core cooling prior to vessel failure. This type of sequence is best identified with the Three Mile Island Unit 2 event. Some core damage and relocation occur along with some radionuclide release into the containment. Hydrogen is generated and the potential exists for a hydrogen burn. Issues related to core concrete interaction and high pressure melt ejection, however, are precluded.

Containment Isolation Status: This category examines the potential for an impaired containment prior to significant core damage. Top events include the potential for either containment isolation failure or sequences which result in containment bypass, e.g., SGTR.

Early Containment Challenges: A major challenge to the containment occurs if the RPV is breached and the debris is discharged to the containment. This category addresses changes that occur at or near the time of RPV failure. The top events address issues related to high pressure melt ejection/direct containment heating, early hydrogen burning or detonation, rapid steam generation, and containment liner attack.

Ex-Vessel Debris Coolability: A major uncertainty related to ex-vessel debris behavior involves the potential for the debris to be cooled once it is outside the RPV and located in the containment. The potential for dispersion, location of the debris along with the presence or absence of water in the containment all are considered by the model to arrive at whether the debris is coolable.

Figure 4–21
SHNPP Containment Event Tree (page 1 of 3)

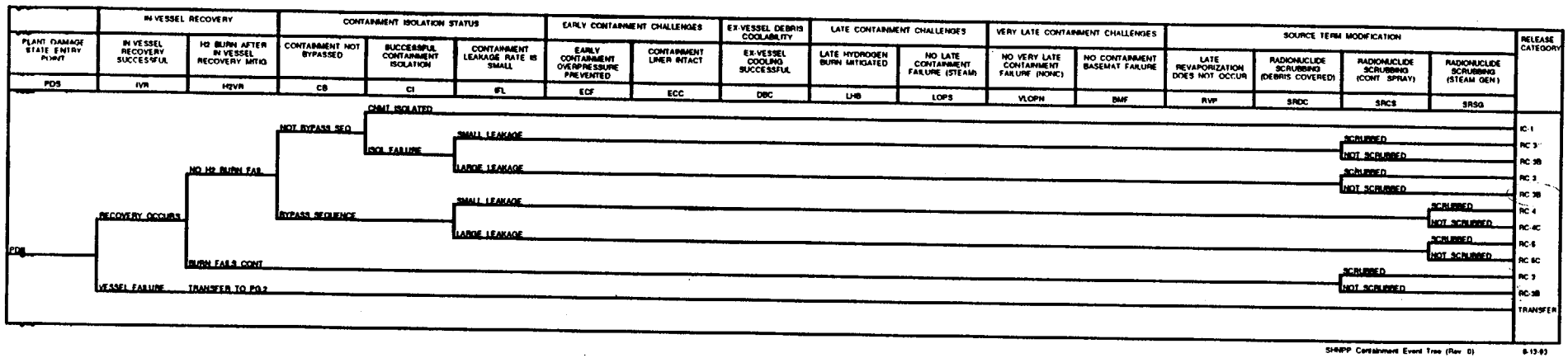


Figure 4–21
SHNPP Containment Event Tree (page 2 of 3)

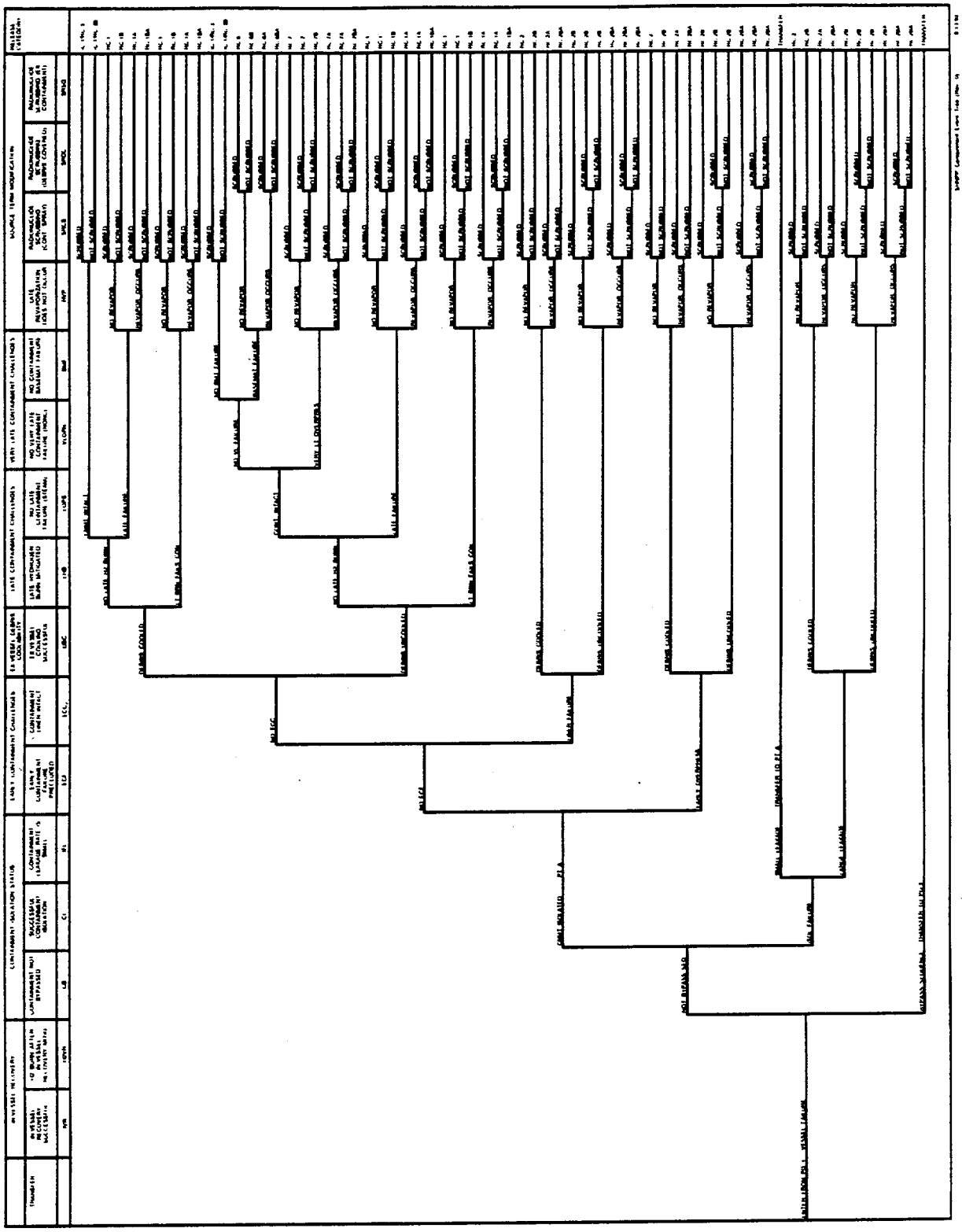
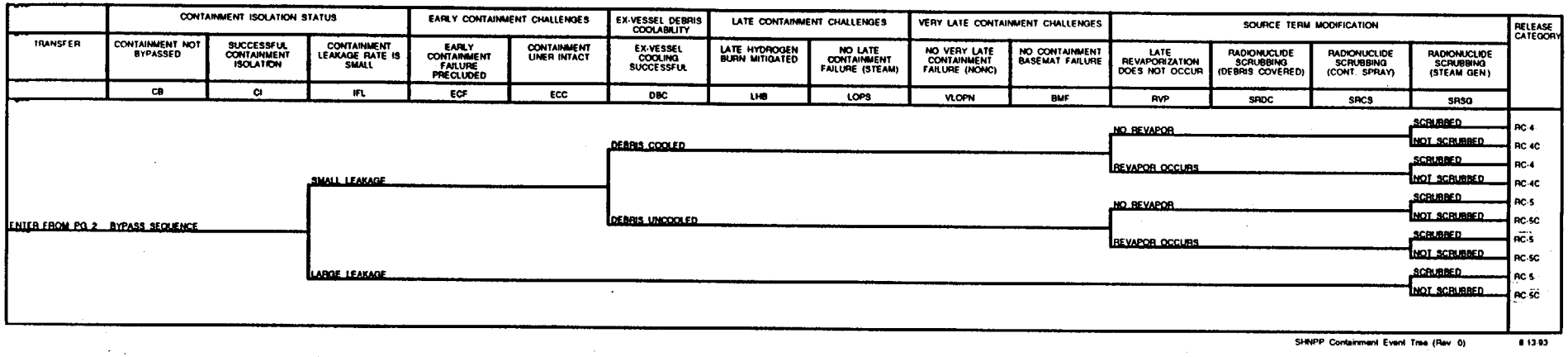


Figure 4–21
SHNPP Containment Event Tree (page 3 of 3)



Late Containment Challenges: If the containment withstands the initial containment challenges, a series of longer term events may result in containment failure. These events address the gradual buildup of steam or hydrogen which result in the potential for containment overpressure due to either steam generation or late hydrogen burn. The potentials for burns following recovery are considered.

Very Late Containment Challenges: Depending on the accident sequence and the success of debris cooling, two containment failure modes can be postulated which may occur in a very slow manner such that containment failure would occur 48 hours or more following the initial transient. If the debris is not coolable core/concrete interaction can ablate the containment basemat and eventually result in a failure of the containment boundary. The concrete ablation process also generates energy and additional gases, mostly steam, which can increase the containment pressure such that failure occurs. These potential failure modes are addressed by the top events comprising this category.

Source Term Modification: The radionuclide source term can be altered by several containment phenomena. Two major issues deal with the potential for late revaporization and the ability to cover the debris with water that can scrub a large portion of the radionuclides and substantially lower any postulated release from the containment. The potential for high pressure sequences to lead to a late revaporization of radionuclides is addressed along with the potential for radionuclides to be scrubbed prior release. The scrubbing consideration includes overlying pools of water, containment sprays, and ex-containment scrubbing for sequences involving SGTR.

When the potential combinations of the top events are considered, 86 end states are defined which represent somewhat unique containment end states. To simplify the analysis, important criteria are defined and the CET end states binned into release categories. Section 4.7 describes this process. The CET top events are described below.

4.5.2 Containment Event Tree Top Events

IVR – In-Vessel Recovery

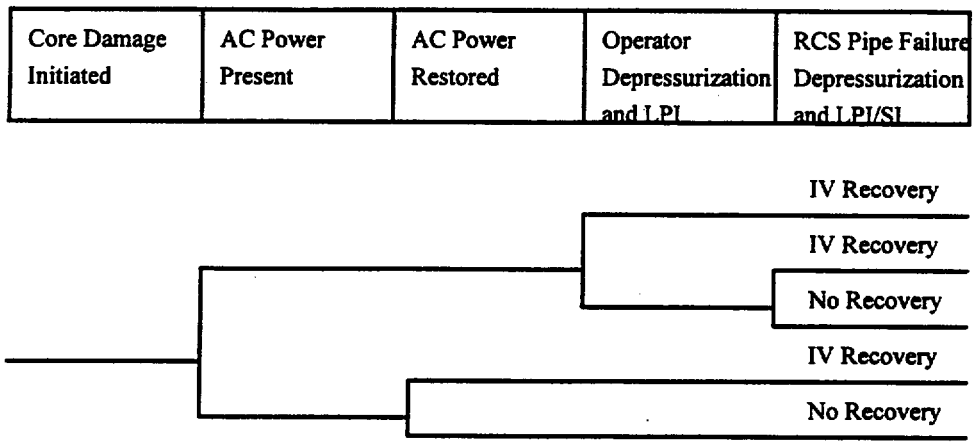
This event addresses the potential for arresting the core damage progression within the RPV. This type of late restoration is best illustrated by the event at TMI. In this case the reflood of the RPV late in the event resulted in the core debris quenching and no RPV breach. The restoration of cooling in the RPV does not preclude some containment challenges, such as a hydrogen burn, but it does significantly reduce the potential for a large release and eliminates concerns related to ex-vessel debris cooling. Sequences ending with the damaged core still within the RPV are of relatively small consequence

and the consideration for recovery is important in determining the potential for containment releases.

The time frame for this event is between the time at which the core becomes uncovered and the time that RPV failure is predicted. Using this definition, fast-acting sequences, such as a large LOCA without injection, will provide little time for in-vessel recovery. In contrast, loss of feedwater events which result in a slow heatup of the RCS provide considerable time for recovery.

The top logic developed for this event addresses three possible means by which in-vessel recovery could occur. The logic for the success of each is illustrated in Figure 4–22. All of the alternatives are based on the RCS maintaining a high pressure prior to in-vessel recovery. This is indicative of a slow acting event such as a loss of feedwater or S1 LOCA.

Figure 4–22
In-Vessel Recovery Path Evaluation



The first situation involves a station blackout sequence which has continued until the core has uncovered. An RCP seal LOCA may be present at that time and the RCS pressure may still be within the high pressure range. Based on the plant damage state contributions, the dominant sequence for station blackout deals with the start failure of the diesel generators with a subsequent failure of the AFW turbine-driven pump. The loss of all AC power results in a loss of all safeguards equipment and precludes operator actions to restore RCS inventory and cooling.

The start failures are dominant because run failures are predicted to occur later in the event which provides considerable time for offsite power restoration. In considering the potential for offsite power restoration to preclude core damage, only the time until core uncovery was considered. This event carries the process further and examines the time

from core uncovery until RPV failure is predicted. Based on MAAP analyses for the potential RCP seal LOCA sizes (Table 4.20) the available time between core uncovery and RPV failure are predicted.

Table 4–20
MAAP Predicted Event Timing

RCP Seal LOCA Size	Weighting Probability	Time of Core Uncovery (hr.)	Time of RPV Failure (hr.)	Available Time (hr.)
250 gpm	0.47	2.5	7.7	5.2
21 gpm	0.36	5.8	12.3	6.5
170 gpm	0.18	3.0	9.1	6.0
480 gpm	0.0045	2.0	6.6	4.6

This information is combined with the loss of offsite power event tree to develop the probability of recovery between these two events. This value is then used to determine the probability of in-vessel recovery. Given AC power is restored recovery is possible using the safety injection system to restore RCS inventory and secondary-side heat removal to restore cooling. It is also possible that the core geometry will have degraded such that recovery is not possible and that restoration of safety functions cannot preclude RPV failure. An event which represents this potential is included in the model.

If AC power is available, other occurrences can lead to the depressurization of the RCS and allow for RPV flooding to stop core melt progression.. The operators may initiate feed-and-bleed cooling at a point too late to preclude core damage but prior to the time of RPV failure. This would preclude RPV failure. Further, the operators are required by procedures to depressurize the RCS in order to allow for accumulator injection in response to a loss of heat sink. The procedures instruct the operators to open the pressurizer PORVs and reactor head vents. Based on MAAP analysis, if this action is performed in a timely manner it is possible to reduce RCS pressure to below the low pressure injection setpoint and reflood the RPV. If the debris is in a coolable geometry, the core melt progression is arrested and debris material is not expelled to the containment. The model accounts for this possibility by including a model for late RCS depressurization by the operators and operation of the low pressure injection system.

Thermal-hydraulic analyses performed on similar plants have identified the potential for thermally-induced ruptures of the RCS. Some analyses have predicted that the temperature of RCS hot legs and the surge line will exceed the temperatures necessary to cause piping failure which will result in the RCS pressure dropping to a near that of the containment. Two failure points are considered based on the accident sequence. If a pressurizer PORV or SRV is stuck open, much of the energy in the RCS will be relieved through the failed valve. This will allow for heating of the pressurizer surge line and the

potential for surge line failure. If the valves cycle, much less energy will be transferred and additional heating of the hotleg may occur. These two different cases are examined.

For sequences with AC power and low pressure injection available the potential for in-vessel recovery is considered. High pressure sequences, such as those involving a loss of decay heat removal, provide both temperatures and pressures needed to induce piping failure. Based on the core damage sequences for SHNPP, low pressure injection is predominantly available and high pressure injection is failed. Given the late timing of reflood, a rapid, high volume, flooding of the vessel is considered important. For these reasons, only recovery using low pressure injection is considered possible. Another factor to consider in determining the potential for recovery is the configuration of the core when reflood occurs. It is possible that the core will be relocated in such a means that debris cooling is not possible even after reflood.

Uncertainties associated with in-vessel melt progression preclude a detailed assessment. The time at which the operators restore cooling, the degree of core deformation, and the effectiveness of injection systems after prolonged exposure to high temperatures are just some of the many issues involved. It is reasonable to assume that in-vessel recovery is more likely for cases where the operators intervene prior to significant core deformation than for cases which rely on thermaly-induced piping failure after significant core heatup and degredation. Two events are provided in the model to account for this distinction. The first represents a qualitative estimation of the baseline resistance to in-vessel recovery which is present for all cases. A second event is added to express the increased pessimism about in-vessel recovery for cases involving late RCS piping failures.

Figure 4–23 presents the top logic structure for this event. The top logic addresses the failures of each option. Flag events (logic switches) are provided to transfer information contained in the PDS to the CET logic. These switches, such as one for the RCS pressure, provide a means to structure the CET for a specific PDS. In addition to system failures, events which address the likelihood of pipe rupture, in-vessel geometry, and operator actions are included.

H2VR – H2 Burn Occurs After In-vessel Recovery

The event at Three Mile Island identifies the potential for hydrogen burns within the containment regardless of the ability to mitigate the event within the vessel. Reflood and quenching of a hot, partially degraded core will result in additional hydrogen generation. If sufficiently large, hydrogen burns after in-vessel recovery could result in containment failure and an early release of radionuclides which escaped the fuel during the initial heatup. This event addresses the possibility that a hydrogen burn occurs and whether the hydrogen burn is sufficiently large to induce containment failure for cases which involve in-vessel recovery.

Figure 4–23
In-Vessel Recovery (page 1 of 5)

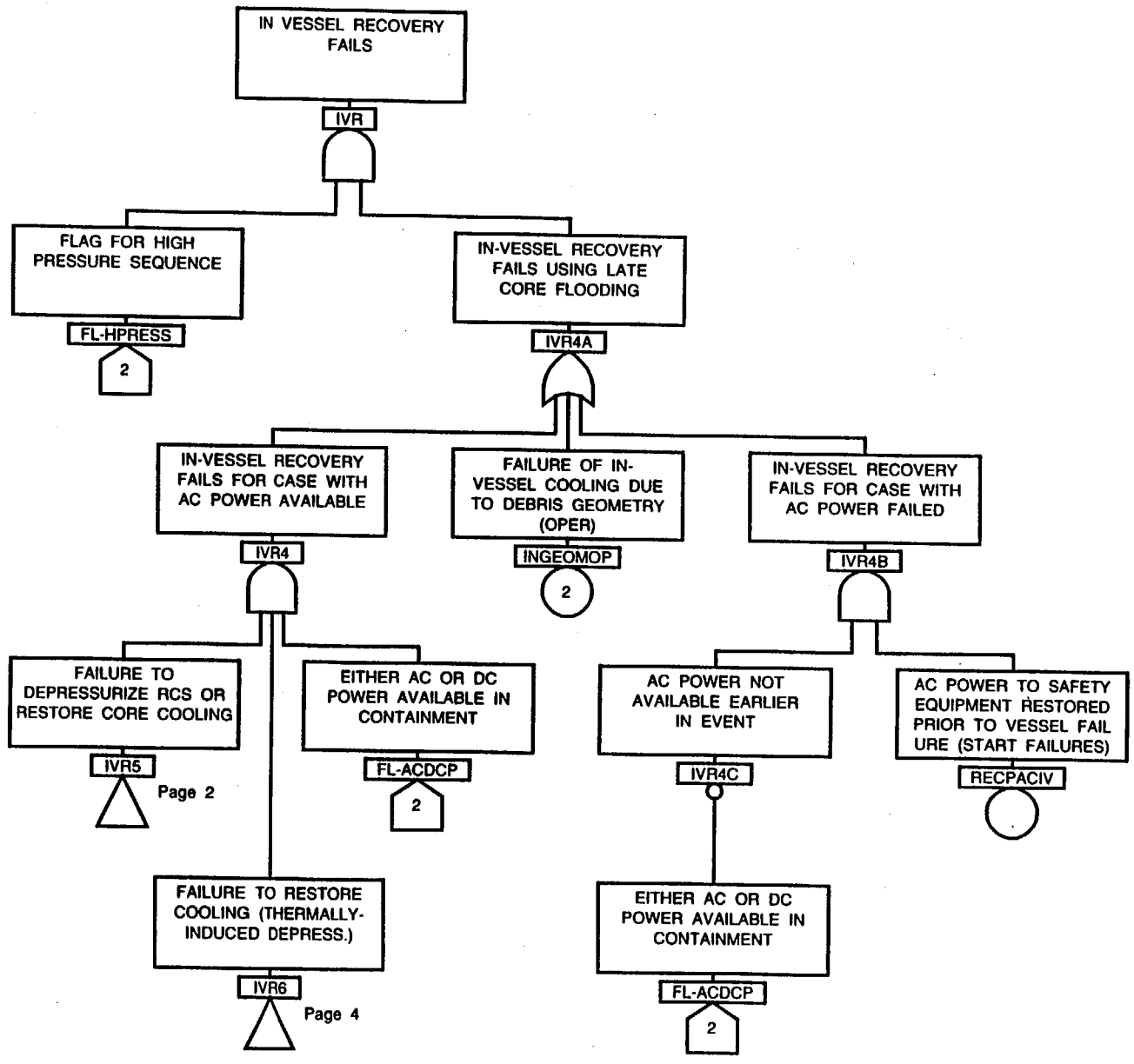


Figure 4–23
In-Vessel Recovery (page 2 of 5)

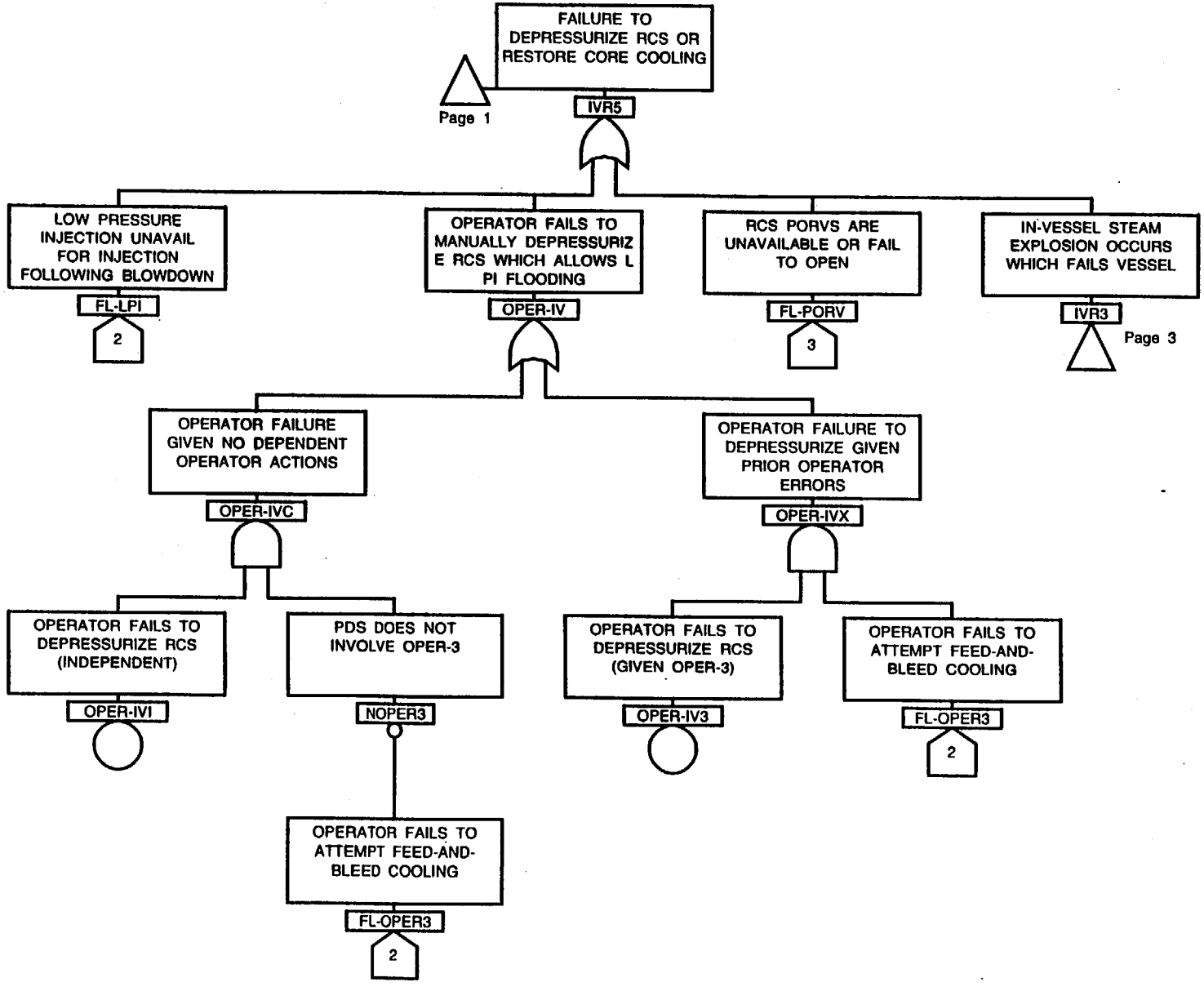


Figure 4–23
In-Vessel Recovery (page 3 of 5)

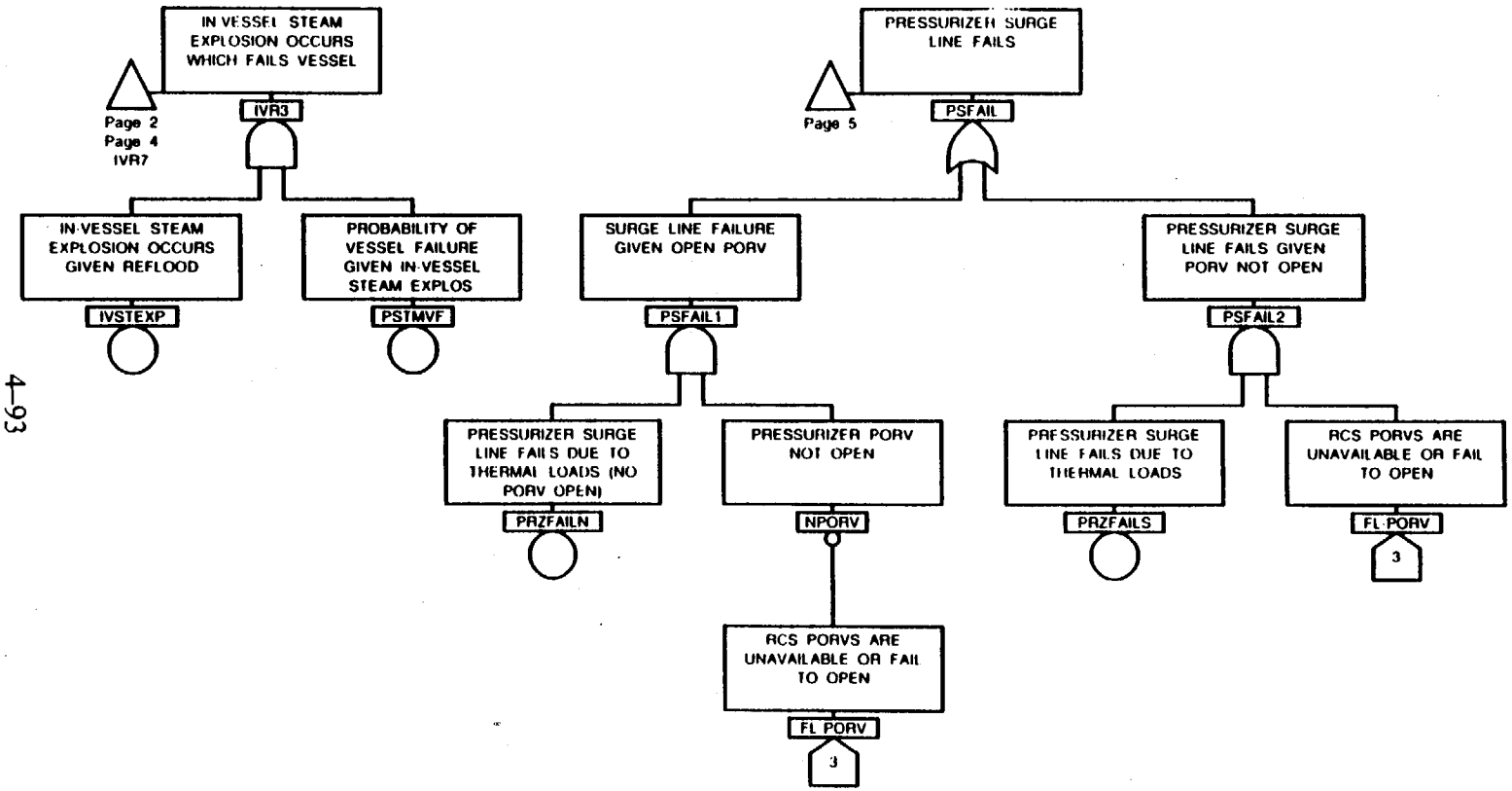


Figure 4–23
In-Vessel Recovery (page 4 of 5)

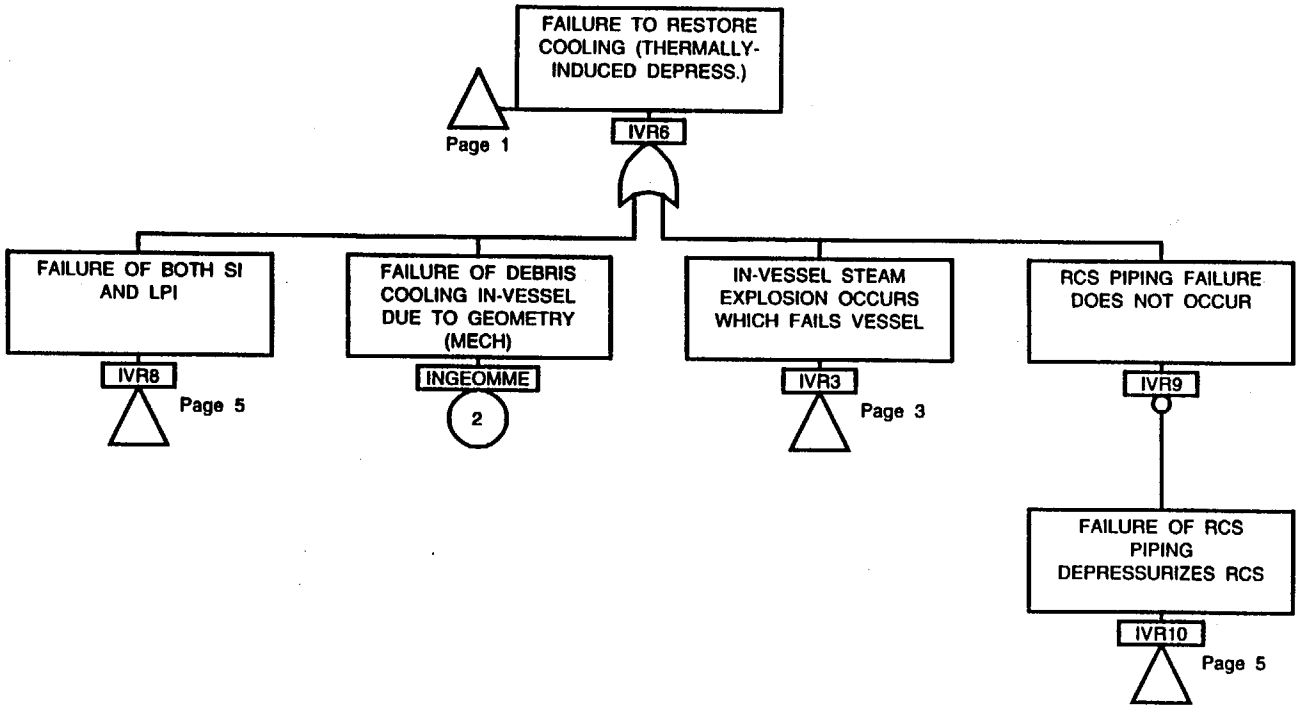
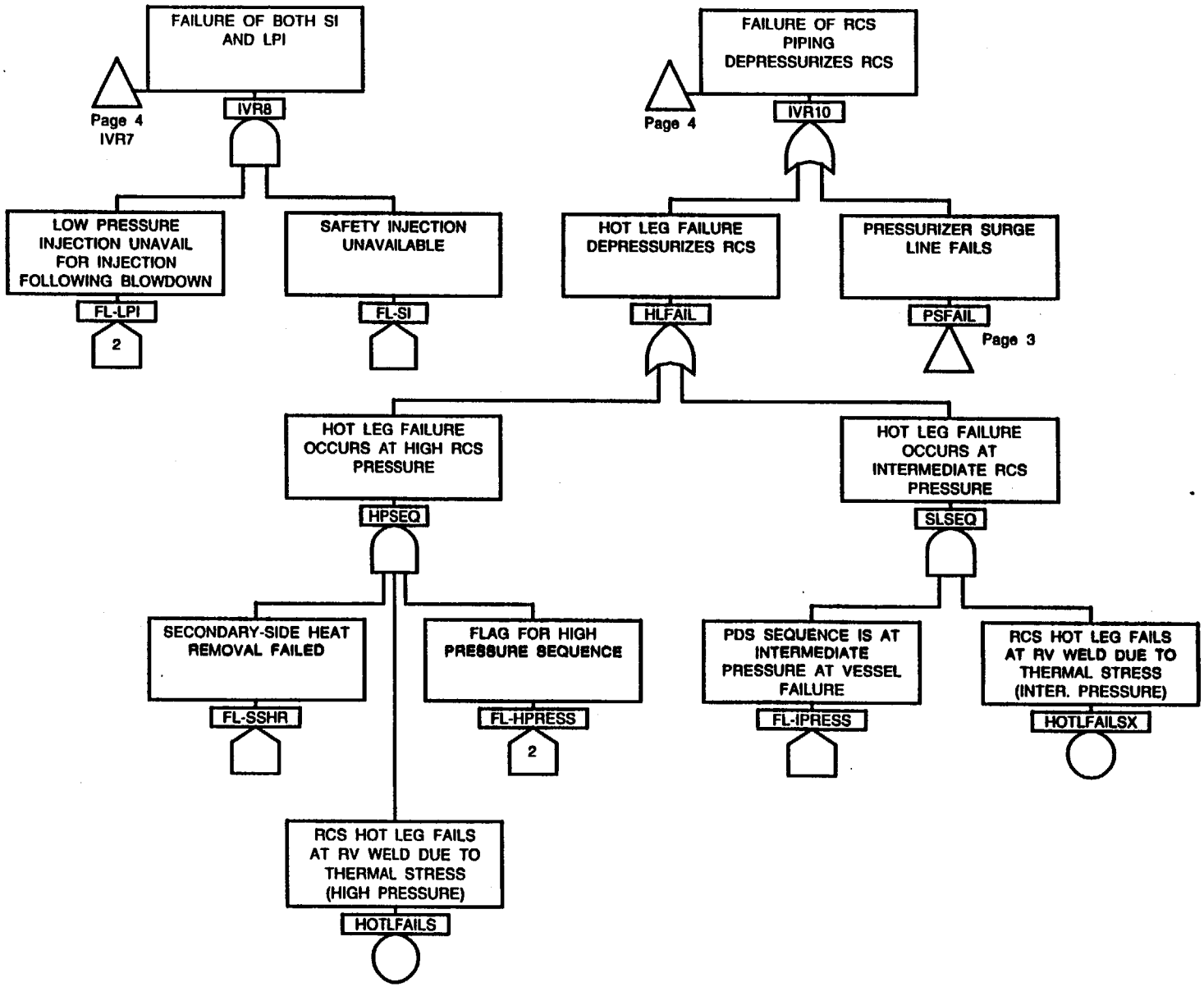


Figure 4-23
In-Vessel Recovery (page 5 of 5)



An important consideration in determining the potential for hydrogen burning is the steam concentration present in the containment. The logic model differentiates two basic configurations with either high or low steam concentrations. Based on plant-specific MAAP analyses, the predominant steam concentrations can be binned in this manner. MAAP analyses show that for cases with containment heat removal, the containment pressure and steam concentration are kept low. If no containment heat removal is present, the containment pressure exceeds 3 atmospheres at the time of significant hydrogen generation. In order to uncover the core water must be boiled away. MAAP results indicate that the quantity of steam generated during this boiloff process is sufficient to raise the steam concentration above 55% which is considered the point that steam inerting would occur and hydrogen burning would be precluded. It is logical to assume, therefore, that these cases will encompass the various less likely possibilities.

Four factors are considered in determining the potential for hydrogen burns failing containment. The first is the hydrogen source term. Depending on the timing of recovery the hydrogen source term may vary. It is, therefore, difficult to predict the quantity present in the containment for burning.

A reasonable upper bound for in-vessel generation is considered to be an equivalent quantity of hydrogen generated from the oxidation of 75% of the available core zirconium. Since no core concrete attack will occur for cases recovered in-vessel, the in-vessel production bounds the overall hydrogen concentration. The 75% value is conservatively adopted as the hydrogen source term for in-vessel recovery cases and exceeds the 50% value estimated for TMI.

In addition to the presence of hydrogen and oxygen, an ignition source must be present for a burn to occur. If AC power is available in the containment, many potential sources of ignition will be present and an ignition source is almost assured. For cases without AC power, it is somewhat less likely. Hot surfaces and the potential for random sparks do provide some potential for ignition and are considered.

The steam concentration is represented by the presence or absence of containment heat removal. For cases with containment heat removal the steam concentration is considered to be low and the potential for burning, given sufficient hydrogen and an ignition source, high. This is the complement to the case with containment heat removal failed. For cases without containment heat removal, the steam concentration present would make hydrogen burning very unlikely.

The final event identifies the potential for the postulated hydrogen burn to fail containment and is evaluated by comparing the expected final containment pressure after the burn to the containment failure curve. The containment base pressure prior to the burn is chosen based on the presence or absence of containment heat removal. For cases

with heat removal the base pressure is essentially 1 atmosphere. Without heat removal, a 3 atmosphere base pressure is used. These values are based on MAAP assessments of several different accident sequences. The results from these assessments tend to support the use of these two points. Figure 4–24 provides the top logic for this event.

CB – Containment Not Bypassed

This top event addresses the status of the containment early in the event and identifies sequences which result in containment bypass. Three different conditions are provided. Two are identified by the PDS entry state with a third being dependent of physical phenomena addressed by the CET.

PDS states which involve SGTR or ISLOCA result in a direct path from the RCS outside the containment. Flag events are included under this event to identify these sequences. In addition to these bypass sequences, the potential exists for a late SGTR which is induced by stresses on the steam generator tubes much like those considered in addressing the potential for hot leg failure. The development of this failure mode is discussed below.

If the RCS remains at high pressure, the decay heat will be transferred through the RCS. If the operators operate the RCPs, as directed by procedures, a path from the core to the steam generators will be established, by emptying the loop seals of water, and energy will be transmitted to the steam generator tubes.

If secondary-side cooling is not present, the tubes could heatup until thermally-induced failure occurred. Restarting the RCPs will clear the water present in the loop seals and increase heat transfer to the steam generator tubes, which will increase the potential for thermally-induced failure. The presence of secondary-side heat removal precludes this failure mode. Secondary-side depressurization, also included in the procedures for restoration of heat removal, can increase the pressure differential across the tubes and may increase the potential for failure. These consideration are included in covering this issue. Events which address the potential for tube failure, RCP status, and secondary-side pressure are included and the top logic is presented in Figure 4–25.

CI – Successful Containment Isolation

In addition to the potential for containment bypasses, the failure of the containment isolation system can result in early leakage of radionuclides. This event comprises the two states identified in the containment safeguards event tree (CSET). Large and small isolation failures are addressed by flag events. The definition of these two failure size ranges is discussed in Section 4.3. Figure 4–26 provides the simple model for this event.

Figure 4–24
H_2 Burn Occurs After In-Vessel Recovery (page 1 of 2)

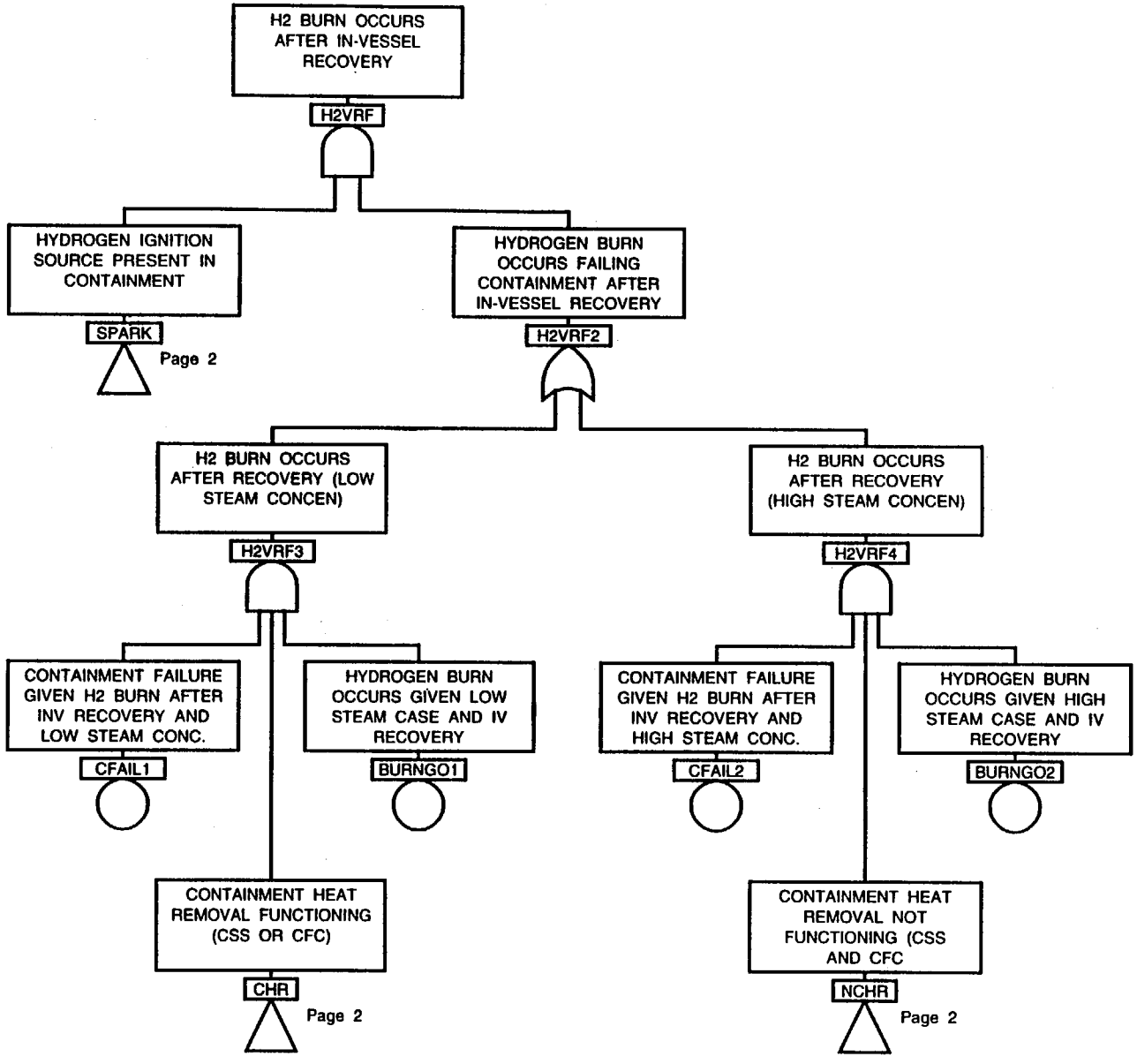


Figure 4–24
H_2 Burn Occurs After In-Vessel Recovery (page 2 of 2)

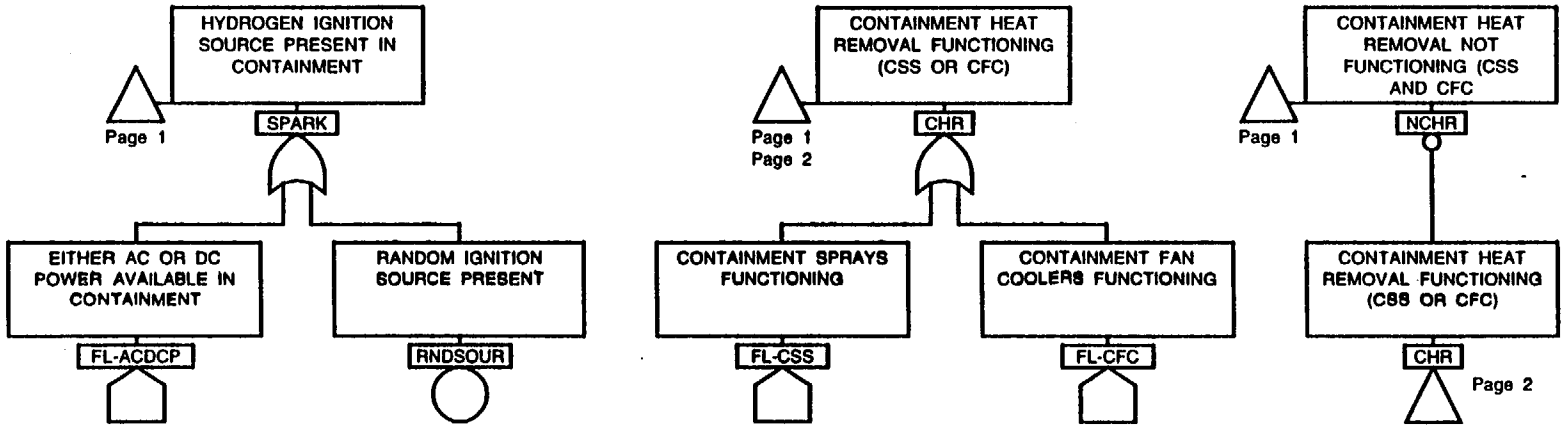


Figure 4–25
Containment Bypass Occurs (page 1 of 2)

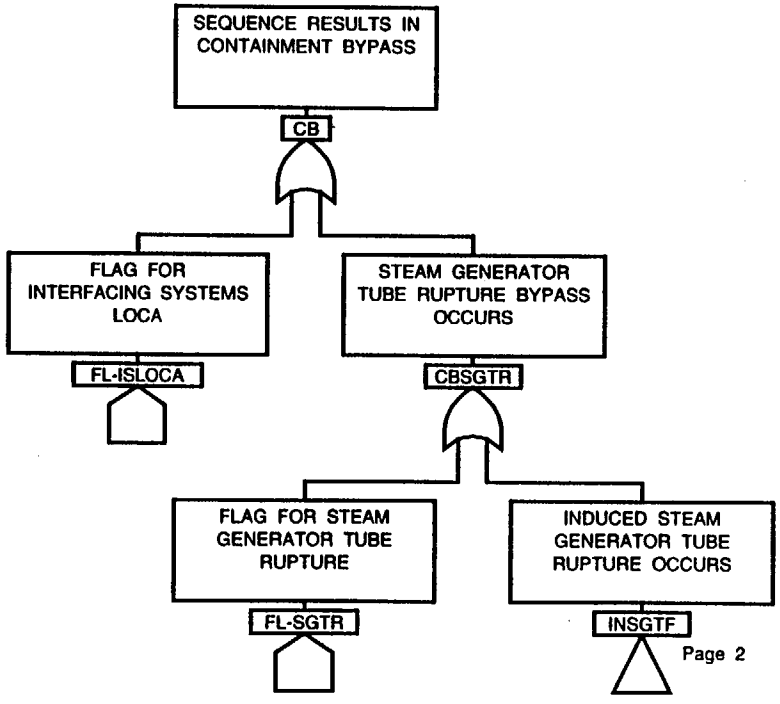


Figure 4–25
Containment Bypass Occurs (page 2 of 2)

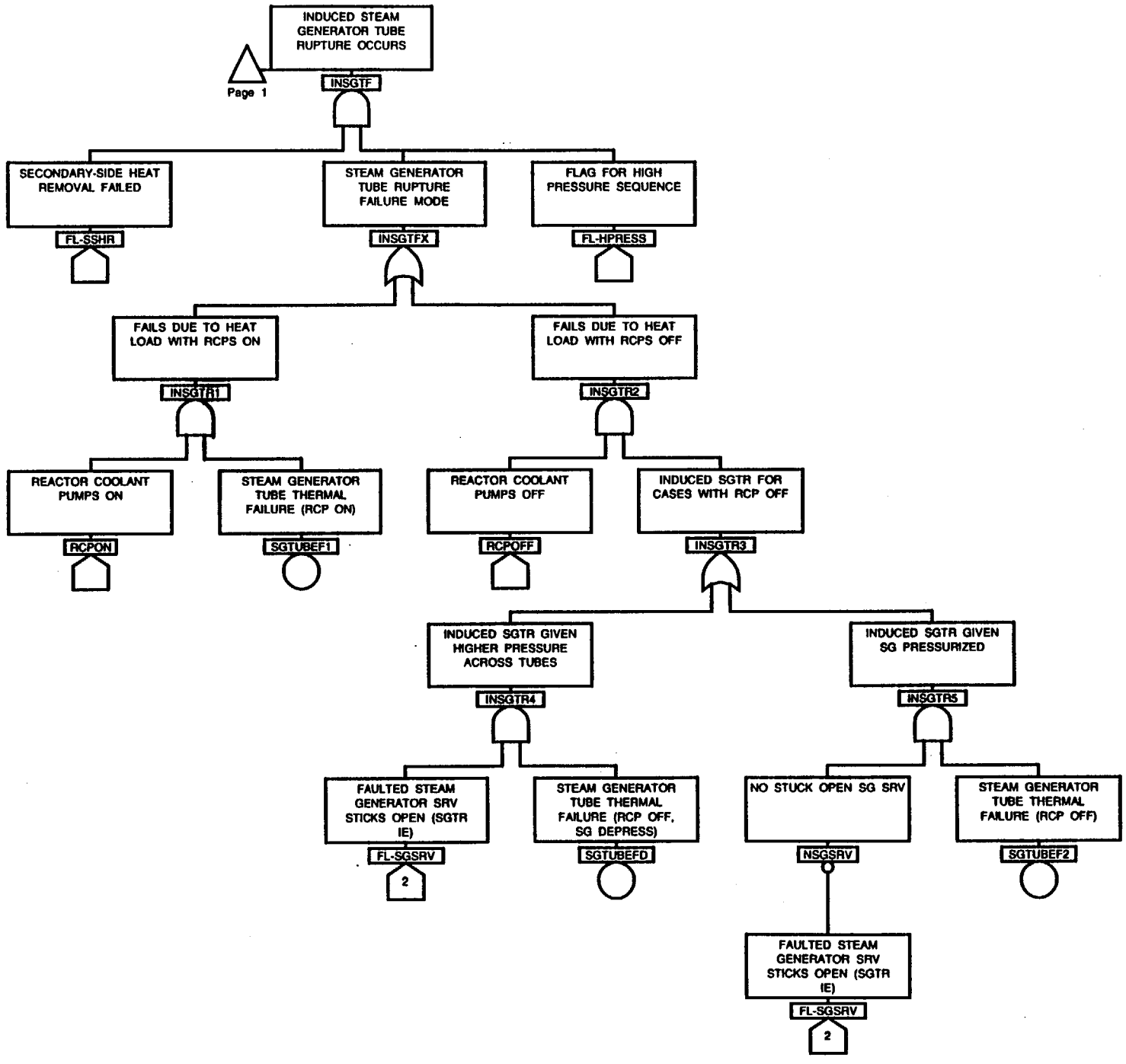
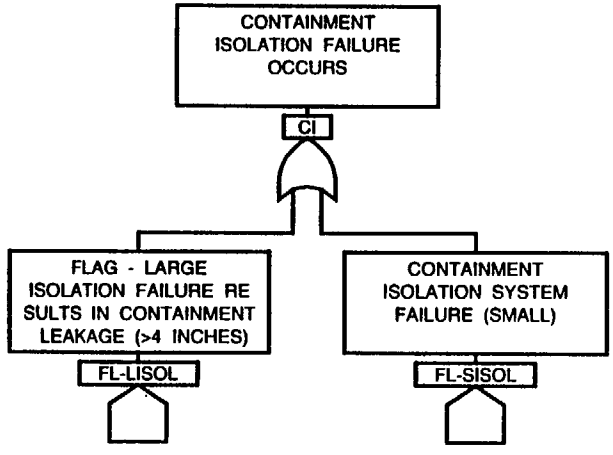


Figure 4–26
Containment Isolation Fails



IFL – Containment Leakage Rate is Small

If the containment is impaired due to either an isolation failure or bypass sequence the rate at which the containment is leaking becomes important for several reasons. First, this leakage rate determines radionuclide holdup time in containment which, in turn, determines the likelihood of radionuclide settling in the containment and the resultant source term. Additionally, if the containment leakage is large, further challenges to the containment will be less important since the existing breach will tend to preclude significant containment pressurization. Small leakage, however, have the potential for becoming larger releases later in the sequence due to loads which may challenge the containment later in the sequence.

For SHNPP, two cases, large and small, are used (illustrated by Figure 4–27). For isolation failures the choice of size ranges is based on the ability to prevent further containment loads. The bypass sequences are chosen based on the relative bypass size. SGTR sequences are binned as small, and ISLOCA events, dominated by RHR bypass sequences, are binned as large leakage rates. The source terms expected for large and small leakage rates are also adjusted based on the leakage size.

ECF – Early Containment Overpressure Prevented

This question addresses the possibility of an early containment overpressurization failure. The term early addresses the time period up to and some time slightly after RPV failure. The use of the term early is somewhat arbitrary but is used to symbolize the break point between prompt containment failures and those associated with long term pressurization and prolonged CCI.

The model encompasses the most probable pressurization events which have been assessed in prior PRAs and are specified in NUREG–1335. The effect the associated pressure load has on containment integrity for the following events is assessed.

- ex-vessel steam explosions
- hydrogen deflagration to detonation transition (DDT) and detonation
- hydrogen deflagration
- DCH

Figure 4–27
Containment Leakage Rate (page 1 of 3)

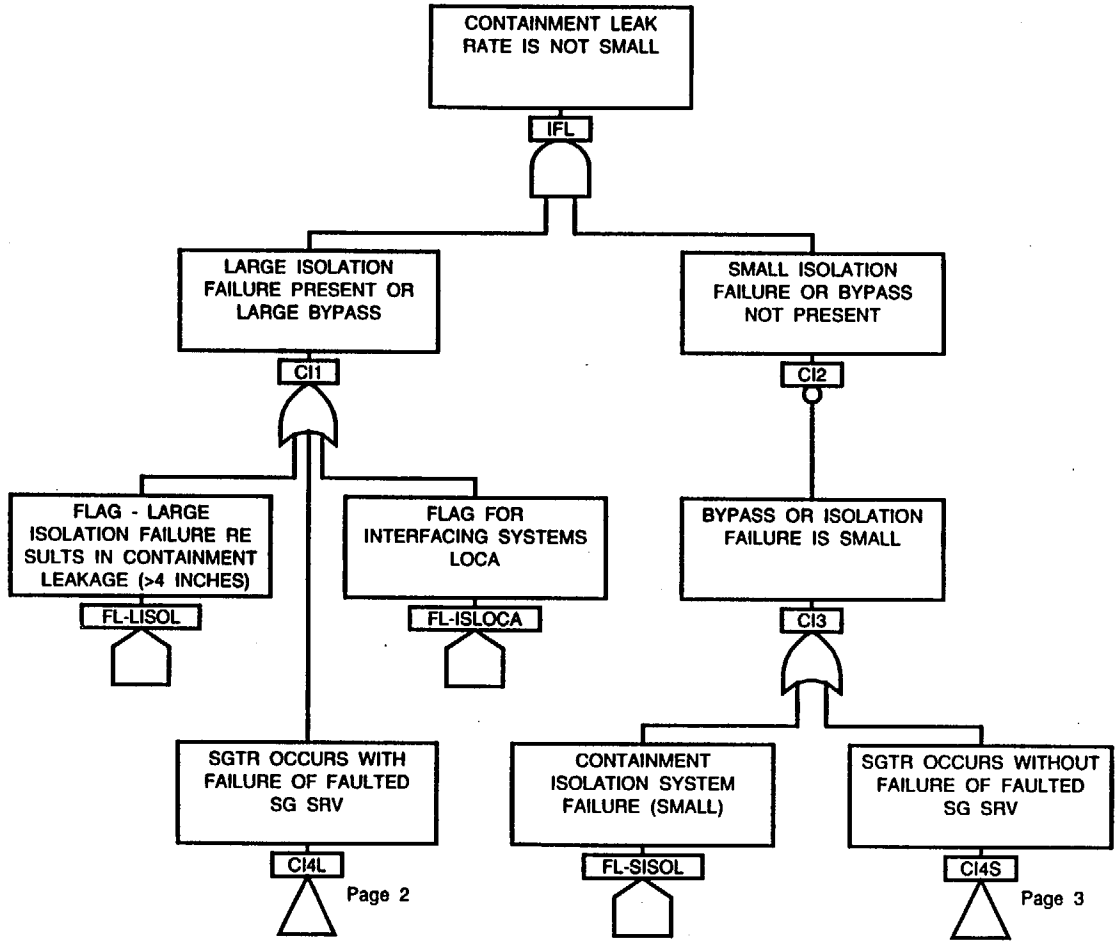


Figure 4–27
Containment Leakage Rate (page 2 of 3)

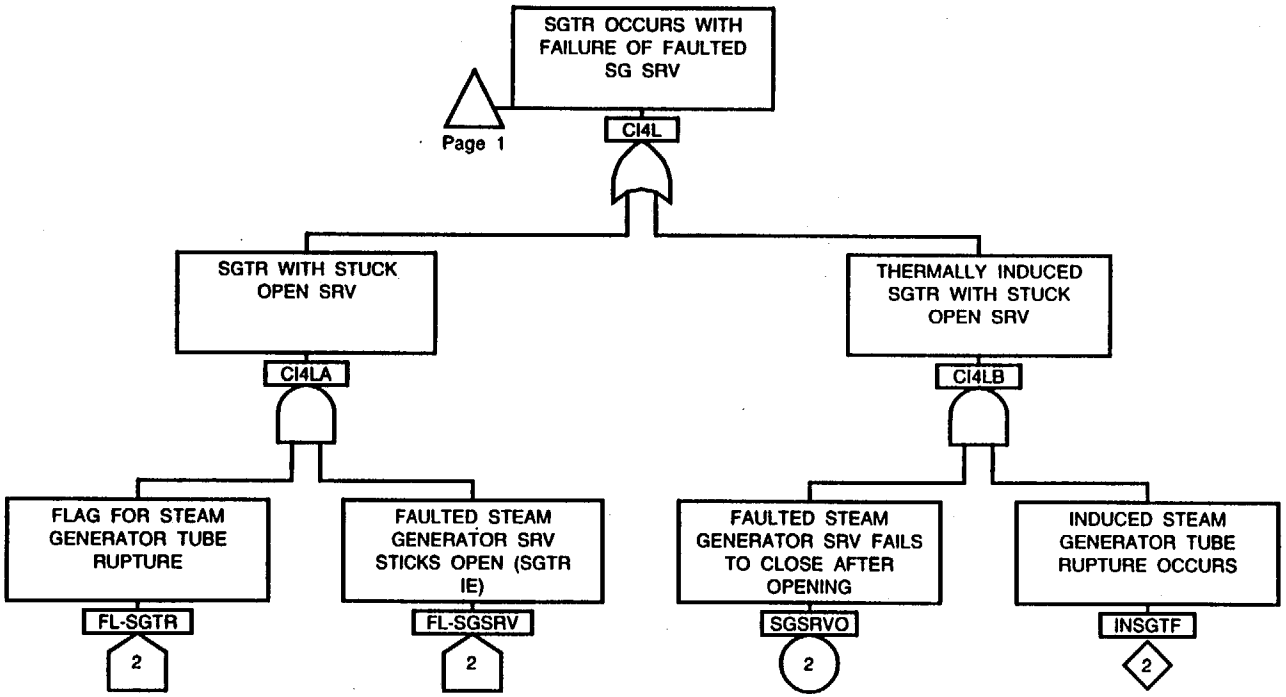
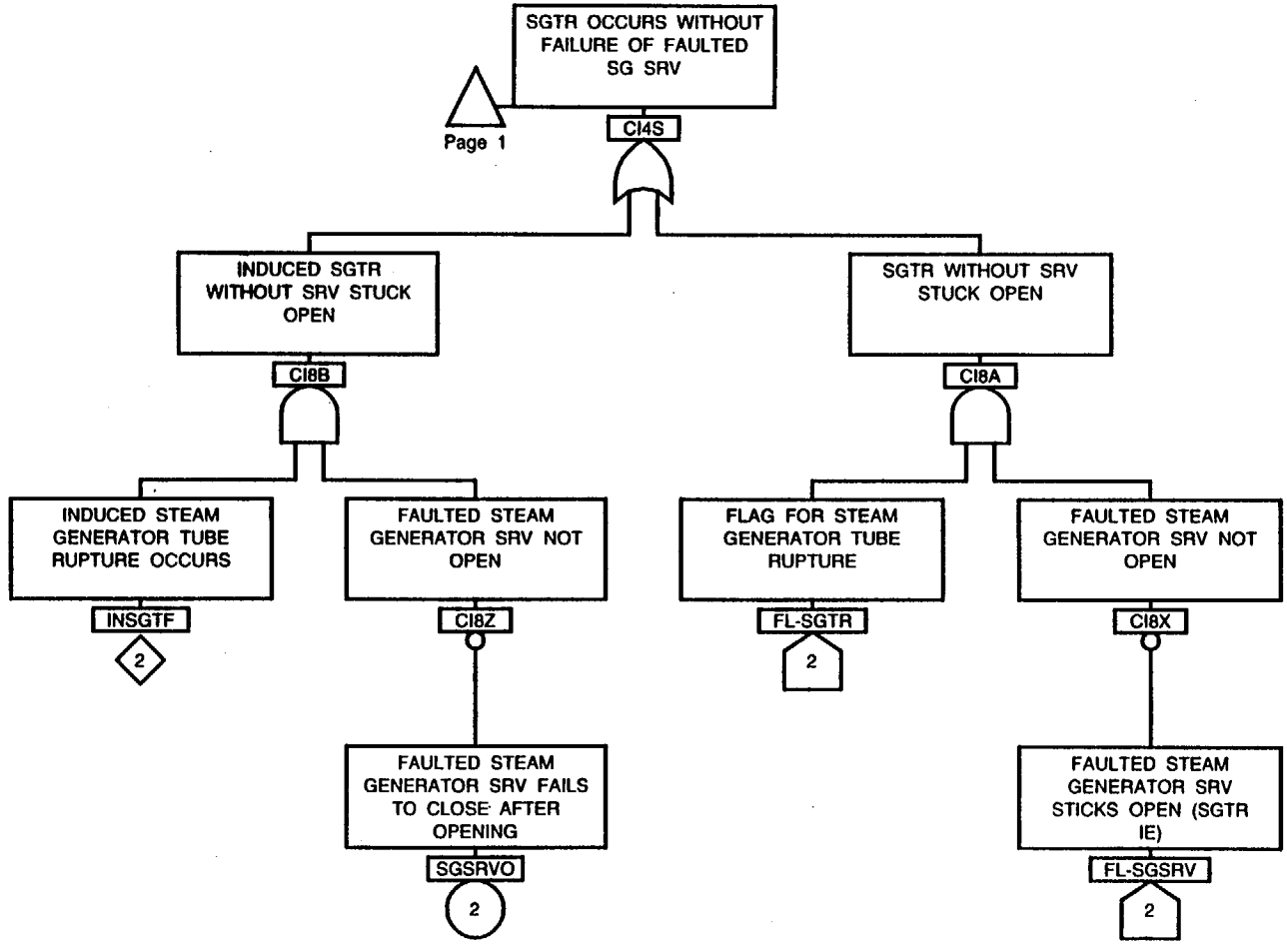


Figure 4–27
Containment Leakage Rate (page 3 of 3)



As appropriate, containment loads are addressed both individually and in combination as has been postulated based on prior analyses and experimental evidence. In addition, the containment pressure prior to the event is also addressed in determining the resultant containment pressurization. Containment loading from multiple sources have been postulated to occur simultaneously for events such as DCH and hydrogen deflagration.

In addition to the loads already discussed, the potential for containment overpressurization from steam generation is considered. Some plants have identified situations where, given no containment heat removal, the containment failed while the core was successfully cooled. For PWRs, this usually involved the use primary feed-and-bleed for cooling coupled with a small containment volume. Decay heat is deposited into containment and containment pressure exceeded design pressures.

Based on MAAP analyses performed for SHNPP and using a plant-specific model, this challenge is not of concern. The containment pressure remains well below the median capacity and decreases once recirculation is initiated. Therefore, this failure mode is not addressed in the CET. Each of the other potential containment loads are addressed below in terms of how the CET model reflects the particular phenomenon.

Ex-Vessel Steam Explosion

When extremely hot material comes into contact with water a rapid exchange of energy occurs and the conditions may be such that a steam explosion occurs. The potential for such an occurrence is dependent on such factors as the mass of material, the degree of mixing, and the volume of water into which the material is injected.

Based on the SHNPP design, the cavity is expected to be flooded if the RWST is emptied into the containment. If the RWST is not injected, only a small quantity of water will be located in the cavity and steam explosions are not considered likely. Only cases with the RWST injected, therefore, are of interest and considered in the CET.

Upon failure of the RPV lower head, melted debris will be released into the cavity. This provides a potential condition for a steam explosion to occur. The steam explosion would generate a pressure wave that may be transmitted to the cavity walls and result in substantial loading on these surfaces and weakening of the cavity structures. This, in combination with loads on the RPV, could result in a loss of RPV support which could in turn result in movement of the vessel, connecting piping, and result in a loss of containment integrity at some piping penetrations. In addition to this failure mode, experiments performed to date have sometimes resulted in the dynamic destruction of the test apparatus. Since the cavity is embedded, the forces acting on the cavity walls are not expected to pose a threat to the cavity. This is, however, a source of uncertainty since detailed loading calculations are not available. To address this issue, the model includes events which address the likelihood that a steam explosion will occur in the reactor cavity

and if the resultant pressurization will be sufficient to fail the cavity. The assumption is made that cavity failure will result in a loss of RPV support. A flag event requiring RWST injection is provided to switch the logic on only for cases with the RWST injected and were sufficient water would be present in the cavity.

Hydrogen DDT/Detonation

As the core overheats, hydrogen is generated during the metal-water reaction and is transported into the containment. If sufficient hydrogen is generated, the potential exists for hydrogen deflagration. Delaying burns until the hydrogen concentration is high may lead to a transition from deflagration to detonation (DDT), or to a pure detonation.

DDT is the process of transitioning a deflagration into a detonation by accelerating the flame propagation speed until the velocity of the wave approaches sonic velocities. Typically, this occurs when deflagration occur in long and thin chambers. For dry air detonations have been caused in hydrogen concentrations of approximately 10% by volume for some very specific geometries and using a high power ignition source. As steam is introduced, the likelihood of detonation or deflagration decreases proportionally until hydrogen burning of any form is precluded due to steam inerting.

Information contained in Section 4.2 describes the SHNPP containment in relation to hydrogen generation and the potential for DDT and detonation. Based on the excellent mixing of containment volumes, and the potential hydrogen behavior, the likelihood of detonation is believed to be remote which is consistent with most other studies which examine large-dry containments. Since the possibility of DDT or detonation cannot be completely excluded, the model does address the potential for a hydrogen detonation or DDT. Two potential events are provided which address local and global events. If high local concentrations occur, the potential exists for a localized detonation to occur resulting in a dynamic pressure on a localized basis which may be sufficient to fail the containment. The global concentration must approach the minimum hydrogen concentration necessary for a detonation to occur to produce a global detonation.

In assessing the potential for DDT or detonation, it is important to consider several factors, including steam concentration, maximum hydrogen concentration, potential for containment mixing and hydrogen burning to preclude high hydrogen concentrations, and the availability of an ignition source sufficient to result in hydrogen detonation. Without adequate hydrogen or an ignition source, detonation is precluded.

From information contained in Section 4.2, sufficient communication between compartments and adequate flow through the various subcompartments exist to consider the containment well mixed. This mixing is important for precluding local pockets of high hydrogen concentration. In addition, the containment volume to zirconium ratio is such that almost complete reaction of the metal in the core is necessary to generate

hydrogen concentrations which would promote global detonations in the absence of steam. Since steam concentrations are expected to be high for most sequences and likelihood of generating this quantity of hydrogen by in-vessel generation are very remote, DDT/detonation is not expected.

Containment sprays and fan coolers also play a part in assessing the potential for DDT/detonation. If either is functioning, the containment steam concentration will be low which reduces the hydrogen concentrations necessary for DDT/detonation. It also, however, will encourage global burning at lower concentrations which will prohibit DDT or detonation. Additionally, spray flow or fan cooler operation will result in turbulence in the containment which will promote mixing and will lower the potential for locally high hydrogen concentrations.

The model includes events which assess local and global DDT/detonation, the presence of an ignition source, and the operation of containment cooling. As described earlier, the status of containment cooling provides a direct correlation to steam concentration. If global hydrogen burning occurs prior to reaching DDT/detonation concentrations, the available hydrogen and oxygen concentrations will be reduced and will be insufficient to support DDT or detonation. As such, the potential for an early hydrogen burn is included as a means of precluding DDT/detonation.

Hydrogen Deflagration

The hydrogen generated during core degradation due to the metal-water reaction may be ignited in the containment resulting in energy generation and increased containment pressure and temperature. If sufficient hydrogen is present and is burned completely, the resultant pressure rise may be sufficient to challenge containment integrity. If the flame speed is slow, the ignition is referred to as a deflagration. The potential for hydrogen burns is related to parameters previously described for DDT/detonation. In contrast to detonation, hydrogen burning can be supported in concentrations as low as 8% by volume in dry air. Larger concentrations are needed for cases involving partial steam environments. Hydrogen burning can not occur in volumes with a steam concentration in excess of 55%. In addition to an adequate hydrogen to steam ratio, an ignition source must be present such as electrical equipment or hot surfaces.

As discussed above, several factors influence the potential for hydrogen burning. They may increase or decrease the likelihood of a burn or effect the resultant containment pressurization given a burn occurs. Operation of containment heat removal equipment tends to increase the potential for global hydrogen burns by providing an ignition source and reducing containment steam concentrations. Other factors are also important for assessing hydrogen deflagration likelihood, e.g., the rate of hydrogen generation. These issues must be accounted for the particular accident sequence and in evaluating the sensitivity of the results to changes in parameters.

MAAP analyses and hand calculations indicate that hydrogen deflagrations can be ruled out in the best estimate assessment, with the exception of those associated with direct containment heating, if the containment pressure is in excess of 45 psia. In order to achieve this pressure, the steam concentration inside containment must exceed the 55% steam inerting limit and hydrogen burning is precluded. MAAP analysis shows that inerting is assured for sequences not involving bypass prior to the core being uncovered. Therefore, the containment would be inerted prior to any large hydrogen concentrations.

The operation of containment sprays or fan coolers will result in low steam concentrations and will promote and increased likelihood of hydrogen burning by reducing the concentration below the inerting limit. Therefore, a general criterion for hydrogen burning can be defined that requires either containment sprays or fan coolers to function in order to prevent steam inerting.

Another factor to consider in evaluating the potential for a burn is the hydrogen source term. Hydrogen is generated when hot metals, such as zirconium in the core, react with steam. The amount generated is generally limited by the availability of steam. It is reasonable, and also supported by MAAP analysis, to assume that sequences which involve a rapid loss of RCS inventory from the RCS will result in lower hydrogen source terms than those which slowly lose inventory due to boiloff. Therefore, sequences, such as large LOCAs without injection, with rapid loss of inventory have somewhat lower hydrogen source terms. Put in other terms, high pressure sequences demonstrate higher source terms than those found for low pressure sequences.

The model for early hydrogen deflagration includes the factors described above including events which address the availability of ignition sources, presence of containment cooling (sprays or fans), the likelihood of a burn, and the ability of the containment to accommodate the pressure rise. The hydrogen source term is adjusted based on the RCS pressure.

Direct Containment Heating

If the debris is ejected from the RCS following vessel failure at high pressure (called high pressure melt ejection, or HPME), the potential exists for finely fragmented debris to be entrained in the escaping gases and be transported from the cavity to the lower containment. For some designs, the fragmented debris may be transported to the upper compartment. Two heating effects can occur due to DCH. First, the hot debris can, by means of its large surface area, transfer its energy to the surrounding environment which will result in an increase in containment temperature and pressure. In addition, the fragmentation of the debris exposes unoxidized metals to steam in the containment which results in additional oxidation and hydrogen generation. Given the temperature of the debris and the generation of hydrogen, hydrogen burning is likely to occur.

The exact phenomena associated with DCH is uncertain. However, it is generally understood that the risk from DCH is dependent on several factors such as the degree of debris fragmentation and the dispersal time. The potential for HPME, and therefore DCH, is dependent on the initial RCS pressure at the time of vessel failure. Lower RCS pressures reduce the potential for fragmentation and transport of materials from the cavity. The discussion in Section 4.2 also states that recent experiments tend to support the statement that the heating effects due to DCH are only a small input to the overall pressure rise. The hydrogen generation and burning is more important. Other factors which may influence the degree of DCH which occurs relate to the amount of debris that initially leaves the RPV. The less material found in the lower head at the time of vessel failure, the less severe the impact of DCH.

The amount of material which participates in the blowdown is uncertain and depends on the progression of the accident of which our present knowledge is limited due to a lack of conclusive data. If more debris is initially released to the lower head, the potential for more severe DCH is increased. This uncertainty must be considered when assessing the potential for HPME and DCH.

The mitigating effects of containment sprays or fan coolers are not considered for cases involving HPME/DCH. The presence of hot material is assumed to provide adequate ignition sources to allow hydrogen burns. Additionally, the time scale associated with energy addition due to DCH is on the order of 50 times faster than the rate of heat removal by the sprays or containment fan coolers. The operation of the containment sprays or fan coolers do, however, lower the base pressure of the containment. There is a direct relation between containment base pressure and the final pressure. If the base pressure is lower, the final pressure is reduced proportionally. Therefore, the operation of containment cooling will impact the final pressure and is included.

The likelihood of HPME increases with increasing RCS pressure. If the RCS pressure is low, the debris will not be significantly fragmented, and may be considered more a continuous wave which may be retained in the cavity, which effectively mitigates HPME. Based on a review of available literature, an acceptable cutoff appears to be on the order of 200 psia.

Higher cutoff values have been proposed but are not widely accepted. High pressure sequences are considered the most likely for HPME to occur with the likelihood decreasing with decreased RCS pressure. An intermediate pressure range is used to credit the reduced potential for HPME at pressure less than full RCS pressure. The major classes of accident sequences are grouped using these three pressure regimes. Events which occur after the onset of core damage can influence the RCS pressure at the time of vessel failure. Two important events deal with thermally-induced RCS piping failure and

repressurization when the debris comes into contact with the water in the bottom of the vessel.

As hot gases escape from the RPV, they impart energy into the RCS piping, specifically the hot leg and surge lines. This causes the temperature of the piping to increase. As the temperature increases the probability of stress failure of the piping increases. If piping failure occurs, then RCS will depressurize prior to vessel failure and reduce the potential for HPME. The model addresses this possibility for high pressure sequences. The phenomenon is related to the temperature at pressure for a length of time. As the RCS pressure is reduced, the probability of failure decreases. Intermediate pressure and low pressure sequences, therefore, are not included since the potential for failure is low.

After the core is uncovered, some water remains in the bottom of the RPV below the lower support plate. As the debris falls on the lower support plate, it progresses to the bottom of the vessel and interacts with this body of water. Depending on the actual progression, significant steam generation can occur and can repressurize the RCS. If the repressurization occurs and vessel failure is prompt, the RCS pressure may be high for some sequences initially placed in the intermediate pressure range. MAAP analyses, however, indicates that pressurization to above 1000 psia following support failure does not occur. Due to uncertainties in the physical phenomena, repressurization is considered in the model as a means of shifting a medium pressure sequence to the high pressure category. Due to the large RCS breach necessary to achieve a low pressure sequence, repressurization is not addressed for low pressure cases.

The model addresses pressurization from HPME and DCH loads at vessel failure and includes the blowdown loads and the potential for an associated hydrogen burn. The logic includes events which address debris fragmentation, RCS pressure at the time of RPV failure, and the potential for hydrogen burns concurrent with DCH. An important consideration related to hydrogen burning is the presence of an ignition source. For DCH, the primary ignition source is considered to be the debris. Thus, an ignition source is almost guaranteed and is included in the model.

The four phenomena described above represent the early containment pressurization loads assessed for cases involving RPV failure. The supporting fault tree logic for early containment failure is provided as Figure 4–28. Hydrogen detonation is not considered plausible but is provided for use in addressing any future sensitivity studies and to indicate its relative importance to other early failure modes.

ECC – Containment Liner Intact

A plant-specific review of the SHNPP containment identified a potential means for debris ejected following vessel failure to impact the containment liner. If sufficient material is transported to the liner it is possible that liner failure will occur due to ablation. The

basic path for the debris is to exit the RPV, traverse the cavity, enter the lower compartment, and exit through one or more ports located in the area. These holes serve as interfaces with the annular area outside the crane wall. Material ejected through these ports could strike the liner located outside the crane wall and pile up against the wall. Thermal effects could then weaken and fail the containment liner. In order to have a driving force to push the debris to the liner the RCS pressure must be sufficient to push the debris from the cavity. Based on prior discussions, this requires that the RCS pressure not be in the low pressure range. For the low pressure range, the majority of the debris will be retained in the cavity and only a very small fraction would be expelled which could attack the liner. If the RCS is at low pressure at the time of RPV failure, liner failure is considered remote and is not addressed.

For higher pressure sequences, as the debris is expelled immediately following RPV failure some of the debris will be fragmented and entrained in the exiting gases. The rest of the debris may either be fragmented or may form a coherent film at the base of the cavity which may be pushed up to the lower compartment by the exhaust gases. This film could then flow to the drain holes and be forced through the drains by pressurization in the lower compartment. Once through the holes, the material would spread across the flow and strike the liner.

Due to the presence of curbs, water would be present in the lower compartment with or without injection of the RWST. The quantity of water will be greater, about 6 ft, if the RWST is injected. The drain lines, therefore, will be either partially or totally underwater. This could impede the flow of the debris and serve as a means for debris quenching which could preclude continual contact with the liner by the debris. The water will aid in fragmenting and quenching the debris before it can attack the liner. The available area for the debris to spread is also important since it determines the depth of the material given that the amount of material expelled is known.

Figure 4–28
Early Containment Failure (page 1 of 15)

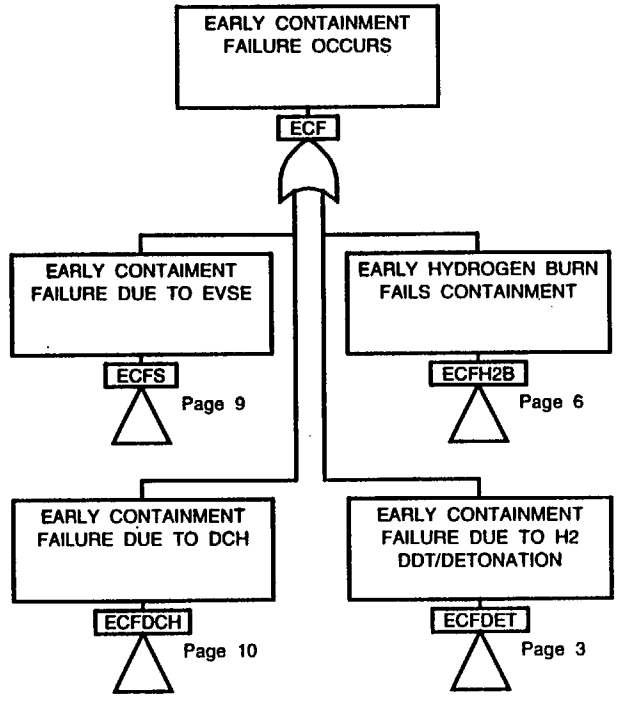


Figure 4–28
Early Containment Failure (page 2 of 15)

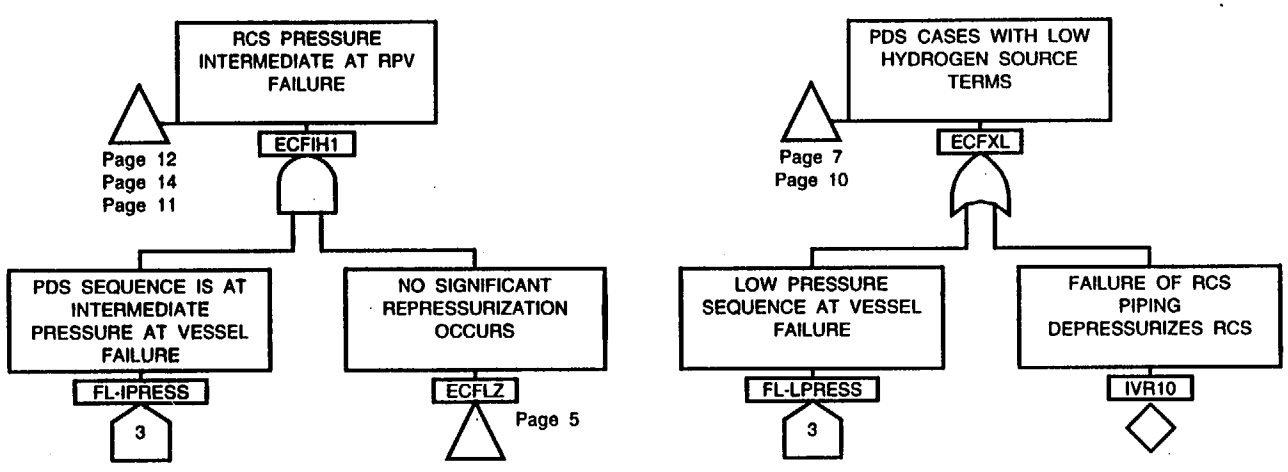


Figure 4–28
Early Containment Failure (page 3 of 15)

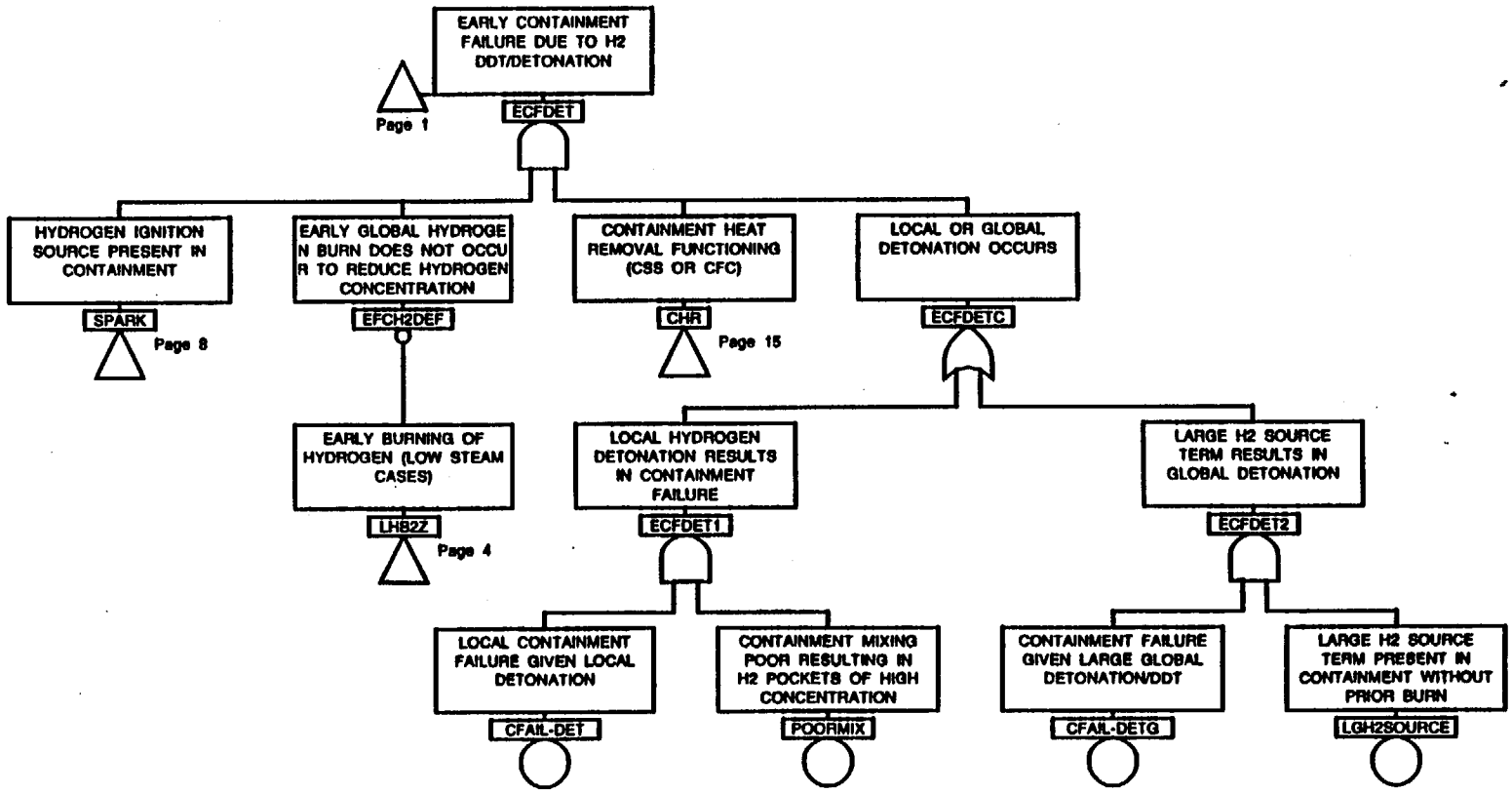


Figure 4–28
Early Containment Failure (page 4 of 15)

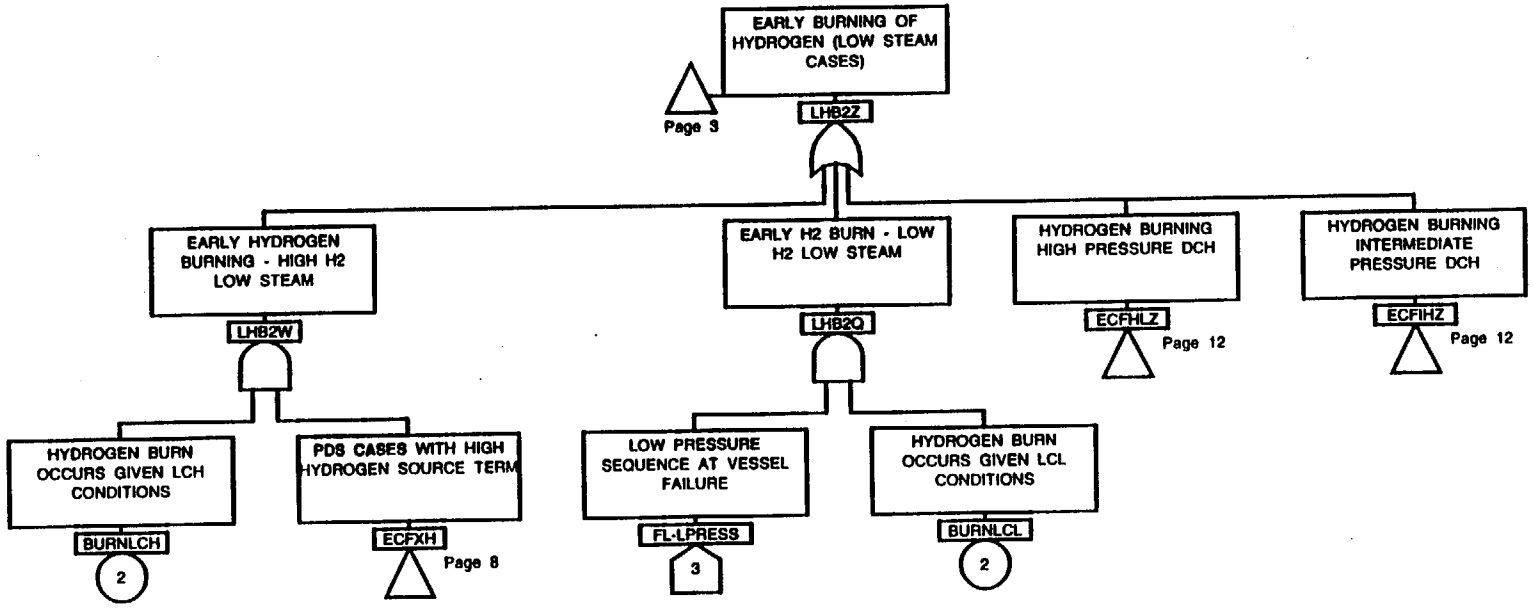


Figure 4–28
Early Containment Failure (page 5 of 15)

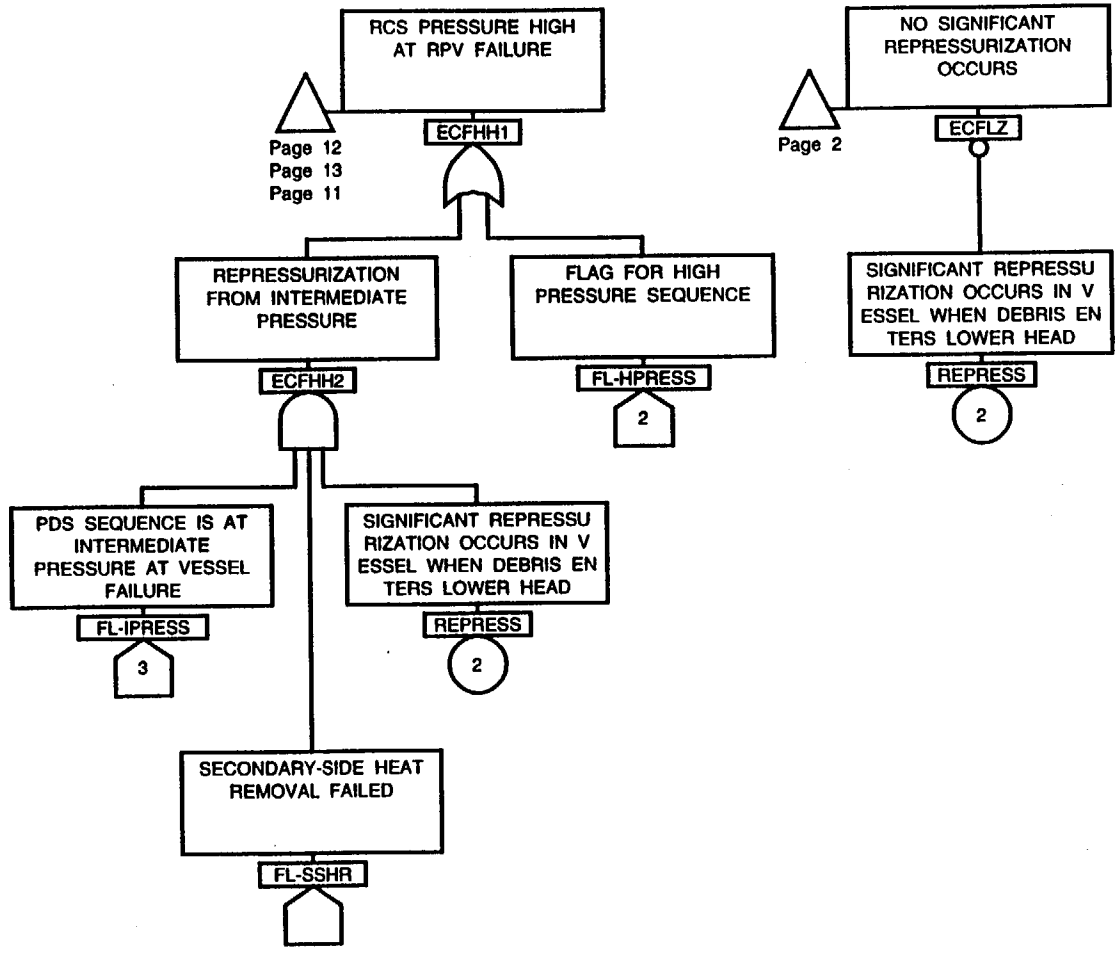


Figure 4–28
Early Containment Failure (page 6 of 15)

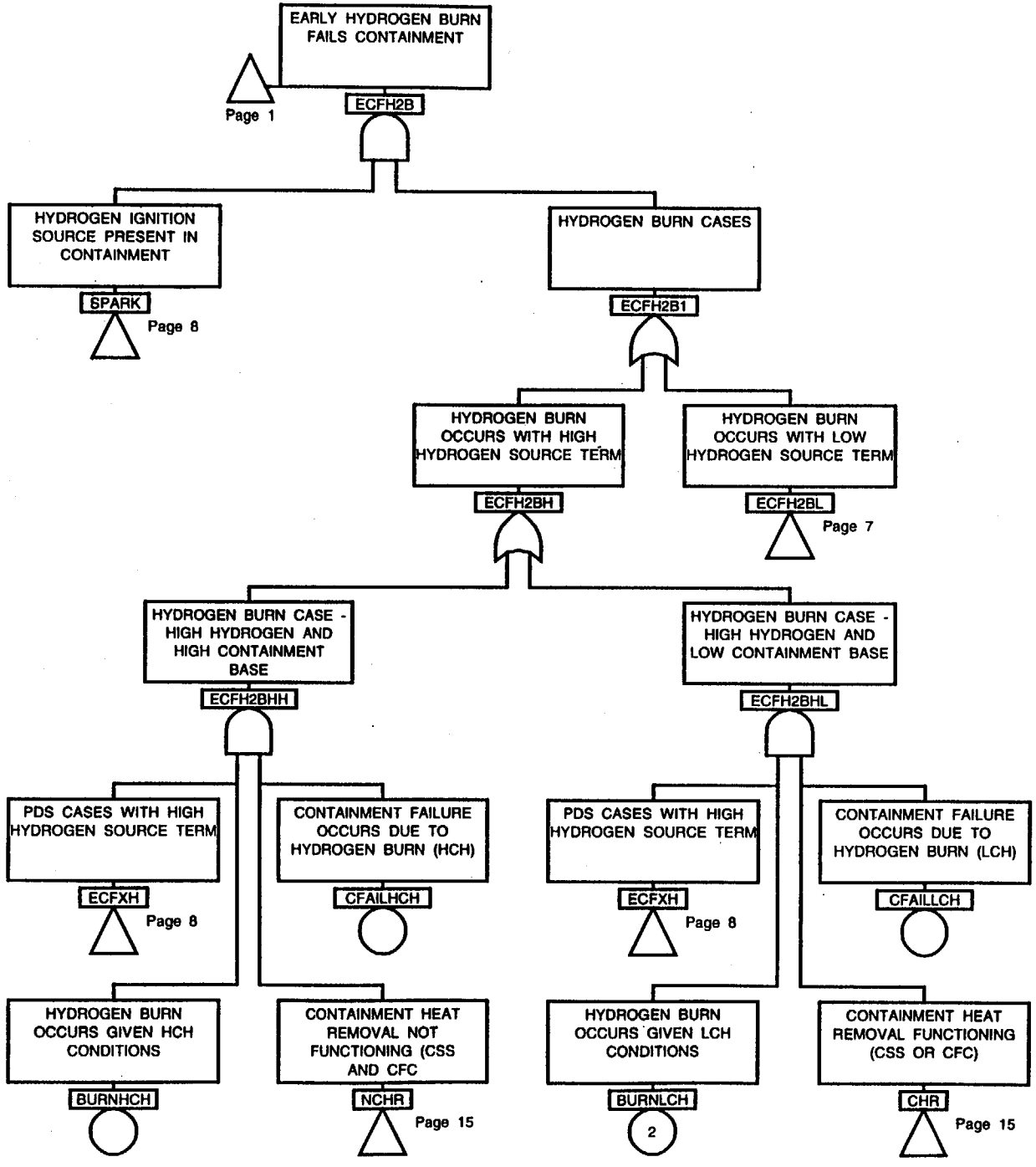


Figure 4–28
Early Containment Failure (page 7 of 15)

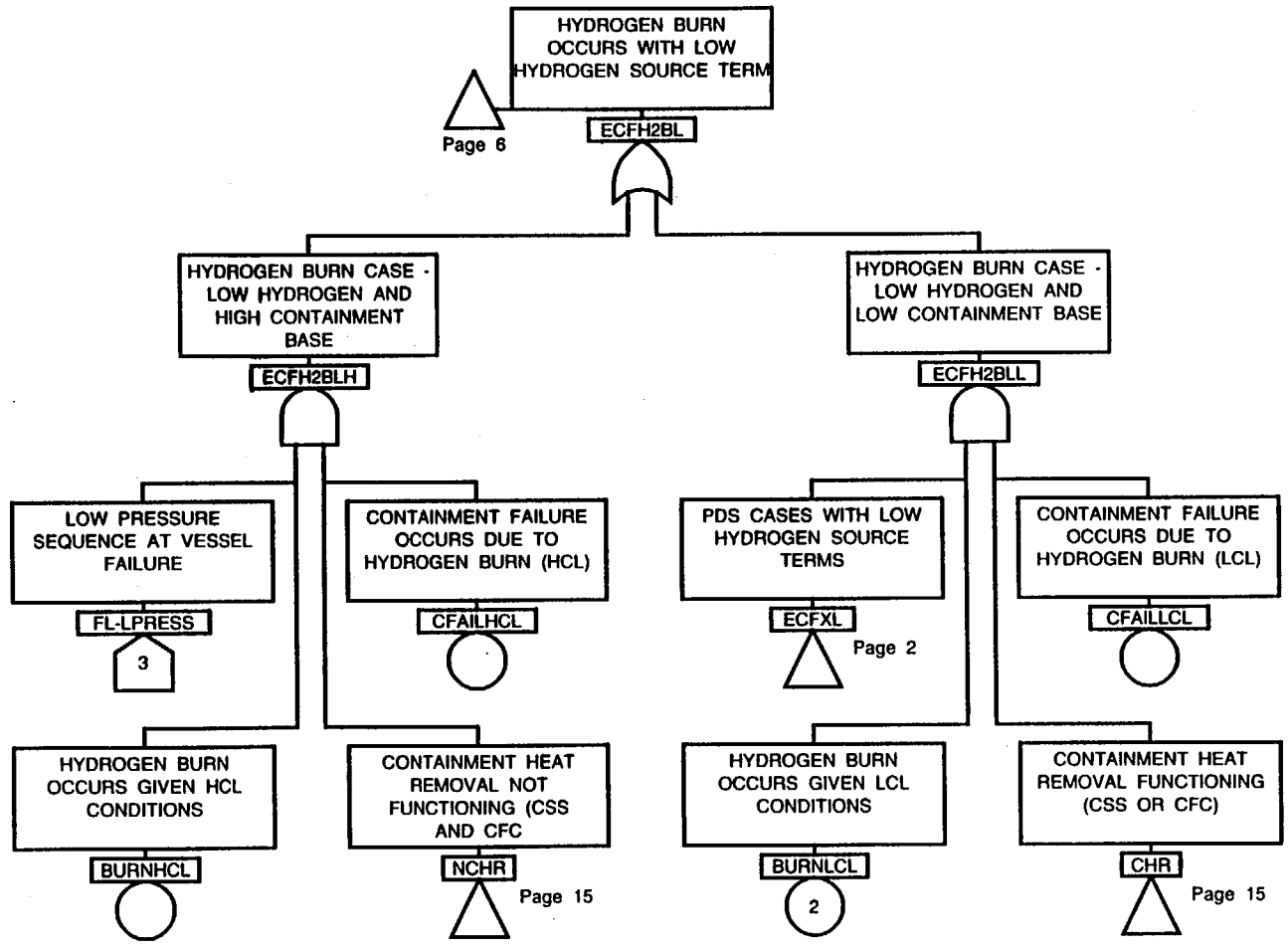


Figure 4–28
Early Containment Failure (page 8 of 15)

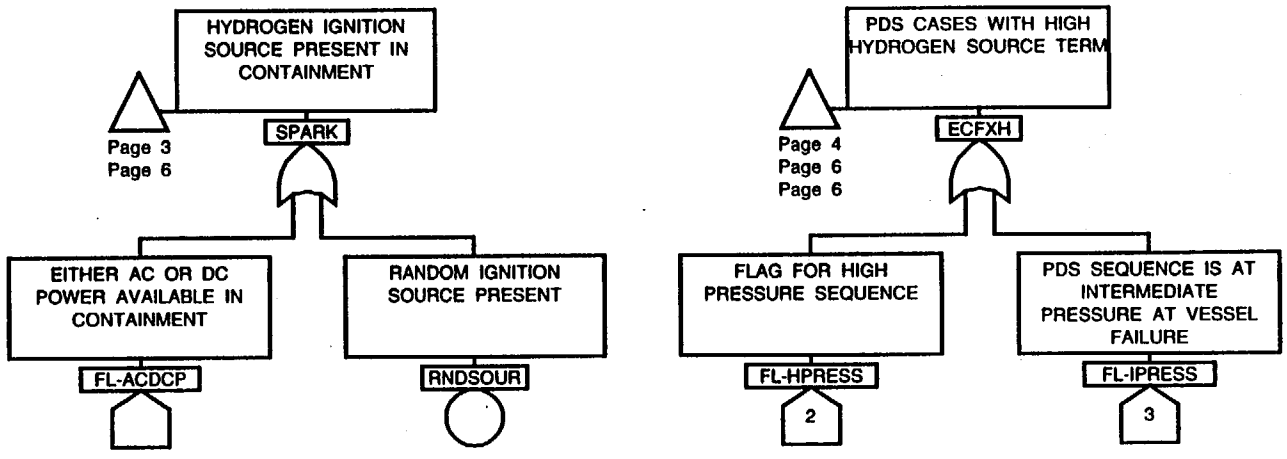


Figure 4–28
Early Containment Failure (page 9 of 15)

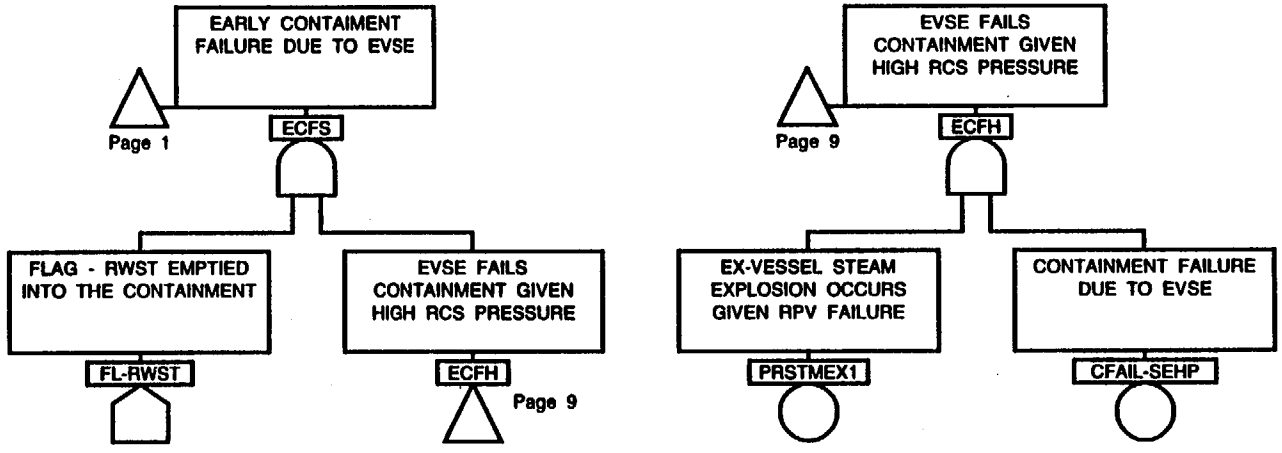


Figure 4–28
Early Containment Failure (page 10 of 15)

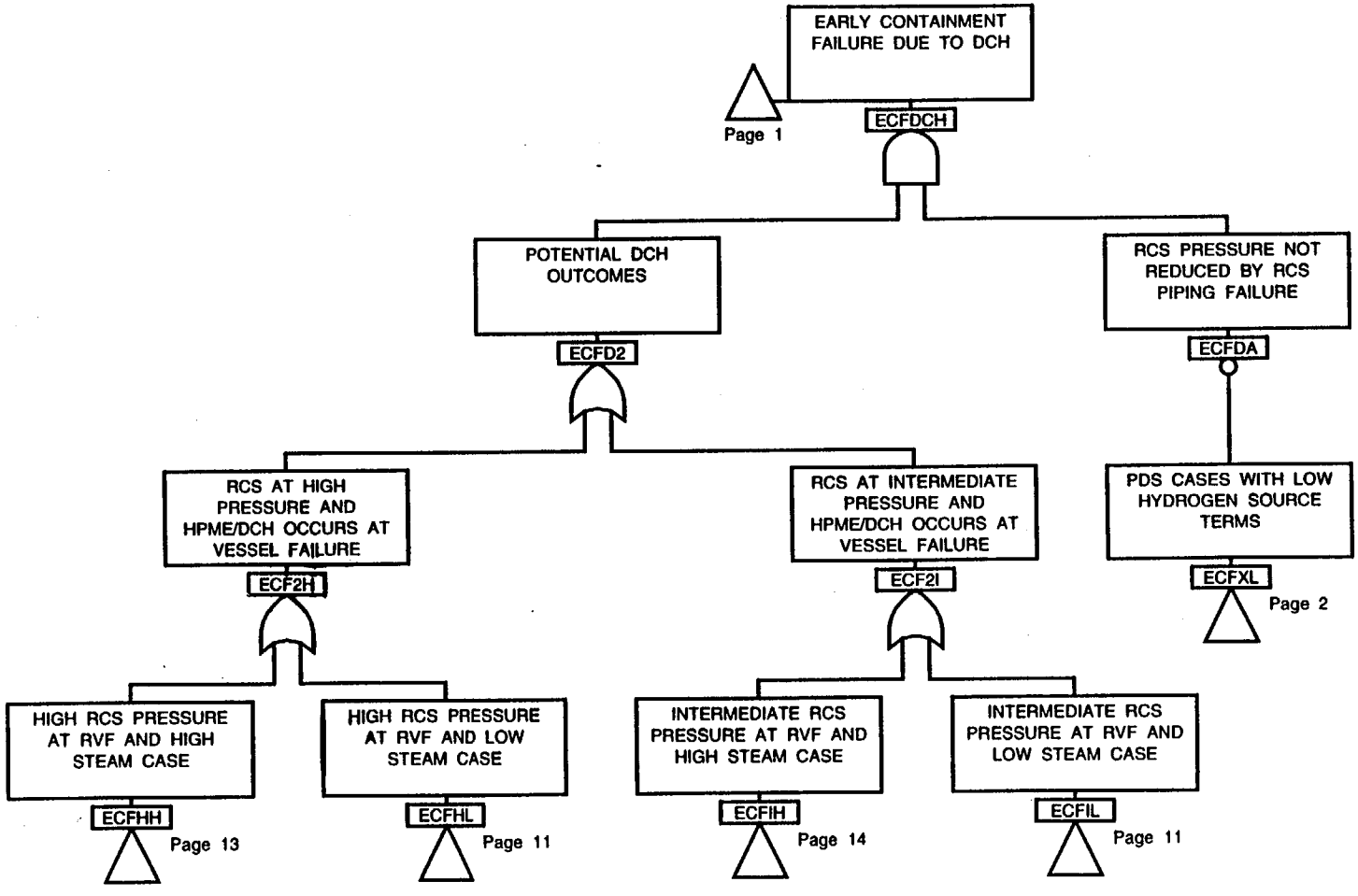


Figure 4–28
Early Containment Failure (page 11 of 15)

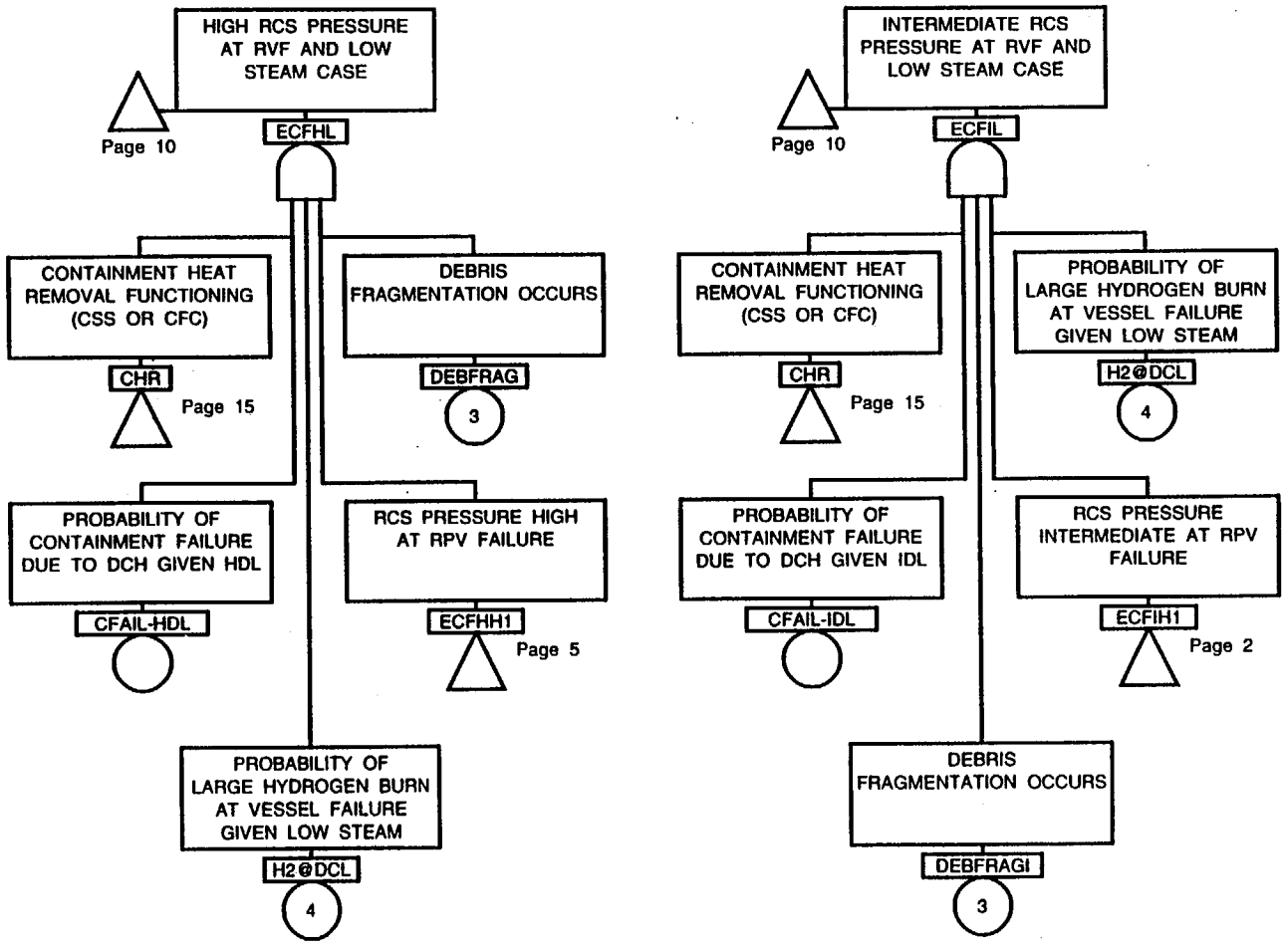


Figure 4–28
Early Containment Failure (page 12 of 15)

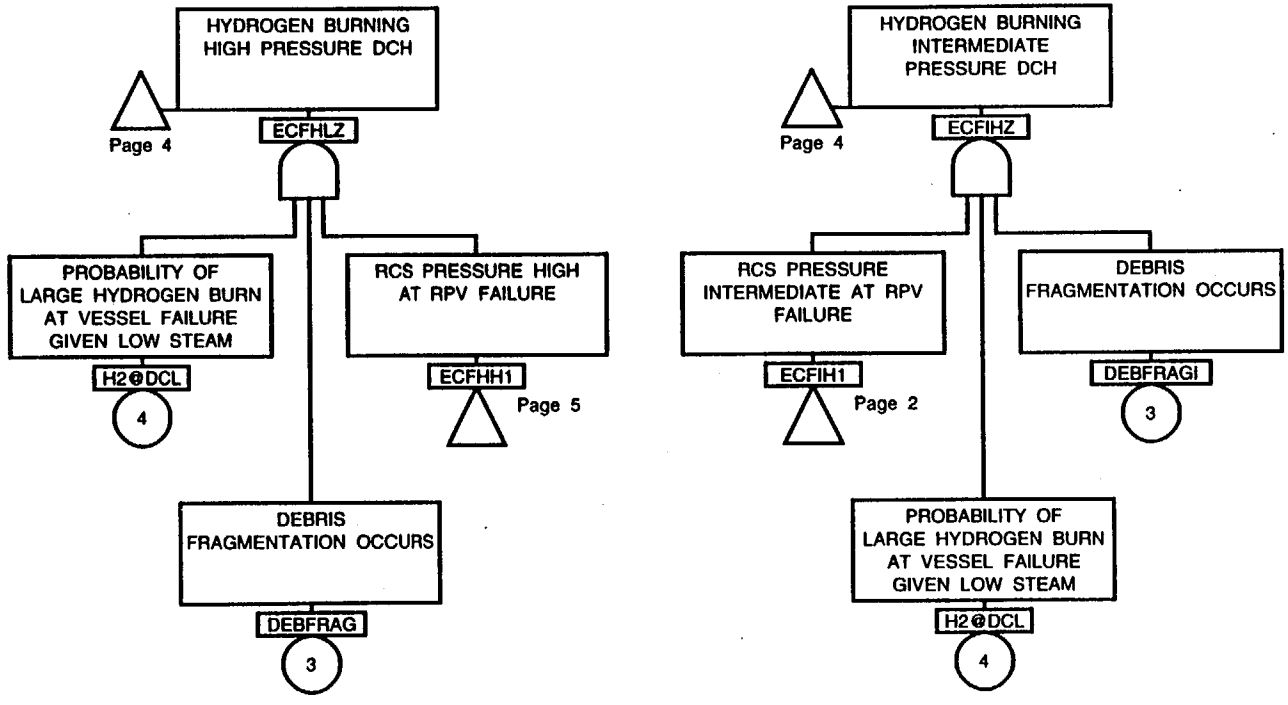


Figure 4–28
Early Containment Failure (page 13 of 15)

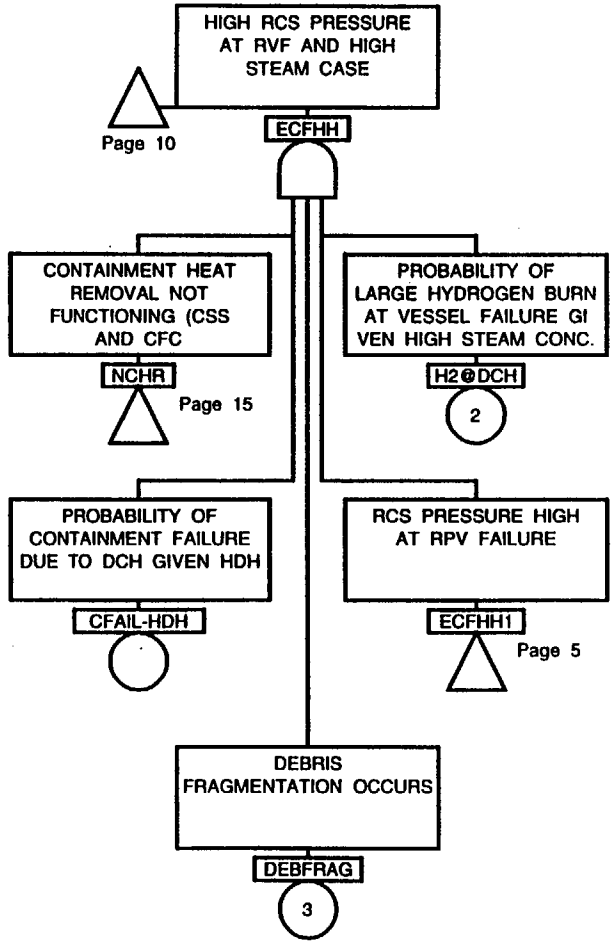


Figure 4–28
Early Containment Failure (page 14 of 15)

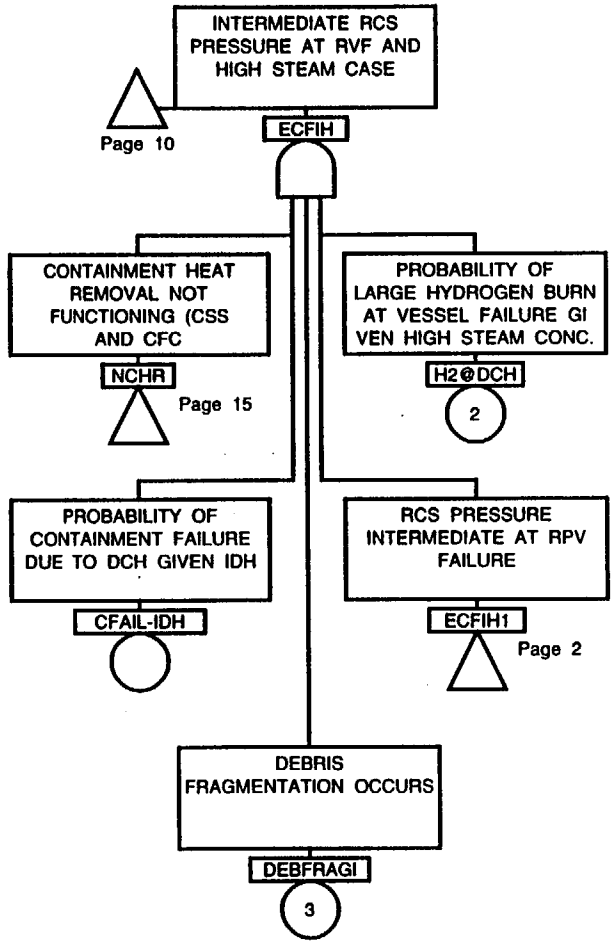
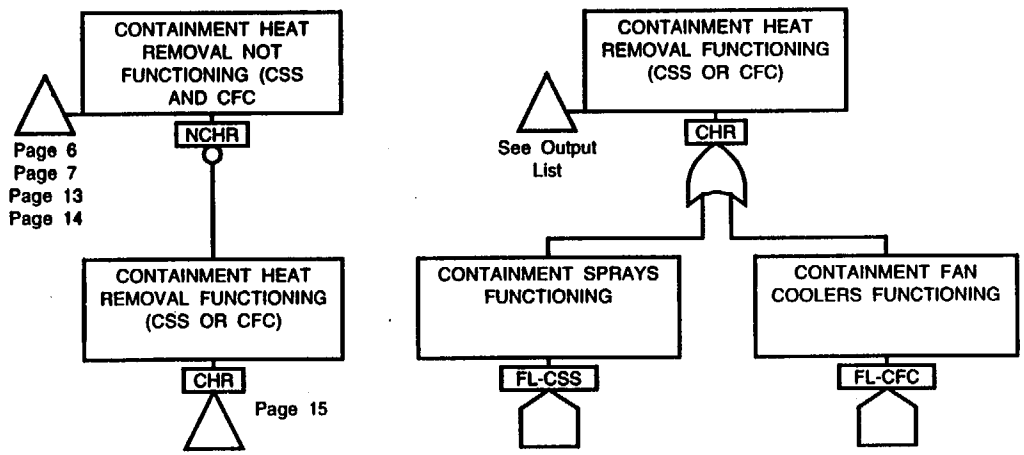


Figure 4–28
Early Containment Failure (page 15 of 15)

CHR Outputs:
Page 15, Page 3, Page 6, Page 7, Page 11, Page 11



If the RWST is not injected, the potential for liner attack is higher. The drain holes will be only partially filled and much less water will be available for cooling the debris. Therefore, liner attack may occur either at the time of vessel failure or after the available water has boiled off assuming a large quantity of core material is available. As stated initially, the escaping debris may be fragmented which will promote debris quenching. If a substantial amount of the debris is fragmented, early debris quenching is expected and liner failure is not expected.

In the event that the debris does interact with the liner, it is possible that the liner may fail but a loss of containment will not occur because the debris will freeze in the failure location and maintain containment. This idea has been proposed in NUREG/CR–5423 and is considered reasonable for this arrangement and an event is provided to address this possibility.

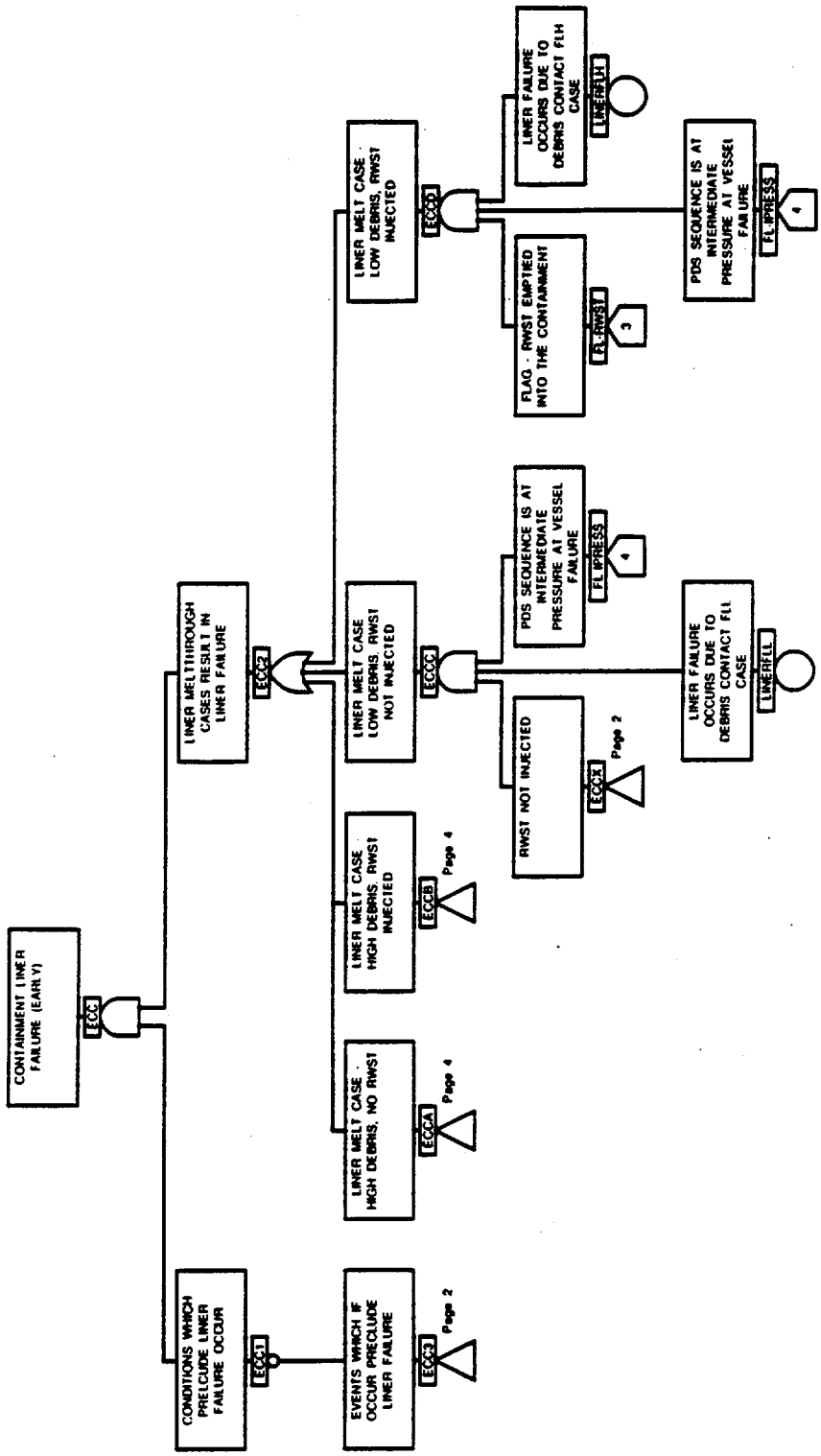
Based on these insights, the liner failure model was developed (see Figure 4–29). The model includes events which address RCS pressure at vessel failure, the status of the RWST, and the potential for the debris failing the liner. For low pressure cases the material is not expected to leave the cavity and liner failure is not expected. If the RWST has been injected into the containment there is a high likelihood of quenching and liner failure is not expected. The chance of liner failure is highest for the case with both high pressure and the RWST not injected.

DBC – Ex-vessel Cooling Successful

This event addresses whether the debris once relocated outside the RPV can be cooled, and considers the complex interaction between the debris bed and available water. The ability to remove the energy from the containment, by steam condensation, is addressed in later questions. Considerable uncertainty exists as to the ability of an overlying pool of water to remove sufficient heat from the debris to induce freezing of the debris and to stop concrete ablation. For SHNPP the potential for debris cooling is helped by a large cavity spreading area which reduces the debris thickness. Given debris cooling and containment cooling, the debris will cool until the material freezes and the accident sequence terminated.

Along with determining if the accident is terminated, the degree to which debris cooling is successful is important in assessing other long term containment issues such as late overpressure. If the heat transfer between the debris and the overlying pool of water is relatively low or if no water pool exists, steam generation will be limited. In contrast, heat transfer adequate to ensure debris cooling will require substantial steam generation which must be condensed in order to maintain containment pressure within acceptable levels. Failure to achieve a coolable debris bed also effects late hydrogen concentrations and the potential for basemat failure.

Figure 4–29
Containment Liner Failure (page 1 of 4)



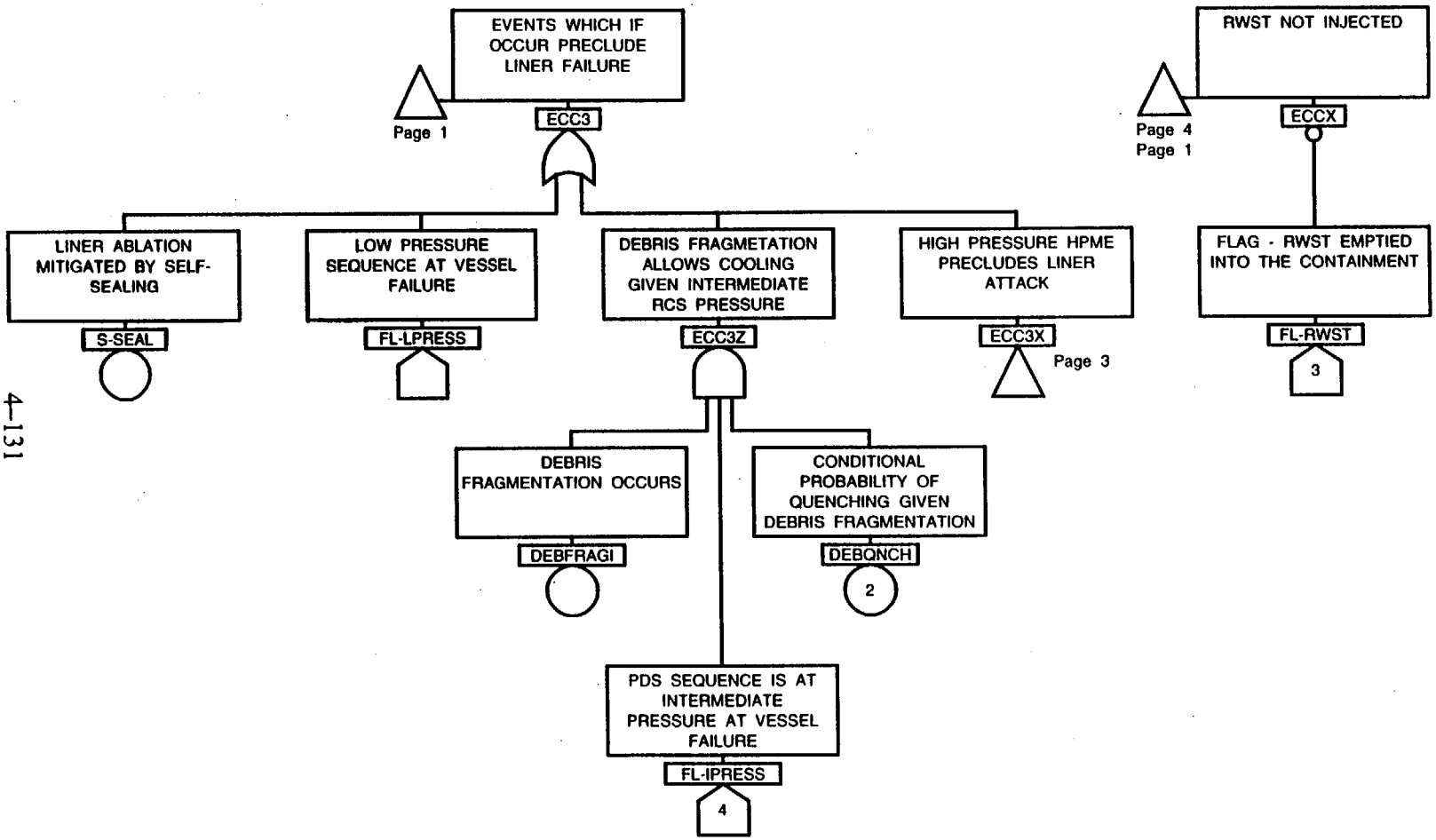


Figure 4–29
Containment Liner Failure (page 2 of 4)

Figure 4–29
Containment Liner Failure (page 3 of 4)

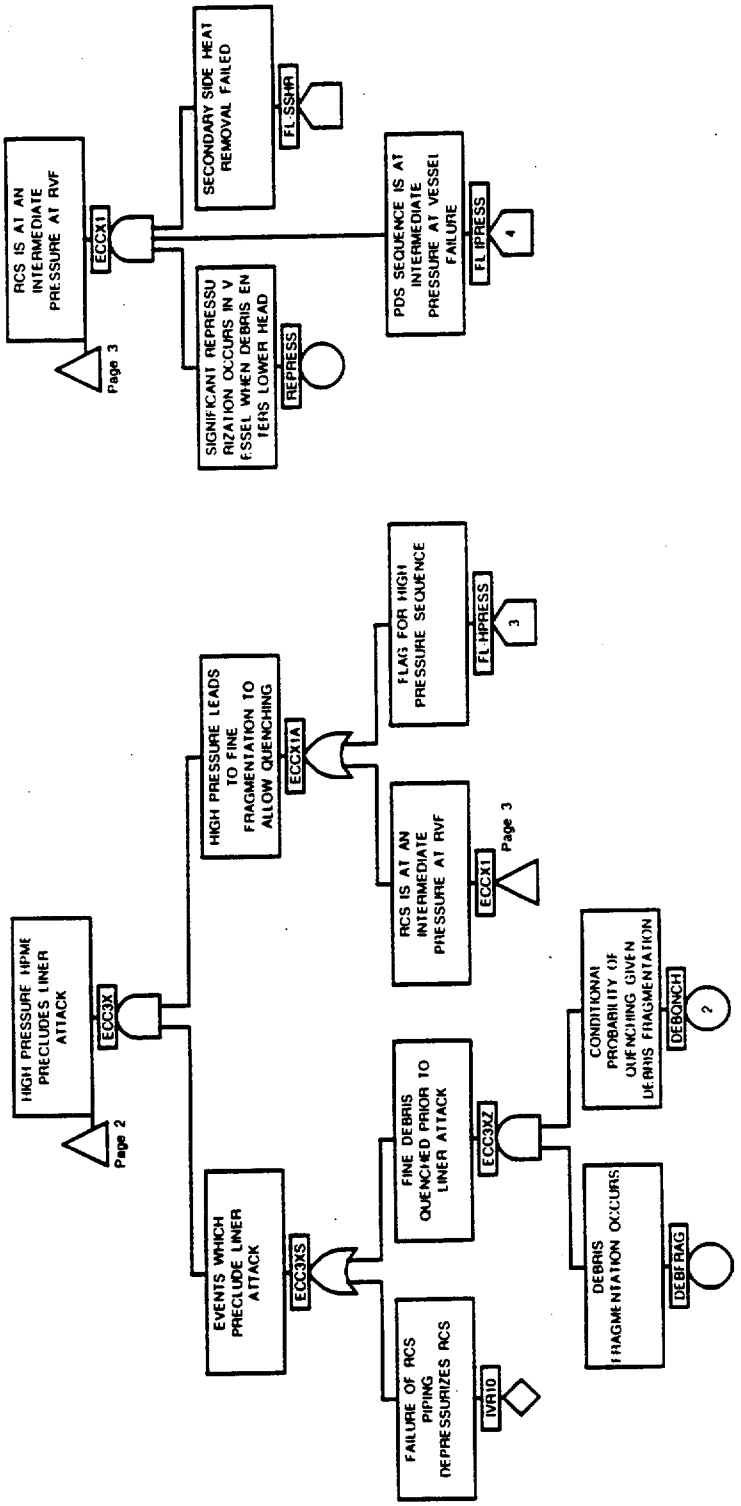
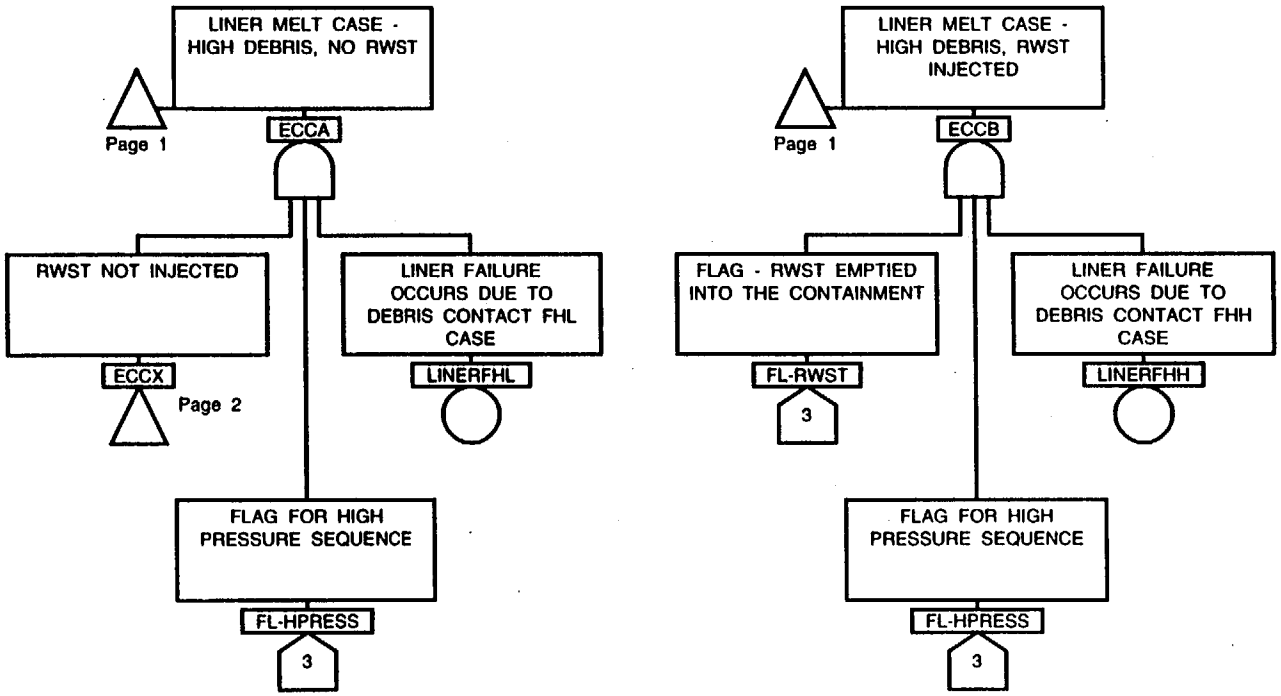


Figure 4–29
Containment Liner Failure (page 4 of 4)



Many factors, some not well understood, are important in determining whether the debris is coolable. Section 4.2 provides a summary of the issues related to debris cooling and a discussion of the issue as related to SHNPP. It is generally accepted that two of the more important, and possibly controlling, factors associated with debris cooling are the availability of a pool of water over the debris and the ability to spread the debris over a large area. Additionally, the RCS pressure at the time of RPV failure will have an impact on the potential for spreading the debris and debris cooling.

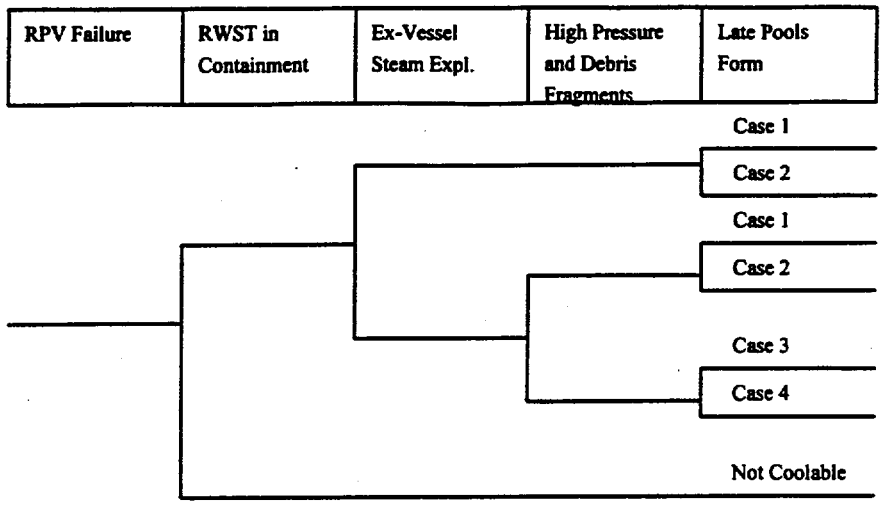
The presence of water in the cavity at the time of vessel failure may result in the exiting debris being partially quenched as it contacts the water and fragmentation may occur due to steam explosions. If the debris can be broken into fragments instead of forming a debris pool, the ability to cool the debris is enhanced. For SHNPP the presence of curbs around the entrance to the cavity results in little water being present in the cavity for cases without the RWST being injected into the containment. As such, debris located in the cavity would not be considered coolable. For cases with the RWST injected, the cavity would contain considerable water and debris cooling should be enhanced.

The RCS pressure at the time of vessel failure will also influence the potential for debris cooling. If the debris ejection is energetic, the potential for dispersal is high. Debris dispersal improves the changes for successful debris cooling. Sequences which involve HPME or ex-vessel steam explosions in the cavity are examples of when dispersal can occur. In contrast, if the debris material release is more driven by gravity, then the potential exists for the debris to stream down through any water and form a thick pool under the RPV. Depending on the quantity of water present in the cavity the debris material could be partially quenched and fragment which could promote cooling.

Dispersal of the debris and the presence of water in the cavity provide a situation which is likely to promote debris cooling for the initial debris which exits shortly after RPV failure. Later in the event, however, the remaining material in the RPV will relocate to the cavity. This material, up to roughly 50% of the total debris mass, could form a pool under the RPV which would preclude debris cooling. Thus, two distinct time regimes must be considered and are important to the establishment of debris cooling.

Figure 4–30 illustrates the possible outcomes for different situations associated with debris cooling. The factors considered include the status of the RWST, RCS pressure at RPV failure, potential for ex-vessel steam explosions, and the potential for forming deep debris pools which cannot be quenched.

Figure 4–30
Debris Cooling Outcomes Addressed in CET



The four cases represent different situations related to debris cooling. Case 2 identifies the optimal conditions where the debris is dispersed, the RWST is injected, and no late pools of debris form. Case 1 is similar to case 2 with late pools forming. Pooling has the effect of mitigating any early debris spreading. The 50% of the debris mass released after RPV failure is assumed not to spread, and will form a thick pool directly under the RPV. The thickness of this debris bed isgnificantly reduces the changes for successful debris cooling. Thus, the ability to disperse debris early is not important if pooling occurs, and the potential for cooling is driven by late debris releases. Cases 3 and 4 do not have debris dispersal and are similar with the exception of the presence or absence of late pools. If the RWST is not injected, cooling is assumed not possible. The supporting fault tree logic (see Figure 4–31) for this question addresses the basic issues associated with debris cooling. Because of uncertainties in the success of debris cooling outside the vessel, guidance provided in NUREG–1335 indicates that the IPE submittal must address both coolable and non-coolable alternatives for ex-vessel cooling. Due to the available areas which result in small debris thickness and the high reliability of cavity flooding, the best estimate assessment assumes that the debris is coolable if the conditions for cooling are met. The RWST flag can be used to examine the possible outcomes if debris cooling is conservatively assumed to be impossible.

LHB – Late Hydrogen Burn Mitigated

The event addresses the potential for hydrogen burning after vessel failure. Late hydrogen burns could occur as a result of recovery (containment cooling) or due to additional hydrogen generation from ex-vessel sources.

Figure 4–31
Debris Cooling Fails (page 1 of 6)

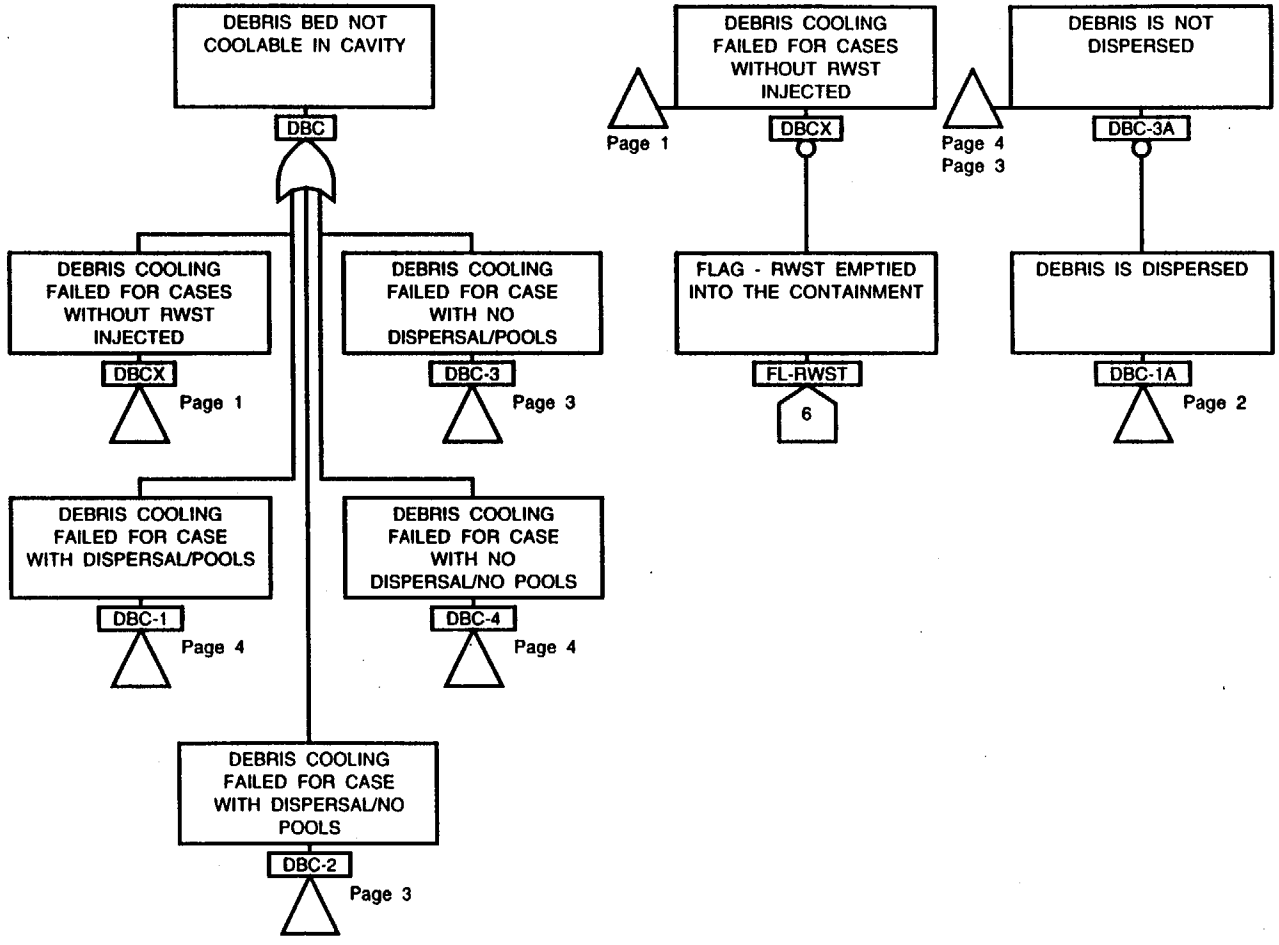


Figure 4–31
Debris Cooling Fails (page 2 of 6)

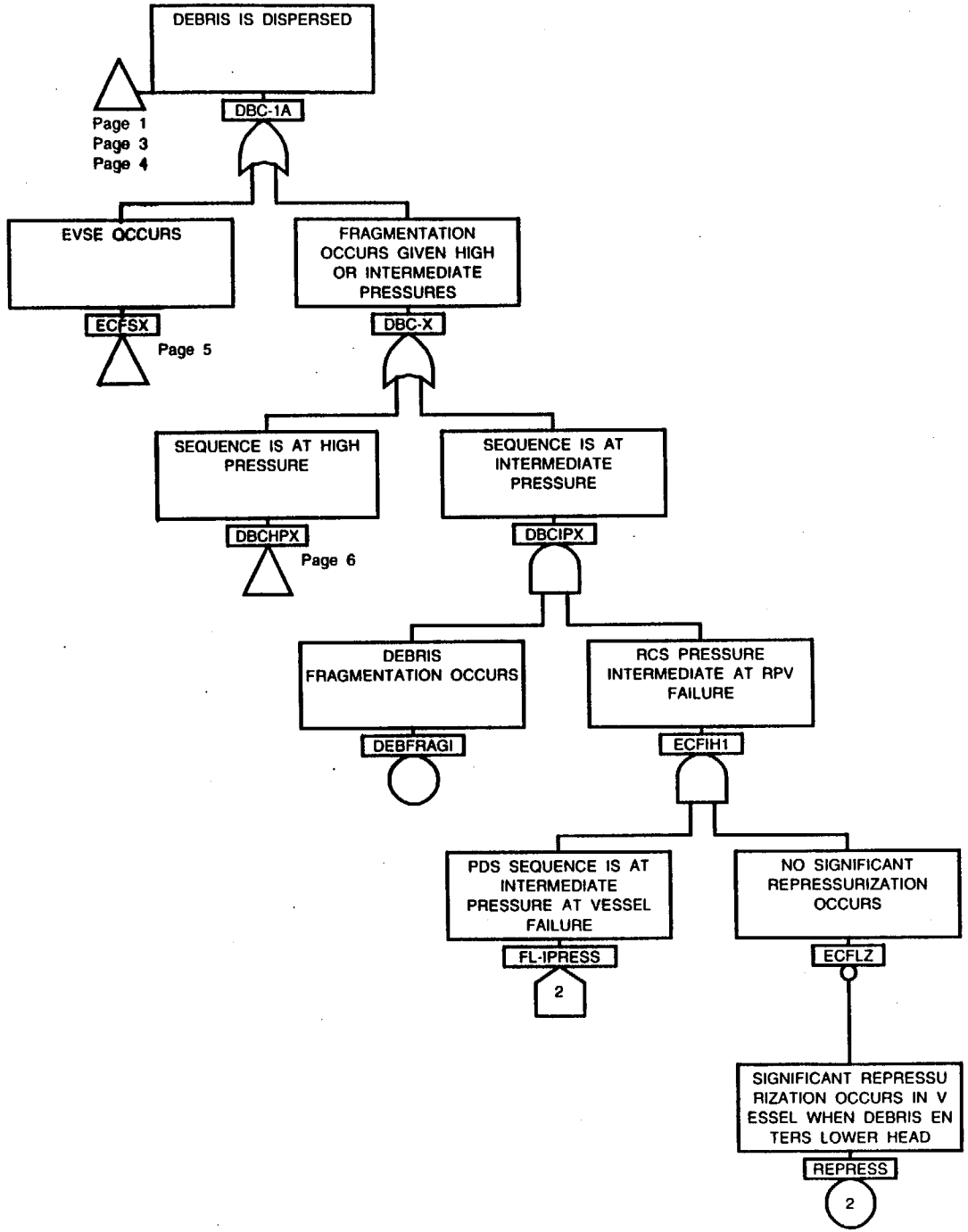


Figure 4–31
Debris Cooling Fails (page 3 of 6)

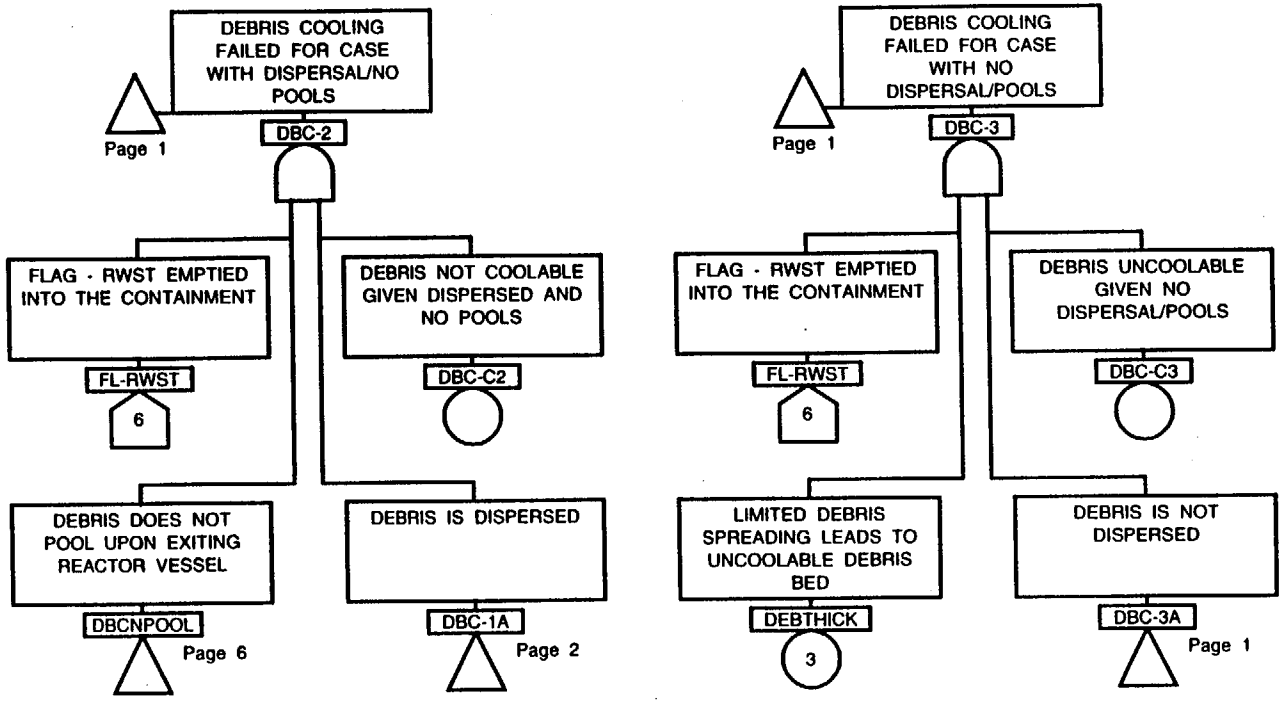


Figure 4–31
Debris Cooling Fails (page 4 of 6)

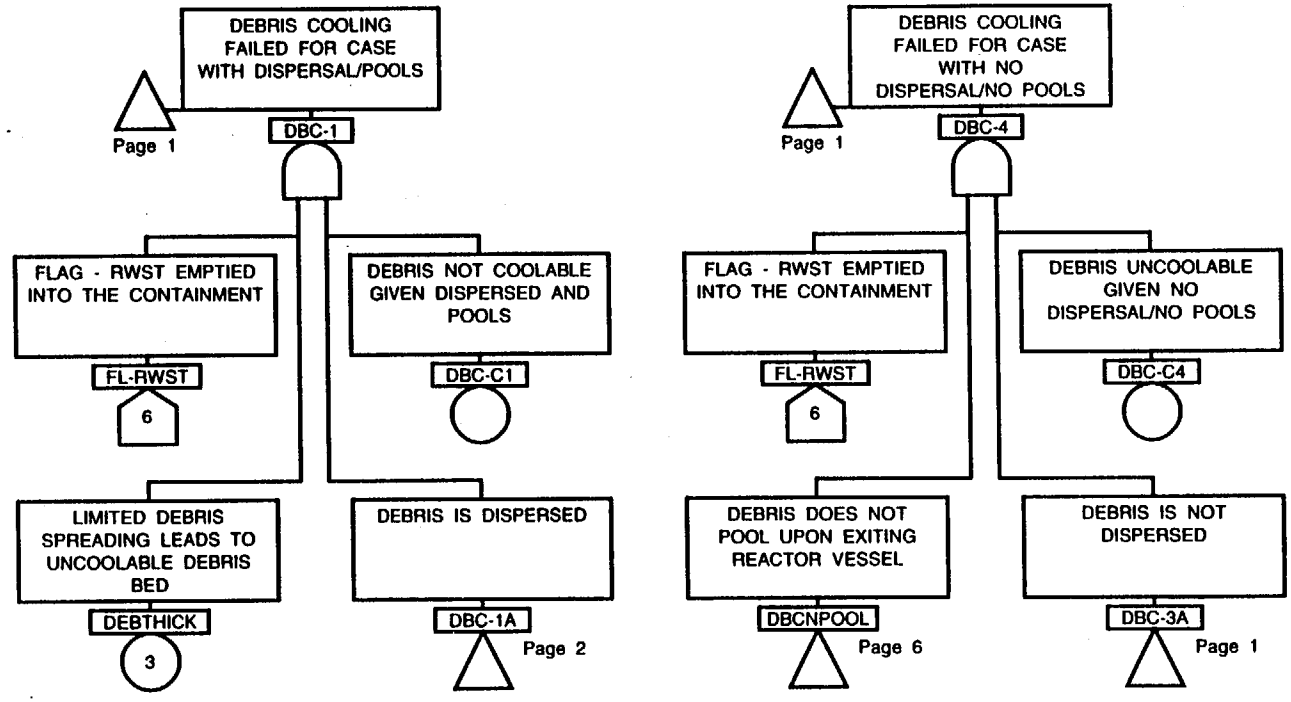


Figure 4–31
Debris Cooling Fails (page 5 of 6)

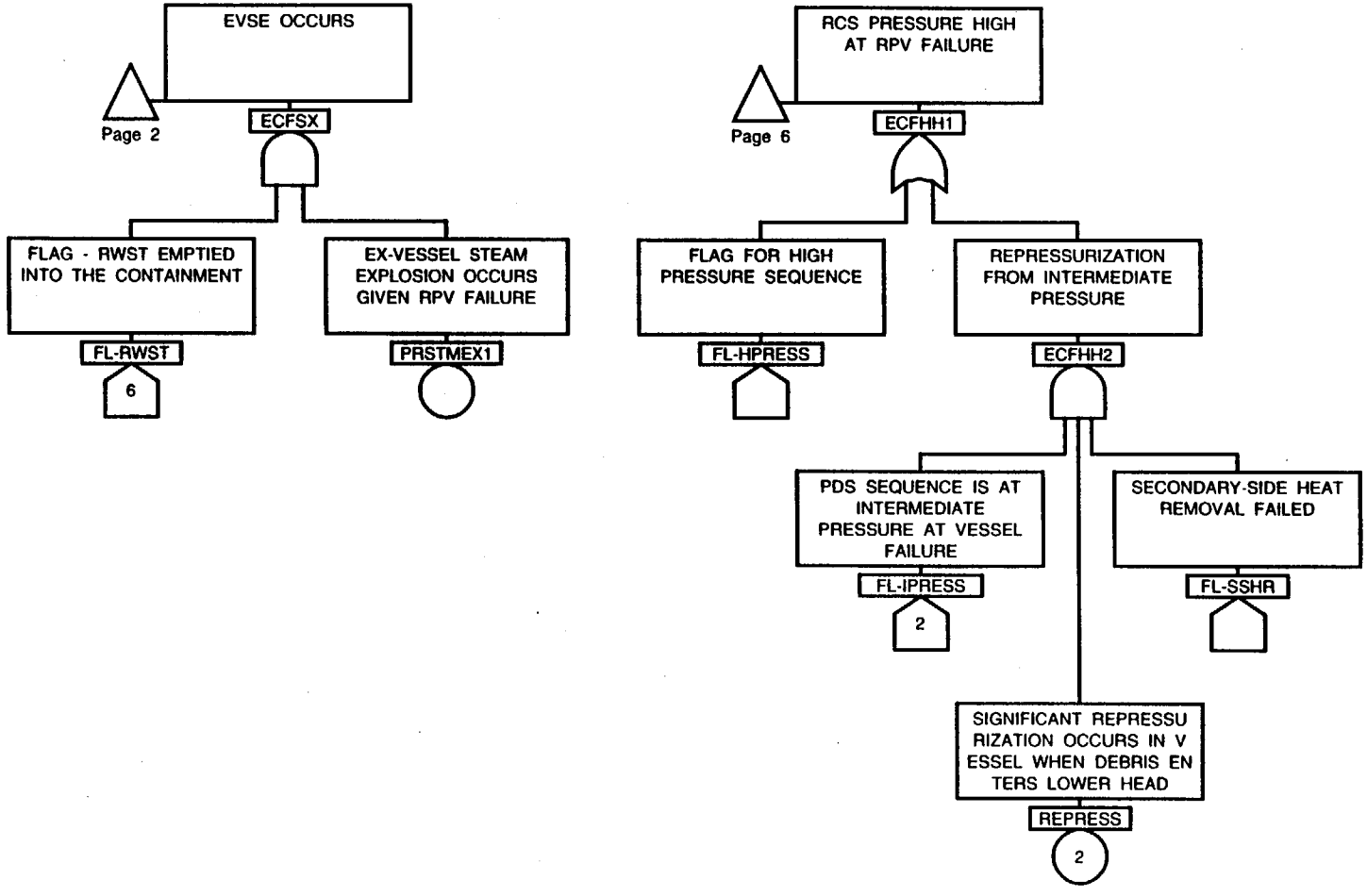
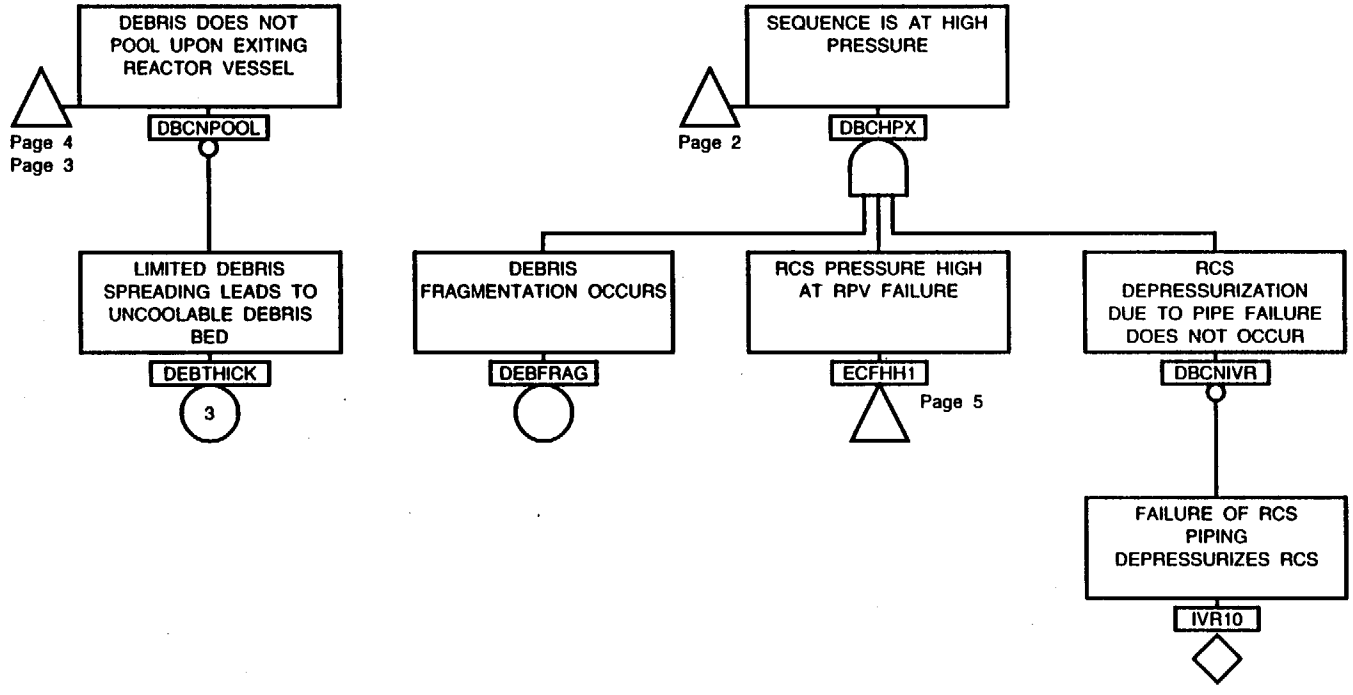


Figure 4–31
Debris Cooling Fails (page 6 of 6)



The restoration of containment heat removal prior to containment failure is the preferred means to reduce containment pressure by means of steam condensation. During this process, the steam concentration inside containment may be reduced to a level that the containment is deinerted and hydrogen burns are possible. With somewhat accelerated condensation and the availability of an ignition source, hydrogen concentrations may be sufficient to support detonation with the capacity to challenge containment integrity. The possibility of a burn or detonation and the resultant containment loading is based on several factors which include whether early burning occurred, ex-vessel hydrogen production, and the effectiveness of the sprays or fans in providing good mixing between compartments.

If burns occur prior to vessel failure, the containment hydrogen concentration will be controlled by ex-vessel production. For cases with debris cooling, the amount generated will be quite low and the potential for hydrogen burns will be low. Given core-concrete attack, the quantity of hydrogen generated ex-vessel will be substantial. It is possible that the total amount generated can exceed that expected from in-vessel production alone.

Cases without containment heat removal may result in large hydrogen source terms by combining ex-vessel production with in-vessel hydrogen production, i.e., no early burns. The steam concentration, however, should be more than adequate to retain steam inerting and preclude late burns without containment cooling recovery.

If ex-vessel hydrogen production remains unchecked and containment heat removal is present, i.e. limited steam in the containment, the potential exists for hydrogen concentrations necessary for a burn to occur. For example, if the heat transfer between the debris pool and the overlying water is small and containment cooling is present, then hydrogen will continue to be made due to concrete attack but the steam concentration will be small. For this case, hydrogen burns may be possible and the integral hydrogen source term may be large. Hydrogen burns which occur for this situation or following recovery may involve elevated concentrations of hydrogen. The possibility of detonation, therefore, should be considered.

The probability of a hydrogen detonation is quite small, however, since the containment steam concentration will either be low as the hydrogen is generated, or will be slowly reduced below the inerting limits. Both of these conditions support global burns as soon as conditions permit and before detonation limits are met. Further, geometric conditions within the SHNPP containment are not favorable for detonation due to the large areas of communication between compartments. Thus, a series of deflagrations seem more likely than a sudden ignition of a hydrogen concentration sufficient for detonation. DDT is also not expected since the geometric configuration is not conducive to DDT. The model (Figure 4–32) addresses the possibility of a late hydrogen deflagration given these two possibilities and includes events which address containment spray operation, debris

cooling, early hydrogen burning, operator recovery, and the presence of an ignition source.

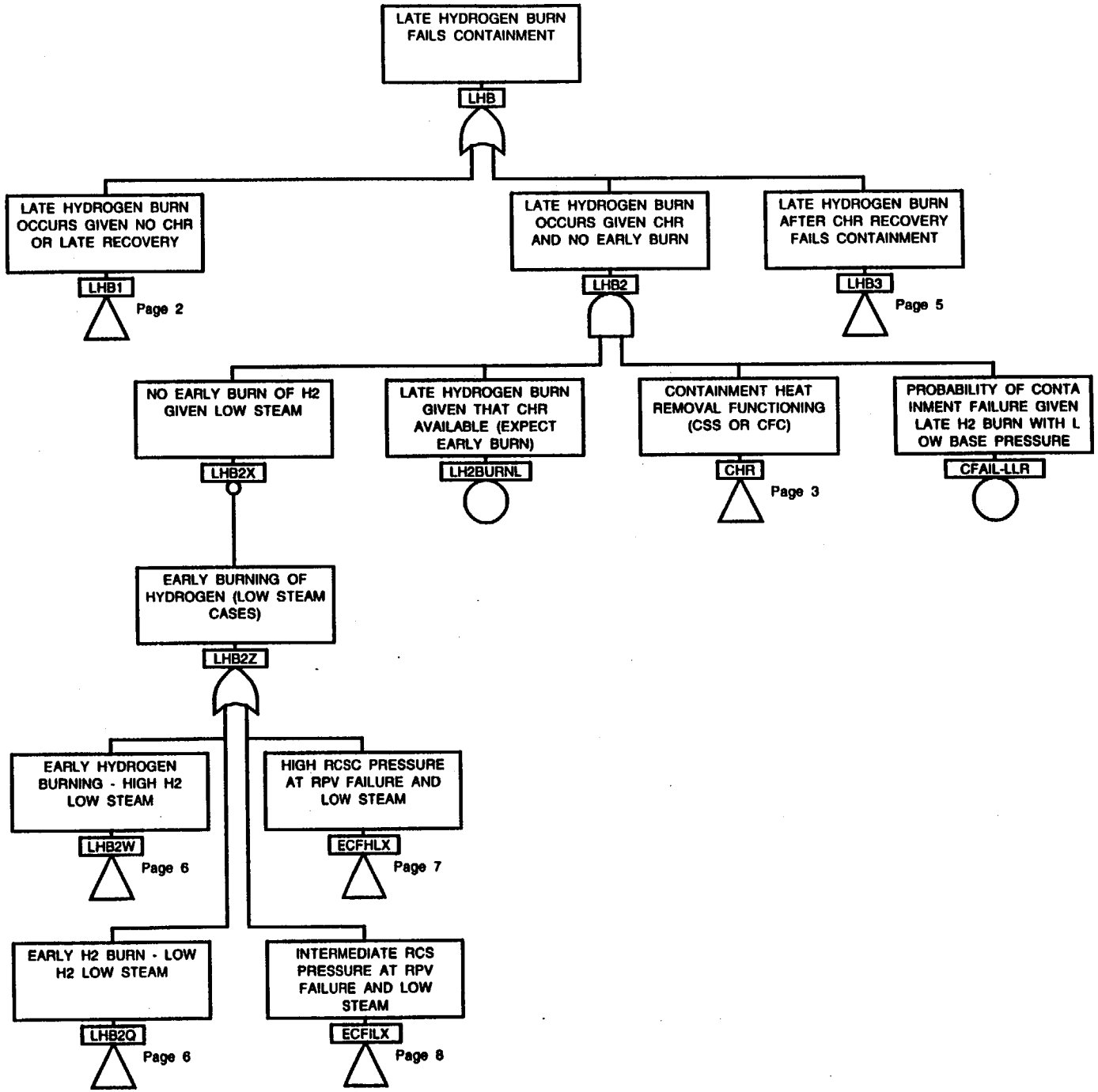


Figure 4–32
Late Hydrogen Burn (page 1 of 9)

Figure 4–32
Late Hydrogen Burn (page 2 of 9)

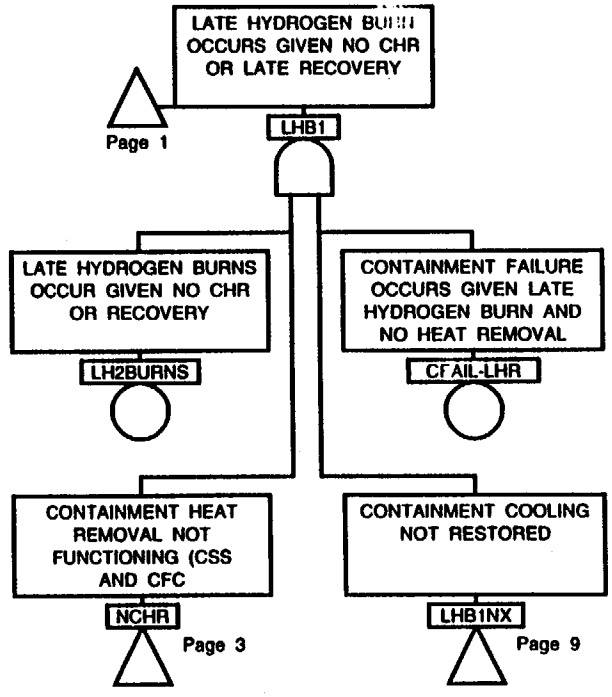


Figure 4–32
Late Hydrogen Burn (page 3 of 9)

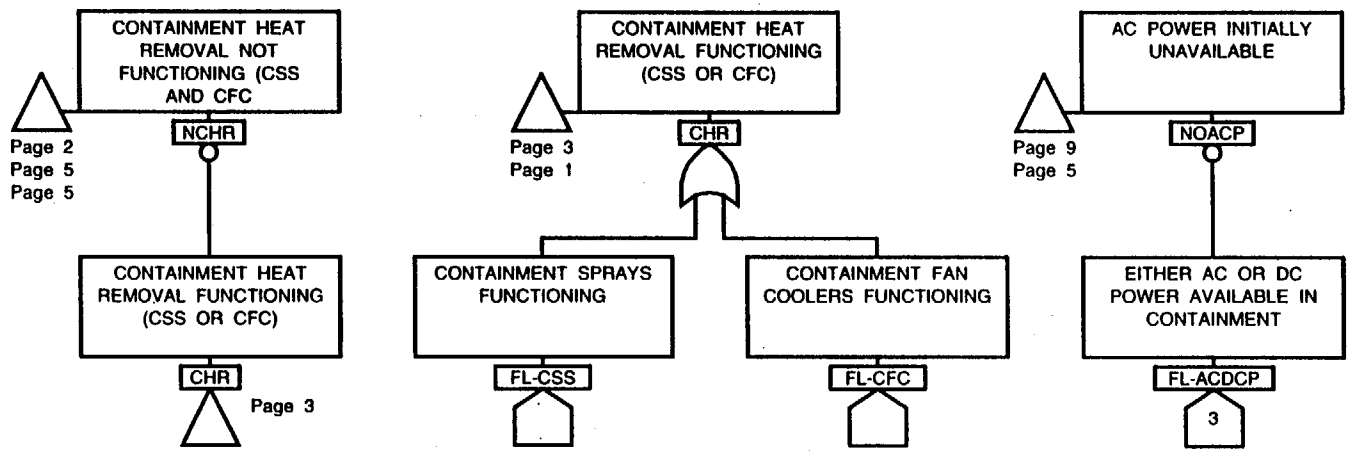
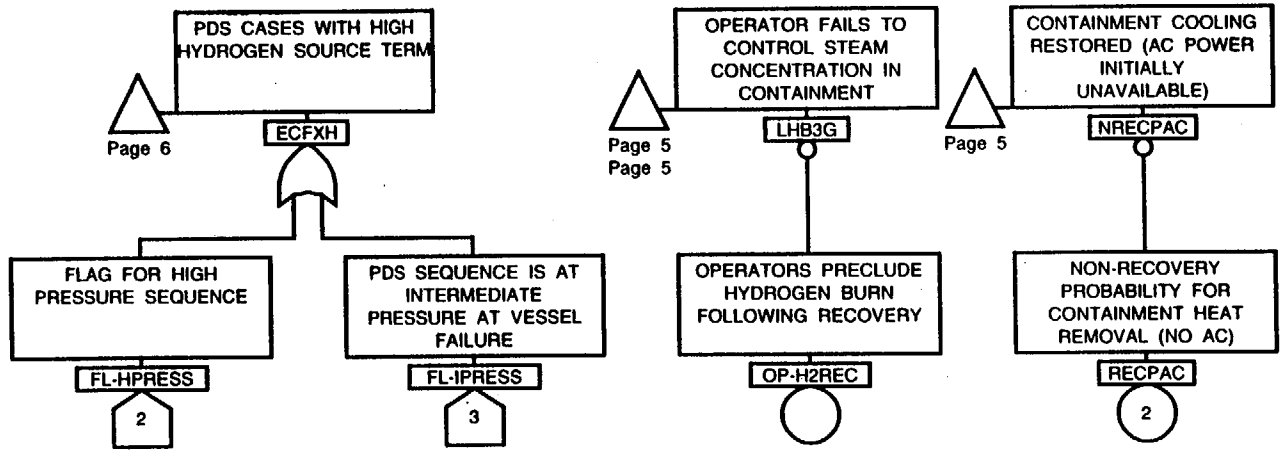


Figure 4–32
Late Hydrogen Burn (page 4 of 9)



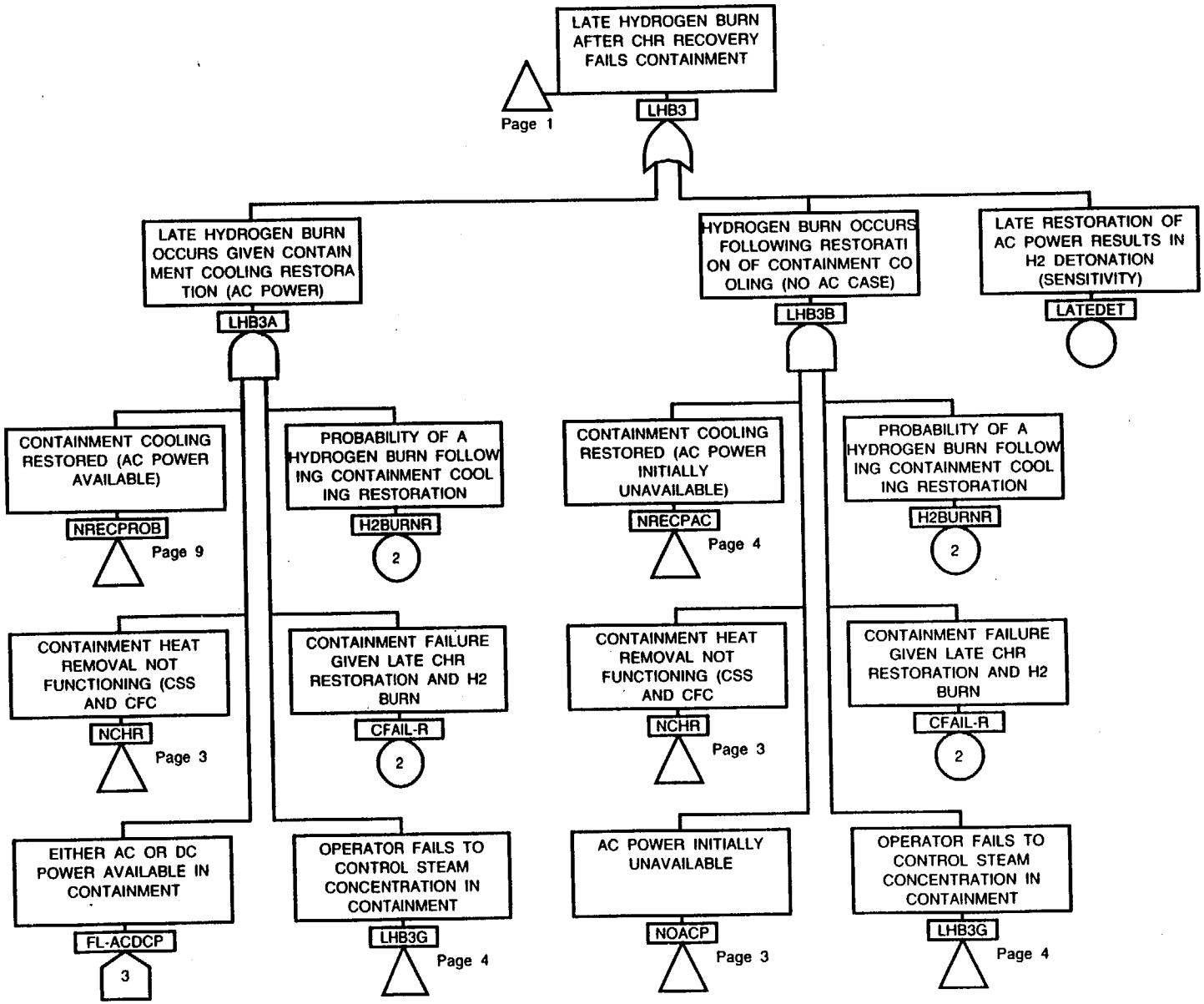


Figure 4–32
Late Hydrogen Burn (page 5 of 9)

Figure 4–32
Late Hydrogen Burn (page 6 of 9)

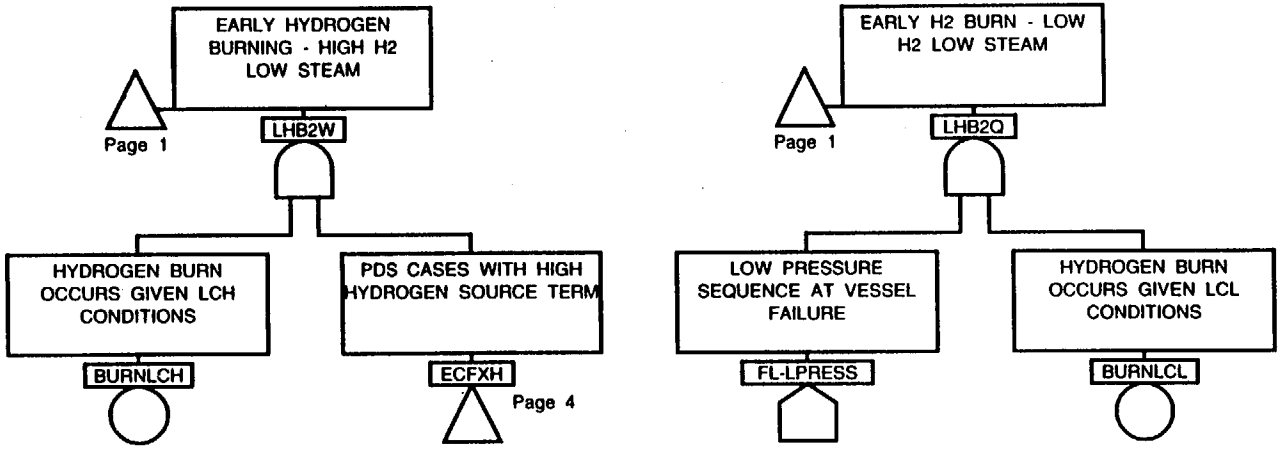


Figure 4–32
Late Hydrogen Burn (page 7 of 9)

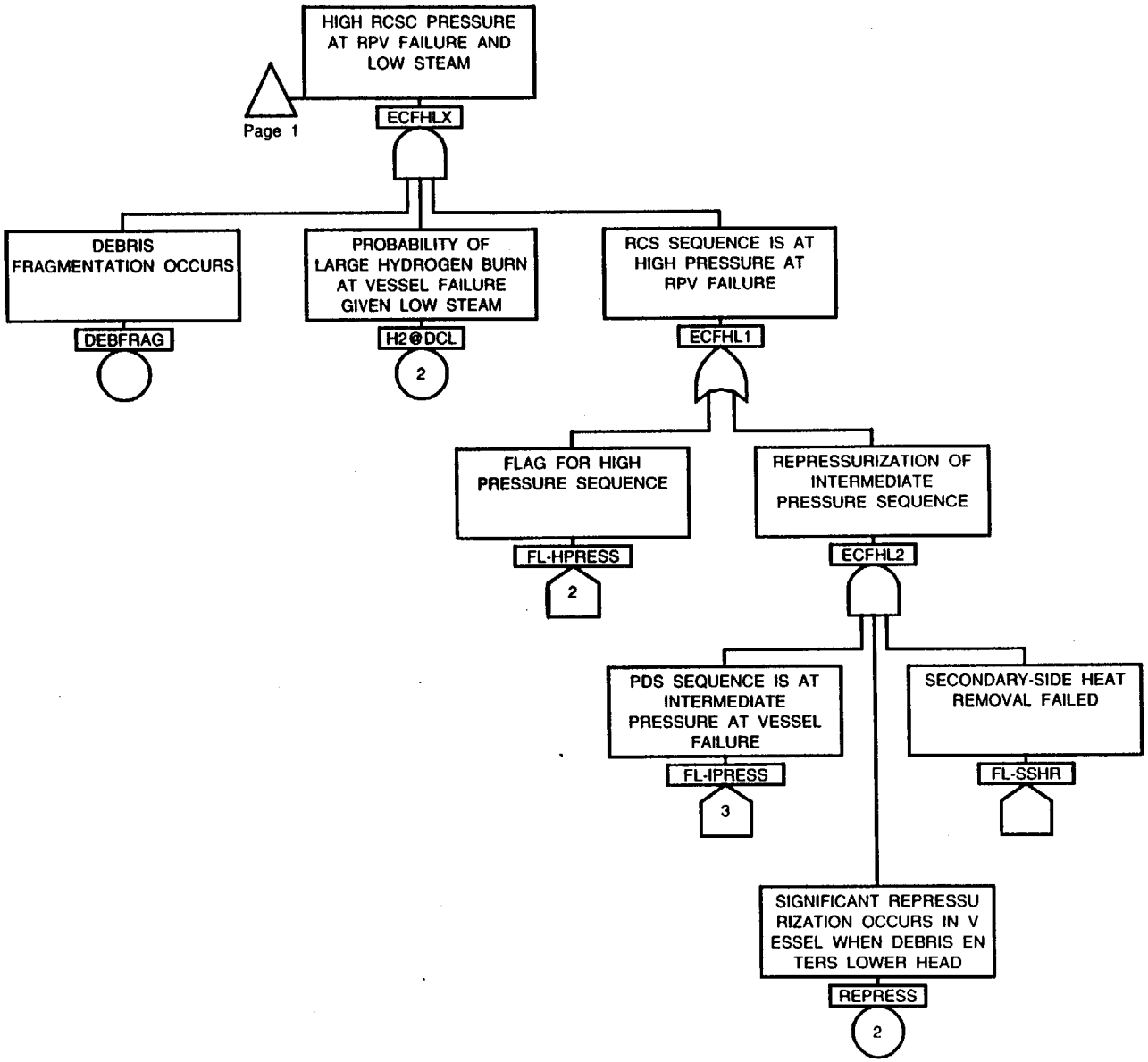


Figure 4–32
Late Hydrogen Burn (page 8 of 9)

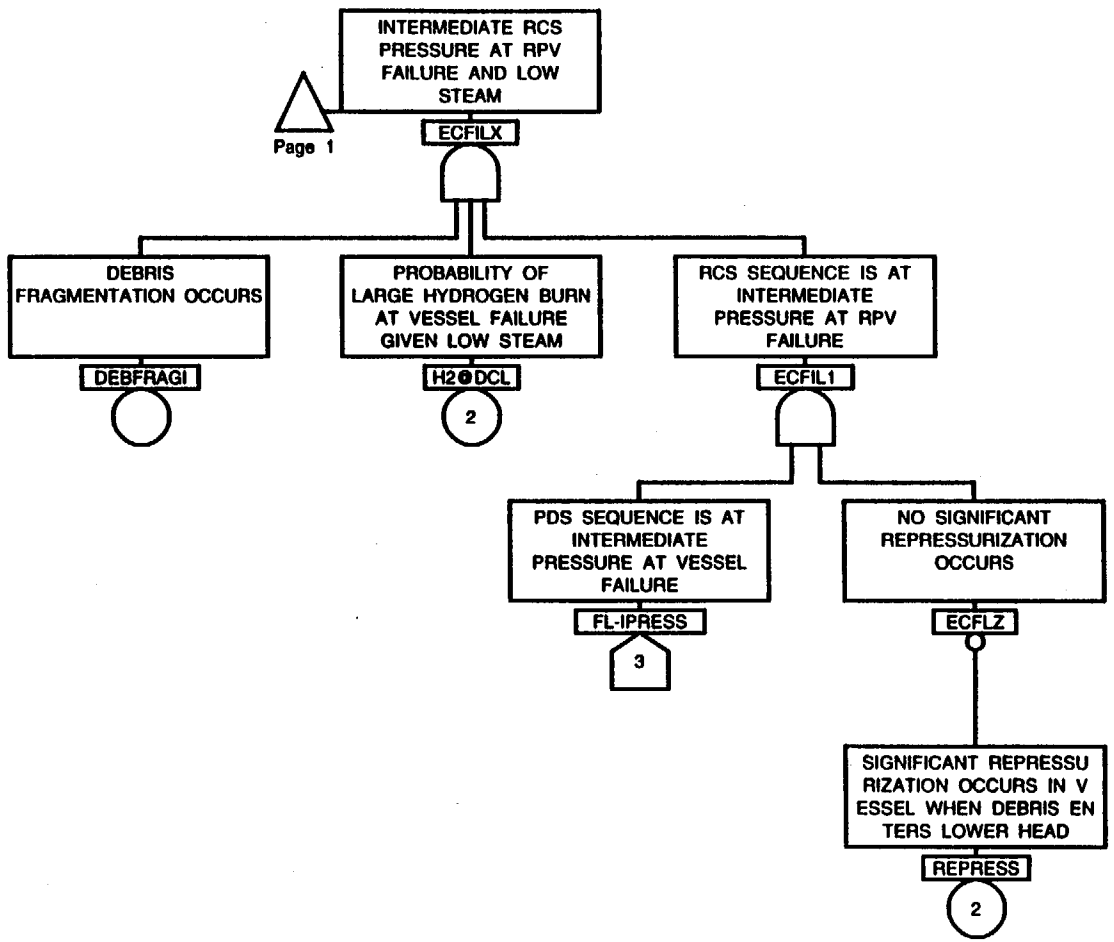
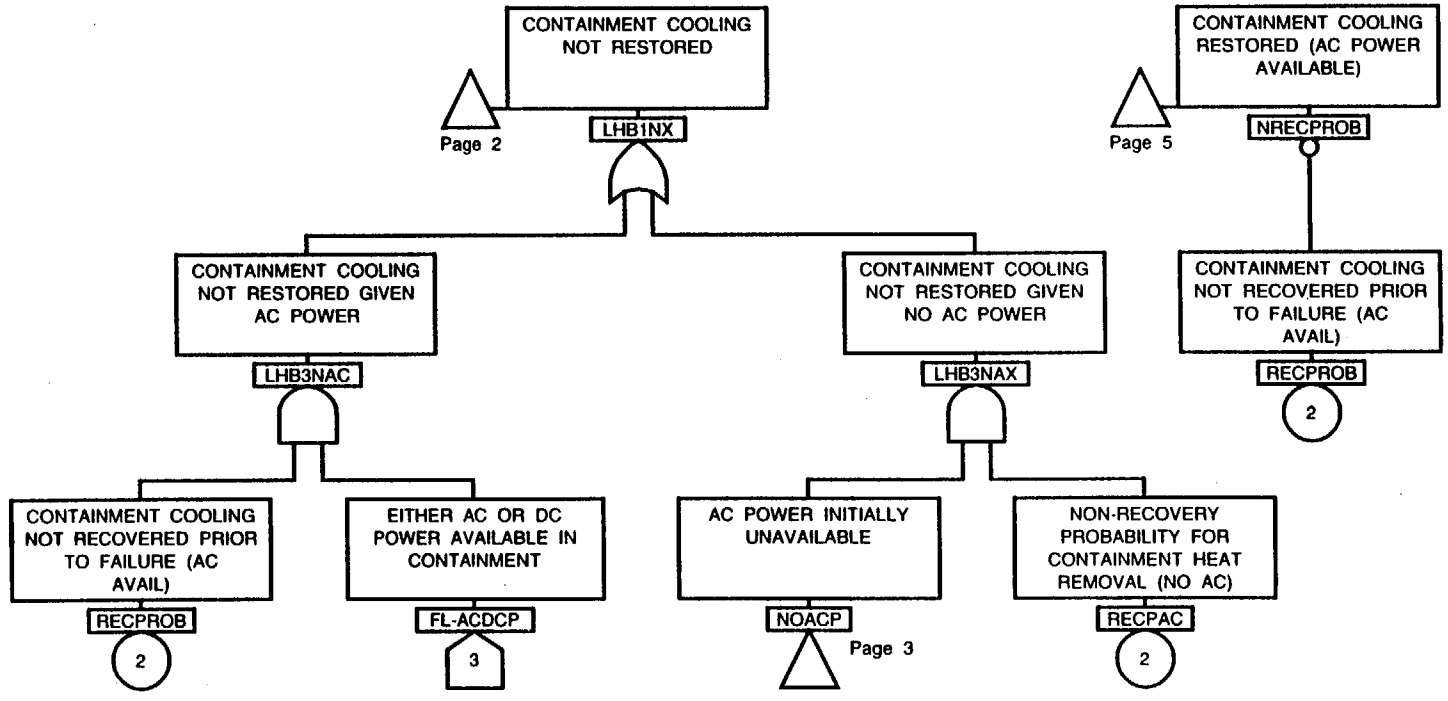


Figure 4–32
Late Hydrogen Burn (page 9 of 9)



LOPS – No Late Containment Failure (Steam Generation)

This event and the following event (VLOPN) address the different long term overpressure failure modes which are related but are separated by timing. In both cases no containment heat removal is present. The rate of pressurization, however, is different for cases with a water pool covering the debris than for cases without a water pool covering the debris. For cases without a water pool, the steam addition after dryout is limited to the steam generated during concrete attack. The different steaming rates of these two cases results in a significant timing difference which is distinguished in the CET by these events. This top event addresses the case with a water pool covering the debris.

If water is covering the debris and the debris is coolable, the containment will be pressurized by the steam generated as the water absorbs the decay heat present in the debris. If containment heat removal is not present and is not recovered, the containment pressure can increase until containment overpressure occurs. In order to maintain a water pool over the debris, the RWST must be emptied into the containment. Without the inventory contained in the RWST, the debris bed will dryout and containment pressurization will increase at a slower rate. Two different cases are addressed by the top logic and are related to the actions necessary to recover containment heat removal. If the loss of AC power is the cause of containment heat removal failure then the recovery involves the restoration of AC power prior to containment overpressure. The time available for recovery of AC power is determined based on the pressurization rate and the probability of non-recovery is determined by evaluating the AC power recovery curve for the period of time between RPV failure and containment overpressure.

For cases involving other failures of the containment heat removal system a recovery value is chosen based on prior studies and the time available for recovery. Figure 4–33 presents the top logic for this top event.

VLOPN – No Very Late Containment Failure (Slow Pressurization)

Given that the debris is not coolable or no water pool over the debris is present, CCIs will continue until the basemat below the debris is ablated. For the analysis, however, only the attack which occurs prior to containment overpressure is of interest. This event and the event which follows (BMF) are used to bin the two potential outcomes. The most likely event, based on pressurization and ablation rates, is given the greater chance of occurrence.

Figure 4–33
Late Containment Failure (page 1 of 2)

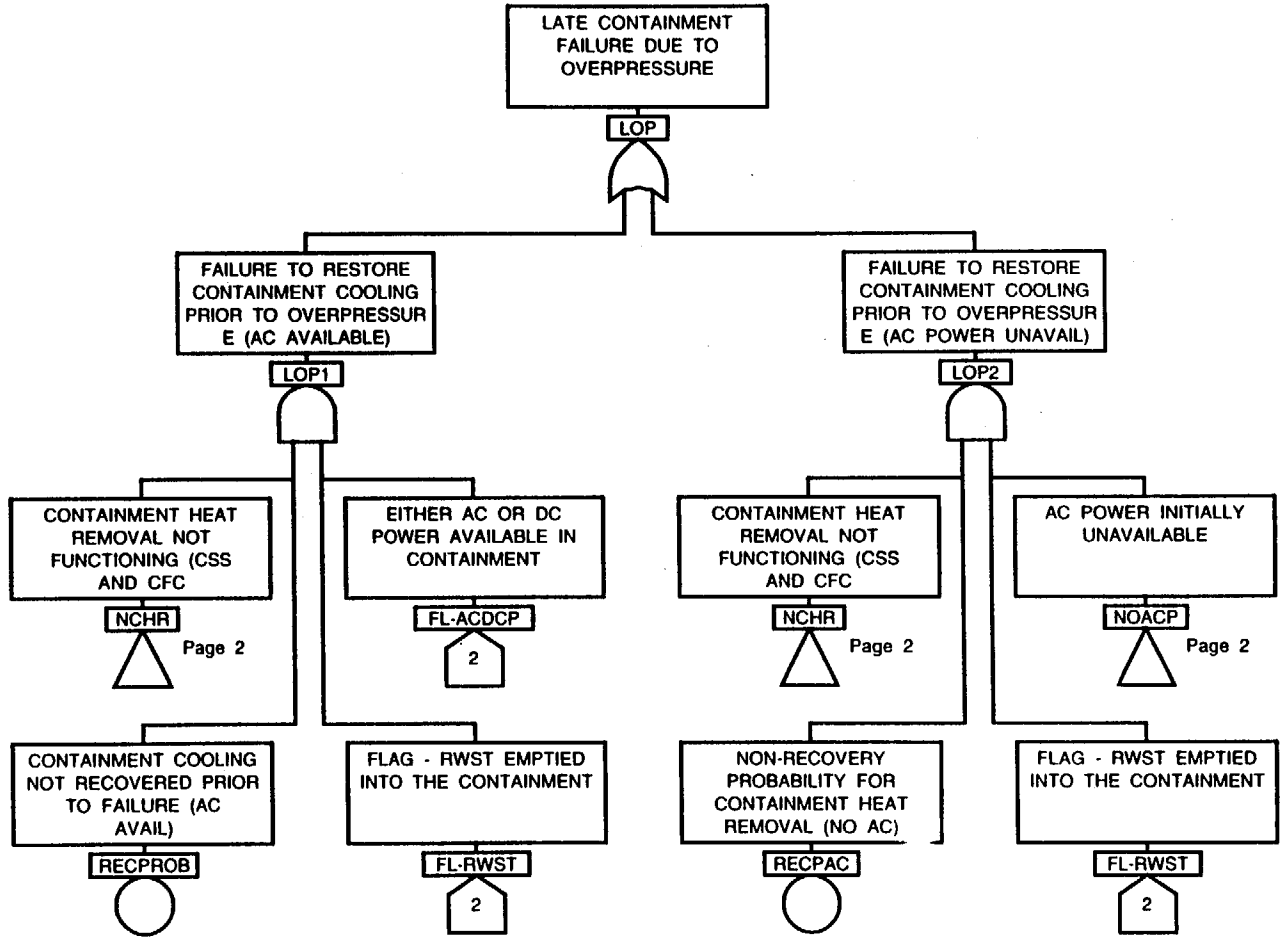
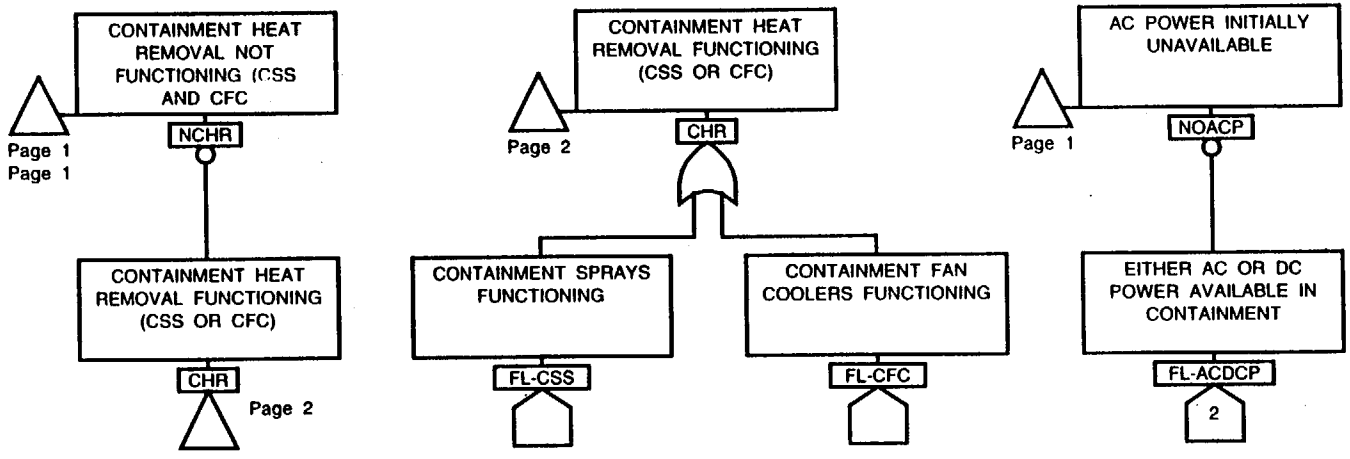


Figure 4–33
Late Containment Failure (page 2 of 2)



Either event is expected to require considerable time, greater than 40 hours, to evolve to the point of containment failure. The long times available for recovery provide considerable opportunities for operator action and recovery. Because of the importance of recovery to these sequences, two different situations are addressed. The first addresses the potential for overpressure failure given that AC power is available during the event. The model addresses the potential for restoring containment cooling given AC power is resent and weights this likelihood by the probability that basemat failure does not occur first. The second case is similar but assumes that AC power is not initially available and that recovery of AC power will result in the restoration of containment cooling which will preclude containment failure.

For either case, the restoration of cooling will preclude containment failure due to pressurization based on the cooling rate expected and the low heat load generated late in the event. It is important to note that recovery does not, however, preclude late hydrogen burns which is addressed by the prior event. The top logic for this event is illustrated in Figure 4–34.

BMF – No Containment Basemat Failure

This event is the complement to the very late overpressure event. If core concrete interaction continues unchecked, basemat failure may occur regardless of the status of containment cooling. The thickness of the basemat, 12 feet, ensures that considerable time will be available prior to containment breach. The failure of the basemat may not result in actual release since the basemat is located upon an indeterminate thickness of rock. Failure, however, is conservatively assumed to occur once the basemat is ablated.

If considerable concrete ablation occurs, the debris may be in a noncoolable configuration or may form a crust which could impede debris quenching. Conversely it may be possible that as the debris interacts with the concrete the decay heat will drop such that ablation will cease. An event is included in the model to address the potential that, even after recovery, the debris will not be coolable and basemat failure will eventually occur.

Two different cases, which mirror those described in the prior event, are provided to address recovery for cases with and without AC power. An event is provided under each case to provide for cases where the debris freezes prior to basemat failure. Figure 4–35 provides the top logic for this event.

Figure 4–34
Very Late Containment Failure

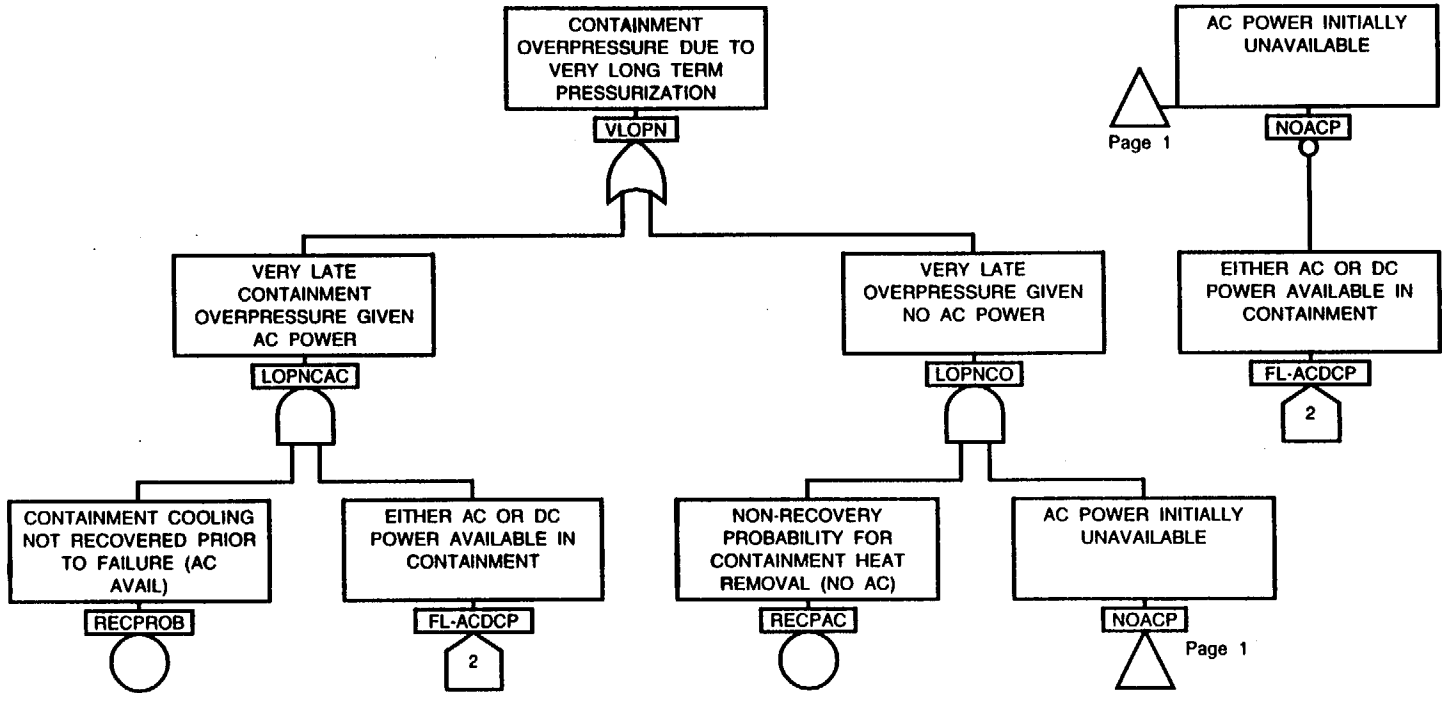
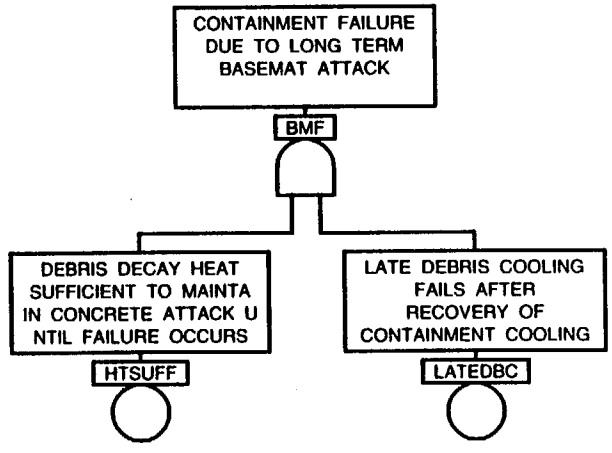


Figure 4–35
Basemat Failure



RVP – Late Revaporization does not Occur

This event addresses whether radionuclides initially deposited in the RCS are released into the containment late in the sequence, perhaps after containment failure, and result in a higher source term. The model addresses cooling of the radionuclides and the retention of the radionuclides within the RCS during core damage.

The amount of volatile radionuclides retained within the RCS is dependent on the type of accident sequence. Sequences which provide ample interfaces between the core and the containment, i.e., large RCS breaks, will have lower RCS retention and, therefore, less chance for late revaporization. Large LOCA sequences are the best example of a sequence which provides the necessary conditions to reduce RCS retention. If the radionuclides are released early into the containment, they may be scrubbed or may settle onto surfaces within the containment prior to containment failure. As such, they will not be available for revaporization release. This potential is accounted for in the model.

Radionuclide cooling within the RCS is another important issue related to revaporization. If cooling is available, the radionuclides deposited in the RCS may have insufficient energy to revaporize and leave the RCS. The presence of cooling will also cause much of the material evolved during core damage to migrate to cooler regions in the RCS such as the steam generator tubes. If cooling water is maintained to the steam generators, the radionuclides will adhere to the tubes and significant revaporization will not occur. The success of secondary-side heat removal is assumed to result in adequate cooling and provide high assurance that revaporization will be precluded. Figure 4–36 provides the top logic for assigning the likelihood of revaporization.

SRDC – Radionuclide Scrubbing (Debris Covered)

If the debris bed is covered by an overlying pool of water, and the release is not prompt, the source term will be lowered. The water pool will wash out, or scrub, a large portion of the released radionuclides. Analysis has indicated that if the RWST is injected into the containment the cavity and lower compartment will be flooded and the debris will be covered by a pool of water. For sequences which do not involve debris bed dryout, any release should be scrubbed and the source term reduced appropriately.

Figure 4–36
Revaporization Occurs (page 1 of 2)

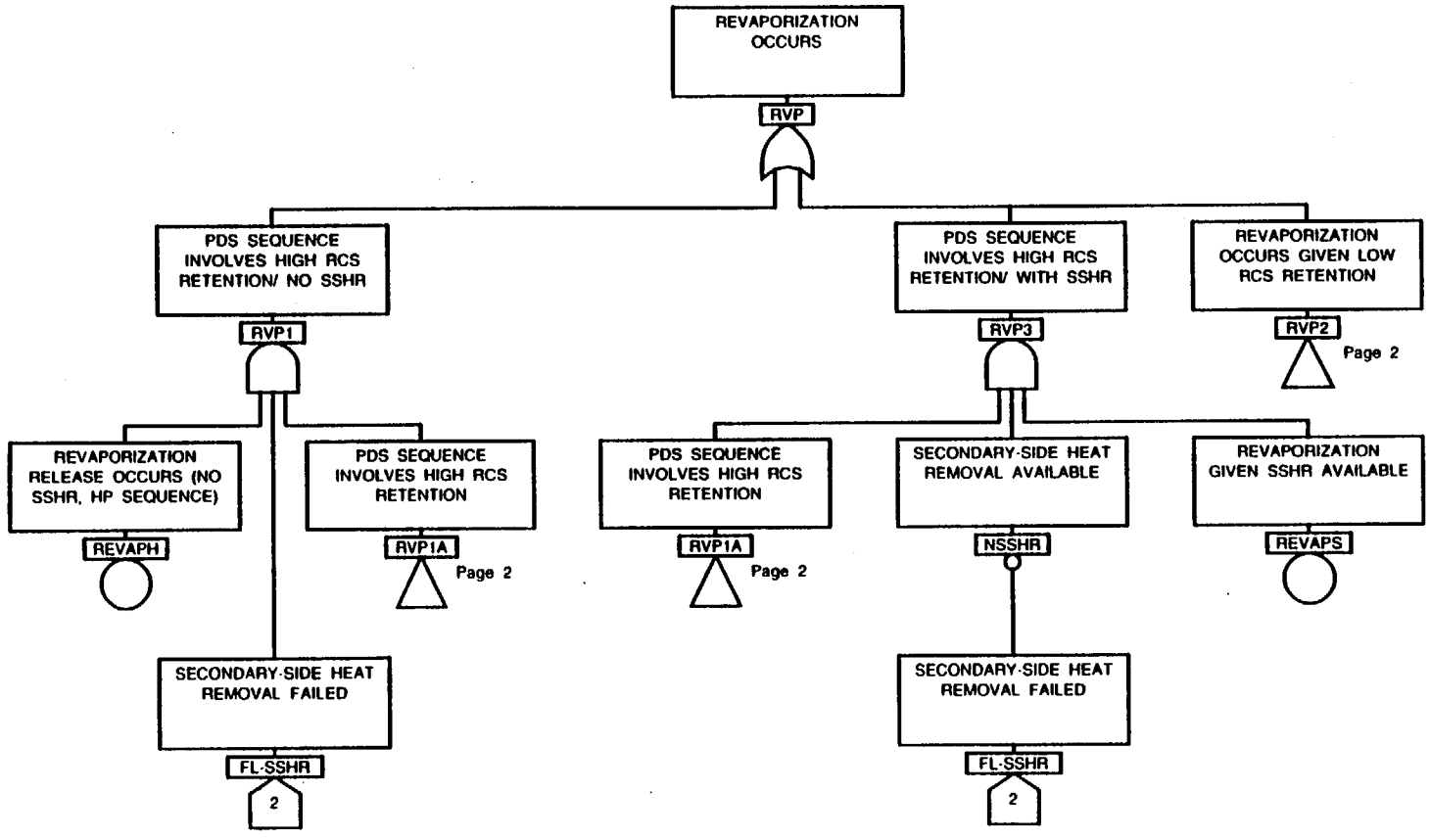
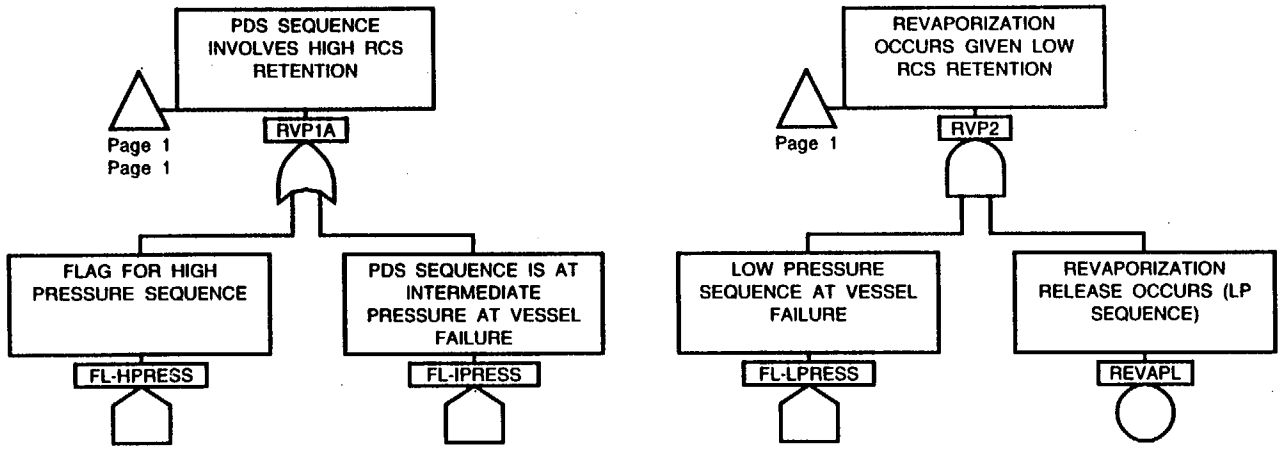


Figure 4–36
Revaporization Occurs (page 2 of 2)



SRCS – Radionuclide Scrubbing (Containment Spray)

The presence of containment sprays provides a means to flood the cavity and lower compartment and provides for debris coverage. In addition, the water spray will scrub the containment atmosphere and remove a significant quantity of radionuclides which will reduce the potential source term. Based on the PDS input, this event is either possible or not. The PDS provides the information necessary to determine whether or not containment sprays are available. Since the only component of the containment spray system which would be exposed to the containment conditions is the spray nozzles the potential for containment spray failure due to environmental effects is considered remote and is not addressed. The containment spray pumps and associated equipment are outside containment and are not exposed to adverse containment conditions..

SRSG – Radionuclide Scrubbing (Steam Generators)

In the case of SGTR events, releases through the containment are not as important as those which may occur through the ruptured tube of the steam generator. This provides a direct path to the environment and can result in significant releases if not scrubbed. The ability to scrub releases during a SGTR accident sequence is based on whether water exists in the ruptured steam generator. If water is maintained in the steam generator or added at some point, then the release through the ruptured steam generator will be scrubbed. Two factors are important for scrubbing. First, is whether water has been retained in the steam generator such that the rupture is covered. If the ruptured steam generator is isolated, there is a reasonable chance that the break will be covered from some period after the event. If the steam generator is not isolated or an SRV is stuck open, then the chances for a submerged release is small.

The operators are instructed to isolate the faulted steam generator, but to maintain feedwater flow to the ruptured steam generator sufficient to keep the tubes covered. If the ruptured steam generator is also faulted (i.e., a relief valve open), however, procedures direct the operators to isolate feedwater to the ruptured steam generator. If no other steam generators are available, then the procedures do direct re-initiation of feedwater to a ruptured steam generator. If feedwater flow is available to an intact steam generator, it is unlikely that the operators will restore cooling to the faulted steam generator prior to release. The option to do this in order to reduce releases is not proceduralized. Many factors would need to be weighed and the possibility for success is questionable. This option is not included in the analysis at this time but may be considered later as accident management alternatives are addressed.

4.5.3 Comparison of CET to NUREG–1335 Guidance

NUREG–1335 provides guidance in performing the containment analysis. This guidance includes a listing of issues which should be addressed in the CET during the IPE process and provides a reasonable listing of the major containment loads which have been postulated for examination during the level 2 portion of the PRA. As a check of the completeness of the SHNPP CET model, the listing provided in NUREG–1335 was compared to the SHNPP model. The comparison indicates that the SHNPP model addresses the issues presented in NUREG–1335 and is an adequate model for studying containment loads during severe accidents. The results of the comparison are summarized in Table 4–21.

Table 4–21
Comparison of Containment Failure Modes Addressed to Those Identified in NUREG–1335 (taken from Table A.4)

Potential Failure Mode	NUREG–1335	SHNPP CET	How Addressed in SHNPP CET
Interfacing Systems LOCA	Yes	Yes	PDS identifies ISLOCA sequences and addresses them explicitly.
Failure to isolate containment	Yes	Yes	Plant-specific isolation fault tree model is used to quantify frequency of sequences without isolation. Isolation failures are identified in PDS.
Steam generator tube rupture	Yes	Yes	PDS identifies SGTR sequences and differentiates those involving stuck open steam generator SRVs.
Overpressurization due to noncondensible gases and steam (early)	Yes	Yes	Early pressurization due to steam is addressed in CET.
Overpressurization due to combustion processes (early)	Yes	Yes	Addressed explicitly for cases with and without in-vessel recovery. Discussion includes both detonation and deflagration potential.
Overpressurization due to direct containment heating	Yes	Yes	Addressed explicitly for cases with and without hydrogen burns.
Missiles or pressure loads due to steam explosions	Yes	Yes	Both in-vessel and ex-vessel steam explosions are addressed in the CET.

Table 4–21 (cont.)
Comparison of Containment Failure Modes Addressed in the SHNPP CET to those Identified in NUREG–1335 (taken from Table A.4)

Potential Failure Mode	NUREG–1335	SHNPP CET	How Addressed in SHNPP CET
Meltthrough due to direct contact between core debris and containment	No	Yes	Plant-specific configuration is such that potential exists for core material to impact liner for high pressure sequences.
Vessel thrust force due to blowdown at high pressure	Yes	Yes	Potential for significant thrust force involving stress failure of lower head is not probable due to lower head penetrations.
Overpressurization with high temperatures due to noncondensible gases and steam (late)	Yes	Yes	Explicitly considered in the CET and includes interactions associated with late steam condensation and the potential for hydrogen burns.
Overpressurization with high temperatures due to combustion processes	Yes	Yes	Late hydrogen burns due to ex-vessel hydrogen generation are addressed along with the potential for burns following late recovery.
Meltthrough due to basemat penetration by core debris	Yes	Yes	Basemat attack following vessel failure is included along with evaluation of debris cooling.
Vessel structural support failure due to core debris erosion	No	No	Design is such that this failure mode is not plausible.

4.6 Severe Accident Progression Analysis and CET Quantification

This section describes the development of the branch point probabilities for the CET and the basis for the basic events which are provided in the CET top logic. In addition, the quantification of the CET end states is provided along with the overall results.

4.6.1 Methods for CET Quantification

This section outlines the process of developing probability estimates for each of the containment event tree branch points. Ideally, for each given plant damage state there is a specific path through the containment event tree which would be followed. The use of probabilities, as opposed to binary states 0 and 1, for each branch point represents our state of knowledge concerning the phenomena which influence the paths chosen. As such, it represents an uncertainty in the accident progression. The use of probabilities also provides a tool for understanding the range of potential outcomes given varying assumptions on containment phenomena and how those assumptions affect overall results of the analysis.

There are two types of event probabilities to be developed. The first comprises those probabilities which are associated with statistical random failures of systems and equipment and are based on available PRA data. An example of this type of event is the probability of containment spray operation. This involves the successful operation of mechanical components and electrical systems and the probability is based on historical performance of similar systems and equipment. From this historical information, a range of possible values can be identified which can be used to define a statistical uncertainty and mean value.

The other type of probability in the CET is a quantitative measure of understanding of how certain phenomena may occur. This usage is a means of representing engineering judgment in the analysis. The probabilities are based on a limited number of deterministic analyses using the MAAP code and available test data from industry and laboratory experiments. This data is incomplete and not well understood for some containment phenomena which tends to increase the uncertainty.

To represent these events in probabilistic terms, a standard set of probabilities has been chosen which represents a likelihood scale. The probabilities are correlated to a qualitative likelihood of the event occurrence based on available information, SHNPP specific design, and engineering judgment. The probability scale is listed in Table 4–22. In some cases, a more refined probability is chosen if it is warranted by either the available design detail or the importance of the event.

Table 4–22
Probability Ranges for Non-Statistical Events

Qualitative Description	Quantitative Value
Certain to Occur	1.0
Highly Likely to Occur	0.999
Very Likely	0.99
Likely to Occur	0.9
Insufficient Information to Draw a Conclusion or Indeterminate	0.5
Unlikely to Occur	0.1
Very Unlikely	0.01
Highly Unlikely to Occur	0.001
Cannot Occur or is Excluded by Previous Events	0.0

In evaluating the potential for events to occur, several sources were used as input. A major source of information was the plant-specific MAAP assessments performed for SHNPP.

The MAAP runs were chosen to represent the dominant sequences for each core damage bin and include consideration of the most important containment safeguards states. For example, the dominant sequence for core damage bin 2 is a loss of feedwater with failure of AFW and the failure of feed-and-bleed cooling. This sequence is chosen as the representative sequence for that bin. In addition, the containment safeguards event tree was solved to address the predominant status for the containment safeguards. For the representative sequence, the most likely case involves the successful operation of all safeguards equipment and containment isolation (state A) and is used in the MAAP analysis to examine the plant response for PDS 2A. MAAP cases are also used to investigate other important issues, e.g., hot leg failure and isolation failures. In this way, important accident characteristics are identified and considered. The number and variety of cases is determined based on the number of probabilistically significant CSET end states, the probabilistic importance of the core damage sequence, and the expected impact on release magnitude. At least one MAAP run is provided for each core damage bin and dominant plant damage state. The representative sequences are identified in Section 4.4.

In addition to this plant-specific assessment for the PDS cases, a deterministic evaluation of the SHNPP containment bounds is documented in Section 4.2. This evaluation looks at the potential ranges and uncertainties for various important phenomena and includes hydrogen source term and burning, hydrogen detonation, in-vessel recovery, induced RCS failure, high pressure melt ejection and direct containment heating, potential for

debris and liner interaction, and ex-vessel debris cooling. The insights provided in this summary are used to formulate many of the upper and lower bounds assumed for the SHNPP analysis.

NUREG–1150 and its supporting documentation are also used in developing the estimates for event probability. Two specific areas involved early hydrogen burn source term and pressurization due to high pressure melt ejection and associated hydrogen burning. For these issues, the expert option in the supporting documenetation serves as a prior distribution which is updated based on the ranges identified by the plant-specific assessment using MAAP. The result is a new distribution which reflects the finding of both NUREG–1150 and the plant-specific assessment.

Finally, other probabilistic risk assessments and other related documentation were reviewed to gain insights into the findings of other studies and to ensure that the quantification and models are reasonable given the present state of knowledge.

4.6.2 CET Basic Event Quantification

This section provides a summary of the selection process for determining the probabilities assigned to events in the CET logic structure and presents the values chosen. For many of cases, the PDS will impact the probabilities assigned and the events are quantified in terms of the particular PDS. In other cases, the event is applicable for all cases and a single value is determined which is appropriate for all PDS cases. Values assigned for each event are provided along with the method used to quantify the events. Some of the events are not based on either statistical information or the probability ranges for non-statistical events. These events represent flag events which are set for each PDS. The flag events used in the CET serve as switches to activate the particular logic when the event condition is met. The flag events are set either true or false. In this way, the event top logic is controlled.

The status of the flag events are based on information provided by the PDS. Although not specifically addressed in the CET, the PDS event is important for two reasons. It provides the initiating event frequency for the CET and also transmits physical conditions and plant status from the front end to the back end analyst. It is important that the PDS provides sufficient resolution to allow differences in the PDS cases to be examined. Based on the solution of the CSET for each core damage bin (Section 4.4), a total of 16 plant damage states exist which exceed the truncation value of 1×10^{-7}/year.

Although each PDS represents many different combinations of failures, the basic accident progression is similar and one or two sequences can be chosen to represent the other sequences. The PDS is used to transfer information from the front end analysis to the back end analysis. Each PDS is assigned a two character descriptor for convenience. For example, the first entry is 2A, corresponding to core damage bin 2 and CSET state A.

From this information, a complete record can be developed for use in evaluating the CET. Table 4–23 presents an example of the information transmitted by the PDS.

Table 4–23
Example of PDS Information Using PDS 2A

Parameter	Status	Indicated By:
RCS pressure	High[1]	Core damage bin 2
Secondary-side cooling	Failed	Core damage bin 2
RCS leakage rate	Cycling relief valve	Core damage bin 2
Reactor cavity status at time of vessel failure	Low water level	Core damage bin 2
Reactor cavity status after vessel failure	High water level	Core damage bin 2
Containment isolation	Isolated	CSET state A
Containment sprays	Functioning during event	CSET state A
Containment fan coolers	Functioning during event	CSET state A
Safety injection after vessel failure	Unavailable	Core damage bin 2, CSET state A, and review of representative sequence

1 The RCS pressure may be low if thermally-induced piping failure occurs. This possibility, however, is determined in the CET.

Flag Events

As described above the flag events are used to activate and truncate branches of the CET which may or may not be appropriate for a particular PDS. For example, the flag event FL–HPRESS is used to distinguish PDS cases where the RCS is at a high pressure at RPV failure. The flag events for the SHNPP CET and a brief description are provided in Table 4–24 and are set either to true or false, depending upon the PDS.

CET Quantified Events

The probabilities for the basic events which comprise the CET logic are derived using several different methods. Table 4–25 presents the basic events, description, assigned value, and the basis for the assigned value. Three categories are used to describe the basis for determining the value chosen. These categories are described below.

Table 4–24
CET Flag Events

Flag Event	Description When True
FL–HPRESS	RCS is at high pressure when RPV fails
FL–SRVO	Pressurizer SRV stuck open
FL–RWST	RWST is injected into the containment
FL–CFC	Containment fan coolers functioning
FL–CSS	Containment sprays functioning
FL–LPI	Low pressure injection failed
FL–SI	High pressure injection failed
FL–PORV	Pressurizer PORV failed or closed
FL–ISLOCA	Interfacing systems LOCA present
FL–SGTR	SGTR present
FL–SSHR	Secondary-side heat removal failed
FL–LISOL	Large containment isolation failure present
FL–SISOL	Small containment isolation failure present
FL–SGSRV	Steam generator SRV stuck open
FL–ACDCP	Ac power available
FL–IPRESS	RCS is at intermediate pressure at RPV failure
FL–LPRESS	RCS is at low pressure at RPV failure

Table 4–25
CET Basic Events

Event Name	Value	Description	Calculational Basis
HOTLFAILS	0.9	RCS hot leg fails at RPV weld due to thermal stress (high pressure)	QUAL
PRZFAILS	0.1	Pressurizer surge line fails due to thermal loads (no PORV open)	QUAL
IVSTEXP	0.5	In-vessel steam explosion occurs given reflood	QUAL
PSTMVF	0.01	Probability of RPV failure given in-vessel steam explosion	QUAL
INGEOMOP	0.1	Failure of in-vessel cooling due to debris geometry (operator action or depressurize)	QUAL
INGEOMME	0.5	Failure of debris cooling in-vessel due to geometry (depressurize case only)	QUAL
OPER–IV	0.1	Operator fails to manually depressurize RCS which allows LPI flooding (undeveloped event)	QUAL
RECPACIV	0.13	Ac power to safety equipment restored prior to RPV failure (demand ac failures)	LOSP
PRZFAILN	0.5	Pressurizer surge line fails due to thermal loads (RCS PORV open)	QUAL
HOTLFAILSX	0.1	RCS hot leg fails at RPV weld due to thermal stress (intermediate pressure)	QUAL
RNDSOUR	0.5	Random ignition source present in containment	QUAL
CFAIL1	0.003	Containment failure given hydrogen burn after in-vessel recovery and low steam concentration	CALC
BURNGO1	0.1	Hydrogen burn occurs given low steam concentration and in-vessel recovery	QUAL
CFAIL2	0.003	Containment failure given hydrogen burn after in-vessel recovery and high steam concentration	CALC
BURNGO2	0.001	Hydrogen burn occurs given high steam concentration and in-vessel recovery	QUAL
SGTUBEF2	0.01	Steam generator tube thermal failure (RCP off)	QUAL
SGTUBEF1	0.5	Steam generator tube thermal failure (RCP on)	QUAL
SGTUBEFD	0.1	Steam generator tube thermal failure (RCP off, steam generator depressurized)	QUAL
SGSRVO	0.1	Ruputred steam generator SRV fails to close after opening	QUAL
POORMIX	0.001	Containment poor mixing resulting in pockets of high hydrogen concentration	QUAL
CFAIL–DET	0.1	Local containment failure given local detonation	QUAL
CFAIL–DETG	1	Containment failure given large global detonation/DDT	QUAL
LGH2SOURCE	0.001	Large hydrogen source term present in containment without prior burn	QUAL

Table 4–25 (cont.)
CET Basic Events

Event Name	Value(s)	Description	Calculational Basis
BURNHCH	0.001	Hydrogen burn occurs given high steam and high hydrogen source term	QUAL
CFAILHCH	0.077	Containment failure occurs due to hydrogen burn (HCH case)	CALC
BURNHCL	0.001	Hydrogen burn occurs given high steam and low hydrogen source term	QUAL
CFAILHCL	0.001	Containment failure occurs due to hydrogen burn (HCL case)	CALC
BURNLCH	0.9	Hydrogen burn occurs given low steam and high hydrogen source term	QUAL
CFAILLCH	0.015	Containment failure occurs due to hydrogen burn (LCH case)	CALC
BURNLCL	0.5	Hydrogen burn occurs given low steam and low hydrogen source term	QUAL
CFAILLCL	0.001	Containment failure occurs due to hydrogen burn (LCL case)	CALC
PRSTMEX1	0.25	Ex-vessel steam explosion occurs given RPV failure	QUAL
CFAIL–SEHP	0.001	Containment failure due to ex-vessel steam explosion	QUAL
H2@DCH	0.1	Probability of large hydrogen burn at RPV failure given high steam concentration	QUAL
DEBFRAG	0.99	Debris fragmentation occurs (HP case)	QUAL
H2@DCL	0.9	Probability of large hydrogen burn at RPV failure given low steam concentration	QUAL
DEBFRAGI	0.5	Debris fragmentation occurs (IP case)	QUAL
CFAIL–HDH	0.01	Probability of containment failure due to DCH given high RCS pressure and high steam	CALC
CFAIL–HDL	0.001	Probability of containment failure due to DCH given high RCS pressure and low steam	CALC
CFAIL–IDH	0.001	Probability of containment failure due to DCH given intermediate RCS pressure and high steam	CALC
CFAIL–IDL	0.001	Probability of containment failure due to DCH given intermediate RCS pressure and low steam	CALC
REPRESS	0.1	Significant repressurization occurs in-vessel when debris enters lower head	QUAL
S–SEAL	0.5	Liner ablation mitigated by self-sealing	QUAL
LINERFHL	0.1	Liner failure occurs due to debris contact (high debris, low water case)	QUAL
LINERFHH	0.01	Liner failure occurs due to debris contact (high debris, high water case)	QUAL
LINERFLL	0.1	Liner failure occurs due to debris contact (low debris, low water case)	QUAL
LINERFLH	0.001	Liner failure occurs due to debris contact (low debris, high water case)	QUAL

Table 4–25 (cont.)
CET Basic Events

Event Name	Value(s)	Description	Calculational Basis
DEBQNCH	0.5	Conditional probability of quenching given debris fragmentation (liner attack case)	QUAL
DBC–C4	0.1	Debris not coolable given no dispersal and no late pools	QUAL
DBC–C3	0.5	Debris not coolable given no dispersal and late pools	QUAL
DBC–C2	0.01	Debris not coolable given dispersal and no late pools	QUAL
DBC–C1	0.5	Debris not coolable given dispersed and late pools	QUAL
DEBTHICK	0.1	Limited debris spreading leads to uncoolable debris bed	QUAL
LH2BURNL	0.99	Late hydrogen burn given that containment heat removal available (expect early burn)	QUAL
H2BURNR	0.99	Probability of a hydrogen burn following containment cooling restoration	QUAL
CFAIL–LLR	0.001	Probability of containment failure given late H2 burn with low base pressure	CALC
LH2BURNS	0.001	Late hydrogen burn occurs given no containment heat removal or recovery	QUAL
CFAIL–LHR	0.001	Containment failure occurs given late hydrogen burn and no containment hear removal	CALC
CFAIL–R	0.1	Containment failure given late containment heat removal recovered and hydrogen burn	CALC
RECPROB	0.8	Containment cooling not recovered prior to containment failure (ac available)	QUAL
RECPAC	0.2	Non-recovery probability for containment heat removal (no ac power)	LOSP
OP–H2REC	0.001	Operators preclude hydrogen burn following recovery of containment heat removal	QUAL
LATEDET	0.001	Late restoration of AC power results in hydrogen detonation	QUAL
HTSUFF	0.5	Debris decay heat sufficient to maintain concrete attack until containment failure occurs	QUAL
LATEDBC	0.5	Late debris cooling fails after recovery of containment cooling	QUAL
REVAPL	0.01	Revaporization release occurs (low pressure sequences)	QUAL
REVAPH	0.1	Revaporization release occurs (high pressure sequences)	QUAL
REVAPS	0.001	Revaporization release occurs given secondary-side heat removal available	QUAL

QUAL – As discussed above, many of the events present in the CET are not based on statistics but are expressions of uncertainty about the potential for an event to occur. Events of this type are included in the "QUAL" category. The event probability represents a qualitative uncertainty about the issue and is based on a review of the literature, plant-specific MAAP assessments, and hand calculations. For example, MAAP calculations for SHNPP indicate that the containment will be above the limits stated in the literature required for steam inerting by the time core uncovery occurs if containment heat removal is not present. Given this fact, it is very unlikely that a hydrogen burn would occur and a value of 0.01 is assessed.

CALC – Events in this category are based on an evaluation of calculated pressures, temperatures, and the containment failure curve Section 4.3. Given the conditions described by other events, a containment load is postulated, such as a hydrogen burn, and the resultant pressure load is developed. A distribution of possible pressures is developed using information contained in NUREG–1150 updated with plant-specific analysis. The resultant pressure distribution is convoluted with the containment failure curve to arrive at the probability of containment failure.

LOSP – The events assigned to this category represent ac power restoration events and are based on the loss of offsite power restoration curve. The core damage model addressed AC power recovery up to the point of core uncovery. In the CET two different time frames are addressed for situations were the restoration of AC power could impact overall containment performance. The first time frame deals with in-vessel recovery. The probability of restoring AC power between the time of core damage and RPV failure and the AC power recovery curve are used to determine the incremental AC power recovery probability. For long term station blackout sequences with containment heatup and core concrete attack a best estimate time to recovery is determined using available time from RPV failure until containment overpressure and the AC power recovery curve..

4.6.3 CET Quantification Results

This section provides an overview of the results of the CET quantification and release category assignment. In addition, the important PDS sequences are discussed in terms of containment failure categories and release categories.

<u>Containment Failure Category Results</u>

The CET was solved for each of the PDS cases identified in Section 4.4. The various basic events in the CET top logic models were adjusted, using the flag events, as necessary to represent the specifics of the PDS. Based on this solution, the CET end state frequencies were determined. The dominant PDS results are presented Table 4–26 and account for almost 90% of the total frequency. The end states are provided in terms of one of eight categories, described in Table 4–27.

Table 4–26
Results of CET Quantification for Major Contributors

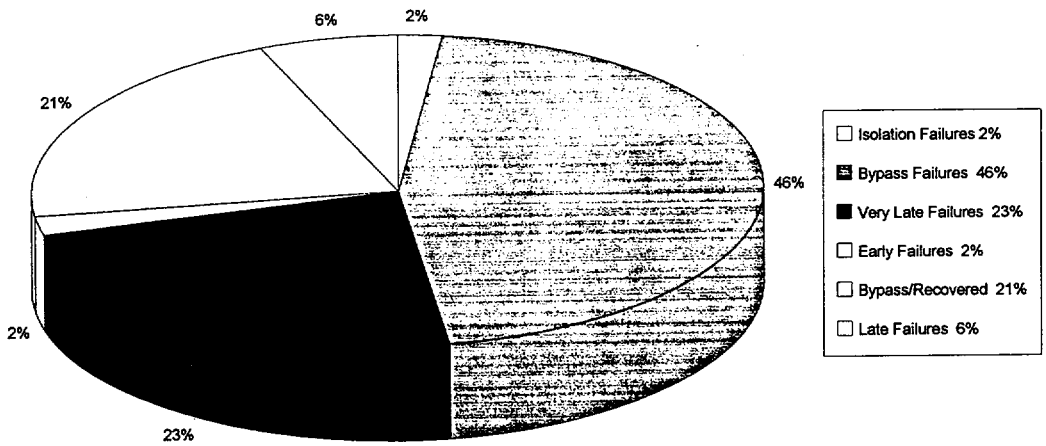
Plant Damage State	1P	15A	13A	2A	7A	12A
PDS Frequency	2.30×10^{-5}	1.17×10^{-5}	1.16×10^{-5}	7.05×10^{-6}	6.56×10^{-6}	2.31×10^{-6}
Containment Intact	4.3×10^{-6}	1.13×10^{-5}	1.12×10^{-5}	1.9×10^{-6}	6.35×10^{-6}	2.2×10^{-6}
In-vessel Recovery and Containment Intact	1.65×10^{-5}	–	–	1.58×10^{-6}	–	–
In-vessel Recovery and Containment Failed	2.8×10^{-8}	–	–	1.6×10^{-6}	–	–
Containment Isolation/Bypass Failure	–	–	–	1.9×10^{-6}	–	–
Early Containment Failure	–	1.6×10^{-7}	1.5×10^{-8}	–	–	–
Late Containment Failure	6.4×10^{-7}	1.2×10^{-8}	1.16×10^{-8}	–	–	–
Very Late Containment Failure	1.4×10^{-6}	2.4×10^{-7}	3.5×10^{-7}	5.8×10^{-8}	2.0×10^{-7}	6.9×10^{-8}

Table 4–27
Descriptions of CET Binning Categories

Category	Description
Very Late Failure	Very late containment failure occurring due to prolonged core concrete attack which results in overpressure or basemat failure.
Late Failure	Late failures are defined as containment failures which occur at some time after RPV failure and involve failures related to ex-vessel debris cooling, recovery, and late overpressurization.
Early Failure	This category includes early containment failures which occur either before or shortly after RPV failure, e.g., DCH, early hydrogen burns, liner failure.
Small Isolation Failure	This category addresses sequences which have a small containment isolation failure. These sequences are precluded from having successful isolation since the leakage path provides a means for radionuclides to exit the containment. The path is small enough, however, to allow for some containment retention of radionuclides. The isolation failure is assumed to occur prior to core damage.
Large Isolation Failure	This category is similar to the small failure except that no containment retention is credited.
Small Containment Bypass	Some initiating events provide for containment isolation failure by providing a bypass path, e.g., SGTR. This category identifies the contribution from these sequences.
Large Containment Bypass	This category is similar to the small bypass except that little retention is provided in the RCS and the potential for substantial releases is present.
Intact Containment	Containment isolation has been maintained and leakage is limited to design basis.
In-vessel Recovery with Intact Containment	Core damage is arrested in-vessel and containment integrity is maintained.
In-vessel Recovery with Failure of Containment	This category includes the likelihood that, following in-vessel recovery, containment failure occurs.

As the results indicate, there is a high probability of maintaining containment integrity. Based on the results, the conditional probability of containment failure is estimated to be about 0.15. The distribution of the containment failure modes is graphically illustrated in Figure 4–37. From this figure, it is observed that containment bypass failures represent the most likely failure means (~55%) with the next contributor being that of very late containment failure (~23%). Bypass failures are dominated by the potential for an induced SGTR following heatup of the steam generator tubes. Very late failures are dominated by failure to recovery containment heat removal before containment overpressure.

Figure 4–37
CET Category Results



Early containment failures represent only about 2% of the potential failure outcomes. The potential for early containment failures has been identified in prior studies to be an important characteristic for measuring containment performance. At SHNPP the conditional probability of early containment failure is low due to factors described in Section 4.8, and represents less than 0.1% of all potential outcomes.

PDS case 15A is the highest contributor to early containment failure (~92%) and involves a hydrogen burn which overpressurizes containment as the dominant mode of containment failure. No other sequence contributes significantly to early overpressure failure. The specific results are addressed in additional detail below and are grouped to simplify the discussion and to be able to easily understand containment response to severe accident loadings. The categories are chosen to address important issues which have been identified in prior PRAs and by the guidance provided by the NRC in regards to the performance of the IPE. Three of the impaired containment categories contribute over 90% of the total frequency of impaired containments. The accident progression for the most important accident sequence for each of the impaired containment categories is discussed. Other contributors are identified if important. Sequences which result in an intact containment state with either recovery of core cooling invessel or a coolable debris bed represent a total of 85% of the end state outcomes and are not addressed in this section.

Containment Bypass Failure/In-Vessel Recovery and Containment Bypass (5.0×10^{-6}/year)

Due to the similar nature of these categories they can be discussed together. These categories represent an accident sequence with either failure of the RPV or recovery in-vessel and the bypass of the containment. PDS 16B and PDS 2A both contribute about equally to these categories and are very similar in nature. In both cases, the bypass is caused by a SGTR. For PDS 16B the representative sequence is a SGTR with failure of makeup and a failed open steam generator SRV (sequence RUW). The SGTR results in a containment bypass. PDS 2A represents a loss of all feedwater and failure of feed-and-bleed cooling, and contributes as a result of an induced SGTR following RCS heatup. The operation of the RCPs for this PDS results in decay heat being transferred to the steam generator tubes which then fail due to thermal stresses.

Very Late Containment Failure (2.5×10^{-6}/year)

This category is the second most frequent containment failure mode and represents containment failure after a very long period of time (about 90 hours). Seven PDSs contribute significantly to this category with each having the same basic containment failure mechanism. The failure of containment is due to prolonged core concrete interactions which result in either very long term containment overpressure or basemat failure. The driving factor for these sequences is the uncertainty associated with debris cooling and the potential for late pools of debris to form under the RPV as the debris leaves the RPV after vessel failure. The inability to cool the debris results in prolonged steam generation at a very slow rate (about 1.5 psi/hr) as the debris ablates the concrete. The basemat is about 12 feet thick and concrete ablation results in very long term basemat failure which would occur near the same time (90 hours).

Given the extremely long time involved, considerable actions could be attempted to establish debris cooling and to mitigate possible releases. Due to uncertainties in the effectiveness of these very long term recoveries, only limited credit was taken for cases which involved equipment restoration. Due to the long times involved many PRAs have considered these types of sequences as successful containment integrity. If this assumption is applied to SHNPP, the conditional probability of successful containment performance is increased from 85% to about 90%.

Late Containment Failure (6.7×10^{-7}/year)

The major PDS for this category is PDS 1P which represents about 92% of the total frequency. PDS 1P represents a station blackout sequence without containment heat removal, secondary-side heat removal, or safety injection available. The RCS pressure at

RPV failure may be high, and the containment is isolated. The loss of RCP seal injection and thermal barrier cooling may result in an RCP seal LOCA which, without RCS injection, which uncovers the core and results in core damage.

For this sequence, the potential for recovering ac power is assessed and two cases developed. If ac power is not recovered, the containment will gradually pressurize until failure occurs (about 90 hours into the event). If ac power is restored, the steam concentration will be greatly reduced and a hydrogen burn is possible which could fail containment. This failure model is the dominant contributor to this category.

Containment Isolation Failure (2.0×10^{-7}/year)

PDS 15B contributes entirely to this category. PDS 15B represents an S2 LOCA with the failure of all injection. An isolation failure occurs which provides a path for radionuclide release. The accident progression is similar to most LOCA events with the loss of RCS inventory resulting in a loss of decay heat removal and core damage. The accident progresses to RPV failure and the debris relocates to the cavity. The availability of containment sprays allows for a continuous water pool over the debris and radionuclide scrubbing. The debris is coolable which reduces concern over basemat attack. Although specifically addressed by the CSET, large isolation failures do not contribute significantly to the probability of containment failure.

Early Containment Failure (1.8×10^{-7}/year)

PDS 15A provides about 92% of the early failure contributions, with the dominant cause of containment failure being due to hydrogen burning.

PDS 15A is represented by an S2 LOCA with a failure of injection. Given the successful operation of the containment sprays and fan coolers, the containment pressure and the steam concentration remain low throughout the accident progression. This provides the opportunity for hydrogen burning. Since AC power is available, sufficient ignition sources are available in containment to ensure ignition of hydrogen at low concentrations. The operation of the containment sprays, in combination with the communication pathways between the various regions of the containment result in a negligible potential for high hydrogen concentrations and detonation.

A hydrogen burn is assumed to occur at or near vessel failure which is sufficient to fail containment at some modest probability. The injection of the RWST along with the operation of containment sprays and/or fan coolers results in the debris being continuously covered after vessel failure. Given the available floor area and the quantity of debris expected, the depth of the debris, ex-vessel cooling is essentially assured. The overlying pool also provides a scrubbing mechanism for radionuclide scrubbing.

4.7 Radionuclide Release Characterization

4.7.1 Release Category Characterization

The end points of the CET correspond to the outcome of possible severe accident sequences. Each of these end points defines a different containment end state with an associated release. Simplifications can be attained by grouping sequences with similar release characteristics into a release category that can be applied to the containment end states. A set of release categories is defined such that all accidents assigned to the same category are assumed to have the same set of release fractions.

The main characteristics of the containment end states considered when developing these release categories were:

- release energy
- containment isolation failure size
- Timing
- isotopic composition

These characteristics, and how they were used to group the events into release categories, are discussed below.

Energy of Release

The energy of the release is important in determining dispersal of radionuclides in the environment. If the containment fails while at high pressure, the radionuclides will be injected into the environment quickly, and the release will be similar to a puff of radionuclides. A low energy release, such as a small containment isolation failure that prevents significant containment pressure buildup, will be a slower release over a longer period of time. Those sequences that occur suddenly, such as containment rupture, will be high-energy releases. In addition, large isolation failures will be included in the high energy release category.

The difference in the energy of the release is important for analyses in which evacuation and meteorological conditions are addressed. This would be considered in a level 3 PRA, which is not required at this time for SHNPP. Thus the effect of the energy of the release can be neglected for this study. This leads to a significant reduction in the number of release categories. That is if the energy of the release is disregarded, the same source term may be applied to:

- Sequences with late failure due to gradual overpressurization may be grouped with sequences with late failure caused by a hydrogen burn.

- Sequences with early over pressurization failures (e.g., due to early large H_2 burns or DCH) may be grouped sequences with early large isolation failures.

Containment Isolation Failure Size and Location

The isolation failure size and location will determine the rates at which radionuclides leave the containment, i.e., the holdup time. If the radionuclides can be held in the containment they will have time to decay and the source term will be reduced. If there is an isolation failure, the holdup time will be reduced. For large isolation failures, the radionuclides will not be held in the containment and a larger release will result. Three distinct alternatives are chosen for this parameter: large, small, and intact. The last alternative is chosen to represent those core damage sequences in which core damage has occurred but containment integrity has been maintained. In addition, the location of the break may be either the auxiliary building or the environment. However, no credit has been taken for fission product reduction in the auxiliary building in this study. Releases to the environment would be above ground for the most part, but may also be underground; i.e., in the case of basemat melt through.

Timing of the Release

The timing of the release impacts the source term and the warning time for most PRAs. Since this analysis is not required to address evacuation, warning time is not an issue. Past PRAs have, however, made a distinction between early and late failures of the containment, and this convention is retained. For this analysis, "early" is defined as any time up to and including the time shortly after RPV failure. "Late" is defined as any time following RPV failure until the end of the analysis (usually 48 hours).

An additional distinction "very late" was used to address cases in which the containment pressure rise is very slow (due mainly to CCI), and containment failure (i.e., due to over pressure or basemat melt through) is not expected for several days.

Isotopic Concentration

This characteristic addresses the makeup of the source term and is based on the events that have occurred during the accident. The radionuclide concentration varies according to what systems are functioning and the time of the release. The different concentrations correspond to releases in which CCI occurs, releases in which all releases are scrubbed prior to release, releases in which revaporization of fission products that had previously been retained in the RCS occurs following containment failure, and releases associated with an intact containment. In general, the releases that have CCI have higher non-volatile fission products (i.e., mainly strontium, tellurium, and barium, produced ex-vessel) release fractions, and revaporization releases increase the volatile fission products

(i.e., mainly cesium and iodide produced in-vessel) release. The intact containment releases are small due to the low containment design leakage rate.

If containment sprays are present, radionuclide aerosols are scrubbed from the containment atmosphere and the major release constituents are noble gases.

A set of release category definitions was developed to be applied to each containment end state. The following paragraphs define each release category and related assumptions.

Containment Intact (IC–1)

This release category represents an accident sequence in which the containment is intact. The source term for this type of sequence is very small and limited to the containment design leakage rate.

Release Category 1 (RC–1)

This release category is a late containment failure caused by gradual over pressurization. The core debris is assumed to be coolable. This type of gradual pressure increase is assumed to result in a benign containment failure and the duration of the release could be over a long period of time. The release from the containment is scrubbed by either the containment sprays or a pool of water over the core debris.

Release Category 1A (RC–1A)

This release category is similar to RC–1 except that revaporization occurs. Revaporization is caused by the self-heating of radionuclides plated out on the reactor coolant system. This revaporization is postulated to occur late in the accident sequence after the containment has failed. This allows the radionuclides to be released from the containment after only a limited holdup time. The impact of revaporization on the source term is to increase the contribution of volatile radionuclides.

Release Category 1B (RC–1B)

This release category is similar to RC–1 except that no scrubbing by containment sprays and/or water pools is available. If containment sprays function, or the RWST inventory is otherwise dumped into containment, then both debris cooling and scrubbing will be attained (unless debris uncoolability is assumed). This can be assumed because for the SHNPP containment when the RWST is discharged the water level reaches several feet over the debris, completely covering the debris bed for the duration of all applicable sequences studied. Thus, this category applies to sequences in which the RWST is not injected and the debris bed eventually dries up resulting in considerable CCI.

Release Category 1BA (RC–1BA)

This release category is similar to RC–1 except that both revaporization and no containment scrubbing are assumed to occur.

Release Category 2 (RC–2)

This release category represents a large early containment failure. The core debris is assumed to be coolable. The large failure significantly reduces radionuclide holdup time in the containment. The SHNPP specific liner failure releases are assumed to belong to this category. The release from the containment is scrubbed by containment spray operation at the time following fission product releases from the primary side. In this case the releases will be driven by the prompt release of fission products at containment failure and the effect of revaporization, if any, should be small. Thus, release categories with revaporization will not be postulated for the large early containment failures. However, care will be taken when assigning source terms to pick a representative sequence for RC–2 (and RC–2B) that exhibits revaporization.

Release Category 2B (RC–2B)

This release category is similar to RC–2 except that no scrubbing by containment sprays and/or water pools is assumed to happen.

Release Category 3 (RC–3)

This release category represents an early containment isolation failure with a small leakage rate (<4" diameter). The core debris is assumed to be coolable. The release from the containment is scrubbed by either the containment sprays or a pool of water over the core debris. For the larger of the small leakage failures (i.e. close to 4" in diameter) the releases will be driven by the prompt release of fission products at containment failure and the effect of revaporization, if any, should be small. Smaller diameter isolation failures will result in reduced source terms due to the longer time available for natural removal mechanisms, such as gravitational settling, to take place. Release categories with revaporization will not be postulated for the small early containment failures. However, care will be taken when assigning source terms to pick a representative sequence for RC–3 (and RC–3B) that exhibits revaporization.

Release Category 3B (RC–3B)

This release category is similar to RC–3 except that no scrubbing by containment sprays and/or water pools is assumed to happen.

Release Category 4 (RC–4)

This release category represents a containment bypass accident sequence with a small leakage rate. The leakage rate that would correspond to a SGTR sequence with cycling SRVs, or an ISLOCA in which operators react in time to mitigate effects by closing the valves on the RHR suction line. The core debris is assumed to be coolable and releases from the containment scrubbed. Scrubbing by water in the ruptured SG above the break is assumed to occur, since the procedures would direct the operators to maintain a minimum level in the ruptured steam generator.

Release Category 4C (RC–4C)

This release category is similar to RC–4 except that no scrubbing by water in the ruptured steam generator above the break occurs. The core debris is assumed to be coolable and releases from the containment scrubbed.

Note that a release category for no scrubbing by containment sprays and/or water pools is not postulated in this case. This is so because for the bypass sequences most of the release would be directly from the primary to the environment or the auxiliary building. The non-volatile contribution to the source term should be negligible since CCI is unlikely in all the dominant SGTR sequences for SHNPP. This is so, because core damage is averted in sequences for which the high pressure injection pumps are available (assuming the RWST is refilled, which is likely considering MAAP predicts at least 20 hours would pass before this tank empties). When HPI is not available the RWST inventory would be discharged to the containment via containment spray operation and cool down the debris, or at least scrub the releases. At any rate, even if CCI occurs, the containment would be at low pressure (depressurizes through the stuck open relief valve for the cases involving loss of RWST inventory outside containment) and thus, there would be little driving force for further releases.

Release Category 5 (RC–5)

This sequence represents a containment bypass accident with a large leakage rate. Such rate is representative of a SGTR accident with a stuck open SRV on the ruptured steam generator, or an unmitigated ISLOCA accident. The core debris is assumed to be coolable and releases from the containment scrubbed. The releases from the ruptured steam generator are assumed to be scrubbed by water above the break line. However the probability of scrubbed releases is very low due to present procedures. Thus, the unscrubbed source term (RC–5C) will be conservatively assigned to these low probability branches.

Release Category 5C (RC–5C)

This release category is similar to RC–5 except that scrubbing by water in the ruptured steam generator does not occur.

Release Category 6 (RC–6)

This category represents cases for which the containment failure mode would be a very late failure due to basemat meltthrough.

Release Category 7 (RC–7)

This category represents cases for which containment fails "very late" due to over pressurization. That is, in this case the pressure rise in containment is mainly caused by the build up of noncondensible gases from CCI.

Note that all the debris categories discussed above apply strictly only to CET branches labeled "debris cooled". That is, the "debris uncooled" branches include the possibility of debris uncoolability, as well as RWST not injected. However, the lack of debris cooling is still addressed in the CET. Two simplifying assumptions were used to apply the above release categories to the "debris uncooled" branches. These assumptions were:

- The release categories for coolable debris with unscrubbed releases (i.e., debris bed coolable but no water cover available) can be used in place of both "non-coolable" scrubbed and not scrubbed cases. Note that in the former substitution (coolable unscrubbed substitutes non-coolable scrubbed) the release is conservatively over predicted.

- The release categories for coolable debris with unscrubbed releases and revaporization (i.e., debris bed coolable but no water cover available and fission product revaporization) can be used in place of both "non-coolable" with revaporization, scrubbed and not scrubbed cases. Note that in the former substitution (coolable unscrubbed with revaporization substitutes non-coolable scrubbed with revaporization) the release is conservatively over predicted.

As used here, the term "coolable" is understood to signify that the debris pool on the containment floor can be cooled if the mean to cool it exists. This cooling is generally provided by an overlying pool of water since convective cooling by containment atmosphere would be insufficient. However, the debris geometry could be such (e.g., due to formation of surface crusts, or debris piled up into an uncoolable geometry) that cooling would not be possible even when overlying water pools are available. On the other hand, if no overlying water pool exists the debris will not be cooled, and corium

concrete attack would continue regardless of whether the debris would be coolable or not coolable.

4.7.2 Source Term Estimates

The estimate of the source term for each release category, used results from deterministic analysis of representative PDS sequences. The analysis used the MAAP code, which calculates source terms for severe accident progressions.

The releases predicted by a particular representative sequence were used to define release fractions for a release category, whenever the characteristics of the sequence closely matched a containment end state. Otherwise, information implied from the complete set of PDS runs, or new sequences designed for a specific release category, were used to complement the results of representative sequence in assigning release fractions.

The first step in this process was to assign each sequence (see Table 4–28) to one of the release categories defined earlier. Several observations can be made with respect to these assignments:

- The principal variables taken into account when assigning the late release categories were the state of the containment sprays, RWST discharge, AFW, and whether revaporization occurred or not. The containment sprays are very efficient in scrubbing the containment atmosphere. However for most late sequences containment spray operation is not expected, since their availability would for most cases, prevent containment failure. The status of the RWST gives an indication of fission product generation by CCI, and whether scrubbing occurs. If the RWST is injected the debris bed would most likely be cooled, or at least the releases due to CCI would be scrubbed. If AFW is on, revaporization of volatile fission products in the RCS would probably not occur, and in most cases there would be sufficient time for natural removal mechanisms to mitigate the volatile fission products released at and before vessel failure from the RCS.

Table 4–28
Release Category Assignments

RUN#	TYPE	TIME RVF (HR)	PRESS RVF, PSIA	CONTM FAIL (HR)	RELS time	HPME	CS	CFC	RWST DISCH	CAV WET	AFW	REVAP (RCS) %	PS RET %	NOB % (4)	CsI %	CsOH %	SrO %	Te2 %	TeO2 %	CNMT STATE	REL CHAR	RELS CATEG
SH-12A-1	MLOCA, SI(R)	4.0	25	N		N	Y	Y	Y	Y	Y	3	42	L	L	L	L	L	L	I		IC-1
SH-12A-2	MLOCA, SI(R), CS(R)	4.0	25	N		N	N	Y	Y	Y	Y	6	40	L	L	L	L	L	L	I		IC-1
SH-13A-1	SLOCA(3"),SI(R)	5.3	34	N		N	Y	Y	Y	Y	Y	5	50	L	L	L	L	L	L	I		IC-1
SH-2G-1	LOFW,HPI(S),CFC(S)	3.0	2270	N		Y	Y	N	Y	Y	N	0	93	L	L	L	L	L	L	I		IC-1
SH-9A-1	MLOCA, LPI(S)	4.5	25	N		N	Y	Y	Y	Y	Y	5	43	L	L	L	L	L	L	I		IC-1
SH-11A-1	LLOCA, SI(R)	2.9	24	N		N	Y	Y	Y	Y	Y	3	44	L	L	L	L	L	L	I		IC-1
SH-13A-2	SLOCA(4"),SI(R)	5.1	29	N		N	Y	Y	Y	Y	Y	3	42	L	L	L	L	L	L	I		IC-1
SH-13A-3	SLOCA(5"),SI(R)	5.1	27	N		N	Y	Y	Y	Y	Y	4	41	L	L	L	L	L	L	I		IC-1
SH-2A-3	LOFW,PFB,PORV(O)	2.5	2290	N		Y	Y	Y	Y	Y	N	0	94	L	L	L	L	L	L	I		IC-1
SH-2A-1	LOFW,PFB(S)	NO CORE DAMAGE																		I		IC-1
SH-2G-1HL	SH2G-1 W/ HLF	CORE RECOVERED (IN VESSEL H2: 650 LB)																		I		IC-1
SH-2A-2	LOFW,PFB,PORV(O)	NO CORE DAMAGE																		I		IC-1
SH-2G-3	SH2G-1HL W/ RCP(OFF)	CORE RECOVERED (IN VESSEL H2: 31 LB)																		I		IC-1
SH-16B-2	SGTR,HPI(S),SEC DEPRESS	NO CORE DAMAGE																		I		IC-1
SH1P-5	SH1P-3 W/ HLF	14.2	54	(SEE SH1P5-RC6)		N	N	N	N	N	N	29	44	(SEE SH1P5-RC6) (BACK-UP)								
SH-16B-1	SGTR, HPI(S),SRV(O)	6.4	552	2 min	E	N	Y	Y	Y	Y	Y(1)	30	37	100	59.05	52.59	0.011	9.40E-3	0.0	B	HM	RC-5/RC-5C
SH-RC-4C2	SGTR,CYCL SRV,UNSCRUBBED	5.1	1310	0	E	Y	Y	Y	Y	Y	N	34	62	100	3.71	3.39	8.35E-5	0.0	0.0	B	ML	RC-4C
SH-RC-4	SGTR,CYCL SRV,SCRUBBED	16.6	1240	0	E	Y	Y	Y	Y	Y	Y	17	80	100	0.223	0.176	4.76E-5	0.0	0.0	B	LL	RC-4
CTFLSZ-7	SBO(250 GPM),ISO(4.5")	7.7	567	0.0	E	N	N	N	N	Y(2)	N	0	85	100	2.68	1.78	0.17	11.78	0.0	IS	MH	RC-2B
SH-RC-3B	CTFLSZ5: reduc deb. area	8.0	593	0.0	E	N	N	N	N	Y(2)	N	14	76	100	1.83	1.32	9.89E-4	0.0059	0.0	IS	MM	RC-3B
CTFLSZ-8	SBO(250 GPM),ISO(5")	7.7	567	0.0	E	N	N	N	N	Y(2)	N	0	90	100	1.76	1.11	0.21	12.38	0.0	IS	MH	RC-2B
CTFLSZ-5	SBO(250 GPM),ISO(3.5")	7.7	567	0.0	E	N	N	N	N	Y(2)	N	0	91	100	0.83	0.55	0.03	2.70	0.0	IS	LM	RC-3B
CTFLSZ-4	SBO(250 GPM),ISO(3")	7.7	567	0.0	E	N	N	N	N	Y(2)	N	0	91	100	0.77	0.48	0.04	4.23	0.0	IS	LM	RC-3B
CTFLSZ-0	SBO(250 GPM),ISO(1")	7.7	567	0.0	E	N	N	N	N	Y(2)	N	0	91	100	0.160	0.100	0.0026	0.31	0.0	IS	LM	RC-3B
SH-13A-4	SLOCA(3"),SI(R),ISOL(2")	5.3	34	0.0	E	N	Y	Y	Y	Y	Y	8	47	100	0.069	0.069	3.00E-5	3.20E-5	0.0	IS	LL	RC-3
SH-13A-5	SH-13A-4 W/ ACCUM(S)	4.2	34	0.0	E	N	Y	Y	Y	Y	Y	9	47	100	0.045	0.046	1.80E-5	1.50E-5	0.0	IS	LL	RC-3
SH-12A-3	LLOCA,SI(O), ISO(5")	1.1	40	0.0	E	N	N	N	N	N	N	2	43	100	4.1	4.56	2.50E-1	12.40	0.0	IS	MH	RC-2B
SH-RC-2B(3	LOFW,HPI(S),CHR(O)	2.5	2320	2.5	E	Y	N	N	N	N	N	20	78	100	2.890	3.260	3.13E-4	0.0	0.0	R	MM	RC-2B
SH-RC-2(3)	LOFW, SI(O), CHR(O)	2.5	2320	2.5	E	Y	Y	Y	Y	Y	N	7	91	100	0.47	0.40	1.65E-4	0.0	0.0	R	LM	RC-2
SH-9A-1AX	MLOCA,LPI(S),CHR(T=0)	11.5	51	49.1	L	N	N	N	Y	Y	Y	6	63	100	4.00	3.74	1.85E-7	8.65E-4	0.0	R	HL	RC-1A
SH-2G-2	LOFW,HPI(S),CHR(RVF)	3.0	2270	56.8	L	Y	N	N	Y	Y	N	32	61	100	2.97	2.91	7.10E-7	0.0	0.0	R	HL	RC-1A
SH-13A-1BX	SLOCA(3"),SI(R),CHR(T=0)	12.6	72	49.6	L	N	N	N	Y	Y	Y	5	61	100	2.72	2.50	2.43E-7	8.05E-4	0.0	R	HL	RC-1
SH-13A-3AX	SH-13A-3 W/ CHR(T=0)	11.5	51	48.0	L	N	N	N	Y	Y	Y	3	65	100	1.50	1.88	1.87E-7	8.06E-4	0.0	R	HL	RC-1
SH-RC-1B(5	SLOCA(3"),SI(O),CHR(O)	4.2	65	40.9	L	N	N	N	N	Y(2)	Y	5	46	100	0.13	0.16	2.30E-4	0.7	0.0	R	MH	RC-1B
SH1P-2	SBO,SEAL LOC(170 GPM)	9.0	768	71.4	VL	N	N	N	N	Y(2)	N	0	82	100	0.15	0.19	2.40E-7	0.0	0.0	R	LL	RC-1B
SH1P-1	SBO,SEAL LOC(250 GPM)	7.9	588	84.8	VL	N	N	N	N	Y(2)	N	0	78	100	0.039	0.077	1.20E-6	0.0	0.0	R	LM	RC-1B
SH1P-3	SBO,SEAL LOCA(21 GPM)	12.3	2440	82.3	VL	Y	N	N	N	Y(2)	N	0	88	100	0.017	0.031	8.90E-7	0.0	0.0	R	LL	RC-1B
SH13A1BY	SLOCA(3"),SI(O),CHR(O)	4.2	7	85.0	VL	N	N	N	N	N	Y	3	46	100	0.017	0.017	4.30E-5	0.0	0.0	R	LM	RC-1B
SH1P5-RC6	SH1P-5 - RESTART	14.2	54	89.6	VL	N	N	N	N	N	N	30	43	100	0.014	0.190	6.20E-4	0.31	0.0	R	LH	RC-6/RC-1BA

(1) Failed SG isolated per operating procedures.
(2) Accumulators discharge following RVF.
(3) Assumed DCH=100%, hydrogen burns forced at RVF, and containment failure pressure lowered to 90 psia.
(4) Assumes 100% of noble gases released - MAAP groups noble gases with all other inert aereosols/gases.
(5) Containment failure pressure lowered to obtain late (rather than "very late") failure.

- When assigning the early release categories to sequences with isolation failures, the main considerations were the state of the containment sprays, AFW, revaporization and, fission product retention in the RCS. Containment sprays operation is a necessary condition in attaining fission product reduction for early containment failures, since there is insufficient time for natural removal mechanisms (e.g., settling) to be effective. If the debris is covered but no sprays are available the source term should be of "medium" magnitude (with respect to other early source terms). This accounts for the possibility of imperfect scrubbing in the overlying water pool and the short time available for gravitational settling and other processes to work on any aerosol escaping the pool. High source terms are associated with uncovered debris and no sprays. The volatile fission products contribution to this high source term will be medium if primary system retention is high, and high if retention is low.

- When assigning the early release categories for the steam generator bypass sequences, the main variables were the amount of fission product retention in the RCS, the state of AFW, and the water level in the ruptured steam generator. In most of the postulated bypass cases AFW is available, however the operating procedures direct the operator to isolate the steam generator for the dominant SGTR sequences (i.e., those in which the steam generator is also faulted with an open relief valve). Thus, the ruptured steam generator will be dry in the majority of the cases, and no fission product scrubbing would occur.

- Note that those accident sequences in which containment failure is due to DCH should be assigned to early failure containment end points with releases not scrubbed. This is because this type of sequence can be thought of as having the debris not covered, to account for the enhanced release of non-volatile fission products during the DCH event itself. For these sequences only, it may also be prudent to assume that the containment sprays are ineffective in reducing the source term because of the promptness of the releases.

Note that one of the entries in Table 4–28 is the release characteristic. This entry was used to differentiate between the amounts of volatile and non-volatile fission product releases. The first digit represents the volatile releases, and the second the non-volatile contribution. The release fractions are grouped into high, medium and low groupings using the containment faiure timing for the PDS (early or late).

The break up is as follows:

Lates:		Earlies:	
volatiles (CsI-typ,%)	non-volatiles (SrO-typ,%)	volatiles (CsI-typ,%)	non-volatiles (SrO-typ,%)
H>1	H>1 x 10^{-4}	H>5	H>.1
.1<M<1	1 x 10^{-6}<M<1 x 10^{-4}	5<M<1	1 x 10^{-4}<M<.1
L<.1	L<1 x 10^{-4}	L<1	L<1 x 10^{-4}

When assigning release categories, for each group of representative sequences within a release category, that sequence with the highest release fraction was conservatively chosen to represent the particular category. The results of these groupings are presented in Table 4–29.

Table 4–29
Recommended Sequences for Release Category Assignment

RUN #	TYPE	RELEASE CATEGORY
SH–13A–1BX	SLOCA(3"),SI(R),CHR(T=0)	RC–1
SH–RC–1B	SLOCA(3"),SI(S),CHR(T=0)	RC–1B
SH–9A–1AX	MLOCA,LPI(S),CHR(T=0)	RC–1A
SH–RC–2	LOFW,SI(0),100% DCH,H2 BURNS,CHR(OK)	RC–2
SH–RC2–2	LLOCA,SI(0),CHR(0), ISOL(5")	RC–2B
SH–13A–4	SLOCA(3"),SI(R),ISOL(2") SRV	RC–3
SH–RC–3B	SBO, SEAL LOCA (250 GPM), AND 3.5" ISOL FAIL	RC–3B
SH–RC–4	SGTR,CYCLING SRV, SCRUBBED	RC–4
SH–RC–4C2	SGTR CYCLING SRV, UNSCRUBBED	RC–4C
SH–16B–1	SGTR, HPI(S), SRV STUCK OPEN	RC–5
SH1P5–RC–6	SBO, SEAL LOCA (21 GPM), AND HLF	RC–7

Due to the robustness of the SHNPP containment design, it was necessary to alter the PDS sequences in order to obtain meaningful release fractions. For example, to develop release categories RC–2 and RC–2B, DCH was set to a maximum, hydrogen burns were forced to occur at RPV failure, and the containment failure pressure was lowered below the estimated median failure pressure for SHNPP. This was done in order to make the containment rupture at RPV failure; otherwise no early catastrophic containment ruptures are predicted to occur. By implementing these extreme changes, an early release was generated and the source term identified. An additional MAAP run (SH_RC2–2), consisting of a large LOCA with large isolation failure and no containment heat removal (CHR), was performed in order to increase the source term (i.e. since LLOCA has low primary side FP retention). MAAP calculations RC–4 and RC–4C were designed to analyze SGTR with cycling SRVs sequences, which did not show up with the statistically significant PDSs.

Also note that the only difference between runs SH–13A–1BX and SH–RC–1 is that in the latter safety injection is assumed to fail to start, and the containment failure pressure was lowered to obtain containment rupture within the same time frame for both runs. Safety injection was deactivated from the start in SH–RC–1 so that the RWST would not discharge, and fission product releases would be unscrubbed. As expected, the non-volatile fission product contribution was higher in SH–RC–1; however MAAP calculated lower volatile fission product releases. This seems to be due to the lower containment failure pressure, assumed in SH–RC–1, providing a lower driving force for fission product emission in the second case. Nevertheless, when the sequence (SH–RC–1) was run allowing containment rupture at its median failure pressure (150 psig), then the sequence becomes a "very late" case with releases lowered due to the long time for natural fission product removal mechanisms (e.g., gravity settling, inertial impaction, and thermophoresis).

For all other release categories postulated, several conservative assumptions were made to assign source terms to the particular category. The following Table 4–30 presents the remaining release categories and the arguments used to resolve them.

Note that tellurium is also a volatile fission product but it is assumed in the analysis to be bound in the clad as a telluride (see section 4.2.2) and is released from the debris only if ex-vessel CCI occurs. It is released in large quantities soon after CCI begins and gives a good indication of releases due to ex-vessel CCI.

Table 4–30
Recommendations for Source Terms Assignment of Unanalyzed Categories

RELEASE CATEGORY	RECOMMENDATION	RECOMMENDATION REASONS
RC–5C	Use the same source term as RC–5.	This predicts conservatively large source terms. In any event, the likelihood of scrubbing in bypass sequences (STGRs) is low due to procedure directions to isolate the faulted, ruptured steam generator.
RC–1BA	Use volatiles from SH–9A–1AX and non-volatiles from SH1P5–RC6	SH–9A–1AX yields the largest volatile FP contribution for this group, but there is no CCI in this run. SH1P5–RC6 has the largest non-volatile contribution in this group. It is thus, conservative to use it.
RC–6	Use SH1P5–RC6 (i.e. RC–7) with releases reduced by a factor of 10	Containment failure mode for RC–6 is basemat melt through. The releases would be underground, thus the assumption was made to account for the reduction in atmospheric releases.

4.7.3 Plant Damage States Release Source Terms.

Using the results of MAAP runs for the representative sequences, the release fractions were obtained. The calculated radionuclide release fractions assigned to each release category are shown in Table 4–31. The release fractions in this table are listed by MAAP "groups" shown in Table 4–32. The isotopes within each group have similar chemical behavior.

Note that low tellurium releases, indicative of low CCI, is seen for most of the release categories. This is due to a number of sequence specific factors. For all the early release cases, the sequence postulated to fail containment had a RPV failure at medium to high pressure. Thus, for these cases MAAP calculates the majority of the debris is entrained into the lower compartment, where it is thinly spread and easily cooled avoiding significant CCI. For late sequences, the containment takes so long (2 to 3 days) to fail that natural fission product removal mechanisms (e.g., gravitational settling, and thermophoresis) may account for the low releases.

Table 4–31
Release Category Source Terms

Release Category	Nobles %	CsI %	TeO_2 %	SrO %	MoO_2 %	CsOH %	BaO %	La_2O_3 %	CeO_2 %	Sb %	Te_2 %	UO_2,ACT %
					Standard Source Term							
RC–1	100	2.72	0.0	2.4×10^{-7}	3.2×10^{-6}	2.50	2.0×10^{-6}	2.0×10^{-5}	2.0×10^{-5}	0.31	8.1×10^{-4}	(2)
RC–1A	100	4.0	0.0	1.8×10^{-7}	1.8×10^{-6}	3.74	2.1×10^{-6}	1.6×10^{-5}	1.6×10^{-5}	0.087	8.7×10^{-4}	(2)
RC–1B[1]	100	2.72	0.0	2.3×10^{-4}	3.2×10^{-6}	2.50	3.0×10^{-4}	1.5×10^{-4}	2.3×10^{-4}	0.31	0.718	(2)
RC–1BA	100	4.0	0.0	6.2×10^{-4}	0.077	2.50	0.002	2.7×10^{-5}	3.8×10^{-5}	0.046	0.31	(2)
RC–2	100	0.47	0.0	1.6×10^{-4}	0.0018	0.39	0.0035	0.013	0.018	0.15	0.0	0.0
RC–2B	100	4.1	0.0	0.250	0.0073	4.56	0.118	.341	.462	3.40	12.42	0.0
RC–3	100	0.067	0.0	3.0×10^{-5}	2.2×10^{-4}	0.069	2.6×10^{-4}	6.8×10^{-4}	6.8×10^{-4}	0.006	3.2×10^{-5}	(2)
RC–3B	100	1.82	0.0	9.9×10^{-4}	0.021	1.32	0.031	0.26	0.26	0.33	0.006	(2)
RC–4	100	0.22	0.0	4.7×10^{-5}	2.3×10^{-4}	0.176	2.6×10^{-4}	1.0×10^{-5}	1.0×10^{-5}	0.013	0.0	0.0
RC–4C	100	3.7	0.0	8.35×10^{-4}	0.0047	3.39	0.005	0.0013	0.0013	0.50	0.0	0.0
RC–5	100	59.05	0.0	0.011	0.28	52.6	0.067	9.2×10^{-5}	9.6×10^{-5}	3.3	0.009	(2)
RC–5C	100	59.05	0.0	0.011	0.28	52.6	0.067	9.2×10^{-5}	9.6×10^{-5}	3.3	0.009	(2)
RC–6	100	0.0014	0.0	6.4×10^{-5}	0.0078	0.019	0.0002	2.7×10^{-6}	3.8×10^{-6}	0.0046	0.031	(2)
RC–7	100	0.014	0.0	6.4×10^{-4}	0.078	0.191	0.002	2.7×10^{-5}	3.8×10^{-5}	0.046	0.31	(2)

1 Some values (volatile source terms) taken from the sequence for release category 1 since it predicts a higher release.

2 This release fraction is smaller than 1.0×10^{-8}%.

Table 4–32
MAAP Fission Product Groups

Group #	Group I.D.	Composition
1	Nobles	Noble gases and radioactively inert aerosols
2	CsI	CsI + RbI
3	TeO_2	TeO_2
4	SrO	SrO
5	MoO_2	MoO_2
6	CsOH	CsOH + RbOH
7	BaO	BaO
8	$La2O_3$	La_2O_3+Pr_2O_3+Nd_2O_3+Sm_2O_3+Y_2O_3
9	CeO_2	CeO_2
10	Sb	Sb
11	Te_2	Te_2
12	UO_2	UO_2 + NpO_2 + PuO_2

4.8 Summary and Conclusion

This section summarizes the insights gained into the SHNPP containment and discusses their importance in relation to the analysis results. The response of the containment to specific severe accident loads is also provided.

4.8.1 Summary of Results of the Containment Assessment

The SHNPP containment has a design pressure of 45 psig. Based on an assessment of the containment materials and structure, different containment failure modes are identified and examined to determine the most likely failure modes for the SHNPP containment (see Section 4.4). Basemat shear failure is found to be the most limiting containment failure mode. The capacity calculated for this failure mode is approximately 50 psi lower than the next likely failure mode. Examination of piping penetrations and the equipment hatch found that they possess higher capacities than the shear failure mode for the range of temperatures and pressures expected based on MAAP calculations.

A bounding curve, strongly influenced by the shear failure mode, is developed which enveloped the other potential failure modes. A mean capacity of 150 psig is generated. The omission of temperature from the selection of the mean capacity is based on analysis results which indicate that the containment strength does not change significantly over the temperature range of interest. The 150 psig value is similar to values found for other prestressed containment buildings and illustrates that the SHNPP containment is comparable to other sites.

Combinations of core damage sequences with frequencies in excess of 1.0×10^{-8}/year with the possible containment end states yielded 29 possible plant damage states. These plant damage states contained representative sequences which addressed a wide range of potential accident sequences. The wide variety of sequences provides added assurance that the CET results are valid over the range of accident sequences and initiators addressed in the core damage analysis.

The conditional probability of an impaired containment is 0.15. This compares favorably with other large dry containments and illustrates the strength of the containment. The dominant class of containment failure, accounting for 55% of the total containment failures, involves a small containment bypass caused by a steam generator tube failure. Very late containment failures, dominated by overpressure and basemat failures, account for 23% of the impaired containment probability. Late failures due mainly to hydrogen burns account for about 6% of the containment failure outcomes. Early containment failure contributes about 2% and isolation failures contribute another 2%. The small bypass category is represented by two sequences which involve SGTRs. SGTR sequences without makeup and a failure of the SRV on the ruptured steam generator contribute about 50% to this category. Loss of decay heat removal sequences which

result in an SGTR due to thermal failure of the steam generator tubes also contributes about 50%.

Ten sequences contribute to the very late containment failure category. These sequences are characterized by cases involving very long term containment failure due to basemat failure or late overpressurization caused by a failure to cool the debris ex-vessel. Station blackout sequences and low pressure sequences with the majority of the debris trapped in the cavity contribute to this category. Early containment failures are dominated by sequences which involve hydrogen burns that result in containment overpressure failure. These sequences all have containment heat removal available and low containment steam concentrations which support early hydrogen burning.

4.8.2 Conclusions Based on Containment Assessment

As important as understanding the results of the analysis is understanding how SHNPP characteristics affected the containment assessment. As previously stated, the containment mean failure pressure is similar to other large dry containments. As such, major issues related to large dry containments are used as a focus for conclusions gained into SHNPP containment performance.

Direct Containment Heating

DCH is an important challenge for large dry containments. If sufficiently large, the resultant pressurization could cause containment failure. For SHNPP, the calculated DCH pressure load, assuming complete dispersal and hydrogen burn, is much less than the mean containment capacity. This assessment, however, is conservative and a more realistic assessment is that less than 50% of the material will be involved and an even smaller fraction will be finely fragmented. For high pressure sequences, consideration of both MAAP and NUREG–1150 results, yielded peak containment pressures on the order of 100 psi and represents a negligible likelihood of containment failure. Offsetting the potential for DCH is the potential that the RCS will be depressurized prior to vessel failure due to thermal failure of RCS piping. SHNPP hot leg piping has a higher capacity than earlier designs that somewhat reduces the potential for this effect. A review of temperature profiles predicted by MAAP combined with considerations present in the NUREG–1150 analysis leads to the conclusion that it is likely that RCS piping failure will occur. The piping failure results in depressurization of the RCS prior to vessel failure and considerably reduced the contribution to containment failure due to DCH loading.

The assessment of the SHNPP cavity design supports the overall insight that DCH is not an important contributor for SHNPP. The interface between the cavity and the upper containment is not direct and this configuration helps to promote deposition of fragmented debris in the lower compartment. By preventing large scale communication

between the upper compartment and the cavity, heating effects are limited to the escaping gas and large transfers of heat from the debris to the containment are not possible.

Hydrogen Generation and Burning

In assessing the hydrogen source terms for SHNPP, the information contained in NUREG–1150 was combined with plant-specific MAAP assessments to develop potential hydrogen source term distributions. This approach allows the uncertainties present in hydrogen generation and burning to be addressed

Bounding assessments predict that oxidation reactions must generate a 96% metal-water reaction to raise the hydrogen concentration to about 14%, which is a lower bound for hydrogen detonation to occur in dry air. In a more realistic case, the steam concentration will be high enough to preclude detonation. With regard to detonation and DDT, the SHNPP containment provides adequate communication between the containment compartments to preclude large, local hydrogen pockets. Additionally, the main discharge points for hydrogen, e.g., pressurizer relief tank, are found in large open areas which should result in quick hydrogen dispersal. The presence of containment sprays or fan coolers for the vast majority of sequences increases circulation and reduces the potential for large hydrogen gradients and the potential for local detonations or DDT.

Late restoration of containment cooling presents another opportunity for hydrogen detonation. If, after prolonged CCI and additional hydrogen generation, containment cooling is restored, the hydrogen source term in the containment may support detonation. Two factors, however, support the insight that detonation is not likely for this situation. First, the containment sprays and fan coolers will require many hours to decrease the steam concentration to flammability limits. Thus, no sudden reduction in steam concentration will occur. Second, as steam concentration decreases, global hydrogen deflagration will occur as soon as conditions permit. This will deplete oxygen and hydrogen and prevent detonation.

Based on calculated steam concentrations and MAAP analysis, the containment will be steam inerted prior to the core being uncovered for all cases except those involving containment bypass. Thus, hydrogen burning is very unlikely for cases without a means of condensing steam. For cases with containment heat removal, hydrogen burning is not expected to occur due to low hydrogen concentrations that are below the 8% concentration needed for global burning to occur. For upper limit in-vessel generation, however, burns are possible. If 75% metal-water reaction is assumed, a complete burn will result in a final pressure of 80 psi. This represents a negligible probability of containment failure and is well below the mean capacity.

If containment cooling is restored late in a sequence, hydrogen burning is almost assured. The main uncertainty in this instance is the amount of available hydrogen, which is a

direct function of time. The later recovery occurs, the higher the hydrogen source term. Cases with debris cooling, are dominated by in-vessel generation. For noncoolable debris cases, the hydrogen source term is expected to exceed in-vessel generation. Since in-vessel generation provides only a limited potential for hydrogen burning the combination of in-vessel and ex-vessel sources would result in higher likelihood of a burn occurring. The resultant pressurization due to the burn would be dependent on the timing of the burn which would determine the amount of basemat ablation and, thus, the hydrogen source term. Given the containment capacity, however, the containment failure probability is modest for the most likely hydrogen source terms.

The potential for detonation or DDT is considered remote for the SHNPP containment due to adequate mixing and the presence of steam for most sequences. Hydrogen burning can occur if containment cooling is present. Without containment cooling, steam inerting will occur prior to significant hydrogen production and will preclude early hydrogen burns. Assuming a burn the resultant pressure loads are expected to be small for the best estimate assessment and moderate for late burns following recovery of containment heat removal. The probability of containment failure due to hydrogen burning, however, is low.

Ex-Vessel Debris Cooling

Addressing ex-vessel debris coolability is explicitly requested in the guidance for completion of the IPE submittal. The CET provides a detailed model for assessing the ability of the debris to be cooled ex-vessel. In addition, assessments were also performed and the results documented in Section 4.2. Based on these discussions, the conclusion is made that ex-vessel debris cooling is likely for SHNPP if the RWST is injected into the containment and the debris is dispersed by fragmentation at high pressure or if a steam explosion occurs. The supporting evidence is summarized below. If the RWST is not injected, debris cooling is not expected.

Based on SHNPP design considerations the expected steady state heat flux needed to cool the debris is on the order of 100 kw/m^2. This heat flux is low and supports the notion that the debris is coolable if dispersed. If dispersal does not occur and the entire core expelled to the cavity, a debris depth of about 22 cm is found which is below the value (25 cm) referenced in generic letter 88–20 as the point at which debris cooling could be assumed successful. Thus, cooling can be accomplished for a wide range of debris expulsion rates if water is available. One potential issue which can change this conclusion is the presence of debris pooling under the vessel after RPV failure. It is believed that about 50% of the debris will be retained after RPV failure. This material will melt and relocate to the cavity after the initial RPV blowdown. If the debris piles under the RPV, as opposed to spreading, debris cooling could be significantly effected and perhaps precluded.

Two other issues related to debris cooling are basemat failure and gas generation during prolonged concrete attack which leads to a loss of containment integrity. The depth of the basemat is approximately 12 ft at SHNPP. The basemat rests on rock of undetermined thickness. The rock may provide additional protection but no credit is taken in the analysis. Based on ablation rates predicted by MAAP, the expected time required to breach the basemat is on the order of 90 hours after the start of the event.

Containment overpressure due to steam generation is expected to occur at about the same time due to the slow gas generation rate. The SHNPP concrete aggregate is similar to basalt and is not subject to the large gas generation rates attributed to limestone concrete. In either case, the time to containment failure is very long which provides considerable time for accident management and the potential for recovery. Some prior studies have considered the time available adequate to ensure recovery and not considered the possibility of containment failure.

Overall Containment Performance

The SHNPP containment is robust with regard to postulated severe accident loadings. The frequency of potential containment failures are provided in Table 4–33. The low probability of early containment failure and the long time to failure for the majority of sequences confirms that the operations staff will have time to implement accident management actions. The large area available for the debris to spread and presence of water in the cavity for most cases, with the exception of station blackout sequences, provides high confidence that the debris will be coolable following RPV failure.

Table 4–33
Containment State Frequencies

Containment State	Frequency (/year)
Early Containment Failure	1.80×10^{-7}
Late Containment Failure	6.70×10^{-7}
Very Late Containment Failure	2.50×10^{-6}
Small Isolation Failure	1.90×10^{-7}
Large Isolation Failure	1.00×10^{-8}
Small Containment Bypass	2.53×10^{-6}
Large Containment Bypass	2.49×10^{-6}
Containment Failure after In-Vessel Recovery	2.27×10^{-6}

The release classes are provided in Table 4–34. The dominant release category (RC–4) involves containment releases through the steam generator. The ruptured steam generator does not contain water and radionuclide scrubbing through the break does not occur. As

a result, the volatile source terms are large. For cases with very late containment failure the release fractions are very small due to radionuclide deposition in the containment.

The dominant PDS contributor for RC–4 is PDS 2A and involves a loss of all decay heat removal. In this case, containment cooling is available but a steam generator tube rupture occurs late in the accident progression due to thermal stress. The failure of the tubes provides a path for the radionuclides to escape the RCS and bypass the containment. The release is not scrubbed due to the failure of secondary-side heat removal and isolation of the steam generator. The releases from this sequence are large due to the bypass leakage and approach the release fractions predicted in WASH–1400 for release category PWR–4.

Table 4–34
Release Category Frequencies and Major Release Constituents

Release Category	Frequency (/year)	Nobles	CsI	TeO_2
RC–4	3.83×10^{-6}	100%	0.22%	e[1]
RC–5	2.81×10^{-6}	100%	59.05%	0.01%
RC–6	2.47×10^{-6}	100%	0.001%	0.031%
RC–1B	5.82×10^{-7}	100%	2.72%	0.718
RC–4C	5.48×10^{-7}	100%	3.7%	e
RC–3	1.89×10^{-7}	100%	0.067%	e

[1] e– less than 1.0×10^{-4} 1.04 E-5

The next highest release category is RC–5. Release category RC–5 represents the largest release and is a result of an ISLOCA or an induced SGTR with failure of a steam generator SRV. The failure of the SRV results in a large release without scrubbing which is similar to the ISLOCA. The ISLOCA is due to a failure in the RHR system. Both cases result in large release fractions due to the early failure of containment and the lack of radionuclide retention in the containment.

The third highest contributor, category RC–6 is comprised of contributions from over 20 PDSs. The major contributor is PDS 1P which is dominated by station blackout sequences. These sequences involve failure of the containment due to basemat failure. Due to the very long times involved, a large portion of the radionuclides have settled in the containment and the release is quite small. The other PDS which contributes is an S2 LOCA with a failure of recirculation. Basemat failure occurs associated with concrete ablation due to a failure to cool the debris ex-vessel.

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5.0 UTILITY PARTICIPATION AND REVIEW TEAMS

5.1 Project Organization

5.1.1 Project Management

Since the early 1980's, an overall objective of CP&L has been to develop and use probabilistic analysis methods to understand nuclear power plant risks. Meeting this objective provides information to help plant and corporate management make better decisions affecting nuclear safety. CP&L development of PRA expertise began in the Corporate Nuclear Safety organization with systems level probabilistic analyses. In 1988, a preliminary level I PRA was completed for SHNPP with the help of an outside contractor, Science Applications International Corporation (SAIC). The project was managed by SAIC with technology transfer as a major objective. CP&L personnel performed approximately 50% of the project work in meeting that objective.

When the regulatory requirement for development and submittal of IPEs was promulgated in 1988, CP&L was in a strong position to build on its in-house PRA expertise and the work completed for the SHNPP, and to perform a significant amount of the PRA work in house. Work was begun on updating the SHNPP Level I PRA to comply with the requirements of Generic Letter 88–20 (Ref 5–1) for the IPE submittal.

Because of personnel changes in the PRA organization and the demands of completing three essentially unique IPEs for SHNPP, H. B. Robinson plant and Brunswick plant, it was decided to proceed with the other two nuclear plant PRAs based on the expected low risk for SHNPP. Work was suspended on the SHNPP PRA, and was not resumed until March 1992.

All the previous level 1 effort was carefully reviewed and updated to reflect the plant configuration as well as incorporating new advances in PRA technology and computer applications. The previous modelling efforts were used as guides to finalize the system and event tree models. The database was completely updated using the most current information available and plant-specific data for the period of commercial operation of SHNPP. The common cause failure analysis was completely redone, as was the human reliability analysis. New internal flooding and ISLOCA analyses were completed and incorporated into the PRA. All effort associated with the level 2 PRA work represented new analyses which were completed in early 1993.

The preceding discussion provides the background for descriptions of the SHNPP IPE project management and review processes. The project has evolved over a considerable time period and is the result of contributions from many CP&L employees. Plant

operations, technical support and training personnel have been involved since the beginning and in all phases of the project.

CP&L's PRA organization has also evolved during development and application of the SHNPP PRA. The IPE described in this report was developed from the PRA by the project team shown in Figure 5–1. This figure reflects the organization that existed in 1992 when the bulk of the IPE level I work was performed. The PRA organization (Risk Assessment Section) currently resides in the Nuclear Services Department since its transfer from the Nuclear Engineering Department in June 1993.

Figure 5–2 shows the SHNPP Severe Accident Issues Project Team that was formed in early 1993 to assess potential vulnerabilities, evaluate alternatives to address them and recommend actions to resolve severe accident issues using the NUMARC IPE closure guidelines. This team served several additional purposes including internal review of results and their implications. The depth of experience of the plant operations, training, and technical support personnel assigned to this team was a significant factor in its successful resolution of the issues. This review forum also fostered an appreciation for behavior of the plant during severe accident conditions, the likelihood of severe accidents and the potential contributors to such accidents. The plant members of the team have considerable nuclear plant experience and demonstrated abilities. Their credibility will result in a growing appreciation throughout the plant staff for severe accident issues and appropriate actions to address them. This team will continue to address severe accident issues from the IPE and other elements of the NRC Severe Accident Program, e.g. Accident Management, IPEEE.

Mr. R. E. Oliver, manager of Risk Assessment, is responsible for the PRA efforts at CP&L, including management responsibility for the IPEs for all three nuclear plants. His responsibilities also entail leading the efforts to integrate PRA and its insights with other corporate safety and licensing efforts. For the last six years Mr. Oliver has directed the overall efforts to complete the IPEs while ensuring that the CP&L staff continually gained PRA expertise. In addition, CP&L staff not directly involved in performance of PRA were familiarized with the technology to facilitate interactions with personnel who might deal with the PRA group or have occasions to use PRA insights.

Project management of the SHNPP IPE was provided mainly by the lead engineers for the project, Mr. A. J. Howe of CP&L and Mr. R. L. Summitt of Safety and Reliability Optimization Services (SAROS), Inc. For the SHNPP PRA and IPE submittal, Mr. Howe also had overall project technical responsibility, and developed or reviewed every major portion of the PRA. Mr. Howe is experienced in fault tree analysis techniques and technology, and has previously worked at SHNPP, having held an SRO license at the plant. Mr. Howe and Mr. Summitt were directly involved as technical leads for a number of tasks in the study.

Figure 5–1
SHNPP PRA Organization Chart

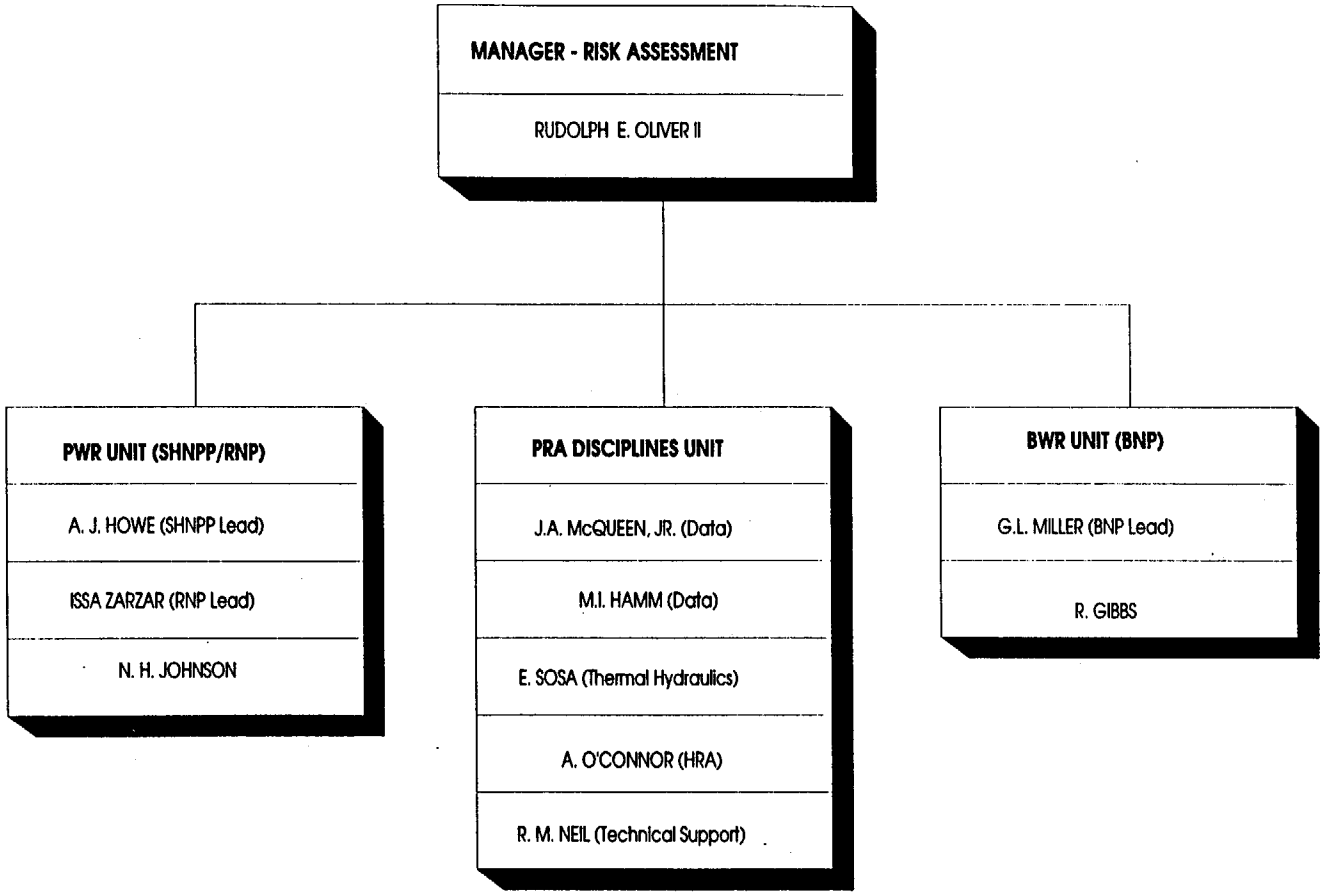
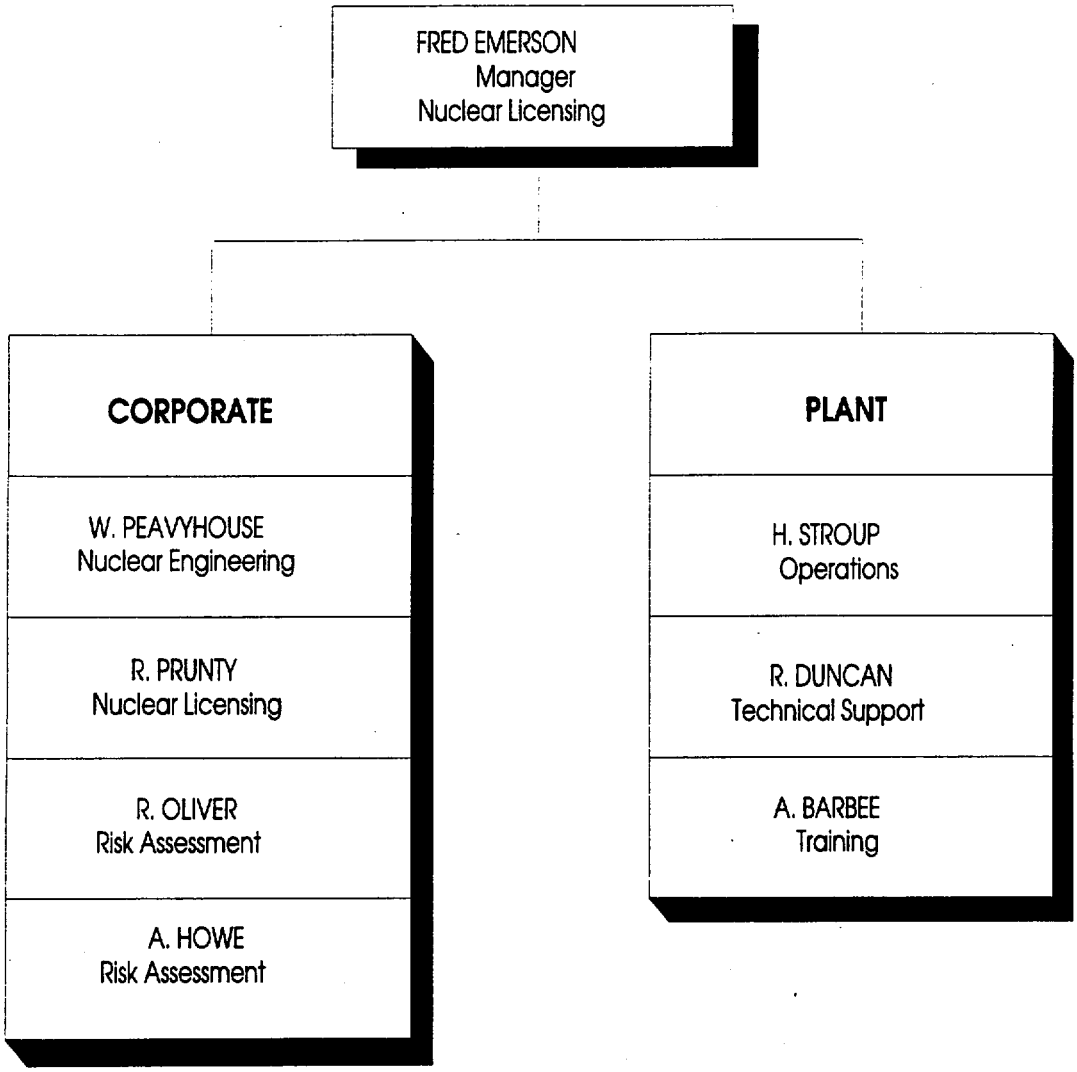


Figure 5–2
SHNPP PRA Severe Accident Issues Project Team



The performance of the tasks generally involved teams, with one individual acting as task leader for particular areas. The technical responsibilities for the PRA tasks are discussed in the next section.

5.1.2 Technical Organization

As previously discussed, the technical work on the study involved numerous CP&L people. Outside consultants with specialized expertise were called in to assist the CP&L staff in the methodology and in the performance of some tasks. A brief review of the conduct of the study is provided here.

As noted previously, a PRA effort was initiated by CP&L prior to the generic letter. Although it formed a solid basis for analysis, all tasks in the initial PRA were redone when the study was formally tied to the IPE objectives. The remaining description applies to this phase, which involved reworking all the initial PRA tasks to upgrade the technology, update the study to reflect the current plant, and to address IPE-specific objectives.

The systems analysis tasks involved the largest number of analysts. The accident sequence delineation and initiating event and event tree development work was done by Mr. Summitt of SAROS, Inc. and was reviewed by Mr. Howe of CP&L. This team effort ensured that the tasks had the perspective of years of PRA experience in different PWR PRAs provided by Mr. Summitt as well as the plant-specific input provided by Mr. Howe. This team was assisted in specific areas by other CP&L staff. Mr. M. Pope (Fuels Section) and Mr. E. Sosa supported the success criteria development through thermal-hydraulic analyses specific to the SHNPP plant. Mr. S. Thomasson of CP&L performed the room heatup analysis to determine plant-specific HVAC requirements.

The fault tree task was a more labor intensive effort, and some additional contractors supplemented the CP&L staff. Mr. A. Howe, and Mr. M. Hamm and Mr. R. Neil, also of CP&L, were assisted by Ms. J. Jansen and Mr. W. Gough, of SAIC, and Mr. R. Brooks of Tenera in the notebook development and fault tree effort. These analysts all have expertise in fault tree construction and had assignments consistent with their prior experience. The major portion of the work was completed onsite and the analysts had access to all current plant information, as well as to systems engineers and others with knowledge essential to realistic model development. The fault trees and notebooks were subjected to several review cycles, as discussed in the next section.

The data development effort was led by Mr. J. McQueen of CP&L, who has been involved in PRA data development efforts for all CP&L plants for several years. Mr. M. Hamm, also of CP&L, assisted in the data development effort as did Mr. E. Tibbs of SAIC. Mr. Hamm was also responsible for the development of common cause failure

analysis. Mr. Summitt provided technology transfer and developed the generic data base used in the study.

The human reliability analysis task was led by a CP&L staff member, Ms. A. O'Connor, with technology transfer from a specialist in the discipline, Mr. G. Parry, of NUS.

The ISLOCA analysis was performed by Mr. Brooks of Tenera, and the internal flooding analysis was performed by Mr. Boyd or SAROS, with assistance from CP&L staff on equipment locations and flood response modeling. In all of these specialty areas CP&L personnel reviewed and became fully cognizant of the results and implications of the analyses. Documentation was designed to ensure that information, rationale and insights provided in these areas were clear, concise and reproducible.

The accident sequence results were quantified by Mr. Howe of CP&L, assisted by Mr. Gough of SAIC. Mr. Howe, Mr. Gough and Mr. Summitt participated in the cutset review and reiteration. Once final results were generated, additional CP&L staff from the plant and the corporate office were involved in the review and validation of the results.

The back-end tasks required significantly more contractor assistance, with the lead in those task area being provided by Mr. Summitt with assistance from Mr. Howe. Mr. Sosa of CP&L generated the plant specific MAAP runs and participated in the back-end analysis. Additional expertise concerning back-end phenomena was provided by Mr. M. Kenton of Gabor, Kenton and Associates, Inc. As with the specialty areas, CP&L personnel reviewed and became cognizant of all results and implications. Documentation was designed to ensure that information, rationale and insights provided in this area were clear, concise and reproducible.

5.2 Review Processes

The initial SHNPP PRA was subjected to an independent review effort (Ref 5–2), consistent with the review methodology described in NSAC–67 (Ref. 5–3). The review team consisted of the following highly experienced practitioners: Mr. S. Lewis of SAROS, Mr. J. Gaertner of Electric Power Research Institute, Mr. E. Hughes of ERIN Engineering & Research, Inc., Mr. G. Parry of NUS Corporation; and Mr. J. Stetkar of Pickard, Lowe and Garrick, Inc. At the time of this review, all of the reviewers were completely independent of the SHNPP analysis that had been completed to that date (December 1988).

CP&L used these comments in planning for the PRA revision and update. The analysis team generated responses to each comment and identified action items to ensure that the comment resolutions would be represented in the PRA model update. A notebook was

developed as part of the PRA documentation that provides a response to each comment from the external review.

The PRA revision and update process included additional review steps. Internal to the project team, there were steps to ensure that the task products were correct, sufficiently complete, and consistent. Task outputs were reviewed by other PRA team members. The systems analysis fault trees were subjected to independent review by experienced PRA practitioners not involved in their initial development. These reviews were documented, and systems analysts responded to the review comments in the update of their trees and notebooks.

The next step involved the review of the systems analysis by plant Technical Support (systems engineers). The reviewers were independent of the PRA staff that prepared the models and notebooks. This review step helped to ensure that the analyses represented the facility as currently designed and operated. After receipt of comments from this review team, the PRA analysts developed and documented responses, and changed the systems analyses as necessary.

After the PRA results had been developed, an internal review of the quantified cutset results was performed with plant Operations and Training personnel. The individuals involved were highly experienced shift operating supervisors and shift technical advisors with current SRO licenses and equally experienced members of the training unit. Additional expertise was provided by a member of the corporate engineering and nuclear licensing departments, as well as outside contractors and a member from Institute of Nuclear Power Operations. This review provided assurance that the cutsets were valid and also identified where recovery actions might be considered.

This review substantially validated the results of the PRA. Especially relevant were discussions of the operator actions in the sequences, and verification that the sequence results were consistent with plant operating procedures and practices.

Review comments and recovery actions were incorporated into the model and documented. The model was then requantified and the new results evaluated for potential vulnerabilities to severe accidents and their contributors.

For the level 2 analysis, an internal review process was used. The backend analysis was performed using a team comprised of SAROS, GKA, and CP&L. A team member was assigned the responsibility for a particular area of the analysis and the other team members served as internal peer reviewers. Peer reviews were held during periodic review meetings were held between CP&L, SAROS, and GKA. For example, the team member from SAROS was responsible for the development and quantification of the SHNPP CET. During these meetings, GKA and CP&L representatives served as team

peer reviewers for the CET development. In a likewise fashion GKA and SAROS reviewed MAAP analyses performed by CP&L for accuracy. Although some interdependency existed, these reviews served as a means of ensuring technical accuracy.

Table 5–1 provides a breakdown of backend responsibility.

Table 5–1
Backend Task Responsibilities

Task	CP&L PRA	SAROS	GKA
Containment Event Tree Development	R	D	R
Containment Event Tree Event Quantification	R	D	R
PDS MAAP Analysis	D	R	R
Special Issues Analysis and White Paper	R	R	D
Release Category Definition	D	R	R

Legend: D – Primary responsibility for development, R – Internal reviewer.

After the results were essentially finalized, a team was established to examine the results, develop insights, and consider any potential enhancements to address the insights. The organization, purpose and function of this Severe Accident Issues Project team was discussed in Section 5.1. This team held a series of meetings in May and June 1993. The team reviewed all significant accident sequences and considered alternatives such as plant modifications, procedure changes and training enhancements to address them. This team will continue to review and assess severe accident analysis results from future SHNPP PRA updates and other elements of the NRC Severe Accident Program.

The review activities associated with the project from its initiation through completion provide assurance that the model represents the current facility and the results are valid. A brief summary of the review findings is provided in the next section.

5.3 References for Section 5

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6.0 PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

6.1 Plant Improvements

As described in Section 3.4, a multi-disciplinary team was established to evaluate the IPE results and suggest areas for potential improvements. The results of the evaluation are summarized below. The recommendations include some immediate changes and areas for further investigation to see if other changes are cost effective.

The team, plant staff, and management agreed to the following actions:

- Revise operating procedures to provide explicit instructions for locally aligning offsite AC power if the breakers fail to automatically actuate and cannot be controlled from the main control board. These procedures are now in effect.

- Investigate the feasibility of installing instrumentation for improved battery monitoring capability, especially for detection of open circuit faults while the bus is carried by the battery charger. This would eliminate the potential for the open circuit or loose terminal failure mode.

- Ensure the testing and maintenance practices for the non-vital 125 VDC battery to be the equivalent of practices for the safety-related batteries. This was verified to be the plant practice, and no changes to maintenance or testing procedures were required.

There were no changes resulting from the analysis of containment response.

6.2 Unique Plant Features Important to Safety

As noted in previous sections, the SHNPP PRA provided insights concerning important plant features. As discussed in Section 6.1, procedure changes were accomplished, and additional efforts are underway to improve the reliability of the non-vital 125 VDC system. A summary of important plant features is provided below.

Level 1 Analysis

The SHNPP results indicate that accidents initiated by small LOCAs are the dominant contributors to core damage frequency. One important reason for this is the combined function charging and high pressure safety injection pumps (CSIPs). SHNPP design does not provide separate intermediate pressure safety injection pumps. The CSIPs are normally aligned to the VCT, and on a SI signal, must automatically realign to the

RWST. This flowpath cannot be routinely tested during power operations since the borated RWST water would enter the RCS and cause a temperature and power transient. As a result of the long testing interval, the motor-operated valves at the suction of the CSIPs are a dominant failure mode of the high pressure safety injection function. Limited credit for operator recovery is taken, since if the CSIP suction alignment fails, the VCT will provide only a short period of operation for the CSIPs when they realign the discharge to the safety injection headers. Once the VCT is depleted, the CSIPs would fail due to a loss of suction.

Another effect of the combined function CSIPs is the impact on RCP seal cooling and mitigation of LOCAs. The CSIPs provide RCP seal injection flow, which together with the CCW cooling of the RCP thermal barrier heat exchangers, maintains the seal cooling function. The CSIPs also provide the high pressure safety injection function, which is required to mitigate small LOCAs, including those resulting from RCP seal failures following a sustained loss of seal cooling. In other plant designs, the separate charging pumps provide RCP seal injection, and the safety injection function is provided by separate safety injection pumps. Failures of the CSIPs at SHNPP cause a loss of redundancy of the seal cooling function, and also cause a loss of the primary means of mitigating a RCP seal failure, specifically the high pressure safety injection function. Thus the total number of system failures which must occur to cause a RCP seal failure and mitigating system failure is reduced to two systems.

Another dominant contributor to core damage frequency is loss of offsite power, either loss of power at the switchyard, or interruption of offsite power coming into the plant. Although the contribution of these failure modes is not unique to SHNPP, the configuration of the non-vital 125 VDC system at SHNPP does significantly impact the probability of such events occurring.

As previously discussed, SHNPP uses an automatic fast bus transfer on any main generator trip to switch the source of plant AC power from the main generator to the offsite power grid. There are eight breakers which must change position to accomplish this transfer, and each receives control power from the non-vital 125 VDC system. Failures of this DC system, therefore, can cause a failure of the automatic bus transfer and induce a loss of offsite power, requiring a demand on the EDGs.

A failure of the non-vital 125 VDC system will cause a plant trip due to effects on the secondary plant. In addition, for any plant initiating event, if the battery which supplies backup power to the DC bus fails by an open circuit failure mode (e.g., loose or corroded terminal connections), then this will also cause a failure of the automatic bus transfer since AC power to the non-vital battery chargers is momentarily interrupted during the transfer process. The various failure modes of the non-vital 125 VDC system would

dominate the SHNPP PRA results in the absence of operator action to locally operate the failed electrical breakers.

The SW system at SHNPP is a positive unique feature. It is comprised of two independent subsystems the NSW system and the ESW system.

The NSW system has two 100% capacity non-safety-related pumps which draw suction from the cooling tower basin and recirculate the water through the plant heat exchangers and back to the tower. During normal operations, only one pump is in operation providing flow to all normal and safety-related loads, including the ESW headers. If the operating pump fails due to electrical malfunction, the standby pump is automatically started.

The ESW system has two independent 100% capacity safety-related pumps. There are no dependencies between these two trains. The ESW system is automatically actuated on an SI signal or on a loss of offsite AC power. In addition, the ESW pumps automatically start on low ESW header pressure in the event of a failure of the NSW system.

Due to the diversity and redundancy of the SW system, multiple failures are required to cause a loss of function, and so the contribution of SW to core damage frequency is not as great as might be expected.

Level 2 Analysis

The analysis of accident progression, containment response, and radionuclide release yielded other insights about important plant features. The most important aspects of the plant design that influenced the level 2 analysis are discussed below.

The evaluation of containment strength yielded an estimate that the ultimate containment pressure capacity is 150 psig. This substantial pressure margin ensures that most severe accident containment loads are well within the containment capability, reducing the probability of containment failures due to overpressurization. This, in turn, reduced the probability of early containment failure since for most cases the pressurization due to steam, hydrogen burn, and possible direct containment heating are less than the containment strength. This is not a unique finding, as other plants with prestressed concrete containments have exhibited a resistance to failure due to overpressurization. Localized failures, isolation failures, and large containment bypass accidents were found to be low in frequency, offering further assurance that the high capacity would not be compromised by other failure modes.

Contributing to the low probability of containment overpressurization are features that reduce the containment loads. For example, there are two design features that minimize

the risk of containment failure due to direct heating. The first was noted above—the containment ultimate strength is high relative the any potential overpressurization associated with direct heating. The other feature is the cavity/upper containment interface design, which limits direct communication. This design limits the transport of particles to the upper containment, reducing the immediate heat load and therefore reducing the expected pressurization. The loads associated with hydrogen deflagration are typically less than containment strength and there is a very limited potential for detonation of hydrogen. The containment is well mixed without significant localized hydrogen collection and detonation potential.

With regard to the response of the SHNPP containment to sequences involving failure of the reactor vessel, two features are important. First, the core debris was assessed to be frequently coolable because the geometry of the plant will generally allow the core debris to be spread over large surface areas, thus lowering the depth of the material and increasing the probability of successful cooling. The second important containment design feature is the basaltic-based concrete in the basemat. This type of concrete exhibits considerably less gas generation than limestone concretes, and most accidents at SHNNP were assessed to have very slow late pressurizations for sequences involving basemat attack. In most cases it would be a very long time before overpressurization in these types of accidents, allowing time for contingency actions and accident management. This feature will be considered more directly in future considerations of accident management potential at SHNNP.

The only specific design feature which exacerbates the outcomes for some of the accident sequences is a procedural action that, in an effort to provide some core cooling, results in restart of the RCPs when there may be very hot gases in the vessel, which would then be transferred to the steam generators. In sequences involving this type of procedural response there is an increased potential for a temperature-induced failure of a steam generator tubes with resultant loss of integrity and small containment bypass. This type of sequence contributes about 25% of the containment failure frequency.

The radionuclide releases associated with accidents are largely affected by the design features already noted. The large capacity reduces the potential for early releases, and in combination with the basaltic concrete assures long holdup times prior to ultimate release. The holdup time allows deposition and decay of the radionuclides, reducing the size of the ultimate release. Containment sprays were found to reduce the release for all sequences in which they were available. The limited core-concrete potential due to the large potential for successful debris cooling further reduced the releases associated with late basemat attack. The most important releases were found to be associated with SGTR sequences without water scrubbing.

7.0 SUMMARY AND CONCLUSIONS

7.1 Summary

Carolina Power and Light (CP&L) has completed a probabilistic risk assessment (PRA) of the Shearon Harris Nuclear Power Plant, Unit No. 1 (SHNPP). This report describes the results and illustrates the accomplishment of one of the principal goals of the study, compliance with the Nuclear Regulatory Commission's (NRC) Generic Letter 88–20 (Ref 7–1), requesting individual plant examinations (IPEs). The methods and results of the study are contained in this report in form and content called for in the *Individual Plant Examination: Submittal Guidance*, NUREG–1335 (Ref 7–2). The report demonstrates achievement of the goals called out in the generic letter. Specifically, CP&L has shown that it has:

- an appreciation of severe accident behavior from initiating event through the potential physical processes of core damage and possible containment response;
- an understanding of the significant characteristics of the potential severe accident sequences that could occur, including the potential failures, involvement of the operations staff, the timing, and the potential for recovery;
- a recognition of the quantitative aspects of the potential for severe accidents, including the most likely sequences and failure modes, and the sensitivity of the results to quantitative input; and
- an understanding of the results with identification of improvements that will be implemented to reduce the frequency or consequences of any significant sequences.

These goals were not the only ones established by CP&L in developing a PRA model. In particular, CP&L has developed models that can be easily maintained and exercised to study plant safety in the future, including assessment of any changes in data or understanding or evaluation of plant or procedural modifications. In order to facilitate its use for these purposes, additional PRA documentation is available which exceeds the submittal requirements, allowing recreation of all of the analyses.

The results of the study include the definition and quantification of potential accident sequences that could result in core damage, as well as an evaluation of the accident progression, resultant containment response and potential radionuclide releases. A brief

summary of those findings is provided here. Sections 3.4 and 4.8 describe the study results in more detail.

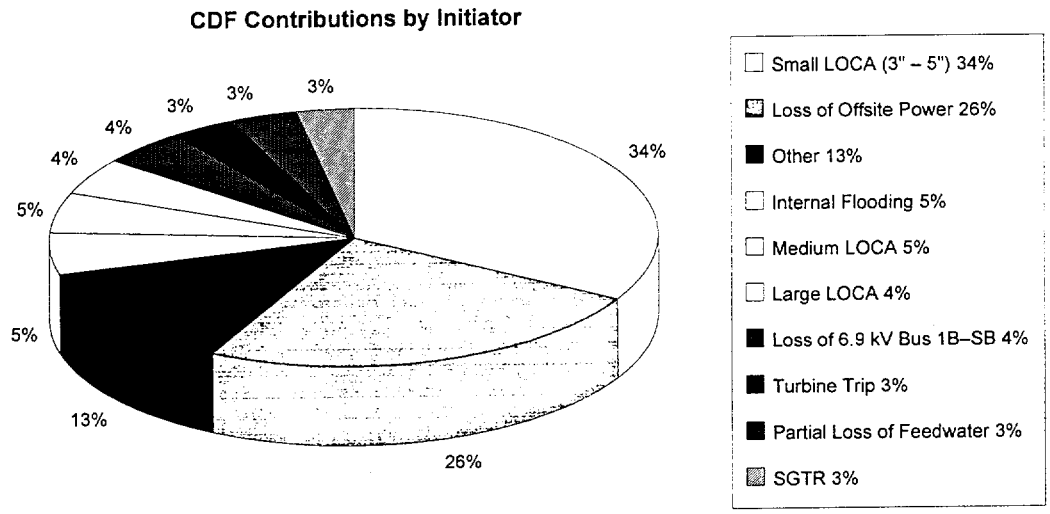
The types and frequencies of core damage accident sequences are shown in Figure 7–1.

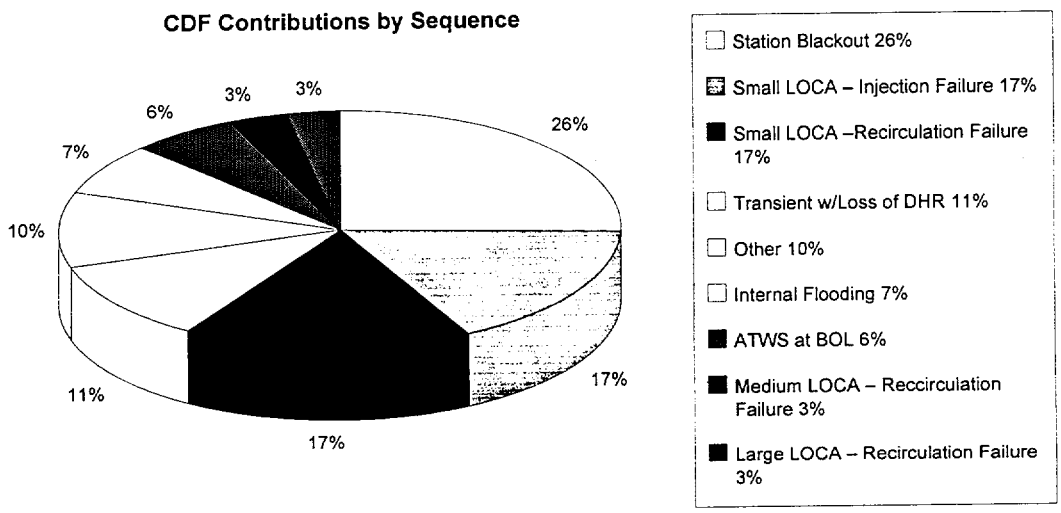
The first feature that can be derived from the results is that the SHNPP core damage profile is not dominated by any one particular accident sequence. This is illustrated in Figure 7–1. As is indicated in the first chart in the figure, no single type of initiating event is responsible for a large part of the overall core damage frequency, although small (3 – 5 inch) loss of coolant accidents (LOCAs) and loss of offsite power are the top two initiating event contributors, accounting for about 60% of overall core damage frequency. The second chart illustrates this point further in terms of accident sequences. The four most frequent individual event tree sequences account for about 70% of overall core damage frequency. These results suggest that there is no particular plant feature that creates a unique accident type that is predominant.

The evaluation of the insights from this study requires consideration of what makes individual accident sequences more important than others, with a special emphasis on identifying any commonalties that might influence a number of accident sequences. The overall core damage frequency of 7.0×10^{-5}/year is not atypical compared to PRA results of other similarly designed Westinghouse pressurized water reactors (PWRs). After appropriate reviews to verify that the study accurately represented the facility, it was necessary to more thoroughly consider any insights. In order to determine the possibilities for improvement at SHNPP, a special team was formed with responsibility for examining the results and evaluating potential enhancements. This team provided the broad perspective needed to evaluate effective, cost-efficient improvements. It was composed of representatives from operations, training, technical support, corporate engineering, licensing, and PRA analysis. This team examined the results and suggested some changes and areas for further investigation, the highlights of which are presented below.

Sequences involving failure of the non-vital 125 V DC electrical system were found to be significant contributors to overall core damage frequency. The DC system provides control power to all breakers which must change position to transfer plant power from the main generator to the switchyard on a plant trip. A failure of the DC system would inhibit this transfer, causing a loss of offsite power and a reliance upon the emergency diesel generators. Because of the significance of this failure mode to overall PRA results, procedure changes were implemented to direct local operation of the breakers in question if DC control power failed. In addition to this short term action, modifications were investigated to improve monitoring capabilities for the non-vital battery, as well as other changes to operating practices which would lessen the likelihood of this failure.

Figure 7–1
Initiating Events and Functional Sequences Contributions to Core Damage Frequency





As indicated above, the IPE has generated significant insights which have been or are being addressed by consideration of potential improvements. More importantly, CP&L is planning to maintain a living PRA and to make full use of this study in the future to address any new safety issues and assess plant changes. This commitment to continued use of this valuable tool as an additional means of safety assurance achieves CP&L's goals for continuing safety improvement as well as NRC's objectives established in Generic Letter 88–20.

The containment assessment provided insights into the response of the SHNPP containment to severe accident loading. The results of the containment assessment and a brief description of the insights are provided below. The information obtained from the containment assessment will provide input into CP&L's efforts related to containment performance and severe accident management.

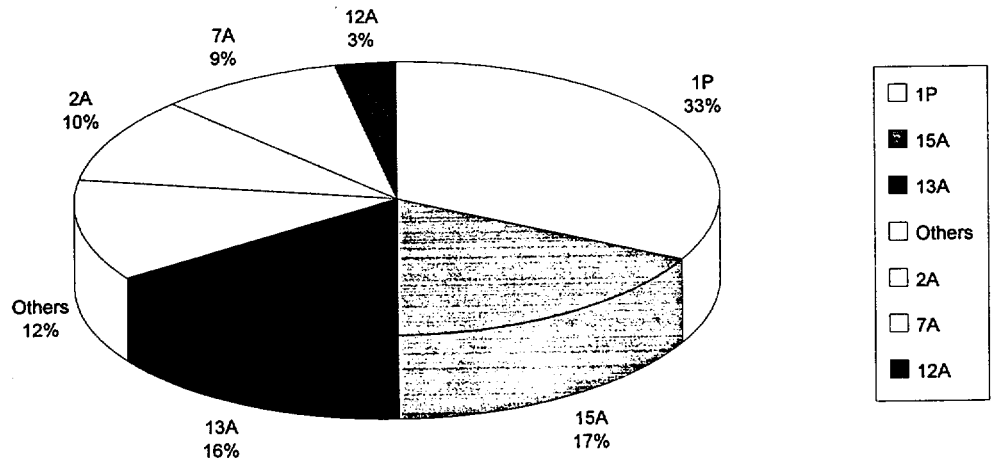
An important aspect of the containment analysis was the determination of the containment's real strength. Although the design pressure is 45 psia, the containment fragility assessment showed the best estimate median failure pressure to be 150 psig (95% confidence level value). This demonstrates the margins present in the containment design which can be utilized in addressing severe accidents.

The plant damage states (PDSs), containment failure modes, and release categories represent the results of the three steps of the analysis that provide insights concerning containment performance. Each of these areas is discussed below.

The PDSs represent specific categories of accident scenarios. Each PDS comprises a group of individual accident sequences that involve similar core damage sequence characteristics as well as containment safeguards systems response. The use of PDSs allows the containment analysis to be accomplished more efficiently, since all accidents within a PDS have similar enough accident progression and containment response to be treated as a single accident type. A review of the sequences which comprise the PDSs identifies the types of accidents that are most frequent in order to determine a representative sequence for the containment assessment. The SHNPP PDS results are illustrated in Figure 7–2.

Three plant damage states have frequencies in excess of 1×10^{-5}/year and contribute approximately 70% of the total PDS frequency. The highest contributing plant damage state is PDS 1P (frequency 2.5×10^{-5}/year) which contributes 34% of the total. The representative sequence for this PDS is a loss of offsite power and failure of both emergency diesel generators to start, and a failure to restore offsite power in time to avert the onset of core damage (station blackout). The containment sprays and fan coolers are unavailable due to the unavailability of AC power, but containment isolation is successful

Figure 7–2
Plant Damage States



following local valve operation. The next highest contributor is PDS 15A (frequency 1.2×10^{-5}/year) which contributes 17% of the total. The representative sequence for this PDS is a small LOCA (between 3 – 5 inches) with a failure of the high pressure safety injection system. Low pressure safety injection systems as well as all containment safeguards are available. The third highest is PDS 13A (frequency $1.2\text{x } 10^{-5}$/year) which also contributes 17% of the total. The representative sequence for this PDS is the same small LOCA with failure of the low pressure safety injection system during recirculation. Again, for this sequence all containment safeguards are available.

Comparing the three dominant PDSs highlights an important insight. For the case involving the loss of AC power, a major support system, both core cooling and containment safeguards systems are lost due to the common dependence on support systems. This insight is not unique to SHNPP. In fact, most PWR designs have similar dependencies. It is important, however, that this be considered in assessing accident management alternatives. It is also important to note that the containment isolation function is normally achieved for the dominant core damage sequences.

The assessment of accident progression, containment response, and radionuclide release also yielded insights concerning possible severe accident responses, with particular value relative to severe accident management.

Table 7–1 lists the outcomes of the next step of the analysis, the assessment of containment failure potential.

Table 7–1
Containment State Frequencies

Containment State	Frequency (/year)
Early Containment Failure	1.80×10^{-7}
Late Containment Failure	6.70×10^{-7}
Very Late Containment Failure	2.50×10^{-6}
Small Isolation Failure	1.90×10^{-7}
Large Isolation Failure	1.00×10^{-8}
Small Containment Bypass	2.53×10^{-6}
Large Containment Bypass	2.49×10^{-6}
Containment Failure after In-Vessel Recovery	2.27×10^{-6}

The outcomes can be grouped into the broad categories listed in the table which are consistent with the assessment of containment performance in other risk assessments. A review of the frequencies in the table indicate that the containment does not fail for a large portion of the overall core damage frequency. For the cases in which failure is expected, the most frequent failure modes are small containment isolation failures and late containment failures. These failure modes would result in small releases, i.e. well below PWR 2 category releases as defined in the *Reactor Safety Study* (Ref 7–3).

The conditional probability of an impaired containment is 0.15. This compares favorably with other large dry containments and illustrates the strength of the containment. The dominant class of containment failure involves a small containment bypass (55%) caused by a steam generator tube failure. Very late containment failures, dominated by overpressure and basemat failures, account for 23% of the impaired containment probability. Late failures due mainly to hydrogen burns account for about 6% of the containment failure outcomes. Early containment failure contributes about 2% and isolation failures contribute another 2%. The small bypass category is represented by two sequences which address SGTRs. SGTR sequences without makeup and a failure of the SRV on the faulted steam generator contribute about 50% to this category. Loss of decay heat removal sequences which result in a SGTR due to thermal failure of the steam generator tubes also contributes about 50%.

Ten sequences contribute to the very late containment failure category. These sequences are characterized by cases involving very long term containment failure due to basemat failure or late overpressurization caused by a failure to cool the debris ex-vessel. Station blackout sequences and low pressure sequences with the majority of the debris trapped in the cavity contribute to this category. Early containment failures are dominated by sequences which involve hydrogen burns which result in containment overpressure

failure. These sequences all have containment heat removal available and low containment steam concentrations which support early hydrogen burning.

The release category results are summarized in Table 7–2.

Table 7–2
Release Category Frequencies and Major Release Constituents

Release Category	Frequency (/yr)	Nobles	CsI	TeO_2
RC–4	3.83×10^{-6}	100%	0.22%	e[1]
RC–5	2.81×10^{-6}	100%	59.05%	0.01%
RC–6	2.47×10^{-6}	100%	0.001%	0.031%
RC–1B	5.82×10^{-7}	100%	2.72%	0.718
RC–4C	5.48×10^{-7}	100%	3.7%	e
RC–3	1.89×10^{-7}	100%	0.067%	e

1. e– less than 1.0×10^{-4}

The dominant release category (RC–4) involves containment releases through the steam generator. The faulted steam generator does not contain water and radionuclide scrubbing through the break does not occur. As a result, the volatile source terms are large. For cases with very late containment failure the release fractions are very small due to radionuclide deposition in the containment.

The dominant PDS contributor for RC–4 is PDS 2A and involves a loss of all decay heat removal. In this case, containment cooling is available but a steam generator tube rupture occurs late in the accident progression due to thermal stress. The failure of the tubes provides a path for the radionuclides to escape the RCS and bypass the containment. The release is not scrubbed due to the failure of secondary-side heat removal and isolation of the steam generator. The releases from this sequence are large due to the bypass leakage and approach the release fractions predicted in WASH–1400 for release category PWR–4.

The next highest release category is RC–5. Release category RC–5 represents the largest release and is a result of an ISLOCA or an induced SGTR with failure of a steam generator SRV. The failure of the SRV results in a large release without scrubbing which is similar to the ISLOCA. The ISLOCA is due to a failure in the RHR system. Both cases result in large release fractions due to the early failure of containment and the lack of radionuclide retention in the containment.

The third highest contributor (RC–6) is comprised of contributions from over 20 PDSs. The major contributor is PDS 1P which is dominated by station blackout sequences.

These sequences involve failure of the containment due to basemat failure. Due to the very long times involved, a large portion of the radionuclides have settled in the containment and the release is quite small. The other PDS which contributes is an S2 LOCA with a failure of recirculation. Basemat failure occurs associated with concrete ablation due to a failure to cool the debris ex-vessel.

7.2 Conclusions

The study has achieved the most important goal—provision of new insights regarding the safety of SHNPP. These insights have been developed from the process itself, during creation of an integrated model of the facility. In addition, insights have been derived from the results of the study, a detailed listing of potential accidents and their estimated frequencies. The results have been evaluated by a multidisciplinary team and a set of improvements and areas for further investigation have been identified. They demonstrate the value of PRA as a supplementary tool for safety assurance.

To determine the possibilities for improvement at SHNPP, a special team was formed with responsibility for examining the results and evaluating potential enhancements. This team included the broad perspective needed to evaluate effective and efficient improvements, including representatives from operations, training, technical support, corporate engineering, licensing, and PRA analysis. This team examined the results and suggested changes to plant procedures to allow for manual operation of breakers in transferring to offsite AC power, as well as areas for further investigation improving reliability of the non-vital DC system.

The study is also not being regarded as a static assessment. Evaluations are continuing to ensure that the study is a realistic portrayal of SHNPP.

With regard to the analysis of accident progression and containment response, it is expected that most of the insights derived from the analysis project will be used during future enhancements of the severe accident management capabilities at SHNPP.

While it can be directly concluded from examination of this submittal report that the NRC's goals relative to the IPE have been achieved, this step in the process does not actually represent a conclusion. The SHNPP PRA study, models, and results will be maintained as an additional tool in CP&L's inventory of methods to assure continued safe plant operations.

7.3 References for Section 7

7–1. U.S. Nuclear Regulatory Commission, *Individual Plant Examination for Severe Accident Vulnerabilities*, 10CFR50.54(f), Generic Letter 88–20, November, 23, 1988.

7–2. U.S. Nuclear Regulatory Commission, *Individual Plant Examination Submittal Guidance*, NUREG–1335, August 1989.

7–3. U.S. Nuclear Regulatory Commission, *Reactor Safety Study, An assessment of Accident Risk in U.S. Commercial Nuclear Power Plants,* WASH–1400, 1975.