ATTACHMENT 1

HOPE CREEK GENERATING STATION
INDIVIDUAL PLANT EXAMINATION
LEVEL II ANALYSIS
PRELIMINARY SUMMARY

OCTOBER, 1992

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(Partial for Preliminary Summary)

INDIVIDUAL PLANT EXAMINATION

HOPE CREEK GENERATING STATION

Level II Analysis

Preliminary Summary

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4.0 Back-End Analysis

The back-end (or Level 2) portion of the IPE undertakes an assessment of the progression of accident sequences beyond the point of core damage. Evaluation of the likely modes of containment failure is the principal objective. This evaluation includes both the probability of each of the potential modes of containment failure and the characterization of the radionuclide releases that may accompany each mode. The results are reported in terms of the frequency (i.e., expected number of occurrences per year) of specific release categories. A release category is characterized by the fraction of the initial core inventory of fission products that is released from the containment and the timing of that release.

The approach applied for this analysis is based on that developed for the Nuclear Safety Analysis Center (NSAC) of the Electric Power Research Institute (EPRI) [1]. Under this methodology accident progression, the containment loads resulting from that progression, and the response of containments to those loads are predicted probabilistically. Logic models, similar to those used for the front-end (i.e., Level 1) analysis are used to support the evaluation of results. Quantification of the logic models is based on mechanistic analysis of the plant features and the phenomena involved. The Modular Accident Analysis Program (MAAP) [2] was used as the principal tool for developing an integrated perspective on plant response to postulated severe accidents.

4.1 Plant Description

Hope Creek Generating Station (HCGS) is a General Electric Company BWR/4-251 (251" diameter vessel) in a Mark I containment. HCGS is very similar to Peach Bottom, Unit 2, which was used as the reference plant for this analysis. The principal difference is the use of a secondary containment building rather than a reactor building to house support systems. Figure 4.1-1 provides an elevation plan view of the plant. The Mark I primary containment is composed of two connecting structures. The first structure, the drywell, is an inverted light-bulb shape steel pressure vessel containing the reactor vessel, the reactor coolant recirculation system, and other primary system piping. The second structure, the wetwell or torus, is a toroidal shaped steel pressure vessel placed below and encircling the drywell. The drywell is connected to the wetwell via eight 6'-2" ID vent pipes [1], that feed into a header inside the wetwell and then to downcomers which extend down into the 118,800 ft^3 [2] volume of water that forms the suppression pool. Dual isolation valves in series are provided on the various process lines (e.g., main steam lines) penetrating the containment to ensure containment of radioactive materials released from the primary system in the event of an accident.

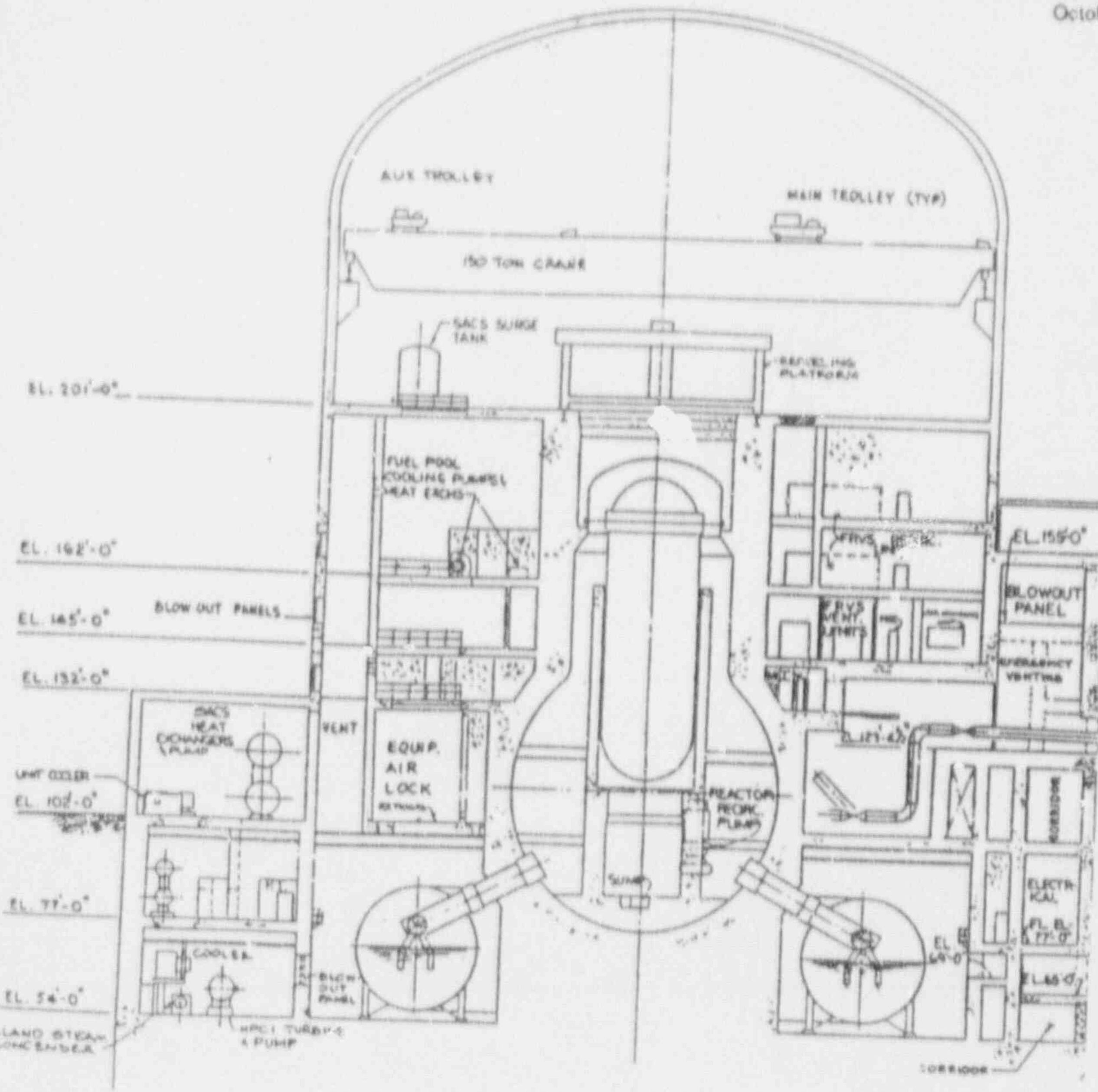

Figure 4.1-1
Containment Elevation Plan - Looking North
(Taken From Reference 10)

4.1.1 Engineered Safeguards

This section provides a summary discussion of the systems that were considered as having the potential to affect or mitigate the progression of postulated severe accidents. All of these systems require power to operate. With the exception of reactor pressure vessel (RPV) depressurization, AC power, either from the diesels or from offsite, is required for their operation. It is recognized that these systems would be required to operate under conditions that are more severe than their design basis. This has been accounted for both in the structure of the back-end analysis model and the evaluation of failure probabilities.

Residual Heat Removal

Low Pressure Coolant Injection (LPCI), Suppression Pool Cooling (SPC) and Containment Spray System (CSS) are three modes of the Residual Heat Removal (RHR) system that are relevant to postulated severe accident progression. SPC and CSS can be used to remove heat from the containment. LPCI can be used to inject coolant into the primary system but does not remove heat. In the current analysis, the Low Pressure Core Spray (LPCS) is modeled identically to LPCI. The term Low Pressure Emergency Core Cooling System (ECCS) is applied to either or both. The SPC system takes water from the suppression pool, passes it through heat exchangers, and discharges it back into suppression pool. The CSS system takes water from the suppression pool, passes it through heat exchangers, and discharges the water through spray headers in the drywell and torus. Energy is removed from the primary containment to lower temperature and pressure.

The RHR system consists of two independent system loops with two independent trains in each loop. Each train contains one pump. There is one heat exchanger for each loop. The RHR system is capable of performing its intended function if either one of the two loops are operating. Valve positions in discharge lines are changed to operate the different modes. Both SPC and CSS are emergency AC powered and are unavailable in station blackout scenarios unless offsite power is recovered. Interlocks are provided to prevent LPCI flow from being diverted to the containment spray mode within ten minutes of a low water level (i.e., RPV Level 1) signal. This lockout is controlled by a timer. A keylock switch in the control room permits the overriding of this interlock to reduce containment pressure if required.

The containment spray subsystems have two loops. The first, consisting of one heat exchanger, two main system pumps in parallel, and associated piping, is located in the northwest quadrant of the reactor building at the 54′ elevation. The two pumps are located in separate compartments. The second loop consists of a heat exchanger, two pumps, and piping and is located in the southwest quadrant of the reactor building at the 54′ elevation to minimize the possibility of a single physical event causing the loss of the entire system. The two pumps are also located in separate compartments. Failure of the RHR system due to severe environments in the secondary containment has been considered.

Reactor Vessel Depressurization

The function of the pressure relief system is to prevent overpressurization of the RPV and provide the means for depressurization to allow water make-up from low pressure systems. The automatic depressurization system (ADS) portion of the pressure relief system operates in conjunction with the ECCS to reflood the core following transients and small breaks in the primary system. ADS is designed to reduce the RPV pressure by discharging steam to the suppression pool.

The RPV pressure relief system consists of fourteen safety/relief valves (S/RVs), five of which are part of the ADS. All fourteen valves are located upstream of the main steam isolation valves (MSIVs) on the main steam lines and within the drywell. S/RVs will open when RPV pressure exceeds the spring set point. They are also pilot operated allowing them to be operated by pneumatic pressure. When opened at the spring setpoint, steam pressure is relieved until the RPV pressure has dropped 50 to 100 psig below the S/RV opening setpoint. Use of pneumatic pressure allows operation of the relief valves for automatic depressurization, low-low set (LLS), and manual pressure relief operations. Pneumatic pressure is provided by the instrument nitrogen system. Each S/RV is supplied by a nitrogen accumulator within the drywell.

The fourteen S/RVs all discharge through piping routed to the torus and then through "T"-quenchers located below the water level, forcing the steam to condense in the torus. Following a S/RV overpressure, ADS, or manual actuation, steam trapped in the relief valve piping after the relief valve has closed will condense, creating a vacuum within the discharge piping. This vacuum would draw water up into the piping from the suppression pool, and if the valves were to operate again, a water hammer would occur, possibly causing damage to the relief valves, and their piping, and to the torus. To protect against this, two 6-inch vacuum breakers are installed in each of the fourteen relief valve discharge lines [2].

Automatic initiation of the ADS system requires that a low pressure injection pump must be operating (one LPCI or two LPCS). Thus in the event of station blackout condition, ADS will not automatically initiate since no low pressure injection pumps will be available. The operator must then act to manually depressurize the primary system. Both the ADS system and manual S/RV actuation requires DC power, therefore, the reactor pressure vessel (RPV) can not be or remain depressurized in sequences with initial DC failure or battery depletion.

Venting

In compliance with Generic Letter 89-16, PSE&G is installing a 12"-diameter hard pipe vent line. This line connects the torus air space with the environment. The line is equipped with dual isolation valves and a 35 psig rupture disk. It will be possible to perform containment venting can be initiated from the control room if AC power is available. Manual local activation is also possible. Venting will be directed by procedure once the containment reaches 60 psig (HCGS has 56 psig containment design pressure). Opening the isolation valves at this pressure fails the rupture disk and results in containment depressurization to the atmosphere. Procedures will direct throttling (or periodic reclosure) of the isolation valve(s) to maintain containment pressure at approximately 60 psig. When the isolation valves are fully open, the effective flow area of this line is equivalent to a 6"-diameter opening. The HCGS IPE was performed assuming that this system is in place.

Alternate Injection

The back-end analysis considers the use of alternate means to supply water make-up to the vessel. The condensate and/or condensate booster pumps can be used, when the RPV is at low pressure and AC power is available, to provide water make-up from the condenser hot well and condensate storage tank (CST). A cross-tie between RHR and the service water system can provide water makeup from the ultimate heat sink. Again, low RPV pressure and AC power are required. Failure of these systems due to severe environments in the secondary containment is considered.

The emergency operating procedures (EOPs) also direct that the main fire pump system (MFPS) be used for coolant make-up if other systems are inoperable. The MFPS has dedicated diesels and is independent of normal and emergency AC power. However, for the MFPS to provide RPV injection. DC power must be available in order to maintain RPV pressure below the MFPS pump shut-off head. Thus, for instance, the MFPS can not be used in long-term station blackout sequences in which the station batteries are drained prior to core damage.

4.1.2 Drywell

The drywell for the HCGS containment vessel is made of SA-516, Grade 70 steel. The drywell head, is hemi-ellipsoidal with a major diameter of 33'-2" [8] and a head thickness of 1.5" [10]. The drywell head is connected to a cylindrical shell, which has a radius of 16'-7" [10] and a shell thickness of 1.5" [10]. A cone shaped structure connects the cylindrical upper shell to a lower cylindrical shell. The cone thickness is 1.5" [10]. The lower cylindrical shell is divided into two portions. The radius and thickness values for the top portion of the cylindrical shell are 20'-3" [10] and 1.5" [10]. The lower portion of the cylindrical shell a 20'-3" radius [10] and a thickness of 0.9375" [10]. A transition knuckle connects the cylindrical shell and spherical bottom of the drywell. The knuckle has a radius of 8'-4 1/8" [10] and a thickness of 2.875" [10]. The spherical bottom shell has a radius of 34'[2] and a thickness of 1.5" [6]. The drywell has an overall height of 114'-9" [8], and free volume of approximately 165,620 cubic feet (including the suppression pool vent system) [9].

The HCGS drywell has the following design parameters:

a. 56.0 psig internal design pressure, 58.0 psig maximum calculated accident design pressure (internal), and 62.0 psig maximum internal design pressure allowed by ASME code (110 percent of design pressure).

b. 3.0 psig maximum external design pressure.

c. 340°F maximum temperature.

The drywell is closed at the top by a removable, double gasket, bolted head to facilitate reactor refueling. It is also enclosed in reinforced concrete for shielding purposes and to provide additional resistance to deformation and buckling in areas where the concrete backs up the steel shell. Above the foundation transition zone, the drywell is separated from the reinforced concrete by an air gap of approximately 1 3/4" to 2 1/2" in thickness [8] to accommodate thermal expansion of the drywell shell.

Four drain sumps are located in the drywell floor. Two of these sumps are located in the RPV pedestal floor. The other two sumps are smaller pump sumps that are located ex-pedestal. Access to the drywell is provided by one 8'-10 1/2" diameter personnel access lock [10]. One 2-foot diameter personnel access hatch is also provided on the drywell head [7]. A 2-foot ID construction access hatch is located at an elevation of 77'-10" [7]. Both 2-foot access hatches are bolted in place. Two equipment access hatches having 12-foot diameters are also provided [7]. One of these hatches is connected to the personnel access lock. The other hatch is bolted in place. The CRD removal hatch is located at an elevation of 103'-6" and has an ID of 3'-0" [7].

4.1.3 Torus

The wetwell is a steel pressure vessel in the shape of a torus below and encircling the drywell. The torus centerline diameter is approximately 112'-8" [8]. Its cross-sectional diameter is 30'-8" [8], and its thickness is 1.0" [5]. The pressure suppression chamber contains approximately 118,800 cubic feet of water [8] and has a net air space above the water pool of approximately 142,638 cubic feet. The wetwell is not directly enclosed by concrete, but is located in a large room below ground level called the torus room.

Vent pipes connect the drywell and the pressure suppression chamber. A total of eight circular vent pipes, each having a diameter of 6' - 2" inches are anchored at the drywell, radiate outward at 45 degree intervals, and penetrate the torus shell at alternating segments midway between ring girders. The drywell vents are connected to a 4' - 3" inch diameter vent header in the form of a torus (ring header) which is contained within the airspace of the suppression chamber. The vent header has the same temperature and pressure design requirements as the vent pipes. Projecting downward from the header are 80 downcomer pipes, each 24 inches in diameter. The vent header system is supported by 16 pairs of 6 inch pipe columns each pair pinned to the bottom of a ring girder. The downcomers terminate a minimum of four feet below the surface of the pool. In addition to its function as a path for energy and mass transfer from the drywell to the suppression chamber, the vent pipes also allow the passage of the S/RV tailpipes to the suppression pool.

The vacuum relief system is composed of two separate systems: the wetwell-to-drywell vacuum breakers and the secondary containment-to-wetwell vacuum breakers. The torus to drywell vacuum breakers relieve pressure from the wetwell-to-drywell if there is a pressure differential greater than 0.2 psid. There are eight pairs of torus to drywell vacuum breakers [10]. Each one is 20-inch swing-check type valve with an attached air operator for testing. The eight pairs of valves are installed in lines connecting the wetwell airspace to the vent pipes.

The secondary containment-to-torus vacuum breaker system relieves pressure from the reactor building to the torus if external (secondary containment) pressure is 0.25 psid above torus pressure. Operation of the vacuum breakers will maintain a pressure differential of less than 3 psid, the external design pressure.

4.1.4 Reactor Vessel Pedestal

The primary function of the reactor pedestal is to provide the necessary support required to hold the reactor pressure vessel (RPV) in its lateral and vertical positions. It also acts as a radiation and missile barrier. The pedestal houses the control rod drives and contains drain sumps for collecting water that leaks from the recirculation pumps and other equipment.

The reactor pedestal is essentially a right circular cylinder with a diameter of 20.25 feet having a reinforced concrete wall that supports both the reactor pressure vessel and the sacrificial shield wall. The pedestal wall is 4' - 10" thick at the bottom and 5' - 9" thick at the top and is imbedded into the concrete forming the drywell floor. The inside of the pedestal wall has a stainless steel liner. There is no liner outside at the floor level of the pedestal wall. There is a 3-foot by 7-foot personnel access door flush with the drywell floor.

There are two drain sumps inside the pedestal, Clean Radwaste (CRW) and the Dirty Radwaste (DRW) which are used to collect water from known and unidentified sources respectively. Each sump is connected to a separate sump on the outside of the pedestal wall by a pipe approximately 4 feet under the concrete. The two inside sumps have a cross sectional area of 36 square feet each, and are 2′ 11" deep. Each sump has a stainless steel grating cover 3/16 inch thick. Both sumps are equipped with pumps and flow rate measuring devices. The pumps pump water to the radwaste facilities. The bottom of the outside sumps is very close to the drywell shell. The closest distance from the bottom of these sumps to the drywell shell has been calculated to be about 7.4 inches.

4.1.5 Secondary Containment

The secondary containment at HCGS completely encloses the reactor and its pressure suppression primary containment system (Figure 4.1-1). It houses the refueling and reactor servicing equipment, new and spent fuel storage facilities and other reactor auxiliary and service equipment. Also housed within the secondary containment are the emergency core cooling systems, reactor water clean-up demineralizer system, standby liquid control system, control rod drive system, instrumentation for the reactor protection system, and electrical equipment components.

The secondary containment is a Seismic Category I structure. It is constructed of reinforced concrete both above and below the refueling floor. The reactor well dryer-separator pool and fuel storage pool are lined with stainless steel [11].

The secondary containment at HCGS has several floors which are isolated from each other except for a series of large open hatches that run from Elevation 132′ up to the refueling floor on the west side. There are two enclosed stairways, one on the west side and one in the southeast corner of the reactor building. There are several pipe chases between floors. It is expected that steam released into the reactor building will mostly go up the open hatch to the refueling floor and then out the blowout panels (rated at 0.5 psig) to the environment. There is a path from the secondary containment to the turbine building via the steam tunnel. There are several blowout panels at the end of the steam tunnel which opens to the turbine building. The hall connecting the steam tunnel to the turbine building is also equipped with blowout panels. These are expected to reduce the potential for severe environments in the turbine building should overpressurization of the secondary containment occur.

4.1.6 Plant Data

Table 4.1-1 presents a summary of basic RPV and containment data needed for the HCGS IPE. The HCGS values are compared to those for Peach Bottom, Unit 2 [12].

Table 4.1.1. Comparison of Basic RPV and Containment Features of HCGS and Peach Bottom, Unit 2

Parameter Description		
PLANT NAME	HCGS	PEACH BOTTOM
TYPE OF REACTOR	BWR/4 MARK I	BWR/4
TYPE OF CONTAINMENT	4/11/86	MARK I
DATE OF COMMERCIAL OPERATION		
Reactor Core		
Thermal Power (MW_t)	3,293	3,293
Number of Fuel Assemblies	764	764
Number of Control Rods	185	185
Core Weight, Total (lbm)	---	511,502
Uranium Dioxide (lbm)	365,236	351,440
Zircaloy (lbm)	49,869	144,382
Miscellaneous	---	15,600
Reactor Vessel		
Inside Diameter (in)	251	251
Inside Height (ft)	72.54	72.92
Operating Pressure (psig)	1,020	1,020
Primary system oper. temp (°F)	555	555
RPV liquid mass(klbm)	≈610	---
RPV steam mass (lbm)	24,500	---
Number Safety Valves	0	2
Lowest Safety Valve SP (psig)	1,130	1,230
Safety Valve Capacity (klb/hr)		925
Number of Safety/Relief Valves	14	11
Lowest Rel. Valve SP (psig)	1,017	1,105
Relief Valves Cap. (klb/hr)	---	819

Table 4.1.1. Comparison of Basic RPV and Containment Features of HCGS and Peach Bottom, Unit 2 (Continued)

Parameter Description		
PLANT NAME	HCGS	PEACH BOTTOM
TYPE OF REACTOR	BWR/4 MARK I	BWR/4
TYPE OF CONTAINMENT	4/11/86	MARK I
DATE OF COMMERCIAL OPERATION		
	PRIMARY CONTAINMENT STRUCTURE	
Reactor Coolant Recirculation		
Number of Loops	2	2
Number of Pumps	2	2
Inlet Pressure (psig)	1,148	1,148
Outlet Pressure (psig)	2,326	2,326
Number of Jet Pumps	20	20
Flow Rate/Pump (gpm)	45,200	45,200
RHR System		
Number of Loops	2	2
Number of Pumps	4	4
Flow Rate/Pump (gpm @ psig)	10,000 @ 175	10,000 @ ---
Number of Heat Exchanger (Hx)	2	4
Max. Cap. of Hx (BTU/hr) (per 2 HX set)	130,000,000 (for containment cooling, MTD = 88.7°F)	70,000,000

4.2 Plant Models and Methods for Physical Processes

A discussion of the use of MAAP and NRC-contractor evaluations of severe accident phenomena will be included in the final submittal.

4.3 Plant Damage States

Plant damage states (PDS) are bins that group together (or "bin") sequences that present similar initial and boundary conditions to the Containment Event Tree (CET) analysis. To establish the appropriate plant damage states, a two-step process, similar to that used for NUREG-1150 [2] was applied. The first step involved grouping the Level 1 sequences based on system or component failures that had the potential to impact the CET results. A larger number of groups than could reasonably be accommodated within the Level 2 framework resulted. Combining these initial sequence groups, based on implied timing and combinations of failures that actually altered the CET results, constituted the second step. Differences within the groups that had negligible impact on the CET results were essentially screened out. Once the screening criteria had been applied, a total of eight plant damage states were defined.

4.3.1 Summary of Front End Analysis

The front end analysis involved the construction of systemic event trees to delineate the potential accident sequences for each initiating event, including special initiators. Fault tree models for both the front line (e.g., low-pressure core spray, reactor protection) and support (e.g., electric power, service water) systems were developed. A fault tree linking approach was used to evaluate the frequency of sequences leading to core damage. Approximately eighty systemic sequences were evaluated. The frequencies of these sequences ranged from 2×10^{-5}/yr down to 1.4×10^{-10}/yr.

As is typical for BWRs, the core damage results are dominated by transient-initiated sequences involving failures of a support system (i.e., AC power or service water). Sequences initiated by primary system breaks also contribute. The use of an automatic system for injection of standby liquid control at HCGS makes the contribution from anticipated transients without scram (ATWS) sequences negligible. Eight PDS were required to reflect core damage sequences exceeding the screening criteria provided in Section 2.1.6 of NUREG-1335. A reevaluation of sequences initiated by a anticipated transients without scram (ATWS) is currently being made. The results may lead to the inclusion of ATWS-initiated PDS in the final Level 2 analysis.

4.3.2 Plant Damage State Binning Criteria

The binning criteria were developed based on the criteria applied in the NUREG-1150 analysis of Peach Bottom Unit 2 [2]. These criteria are reflected in the logic models that constitute the EPRI Generic Methodology [7] and in the logic models that support quantification of the HCGS CET (described in Section 4.5). The initial screening criteria include all system and component

Table 4.3-1. PDS Binning Criteria Description

Name	Description	Importance
Initiator	Transient, LOCA, ATWS	Determines, in part, the initial pressure in containment and the leakage pathway from the RPV to the containment.
Time to core damage	Estimated based on cause of failure for vessel make-up systems: Immediate (30-60 min); Long-term (~4 hrs); and delayed failure (10-15 hrs).	Release timing, containment pressure.
RPV pressure prior to core damage	High/Low	Vessel pressure during core damage (may effect H_2 generation).
S/RV Operability	Yes/No	Vessel pressure at failure.
Stuck-open S/RV (SORV)	Yes/No	Vessel pressure at failure.
AC Power	Yes/No	Ability to recover available water make-up systems.
DC Power	Yes/No	S/RV operability
HPCI/RCIC	Failed/Working	Accident timing if no depressurization.
CRD Hydraulic System	Available/Failed	Accident recovery potential cooling of core debris.
LPCI/LPCS	Available/Failed	Accident recovery potential cooling of core debris.
SW/Condensate	Available/Failed	Alternate source of make-up for accident recovery.
SPC	Working/Available/Failed	Containment pressure.
Venting	Done/Available/Failed	Containment pressure/integrity.
SGTS	Working/Available/Failed	Mitigation of releases outside secondary containment.

Table 4.3-2. Example Application of Binning Criteria

Sequence	Initiator	Time to Core Damage	RPV Pressure	S/RV Operability	SORV	AC Power	DC Power	HPCI/ RCIC	CRD	LPCS/ LPCI	SW/ Condensate	SPC	Venting	SGTS
TCWW1	Trans.	10-15 hrs.	High	Y	N	Y	Y	F	F	F	F	F	F	W
TIAPQWW1	Trans.	10-15 hrs.	High	Y	Y	Y	Y	F	F	F	F	F	F	W
TSAQWUV	Trans.	10-15 hrs.	Low	Y	N	Y	Y	F	F	F	F	F	D	W
AWW1	LOCA	10-15 hrs.	Low	Y	N	Y	Y	F	F	F	F	F	D	W
T123IIR2U2U1UV	Trans.	30-60 min.	Low	Y	N	N	Y	F	F	A	A	A	A	A

failures that could affect accident progression and containment response. These are shown in Table 4.3-1. In the first stage of the binning process, a matrix was developed that indicated the status of each criteria based on the sequence definition. Table 4.3-2 illustrates the use of this matrix for a few sequences selected at random. The approach used in NUREG-1150 examined sequences of the cut set level. This level of detail was judged to be inconsistent with the requirements of the IPE. Where the potential existed for different cut sets in the same sequence to indicate a different status of the same system [1], the most likely status was assumed.

Based on the results obtained from completing the matrix illustrated by Table 4.3-2, the number of unique sequence groups was larger than could be practically handled by the Level 2 process. To reduce this number of groups, the back-end analysts ranked the binning criteria based on their importance with respect to accident progression and containment response. For instance, failures affecting the timing between the initiating event and core damage were ranked highly. Timing is an important element in the source term in that it impacts the potential consequence of a radionuclide release. In addition, conditional rankings (i.e., rankings that depend on higher ranked failures) were applied. The operability of suppression pool cooling (SPC) has a significant impact for long term sequences in reducing the containment pressure. For short term sequences the decay heat load to the suppression pool is small so SPC does not impact containment pressure and hence the probability of subsequent failure[2].

Core melt sequences were then sorted according to the ranking of the binning criteria for which failures were indicated. Final PDS groups were identified based on unique combinations of failures in the high ranking criteria. Differences between sequences within a PDS that were considered too significant to neglect, or treat conservatively, were included by incorporating split fractions in the CET logic models.

For example, the PDS representing transient initiators followed by a loss of containment cooling includes sequences with both successful and failed venting of containment. The evaluation for this PDS in the CET includes probability for containment venting prior to core damage equal to the ratio of the frequency of sequences where venting failed to the total frequency of the PDS[3].

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[1]For instance, a station blackout sequence may contain cut sets where the diesel generators failed to start resulting in failure of low pressure make-up systems. If offsite power were restored low pressure make-up could be initiated. In such cases the system is classified as "available". Other cut sets in this sequence may involve service water failure following the loss of offsite power. The loss of cooling fails both the diesels and the low pressure pumps. If power were restored given these failures low pressure make-up remains inoperable. In those cases a classification of "failed" would be applied.

[2]Unless the core debris is subsequently cooled. The potential for recovering make-up was thus also considered in ranking SPC failure.

[3]Frequency of a PDS is equal to the sum of the frequencies of the sequences in that bin.

4.3.3 Plant Damage State Descriptions

The eight PDS for the HCGS back-end analysis are shown in Table 4.3-3. A short summary of their characteristics is provided. An approximate range for the frequency is indicated. A fractional contribution to the total frequency of accidents considered in the back-end analysis will be provided in the final submittal.

4.4 Containment Failure Characterization

This section describes the assessment of containment performance in response to postulated severe accident loads. The full range of potential loads, including static pressurization, elevated temperatures, and dynamic loads resulting from energetic phenomena (e.g., fuel-coolant interaction) have been considered. The objective of this assessment was the characterization of the potential containment failure modes. Determination of the timing of failure relative to the predicted progression of accidents was the principal focus. Failure location, and the corresponding pathway for fission product release to the secondary containment or environment was also a primary consideration. It is recognized that the rate at which fission products released during an accident escape from the containment can have a significant effect on the magnitude of the overall releases. An assessment of the size of induced failures was also undertaken.

4.4.1 Structural Analysis

A detailed evaluation of the fragility of the HCGS containment has been undertaken [5]. A complete structural analysis of the primary containment was performed. Only quasi-static pressurization was analyzed. The analysis included consideration of all potential failure locations including:

(1) Drywell shell,
(2) Drywell head flange,
(3) Vent lines from the drywell to the suppression pool,
(4) Torus shell (wetwell),
(5) Drywell equipment hatch,
(6) Personnel airlock,
(7) Control rod drive (CRD) removal hatch, and
(8) Piping penetrations.
(9) Electrical penetrations

Table 4.3-3. Plant Damage States for Back-End Analysis

Name	Symbol	Description	Frequency	Fractional Contribution
PDS-1	TW	*Transient Accompanied by Containment Heat Removal Failure.* Transient (an MSIV closure transient has been selected as being representative) accompanied by (1) failure to remove decay heat using residual heat removal (RHR) suppression pool cooling (SPC), containment spray (CSS), and the shutdown cooling system (SCS) and (2) failure to remove decay heat by venting the containment. This PDS includes both sequences in which containment venting fails (W1) and in which containment venting succeeds. If venting fails, core damage and vessel failure will occur with both the vessel and containment at high pressure. If venting is successful, core damage and vessel failure will occur with the vessel at low pressure and containment at a lower pressure.	$5x10^{-5}$ to $3x10^{-6}$/yr	
PDS-2	TQUV	*Transient with High and Low Pressure Injection Failure.* Transient (an MSIV closure transient has been selected as being representative) accompanied by feedwater isolation and high and low pressure coolant injection failure. In this PDS, core degradation and vessel failure occur with low primary system pressure and low containment temperature and pressure.	$5x10^{-6}$ to $1x10^{-6}$/yr	
PDS-3	ST-SBO	*Short-Term Station Blackout.* Loss-of-offsite power (LOOP) transient accompanied by simultaneous failure of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems. Low pressure injection systems are available if offsite power is restored. All other systems function normally under DC power.	$4x10^{-6}$ to $9x10^{-7}$/yr	
PDS-4	AW	*Large LOCA with Containment Heat Removal and Containment Venting Failure.* Large LOCA followed by failure to remove residual heat with any mode of RHR (SPC, SPS, or SCS). Suppression pool temperature increase is assumed to result in failure of all injection systems.	$3x10^{-6}$ to $9x10^{-7}$/yr	

Table 4.3-3. Plant Damage States for Back-End Analysis (Continued)

Name	Symbol	Description	Frequency	Fractional Contribution
PDS-5	TQUX	*Transient with Depressurization and High Pressure Injection Failure.* Transient (an MSIV closure transient has been selected as being representative) accompanied by feedwater isolation, depressurization failure and failure of HPCI and RCIC. Operator failure to depressurize makes low pressure systems unable to supply injection. However, these systems (LPCI, LPCS) are available if vessel pressure is reduced subsequent to core damage.	$6x10^{-7}$ to $1x10^{[illegible]}$/yr	
PDS-6	SQUV	*Small/Medium LOCA and Low Pressure Injection Failure.* Either small-or medium-sized primary system break occurs. Vessel is depressurized by either the break flow or operator action. Both high and low pressure injection systems fail.	$6x10^{-7}$ to $1x10^{[illegible]}$/yr	
PDS-7	LT-SBO	*Small/Medium LOCA and High Pressure Injection Failure.* Either small- or medium-sized primary system break. Operator fails to depressurize. Available high pressure system (if any) is assumed unable to maintain core water level. It is conservatively assumed that vessel pressure stays above the LPCI/LPCS shut-off head prior to core damage. These systems are available to inject once vessel pressure is reduced.	$6x10^{-7}$ to $1x10^{[illegible]}$/yr	
PDS-8	TB	*Long-Term Station Blackout.* LOOP followed by diesel/generator failure. HPCI and/or RCIC provide makeup until battery depletion. Low pressure systems available if power restored and vessel depressurized.	$6x10^{-7}$ to $1x10^{[illegible]}$/yr	

The analysis concluded that the drywell head flange was the most likely failure location. At low temperature (200°F) a value of 78 psig for which there was high confidence of a low probability of failure (HCLPF) was determined. The equipment hatch and the drywell shell were found to have the next lowest HCLPF values, at 134 and 135 psig respectively. At high temperature (600°F), the CRD removal hatch HCLPF value was assessed to be 15 psig. The drywell head access hatch and flange HCLPF values at these temperatures were 45 and 60 psig respectively. Drywell failure is thus expected to dominate in postulated severe accidents at HCGS. For early failures, the drywell head seal predominates as the leakage location. For late failures the CRD and drywell head access hatches are additional considerations with respect to leakage location.

4.4.2 Bypass and Isolation Failure Potential

A thorough investigation of potential bypass and isolation failures was conducted. As part of the Level 1 analysis, PSE&G undertook a complete, separate, and independent assessment of the potential for primary system failure through an interfacing system. The results of this study lead to the conclusion that the frequency of such events was negligible with respect to the IPE. As part of the back-end analysis, a thorough revie of all containment penetrations was conducted. Penetrations were identified based on the HCGS Final Safety Analysis Report (FSAR). Lines connecting to the primary system received special attention since the potential for bypass existed. It was concluded that the probability of either containment bypass or containment isolation failure for HCGS was sufficiently small that sequences involving either one had frequencies below the NUREG-1335 screening criteria. This conclusion is consistent with that in the NUREG-1150 assessment of Peach Bottom.

4.4.3 Containment Response to Severe Accident Loads

Temperatures and pressures beyond the containment design basis may accompany postulated severe accidents. The MAAP code was the principal tool used to predict temperature histories within the containment. These predictions were used in conjunction with the capacity assessments to predict the timing and location of containment failure. Available information from NRC contractor studies of the Peach Bottom and Browns Ferry plants was also considered as part of the assessment of uncertainty in the results obtained based on the MAAP predictions. Loads due to high pressure melt ejection (which are not modeled in the BWR version of MAAP) were also assessed on the basis of NRC contractor studies. The basis for quantifying the probability of containment failure is discussed more completely in Section 4.6. Included in this section is the treatment of the probability of containment failure due to direct contact between the core debris and the drywell shell.

An assessment of the potential for containment leakage through the seals on electrical penetrations was made. The principal conclusion from that assessment was that the electrical penetrations had significantly higher leak resistance at elevated temperatures than e... the CRD removal hatch or the drywell head access hatch. Electrical penetration failure was not included as a distinct failure mode.

Response to the dynamic loading that has been postulated as a possible consequence of fuel-coolant interaction (FCI) on the drywell floor was not explicitly examined. Such an assessment was considered to be outside the scope of the IPE. Containment failure probabilities for FCI used in the analysis of Peach Bottom for NUREG-1150 were adopted for this study. Since the core and primary containment designs for HCGS and Peach Bottom are essentially similar, this approach was judged to be adequate.

4.5 Hope Creek Containment Event Tree

The containment performance logic model for the Hope Creek Generating Station (HCGS) has been developed in the form of linked event trees. In this context, the term "linked" means that there are common events among the event trees that have been developed to reflect each aspect of containment response. The dependencies between the event tree that represent the different phenomena considered are fully treated. This is consistent with the EPRI Generic Methodology [1]. However, the fault tree models for the CET top events have been replaced with event tree models. The CETs consider all the relevant events and phenomena included in the EPRI generic Methodology. The events and phenomena included in the EPRI methodology were identified based on an in-depth review of the analysis of Peach Bottom Unit 2 performed in support of the first draft on NUREG-1150 [3]. This includes the phenomena listed in Table 2.2 of NUREG-1335 pertinent to BWRs with Mark I containments. Event tree format has been used to display the logic. There are two motivations for this approach: (1) event trees have historically been applied to back-end analysis and are more amenable to review and (2) the event trees are more flexible and powerful tools for modeling Level 2 accident sequences in that success paths and dependent probability assignments are more readily handled. The discussion that follows provides a brief summary of the basis for the CETs employed.

4.5.1 Methodology

The HCGS CETs, are altered slightly from that appearing in the EPRI Generic Methodology. Thirteen subtrees, one supporting the quantification of each CET top event have been developed. Some of the subtrees have sub-subtrees for specific phenomena. These provide the basis for the evaluation of specific top events in the subtrees. The linked subtrees (and sub-subtrees) are solved simultaneously using the EVNTRE [5] software. Graphical display of the results has been provided by re-formatting the output of EVNTRE and loading the results into event tree editing software. The event tree figures provided in this document were produced in this manner.

Branch points in the CET are each evaluated based on the subtree model results. The probability assigned to a branch is simply the sum of the probabilities of the subtree end states that correspond to the indicated outcome. A few top events in the subtrees are evaluated in the same manner using sub-subtrees. Branch points in the sub-subtrees, and those in the subtrees not evaluated with sub-subtrees, are evaluated directly by the analyst. These are referred to as *basic events* in order to emphasize the similarity between this approach and the fault tree linking approach used for the Level 1. Evaluation of the basic events is discussed in Section 4.6.

The same CET is applied for each PDS. Based on the subtree and sub-subtree logic structure the evaluation of the split fractions varies. Basic events that reflect the PDS characteristics are incorporated directly into the subtrees and sub-subtrees. When these are evaluated as either zero, unity, or a split fraction that has been determined based on the Level 1 results, the subtree structure is altered. (Assignment of zero or unity to a basic event eliminates the branch point and indicates success or failure, respectively, for the corresponding top event.) This alteration of the subtree structure changes the end state probabilities. The corresponding CET top event probabilities are thus adjusted based on the PDS characteristics. This approach is essentially similar to that used in NUREG-1150 since there is in fact only one CET and the quantification is varied based on the PDS definition. Table 4.5-1 indicates the basic events that reflect the PDS definitions.

4.5.2 Containment Event Tree Structure

Figure 4.5-1 depicts the CET for HCGS. A description is provided below. A summary of the basic events considered in the supporting subtrees is then provided.

Initiating Event (I-Event)

Based on a thorough review of the potential for containment bypass and isolation failure, the CET top event representing this possibility was removed. This review encompassed by the Level 1 and Level 2 analysis. The Level 1 analysis determined a negligible frequency[4] of interfacing system LOCA leading to core damage. Under the Level 2, a thorough investigation of the potential for primary system and containment isolation failure was conducted. The results of this study, which is summarized in Section 4.4, concluded that the probability of isolation failure, given any of the core melt sequences treated in the Level 2, was sufficiently low that the combined frequency of the core melt accident and the isolation failure was less than 1×10^{-8}/reactor-yr. Consequently, such sequences were not analyzed further.

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[4] On the order of 2×10^{-9}/reactor-yr

Table 4.5-1. PDS Basic Events

Event	Outcomes	Definition
I-Event	S-LOCA	Small LOCA. Vessel depressurization required to prevent core damage if high pressure make-up is unavailable.
	S-MLOCA	Small to medium LOCA. Vessel depressurization with out S/RV (or ADS) to 500 psig, which is sufficient to allow condensate injection, prior to extensive core damage.
	L-LOCA	Large LOCA. Vessel depressurization sufficiently rapid for low pressure make-up systems to prevent core damage if they are operable.
	ST-SB	Short term station blackout. Total loss of AC power and turbine-driven make-up systems (i.e., HPCI and RCIC).
	LT-SB	Long term station blackout. Total loss of AC power. Turbine-driven make-up supplied until battery depletion results in failure.
	Trans.	Any transient leading to reactor scram without leakage from primary and with AC power available at least to one emergency bus.
SORV	SORV	One or more S/RV stuck open
	nSORV	S/RVs reclose and reseat
OP-S/RV	OP-SRV	At least one S/RV operates in relief mode
	nOP-SRV	No S/RV operable in relief mode due to hardware failure
DC	DC	DC power available on at least one bus
	nDC	No DC power

Table 4.5-1. PDS Basic Events (Continued)

Event	Outcomes	Definition
DP	SRV-DP	RPV depressurized using S/RVs prior to core damage
	nSRV-DP	S/RVs not used to depressurize RPV prior to core damage
ECP	ECP-Lo	Containment pressure at onset of core damage does not force S/Rv reclosure
	ECP-Hi	Containment pressure at core damage prevents S/Rv operation in relief mode
Vent	PCD-Vnt	Containment vented (12" - ?) prior to core damage
	PCDnVnt	Containment not vented prior to core damage
CRD	aCRD	CRD hydraulic system make-up available if AC power available
	fCRD	CRD failed
ECCS	aECCS	LPCI and/or LPCS available if AC power available
	fECCS	LPCI and LPCS failed
ALT	aALTINJ	Condensate and/or service water available for water make-up if AC power available
	fALTINJ	Both condensate and service water failed
SPRAY	aSPR	Drywell spray available
	fSPR	Drywell spray failed

Containment Event Tree Description

The CET shown in Figure 4.5-1 was developed based on considerations that had the potential to effect tne source term. Both timing, as well as magnitude, of the release contribute to the source term.

Six initiating events (or more appropriately, accident classes) were identified from the PDS. In order to facilitate interpretation of the CET, the initiating event is identified by a branch label corresponding to the definitions in Table 4.5-1.

Event 1: Vessel at Low Pressure (DP)

This top event establishes depressurization of the RPV prior to vessel breach. Success in this branch implies that RPV pressure is reduced either through the capability of the operator to depressurize the reactor or through a phenomenological condition that could induce RPV depressurization. Conversely, transient accident sequences in which the RPV is at low pressure (through opening the S/RVs) may be repressurized if the ADS valves cannot be maintained open. This event node is considered for high pressure PDSs to indicate a potential recovery or mitigating condition during core melt prior to vessel breach.

For accident sequences with the RPV at high pressure (and low pressure coolant injection initially unable to deliver makeup to the vessel due to the high pressure in the vessel), depressurization of the RPV can mean either of the following:

- The condition that precludes low pressure coolant injection is removed, and coolant makeup is likely to occur; or

- High RPV pressure that could exacerbate containment challenges at vessel breach (such as high pressure melt ejection) is removed.

This event node directly impacts the likelihood of the subsequent CET event nodes related to in-vessel recovery and early containment challenge.

Event 2: Injection Recovered (INJ)

The question asked in this top event is related to recovery of coolant injection after core degradation and prior to vessel breach. This event addresses the vessel injection recovery measures that have the potential for arresting core melting and subsequent thermal failure of the reactor vessel. It considers the possibility of low pressure injection systems working once the RPV is depressurized.

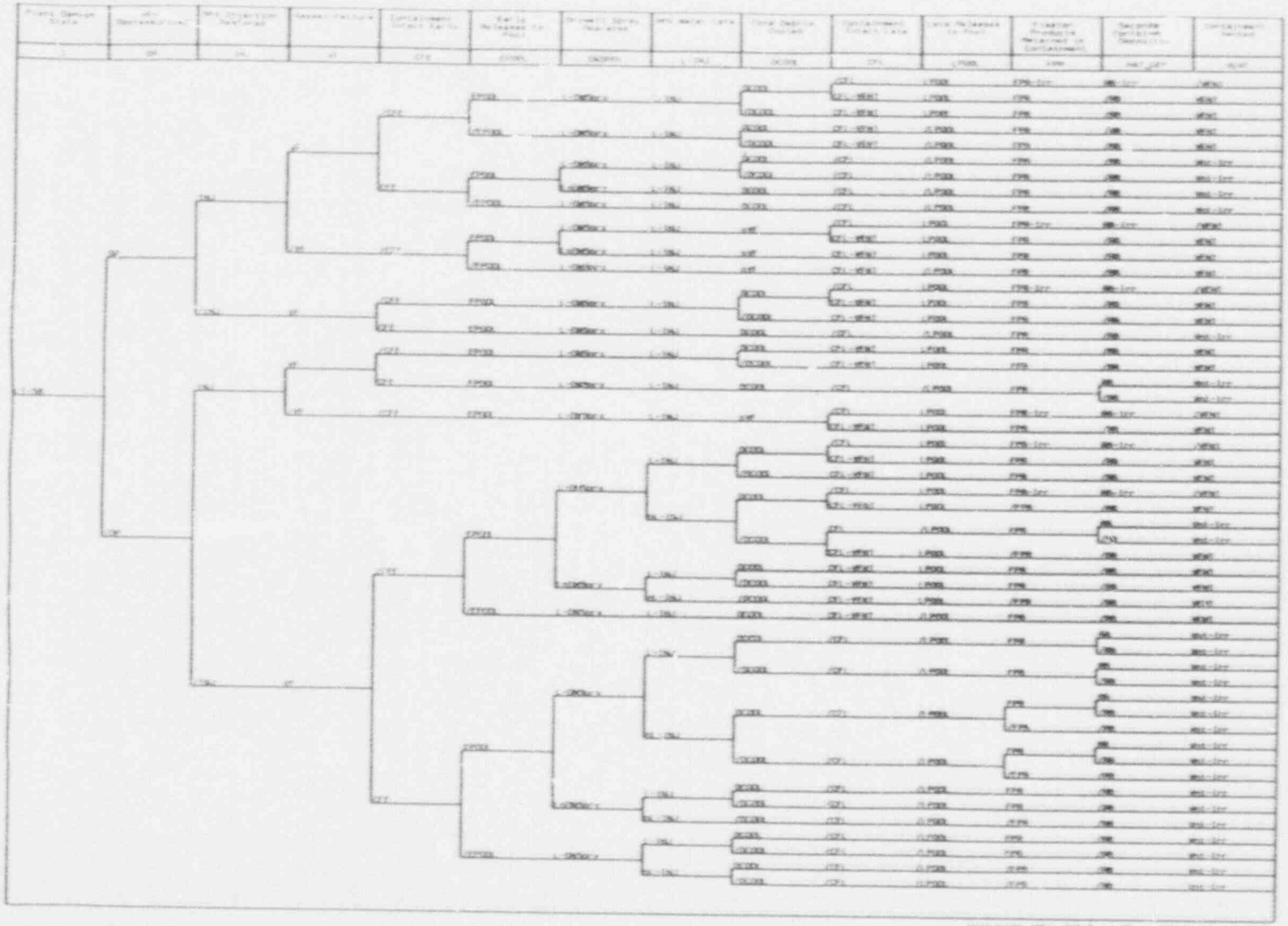

Figure 4.5-1. Generic Containment Event Tree

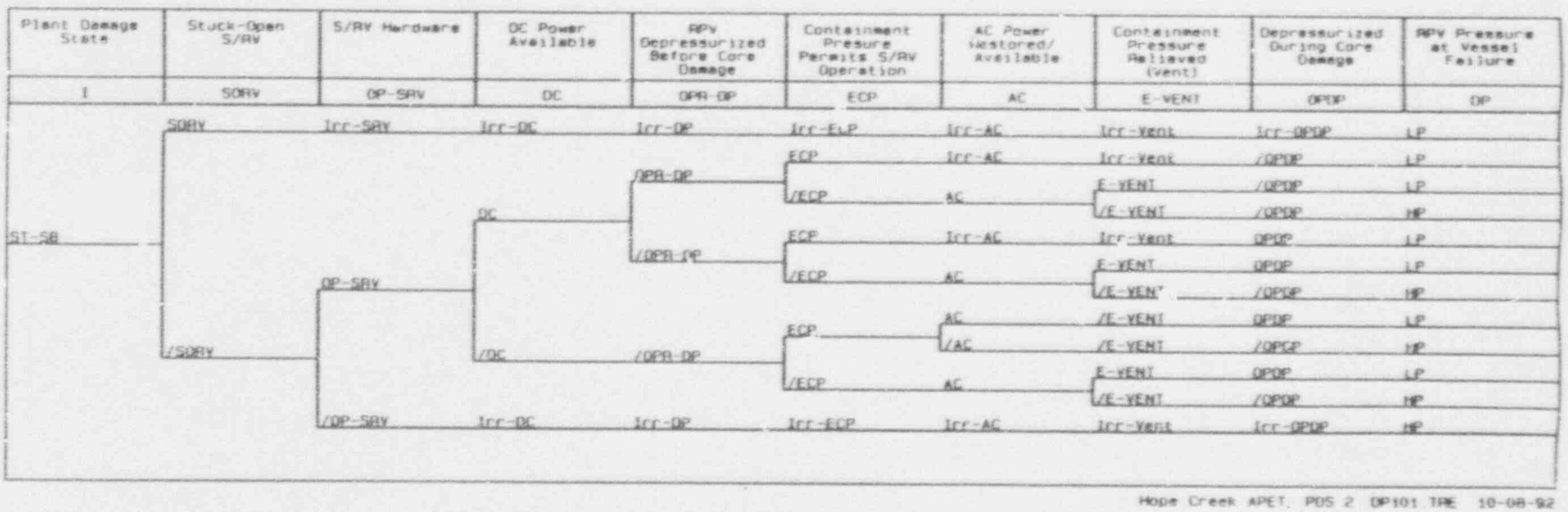

Figure 4.5-2. Depressurization (DP) Subtree

For PDSs where the containment is initially intact and the RPV is at pressure, core damage might be induced by lack of coolant make-up due to failure of the high pressure injection systems. However, the low pressure injection systems may be available, but coolant injection is prevented by conditions that preclude pump operation (i.e., RPV pressure exceeding the shut-off head). Once the high pressure condition is removed (as modeled in the previous event node, DP), coolant injection would most likely be recovered. This event considers the possibility of human intervention along with successful recovery of alternative systems that may have failed prior to core damage, but could potentially succeed given additional time for operator action such as high pressure systems. Success at this branch implies fission product releases from the fuel would be mitigated and establishment of a heat transfer pathway may assure maintenance of containment integrity.

For PDSs involving low RPV pressure (i.e., large LOCAs with failure to provide adequate coolant makeup), success at this event node is not likely, as implied by the accident sequence definition. The accident sequence cut-sets generally include human intervention in providing alternative injection systems; core damage occurs given failure to recover.

This event node directly affects the subsequent event node relative to arresting core melting and precluding thermal failure of the vessel bottom head.

Event 3: Vessel Failure (VF)

The question raised in this event addresses recovery of a degraded core within the vessel, which prevents vessel lower head thermal attack. Core melt recovery within the vessel is considered only to the extent that coolant make up has been successful in the previous event node (DP). This requires that core cooling be recovered prior to core blocking (MAAP model) or relocation of molten debris to the lower plenum and thermal attack of vessel head. Therefore, the primary consideration for successful in-vessel recovery is the time available from incipient core degradation to the point of non recovery.

Event 4: Early Containment Failure (CFE)

CFE not does include intentional venting. While venting constitutes a failure of the containment function, it is not a failure of the structure as could be implied. This model includes a venting event which distinguishes between containment venting and containment failure.

Early containment failure can be induced by the phenomena accompanying vessel failure. These are addressed specifically in the subtree. Containment leakage (failure size sufficient to prevent further long-term pressure increase and lead to reduction to atmospheric pressure over a period longer than ten to twelve hours) and containment rupture (failure size sufficient to reduce containment pressure to atmospheric within half an hour) are separated by the subsequent fission product retention (FPR) event. This distinction can significantly impact source term.

Event 5: Early Release to Pool (EPOOL)

The EPOOL event only considers complete suppression pool bypass. (i.e., releases to the drywell with wetwell-to-drywell vacuum breakers stuck open.) Under these conditions, fission product releases from the vessel during core damage are not subject to suppression pool scrubbing. Events that change the fission product transport pathways (as indicated in the EPOOL Subtree) occur either prior to core damage or vessel failure. The outcome of the Late Release to Pool (LPOOL) event is used to determine fission product scrubbing for releases occurring subsequent to vessel failure. This approach is slightly conservative (i.e., tends to overestimate releases) but is far more defensible than assuming the early release transport pathway pertains at vessel breach.

Event 6: Drywell Spray Operates Late (DWSPRY)

Water supplies to the core debris on the drywell floor following vessel failure has the potential to enhance fission product retention even if the debris is not cooled by this water. Scrubbing of releases from the core-concrete interaction is the principal retention mechanism. Cooling of the released gases (i.e., H_2, CO, and CO_2) and condensation of water vapor released from concrete ablation are secondary effects in that fission product residence time within the containment is increased. This event reflects the inclusion of these important considerations.

The HCGS drywell spray headers are located on the upper wall of the spherical section of the drywell; just below the knuckle. Other Mark I containments have their spray headers in the cylindrical section, above the knuckle. The position of the HCGS spray header suggests that more effective fission product removal from the drywell atmosphere could be accomplished because the spray droplets have a less obstructed area through which to fall; although the fall distance is reduced. Even with this consideration, it was decided to treat drywell spray and other sources of water to the debris that do not enter the drywell through the RPV (e.g., CRD flow from the severed hydraulic lines) as if they had the same effect. Thus, this event is a surrogate for all water that is supplied to the debris other than from "normal" injection, which is treated in Event 7. This simplification is expected to underestimate the retention of fission products by the containment when drywell sprays are operable.

Event 7: Injection to RPV Following Failure (L-INJ)

Volatile fission products released during the core degradation process are deposited on the RPV internals and vessel surfaces in significant quantities. Estimates that range up to ninety percent of the total release have been made. Following vessel failure the surfaces holding these deposits will be heated by their decay heat. The result is the revolatilization of these deposits. Revolatilization represents a significant source term to the drywell during the later phases of an accident. If injection to the RPV is reestablished during the later phases of a postulated accident, these releases will be largely prevented. The effect of this water will be to substantially lower RPV temperatures. If surface temperatures are maintained below 600-700°F, these release are expected to be insignificant. It is expected that water injection to the RPV will keep the surface temperatures low enough to eliminate the revolatilization source term. This event reflects this mitigation mechanism.

Event 8: Coolable Debris Forms Ex-Vessel (DCOOL)

This event is included in the CET to signify the termination of the core melt progression subsequent to vessel breach. The success branch at the CET node means that a coolable debris bed is formed, terminating concrete attack, and thus precluding ex-vessel fission product releases from core-concrete interactions. Following the success branch also implies that containment overpressure challenges from non-condensible gas generation is precluded, thus containment integrity may be maintained in the long term if heat removal or sufficient make-up is available. For example, for PDSs where the low pressure injection systems were previously unable to inject due to high RPV pressure, these systems would likely start to deliver coolant when the vessel is breached. Coolant injection could potentially quench the debris.

Failure at this branch implies that concrete attack occurs in the sumps on the drywell floor. Core debris remains hot and sparging of the concrete decomposition products through the melt releases the less volatile fission products to the containment atmosphere. This condition is considered more likely if a deep core debris bed is formed in the pedestal and, absent coolant addition, the debris is not able to effectively dissipate the decay heat to the surroundings. Should an impervious crust form, coolant addition would not likely terminate concrete attack, although the released fission product aerosols would be scrubbed by the overlying water pool.

Event 9: Late Containment Failure (CFL)

This event is included in the event tree to address the potential loss of containment integrity in the long term, after vessel breach and core-concrete interactions (CCI). Event CFL includes such events as overpressure failure of the primary containment, loss of containment integrity due to overtemperature, and basemat penetration (a less likely condition). The success path here depends strongly on the recovery of systems that establish a complete heat transfer pathway from the core debris to the ultimate heat sink. One of the most important considerations is related to the potential for overpressure failure due to the relatively smaller containment volume of the Mark I containment.

Event 10: Late Release to Pool (LPOOL)

This event is similar to event EPOOL. It addresses the importance of suppression pool scrubbing in mitigating the magnitude of fission products released from the debris. The success branch at this event node implies that the fission product transport path subsequent to late containment failure is through the suppression pool and the wetwell airspace. The suppression pool is not bypassed (e.g., the failure location is at the wetwell airspace and the pool water is not drained); fission products released during core-concrete interaction (ex-vessel releases) and revolatilization from vessel surfaces are scrubbed by the pool. The dependency of LPOOL on failure in EPOOL is noted in the CET.

Event 11: Fission Product Retention (FPR)

To quantify this event, the generic framework considers the effects of drywell spray and water on the drywell floor on fission product retention within the containment. Also, considered is the potential for revolatilization of fission products from the reactor pressure vessel (RPV) internal surfaces. Consideration of whether containment failure occurs through a leak or a rupture is also included (these failure modes are described above with reference to Event 4). Leak-size failures enhance fission product retention by increasing the extent of deposition within containment. This increase results from the greater residence time for fission products when the rate of egress is slowed by a small failure. Within the level of accuracy appropriate for the IPE, the added retention afforded by a leak-size failure is comparable to that afforded by water scrubbing of core-concrete interaction releases or deposition of airborne materials due to drywell spray operation in the Mark I configuration.

Event 12: Secondary Containment Retention (NAT_DEP)

This event is included in the CET to characterize the impact of mitigation afforded by the secondary containment following containment failure. This event is considered only for the CET progression paths that by-pass the suppression pool. This is done because the effects of suppression pool scrubbing completely overwhelm potential reactor building decontamination. The magnitude of fission products released from the primary containment can be significantly reduced in the secondary containment through removal mechanisms within the reactor building provided the residence time is significant.

Following primary containment loss of integrity, the secondary containment provides another barrier for mitigating release of radioactive materials to the environment since the secondary completely surrounds the primary containment. The secondary containment structure is designed with a 3 psig internal capacity. Blowout panels will relieves gases escaping from the primary containment into the secondary at a much lower pressure (~0.5 psid. Therefore, while the secondary containment does not provide containment for releases following primary containment failure, it does present substantial deposition sites for aerosol removal. Significant retention of fission products released from the primary containment is expected. The higher overall structural capacity is expected to lead to somewhat greater retention that the normal BWR Mark I and II reactor building designs. The principal reason is the absence of the sheet metal panelling surrounding the refueling floor, which is vulnerable to failure, providing a direct release pathway.

Event 13: Vent

The VENT event is a summary event that gives the model the capability to differentiate between containment failure and intentional venting. This avoids including containment venting as a form of containment failure. The model can now identify those sequences in which venting occurs but the containment does not fail.

Subtree Structure

Having described the events that comprise the main CET, the remainder of this section describes the structure for the logic trees that support the quantification of the CET top events. Only one of the subtrees is illustrated. For the remainder, a brief summary of the events that are included in each subtree is provided.

CET Event 1: Vessel at Low Pressure (DP)

The DP Subtree is shown in Figure 4.5-2. The first six events in the subtree reflect plant damage state (PDS) characteristics and are designated as PDS basic event in the logic model. Events seven through nine each represent at least one unique basic event.

Event 1: I-Event - Initiating Event

Event 2: SORV - Represents a PDS-dependent basic event indicating a stuck open relief valve.

Event 3: OP-SRV - operable Safety/Relief Valve

Event 4: DC - DC power available

Event 5: OPR-DP - Automatic or operator depressurization prior core damage.

Event 6: ECP - PDS-dependent basic event with success indicating that the containment pressure prior to core damage was below the level at which the S/RVs are operable in relief mode.

Event 7: AC - This event represents either a PDS-dependent basic event indicating the power recovery frequency or the availability of AC power for non-blackout PDSs.

Event 8: E-VENT - Venting is shown as if it will permit lowering the containment pressure to the point at which depressurization could occur within the time frame of core damage (i.e., within two hours).

Event 9: OPDP - RPV depressurization by operator after core damage, given previous failure to depressurize, is considered for four different sequences. The upper two sequences involve a prior error of omission.

Event 10: DP (Outcome) - System may either remain at High Pressure (HP) or it may be depressurized [Low Pressure (LP)].

CET Event 2: Injection Recovered (INJ)

The INJ Subtree contains eleven tow events relating to the recovery of low pressure injection to the RPV.

Event 1: I-Event - Initiating Event.

Event 2: AC - Same as Event 7 in the DP Subtree.

Event 3: CRD-ADQ - Adequacy of CRD flow to arrest in-vessel core-melt progression is addressed by this event. Reflects the analysts' level of confidence that CRD alone could arrest core damage.

Event 4: aCRD - PDS-dependent basic event.

Event 5: E_CRD - Represents probability that the operator maintains or recovers CRD flow.

Event 6: HP - Result of DP Subtree that indicates whether the vessel is depressurized prior to breach.

Event 7: aECCS - PDS-dependent basic event.

Event 8: E_ECCS - Represents probability that ECCS flow will be recovered once all conditions necessary to permit it are met.

Event 9: aALTINJ - PDS-dependent basic event.

Event 10: E_ALTI - Represents probability that alternate injection sources would not be aligned once all conditions making it possible were realized.

Event 11: INJ (Outcome) - Injection restored during the core damage process.

CET Event 3: Vessel Failure (VF)

The VF Subtree consists of four events that determine the probability of vessel failure due to thermal attack.

Event 1: INJ - Result of the INJ Subtree indicating the restoration of injection during core damage process.

Event 2: <26%_CM - This event represents the frequency with which injection would be recovered prior to the fraction of the fuel that has melted attaining the 26% level. It is assumed that recovery is virtually assured before this point.

Event 3: SLUMP - This event represents the subjective probability that core collapse to an uncoolable geometry would occur once more than 26% of the fuel had melted.

Event 4: VF - Represents the subjective probability that injection will arrest core melt progression. Also determines the outcome for this subtree.

CET Event 4: Early Containment Failure (CFE)

The CFE Subtree consists of eight events that establish the logic for containment failure due to phenomena associated with vessel failure. Containment failure due to overpressurization resulting from vessel blowdown is treated by looking at the sum of the pressure rise in containment resulting from vessel blowdown and the pressure in containment when the vessel fails. Both the pressure rise and the base pressure have been subdivided into low, medium, and high ranges. The numerical values for pressure corresponding to these ranges have established during CET quantification.

Event 1: E-Vnt - Represents the probability the pressure rise during core damage is mitigated by venting.

Event 2: BLOW-Hi - Pressure rise in containment due to vessel blowdown at failure in the high range. This pressure rise is determined based on the Vessel Blowdown (BLOW) Sub-subtree.

Event 3: CP-Hi - Base pressure in the containment immediately prior to vessel breach in the high range. This base pressure is determined based on the Containment Pressure Rise (CPRISE) Sub-subtree.

Event 4: BLOW-Md - Pressure rise in containment due to vessel blowdown at failure in the medium range given that it was not in the high range. BLOW Sub-subtree outcome.

Event 5: CP-Mod - Base pressure in the containment immediately prior to vessel breach in the moderate range given that it was not in the high range. CPRISE Sub-subtree outcome.

Event 6: CF-FCI - Indicates the occurrence of an FCI that produces containment failure. The FCI Sub-subtree is used to determine the outcome of this event.

Event 7: CF-MLT - Indicates the occurrence of drywell shell melt-through due to either debris impingement or spreading. The MELT Sub-subtree is used to determine the outcome of this event.

Event 8: CFE (Outcome) - Indicates the occurrence of early containment failure.

CET Event 5: Early Release to Pool (EPOOL)

The EPOOL Subtree consists of seven events that examine the potential for in-vessel releases to bypass the suppression pool. For bypass to occur there must be a pathway from the RPV to the drywell <u>and</u> either the drywell must have been vented, or a flow path must exist between drywell and wetwell airspaces that will cut off flow through the vents (i.e., torus-to-drywell vacuum breaker failure).

Event 1: LOCA - PDS-dependent basic event.

Event 2: SORV - PDS-dependent basic event.

Event 3: SO_TVB - Represents the frequency for a S/RV tailpipe vacuum breaker failing to reclose.

Event 4: E-Vnt - Probability of early venting.

Event 5: DWVENT - Probability that venting was from the drywell.

Event 6: C-DWVB - Represents the probability that one or more wetwell-to-drywell vacuum breakers fails to reclose.

Event 7: EPOOL (Outcome) - Represents sequence-dependent flag indicating pool bypass by releases.

CET Event 6: Drywell Spray Operates Late (DWSPRY)

The Drywell Spray Subtree establishes the status of drywell sprays following vessel failure. There are six events in the subtree.

Event 1: aSPRY - PDS-dependent basic event that establishes the availability of the CSS mode of RHR. Available implies operable if AC power is available.

Event 2: SBO - PDS-dependent basic event that establishes whether this accident is a station blackout.

Event 3: EAC - Represents the recovery of AC power prior to vessel failure.

Event 4: E-Spry - Represents the operation of drywell sprays prior to vessel failure.

Event 5: LAC - Represents the recovery of AC power following vessel failure (up until the time that a significant fraction of the total CCI release has occurred).

Event 6: L-Spry - Represents the operation of drywell sprays following vessel failure and the opportunity for AC power recovery.

CET Event 7: Injection to RPV Following Failure (L-INJ)

The Late Injection Subtree determines the probability that injection has been restored to the vessel before significant fission product revolatilization occurs. There are nine events in this subtree.

Event 1: aECCS - PDS-dependent event that establishes the operability of the LPCI mode of RHR and/or LPCS.

Event 2: aALTINJ - PDS-dependent event that establishes the operability of service water and/or condensate as sources of water injection.

Event 3: EAC - Represents the recovery of AC power prior to vessel failure.

Event 4: E-ECCS - Represents the operation of a low pressure ECCS injection system prior to vessel failure.

Event 5: E-ALT - Represents the operation of service water or condensate for RPV injection prior to vessel failure.

Event 6: LAC - Represents the recovery of AC power following vessel failure (up until the time that a significant fraction of the total CCI release has occurred).

Event 7: L-ECCS - Represents the operation of a low pressure ECCS injection after to vessel failure.

Event 8: L-ALT - Represents the operation of service water or condensate for RPV injection after to vessel failure.

Event 9: L-Fire - Represents the operation of the main fire pump system for RPV injection after to vessel failure.

CET Event 8: Coolable Debris Forms Ex-Vessel (DCOOL)

The DCOOL Subtree consists of seven top events that establish the conditions in the drywell following vessel failure. Coolability is assessed separately in the subtree for seven different possible sets of conditions. The subtree thus involves significant decomposition in determining coolability. The probability of coolability conditions to be different if the debris falls into water at vessel failure instead of water being added to the top after vessel failure.

Event 1: VF - Outcome of the VF Subtree.

Event 2: INJ - Outcome of the INJ Subtree.

Event 3: SPRAY - Represents the probability of operating the drywell sprays prior to vessel breach.

Event 4: WATER - Represents the late addition of water to the debris based on the WATER Sub-subtree.

Event 5: DSPRS - Represents the dispersion of core debris by an energetic event associated with vessel breach. Dispersal is based on the DISPERSE Sub-subtree.

Event 6: SLUMP - Event 3 in the VF Subtree.

Event 7: DCOOL (Outcome) - Represents the coolability of core debris.

Conditions under which coolability was assessed are discussed in the Appendix.

CET Event 9: Late Containment Failure (CFL)

The CFL Subtree consists of six top events that address the phenomena that could lead to early containment failure. The principal failure mechanisms considered are failure due to noncondensable gas generation and thermal failure of seals. Hydrogen burn is assumed not to contribute to late containment failure. This assumption is believed to be valid for a Mark I containment, independent of nitrogen inerting. If the containment were not inerted a hydrogen burn would certainly occur at vessel breach consuming the available oxygen and contributing to the pressure rise that may fail containment early. A late burn would thus be excluded due to lack of oxygen.

Event 1: CFE - Outcome of the CFE Subtree.

Event 2: WATER - Represents the late addition of water to the debris based on the WATER Sub-subtree.

Event 3: DCOOL - Outcome of the DCOOL Subtree.

Event 4: TEMP - Represents the probability of thermal failure of the drywell hatch or electrical penetration seals.

Event 5: SUMP - Represents CCI ablation of the drywell sumps followed by failure of the drywell shell below the drywell floor.

Event 6: CFL (Outcome) - Indicates the occurrence of late containment failure.

CET Event 10: Late Release to Pool (LPOOL)

The LPOOL Subtree consists of seven top events that examine the potential for ex-vessel releases (i.e., revolatilization and CCI) to bypass the suppression pool.

Event 1: CFE - Outcome of the CFE Subtree.

Event 2: CFL - Outcome of the CFL Subtree.

Event 3: TEMP - Same event as Event 5 in the CFL Subtree.

Event 4: WW_Fail - Represents the probability that containment failure occurred in the drywell (i.e. failure not in the torus).

Event 5: C-DWVB - Represents the probability that one or more wetwell-to-drywell vacuum breakers fail to reclose.

Event 6: POOL - Represents the probability of loss of suppression pool level below the downcommer due to containment failure in the torus.

Event 7: LPOOL (Outcome) - Indicates the occurrence of late pool bypass.

CET Event 11: Fission Product Retention (FPR)

The FPR Subtree consists of nine top events that establish the conditions in the containment during the late phases of the postulated accident. The outcome of this tree establishes the containment decontamination factor for early and late releases. Values are assigned for leak or rupture, with and without water overlying the debris on the drywell floor.

Event 1: VF - Outcome of the VF Subtree

Event 2: L-INJ - Outcome of the L-INJ Subtree.

Event 3: SPRAY - Represents the late operation of drywell spray based on the DWSPRY Subtree.

Event 4: CFE - Outcome of the CFE Subtree.

Event 5: CFL - Outcome of the CFL Subtree.

Event 6: TEMP - Same event as Event 4 in the CFL Subtree.

Event 7: SUMP - Same event as Event 5 in the CFL Subtree.

Event 8: LEAK - Represents the probability that effective containment leakage area is equal to that of a rupture.

Event 9: FPR (Outcome) - Represents the outcome of fission product retention.

CET Event 12: Secondary Containment Retention (NAT_DEP)

The NAT_DEP Subtree includes twelve top events that establish the decontamination factor applied to represent retention in the secondary containment. Ascertaining the likely residence time for fission products in the secondary containment based on the established accident progression is the principal focus. Providing a basis for determining the operability of the standby gas treatment system (SGTS) is also addressed.

Event 1: CFE - Outcome of the CFE Subtree.

Event 2: CFL - Outcome of the CFL Subtree.

Event 3: RB-By - Represents the determination of whether containment failure was due to venting using a pathway that bypasses the reactor building.

Event 4: VENT - Represents the occurrence of venting based on the VENT Subtree.

Event 5: DCOOL - Outcome of the DCOOL Subtree.

Event 6: HIH2 - Represents the determination of whether high hydrogen production occurred during the core damage process.

Event 7: LEAK - Represents the determination of whether containment failure resulted in rupture.

Event 8: BURN - Represents two different probabilities relative to determination of whether hydrogen combustion occurs in the reactor building. These are:

(a) With sufficient hydrogen and containment leak.

(b) With sufficient hydrogen and containment rupture.

Event 9: SGTS - Represents the operation of the SGTS after containment failure. SGTS is only assumed to handle leak-type containment failures.

Event 10: HI_Fail - Represents the determination of whether containment leakage was to the refueling floor.

Event 11: SUMP - Same event as Event 5 in CFL Subtree.

Event 12: Nat_Dep - Represents the determination of whether the reactor building would provide significant fission product retention by natural deposition mechanisms.

CET Event 13: VENT

The VENT Subtree consists of eleven events. It is used to determine the occurrence of containment venting during core damage (i.e., subsequent to the start of core damage but prior to vessel failure). Venting is separated from containment failure since it represents an intentional action rather than a consequence of phenomena associated with the progression of a postulated accident.

Event 1: aVnt - PDS-dependent basic event.

Event 2: VE-Vnt - PDS-dependent basic event (PCD-Vnt).

Event 3: ECP - PDS-dependent basic event.

Event 4: H2Hi - This event represents the subjective probability that hydrogen generation during core damage will exceed the level required to increase containment pressure beyond the level at which procedures would require venting.

Event 5: SBO - PDS-dependent basic event.

Event 6: E-AC - Same event as Event 7 in the DP Subtree.

Event 7: E-Vnt - This event represents the probability that venting did not occur early.

Event 8: DCOOL - Outcome of the DCOOL Subtree.

Event 9: L-AC - Represents the probability that AC power was not restored during the period over which CCI would result in the significant release of fission products.

Event 10: L-Vnt - Represents the probability that the containment would be vented late given that no leak has occurred, AC power is available, and the pressure continues to rise due to noncondensable gas generation.

Event 11: VENT (Outcome) - This event represents the outcome of containment venting.

4.6 Accident Progression and CET Quantification

4.6.1 Containment Load Assessment

This section will discuss the use of MAAP to predict containment loads and accident timing.

4.6.2 Survivability of Engineered Safeguards

The probability of failure of injection systems following containment failure was found to be significant. A discussion of the basis for this will be incorporated in this section.

4.6.3 CET Quantification

The branch point probabilities used in the CET were quantified using several different methods depending on whether they were specific to the HCGS plant or generic to Mark I BWRs, and on whether they involved human reliability or system reliability. The following discussion outlines the methodology used for each of these CET variable classes. Table A-1 in Appendix A provides a list of the variables in the CET, along with the variable class, a brief description of each, and the values assigned in the CET.

Many of the plant-specific CET variables were evaluated based on results of computer code calculations (shown as Class 2 in Table A-1). The primary source for these results was the set of calculations performed by PSE&G staff using MAAP [12]. In some cases, the results calculated using the MELCOR [13] and CONTAIN [14] computer codes available in the literature were used to supplement the MAAP results. For example, variables associated with the magnitude of hydrogen production under various conditions were easily determined from the MAAP calculations. The possible range was evaluated based on NRC-contractor assessments (i.e., BWRSAR/CONTAIN, MELCOR, and STCP calculations as they were available and relevant to scenarios postulated for HCGS). Other variables, such as those associated with radionuclide retention in the primary system, reactor building, or containment building are also determined principally from MAAP calculations.

Other plant specific variables were not available from computer code calculations, but could be determined by comparison to analyses performed for similar plants (Class 3 in Table A-1). An important example of this class of variable is the probability of drywell shell meltthrough. An analysis of drywell shell failure for a plant similar to HCGS has been published by Theofanous as NUREG/CR-5423 [15]. In assessing the probabilities of drywell shell failure under a variety of accident conditions, the results in NUREG/CR-5423 were used, but were modified to account for differences between HCGS and the reference plant used in the analysis.

Generic CET variables were variables for which specific features of the HCGS plant are not important or in which the uncertainty in the phenomena is far greater than the potential impact of plant design (Class 1 in Table A-1). For these variables, CET values were taken from other analyses or other PRAs. Prime examples of generic CET variables are the probabilities of an in-vessel steam explosion, or of vessel failure given an in-vessel steam explosion. In both of these examples, probabilities were taken from the NUREG-1150 analysis for the Peach Bottom plant.

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Human reliability analyses (HRA) (Class 4 in Table A-1) were performed for CET variables associated with operator actions during the accident. In some cases, the timing of the operator actions (i.e., the time period in which the operator can respond) was determined from the MAAP analyses of the accident sequence. Timing was then considered in the HRA assuming high stress conditions.

System reliability analyses (Class 5 in Table A-1) were performed for CET variables concerned with functionality (i.e., availability) of equipment that could potentially mitigate the accident. Standard fault tree techniques were used in this analysis. If a similar analysis was performed as part of the front-end (Level 1) assessment, those results were used, but were modified to account for conditions present during the back-end (Level 2) portion of the accident.

4.7 Radionuclide Release Characterization

The consequences of a reactor accident are determined to a large extent by the magnitude of the radionuclide release to the environment. Thus to complete the IPE assessment of plant vulnerability, predictions of the radionuclide source term are required. This section describes the source term algorithm utilized in the IPE HCGS, the quantification of the source term parameters, and the results obtained.

4.7.1 Source Term Prediction

Source terms for the HCGS IPE were calculated with a source term algorithm built into the containment event tree model. The source terms are defined in terms of the fractional release of several key radionuclide groups. The algorithm estimates source terms based on sequence-dependent radionuclide release fractions (RFs) and decontamination factors (DFs) that are input as part of the containment event tree (CET).

A wide variety of radioactive fission products build up in an operating reactor core. Different fission product species would be expected to behave in very different ways during the course of an accident. For example, noble gases (primarily krypton and xenon) evolve from the fuel during the fuel degradation process. Whereas, ruthenium stays within the fuel matrix until very high temperatures are achieved during molten core-concrete interactions. Furthermore, some species are readily decontaminated by pool scrubbing (such as iodine), while others are unaffected by decontamination mechanisms (such as noble gases). Because so many fission product isotopes exist and because many types of isotopes behave differently throughout the course of an accident, they are typically categorized into groups. Full-scope PRAs typically treat nine radionuclide groups in their source term evaluations. For the IPEs, the EPRI generic methodology tracks only the five most risk significant groups [12]. This approach has been adopted for the HCGS IPE.

The five radionuclide groups being evaluated in the HCGS IPE are (1) noble gases, (2) iodine, (3) cesium, (4) tellurium, and (5) strontium. The IPE source term algorithm is designed to calculate the fractional release of each of these radionuclide groups for each of the main accident sequences. The results of the source term evaluation is a list of release fractions representing the environmental release for each group.

4.7.1.1 Source Term Algorithm

The source term algorithm for the HCGS IPE is comprised of a set of equations that relate release fractions and decontamination factors in a self-consistent fashion to calculated fractional release to the environment. The release fraction for a given radionuclide group is defined as the fraction of the available material that evolves from the core debris and becomes available for release. Once evolved from the core debris, various decontamination mechanisms act on the airborne radionuclides to limit their release to the environment.

The radionuclide release and decontamination mechanisms considered in the source term algorithm are shown below.

Key Radionuclide Release Mechanisms[5]
- (A) In-Vessel - $RF_{IV}(i)$
- (B) Molten Core-Concrete Releases - $RF_{MCCI}(i)$
- (C) In-Vessel Revolatilization - $RF_{REV}(i)$

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[5] Releases from high pressure melt ejection were not included in the model. The Mark I drywell is both small and largely filled with equipment and piping. The pressure suppression system would quickly sweep airborne debris to the suppression pool. Uncertainties with respect to fission product releases from oxidizing airborne debris are very large since there is no experimental basis. Ruthenium release was cited as likely in NUREG-1150 but the basis is not clear and the HCGS source term model does not treat this group explicitly. The omission of this release mechanism is appropriate based on these considerations.

Key Decontamination Mechanisms
(1) Early Pool Scrubbing - $DF_{EPOOL}(i)$
(2) Late Pool Scrubbing - $DF_{LPOOL}(i)$
(3) Primary System Natural Deposition - $DF_{VSL}(i)$
(4) Containment Natural Deposition - $DF_{CONT}(i)$
(5) Drywell Sprays - $DF_{SPR}(i)$
(6) Secondary Containment Natural Deposition - $DF_{RB}(i)$

The decontamination mechanisms that are active for the in-vessel release, molten core-concrete release, and revolatilization release are shown below.

(A) In-Vessel Releases
(1) Primary System Natural Deposition
(2) Early Pool Scrubbing
(3) Containment Natural Deposition
(4) Drywell Sprays
(5) Secondary Containment Natural Deposition

(B) Molten Core-Concrete Interactions Releases
(1) Overlying Water Pool
(2) Drywell Sprays
(3) Containment Natural Deposition
(4) Late Pool Scrubbing
(5) Secondary Containment Natural Deposition

(C) Revolatilization Releases
(1) Drywell Sprays
(2) Containment Natural Deposition
(3) Late Pool Scrubbing
(4) Secondary Containment Natural Deposition

The equations used in the source term algorithm to quantify the individual release fractions and the environmental release are shown below.

Total Environmental Release

The total atmospheric release of a particular isotope group is expressed as the fraction of the core inventory escaping from containment failure times the sum of the releases due to in-vessel, molten core-concrete interactions and revolatilization releases for that group. This relationship is estimate based on the following equation:

$$R_{TOT}(i) = \left[R_{IV}(i) + R_{MCCI}(i) \quad R_{REV}(i)\right] \times F_{VENT} \tag{1}$$

where

$R_{TOT}(i)$	=	Total release from containment to atmosphere
$R_{IV}(i)$	=	In-vessel release
$R_{MCCI}(i)$	=	Molten core-concrete release
$R_{REV}(i)$	=	In-vessel revolatilization release
F_{VENT}	=	Escape fraction for containment venting

In Vessel Release

The in-vessel release is calculated by dividing the in-vessel release fraction by the in-vessel decontamination factor in the following relationship:

$$R_{IV}(i) = RF_{IV}(i) \times \frac{1}{DF_{IV}(i)} \tag{2}$$

$$DF_{IV}(i) = DF_{EPOOL}(i) \times DF_{VSL}(i) \times DF_{CONT}(i) \times DF_{SPR}(i) \times DF_{RB}(i) \tag{3}$$

and

$RF_{IV}(i)$	=	In-vessel release fraction
$DF_{IV}(i)$	=	In-vessel decontamination factor
$DF_{EPOOL}(i)$	=	Early pool scrubbing decontamination factor
$DF_{VSL}(i)$	=	Primary system natural deposition decontamination factor
$DF_{CONT}(i)$	=	Containment natural deposition decontamination factor
$DF_{SPR}(i)$	=	Drywell sprays decontamination factor
$DF_{RB}(i)$	=	Secondary containment natural deposition decontamination factor

MCCI Release

Similarly, the release from molten core-concrete interactions is found by multiplying the fraction of the isotope group remaining after in-vessel and DCH releases times the MCCI release fraction and dividing by the MCCI decontamination factor:

$$R_{MCCI}(i) = (1 - RF_{IV}(i)) \times \frac{RF_{MCCI}(i)}{DF_{MCCI}(i)} \tag{4}$$

$$DF_{MCCI}(i) = DF_{LPOOL}(i) \times DF_{SPR}(i) \times DF_{CONT}(i) \times DF_{RB}(i) \tag{5}$$

$RF_{MCCI}(i)$	=	Molten core-concrete release fraction
$DF_{MCCI}(i)$	=	Molten core-concrete decontamination factor
$DF_{LPOOL}(i)$	=	Late pool scrubbing decontamination factor

Revolatilization Release

Release due to Revolatilization is calculated by multiplying the in-vessel releases remaining after vessel decontamination times the ratio of the revolatilization release and the revolatilization decontamination factor:

$$R_{REV}(i) = \left[R_{IV}(i) \times \left(1 - \frac{1}{DF_{VSL}(i)}\right)\right] \times RF_{REV}(i) \times \frac{1}{DF_{REV}(i)} \tag{6}$$

where

$$DF_{REV}(i) = DF_{SPR}(i) \times DF_{CONT}(i) \times DF_{LPOOL}(i) \times DF_{RB}(i) \tag{7}$$

and

$RF_{REV}(i)$	=	Revolatilization release fraction
$DF_{REV}(i)$	=	Revolatilization decontamination factor

The following rules and assumptions are assumed to apply throughout the source term evaluation, and are built into the algorithm:

(1) If the pool scrubbing DF is large (i.e., if $DF_{POOL}(i) > 10$), the effect of drywell sprays is neglected. This is done to prevent double counting of the effect of water and spray decontamination.

(2) All noble gas decontamination factors are unity.

4.7.1.2 Source Term Quantification

The release fractions and decontamination factors must be specified for each of the five radionuclide groups. The values for these parameters are specified within the CET, and are used later in the CET as part of the evaluation of envi onmental release. The parameter values used in the CET are discussed below.

In-Vessel Release Fraction - $RF_{IV}(i)$

This parameter is specified in CET based on the VF Top Event. The range in in-vessel release fractions for each radionuclide group are: Noble gases - 0.2 to 1.0; Iodine - 0.12 to 0.6; Cesium - 0.04 to 0.2; Tellurium - 0.0; and Strontium - 0.0. Higher values are assigned to branches with vessel rupture, no coolant injection or core slumping. Lower values are assigned in cases with coolant injection, or less than 26% core melt.

Primary System Decontamination Factor - $DF_{VSL}(i)$

This parameter is also specified based on the occurrence of vessel failure. The range in the primary system DF for each radionuclide group is: Noble gases - 1.0; Iodine - 2 to 10; Cesium - 2 to 10; Tellurium - 1.5 to 10; and Strontium - 1.5 to 10. High primary system DFs are assigned to cases with coolant injection, or less than 26% core melt. Low primary system DFs are assigned to cases with vessel rupture, no coolant injection, or core slumping.

Suppression Pool Decontamination Factor for Early Releases - $DF_{EPOOL}(i)$

This parameter is specified in conjunction with the EPOOL Top Ev The DFs for each radionuclide group are set to unity in cases with early suppression pool bypass. For cases without early bypass, the DFs for each radionuclide group (except noble gases) are set equal to values ranging from 50 to 1000.

Drywell Spray Decontamination Factor - $DF_{SPR}(i)$

This parameter is specified in conjunction with the SPRAY Top Event. If the sprays have failed early or no AC power is available, the spray DFs are set to unity. In other cases, the spray DFs are assumed to range from 1.0 to 2.0 for each of the radionuclide groups (except noble gases).

Revolatilization Release Fraction - $RF_{REV}(i)$

This parameter is specified in conjunction with the WATER Top Event. The revolatilization release fractions are assumed to have the following ranges: Noble gases - 1.0 (since DF_{IV} is unity for noble gases this value is moot); Iodine - 0.0 to 0.6; Cesium - 0.0 to 0.2; Tellurium - 0.0 to 0.01; and Strontium - 0.0 to 0.1. High values are assigned in scenarios without late injection, while low values are assigned if there is late coolant injection.

Molten Core-Concrete Release Fraction - $RF_{MCCI}(i)$

This parameter is specified in conjunction with the DCOOL Top Event. The MCCI release fractions are assumed to have the following ranges: Noble gases - 0.0 to 1.0; Iodine - 0.0 to 1.0; Cesium - 0.0 to 0.6; Tellurium - 0.0 to 0.4; and Strontium - 0.0 to 0.4. Low values are assigned to cases in which the debris is coolable ex-vessel, while high values are assigned if the debris is not coolable.

Suppression Pool Decontamination Factor for Late Releases - $DF_{LPOOL}(i)$

This parameter is assigned with respect to the LPOOL Top Event. With late suppression pool bypass, the DFs are equal to unity. Without bypass, the late suppression pool DFs are assumed to range from 20 to 500 for iodine, cesium, tellurium, and strontium. The noble gas DF is, of course, 1.0.

Containment Decontamination Factor - $DF_{CONT}(i)$

This parameter is assigned based on the FPR Top Event. This question is used to broadly categorize fission (for later binning purposes) product retention in the containment. The containment DFs are assumed to range from 1.4 to 50.0 for each of the radionuclide groups (except noble gases). High values are assigned in cases without vessel failure or containment failure, or cases with containment failure by leakage. Low values are assigned in cases with containment failure by rupture.

Secondary Containment Decontamination Factor - $DF_{RB}(i)$

This parameter is assigned based on the NAT_DEP Top Event. This question is used to broadly categorize (for binning purposes) fission product retention in the reactor building. The reactor building DFs are assumed to range from 1.0 to 200 for each of the radionuclide groups (except noble gases). High values are assigned in cases with containment failure in the sump or by leakage, and in cases with the Standby Gas Treatment System (SGTS) in operation. Low values are assigned in cases with containment failure by rupture, or reactor building bypass, or cases in which a hydrogen burn has occurred in the reactor building.

4.7.2 Results from the Base Case Evaluation of Radionuclide Release

Radionuclide release was evaluated for each of the five of the eight plant damage states discussed in Section 4.3. This section presents the results of that evaluation, and discusses some of the insights gained from this assessment.

Table 4.7-1 Summary of Significant Accident Progression Bins for PDS-1

Initiating Event - Transient (TW)												
Most Likely Accident Progression Sequences												
Seq. Freq.	DP	INJ	VF	CFE	EPOOL	DWSpry	L-INJ	DCOOL	CFL	LPOOL	FPR	RB
~0.1-0.5	yes	yes	no	no	yes	yes	yes	na	vent	yes	yes	no
~0.2-0.8	yes	yes	yes	no	yes	yes	yes	yes	vent	yes	yes	no
Moderate to High Source Term Sequences												
~0.01-0.05	yes	yes	yes	yes	yes	yes	yes	yes	na	no	yes	no
~10^{4} - 10^{2}	yes	yes	yes	yes	yes	yes	yes	no	na	no	yes	no

Table 4.7-2 Summary of Significant Accident Progression Bins for PDS-2

Initiating Event - Transient (TQUV)												
Most Likely Accident Progression Sequences												
Seq. Freq.	DP	INJ	VF	CFE	EPOOL	DWSpry	L-INJ	DCOOL	CFL	LPOOL	FPR	RB
~0.2-0.8	yes	yes	no	no	yes	yes	yes	na	vent	yes	yes	no
~0.1-0.5	yes	yes	yes	no	yes	yes	yes	yes	vent	yes	yes	no
Moderate to High Source Term Sequences												
~0.01-0.05	yes	yes	yes	yes	yes	yes	yes	yes	na	no	yes	no
~10^{3} - 0.1	yes	no	yes	yes	yes	yes	yes	yes	na	no	yes	no
~10^{3} - 0.1	yes	yes	yes	yes	yes	yes	yes	no	na	no	yes	no
~10^{4} - 0.012	yes	no	yes	yes	yes	yes	yes	no	na	no	yes	no

Table 4. -3 Summary of Significant Accident Progression Bins for PDS-3

Initiating Event - ST-SBO												
Most Likely Accident Progression Sequences												
Seq. Freq.	DP	INJ	VF	CFE	EPOOL	DWSpry	L-INJ	DCOOL	CFL	LPOOL	FPR	RB
~0.1-0.5	yes	no	yes	yes	yes	yes	yes	yes	na	no	yes	no
~0.1-0.5	yes	no	yes	yes	yes	yes	no	yes	na	no	no	no
~0.1-0.5	yes	no	yes	yes	yes	yes	yes	no	na	no	yes	no
~0.1-0.5	yes	no	yes	yes	yes	yes	no	no	na	no	no	no
Moderate to High Source Term Sequences												
~0.1-0.5	yes	no	yes	yes	yes	yes	yes	yes	na	no	yes	no
~0.1-0.5	yes	no	yes	yes	yes	yes	no	yes	na	no	no	no
~0.1-0.5	yes	no	yes	yes	yes	yes	yes	no	na	no	yes	no
~0.1-0.5	yes	no	yes	yes	yes	yes	no	no	na	no	no	no

Table 4.7-4 Summary of Significant Accident Progression Bins for PDS-4

Initiating Event - L-LOCA												
Most Likely Accident Progression Sequences												
Freq.	DP	INJ	VF	CFE	EPOOL	DWSpry	L-INJ	DCOOL	CFL	LPOOL	FPR	RB
2-0.8	yes	yes	no	no	yes	yes	yes	na	vent	yes	yes	no
$\sim 10^{-5}$-0.01	yes	yes	no	no	no	yes	yes	na	vent	no	yes	no
Moderate to High Source Term Sequences												
$\sim 10^{-5}$-0.01	yes	no		yes	no	yes	yes	yes	na	yes	yes	no
$\sim 10^{-5}$-0.01	yes	no	yes	yes	yes	yes	yes	yes	na	yes	yes	no
$\sim 10^{-5}$-0.01	yes	no	yes	yes	no	yes	yes	no	na	yes	yes	no
$\sim 10^{-5}$-0.01	yes	no	yes	yes	yes	yes	yes	no	na	yes	yes	no

Table 4.7-5 Summary of Significant Accident Progression Bins for PDS-5

Initiating Event - Transient (TQUX)												
Most Likely Accident Progression Sequences												
Seq. Freq.	DP	INJ	VF	CFE	EPOOL	DWSpry	L-INJ	DCOOL	CFL	LPOOL	FP	RB
~0.2-0.8	yes	yes	no	no	yes	yes	yes	na	vent	yes	yes	no
~0.1-0.6	yes	yes	yes	no	yes	yes	yes	yes	vent	yes	yes	no
Moderate to High Source Term Sequences												
~10^{5} - 0.1	yes	yes	yes	yes	yes	yes	yes	yes	na	no	yes	no
~10^{5} - 0.1	yes	no	yes	yes	yes	yes	yes	yes	na	no	yes	no
~10^{5} 0.01	yes	yes	yes	yes	yes	yes	yes	no	na	no	yes	no
~10^{5} 0.01	yes	no	yes	yes	yes	yes	yes	no	na	no	yes	no

The CET discussed in Section 4.5 was evaluated using the EVNTRE computer code [16]. Only those sequences shown in the CET (Figure 4.5-1) were evaluated. Each accident progression sequence is characterized by fourteen different characteristics (or top events) each of which has an impact on the magnitude of radionuclide release to the environment.

The following discussion summarizes the radionuclide release results for five plant damage states considered in the current assessment.

Radionuclide Release Results for PDS-1 (TW sequence)

The CET for PDS-1 (a TW sequence) is shown in Figure 4.7-1. The fourteen characteristics are shown across the top of the tree.

The two dominant sequences have the following common features:

* no early containment failure,
* early and late suppression pool decontamination,
* early and late coolant injection,
* active drywell sprays,
* good aerosol retention in the primary system and containment, and
* late venting of containment.

In addition, there are no core-concrete releases in the two sequences. In one case, the reactor vessel does not fail, and, in the other case, the ex-vessel debris is coolable.

The two dominant sequences lead to very low radionuclide release, including less than 50% of the noble gases, less than one percent of the iodine and cesium, and negligible releases of tellurium and strontium. Consequently, these sequences would not be expected to be important from the standpoint of risk.

To screen for sequences more likely to be risk significant, the source term results were examined and sequences with releases of 90% or more of the noble gases, and 1% or more of the iodine were highlighted. The highest frequency sequences in this subgroup are shown in Table 4.7-1.

The two accident progression sequences shown in the table are responsible for more than ninety percent of the higher source term sequences. The sequences are nearly identical, the only difference being that the higher probability sequence has coolable debris ex-vessel, whereas the other sequence does not. The other characteristics of the two sequences are summarized below:

- reactor vessel failure,
- early containment failure,
- early, but not late, suppression pool decontamination,
- early and late coolant injection,
- active drywell sprays,
- good aerosol retention in the primary system and containment, and
- poor aerosol retention in the reactor building.

Both sequences produce moderate to high releases of noble gases, iodine, and cesium. The sequence with coolable debris ex-vessel has negligible release of tellurium and strontium since core-concrete interactions are prevented. In the sequence with uncoolable ex-vessel debris core-concrete interactions occur, and approximately one-tenth of the tellurium and strontium are released. With all sequences considered, the probability of a moderate to high release (as defined by the screening criteria discussed above) is small.

As mentioned, releases of tellurium and strontium are only significant when core-concrete interactions occur, i.e., when the core debris is not coolable. For PDS-1, only a few percent of the accident progression sequences have uncoolable debris.

Reactor vessel failure and early containment failure are characteristics of these and the other sequences with moderate to high source terms. In the CET, reactor vessel failure occurs by two mechanisms: in-vessel fuel-coolant interactions (FCI) or thermal failure of reactor vessel. In the CET evaluation, thermal failure was by far the dominant contributor to vessel failure.

Radionuclide Release Results for PDS-2 (TQUV sequence)

The results of the CET evaluation for PDS-2 are summarized in Table 4.7-2. As was the case with PDS-2, a large fraction of the total sequence frequency is accounted for by two accident progression sequences. The characteristics of these two sequences are identical to those of the dominant sequences for PDS-1. As before, the radionuclide releases from these two sequences are extremely small.

Table 4.7-2 also shows the accident progression sequences likely to be significant from the standpoint of radionuclide release. The same screening criteria (i.e., ≥ 90% noble gas release, and ≥ 1% iodine release) were used to identify the number of sequences to those with moderate to high source terms. The four sequences shown in the table comprise a large fraction of the total frequency of the moderate to high source term sequences.

The first and third sequences shown have accident progression characteristics identical to the dominant source term sequences identified for PDS-1. They differ from one another only in whether ex-vessel core debris is coolable (first sequence) or not (third sequence). The second and fourth sequences differ from the first and third sequences only in the availability of coolant injection during the in-vessel melt progression phase of the accident. Whereas coolant injection is used in the first and third sequences, it is not used in the second and fourth sequences. Again, the only difference between the second and fourth sequences is whether the ex-vessel core debris is coolable or not.

Radionuclide Release Results for PDS-3 (ST-SBO sequence)

The results of the CET evaluation for PDS-3 are summarized in Table 4.7-3. Because all accident progression sequences were found to have both vessel failure and early containment failure, as well as late suppression pool bypass, the source terms for these sequences tended to be generally higher than for the other plant damage states. All sequences were found to meet the screening criteria for moderate to high source term release. Ex-vessel core debris was uncoolable in approximately half of the accident progression sequences. The releases of tellurium and strontium are significant in these sequences.

The four dominant sequences in terms of overall frequency were all found to satisfy the screening criteria for moderate to high source term. These four sequences differed only in whether late injection was available, ex-vessel core debris was coolable, or fission product retention was good in the primary system and containment. Common features of these accident progression scenarios are shown below:

- no early coolant injection,
- vessel failure,
- early containment failure,
- early release to the suppression pool,
- drywell sprays active,
- later release bypasses the suppression pool, and
- poor retention in the reactor building.

Radionuclide Release Results for PDS-4 (L-LOCA sequence)

The results of the CET evaluation for PDS-4 are summarized in Table 4.7-4. The two dominant accident sequences comprised a significant of the total sequence frequency. The sequences are similar to the dominant sequences found for PDS-1 and PDS-2. Characteristics common to both sequences are:

- vessel depressurized,
- no vessel failure or early containment failure,
- early and late coolant injection,
- active drywell sprays,
- late suppression pool decontamination,
- good aerosol retention in the primary system and containment, and
- late venting of containment.

The source terms calculated for these sequences are very low.

A small fraction of the sequences for PDS-4 were found to satisfy the screening criteria of ≥ 90% noble gas release and ≥ 1% iodine release. The four major contributors to this subgroup of sequences are shown in the Table 4.7-4. The four sequences have the following common characteristics:

- no early coolant injection,
- reactor vessel failure,
- early containment failure,
- active drywell sprays,
- late coolant injection,
- late suppression pool decontamination,
- good aerosol retention in the primary system and containment, and
- poor aerosol retention in the reactor building.

The sequences differ from one another in whether there is early suppression pool bypass and whether the ex-vessel core debris is coolable. The probability of early containment failure was found to be less than 0.01. Ex-vessel core debris was uncoolable in very few sequences.

Radionuclide Release Results for PDS-5 (TQUX sequence)

The results of the CET evaluation for PDS-5 are summarized in Table 4.7-5. As shown in the table, the results from PDS-5 (a TQUX sequence) are nearly identical to those from PDS-2 (a TQUV sequence). The important sequences are the same, and their sequence frequencies are extremely close. Consequently, the discussion of PDS-2 is also applicable to PDS-5.

4.7.3 Sensitivity Evaluation

A sensitivity evaluation for the HCGS IPE has not yet been completed. This study will focus on IPE parameters and probabilities that are highly uncertain and that have the potential to significantly effect either the potential for containment failure or the magnitude of the release.

4.7.4 Summary of Results

Containment event trees have been evaluated for the five of the eight plant damage states identified in the Level 1 analysis. An algorithm for evaluating radionuclide release to the environment was built into the CET structure, so source term results were generated for each accident progression sequence.

The highest probability of a large radionuclide release was calculated for a short-term station blackout (PDS-3). The high probability for a large release resulted from the significant probability of vessel failure and early containment failure in these sequences, and the probability of an ex-vessel core-concrete interaction given vessel failure occurred.

The three transient sequences (PDS-1, -2, and -5) produced similar results. For all three transients, the probabilities of early containment failure and moderate to high radionuclide release were small. The probability of an ex-vessel debris-concrete interaction was also small.

The plant damage state representing a large-break LOCA (PDS-4) produced the smallest probability of an early containment failure or a moderate to high radionuclide release. Only a small fraction (< 1%) of the sequences lead to either an early containment failure or a significant radionuclide release. Also, few of the sequences lead to ex-vessel debris-concrete interactions.

The potentially significant containment failure mechanism for all five plant damage states was drywell shell meltthrough.

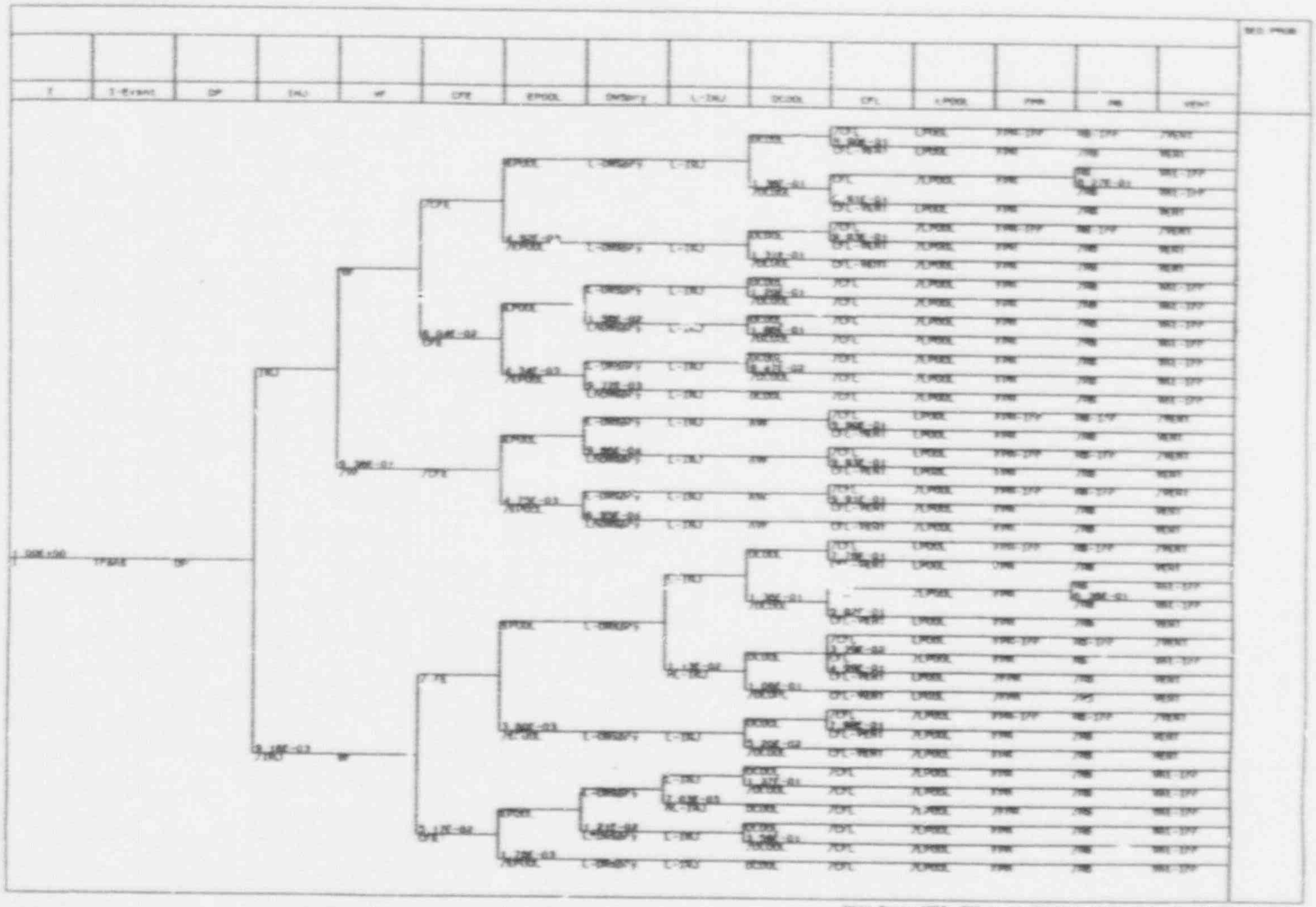

Figure 4.7-1 Containment Event Tree for PDS-1

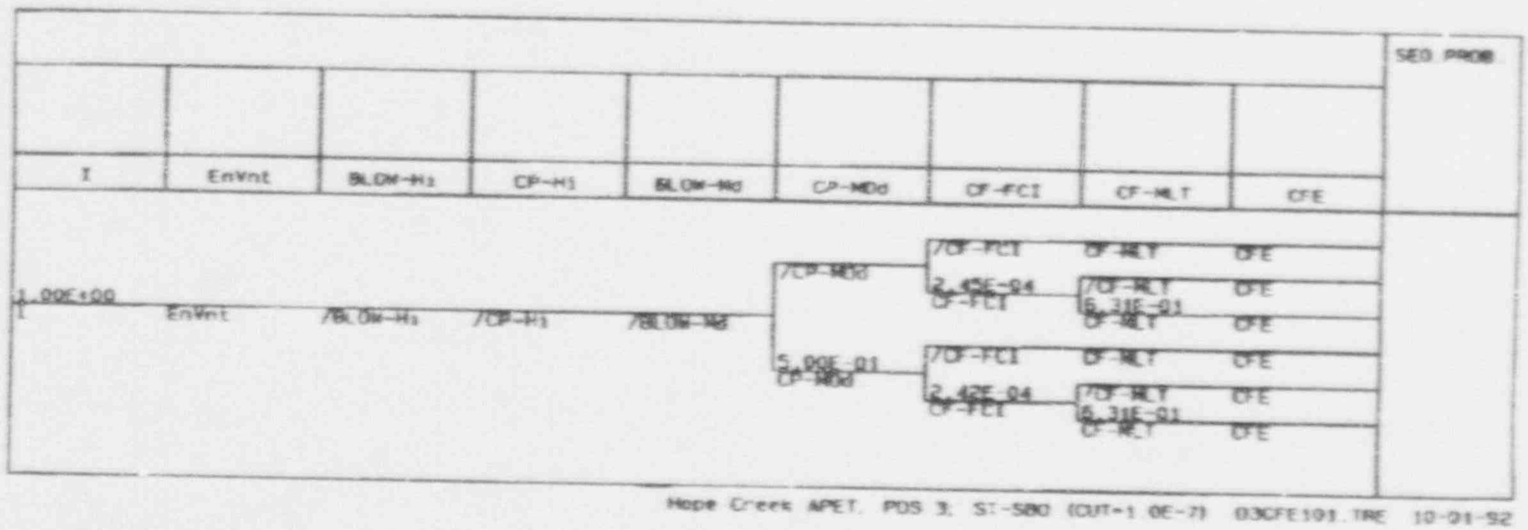

Figure 4.7-2 Early Containment Failure Subtree for PDS-I

4.8 References

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Appendix A

Quantification of CET Base Events for the HCGS IPE

Key to Quantification Classes:

1 = Generic Events

2 = Plant Specific Events - Determined from computer code calculations

3 = Plant Specific Events - Determined using other analysis methods

4 = Events Dependent on Human Reliability

5 = Events Dependent on System Reliability

Basic Event Name	Class	Description	Values
AC-PWR-EARLY	4	AC POWER IS NOT RESTORED EARLY: Probability that AC Power is not restored early given that a station blackout has occurred.	0.1-0.6
ALT-INJ	4	Human error failure to align alternate injection systems (condensate or service water) during core damage given that at least one system is operable (i.e., power available and functional).	0.3-0.6
BRN-LK-1	2	H_2 BURNS IN THE REACTOR BUILDING AFTER CONTAINMENT LEAK: Probability of a Reactor Building H_2 burn, given leakage into the Reactor Building, coolable debris is formed ex-vessel (i.e., no ex-vessel H_2 and in-vessel H_2 production is high.	0.1-1.0
BRN-LK-2	2	H_2 BURNS IN THE REACTOR BUILDING AFTER CONTAINMENT LEAK: Probability of a Reactor Building H_2 burn, given leakage into the Reactor Building and ex-vessel debris is not coolable (i.e., H_2 source is substantial).	0.5-1.0
BRN-RPT1	2	H_2 BURNS IN REACTOR BUILDING AFTER CONTAINMENT RUPTURE: Probability of a Reactor Building H_2 burn, given that there is no reactor building bypass, coolable debris is formed ex-vessel, in-vessel H_2 production is high, and containment rupture occurs.	0.8-1.0
BRN-RPT2	2	H_2 BURNS IN REACTOR BUILDING AFTER CONTAINMENT RUPTURE: Probability of a Reactor Building H_2 burn, given that there is no reactor building bypass, ex-vessel debris is not coolable, and containment rupture occurs.	0.8-1

Basic Event Name	Class	Description	Values
CD --> DP - 1	4	OPERATOR FAILS TO DEPRESSURIZE AFTER CORE DAMAGE: Probability that the operator fails to depressurize the RPV after core damage, given that the SRVs are operable in relief mode and DC Power is available (i.e., prior error of omission).	0.3-1.0
CD --> DP - 2	2	FAILURE TO DEPRESSURIZE AFTER CORE DAMAGE: Operator depressurizes early but containment pressure recloses SRVs prior to core damage. Containment is successfully vented (Containment venting is irrelevant for CD -->DP - 1). This event reflects the failure of the S/RVs to reopen in sufficient time to prevent core damage.	0-1.0
CD --> DP - 3	4	OPERATOR FAILURE TO DEPRESSURIZE AFTER CORE DAMAGE: Probability that the operator fails to depressurize after core damage given that the SRVs are operable in relief mode but DC Power was previously unavailable and AC Power is then restored.	0-1.0
CD --> DP - 4	2, (4)	FAILURE TO DEPRESSURIZE AFTER CORE DAMAGE: Same as CD --> DP - 3, except containment pressure recloses SRVs and containment is successfully vented.	0-1.0
CM>26%	2,4	GREATER THAN 26% CORE MELT: Probability that >26% of the core melts prior to low-pressure injection recovery.	0.1-1.0
COOL-1	3	DEBRIS NOT COOLED: Probability that debris is not cooled given that vessel breach occurs, water is present on the drywell floor at vessel breach due to either spray operation or recovery of injection prior to vessel breach, debris is not dispersed at vessel failure, and the core does not collapse en-masse.	0.1-0.3

Basic Event Name	Class	Description	Values
COOL-2	3	DEBRIS NOT COOLED: Same as COOL-1, except core collapses en-masse (SLUMPS).	0.6- 0.9
COOL-3	3	DEBRIS NOT COOLED: Same as COOL-1, except that debris is dispersed at vessel failure.	0.1- 0.2
COOL-4	3	DEBRIS NOT COOLED: Probability that debris is not cooled given that vessel breach occurs, there is no water on drywell floor at vessel breach, but water addition to the containment is restored following vessel failure. Debris is not dispersed at vessel failure and the core does not collapse en-masse.	0.5- 0.9
COOL-5	3	DEBRIS NOT COOLED: Same as COOL-4, except core collapses en-masse (SLUMPS).	0.5- 0.99
COOL-6	3	DEBRIS NOT COOLED: Same as COOL-4, except debris is dispersed at vessel failure.	0.1- 1.0
COOL-7	3	DEBRIS NOT COOLED: Probability that debris is not cooled given that the debris is dispersed at vessel breach, but that no water is being added to the drywell.	0.7- 1.0
CRD-FLOW	4	HUMAN ERROR FAILURE TO RESTORE CRD: Probability that the operator fails to provide CRD flow to the vessel, given that AC Power is restored, and control rod drive pumps are operable.	0.1- 0.5
DEPOSIT1	2	FISSION PRODUCT RETENTION IS LOW IN REACTOR BUILDING: Probability that natural deposition does not occur given that containment failure results in leakage that does not bypass reactor building, no hydrogen combustion occurs in the building and standby gas treatment fails.	0.01- 0.1

Basic Event Name	Class	Description	Values
DEPOSIT2	2	FISSION PRODUCT RETENTION IS LOW IN REACTOR BUILDING: Same as DEPOSIT1, except containment rupture occurs.	0.-1.0
ECCS-FLOW	4	ECCS FLOW NOT RECOVERED PRIOR TO VESSEL FAILURE: Probability that ECCS flow does not occur given that AC Power is restored prior to vessel breach, the vessel is depressurized, and low pressure ECCS is available.	0.0-0.1
EX-FCI-1	1	EX-VESSEL FUEL COOLANT INTERACTION: Probability that ex-vessel fuel-coolant interaction occurs given that either the core does not collapse en-masse or core collapse induces an in-vessel FCI, in-vessel fuel coolant interaction is prevented, vessel breach occurs, and there is water on the drywell floor.	0.1-0.5
EX-FCI-2	1,2	EX-VESSEL FUEL COOLANT INTERACTION: Probability that ex-vessel fuel coolant interaction occurs given that the core collapses en-masse (SLUMP), in-vessel fuel coolant interaction is prevented, vessel breach occurs, and there is water on the drywell floor.	0.5-0.9
FCI-CF-1	1	FCI INDUCED CONTAINMENT FAILURE: Probability of an FCI-induced containment failure given that an ex-vessel FCI occurs that does not involve a large mass of molten material (i.e., SLUMP does not occur or an in-vessel FCI occurred following SLUMP).	0.001-0.05
FCI-CF-2	1	FCI INDUCED CONTAINMENT FAILURE: Same as FCI-CF-1, except a large mass of molten material is involved.	0.001-0.05

Basic Event Name	Class	Description	Values
HI-RISE1	1,2	HIGH PRESSURE RISE IN CONTAINMENT AT VESSEL BREACH: Probability that the containment pressure rise is high given that the reactor vessel is initially at high-pressure, vessel rupture is prevented, but vessel breach occurs. High-pressure melt ejection occurs with water on the drywell floor, but little debris is liquid (i.e., no SLUMP). High pressure rise implies high probability of containment failure given moderate containment pressure (CP-MED) at vessel breach (i.e., $\Delta P \geq 200$ psi).	0.0-0.1
HI-RISE2	1,2	HIGH PRESSURE RISE IN CONTAINMENT: Similar to HI-RISE1 except that core SLUMP occurs. This implies that liquid debris is ejected under pressure into water on the drywell floor.	0.1-0.5
HI-RISE3	1,2	HIGH PRESSURE RISE IN CONTAINMENT: Same as HI-RISE2, except the water level on the drywell floor is insufficient to lead to significant water entrainment with debris. This event thus considers the potential for direct containment heating leading to containment failure.	0.05-0.4
HIH$_2$-CRD	1,2	HIGH HYDROGEN PRODUCTION GIVEN CRD FLOW: Probability that hydrogen production is high given that CRD flow occurs during core damage and no other injection is restored.	0.2-0.3
HIH$_2$-HP	1,2	HIGH HYDROGEN PRODUCTION GIVEN HIGH PRESSURE: Probability that hydrogen production is high given that the vessel remains at high-pressure during core damage and there is no CRD flow.	0.1-0.3
HIH$_2$-INJ	1,2	HIGH HYDROGEN PRODUCTION GIVEN INJECTION RECOVERY: Probability that hydrogen production is high given that injection is restored during core damage.	0.1-1.0

Basic Event Name	Class	Description	Values
HIH_2-LP	1,2	HIGH HYDROGEN PRODUCTION GIVEN LOW PRESSURE: Probability that hydrogen production is high given that the vessel is depressurized during core damage. There is no CRD flow, and injection is not restored prior to vessel breach.	0.0-0.5
HPME-1	1	HIGH PRESSURE MELT EJECTION: Probability of high-pressure melt ejection given that the vessel is at high-pressure when vessel breach occurs and the core does not collapse en-masse.	0.05-0.2
HPME-2	1	HIGH PRESSURE MELT EJECTION: Same as HPME-1, except the core collapses en-masse (SLUMPS).	0.1-1.0
IN-FCI-2	1,2	IN-VESSEL FUEL COOLANT INTERACTION: Probability of an in-vessel FCI given that the core collapses en-masse (SLUMPS).	0.0-0.3
L-AC (AC-PWR-LATE)	5	LATE AC POWER NOT RESTORED GIVEN EVENT IS A STATION BLACKOUT: Probability that AC Power is not available late given that a station blackout has occurred and AC Power is not restored early.	0.1-0.6
LALT-FL1	4	OPERATOR FAILS TO PROVIDE ALTERNATE COOLING TO DEBRIS: Probability that the operator fails to provide flow from alternate systems after vessel failure, given that there was no station blackout, low-pressure ECCS is not available and alternate injection systems are available. Vessel was not pressurized prior to vessel breach, thus is the first opportunity to use alternate injection.	0.3-0.9

Basic Event Name	Class	Description	Values
LALT-FL2	4	OPERATOR FAILS TO PROVIDE ALTERNATE COOLING TO DEBRIS: Similar to LALT-FL1, except that the vessel was previously depressurized. Operators have previously failed to use alternate injection source as required.	0.5-1.0
LALT-FL3	4	OPERATOR FAILS TO PROVIDE ALTERNATE COOLING TO DEBRIS: Probability that the operator fails to provide flow from alternate systems, given a station blackout, but with AC Power restored late. ECCS is not available.	0.3-1.0
L-CRD-1 (CRD-L-1)	4	OPERATOR FAILS TO PROVIDE CRD FLOW TO DEBRIS: Probability that the operator fails to provide CRD flow to the debris given that there is not a station blackout and CRD pumps are operable. This implies that operators had previously failed to restore CRD flow as required.	0.5-1.0
L-CRD-2 (CRD-L-2)	4	OPERATOR FAILS TO PROVIDE CRD FLOW TO DEBRIS: Probability that the operator fails to provide CRD flow to the debris given a station blackout but with AC Power restored late. CRD pumps are operable.	0.5-1.0
L-DWF (DWF-L)	2	CONTAINMENT FAILS LATE IN DRYWELL: Probability that containment fails late in drywell given that the containment does not fail early and thermal failure is prevented late.	0.1-1.0
L-ECCS-1 (ECCS-L-1)	5	LATE ECCS FLOW NOT PROVIDED TO DEBRIS: Probability that ECCS flow is not provided to the debris late given that there is no station blackout and low-pressure ECCS is available. Vessel pressure was high prior to vessel breach and so ECCS was not previously operable.	0.0-0.5

Basic Event Name	Class	Description	Values
L-ECCS-2 (ECCS-L-2)	5	LATE ECCS FLOW NOT PROVIDED TO DEBRIS: Same as L-ECCS-1, except that the vessel was depressurized prior to vessel breach and ECCS failed to operate as required.	1.0
L-ECCS-3 (ECCS-L-3)	5	LATE ECCS FLOW NOT PROVIDED TO DEBRIS: Probability that ECCS flow is not provided to the debris late given a station blackout, but with the AC Power restored late. Low-pressure ECCS was not available previously.	0.0-0.5
L-SPRY-1 (SPRY-L-1)	4	OPERATOR FAILURE TO ACTUATE SPRAYS TO COOL DEBRIS: Probability that operator fails to initiate drywell sprays to cool core debris given that the vessel has breached and drywell sprays are available.	0.0-1.0
L-SPRY-2 (SPRY-L-1)	4	OPERATOR FAILURE TO ACTUATE SPRAYS TO COOL DEBRIS: Probability that operator fails to initiate drywell sprays to cool core debris after vessel failure given a station blackout with AC Power restored early and sprays available.	0.0-1.0
L-SPRY-3 (SPRY-L-1)	4	OPERATOR FAILURE TO ACTUATE SPRAYS TO COOL DEBRIS: Probability that operator fails to initiate drywell sprays to cool core debris given vessel failure, station blackout, and late restoration of AC Power. Spray system is available.	0.0-1.0
LK-1	2	CONTAINMENT FAILURE LATE RESULTS IN RUPTURE: Containment rupture is the consequence of early containment failure.	0.0
LK-2	2	CONTAINMENT FAILURE LATE RESULTS IN RUPTURE: Probability of containment rupture due to late over-pressure failure.	0.0-1.0

Basic Event Name	Class	Description	Values
LK-3	2	CONTAINMENT FAILURE LATE RESULTS IN RUPTURE: Probability of containment rupture due to late thermal failure.	0.0-1.0
MDH_2-CRD (H_2MD-CRD)	1,2	MODERATE HYDROGEN PRODUCTION GIVEN CRD FLOW: Probability that hydrogen production is moderate given that CRD flow occurs during core damage. No other injection is restored and H_2 production is not high.	0.5-0.9
MDH_2-HP (H_2MD-HP)	1,2	MODERATE HYDROGEN PRODUCTION GIVEN HIGH PRESSURE: Probability that hydrogen production is moderate given that vessel remains at high-pressure during core damage, there is no CRD flow, and H_2 production is not high.	0.5-0.9
MDH_2-D__ (H_2MD-INJ)	1,2	MODERATE HYDROGEN PRODUCTION GIVEN INJECTION RECOVERY: Probability that hydrogen production is moderate given that injection is restored during core damage and H_2 production is not high.	0.5-0.9
MDH_2-LP (H_2MD-LP)	1,2	MODERATE HYDROGEN PRODUCTION GIVEN LOW PRESSURE: Probability that hydrogen production is moderate given that the vessel is depressurized during core damage. There is no CRD flow, injection is not restored prior to vessel breach, and H_2 production is not high.	0.5-0.9
MEDRISE1	1,2	MODERATE PRESSURE RISE IN CONTAINMENT AT VESSEL BREACH: Probability that the containment pressure rise is moderate given that the reactor vessel is at high pressure. Vessel rupture is prevented and vessel breach occurs. The core may or may not collapse en-masse (i.e., SLUMP not relevant without melt ejection) and high-pressure melt ejection is precluded.	0.1-1.0

Basic Event Name	Class	Description	Values
MEDRISE2	1,2	MODERATE PRESSURE RISE IN CONTAINMENT AT VESSEL BREACH: Probability that the containment pressure rise is moderate given that the reactor vessel is at high pressure. Vessel rupture is prevented and vessel breach occurs. The core does not collapse en-masse (no SLUMP), high-pressure melt ejection occurs, but the resultant pressure rise is not high. Debris is ejected into water on the drywell floor.	0.1 1.0
MEDRISE3	1,2	MODERATE PRESSURE RISE IN CONTAINMENT AT VESSEL BREACH: Probability that the containment pressure rise is moderate given that the reactor vessel is at high pressure. Vessel rupture is prevented and vessel breach occurs. The core does not collapse en-masse (no SLUMP), high-pressure melt ejection occurs, and the drywell floor is not covered by water.	0.1-1.0
MEDRISE4	1,2	MODERATE PRESSURE RISE IN CONTAINMENT AT VESSEL BREACH: Same as MEDRISE2 except core collapses en-masse (SLUMP).	0.1-1.0
MEDRISE5	1,2	MODERATE PRESSURE RISE IN CONTAINMENT AT VESSEL BREACH: Same as MEDRISE3 except core collapses en-masse (SLUMP).	0.1-1.0
MELT-1	3	DRYWELL SHELL MELTS AT EMBEDMENT: Probability that the drywell shell melts at the embedment given that the core does not collapse en-masse (no SLUMP), vessel breach occurs, debris does not disperse, and water covers the drywell floor.	0.001-0.1
MELT-2	3	DRYWELL SHELL MELTS AT EMBEDMENT: Same as MELT-1 except water does not cover the drywell floor.	0.4-1.0

Basic Event Name	Class	Description	Values
MELT-3	3	DRYWELL SHELL MELTS AT EMBEDMENT: Probability that the drywell shell melts at the embedment given that the core does not collapse en-masse (no SLUMP), vessel breach occurs, and the debris disperses after vessel failure.	0.1-0.2
MELT-4	3	DRYWELL SHELL MELTS AT EMBEDMENT: Probability that the drywell shell melts at the embedment given that the core collapses en-masse (SLUMP), vessel breach occurs, high-pressure melt ejection is prevented, debris dispersal is prevented, and water covers the drywell floor.	0.001-0.5
MELT-5	3	DRYWELL SHELL MELTS AT EMBEDMENT: Same as MELT-4 except water does not cover the drywell floor.	0.7-1.0
MELT-6	3	DRYWELL SHELL MELTS AT EMBEDMENT: Probability that the drywell shell melts at the embedment given that the core collapses en-masse (SLUMP), vessel breach occurs, high-pressure melt ejection is prevented, but debris dispersal occurs.	0.1-1.0
MELT-7	3	DRYWELL SHELL MELTS AT EMBEDMENT: Same as MELT-3, but water covers drywell floor	0.001-0.1
MELT-8	3	DRYWELL SHELL MELTS AT EMBEDMENT: Same as MELT-6, but water covers drywell floor	0.001-0.5
POOL-1	2	SUPPRESSION POOL DRAINED: Probability that the suppression pool is drained given that containment fails late and not in the drywell.	0.5-1.0

Basic Event Name	Class	Description	Values
POOL-2	2	SUPPRESSION POOL DRAINED: Probability that the suppression pool is drained given that containment fails early and not in the drywell.	0.5-1.0
RUPTURE1	2	PRIMARY CONTAINMENT RUPTURES: Probability that the primary containment ruptures given that the failure is thermally induced in the drywell.	0.0-1.0
RUPTURE2	2	PRIMARY CONTAINMENT RUPTURES: Probability that the primary containment ruptures given that it is induced by over-pressure.	0.0-1.0
SLUMP	1	CORE HAS COLLAPSED EN-MASSE: Probability that the core collapses en-masse.	0.1-0.9
SO-DWVB	5	STUCK OPEN DRYWELL VACUUM BREAKERS: Probability that the drywell vacuum breakers are stuck open given challenge during core damage.	0.03-0.99
SO-TPVB	5	TAIL PIPE VACUUM BREAKERS FAIL OPEN: Probability that a tail pipe vacuum breaker is stuck open given that a LOCA does not occur and there are no SORVs.	0.001-0.5
SUMP	2	CONTAINMENT FAILURE INDUCED AT SUMP: Probability that containment failure induced at sump bottom given no other failure mode has occurred and the debris core is not coolable.	0.1-1.0
TEMP-1	2	THERMAL FAILURE OF CONTAINMENT OCCURS: Probability of thermal failure of containment given that water is supplied to the drywell floor and the debris is not cooled.	0.1-1.0

Basic Event Name	Class	Description	Values
TEMP-2	2	THERMAL FAILURE OF CONTAINMENT OCCURS: Probability of thermal failure of containment given that water is not supplied to the drywell floor and the debris is not cooled.	0.5-1.0
TEMP-3	2	THERMAL FAILURE OF CONTAINMENT OCCURS: Probability of thermal failure of containment given water is not supplied to the drywell floor and the debris is cooled.	0.1-1.0
VB-CM<26%	2	VESSEL FAILURE OCCURS WITH <26% CORE MELT AND INJECTION: Probability that vessel breach occurs given that low-pressure injection is recovered and less than 26% of the core has melted.	0.0-0.5
VB-CM>26%	2	VESSEL FAILURE OCCURS WITH >26% CORE MELT AND INJECTION: Probability that vessel breach occurs given that low-pressure injection is recovered, greater than 26% of the core melts and the core does not collapse en-masse (no SLUMP).	0.5-1.0
VSL-FCI-2	2	VESSEL RUPTURE WITH FCI AND SLUMP: Probability that the vessel ruptures given that the core does collapse en-masse (SLUMP) and in vessel FCI occurs.	0.001-0.05
L-VENT1	3	OPERATOR VENTS CONTAINMENT LATE: Probability that the operator vents containment late given that containment pressure is sufficient to require venting and AC Power is available.	0.0-0.9