Table 2.3-1

SUMMARY OF ACCESSIBILITY LIMITATIONS AS A FUNCTION OF SEVERE ACCIDENT CONDITIONS DUE TO RADIATION

Containment Failure Mode	Location				
	RAB	FHB El. 286' (and 261')	FHB El. 236'	FHB El. 216' N (and 236' N)	FHB El. 216' S
ISLOCA	X	X	X	A	X
SGTR	A/X	A/X	A	A	A/X
Containment Isolation Failure	X	X	A	A	X
Early Containment Failure	X	X	A	A	X
Late Containment Failure	A/X	A/X	A	A	A/X

LEGEND

A - Accessible

X - Means that for the indicated core damage and containment failure mode, the location is NOT accessible for personnel.

A/X - Accessible for a period of time, then inaccessible later in the accident sequence after containment failure. (See Section 2.4 for containment failure times as a function of accident type.)

Table 2.3-2

SUMMARY OF EQUIPMENT SURVIVABILITY AS A FUNCTION OF SEVERE ACCIDENT CONDITIONS

Containment Failure Mode	Locations with Potential Equipment Failures				
	RAB	FHB El. 286' (and 261')	FHB El. 236'	FHB El. 216' N (and 236' N)	FHB El. 216' S
ISLOCA	X	X	X	A	X
SGTR	A/X	A/X	A	A	A
Containment Isolation Failure	X	X	A	A	A
Early Containment Failure	X	X	A	A	A
Late Containment Failure	A/X	A/X	A	A	A

LEGEND

A - Pumps are considered to have survived the environment.

X - Means that for the indicated core damage and containment failure mode pumps in the location are NOT considered to survive the environment.

A/X - Pumps assumed to operate successfully before containment failure. (See Section 2.4 for containment failure times as a function of accident type.)

Table 2.3-3

EX-BUILDING DOSE SUMMARY

Sequence	On-Site Work (WTB, cooling tower basin, intake structure) Work will result in < 25 rem dose for 2 hour exposure time	Entrances to Power Block (plant entrance) Entry will result in < 25 rem dose for 15 minute exposure time
ISLOCA	A	A[1]
Containment Isolation Failure	A	A
Early Containment Failure	A	A[1]
Late Containment Failure	A	A[1] (Note (1))
SGTR	A	A

A Exposure under these conditions is acceptable within a 2 day time period.

A[1] Exposure under these conditions is acceptable within a 2 day period for upwind entry locations. Information on prevailing winds and plant building entry make it highly likely for personnel access.

Note (1): Access is also available prior to containment failure which occurs at 38 to 90 hours.

000771

Survivability

Many motor operated pumps are located in the RAB and the FHB and may be exposed to various degrees of harsh conditions, depending on their spatial relationship to the location of the primary containment failure. These pumps may fail to operate if an adequate room environment is not maintained.

An increase in the ambient temperature, due to loss of room cooling or due to primary containment failure, is the main concern. A conservative approach could be taken by assuming that components fail if the room temperature exceeds the manufacturer recommended value. However, in the case of pump motors, the failure is more a function of time at temperature rather than simply exceeding a temperature limit. Therefore, continued pump operation may be likely even for temperatures exceeding manufacturer specified warranty values. The pump motors may also fail due to moisture intrusion. The humid environment in the pump areas following primary containment failure would likely result in moisture intrusion in the CCW and ESW booster pump motors that could potentially result in shorted or grounded circuits. The CCW and ESW booster pumps are not credited with operability following containment failure scenarios.

The 6.9 kV switchgear located in isolated compartments in the RAB are protected from harsh environment and will not fail during the course of the postulated severe accidents. This is based on personal communication from Walter Schade (CP&L) to Bruce Morgen (CP&L).

2.4 CONTAINMENT FAILURE MODES AND CRITICAL TIMES

The containment failure modes or bypass modes directly influence the ability to maintain the SFPs in a configuration with adequate cooling. This is because the modes of containment failure may cause any of the following:

- Adverse environmental conditions in the FHB that could cause failure of the SFPCCS and cause a loss of cooling and / or makeup to the SFPs;
- Adverse environmental conditions in the Reactor Auxiliary Building that could cause failure of one or more of the systems required to support cooling and/or makeup to the SFPs (e.g., CCW or AC power); or,
- Radionuclide release or high temperature steam release to the RAB or the FHB that could limit the ability for local manual actions to provide makeup to the SFPs given that water makeup may be required.

Figure 2.4-0 Compares the approximate timing associated with severe accidents and the postulated containment failure modes.

Table 2.4-1 qualitatively summarizes the impacts on building environment associated with the various severe accident containment failure modes. These insights are based on MAAP deterministic calculations for SHNPP provided in Appendix E. In addition to the containment failure modes following a severe accident, other effects associated with the Postulated Sequence may limit access by personnel. The principal additional effects identified here are: 1) the potential for SFP boiling; 2) security system failures; and, 3) potential structural failures of other buildings (e.g., hatches).

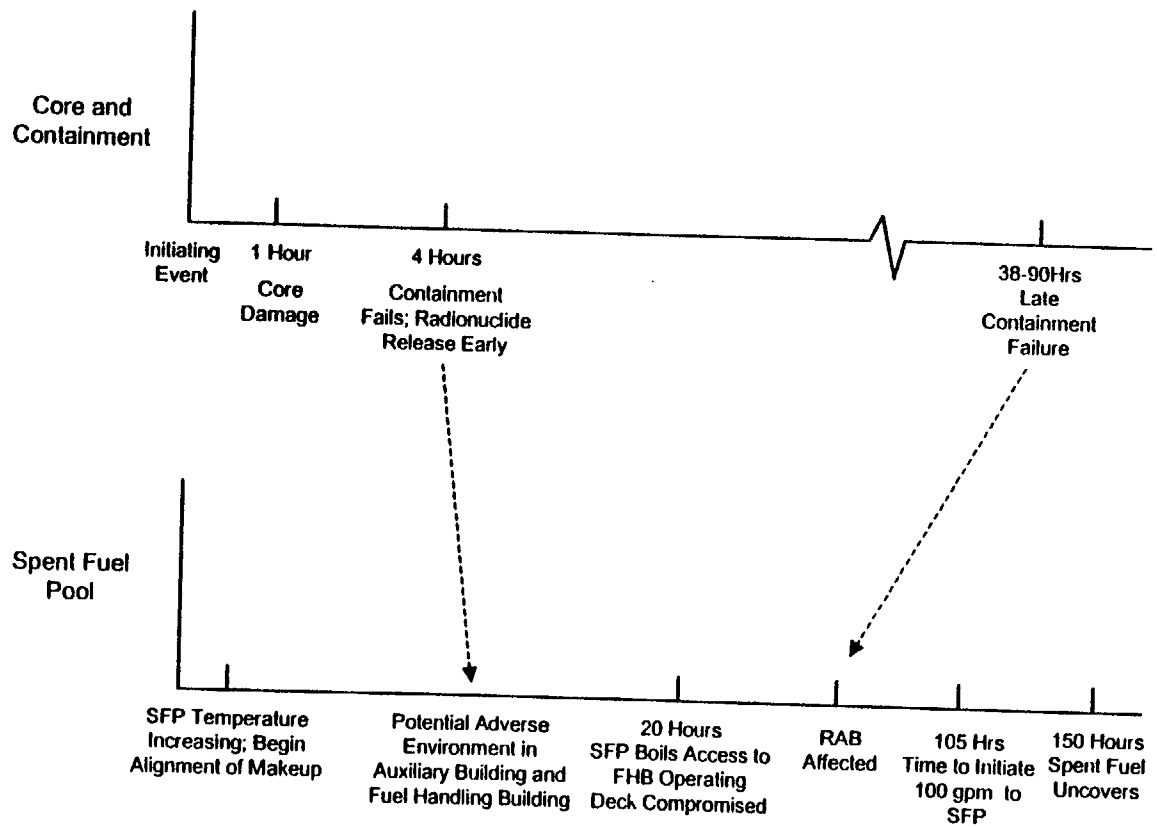


Figure 2.4-0 Comparison of Critical Times Associated with: (a) Core Damage plus Early and Late Containment Failures; and (b) Spent Fuel Pool Evaluation

The timing of containment failure or bypass also influences the operating crew and the TSC ability to provide effective mitigation. These times can be broken down into the following containment failure or bypass cases which will each be discussed in the following subsections:

- Early Containment Failure
- Containment Bypass (including SGTR)
- Containment Isolation Failure
- Late Containment Failure
- Very Late Containment Failure (subsumed within the late containment failure)

2.4.1 Early Containment Failure

Early Containment Failures can be postulated to be energetic (e.g., hydrogen deflagration) and these failures could cause the environment in the RAB and FHB to be sufficiently adverse to prevent personnel access to the FHB above the 236' El. and to most of the RAB. In addition, CCW pump failure is ascribed to the severe conditions of the containment blowdown.

A typical time line for the significant effects associated with an early containment failure is shown in Figure 2.4-1. This figure shows that beyond the time of early containment failure (~3 hours), many of the locations for in-plant alignments of SFP makeup become unavailable.

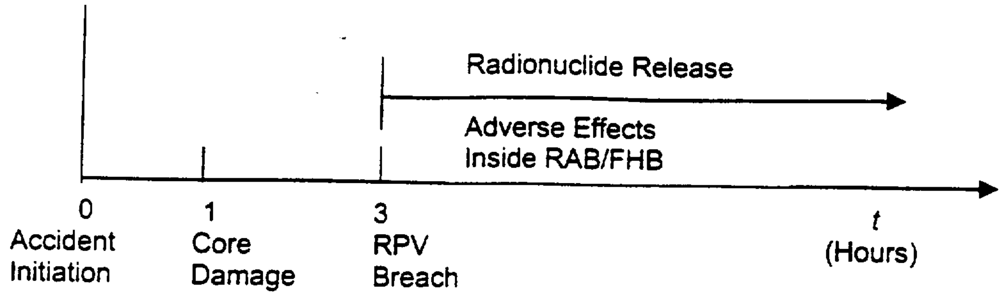


Figure 2.4-1 Typical Time Line for Effects Associated with Postulated Early Containment Failure

2.4.2 Containment Bypass

There are two distinct types of postulated containment bypass which have different potential impacts. These are:

- Steam Generator Tube Rupture (SGTR). See Figure 2.4-2 for the approximate time line.
- Interfacing System Loss of Coolant Accident (ISLOCA). See Figure 2.4-3 for the approximate time line.

2.4.2.1 SGTR

The SGTR could result in radionuclide release to the environment near time 0 to 1 hour. This could limit mobility of the operating crew about the site, but SFP cooling should remain available during this event. Subsequently, containment failure could occur late and lead to the adverse impact on SFP cooling and make-up.

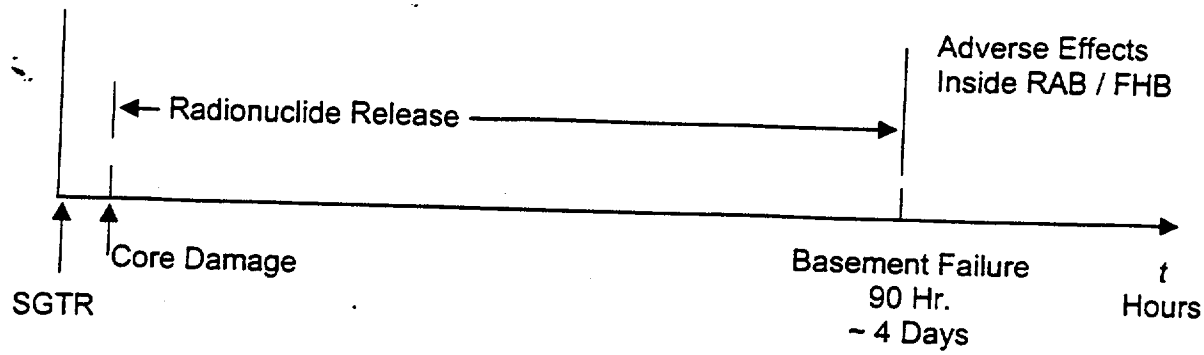
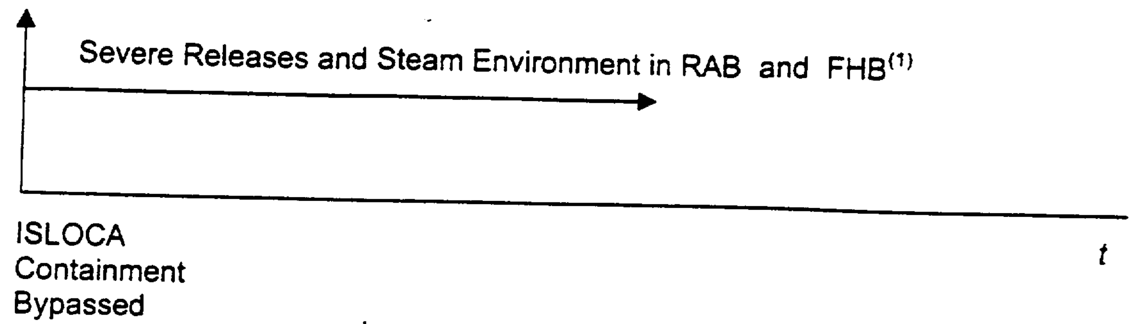


Figure 2.4-2 Approximate Time Line for SGTR

2.4.2.2 ISLOCA

The postulated ISLOCA event is a severe event for the RAB and FHB environments because it is a high energy RPV blowdown. The environment induced in the RAB and FHB would be the most severe of the accidents considered and there is little time available for operating crew local actions.

Figure 2.4-3 is an approximate time line for the ISLOCA scenario. This figure indicates that the radionuclides are released to the RAB and FHB near the time of core damage and containment bypass.



[(1)] Effects on specific locations in the RAB and FHB are discussed in Appendix E and summarized in Section 2.3.3.

Figure 2.4-3 Approximate ISLOCA Time Line

2.4.3 <u>Containment Isolation Failure</u>

The postulated containment isolation failure would result in radionuclide release relatively early for at-power cases. The containment would provide some, but limited, mitigation of radionuclide releases under isolation failure conditions. There are several causes of the isolation failure:

- Pre-existing personnel air lock
- RHR relief valves
- Reactor Shutdown with hatches open
- Seismic events with failure to close sump drain MOVs.

The isolation failure under shutdown conditions is considered to be similar to the at-power case. There is also a potential seismic induced containment isolation failure that causes release to the WPB. This is treated similar to an SGTR in terms of its effect on the timing of releases to the RAB.

Figure 2.4-4 is the approximate time line for containment bypass due to Personnel Access Door failures (at-power, during shutdown).

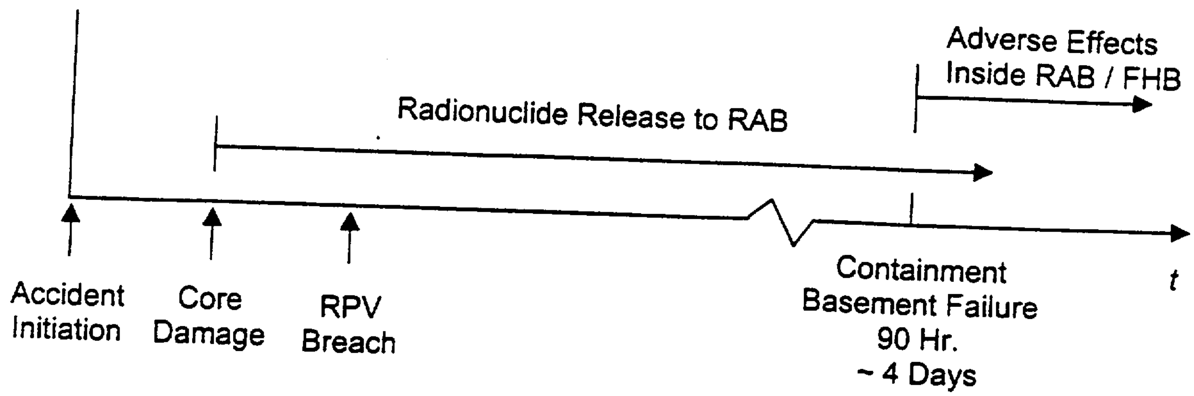


Figure 2.4-4 Approximate Time Line for the Effects Associated with the Containment Isolation Failure

2.4.4 Late Containment Failures

Late containment failures (which subsume the very late failures) are postulated to occur due to one of two potential failure modes. These are the following:

- Basemat melt-through, which would occur at approximately 90 hours (sometimes characterized as very late containment failure).
- Containment pressurization, which would occur due to the increased temperature from core debris and the pressurization from the steam generation and core concrete interaction at 38 hours (if the RWST inventory has been injected to containment)

The postulated late containment failures would provide a long period of time between the time that core damage occurs (approximately the time the TSC is operational) and the time of substantial radionuclide release to the site. This affords a long period of time (30-100hrs) for the TSC and on-site crew to establish that the SFP cooling is impaired (or could become impaired when containment failure occurs). Therefore, for late containment failures there can be two cases postulated:

Case A: TSC and crew seek to place all sources of risk in the most stable and safe condition prior to a late containment failure. This could include actions to place inventory makeup to the SFP.

Case B: A possible sensitivity to Case A where explicit prestaged equipment and guidance for its use is available in the TSC. This could take the form of placing fire hoses and/or quick connect hoses from the demineralized water system in the SFPs given a core damage event and awaiting the effect of imminent failure of containment on spent fuel cooling before initiating a predetermined flow rate to the SFP.

It could also include routing hoses to all pools or deflating the inflatable seals on the gates among pools to allow a single hose or injection point to communicate with all of the pools from the single injection point.

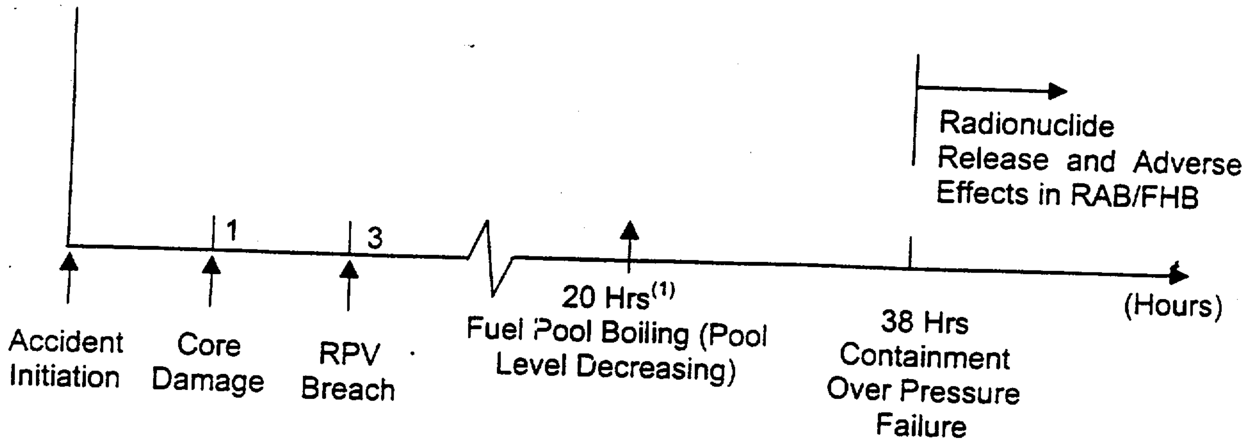


Figure 2.4-5 Approximate Time Line for Late Containment Failures

(1) This occurs only if the severe accident sequence has resulted in failure of the SFP cooling system or its supports. Otherwise, SFP Cooling remains available until adverse conditions following the containment failure causes SFP cooling to fail.

Table 2.4-1

SUMMARY OF IMPACTS ASSOCIATED WITH POSTULATED CONTAINMENT FAILURE MODES

Containment Failure Mode	Timing	Effect	Impact
ISLOCA Bypass	Early	Release of High Energy Steam and Radionuclides to the RAB	Immediate adverse environment introduced into RAB that could affect CCW and ESW booster pumps in RAB. Propagation to FHB occurs.
SGTR Bypass	Early	Release of High Energy Steam and Radionuclides to Environment May Later Cause Containment Failure into RAB	Immediate release of radionuclide to environment causing potential restricted mobility of Aux Operators to perform local actions.
Containment Cylinder Failure	Early	Release of Steam and Radionuclides to RAB with Probability of 0.75[1]	Immediate adverse environment introduced into RAB that could affect CCW and ESW booster pumps in RAB. Propagation to FHB occurs.
	Late	Release of Steam and Radionuclide to RAB with Probability of 0.75[1]	Adverse environment introduced into RAB that could affect CCW and ESW booster pumps in RAB. Propagation to FHB occurs. However, substantial time exists for operating crew action prior to containment failure.
	Very Late	Release of Steam and Radionuclide to RAB with Probability of 0.75[1]	Adverse environment introduced into RAB that could affect CCW and ESW booster pumps in FHB. Propagation to FHB occurs. However, substantial time exists for operating crew action prior to containment failure.

[1] Conditional probability that containment fails such that the release is into the RAB.

Table 2.4-1

SUMMARY OF IMPACTS ASSOCIATED WITH POSTULATED CONTAINMENT FAILURE MODES

Containment Failure Mode	Timing	Effect	Impact
Basemat Failure	Very Late	Release of Steam and Radionuclide to RAB with probability of 0.75	Adverse environment introduced into RAB that could affect CCW and ESW booster pumps in RAB. Propagation to FHB occurs. However, substantial time exists for operating crew action prior to containment failure.
Containment Isolation Failure	Early	Release of Steam and Radionuclides	
A. To RAB (Personnel Access Hatch)		A. Release into the RAB	Immediate adverse environment introduced into RAB that could affect CCW and ESW booster pumps in the RAB. Propagation of adverse condition to the FHB does not occur.
B. Sump Drains		B. Release into the Waste Processing Building	Release is confined to the WPB and potentially the RAB. The FHB will not be affected.
C. Shutdown condition with Access Hatch Open		C. Release into the RAB	Immediate adverse environment introduced into RAB that could affect CCW and ESW booster pumps in the RAB. Propagation of adverse conditions to the FHB does not occur.

2.4.5 Summary of Critical Times

As part of the accident sequence evaluation and the assessment times available, a summary of the critical times affecting human performances were developed. Table 2.4-1 includes some of the critical times that were used in the model. These may be conservative because they are based on bounding (worst case) heat load conditions in the SFPs.

Table 2.4-2

CRITICAL TIMES

Timing Characteristic	Approximate Time	Potential Effects of the Characteristic
Time to SFP boiling for the limiting SFP	~ 20 hours	SFP boiling may create adverse conditions on the operating deck of the FHB which could in turn limit accessibility to the FHB for operator actions without protective clothing.
Time at which 100 gpm injection to the SFP may be inadequate to fill the SFP and spill over the gates to provide makeup to the other SFP prior to spent fuel being uncovered	~ 4 days	This time sets the upper limit on when actions can be effectively taken to begin at least 100 gpm injection to a single SFP.
Time at which the limiting pool with limiting heat load would have spent fuel initially uncovered.	~ 7 days	This time is only for reference; it is not used in the analysis. Radiation on the FHB operating deck would be high and there would be increasing concern for radionuclide release if level continues to decrease. However, radiation release from the spent fuel would require additional evaporation well below this point <u>and</u> would require an exothermic reaction.

2.5 SCOPE, KEY ASSUMPTIONS, AND GROUNDRULES

This section provides a summary of some of the key assumptions and groundrules used in the assessment of SFP cooling given a postulated severe accident.

2.5.1 Success Criteria

Time Available for Spent Fuel Pool Cooling

The time available for passive SFP cooling before some active method could be required to maintain the fuel cool is a function of a number of variables:

- Size of the pool
- Decay heat of the fuel in the pool
- SFP cooling heat removal rate
- Water makeup flow rate

Time to boil for SFPs A, B, C and D is required. All four pools are co-located in the FHB. Access to the local areas for operator intervention to establish SFP makeup can be precluded by adverse environments created by the most limiting pool conditions.

Because recently removed spent fuel can be placed in SFPs A and B, they will generally have the highest heat load and therefore the shortest time to boil in the event of a loss of SFP cooling. Estimates vary from cycle to cycle, but ESR 00-000046, Rev. 0 indicates the SFP Analysis for RFO-09 and Cycle 10 to have heat loads of 15 - 36 MBTU/Hr.

The mitigation measures associated with preserving the adequate cooling of the Spent Fuel consist of the following:

- Maintain water above the fuel and cool the water to prevent boil away of the water.

- Supply make up water to the spent fuel pools to replace any water lost due to boiling or evaporative cooling.

The probabilistic model has been structured in a realistic manner. In addition, the success criteria for the model is also based on a realistic assessment with the following exceptions:

- SFPs C and D are the focus of the evaluation. However, SFPs A and B may lose water inventory prior to SFPs C and D under certain postulated severe accidents. The consequences of loss of water inventory in pools A and B could in turn adversely impact both access and further prevention actions related to pools C and D. Therefore, the success criteria have been structured to require cooling or makeup to all 4 pools. From the standpoint of the Postulated Sequence, this assumption regarding success criteria introduces some slight conservatism.

- The limiting heat load to the SFP is generally that in pools A and B. This is where the fuel with the highest decay heat levels is generally present. For example, consider the following:

Pools	Time to reach boiling temperature	Additional time for water level to reach top of racks	Total time	Makeup required to offset boiling
A and B (Beginning of cycle)	20.57 hours	7.21 days	8.07 days	53.70 gpm
A and B (End of cycle)	38.67 hours	13.56 days	15.17 days	28.57 gpm
C and D (1 MBTU/hr heat load)	384.66 hours	99.99 days	116.02 days	2.15 gpm
C and D (15.6 MBTU/hr heat load)	34.42 hours	8.80 days	10.23 days	33.64 gpm

The limiting heat load is predicated on the full core offload case into pool A. This situation, however, exists for only short periods of time each fuel cycle. Nevertheless, the analysis considered the limiting heat load in pool A as always present.

- Makeup to the SFPs was assessed to be aligned to only one pool. This requires sufficient makeup volume and flow rate to overflow the pool gates and spill into the transfer canals and the other pools to maintain adequate inventory in all pools.

  This is a conservative assumption but is judged not to significantly bias the resulting assessment, i.e., the analysis remains realistic.

- Heat load in SFP, C and D for the current license amendment is limited to 1 MBTU/Hr. However, it is noted that the primary calculations performed in this analysis are based on the long term decay heat load of 15.6 MBTU/Hr. Therefore, the principal cases that have been performed here are done with the maximum anticipated heat load in pools C and D. This is manifested in the probabilistic evaluation in calculating the time available to initiate SFP makeup to preserve the C and D SFP water inventory above the spent fuel.

Effect of Spent Fuel Pool Boiling

With the SFPCCS operating effectively, the water in the SFP has low contamination. Boiling of the SFP is calculated to not create an environment that would preclude accessibility except to the FHB operating deck (286' EL.). Under boiling conditions (or near boiling conditions), the temperature in the 286' EL. is calculated to exceed 190°F. This calculation was performed without FHB ventilation operating.

CP&L has extensive fire brigade training. The results of this training and associated data indicates that entry into an environment of ~ 190°F (FHB operating deck with SFP boiling) can be performed by personnel equipped with available protective gear. This allows access of personnel to the FHB operating deck between the time of SFP initial boiling and the time at which the SFP water level is close to top of the spent fuel (i.e., within approximately 3 ft). This latter time is approximately 5 to 6 days under the highest assumed SFP heat loads.

Limited personnel access under these conditions is possible and is credited for the FHB 286' El. under SFP boiling conditions. However, no credit for local actions beyond 4 days (96 hours) is included.

Makeup Success Criteria

Makeup is adequate if it can fill a pool, overflow the gates and provide flow to adjacent pools via the transfer canal before fuel uncovery in the pool farthest from the injection point. The flow rate required to satisfy this is approximately 75 to 100 gpm.

2.5.2 Mission Time

The mission time for operation of makeup system is chosen as 24 hours. This choice is the same as that used in typical at-power PSAs. The mission time is presumed to result in sufficient time available to make arrangements for alternate system operation if necessary. The mission time associated with various accidents is divided into two categories:

- Degraded core events recovered in-vessel or without containment failure: The mission time investigated in the PSA and in the SFP cooling analysis is 24 hours.
- Degraded core events that produce adverse conditions outside containment may create a continuing challenge to the SFP. A time of 7 days is used as a reasonable time to expect that offsite resources can gain access to the site to install temporary equipment for the purposes of continued spent fuel cooling or makeup. To make SFP cooling last for 7 days, 1 day worth of makeup is required, i.e., approximately 66,000 gal. However, all sources used for success in the model have access to substantially more volume (> 400,000 gal).

2.5.3 Maintenance Unavailability

The purification pumps to be installed for use with SFPs C and D have been identified by CP&L to be operated continuously (i.e., one of the 2 clean-up loops will be aligned to pools C and D with a high availability). This affects the alignment of the demineralized water as a SFP makeup source in response to an accident. CP&L provided an estimate for the unavailability due to maintenance of 5.5E-3 for each loop, based on CP&L judgements of less than 48 hours of maintenance per year requiring a loop out of service (OOS). A value of 1E-2 is used in the model as a bounding assumption.

The Unit 1 purification pumps used in conjunction with SFPs A and B are operated in the same way except for the following:

- 1 week before a refueling they are aligned to the RWST to clean up the RWST
- They are operated during the shutdown to the cavity when the cavity is flooded
- As above, 48 hours/yr can be assumed for maintenance (72 Hrs/18 month cycle)

These facts lead to the following unavailability for the Unit 1 purification loops for demineralized water injection via 1SF 201:

At-Power: $$\frac{168 \text{ Hrs} \quad + \quad 72 \text{ hours}}{13{,}140 \text{ Hrs per cycle}} = \frac{240}{13{,}140} = .0183$$

Shutdown: 1.0

2.5.4 Adverse Environmental Impacts

There are a number of adverse environmental impacts that may result from the postulated degraded core events. These impacts include the following:

High Temperature/Steam: The release of high temperature fluid from the primary system due to containment failure or bypass, e.g., an ISLOCA, can result in a steam environment, high temperatures, high local pressures, and high radiations. The impacts of these adverse conditions affect both: (1) equipment such as Motor Control Centers (MCCs), switchgear, instrumentation, and motors; and, (2) access to areas for local actions of recovery or repair.

The evaluation of the consequences of containment failure has involved the modeling of the open spaces in the RAB and FHB. Enclosed and protected compartments such as the Train A and B switchgear rooms on the RAB 286'El. are not modeled. The adverse environment in the RAB is not judged to affect the enclosed compartments containing the Train A and B switchgear. As such, the preservation of AC power is included in the model unless other MCCs or switchgear are adversely impacted.

Radiation: The discharge of flow from the primary system or containment can cause radiation to migrate to local areas that would severely limit local manual actions at least temporarily.

Hydrogen: The discharge of hydrogen from containment can lead to the collection of hydrogen in local areas in combination with sufficient oxygen and an ignition source to cause a hydrogen burn or deflagration. Such events can cause damage to equipment in the local areas.

Radiation Shine: The containment intact during a degraded core accident will collect radionuclide releases in the containment atmosphere. Two principal cases are of interest:

- With containment sprays
- Without containment sprays

The radiation shine may be sufficient to limit any extensive local actions in adjacent areas. Simple actions are not judged to be substantially affected.

2.5.5 Structural Analysis

The structural analysis has a number of important interfaces with the accident progression analysis. These interfaces include:

- Factoring in the containment failure modes and failure locations as they may affect the ability to successfully maintain adequate cooling of the SFP.
- Factoring in the SFP capability to withstand the postulated boiling condition that may arise as part of a loss of SFP cooling assessment.
- Factoring in the RAB failure modes that may direct adverse conditions to the FHB.

Containment Structural Analysis

The containment failure locations have been evaluated for postulated unmitigated core damage events. The identified failure modes (ranked from highest probability to lowest) are the following:

	Median Failure Pressure
• Containment Basemat Failure	153 psig
• Wall-Basemat Junction	205 psig
• Membrane of Containment Cylinder Wall	210 psig

These are translated into the probabilistic analysis such that the probability of containment failure by location would be as follows:

Location	Conditional Probability
• Containment Basemat Failure	0.9
• Wall-Basemat Junction	0.08
• Membrane of Containment Cylinder Wall	0.02

In addition, to the overpressure structural failure mechanisms identified in the PSA, there is also postulated a containment basemat melt-through due to core debris interaction with concrete.

The basemat melt-through failure could lead to adverse conditions in the RAB similar to that of an over pressure failure. This may be conservative, but current PSA analyses do not support alternative assumptions at this time.

The containment failure modes and their assessed conditional failure probabilities have been treated in a potentially conservative fashion. The dominant late and very late containment failure modes are either: 1) overpressure failure which is calculated to fail at the cylinder basemat juncture; or, 2) basemat melt-through for which a failure location is ill-defined. In addition, the RAB surrounds approximately three fourths (0.75) of the containment. This would imply that at least 25% of the time the containment failure would not affect the RAB or FHB. This factor has not been explicitly modeled in the evaluation because of computer code limitations. Therefore, there is a potential for overestimating the resulting impact on the SFPs due to severe accidents that fail containment.

Spent Fuel Pool Structural Analysis

The SFPs have been evaluated by CP&L relative to their structural capability to withstand boiling. CP&L [2-2] has concluded that the SFP structure is capable of withstanding these temperatures without inducing a SFP excessive leak or rupture causing the loss of inventory. This explicitly recognizes that the SFP concrete design temperature is 150°F and that CP&L evaluates as an "acceptable" abnormal condition the potential for a SFP to be at 212°F (ESR-000046-Rev. 0, PP. 3-3).

Reactor Auxiliary Building

The RAB failure modes have been identified to be into the FHB and the Waste Processing Building. This means containment failures or bypass events leading to releases into the RAB would also result in release propagation into the FHB for containment failures or ISLOCA events occurring from power.

Seismic Capability

It is noted that the Fire Protection System capability to provide SFP makeup may become more complicated under a seismic event. A seismic event may lead to the failure of the Fire Protection Pumps (i.e., they are not seismic). However, the piping is seismic. The SHNPP method of supplying fire protection water is through the use of the ESW pumps, which are seismically qualified through 2 manual cross connect valves located on 236'EL of RAB.

Section 3
SHNPP PSA STATUS AND QUALITY

There are several key characteristics of a PSA that can be used to determine whether the PSA is suitable for a given application.

Among these PSA characteristics are the following which are discussed for each of the potential event frequency contributors in the following subsections:

- Methodology
- PSA quantification
- Uncertainty attributes
- Degree of detail
- PSA Quality

The following provides a brief summary of the models and how they have been used and reviewed for the SHNPP SFP.

3.1 INTERNAL EVENTS

One effective approach to ensuring quality is an independent peer review [3-2] of the plant PSA. Industry PSA peer review methods (see NEI-00-02) [3-1]) can be used to help ensure appropriate scope, level of details, and the quality of the PSA. This section addresses the characteristics of the SHNPP PSA that are important in establishing the probabilistic risk inputs to the Risk-Informed process and discusses the findings of an independent peer review. [3-2]

The independent peer review found the SHNPP PSA is capable of quantifying core damage frequency (CDF) and large early release frequency (LERF) and reasonably reflects the as-built and as-operated plant. The SHNPP PSA is consistent with accepted PSA practices, in terms of the scope and level of detail for internal events.

An evaluation of the SHNPP PSA based on the specific application, assessment of the best estimate probability of the Postulated Sequence, indicated the following:

- The methodology used in the SHNPP PSA is robust and has a significant level of detail that is fully supportive of the proposed application.

- The SHNPP PSA quantification is quite detailed and the results are consistent with PWRs of similar designs.

- A formal uncertainty propagation has not been performed, but there are no SHNPP unique features that would indicate that there are substantive differences in the uncertainty quantification between the SHNPP PSA and other PWRs, such as described in NUREG-1150. Therefore, the specific application is not adversely impacted. Specific sensitivities were performed as part of this analysis.

The one area identified by the independent peer review of the SHNPP PSA for which additional information was suggested in order to provide a more realistic evaluation of the scenario postulated in the ASLB Order was the evaluation of the Interfacing System LOCA (ISLOCA). The ISLOCA analysis in the SHNPP PSA was found to be too conservative because:

- The failure modes included in the evaluation considered failures that are not physically meaningful.
- The pipe failure probability was unrealistically high given the plant-specific pipe characteristics.

The ISLOCA accident was also judged to be important in providing a best estimate of the Postulated Sequence. Therefore, the ISLOCA analysis was updated for this analysis to make the quantification consistent with the state of the technology and more realistic.

3.2 SEISMIC

On the basis of the IPEEE review, the NRC staff concluded that CP&L's IPEEE process was capable of identifying the most likely severe accidents and severe accident vulnerabilities and, therefore, that the SHNPP IPEEE met the intent of Generic Letter 88-20, Supplement 4.

The plant licensing seismic design basis is 0.15g Safe Shutdown Earthquake (SSE) using ground motion design spectra defined by Regulatory Guide 1.60. The plant is binned in the 0.3g focused-scope category in the IPEEE submittal and NUREG-1407.

The licensee used the EPRI methodology for Seismic Margins Assessment (SMA), and, therefore, no estimate of the seismic core damage frequency (CDF) was obtained. The licensee concluded that SHNPP has a plant level high-confidence-low-probability-of-failure (HCPLF) capacity of 0.3g, which is the peak ground acceleration associated with the review level earthquake (RLE).

Because the seismic margins assessment method was used, frequencies of seismic-induced accident sequences were not obtained. The components with the lowest HCLPF capacities were:

- Two RHR heat exchangers (HCLPF capacity of 0.29g)
- Four low voltage switchgears (HCLPF capacity of 0.35g)

The RLE earthquake has a peak ground acceleration (pga) of 0.3g, and consequently the components on the safe shutdown equipment list have HCLPF capacities meeting or exceeding this value. The licensee noted that the calculation of the two RHR heat exchangers is conservative, and that a more refined calculation would increase the

HCLPF capacity of the RHR heat exchangers above 0.3g. In any event, the HCLPF capacity of the RHR heat exchangers is essentially equal to the RLE pga.

Therefore, to support the ASLB required assessment, an approximate methodology was developed to quantify the core damage frequency (CDF) and potential for radionuclide release. This approximate methodology uses the results of the SHNPP seismic margins study and techniques derived from previous seismic PSAs to estimate the CDF and radionuclide release.

The seismic evaluation received an independent review from two senior ERIN PSA analysts [D.E. True and K.N. Fleming]. The results of that independent review indicate that the seismic evaluation is sufficient and adequate to provide the necessary insights to support the application to the ASLB Order.

3.3 FIRE

On the basis of the IPEEE review, the NRC staff concluded that CP&L's IPEEE process was capable of identifying the most likely severe accidents and severe accident vulnerabilities and, therefore, that the SHNPP IPEEE met the intent of Generic Letter 88-20, Supplement 4.

The SHNPP PSA was used directly to assess CDF and the frequency of radionuclide release for the dominant accident sequences.

The fire PSA results for the dominant accident sequences were included in the CAFTA PSA model for SHNPP. These sequences were used to calculate the impact requested in the ASLB Order due to potential fire-induced accident sequences. An independent review of this analysis indicates that the SHNPP application of the EPRI FIVE [3-5] methodology and the incorporation of the dominant fire contributors into the SFP analysis is adequate to support the PSA application to the ASLB Order.

3.4 OTHER EXTERNAL EVENTS

On the basis of the SHNPP IPEEE review, the NRC staff concluded that CP&Ls IPEEE process was capable of identifying the most likely severe accidents and severe accident vulnerabilities and, therefore, that the SHNPP IPEEE has met the intent of Generic Letter 88-20, Supplement 4.

No other external events contribute significantly to the event frequency contribution of severe accidents. Therefore, there is no quantitative measure of these negligible contributors.

3.5 SHUTDOWN

The CDF associated with shutdown has been developed from generic studies. A description of the development of the Shutdown CDF is provided in Section 4. The shutdown event frequency derived from generic studies [3-3] is believed conservative, but adequate for the purpose of demonstrating the limited impact of the results.

The shutdown evaluation received an independent review from two senior ERIN PSA analysts [D.E. True and K.N. Fleming]. The results of that independent review indicate that the shutdown evaluation is sufficient and adequate to provide the necessary insights to support the application to the ASLB Order.

3.6 SUMMARY

The methods used in formulating the response to the ASLB Order are summarized in Table 3-1. In addition, Table 3-1 specifies the method used to ensure that the inputs of the probabilistic analysis are adequate.

Table 3-1

SUMMARY OF APPROACHES USED TO ADDRESS ASLB ORDER AND THE METHODS USED TO ASSURE QUALITY OF THE RESULTS

Potential Contributors	Method	Review
Internal Events	PSA	NEI PSA Peer Review Process Checklists
Seismic	Approximate Method	Independent Review
Fire	PSA (IPEEE)	Independent Review
Other External Hazards	Screened	Independent Review
Shutdown	Approximate Method	Independent Review

The ERIN conclusion, based on independent review of the PSA models developed for SHNPP CDF and containment failure evaluations, is that the models are all adequate to support this PSA application in responding to the ASLB's question regarding the specific accident sequence as it affects the SHNPP spent fuel pools (see ASLB Order).

Section 4
SPENT FUEL POOL COOLING ANALYSIS

This section summarizes the ERIN analysis of the seven step postulated accident scenario set forth in the ASLB Order by examining each of the event frequencies of the potential initiating contributors as follows:

- Internal Initiating Events - Section 4.1
- Seismic Initiating Events - Section 4.2
- Fire Initiating Events - Section 4.3
- Shutdown Initiating Events - Section 4.4
- Other Initiating Events - Section 4.5

Figure 4.0-1 summarizes the accident sequences that are postulated to cause both core damage and containment failure or bypass.

4.1 INTERNAL EVENTS

4.1.1 Accident Sequence Development

The critical task for this analysis was to provide an effective method of identifying the important accident sequences that could result in challenging the SFP cooling or makeup capability to Spent Fuel Pools within the specificity of the seven postulated events as set forth in the ASLB Order. This section addresses the accident sequence development derived from the internal events Level 1 and Level 2 SHNPP PSA.

The approach for internal events was to take the results of the Level 1 and Level 2 SHNPP PSA in the form of individual cutsets and input these cutsets to the assessment of the SFP. Figure 4.1-1 summarizes the overall approach.

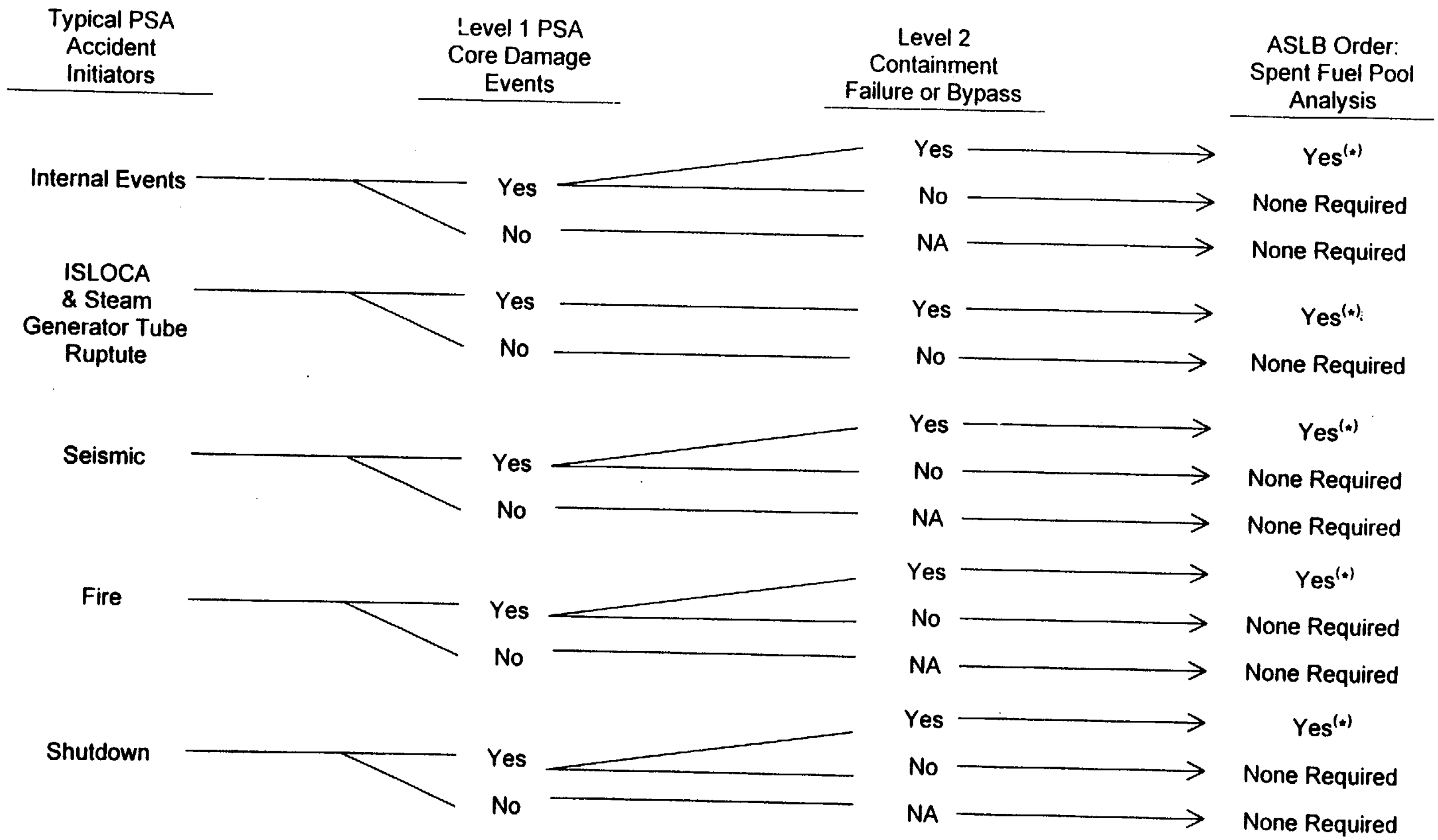
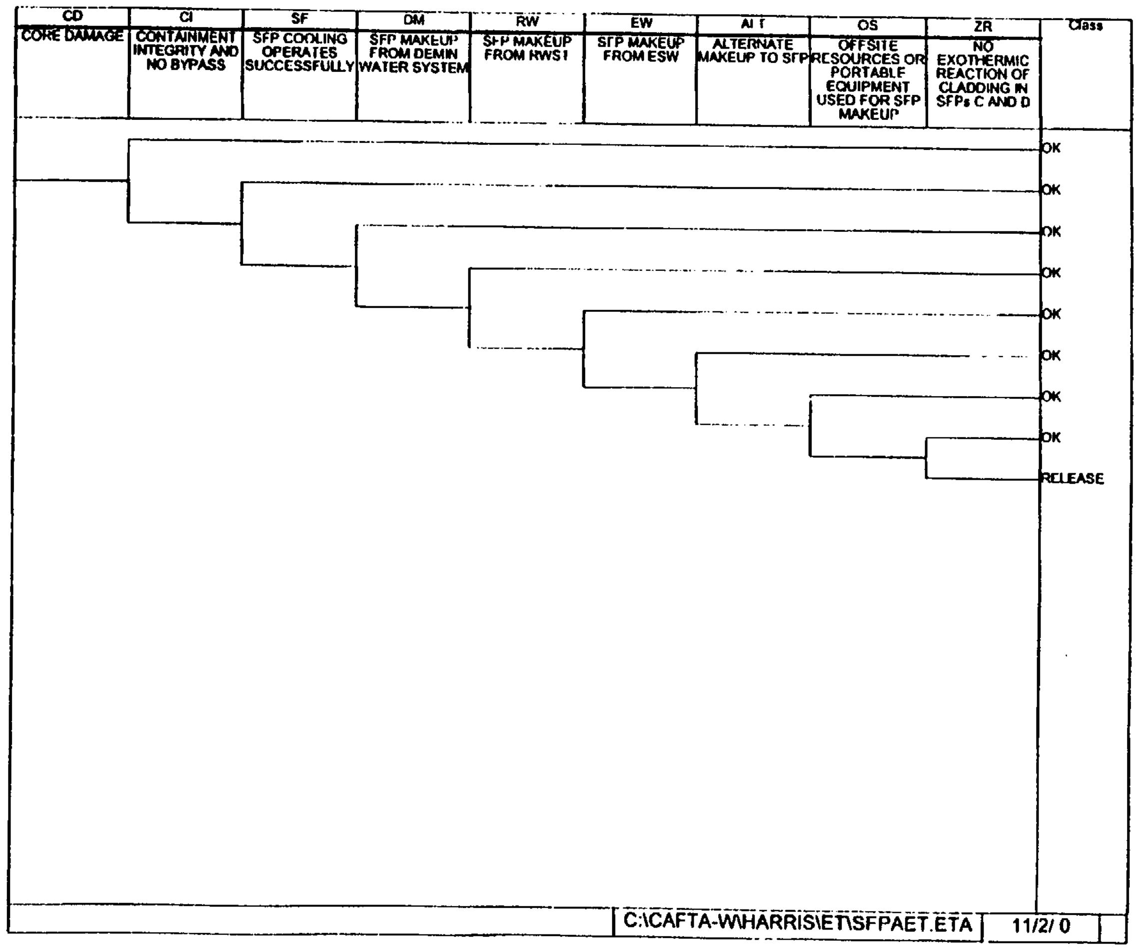


Figure 4.0-1 Summary of Analysis Performed in Support of the ASLB Order (Page 1 of 2)

(*) See Page 2 of 2 Figure 4.0-1

Figure 4.0-1 Summary of Analysis Performed in Support of the ASLB Order
(Page 2 of 2)



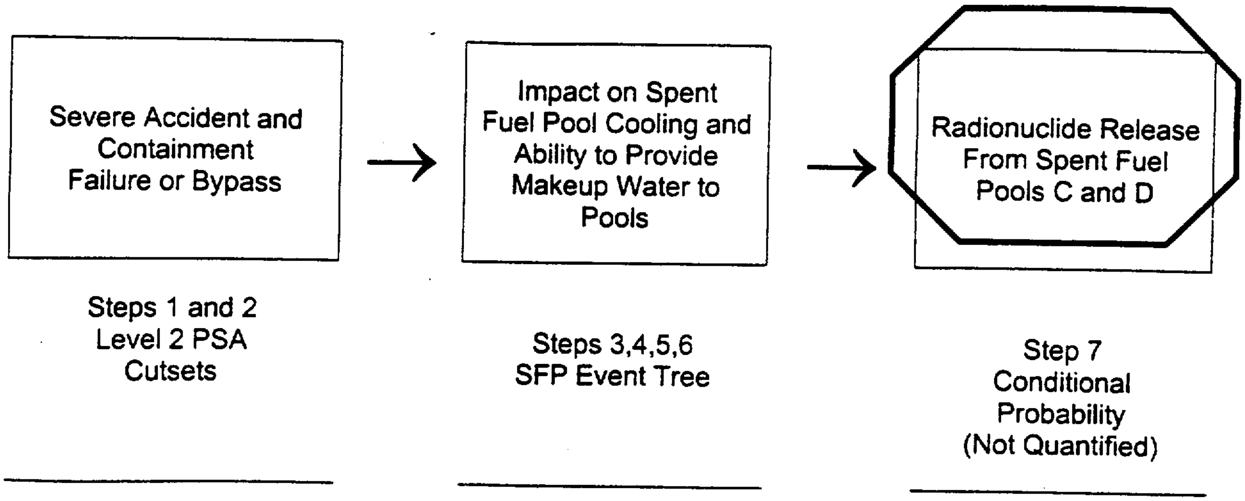


Figure 4.1-1 Internal Event Analysis Approach for Spent Fuel Pool Evaluation According to ASLB Order

The following discussion describes how the PSA methods were employed for the internal events evaluation.

<u>Initiating Events and Conditions for SFP Assessment</u>

In this section, the focus is on the internal initiating events. The initiating events that meet the criteria defined in the ASLB Order are all the initiating events considered in the SHNPP Level 1 PSA for internal events.

The core damage and containment failure or bypass events that are included in the SHNPP PSA Level 1 and Level 2 results were input directly to the Spent Fuel Pool Assessment Event Tree (SFP-AET) described in Appendix D.

<u>Accident Sequences Evaluated for SFP Assessment</u>

In addition to the accident sequences derived from the Level 1 SHNPP PSA and their subsequent challenge of containment, which establish the initial conditions for this analysis, the accident sequence evaluation was then extended to assess the impact on the SFPs.

It is noted that the assessment of the SFPs is dependent on several effects:

- The support system availability
- The consequential effects of the core melt progression and containment failure
- The consequential effects of the loss of SFP cooling and the subsequent SFP boiling and its potential adverse impacts.

The first of the three effects is accounted for by transferring the cutsets for core damage and containment failure from the Level 2 SHNPP PSA into the SFP AET which is described in more detail in Appendix D.

The approach also included separating the cutsets from the Level 2 SHNPP PSA evaluation into the following principal containment failure categories to address the second of the above effects:

CORE MELT PROGRESSION AND
CONTAINMENT FAILURE MODE

- Containment Bypass (Large) (Includes ISLOCA)
- Containment Bypass (SGTR)
- Containment Isolation Failure
- Early Containment Failure
- Late Containment Failure
  - Basemat Failure
  - Overpressure Failure
- In Vessel Recovery and Containment Failure

Table 4.1-1 summarizes the internal event accident sequence types by containment failure categories and their potential consequential effects on the ability to maintain SFP integrity.

Table 4.1-1 includes a description of the following important aspects of the mitigation capability:

- The support system adversely affected.
- The containment conditions and timing.
- The potential methods that could be used to provide SFP makeup recognizing the adverse conditions created by the postulated accident.
- The status of the SFP cooling system initially. It is noted that under the Postulated Sequence, SFP cooling is always assumed to eventually fail in this analysis.

The SFP-AET described in Appendix D gives the analysis structure to evaluate the methods of SFP makeup and cooling. The SFP-AET processes the cutsets from the Level 2 SHNPP PSA. The quantification is performed separately for the different containment failure modes identified above because of the strong dependence of the operating crew and plant equipment response capability as a function of the containment failure mode. This dependence includes both time constraints and spatial effects due to environmental degradation.

<u>Thermal Hydraulic Analysis</u>

Three aspects of the thermal hydraulic analysis are important to the risk assessment:

- The containment failure timing and location is important in the assessment of operating crew response for SFP water inventory control. The analysis is based upon the EQE assessment in the IPE.

- The SFP decay heat, times to boil, and the boil down times are based on CP&L calculations.
- The assessment of RAB and FHB conditions subsequent to a containment failure or bypass is based upon the use of the MAAP code to assess pathway accessibility through the buildings and the CP&L calculations for the effects of the radionuclides dispersed on personnel access.

Systems

A complete fault tree system analysis was performed for the makeup systems and the SFP cooling system. These fault trees are part of the SFP-AET developed in Appendices A and D.

Data

The CP&L SHNPP PSA data base was used where appropriate for similar components in the SFP cooling system and the SFP makeup systems. For other inputs, estimates from the SHNPP Operations Department personnel were used. [4-1]

HRA

The human reliability analysis (HRA) approaches that have been developed over the past few years have primarily been for use in PSAs of nuclear power plants at full power. Methods have been developed for assessing the likelihood of errors associated with routine processes such as restoration of systems to operation following maintenance, and those errors in responding to plant transients or accidents from full power. For SFP operation, there are unique conditions not typical of those found during full-power operation. Thus, the human reliability methods developed for full power operation PSAs, and their associated error probabilities, are not directly applicable. However, some of the methods can be adapted to provide insights into the likelihood of failures in operator performance for the SFP analysis by accommodating the differences in conditions that might impact operating crew performance in the full power and decommissioning phases. There are both positive and negative aspects of the difference in conditions with respect to the reliability of human performance.

Examples of the positive aspects are:

- For most scenarios analyzed here, the time-scale for significant changes in plant condition are protracted. This is in contrast to full power transients or accidents in which response is required in a relatively short time, ranging from a few minutes to a few hours. Times ranging from 60 hours to greater than 200 hours were assumed for heat up and boil off following loss of SFP cooling. Thus, there are many opportunities for different plant personnel to recognize off-normal conditions. A long time is available to take corrective action, such as making repairs, hooking up alternate cooling or inventory makeup systems, or even bringing in help from off site.

- There is only one function to be maintained for success in the analysis performed here, namely SFP decay heat removal, and the systems available to perform this function are relatively simple. By contrast, in the full power case there are several functions that have to be maintained, including criticality control, pressure control, heat removal, and containment integrity.

Examples of the negative aspects that could influence the HRA are:

- Because the back-up systems are not automatically initiated, operator actions are essential to successful response to failures of the SFP cooling function.

- The response is to mitigate challenges that may not be viewed as an immediate threat.

The model considered multiple questions regarding each operator action:

- How is the action diagnosed and by whom?

This is answered by identifying a common basic event for all makeup sources that requires the operating crew or TSC to diagnose the action and direct the proper response. This is quantified using a combination of the cause based, ASEP, and THERP [4-2] procedures.

- How is the action carried out?

  This is represented by an assessment of the manipulation error using the THERP methodology [4-2].

- How does accessibility play a role?

  Accessibility is treated separately from the above diagnosis and execution evaluations. The deterministic MAAP calculations assess whether the conditions in the local areas are adequate to allow the local manual actions. If so, then the manipulation error determined above applies; if not then the action is considered to have failed.

The HRA to support the evaluation of operator actions in the this analysis is a combination of methods that have been used successfully in past nuclear power plant operating PSAs and shutdown PSAs. These methods address both short duration responses which may be time critical and very long duration responses that may be strongly dependent on other performance shaping factors such as local access.

Four quantification methods were applied, and each is briefly described below:

- The Technique for Human Error Prediction (THERP) [4-2]. This method was used to quantify the initial recognition of the problem. Specifically, the annunciator response model (Table 20-23 from Reference 4-2) was used for response to alarms. The THERP approach was also used to assess the likelihood of failure to detect a deviant condition during a walk-down, and also the failure to respond to a fire.

- ASEP Time Reliability Correlation (see Appendix C) to assess the time performance shaping factor. [4-2]

- The EPRI Cause Based HRA method. [4-3]

- An additional diagnosis evaluation to characterize the TSC response. [4-3]

Dependencies Among Operator Actions

It is noted that the multiple human error probabilities (HEPs) in the cutsets have been examined. These HEPs are determined to be completely different actions, occurring in totally different time frames, and performed by different crews. Therefore, there is considered to be no dependence between successive operator actions observed in the resulting cutsets.

In addition, a separate study to set all operator actions to 1.0 was also performed. This separate evaluation determined that the cutsets with multiple HEPs exhibited the same character as those in the dominant cutsets. Therefore, no additional dependent failures needed to be applied.

Dependencies

The treatment of dependencies included the following:

- Common cause failures were included where appropriate.
- Operator and TSC actions that can influence multiple nodes were identified and their dependencies explicitly modeled.
- The failures of support systems or components in Level 1, Level 2, or in the SFP-AET were explicitly tracked to determine their failure in subsequent nodes.
- Spatial interactions that can influence multiple modes or systems were explicitly tracked and the conditions affecting multiple systems were explicitly part of the probabilistic model.

Structural

See Thermal Hydraulic Analysis.

Quantification

The quantification process used the CAFTA code to perform the calculation.

Level 2

The Level 2 SHNPP PSA was used directly as input to assess the radionuclide release pathways and their approximate timing.

Table 4.1-1

COMPARISON OF POSTULATED SCENARIOS WITH POTENTIALLY VIABLE METHODS OF SPENT FUEL POOL COOLING OR MAKEUP

Scenario Description			SFP Make up Methods[1]			SFP Cooling Available Initially
Scenario Characterization	Support System Failures	Containment Condition	Method With Access Constraints Considered	Method With Recovery of Access Required		
				Supplemental Methods	Recovery	
ISLOCA (Containment Bypass-Large)	CCW, ESW Booster, Some MCC	• Containment bypassed into the RAB • Early Failure	<u>PB</u>: Demin water at the FHB 216' El manually aligned (North 216'El through 2SF201)	<u>N1</u>: Fire protection to SFP via hoses <u>N2</u>: Demin water quick connect options at 286' El of FHB	Access to FHB 286' El. required (During the first 8 days, access may be feasible)	No (Adverse Environment)

[1] The RWST is not filled during refuel operations with the cavity flooded, therefore use of the RWST as a makeup water source to the SFP is precluded under those conditions. In addition, the RWST can be used for injection to containment during a severe accident, therefore a substantial portion of the RWST inventory is likely not available for SFP makeup under the conditions postulated in the ASLB Order.

Table 4.1-1

COMPARISON OF POSTULATED SCENARIOS WITH POTENTIALLY VIABLE METHODS OF SPENT FUEL POOL COOLING OR MAKEUP

Scenario Description			SFP Make up Methods[1]			
Scenario Characterization	Support System Failures	Containment Condition	Method With Access Constraints Considered	Method With Recovery of Access Required		SFP Cooling Available Initially
				Supplemental Methods	Recovery	
SGTR (Containment Bypass-Large)	Not directly related to Bypass Mechanism	• Early Containment Failure • Bypassed; release path out the SGTR • Release is to env. outside RAB and FHB • Environment outside RAB and FHB may preclude personal movement • Emergency HVAC for the RAB and FHB may result in taking suction outside the building and discharging to the building. This could contaminate the building interiors	SFPCCS Cooling should remain available PA: ESW alignment in RAB and FHB PB: Demin water in FHB N1: Fire protection to SFP if performed before late containment failure N2: Demin water quick connects at 286' El of FHB if performed before late containment failure	No additional supplemental methods considered.	Following Late Containment Failure when access to FHB 286'El. and RAB is compromised. Access would need to be restored. Access to FHB 286' El. Required (During the first 8 days, access may be feasible)	Yes (Assumed to fail long term when Containment Failure affects RAB and FHB environment)

0C0812

Table 4.1-1

COMPARISON OF POSTULATED SCENARIOS WITH POTENTIALLY VIABLE METHODS OF SPENT FUEL POOL COOLING OR MAKEUP

Scenario Description			SFP Make up Methods[1]			SFP Cooling Available Initially
Scenario Characterization	Support System Failures	Containment Condition	Method With Access Constraints Considered	Method With Recovery of Access Required		
				Supplemental Methods	Recovery	
Early Containment Failures	CCW, ESW Booster, Some MCCs	• Failed Early	<u>PB</u>: Demin water at FHB 216' El. Aligned (North 216'El through 2SF201)	<u>N1</u>: Fire protection to SFP <u>N2</u>: Demin water quick connects at 286' El of FHB	Access to FHB 286' El. required (During the first 8 days, access may be feasible)	No
SBO - Early Failure	AC Power, CCW, ESW Booster, some MCCs, ESW	• Failed Early	<u>PB</u>: Demin water at FHB 216' El. Aligned (North 216'El through 2SF201) Method for motive power required. Offsite AC Power Recovery; portable generator; cut pipe and inject into Demin pipe	<u>PB</u>: Demin water at FHB 216' El. Aligned <u>N1</u>: Fire protection to SFP <u>N2</u>: Demin water quick connects at 286' El of FHB	Access to FHB 286' El. required (During the first 8 days, access may be feasible)	No

Table 4.1-1

COMPARISON OF POSTULATED SCENARIOS WITH POTENTIALLY VIABLE METHODS OF SPENT FUEL POOL COOLING OR MAKEUP

Scenario Description			SFP Make up Methods[1]			
				Method With Recovery of Access Required		
Scenario Characterization	Support System Failures	Containment Condition	Method With Access Constraints Considered	Supplemental Methods	Recovery	SFP Cooling Available Initially
SBO - Late Failure	AC Power, CCW, DC Power, ESW, ESW Booster, some MCCs	• Failed Late	<u>N1</u>: Fire protection to SFP TSC Specifies implementing a) AC Power restoration b) Align M/U (e.g., DFP - Diesel Fire Pump) <u>PB</u>: Demin water at FHB 216' El. Aligned (North 216'El through 2SF201) Method for motive power required. Offsite AC Power Recovery; portable generator; cut pipe and inject into Demin pipe	Restore SFP cooling by recovery of offsite power	AC Power restoration has high probability	No

Table 4.1-1

COMPARISON OF POSTULATED SCENARIOS WITH POTENTIALLY VIABLE METHODS OF SPENT FUEL POOL COOLING OR MAKEUP

Scenario Description			SFP Make up Methods[1]			SFP Cooling Available Initially
Scenario Characterization	Support System Failures	Containment Condition	Method With Access Constraints Considered	Method With Recovery of Access Required: Supplemental Methods	Method With Recovery of Access Required: Recovery	
Very Late Overpressure (88 hrs) Or Basemat Failures (77-122 hrs) Or Late Overpressure (38 hrs)	No specific support system related to this failure mode	• Failed Late • Containment failed very late (48 hrs to 90 hrs) • TSC expected to be manned	SFPCCS cooling should remain available for all sequences except identified support system failures in individual cutsets. • <u>PA</u>: ESW alignment in RAB and FHB This is assumed failed after late containment failure. • <u>N1</u>: DFP to SFP • <u>PB</u>: Demin water • <u>N2</u>: Demin water quick connects at 286' El of FHB if performed before late containment failure	No <u>additional</u> supplemental methods considered.	Following Late Containment Failure when access to FHB 286'El. and RAB is compromised. Access would need to be restored. Access to FHB 286' El. required (During the first 8 days, access may be feasible)	Possible

Table 4.1-1

COMPARISON OF POSTULATED SCENARIOS WITH POTENTIALLY VIABLE METHODS OF SPENT FUEL POOL COOLING OR MAKEUP

Scenario Description			SFP Make up Methods[1]			
Scenario Characterization	Support System Failures	Containment Condition	Method With Access Constraints Considered	Method With Recovery of Access Required: Supplemental Methods	Method With Recovery of Access Required: Recovery	SFP Cooling Available Initially
Large Isolation Failure Transients or Floods with Personnel Access Door Failed	CCW, ESW Booster, Some MCCs	• This is an early impact on radiation • Isolation failure due to personnel access door hardware failure • Release pathway directly to the RAB • Hydrogen, fission products, RPV blowdown steam into RAB	PB: Demin water at the FHB 216'El north manually aligned N1: Fire protection to SFP[*] N2: Demin water quick connects at 286' El of FHB[*]	N1: Fire protection to SFP via hoses N2: Demin water quick connect options at 286' El of FHB	Access to FHB 286' El. required (During the first 8 days, access may be feasible)	No (Adverse environment)

[*] Access to FHB 286' El. required. MAAP indicates that accessibility could be possible. However, sensitivity evaluations indicate that there is limited confidence that access could be obtained. Therefore, in the model, access to FHB 286'El. is not considered for containment isolation failures.

0C0810

Table 4.1-1

COMPARISON OF POSTULATED SCENARIOS WITH POTENTIALLY VIABLE METHODS OF SPENT FUEL POOL COOLING OR MAKEUP

Scenario Description			SFP Make up Methods[1]			
Scenario Characterization	Support System Failures	Containment Condition	Method With Access Constraints Considered	Method With Recovery of Access Required		SFP Cooling Available Initially
				Supplemental Methods	Recovery	
Small Isolation Failure	No specific support system failures identified.	• This is an early impact on radiation • Isolation failure due to personnel access door hardware failure • Release pathway directly to the RAB • Hydrogen, fission products, RPV blowdown steam into RAB	PB: Demin water at the FHB 216'El north manually aligned N1: Fire protection to SFP N2: Demin water quick connects at 286' El of FHB Access to FHB 286' El. Required	N1: Fire protection to SFP via hoses N2: Demin water quick connect options at 286' El of FHB	Access to FHB 286' El. required (During the first 8 days, access may be feasible)	Possible

4.2 SEISMIC EVENTS

The ASLB Order addresses those accident scenarios that result from the loss of SFP water due to evaporation (including boiling). A seismic event can lead to any or all of the following:

- Loss of offsite power
- Diesel generator failure
- SFP cooling or CCSW failure
- FHB failure and SFP draining

The last event is not part of the ASLB specified sequence; therefore, it is not considered in this seismic quantification. As such, the following portion of the full spectrum of postulated seismic events are addressed in this study: seismic events large enough to contribute to the initiating severe accident and containment bypass and disruption in SFP cooling, but of insufficient magnitude to cause FHB failure and draining of the SFP.

CP&L has completed an IPEEE [4-24] for seismic events per Generic Letter 88-20, Supplement 4 that has been accepted by the NRC. The Shearon Harris Seismic IPEEE uses the Seismic Margins Assessment (SMA) methodology. This methodology entails demonstrating a high confidence in low probability of failure (HCLPF) for equipment in designated redundant success paths for seismic event mitigation.

On the basis of the IPEEE review, the NRC staff concluded that CP&L's IPEEE process was capable of identifying the most likely severe accidents and severe accident vulnerabilities and, therefore, that the SHNPP IPEEE has met the intent of Generic Letter 88-20, Supplement 4.

Because the Seismic Margins Assessment method was used in the SHNPP IPEEE, frequencies of seismic-induced core damage accident sequences were not calculated. Therefore, a focused seismic PSA assessment was developed and is summarized here to support the ASLB required assessment. This assessment uses the results of the SHNPP IPEEE and techniques derived from previous seismic PSAs. This streamlined assessment calculates the frequency of the Postulated Sequence when initiated by a seismic event.

The seismic methodology is shown graphically (in event tree format) in Figure 4.2-1. Figure 4.2-1 shows that the analysis addresses the following key steps:

- Seismic Hazard Frequency Assessment
- Seismic-Induced Reactor Core Damage including Seismic Fragility Assessment
- Early Containment Failure Assessment
- Containment Isolation Failure Assessment
- Maintenance of Spent Fuel Coolant Inventory

Seismic events resulting in no core damage are not applicable to this assessment and are not analyzed further (Sequence #1 in Figure 4.2-1). Nor are seismic events which would breach the spent fuel pool and result in a drain down applicable to this analysis because one of the postulated events would be eliminated.

Seismic events are postulated to result in accident scenarios that can lead to the following containment failure modes:

- Early containment failure (sequence #7)
- Containment isolation failure (sequence #5)
- Late containment failure (sequence #3)
- Steam Generator Tube Rupture (SGTR)
- ISLOCA

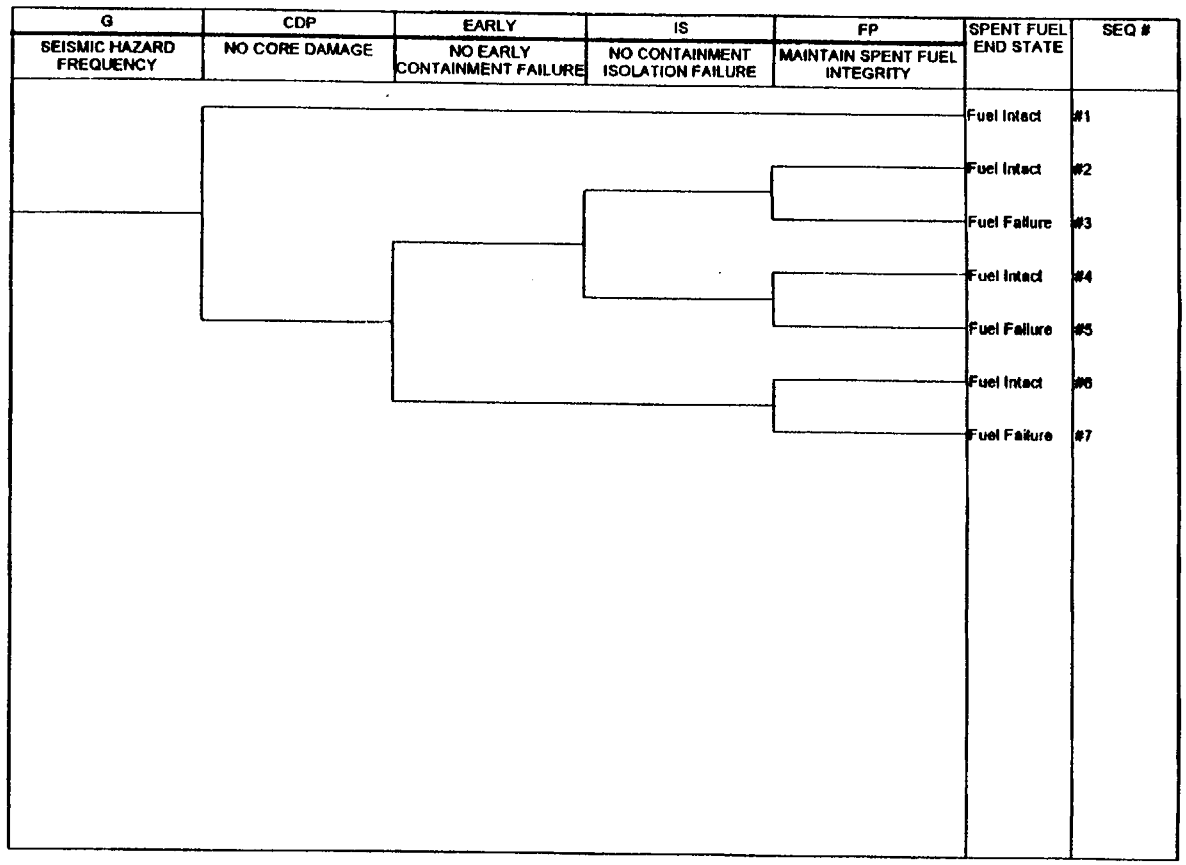


Figure 4.2-1 GRAPHICAL REPRESENTATION OF SEISMIC ANALYSIS

The probability of ISLOCA or SGTR caused by a seismic event has been found to be of low probability and therefore they are not explicitly modeled (or depicted in Figure 4.2-1), i.e., they are less than 1E-8/yr.

Early containment failure events can lead to radionuclide releases to the RAB within approximately 1 to 4 hours of the seismic event. This could limit the time and access to certain areas for SFP related actions (sequence #7).

The containment isolation failure may also lead to early radionuclide releases to the RAB; however, the containment isolation failure leads to significantly milder results in the RAB and FHB than the early containment failure (sequence #5).

Given a seismic-induced core damage scenario, if no early containment failure results and the containment isolation function is successful, the current model assumes that late containment failure will always occur for these seismic severe accidents because no credit is given for repair and recovery under the postulated seismic event. (This assumption may be conservative.) The late containment failure results in a substantial time window (38-90 hrs) during which preparatory actions could be performed by the operating crew or by the OSC at the direction of the TSC (sequence #3).

These failure modes and effects are similar to those discussed for internal events.

4.2.1 Seismic Hazard Frequency

The earthquake hazard frequencies used in this analysis are taken from the latest Lawrence Livermore National Laboratory work on seismic hazard estimates, as discussed below.

Background

The U.S. Nuclear Regulatory Commission (NRC) has been sponsoring the development of probabilistic seismic hazard analysis (PSHA) methodologies by Lawrence Livermore National Laboratory (LLNL) since the 1970's. In the 1980's, the NRC sponsored a LLNL study to develop a seismic hazard methodology for all operating nuclear power plant sites in the eastern United States.

The 1980's LLNL methodology included input data provided by 11 seismicity experts and 5 ground motion experts. The seismicity experts defined maps of source zones of uniform seismicity and then described the seismicity of each zone in terms of the rate of earthquakes versus magnitude for each zone. The ground motion experts each provided several attenuation models for predicting ground motion as a function of distance from the earthquake source. LLNL developed a seismic hazard model that used the experts' input and a Monte Carlo simulation approach to provide an estimate of the probability of exceeding a level of ground motion at a given site. LLNL applied its methodology to develop probabilistic seismic hazard estimates at all 69 eastern United States operating plant sites.

In conjunction with funding LLNL to perform a PSHA study, the NRC recommended that the nuclear power industry perform an independent study to provide the NRC with comparative information. A consortium of nuclear power utilities funded the Electric Power Research Institute (EPRI) to perform a seismic hazard study. EPRI [4-16] developed its own PSHA methodology and PSHA estimates at 56 of the eastern United States sites. The differences between the 1980's LLNL and the EPRI seismic hazard estimates were subsequently assessed in NUREG/CR-4885. [4-17]

LLNL applied its methodology to studies at Department of Energy (DOE) sites. During these applications, LLNL reexamined the expert opinion elicitation process used in the 1980's LLNL studies to better characterize the uncertainty. On the basis of insights gained from these applications, the NRC sponsored a limited re-elicitation of the LLNL experts to refine the estimates of uncertainty in seismicity and ground motion estimates. During 1992 and 1993, LLNL re-elicited input data from the seismicity and ground motion experts using a revised elicitation procedure. LLNL then revised the PSHA computer code and produced updated PSHA estimates at eastern United States sites.

The updated LLNL methodology reduced the seismic hazard estimates below that of the 1980's study, thus reducing the differences between the LLNL and EPRI hazard estimates. The largest differences between the 1993 LLNL and EPRI hazard estimates are at low seismicity sites and soil sites.

According to NUREG-1488 [4-15], the updated LLNL seismic hazard estimates will be considered by the NRC staff in future licensing actions such as safety evaluation reports, reviews of IPEEE submittals, and early site reviews. Therefore, the best seismic hazard data available are used in this analysis.

Frequency Estimation

As stated above, NUREG-1488 provides updated LLNL seismic hazard estimates for the 69 nuclear power plant sites in the eastern United States (i.e., east of the Rocky Mountains). The seismic hazard estimates for the Shearon Harris site, as quoted in NUREG-1488, are presented in Table 4.2-1. These hazard estimates are also presented graphically in Figure 4.2-2 (the data points in the figure are the discrete NUREG-1488 values, the solid curve a curve-fit equation developed as part of this assessment). The estimates are presented in terms of annual exceedance frequency. For example, at 0.1 peak ground acceleration the frequency is 2.11E-4, meaning the frequency of experiencing a seismic event at the SHNPP site with a peak ground acceleration of 0.10g, or greater, is 2.11E-4/yr.

Division of Seismic Hazard Curve

As can be seen from Figure 4.2-2, the seismic hazard curve is characterized by decreasing exceedance frequency with increasing seismic magnitude. Both of these parameters (frequency and magnitude), play key roles in the seismic PSA. Given the broad spectrum of both the frequency and magnitude parameters, it is not appropriate to simply perform a single averaged analysis that represents the entire seismic hazard curve. The typical analytical technique used in seismic PSAs is to divide the seismic hazard curve into a discrete number of ranges and perform a seismic PSA for each of the discrete ranges. The probabilistic results from each range are then integrated to obtain the combined seismic PSA result. This is the approach used in this analysis.

The Shearon Harris seismic hazard curve is divided into the following seven intervals:

- < 0.1 pga
- 0.1 – 0.3 pga
- 0.3 – 0.5 pga
- 0.5 – 0.7 pga
- 0.7 – 1.0 pga
- 1.0 – 1.5 pga
- >1.5 pga

The hazard frequency used in this risk assessment for each of the seismic ranges is calculated as the exceedance frequency at the low end of the range minus the exceedance frequency at the high end of the range. This results in the frequency of a seismic event with a magnitude exceeding the low end of the magnitude range but not the high end of the range. For the > 1.5 g magnitude range, the exceedance frequency for a 1.5 g seismic event is used. At >1.5g, the likelihood that the FHB suffers major damage due to the seismic shock is quite high (>0.50 probability, using a seismic capacity of 1.5g based on the generic class IE building capacity information presented in Table 4.2-2); as such, the >1.5g

Table 4.2-1

SEISMIC HAZARD ESTIMATES FOR SHEARON HARRIS

Peak Ground Acceleration (g's)	Annual Exceedance Probability				
	NUREG-1488 Point Estimates (1)				Curve-Fit Values (2)
	15th Perc.	50th Perc.	Mean	85th Perc.	
0.05	9.4E-5	3.7E-4	5.8E-4	1.1E-3	7.65E-4
0.08	4.1E-5	1.8E-4	3.1E-4	5.6E-4	3.46E-4
0.10	---	---	---	---	2.11E-4
0.15	9.1E-6	5.0E-5	9.1E-5	1.7E-4	9.53E-5
0.20	---	---	---	---	5.10E-5
0.25	2.2E-6	1.5E-5	3.1E-5	5.7E-5	2.78E-5
0.30	---	---	---	---	1.85E-5
0.31	1.3E-6	9.1E-6	2.0E-5	3.6E-5	1.76E-5
0.40	---	---	---	---	8.89E-6
0.41	4.8E-7	3.9E-6	9.2E-6	1.7E-5	8.49E-6
0.50	---	---	---	---	4.64E-6
0.51	1.9E-7	1.9E-6	4.8E-6	8.6E-6	4.34E-6
0.60	---	---	---	---	2.69E-6
0.66	5.0E-8	7.3E-7	2.1E-6	3.6E-6	2.06E-6
0.70	---	---	---	---	1.75E-6
0.80	---	---	---	---	1.14E-6
0.82	1.6E-8	3.0E-7	1.0E-6	1.7E-6	1.06E-6
0.90	---	---	---	---	8.05E-7
1.00	---	---	---	---	6.00E-7
1.02	4.5E-9	1.1E-7	4.6E-7	7.9E-7	5.75E-7
1.10	---	---	---	---	4.82E-7
1.20	---	---	---	---	4.08E-7
1.30	---	---	---	---	3.50E-7
1.40	---	---	---	---	3.09E-7
1.50	---	---	---	---	2.99E-7

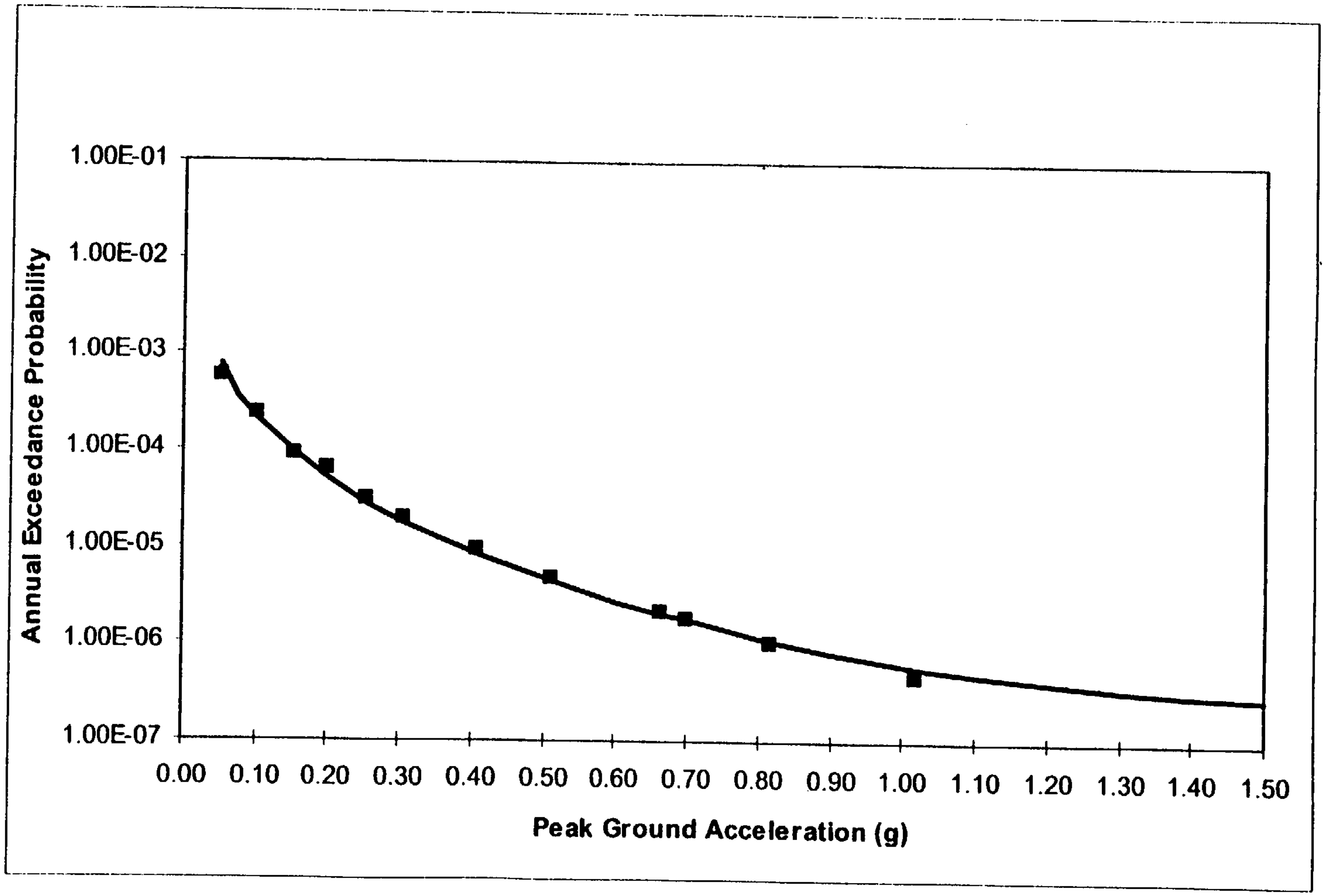
Notes:

(1) Dashes indicate no point estimate data provided in NUREG-1488.

(2) The curve-fit values are calculated by applying an exponential equation to best fit the NUREG-1488 discrete point estimates. These values are employed in the frequency quantifications of this seismic analysis.

Figure 4.2-2

SHEARON HARRIS SEISMIC HAZARD CURVE



interval is defined as the bounding magnitude for this analysis (i.e., seismic events in this magnitude range are assumed to result in FHB failure and as such are not part of the assessed spent fuel failure frequency in this analysis). The < 0.1 g seismic range is not explicitly quantified in this risk assessment as the seismic impacts of this g level are negligible contributors.

4.2.2 Seismic Fragility Assessment

A seismic shock can induce equipment and/or structural failures. As the magnitude of the seismic shock increases, the likelihood of these seismic-induced failures also increases. These issues need to be factored into the analysis.

Seismic fragility is the conditional probability of component or structural failure vs. ground acceleration. Failure is defined as the response level at which the component will no longer perform its intended function. This might be trip of a circuit breaker, failure of equipment anchorage or pressure boundary failure. In some cases, permanent structural deformation will take place at levels substantially below the failure threshold.

Depending on the scope and schedule of the seismic risk analysis, two main approaches to the calculation of seismic fragilities have typically been employed in seismic PSAs: 1) fragility as a function of local response, and 2) fragility as a function of peak ground acceleration. The first approach requires significant resources to evaluate local response parameters (e.g., damping, floor response spectra) for the numerous key components and structures to be addressed in the analysis and is outside the scope of this analysis. This analysis employs the second approach.

The second approach calculates fragility in terms of peak ground acceleration (pga) and is assumed to fit a lognormal distribution with a median acceleration capacity and two variables, β_r and β_u , defined as the logarithmic standard deviations representing randomness and uncertainty about the median. Due to the availability of median seismic component capacities in industry literature since about the mid-1980's, this method has become more attractive for its ease of use. The fragility is defined by the following equation:

$$f' = \Phi\left([\ln(a/A_m) + \beta_u \Phi^{-1}(Q)] / \beta_r \right)$$

The quantity Φ is the standard Gaussian cumulative distribution function, and the quantity Φ^{-1} is its inverse. The parameter ***Q*** is the probability that the conditional frequency of failure, f, is less than f ' for a given acceleration (e.g., a Q of 0.50 indicates a median fragility and a Q of 0.95 indicates a fragility with a 95% confidence level). The parameter ***a*** is the ground acceleration in question. The parameter $\boldsymbol{A_m}$ is the median ground acceleration capacity of the component or structure. The parameter $\boldsymbol{\beta_r}$ is the logarithmic standard deviation representing the inherent randomness of the seismic characteristics (e.g., duration, spectral shape) which can not be significantly reduced by further current analyses or tests. The parameter $\boldsymbol{\beta_u}$ is the logarithmic standard deviation representing the uncertainty (e.g., due to lack of knowledge of material strength, damping factors) in the estimation.

The fragility of a component or structure is fully described by a family of curves representing different confidence levels (refer to Figure 4.2-3). The center solid curve of Figure 4.2-3 represents the median (50% confidence level) fragility curve. The 95% and 5% confidence levels are represented by the left- and right-most curves, respectively. When the analysis is performed using a fragility point estimate (typical approach), the fragility equation reduces to:

$$f' = \Phi\left(\ln(a/A_m) / \beta_c \right)$$

> where the value β_c is the composite deviation and is the square root of the sum of the squares of the randomness and uncertainty components (i.e., β_c = SQR(β_r^2 + β_u^2)).

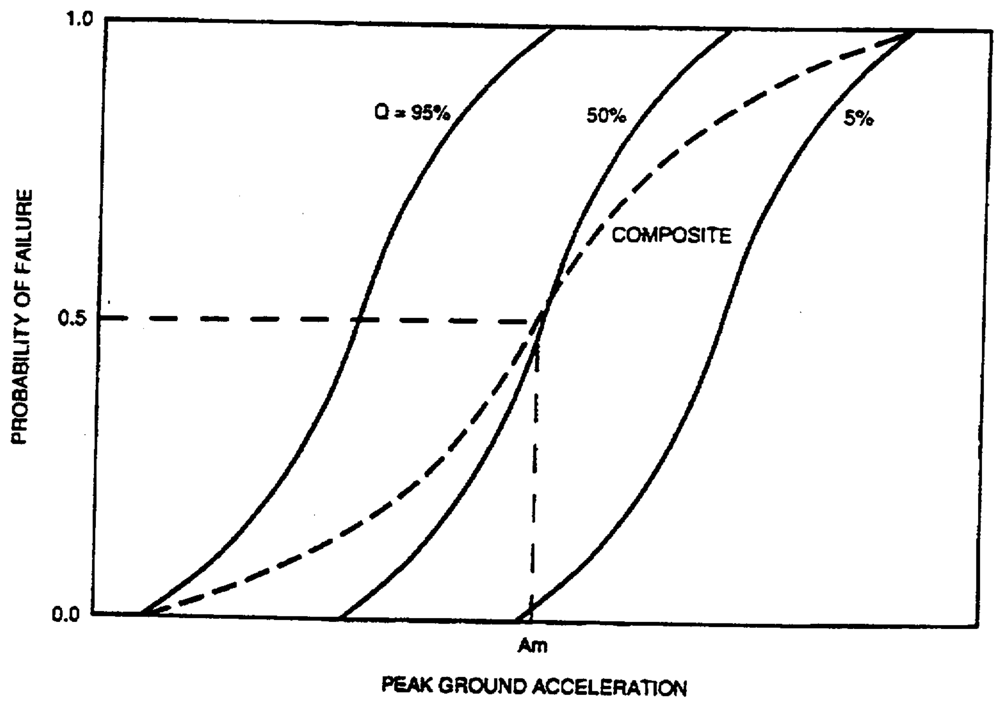


Figure 4.2-3 Typical Fragility Curves

This is the equation used in this evaluation to estimate component and structural fragilities. This composite fragility curve is shown in Figure 4.2-3 as a dashed line.

As an example, consider that the fragility of a certain component is to be calculated for a seismic peak ground acceleration of 0.3g. The median seismic capacity of the component is determined to be 0.7g. The randomness and uncertainty parameters are both assumed to be 0.30 in this example (these are typical values based on past seismic studies). The fragility would be calculated as follows:

$$f' = \Phi(\ln(0.30 / 0.70) / 0.42)$$
$$f' = \Phi(-2.0174)$$

From the equation, the fragility (f ') of the component with a median seismic capacity of 0.7 pga at a 0.3 pga loading is determined to be 2.18E-2.

Median Seismic Capacities

Fragility analyses were performed for the following structures and components considered in this analysis:

- Offsite AC Power
- Emergency Diesel Generators (EDGs)
- Essential Switchgear/MCCs
- Primary Containment Isolation Valves (PCIVs)
- Diesel-driven Fire Pump
- Fuel Handling Building and Spent Fuel Pool Integrity
- Offsite Infrastructure[1]

[1] Includes roads, bridges, communication systems.

In order to calculate the fragilities of these components and structures, the median seismic capacity of each was estimated.

As stated earlier, the use of the "fragility as a function of pga" calculational method is attractive due to the availability of median seismic capacity information in industry literature. Generic information from NUREG/CR-4334 is employed in this analysis. However, the results may be more conservative than if local damping within the buildings was accounted for.

In order to investigate, develop, and provide technical guidance regarding seismic margins analysis, the NRC formed the "Expert Panel on Quantification of Seismic Margins" in 1984. The Expert Panel adopted and employed the HCLPF concept. The HCLPF of a component corresponds to the earthquake level at which it is judged very unlikely that seismic motion induced failure of the component will occur. Expressed statistically, HCLPF values represent a 95% confidence level that the probability of component failure due to seismic motion is not greater than 5%.

Using a combination of judgment, engineering analysis and data, test data, real earthquake data, and past PSA analyses, the Expert Panel developed screening HCLPFs for specific types of equipment and structures and reported these in NUREG/CR-4334 [4-18]. The screening HCLPFs developed by the Panel were assigned to one of three categories:

- less than 0.3g
- 0.3g to 0.5g
- greater than 0.5g.

Table 4.2-2

GENERIC MEDIAN SEISMIC CAPACITIES (Am) CONSIDERED IN THE ESTIMATION OF SHNPP CAPACITIES

Component/ Structure	Am (g) by Seismic PSA [(1), (2), (3)]						
	Zion	Indian Point 2	Indian Point 3	Limerick	Millstone 3	Seabrook	Oconee
Offsite Power Insulators/ Transformers	0.25 (0.20, 0.25)	0.25 (0.20, 0.25)	0.26 (0.20, 0.25)	0.25 (0.20, 0.25)	0.20 (0.20, 0.25)	0.30 (0.25, 0.50)	0.25 (0.20, 0.25)
Emergency Diesel Generators	1.06 (0.35, 0.37)	1.60 (0.20, 0.25)	1.40 (0.26, 0.52)	1.91 (0.28, 0.43)	0.91 (0.24, 0.43)	1.03 (0.39, 0.36)	1.23 (0.25, 0.43)
Essential Switchgear/ MCCs	0.89 (0.35, 0.47)	2.03 (0.41, 0.53)	1.44 (0.24, 0.52)	1.64 (0.35, 0.38)	2.21 (0.28, 0.57)	1.52 (0.32, 0.48)	0.90 (0.24, 0.44)
Class IE Building [(4)]	0.90 (0.30, 0.28)	1.72 (0.30, 0.26)	1.48 (0.16, 0.23)	1.29 (0.31, 0.25)	1.00 (0.24, 0.33)	1.71 (0.41, 0.39)	1.16 (0.23, 0.28)

Notes to Table 4.2-2:

1. Reference: NUREG/CR-4334.
2. Values in parentheses are first the Randomness Factor, Beta(r), and the Uncertainty Factor, Beta(u).
3. Most conservative value listed when multiple options available from reference. For example, if the EDG and the Day Tanks are listed separately, and the Day Tanks have a lower capacity, the Day Tank capacity is used as the representative value for the EDG. Similarly, if a component lists a "Recoverable" capacity and a "Non-Recoverable" capacity, the lower "Recoverable" value is listed here.
4. The following are not included here: EDG Bldg. (already addressed by the EDG component); misc. masonry walls with specific impacts (e.g., masonry wall surrounding battery room); and Turbine Building.

To develop these screening HCLPF values, the Expert Panel reviewed numerous seismic PSAs and summarized a large number of component and structural median seismic capacities, A_m, in an appendix to the report. These generic median seismic capacities were used in this seismic PSA for SHNPP. A summary of generic median capacities from NUREG/CR-4334 for key components and structures in this analysis is provided in Table 4.2-2. Based on this generic information and knowledge of the Shearon Harris plant, median capacities were selected for use in this analysis. These are summarized in Table 4.2-3. The estimated capacities for SHNPP are selected based on judgment and review of the information in Table 4.2-2 (excluding the high and the low values from consideration).

Seismic Fragilities

Using the composite fragility equation presented earlier and the seismic capacities summarized in Table 4.2-3, seismic fragilities were calculated for use in this analysis. These fragilities are summarized in Table 4.2-4.

As this seismic analysis divides the seismic hazard curve into six discrete magnitude intervals, each interval is actually a short range of peak ground accelerations. This analysis uses the midpoint of each magnitude range to calculate the seismic fragilities.

In addition, the β_r and β_u distribution parameters are both assumed to be 0.40 for these fragility calculations.

4.2.3 Seismic-Induced Core Damage

The seismic-induced core damage frequency for Shearon Harris is calculated here as the sum of the following key seismic accident scenarios:

- Seismic Event x Seismic-Induced LOOP x Seismic-Induced Failure of EDGs x AC Power Recovery Failure
- Seismic Event x Seismic-Induced LOOP x Non-Seismic Common Cause Failure of EDGs x AC Power Recovery Failure

- Seismic Event x Seismic-Induced LOOP x Seismic-Induced Essential Switchgear Failure x AC Power Recovery Failure

These accident scenarios are calculated for each of the seismic magnitude ranges.

The seismic-induced fragility contributors for similar components used in this core damage assessment are conservatively assumed to be completely dependent. This represents an analysis conservatism. For example, the seismic-induced failure of one EDG is assumed, in this analysis, to result in seismic-induced failure of both EDGs.

Failure of individual components or structures due to seismic fragility has both statistically independent and dependent characteristics. Component failures due to seismic fragility are statistically independent because individual components may be dissimilar in design, location within the plant, and dynamic characteristics. The same component failures are also statistically dependent because the failure events are all induced by the same shock (a seismic event). The dependence is a function of hazard intensity. In the case of low hazard intensity, the dependence is low. At the theoretical extreme low end of hazard intensity, individual component fragilities are completely independent (i.e., 0.0 fragility dependence). At the high end of hazard intensity, individual component fragilities are theoretically completely dependent (i.e., 1.0 fragility dependence). The core damage frequency assessment in this seismic analysis assumes a 1.0 fragility dependence among similar components, which represents another conservatism.

With respect to loss of offsite power, this analysis conservatively assumes a 1.0 conditional probability for loss of offsite power due to any magnitude seismic event greater than 0.1 g. In addition, the recovery of AC power was assigned a failure probability of 1.0 for these seismic events. Scenarios involving seismic-induced failure of the containment or the FHB which lead to loss of SFP inventory are outside the scope of this analysis.

Table 4.2-3

SHNPP MEDIAN SEISMIC CAPACITIES

COMPONENT / STRUCTURE	MEDIAN SEISMIC CAPACITY (pga)
Offsite Power Insulators	0.0 (1)
Emergency Diesel Generators (EDGs)	1.25
Essential Switchgear / MCCs	1.31 (2)
Primary Containment Isolation Valves (PCIVs)	2.00 (5)
Diesel-Driven Fire Pump	1.25
Fuel Handling Building Flooding	1.25 (3)
Offsite Infrastructure	1.00 (4)

Notes to Table 4.2-3:

(1) This analysis conservatively assumes a seismic-induced loss of offsite power probability of 1.0 for all seismic magnitude ranges evaluated (> 0.1 g).

(2) Certain low voltage essential switchgear was assessed in the Shearon Harris IPEEE Submittal to have a HCLPF of 0.30g. Using the following conversion equation,

$$HCLPF = A_m \, Exp(-1.65(\beta_r + \beta_u))$$

the median capacity of 1.31 is calculated here (a value of 0.30 is used in this case for each of the distribution parameters, β_r and β_u).

(3) Fuel Handling Building Flooding is modeled as an unspecified component or set of components that are insufficiently seismically rugged and may fail with significant probability and result in significant flooding of the building.

(4) The fire truck to be used as a water supply for alternate fuel pool coolant makeup is housed off-site. In addition, a portable generator and pump may be transported to the site for use as an alternate fuel pool coolant pumping supply. This median seismic capacity is used to indicate extreme disruption of offsite infrastructures that prevents transport of the portable generator/pump and the fire truck to the site. This seismic capacity is indicative of the following seismic effects:

- Conspicuous ground fissures
- Broken underground city pipes
- Considerable damage to well-designed city buildings

(5) NUREG/CR-4334 references a median capacity of 2.00 pga for air-operated containment isolation valves.

(6) Conservative estimate of seismic capacity for diesel fire pump. Based on review of generic data in NUREG/CR-4334, focusing on generic values for emergency diesel generators and DC battery nodes.

Table 4.2-4

SHNPP SEISMIC FRAGILITIES

Component / Structure	A_m	Fragility by Seismic Magnitude												
		0.05	0.10	0.20	0.30	0.40	0.50	0.60	0.70	0.80	0.90	1.00	1.25	1.50
Offsite Power Insulators	0.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
Emergency Diesel Generators	1.25	negligible	4.01E-6	5.99E-4	5.82E-3	2.20E-2	5.26E-2	9.72E-2	1.53E-1	2.15E-1	2.81E-1	3.47E-1	5.00E-1	6.26E-1
Essential Switchgear/ MCCs	1.31	negligible	2.71E-6	4.46E-4	4.58E-3	1.80E-2	4.43E-2	8.37E-2	1.34E-1	1.92E-1	2.53E-1	3.17E-1	4.67E-1	5.95E-1
Primary Containment Isolation Valves	2.00	negligible	negligible	2.35E-5	3.99E-4	2.22E-3	7.13E-3	1.67E-2	3.17E-2	5.26E-2	7.90E-2	1.10E-1	2.03E-1	3.06E-1
Diesel Fire Pump	1.25	negligible	4.01E-6	5.99E-4	5.82E-3	2.20E-2	5.26E-2	9.72E-2	1.53E-1	2.15E-1	2.81E-1	3.47E-1	5.00E-1	6.26E-1
Fuel Handling Bldg. Flooding	1.25	negligible	4.01E-6	5.99E-4	5.82E-3	2.20E-2	5.26E-2	9.72E-2	1.53E-1	2.15E-1	2.81E-1	3.47E-1	5.00E-1	6.26E-1
Offsite Infrastructure	1.00	negligible	2.35E-5	2.22E-3	1.67E-2	5.26E-2	1.10E-1	1.83E-1	2.64E-1	3.47E-1	4.26E-1	5.00E-1	6.53E-1	7.63E-1

4.2.4 Early Containment Failure Assessment

Given a core damage event, the timing of the subsequent containment failure (given a containment failure does occur) is key to the likelihood of successfully maintaining coolant inventory to the SFP. An early containment failure following core damage severely limits operator activities in and around the site.

The conditional probability of an early containment failure given core damage, used in this assessment is 3.76E-2 and is taken from the current SHNPP PSA. [4-13] This conditional early containment failure probability is the worst case conditional probability for any plant damage state from the SHNPP PSA results.

4.2.5 Containment Isolation Failure Assessment

This analysis also considers that containment isolation is not successful when demanded during a core damage scenario. Failure of containment isolation will also result in an "early" release state. While not as severe as early containment failure (i.e., early containment failure would result in releases directly to the fuel pool deck; whereas, containment isolation failure was found in the deterministic calculations to have a less severe impact on FHB environments), failure of the containment isolation function will also impact the ability of the operators to perform alternate fuel pool coolant alignment activities.

The probability of containment isolation failure is assessed here as the sum of two contributors:

- Pre-existing containment leakage at the time of the core damage scenario
- Containment isolation functional failure on demand

The pre-existing leakage probability is taken in this study as 1E-3 per core damage scenario, based on NUREG/CR-4551 (Vol. 6) and NUREG/CR-4220.

The containment isolation functional failure on demand contribution considers both seismic and non-seismic failures. The non-seismic containment isolation failure contribution is taken here to be 1E-3 per core damage scenario, based on NUREG/CR-4551 (Vols. 3 and 7).

The seismic contribution is assessed by calculating the seismic fragility of a primary containment isolation valve (PCIV). Given the seismic-induced failure of a PCIV (i.e, failing the valve in the open position), the conditional probability that the second inline PCIV also fails to close was assessed. The concept of fragility dependence was applied here to the assessment of the second valve failure, and was assumed to be an exponential function increasing from 0.05 at a 0.05g seismic event to 1.0 at a 1.5g seismic event.

Manual containment isolation was assigned a failure probability of 1.0.

4.2.6 Maintenance of Spent Fuel Pool Coolant Inventory

For severe seismic events, two pathways for makeup to the SFPs were identified as viable:

- Diesel-driven fire pump and fire hoses aligned to SFPs
- Demineralized water pathway via 2SF-201 valve on 216' El. North

Access to each of these pathways can be discussed relative to the early and late containment failure modes. Access includes an evaluation of:

- Radiation environment
- Temperature environment

- Steam environment
- Door accessibility

The first three environments were addressed using the deterministic code MAAP (see Appendix E). The last item was reviewed to ensure that the accident sequence would not render the door inoperable. All seismic events considered here also lead to SBO conditions. If the security diesel also failed, and security batteries depleted, the doors can still be opened with keys carried bv the security force and auxiliary operators. Therefore, for extended times, security personnel or auxiliary operators with keys would be available to provide access even under SBO conditions.

Diesel Fire Pump Pathway

The use of the Fire Protection System (FPS) piping is the preferred pathway under seismic events because this pathway is comprised of piping with a recognized seismic piping pedigree.

For "late" containment failures following a seismic event, significant time is available (38 hours) to enter the FHB and align the fire hoses to the SFPs. Entering the FHB and aligning fire hoses to the SFPs will setup the pathway that could subsequently be used with any pumping source that can be aligned into the FPS.

For "early" containment failures (including isolation failure), the FHB operating deck (286' El.) is not accessible, as calculated with the deterministic computer analysis (MAAP) documented in Appendix E, and the FPS pathway is therefore not credited. The FHB HVAC system is not functioning because power is not available.

Demineralized Water System Pathway

The use of the demineralized water pathway is also a viable path under a variety of conditions. For all accident scenarios caused by the seismic event, the crew has access to the 216' El. North compartment to make the alignment, as calculated by the deterministic computer analysis (MAAP) and documented in Appendix E.

This access route is well-protected from a potential radiation environment and therefore the status of containment has less impact on the successful alignment of this flow path. Therefore, for either early (including isolation failure) or late containment failures, this flow path should be available with a high likelihood. For higher magnitude seismic events, there may be complicating issues related to seismic-induced failure of pumps to pipe connections that could either:

- Cause flooding in the area, or
- Prevent the piping path from being operable.

Pumping Sources

The diesel fire pump is one primary method of supplying makeup water to the SFP under a seismic event that has caused a SBO. The diesel fire pump will likely survive a substantial portion of the seismic spectrum. Therefore, for a large fraction of the spectrum and for sequences with the FHB accessible, the FPS pipe and the diesel fire pump offer a reliable and viable mitigation method.

Early containment failures can compromise access to the FPS through the FHB operating deck. High seismic magnitude events may disable the diesel fire pump. These would then limit the benefit of this pathway. For large seismic events with "late" containment failures, offsite resources may be available to support this pathway.

For seismic events leading to core damage and containment failure, offsite AC power is likely not available. Therefore, the demineralized water pumps are not available to support the demineralized water system path. In addition, because they are not seismically qualified they may not provide any benefit in a large seismic event even if they could be powered from a portable source. The alternate method of supplying water through the demineralized water pathway (by cutting pipe) is available, using the following water sources:

- Fire truck pump to supply water to the connection
- Portable generator and pump using Shearon Harris Lake as water source

Quantification of Alternate Alignment

In summary, the quantification of failure to align alternate fuel pool coolant makeup following a seismic event considers the following contributors:

- Failure of Fire Hose Alignment
  - Diesel fire pump failure
    - Seismic-induced failure of DFP
    - Failure to Start/Run
  - Fire truck and portable pump/gen. water sources unavailable
    - Seismic-induced failure of offsite infrastructure prevents transport of portable generator/pump and fire truck to site
    - Failure to perform fire truck hook-up
    - Failure to perform portable generator/pump hook-up
  - Diagnosis/Manipulation HEP for fire hose alignment in the FHB:
    - Early containment failure
    - Containment isolation failure
    - No early containment failure or isolation failure
- Failure of Demineralized Water Pathway
  - Building access precluded due to flooding
    - Seismic-induced flooding
    - Flooding prevents access to basement
  - Fire truck and portable pump/generator water sources unavailable

- -- Seismic-induced failure of offsite infrastructure prevents transport of portable generator/pump and fire truck to site
- -- Failure to perform fire truck hook-up
- -- Failure to perform portable generator/pump hook-up

– Diagnosis/Manipulation HEP of demineralized valving in the FHB:

- -- Early containment failure
- -- Containment isolation failure
- -- No early containment failure or isolation failure

<u>Seismic Walkdown Insights</u>

A Shearon Harris supplemental seismic walkdown was also performed by seismic experts to support this analysis.

Based on the supplementary seismic walkdown performed in support of the SHNPP SFP analysis, the following important insights associated with makeup to the SFPs were identified:

- The purification pumps are considered to have extremely low seismic capability. Therefore, these pumps would be unavailable for essentially all seismic events for which core damage is projected.

- The demineralized water pumps are powered from offsite AC power. It is assumed in this analysis that seismic induced LOOP would not be recovered.

- Seismic movement of the purification pump is projected to lead to failures of the attached piping such that manipulation of 1SF-201 may not be feasible. The analysis assumes a relatively high failure probability of 0.5 for access failure to 1SF201 given seismic-induced failure of the piping.

  For late containment failures, accident times of 38-90 hours are available to make alternate alignments. It is noted that SFP boiling is not expected to limit access during these times (see Access discussion in Section 2).

- Containment isolation failure can be postulated for a seismic-induced SBO and failure of local manual closure of MOVs in the normally open pathway from the containment pumps to the WPB sump tank. Under the postulated seismic conditions, an isolation failure could occur resulting in the potential for release of fission products to the WPB early in a severe accident. For purposes of this SFP evaluation, this failure mode can be treated as a release outside the RAB and FHB. Therefore, the consequential impacts on the Spent Fuel Pool due to the containment isolation failure are best characterized by a "late" failure of containment into the RAB.

4.2.7 Seismic Quantification Summary

The quantification of the seismic analysis was performed using Excel spreadsheet equations. The spreadsheet equations include Boolean algebra where necessary. The spreadsheet calculating approach was employed to facilitate sensitivity calculations, and was possible given the bounding scope of this seismic analysis (e.g., loss of offsite power assumed, like component fragilities assumed completely dependent). The spreadsheets used in the seismic quantification are provided in Appendix G.

The results of the analysis are summarized in Table 4.2-5. The total frequency for SFP cooling and makeup failure due to seismic-induced core damage scenarios is calculated to be 8.65E-8/yr. The largest contribution is from the higher magnitude ranges where the fragilities for key components and structures begin to approach 1.0.

Constraints of the ASLB Order

The ASLB Order has specified a specific scenario to be evaluated. This scenario could be caused by a large number of "initiators" and involve a number of different system and component failures. However, the ASLB Order limits the scope of the question to those events that could lead to evaporation in the SFP and subsequent uncovery of the spent fuel plus an exothermic fuel clad reaction. This scenario excludes those very low frequency, very high magnitude seismic events that induce structural failure of the SFPs and lead to draining of the SFPs because this is not consistent with Step 6 of the postulated sequence. As such, these low frequency set of contributors are not included in the seismic-induced spent fuel failure frequency assessed in this report (i.e., seismic events > 1.50 g).

Seismic Assessment Sensitivity Analysis

Sensitivity analyses were appropriately defined and performed to bound the quantitative results of the seismic analysis. Sensitivity analyses were defined to address key steps of the seismic assessment:

- Seismic hazard curve
- Seismic-induced component/structure fragility
- Early containment failure given core damage
- Human interfaces

Ten separate sensitivity cases were defined and quantified. The results are summarized in Table 4.2-6. Each of the ten sensitivity cases are described below.

- (Sensitivity Case 1) Finer Division of Seismic Hazard Curve: This sensitivity case divides the SHNPP seismic hazard curve into 16 intervals (15 intervals between 0 and 1.5g, and one interval for >1.5g) instead of the Base Case 7 intervals. This sensitivity case tests the impact on the quantitative results from the analysis approach of dividing the seismic hazard curve into discrete intervals, quantifying the risk of each magnitude interval, and then integrating the results. Seismic PRAs typically divide the seismic hazard curve into approximately a half dozen intervals – the approach taken in the Base Case. Sixteen intervals is a comparatively extremely fine division of the curve. The first fifteen intervals are 0.1g wide (e.g., 0 – 0.1, 0.1 – 0.2, 0.2 – 0.3, etc.) and the final interval is defined as >1.5g.

  As can be seen from Table 4.2-6, this sensitivity case resulted in a total frequency of 7.42E-8/yr (a 15% reduction in frequency compared to the Base Case). This reduction is not unexpected; the coarser the division of the seismic hazard curve, the more conservative will be the final integrated results.

- (Sensitivity Case 2) No Extrapolation Beyond NUREG-1488 Hazard Curve: This sensitivity case defines the final seismic magnitude range as >1.0g instead of the Base Case >1.5g. In the Base Case, the point at which the FHB is assumed to fail given the seismic shock (and, thus, fall outside the bounds of this analysis) is 1.5g. However, NUREG-1488 only supplies frequency estimates for seismic events up to 1.0g; as such, a case may be made for defining >1.0g as the final magnitude range and assuming that seismic events beyond this are very low likelihood and highly likely to result in FHB failure.

  As can be seen from Table 4.2-6, this sensitivity case resulted in a total frequency of 5.14E8-/yr (a 40% reduction in frequency compared to the Base Case). This reduction is not unexpected; high magnitude seismic events, although low in frequency, impact the quantitative results due to high component and structural fragilities at such g levels.

- (Sensitivity Case 3) Less Conservative Uncertainty Distribution for Seismic Fragilities: This sensitivity case employs less conservative (0.30 and 0.30) randomness and uncertainty parameters in the fragility calculations instead of the Base Case values of 0.40 and 0.40. This sensitivity case tests the impact on the quantitative results from the estimated randomness and uncertainty in the component and structural fragility calculations. Randomness and uncertainty parameters used in seismic PRAs are typically in the 0.20 to 0.40 range. In certain cases, values as low as 0.10 – 0.20 (e.g., offsite power transformers) and as high as 0.50 – 0.70 (e.g., relay chatter failures) are used. The Base Case employs 0.40 and 0.40 as a suitably conservative set of values. This sensitivity case uses 0.30 and 0.30 to represent a less conservative set of values.

  As can be seen from Table 4.2-6, this sensitivity case resulted in a total frequency of 5.40E-8/yr (a 37% reduction in frequency compared to the Base Case). This reduction is not unexpected; all other issues being equal, the tighter the assumed uncertainty around the estimated seismic capacities, the lower are the calculated fragilities.

- (Sensitivity Case 4) Seismic Capacities Increased Approximately 25%: This sensitivity case employs higher component and structural seismic capacities than used in the Base Case. The Base Case uses component and structural capacities estimated based on review of similar components in other seismic PRAs and knowledge of the SHNPP plant. This sensitivity case tests the impact on the quantitative results given the possibility that the selected capacities used in the assessment are conservative. A factor of approximately 1.25 was assumed in this sensitivity to indicate the comparative level of conservatism existing in the selected capacities of the Base Case.

As can be seen from Table 4.2-6, this sensitivity case resulted in a total frequency of 3.65E-8/yr (a 58% reduction in frequency compared to the Base Case). This reduction is not unexpected; all other issues being equal, the higher the estimated seismic capacities, the lower are the calculated fragilities.

- <u>(Sensitivity Case 5) Seismic Capacities Decreased Approximately 25%</u>: This sensitivity case employs lower component and structural seismic capacities than used in the Base Case. The Base Case uses component and structural capacities estimated based on review of similar components in other seismic PRAs and knowledge of the SHNPP plant. This sensitivity case tests the impact on the quantitative results given the possibility that the selected capacities used in the assessment are non-conservative. A factor of approximately 0.75 was assumed in this sensitivity to indicate a comparative level of non-conservatism that may be postulated to exist in the selected capacities of the Base Case.

  As can be seen from Table 4.2-6, this sensitivity case resulted in a total frequency of 1.62E-7/yr (1.9x the Base Case). This increase is not unexpected; all other issues being equal, the lower the estimated seismic capacities, the higher are the calculated fragilities.

- <u>(Sensitivity Case 6) More Conservative Early Containment Failure Probability</u>: This sensitivity case employs a higher early containment failure probability than used in the Base Case. The Base Case uses a conditional (upon core damage) early containment failure probability of 3.67E-2 based on review of the current SHNPP PRA results. The 3.67E-2 value is the most conservative value of the assessed core damage scenarios. This sensitivity case tests the impact on the quantitative results from a higher early containment failure probability. An approximate factor of 3 is applied to the Base Case value, resulting in a nominal early containment failure probability of 0.10 for use in this sensitivity case.

  As can be seen from Table 4.2-6, this sensitivity case resulted in a total frequency of 1.12E-7/yr (a 30% increase in frequency compared to the Base Case). This increase is not unexpected; early containment failure greatly impacts the human error probabilities associated with providing cooling to the SFPs.

- <u>(Sensitivity Case 7) More Conservative Human Error Probabilities</u>: This sensitivity case employs higher human error probabilities than used in the Base Case. The Base Case generally employs conservative human error probabilities (e.g., 1.00 AC power recovery failure probability, 1.00 manual containment isolation failure probability). This sensitivity case applies a conservative element across the board to all human errors. Human error probabilities less than 0.1 are set to 0.1, and human error probabilities greater than or equal to 0.1 are left at the Base Case value.

  As can be seen from Table 4.2-6, this sensitivity case resulted in a total frequency of 1.46E-7/yr (1.7x the Base Case). This increase is not unexpected; human error probabilities play a key role in the assessed spent fuel failure frequency.

- <u>(Sensitivity Case 8) Less Conservative Human Error Probabilities</u>: This sensitivity case employs less conservative human error probabilities for selected human interfaces in the Base Case. The Base Case generally employs conservative human error probabilities (e.g., 1.00 AC power recovery failure probability, 1.00 manual containment isolation failure probability). This sensitivity case reduces the 1.00 failure probabilities to 0.5 for the following selected actions:

  - AC Power Recovery Failure
  - Containment Manual Isolation Failure
  - Fire Hose Alignment Failure Given Early Containment Failure
  - Fire Hose Alignment Failure Given Containment Isolation Failure

  All other human error probabilities are left at the Base Case value.

  As can be seen from Table 4.2-6, this sensitivity case resulted in a total frequency of 3.86E-8/yr (a 55% decrease in frequency compared to the Base Case). This decrease is not unexpected; human error probabilities play a key role in the assessed spent fuel failure frequency.

- <u>(Sensitivity Case 9) Overall Pessimistic Case</u>: This sensitivity case employs all the attributes of Sensitivity Cases 5, 6, and 7. This sensitivity case is aptly described as the overall pessimistic case.

As can be seen from Table 4.2-6, this sensitivity case resulted in a total frequency of 3.43E-7/yr (4x the Base Case).

- (Sensitivity Case 10) Overall Optimistic Case: This sensitivity case employs all the attributes of Sensitivity Cases 1, 2, 3, 4 and 8. This sensitivity case is aptly described as the overall optimistic case.

  As can be seen from Table 4.2-6, this sensitivity case resulted in a total frequency of 2.06E-9/yr (a 97% decrease in frequency compared to the Base Case).

The sensitivity cases described above, and summarized in Table 4.2-6, show an upper bound of approximately 3.5E-7/yr and a lower bound of approximately 2.1E-9/yr. The majority of the sensitivity cases result in frequencies in the range of 3.5E-8/yr to 1.5E-7/yr (a factor of 2 in each direction around the Base Case).

Sensitivity calculations related to uncertainty in the seismic hazard curve are comparatively easy to assess, as the impact on the results is a straight multiplication of the final frequency. As can be seen from Table 4.2-1, the 85th percentile hazard curve ranges from a factor of 1.9 times higher than the Mean curve (the basis of the Base Case analysis) for low magnitude seismic events to a factor of 1.7 for high magnitude seismic events. Increasing the seismic hazard frequency accordingly in each seismic interval results in a failure of SFP cooling and makeup estimated frequency of 1.48E-7/yr.

Similarly, Table 4.2-1 shows that the 15th percentile hazard curve ranges from a factor of 0.15 times lower than the Mean curve for low magnitude seismic events to a factor of 0.01 for high magnitude seismic events. Decreasing the seismic hazard frequency accordingly in each seismic interval results in a spent fuel failure frequency of 2.29E-9/yr. Assessment of the hazard curve uncertainty confirms the results of the other sensitivity cases, that is, the lower bound is in the low E-9/yr range and the upper bound is in the low E-7/yr range. The Base Case value of 8.65E-08/yr remains the best-estimate for the seismic-induced loss of SFP frequency.

Table 4.2-6

SUMMARY OF SEISMIC ASSESSMENT QUANTITATIVE SENSITIVITY CASES

		Seismic Hazard Curve		Seismic Fragility Parameters								Human Interfaces										
														Fire Hose Align HEP			Demin Align HEP					
Sensitivity Case	Case Description (1)	# Seis Mag Intervals	Magnitude of Final Seismic Range	BETA(r), BETA(u)	EDG Am	Ess SWGR Am	PCIV Am	DFP Am	FHB Flooding Am	Offsite Infrastructure Am	Early Containment Failure Probability	AC Recovery Failure Prob	PCIV Manual Isolation	Early Cont. Failure	Cont Isol Failure	Late Cont. Failure	Early Cont. Failure	Cont Isol Failure	Late Cont Failure	Fire Truck Hook-Up HEP	Portable Pump/Gen Hook-Up HEP	Spent Fuel Failure Frequency (1/yr)
0	BASE Case	7	>1.5g	0.4,0.4	1.25	1.31	2.00	1.25	1.25	1.00	3.76E-2	1.00	1.00	1.00	1.00	0.062	0.10	0.019	0.019	1.00	0.05	8.65E-08
1	Finer Division of Seismic Hazard Curve	18	>1.5g	0.4,0.4	1.25	1.31	2.00	1.25	1.25	1.00	3.76E-2	1.00	1.00	1.00	1.00	0.062	0.10	0.019	0.019	1.00	0.05	7.42E-08
2	No Extrapolation Beyond NUREG-1488 Hazard Curve	7	>1.0g	0.4,0.4	1.25	1.31	2.00	1.25	1.25	1.00	3.76E-2	1.00	1.00	1.00	1.00	0.062	0.10	0.019	0.019	1.00	0.05	5.14E-08
3	Less Conservative Uncertainty Distribution for Seismic Fragilities	7	>1.5g	0.3,0.3	1.25	1.31	2.00	1.25	1.25	1.00	3.76E-2	1.00	1.00	1.00	1.00	0.062	0.10	0.019	0.019	1.00	0.05	5.40E-08
4	Seismic Capacities Increased Approximately 25%	7	>1.5g	0.4,0.4	1.50	1.65	2.50	1.50	1.50	1.25	3.76E-2	1.00	1.00	1.00	1.00	0.062	0.10	0.019	0.019	1.00	0.05	3.65E-08
5	Seismic Capacities Decreased Approximately 25%	7	>1.5g	0.4,0.4	1.00	1.00	1.50	1.00	1.00	0.75	3.76E-2	1.00	1.00	1.00	1.00	0.062	0.10	0.019	0.019	1.00	0.05	1.62E-07
6	More Conservative Early Containment Failure Probability	7	>1.5g	0.4,0.4	1.25	1.31	2.00	1.25	1.25	1.00	1.00E-1	1.00	1.00	1.00	1.00	0.062	0.10	0.019	0.019	1.00	0.05	1.12E-07
7	More Conservative Human Error Probabilities	7	>1.5g	0.4,0.4	1.25	1.31	2.00	1.25	1.25	1.00	3.76E-2	1.00	1.00	1.00	1.00	0.10	0.10	0.10	0.10	1.00	0.10	1.46E-07
8	Less Conservative Human Error Probabilities	7	>1.5g	0.4,0.4	1.25	1.31	2.00	1.25	1.25	1.00	3.76E-2	0.50	0.50	0.50	0.50	0.062	0.10	0.019	0.019	1.00	0.05	3.86E-08
9	Overall Pessimistic Case	7	>1.5g	0.4,0.4	1.00	1.00	1.50	1.00	1.00	0.75	1.00E-1	1.00	1.00	1.00	1.00	0.10	0.10	0.10	0.10	1.00	0.10	3.43E-07
10	Overall Optimistic Case	11 (Note 2)	>1.0g	0.3,0.3	1.50	1.85	2.50	1.50	1.50	1.25	3.76E-2	0.50	0.50	0.50	0.50	0.062	0.10	0.019	0.019	1.00	0.05	2.06E-09

NOTES

(1) Shaded cells indicate parameter changes with respect to the BASE Case

(2) Ten seismic hazard intervals between 0.0 and 1.0g, and one interval for >1.0g

0C0051

Table 4.2-5

SPENT FUEL FAILURE DUE TO SEISMIC-INDUCED CORE DAMAGE SCENARIOS

End State	Seismic Hazard Range (pga)						
	< 0.1	0.1 – 0.3	0.3 – 0.5	0.5 – 0.7	0.7 – 1.0	>1.0	>1.5
Spent Fuel Failure Frequency (per year)	Negligible	2.26E-9	7.40E-9	1.30E-8	2.87E-8	3.51E-8	(1)
Total Spent Fuel Failure Frequency: 8.65E-8/year							

(1) Seismic events in the > 1.5 g magnitude range will result in FHB failure (with a high likelihood) and, as such, are outside the scope of this analysis (refer to discussion at the beginning of this section).

4.3 FIRE INITIATED ACCIDENT SEQUENCES

For fire initiated accident sequences, CP&L used the Electric Power Research Institute's fire-induced vulnerability evaluation (FIVE) methodology, with some variations and enhancements of the FIVE and PSA methodologies, as described in the fire portion of the SHNPP IPEEE submittal [4-4]. CP&L estimated the total fire CDF from the scenarios surviving screening to be 1.1E-5 per year.

The fire initiating events that survived the SHNPP IPEEE screening process are listed in Table 4.3-1.

CP&L estimated that switchgear room A fires contributed 3.1E-6 per year to the CDF, switchgear room B fires contributed 4E-6 per year, and fires in the control room contributed 4.3E-6 per year.

The fire evaluation has considered the dominant contributors to core damage frequency induced by fire initiated accident sequences. The SHNPP IPEEE has evaluated these sequences. The fire initiated Level 1 accident sequences primarily impact the containment via either late containment failures (predominant failure mode) or early containment failures. ISLOCA, containment isolation failure, and SGTR are not numerically significant contributors and fall below the model truncation limit of 2E-10/yr used in this SFP analysis. These dominant contributors have been incorporated into the model and the quantitative results can be propagated through the event tree used to model the SFP evaluation.

Table 4.3-1

IPEEE DOMINANT FIRE INITIATORS

IE Designator	Description
%T17	Fire in 6.9kV Bus 1A-SA
%T18	Fire in 6.9kV Bus 1B-SB
%T19	Fire in 6.9kV Bus 1A-SA (Unsuppressed, propagates)
%T20	Fire in 6.9kV Bus 1B-SB (Unsuppressed, propagates)
%T21	Fire in Main Control Room (Isolation and Annunciator Cabinets)
%T22	Fire in Main Control Room and ACP Shutdown

4.3.1 Fire Model

The file names of the Level 2 containment failure minimal cutsets associated with internal fire-induced initiating events are shown in the Table 4.3-2 below. Table 4.3-2 identifies the containment failure modes that have associated minimal cutsets with non-zero probabilities. Containment failure modes that have zero probability cutsets are not of interest and will not be considered further. The non-zero containment failure minimal cutsets were partitioned into two sets; one for early failure and one for late failure. Each set of cutsets was converted into a logically equivalent fault tree to represent the initiating event for the relevant fire induced SFP event trees.

The two SFP-AETs that were analyzed are F-EARLY.ETA and F-LATE.ETA. Each event tree considers the following events:

- CI: Containment Integrity and No Bypass
- SF: SFP Cooling Operates Successfully
- DM: SFP Makeup from Demin Water System
- RW: SFP Makeup from RWST
- EW: SFP Makeup from ESW

- ALT: Alternate Makeup to SFP
- OS: Offsite Resources or Portable Equipment for SFP Makeup
- ZR: No Exothermic Reaction of Cladding in SFPs C and D

The SFP-AET is described in detail in Appendix D.

Table 4.3-2

FIRE MODEL

Fire Induced Containment Failure Cutsets	Containment Failure Mode Descriptions	Above Truncation Limit (Non-zero)	Fire Induced Containment Failure Frequency [per year]	Fire Induced Spent Fuel Pool Event Trees
F-EARLY.CUT	Early	Yes	2.95E-09	F-EARLY.ETA
F-LATE.CUT	Late	Yes		
F-VLATE.CUT	Very Late	Yes		
F-BASMAT.CUT	Basemat (Late)	Yes	9.77E-07	F-LATE.ETA
F-LGBYP.CUT	Large Bypass	No	0	
F-SMBYP.CUT	Small Bypass	No	0	
F-LGISOL.CUT	Large Isolation	No	0	
F-SMISOL.CUT	Small Isolation	No	0	
F-FAILIV.CUT	In Vessel Recovery	No	0	N/A

4.3.2 Quantification

The Level 2 containment failure minimal cutsets, F-EARLY.CUT, were converted into a logically equivalent fault tree using the CAFTA CUTIL function. This fault tree was used to represent the initiator for the F-EARLY.ETA event tree. The Level 2 containment failure minimal cutsets; F-LATE.CUT, F-VLATE.CUT and F-BASMAT.CUT were combined (merged). The combined cutsets were converted into a logically equivalent fault tree using the CAFTA CUTIL function. This fault tree was used to represent the initiator for the F-LATE.ETA event tree. The event trees were quantified using CAFTA PSAQUANT.

Dependencies Among Operator Actions

It is noted that the multiple HEPs in the cutsets have been examined. These HEPs are determined to be completely different actions, occurring in totally different time frames, and performed by different crews. Therefore, there is considered to be no dependence between the HEP couplets observed in the resulting cutsets.

In addition, a separate sensitivity study to set all operator actions to 1.0 was also preformed. This separate evaluation determined that the cutsets with multiple HEPs exhibited the same character on those in the dominant cutsets. Therefore, no additional dependent failures needed to be applied.

4.3.3 Results

The overall results are shown in Table 4.3-3 below. The frequency of spent fuel being uncovered due to loss of makeup initiated by a fire induced early containment failure is 7.98E-11 per year. The frequency of spent fuel being uncovered due to loss of makeup in the Spent Fuel Pool as a result of fire induced late containment failure is 2.86E-09 per year.

TABLE 4.3-3

FREQUENCY OF SPENT FUEL BEING UNCOVERED IN THE SPENT FUEL POOL AS A RESULT OF FIRE INDUCED CONTAINMENT FAILURE

Initiating Event	Level 1 and 2 Frequency Inputs (per year)	Frequency of Spent Fuel Being Uncovered (per year)
Fire Induced Early Containment Failure	2.95E-09	7.98E-11
Fire Induced Late Containment Failure	9.77E-07	2.86E-09
TOTAL	9.80E-07	2.94E-09

4.4 AN ANALYSIS OF PWR SHUTDOWN RISK

The core damage frequency at PWRs associated with refueling outages has been postulated to be on the same order of magnitude as that associated with power operation. Therefore, the contribution of shutdown initiators to the probability of the Postulated Sequence was evaluated.

4.4.1 Core Damage Frequency

Several industry studies and individual plant analyses have been undertaken to quantify shutdown risk using probabilistic methods. [4-6, 4-7, 4-21] These shutdown risk analyses have been performed on various U.S. and international reactors. The analyses have varied from complete Shutdown PSAs, including the impact of external events, to configuration-based Probabilistic Shutdown Safety Assessments (PSSA).

Currently, the accepted surrogate metric for risk while shut down is CDF. In some studies, the end-state is simplified by using the frequency of the fuel being uncovered, which will be conservative compared to the CDF. Some studies have calculated containment performance (i.e., LERF) and early fatalities, but most studies have not.

One of the key observations from the many shutdown assessments is the wide variation in quantified risk for different plants and different outages. Although some variation is expected from plant to plant, the most striking variations can be seen between similar (or the same) plants, by simply considering different outage schedules or modeling assumptions. That is, shutdown risk is sensitive to the configuration of the plant, the time at which certain activities are performed, and the degree of conservative modeling included in the assessment. The configuration and timing differences are primarily due to the time-varying decay heat levels coupled with changing inventory in the RCS, which causes the time available to recover from initiating events to vary significantly.

However, despite the varied results, it is clear that shutdown risk in a PWR is dominated by loss of shutdown cooling events while the RCS is at reduced inventory. Further, the risk is dominated by the early ("front-end") reduced inventory periods. [4-21] In some studies, as much as 85% of the risk for an outage can be accumulated in a very short time period (e.g., the front-end and mid-loop period).

Adding to the uncertainty of the results is the dominance of human errors in the calculated results. Some studies have found that human errors account for 50% or more of the CDF.

Much of the information and data summarized here is taken from presentations made at the NRC Low-Power Shutdown Workshop, documented in Sandia Report SAND99-1815 [4-7], and other data that was presented or referenced in SECY-00-0007 [4-8]. Additional information is also available from the NRC review of shutdown PSAs. [4-21] This latter document is found to include some PWR estimates of CDF which are higher than currently considered reasonable due to suspected errors in modeling. The NRC-summarized results are used to provide a sensitivity to these calculated CDFs.

Therefore, a review of recent (last 5 years) ORAM PSSA results (for Refueling Outages only) was performed and documented in Tables 4.4-1 and 4.4-2. These risk values are from actual or planned outages at various U.S. and two European plants. In general, the individual plants are not named, but the vendor (W = Westinghouse, CE = Combustion Engineering and B&W = Babcock and Wilcox) is listed in the Plant column. The data described in this report are applicable to the Cold Shutdown and Refueling Modes (5 and 6, respectively).

4.4.1.1 Surry Data from NUREG-6144

NUREG-6144 [4-6] is primarily an analysis of CDF from internal events during mid-loop operations at Surry Unit 1, although it does contain other low power and shutdown conditions.

For Mid-loop conditions, including drain-down events, the CDP for the mid-loop periods is approximately 1.8E-6 (on a per year basis) [From Table S.2 of Reference 4-6]. The calculated error factor on the resultant CDF distribution is about 6.

Recent data [4-9] show that the fraction of the year spent in mid-loop is significantly lower (by approximately a factor of 3) than that assumed in the NUREG/CR-6144 analysis.

4.4.1.2 Low Power Shutdown Workshop Information [4-7]

The following information is summarized from the NRC Low Power Shutdown Risk Workshop held in April 1999 [4-7].

EPRI Perspective

An example PWR Risk Profile was presented with the following attributes:

- Average CDF ~ 1.8E-4/yr
- Peak CDF~1E-3/yr

- Minimum CDF ~7E-7/yr
- CDF/yr due to outage (essentially CDP of the outage) = 2.3E-5/yr
- Contribution from peaks (6 days at 1E-3) ~85%

It was noted that some transition-based initiating events which can have a significant impact on risk, such as loss of level control during drain-down to mid-loop and Shutdown Cooling pump switches, are difficult to quantify.

South Texas Project (STP) Experience

An ORAM PSSA and RISKMAN Shutdown PSA were performed and compared. A detailed review of 11 Plant Operating States (POS) identified differences due to specific modeling assumptions. Once the assumptions were reconciled, the PSSA and PSA provided comparable results.

Front-end mid-loop contributes about 25% of the overall shutdown CDF in 1.5% of the total outage hours. It should be noted that STP's mid-loop period is only about 12 hours long, which is significantly shorter than many other PWR outages.

75% of the total CDF for an outage occurs prior to cavity flooding (i.e., front-end work). Results from the analyses of three STP outages are presented in Tables 4.4-1 and 4.4-2.

Seabrook Shutdown PSA

Shutdown CDF was calculated at approximately 4.5E-5 /yr. The uncertainty range (5^{th} to 95^{th} percentiles) is twice as large as the at-power CDF.

CDF Risk Contributors due to Internal Events (which account for approximately 80% of the total shutdown CDF) are:

- Loss of RHR events with RCS in reduced inventory 71%
- Loss of RHR events with RCS filled 11%
- LOCA/Draindowns 18%

Note that "LOCAs" are primarily due to loss of level control or over-draining events, not pipe breaks. Two areas of concern were noted:

- Level at flange: Low thermal margin
- Level at Mid-loop: Low thermal margin and low margin to RHR pump cavitation.

75% of total CDF for outage occurs prior to cavity flooding (i.e., front-end work).

<u>Scientech Safety Monitor Experience</u>

Outage CDF is considered to be on the order of Level I at-power CDF (~1E-5/yr contribution to cumulative risk). Some observations are:

- High "Risk" Evolutions (e.g., RCS level changes) have a higher instantaneous CDF than at-power, but are offset by short duration.
- Most of the outage is spent in very low "risk" configurations.
- Most of the cumulative CDF comes from low inventory configurations and the first few days of the outage.

<u>Westinghouse Experience</u>

Information about the AP600 Shutdown PSA was presented. In the AP600, CDF for shutdown and low power operations is less than one-third the CDF from at-power events. The majority (85%) of the shutdown CDF *still* comes from events during RCS drained conditions.

Additional insights are:

- Time-to-boiling margin is an important parameter in determining periods of high vulnerability.
- Plant shutdown CDF is dominated by a few periods of high CDF.
- Postulated inadvertent losses of coolant while in modes 5 and 6 (with the cavity not flooded) dominate shutdown CDF.
- Offload of the entire core is a way to reduce CDF.

4.4.1.3 SECY-00-0007 Information [4-8]

This section presents additional information from SECY-00-0007, regarding other (mainly international) shutdown risk analyses.

Several shutdown PSA studies indicate that internal fire and flooding, plus seismic-initiated events, are important contributors to shutdown risk. These contributors are not considered in the CDF results presented in Table 4.4-1. The information presented below is the percentage of shutdown risk which is attributed to various other initiators:

- Sizewell B (UK): 30% Fire, 10% Seismic
- Gösgen (Switz): 30% Fire
- Borssele (Neth): 30% Fire
- Mühlenberg (Switz): 55% Fire/Flood/Seismic
- Seabrook (U.S.): 18% Fire/Flood/Seismic

The Gösgen study also determined that 15% of total shutdown CDF is due to outages other than refueling outages (this is significantly lower than the Surry study [4-6], which showed non-refueling outages to contribute twice the CDF as a refueling outage, at least from the perspective of mid-loop operations, which dominate CDF).

Transition Risk is briefly mentioned. It describes work done by the CEOG which determined that the transition CDF contribution for a plant from shutdown to cold shutdown and return to power is on the order of 1.4E-6 to 2.5E-6/yr. (only two studies were performed). These values were comparable with the at-power CDF for that time period.

Additionally, SECY-00-0007 summarized two other shutdown studies.

NUREG/CR-5015

- Generic PWR Shutdown CDF is approximately 5E-5/year
- Loss of Shutdown Cooling (SDC) events (due to various causes) contributes approximately 80%
- Reduced Inventory contributes approximately 65%
- Operator actions contribute approximately 65% (dominated by reduced inventory scenarios)

NSAC 84

- Zion Shutdown CDF is approximately 1.8E-5/year, but uncertainty is high.
- Operator actions contribute approximately 55% (45% is due to reduced inventory scenarios alone)

4.4.1.4 Industry Experience

Table 4.4-1 provides information on plant-specific shutdown risk analyses using primarily the ORAM PSSA methodology. The CDF information generally includes only internal events (not including flooding). Table 4.4-2 provides information on the mean, median, 5^{th} and 95^{th} percentiles for the data in Table 4.4-1.

Figure 4.4-1 provides a "typical" risk profile for a PWR refueling outage. Note that the scale in Figure 4.4-1 is on a per-hour basis.

Table 4.4-1

SUMMARY OF REFUELING OUTAGE CONDITIONAL CORE DAMAGE PROBABILITY (CCDP) FOR PWR ANNUALIZED

Plant	Outage	Duration (days)	Average CDF (/hr)	CCDP Based on 2 Refuel Per Year (cumulative)	Peak CDF (/yr)
W	B1	65	1.0E-09	8.0E-07	3.5E-04
W	B3	22	4.1E-09	1.1E-06	1.3E-04
CE	D1	NA	NA	1.3E-06	3.0E-04
CE	D2	NA	NA	1.3E-06	2.0E-04
W	B4	38	3.5E-09	1.6E-06	6.1E-04
W	E1	32	5.2E-09	2.0E-06	4.6E-04
CE	C1	24	7.8E-09	2.3E-06	3.9E-04
CE	C4	NA	NA	2.9E-06	NA
B&W	A1	36	6.7E-09	2.9E-06	4.5E-05
B&W	J1	35	9.0E-09	3.8E-06	7.9E-04
CE	C2	NA	NA	4.5E-06	NA
CE	C3	NA	NA	5.5E-06	NA
W	F1	26	1.8E-08	5.5E-06	2.0E-04
W	F2	45	2.1E-08	1.2E-05	NA
W	G1	33	4.2E-08	1.7E-05	1.8E-03
STP	1RE07	20	8.2E-08	2.0E-05	NA
STP	2RE06	19	8.7E-08	2.0E-05	NA
STP	1RE08*	28	6.3E-08	2.1E-05	NA
W	B2	48	9.4E-08	5.5E-05	1.8E-02

Effective Average CDF is the CDF accumulated during the outage (outage average CDF * outage duration).

Peak CDF is the Instantaneous CDF (on a per year basis) of the highest risk peak during the outage (typically the front-end mid-loop).

Statistical Information on this data is provided in Table 4.4-2.

Table 4.4-2

SUMMARY OF CCDP FOR PWR REFUEL OUTAGES

	CCDP[1] (cumulative)	Peak CDF (/yr)
Mean	9.5E-06	1.9E-03
Median	3.8E-06	3.7E-04
5th Percentile	1.1E-06	9.3E-05
95th Percentile	2.5E-05	8.8E-03

(1) Conditional Core Damage Probability based 1 refuel outage every 2 years

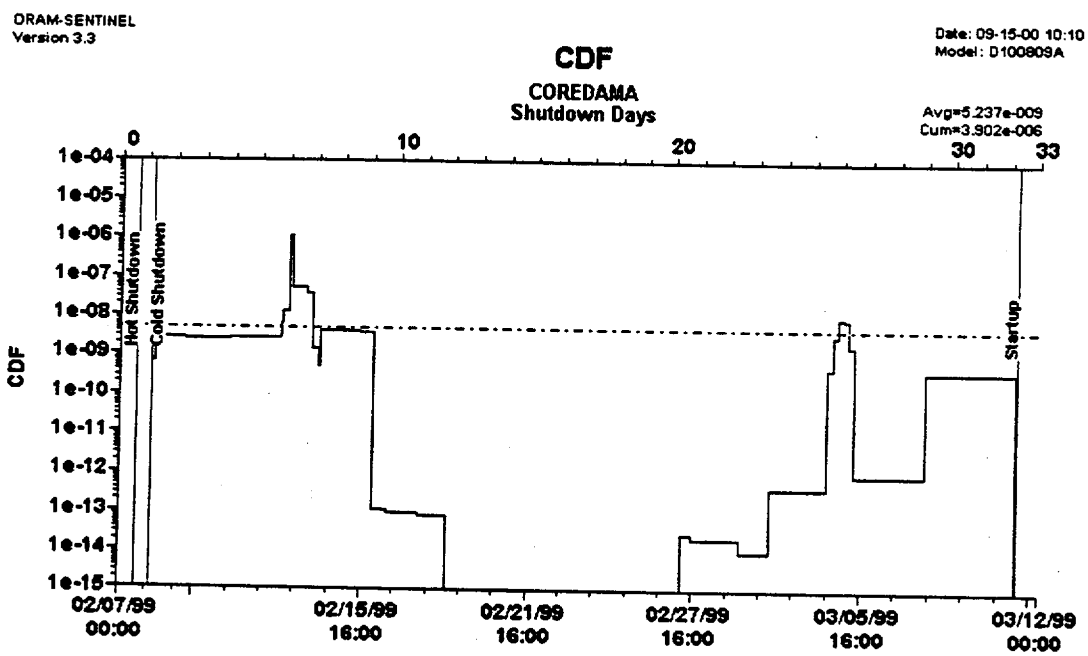


Figure 4.4-1 Typical PWR Refuel Outage CDF Profile

4.4.1.5 Summary of CDF Associated with Refueling Operations

CDF is the most common risk metric used to quantitatively evaluate shutdown risk. Outage risk varies considerably from plant to plant and outage to outage. Evaluation of the available data indicates that the contribution to annual CDF due to a PWR refueling outage is on the order of 1E-5/yr, but can be as high as 5E-5/yr. This includes only internal events. There are indications that considering fires, floods and seismic events may add up to 50% more to the total CDF.

It can be further concluded that shutdown risk in a PWR is dominated by periods of low inventory, especially early in the outage when the decay heat level is still relatively high. The contribution of the "front-end and mid-loop" period to overall outage risk could be as high as an 85% contributor. Operator failures to recover from an event and/or initiate alternate methods of heat removal can contribute as much as 50% to the total risk.

Uncertainty in the results is higher than for comparable at-power studies. The uncertainty is driven mainly by human error probabilities and "transition" type initiating events (such as draining the RCS to mid-loop).

4.4.2 Containment Integrity During Refuel Operations

The purpose of this portion of the evaluation is to develop an estimate of the probability of containment integrity and potential for radionuclide release given a core damage event has occurred during shutdown.

4.4.2.1 Overview of Containment Integrity During Refuel Operations

At SHNPP containment integrity is strictly controlled during refueling outages. This control is provided by several plant procedures as well as plant technical specifications. The plant procedures as well as the applicable technical specifications dictate the actions to be performed, the conditions under which the actions are required, the individual required to perform the actions, and the timeframe in which the action must be performed.

The requirement for containment integrity during refueling conditions is part of the defense-in-depth philosophy of the conduct of refueling outages at the SHNPP [4-10]. In summary, the Outage Shutdown Risk Management procedure (reference [4-10]) requires that:

- Primary containment integrity be maintained any time the RCS temperature is greater than 200° F (Mode 4 and above).
- Containment access doors, PAL, EAL, and equipment hatch may be opened during Modes 5, 6 or defueled. During Modes 5 or 6, they shall be capable of being closed within the more restrictive of either prior to core boiling OR within 4 hours. Additional requirements exist to meet Technical Specification 3.9.4 requirements and GL 88-17 requirements.

Plant Condition	Duration of Plant Condition[1]
Head on	8 days
Head off normal	5.5 days
Head off Mid-loop	1.0 days
Head off Hi water level	6.5 days
Defueled	6.0 days

The projected SHNPP "Standard" Refuel outage is 27 days.

(1) (Based on an e-mail from J.D. Cook (CP&L) to Bruce Morgen, dated October 5, 2000)

4.4.2.2 Methodology

The likelihood that the containment is not isolated following core damage is estimated in a three (3) step process.

- In the first step, the likelihood of core damage in various refueling outage plant configurations is estimated. This step is accomplished and documented in Reference [4-8].
- In the second step, the procedures associated with the control of primary containment integrity during outages are reviewed.
- In the third step, human action error probabilities are estimated for the likelihood that plant personnel restore containment integrity prior to radionuclide release. These human error probabilities are calculated using the SHNPP procedures for control of shutdown risk [4-10, 4-11].

This evaluation serves as input to the development of an event tree which assesses the overall likelihood of radionuclide release from the Shearon Harris Nuclear Generating Station.

4.4.2.3 Evaluation of The Likelihood Of Containment Integrity

The probability associated with fuel damage during shutdown conditions is dependent on the plant configuration. For example, the probability of fuel damage during reduced RCS inventory states can be higher than those plant states where inventory is not reduced.

Because the probability of fuel damage varies in timing and magnitude based on plant configuration, it is appropriate that the human error probability associated with the restoration of achieving primary containment also vary with plant configuration. It should be noted that the likelihood of achieving primary containment integrity for a severe accident is being based on the assumption that the containment is open for refueling activities. Therefore, the probability of successful containment integrity for a severe accident is based on the performance of human actions to restore integrity. This assumption is conservative because it is possible that containment is isolated during mid-loop operation.

Because upwards of 85% of the fuel damage risk can be associated with "front-end" conditions or "mid-loop" operation [4-8], it is appropriate to assess the likelihood that containment is intact or can be restored to intact within an acceptable duration. The additional 15% of fuel damage risk is from a variety of plant configurations associated with lower decay heat levels and higher initial RPV water levels. In these cases, additional time is available for human actions associated with the restoration of primary containment integrity.

Therefore, two potential human actions error likelihoods for the restoration of primary containment integrity are estimated. The first is associated with the more restrictive conditions of mid-loop operation. The second human error likelihood is non-drained down conditions (i.e., normal RCS water level) where more time is available for the restoration of primary containment integrity. It should be noted that the use of "normal RCS water level" for the timing of the human error probability is conservative since some of the fuel damage risk is from cavity flooded configurations in which significantly more time is available.

4.4.2.4 Containment Integrity Human Error Likelihood (mid-loop operation)

During mid-loop operation, less time is available to perform the actions associated with the restoration of primary containment integrity. However, from a review of Shearon Harris procedures, much more restrictive requirements are placed on the plant activities during mid-loop operation. From Reference [4-11], the following conditions apply during mid-loop operation:

Containment Closure

1. *Containment penetrations including PAL, EAL, and Equipment Hatch, may be opened during reduced inventory or mid-loop. Penetrations shall be capable of being closed within the more restrictive of the following:*

   a. *Within 2.5 hours of initial loss of decay heat removal. This time is reduced if the following apply:*

      1. *If openings totaling greater than one square inch exist in the cold legs, RCPs (connecting into the cold leg water space) and crossover pipes of the RCS, this time is reduced to 30 minutes.*

      2. *If the Reactor Head is removed or installed but not yet tensioned, the 30 minutes does not apply, instead the time limit is 2 hours.*

   b. *Within the time to core uncovery from a loss of decay heat removal coupled with an inability to initiate alternative cooling or addition of water to the RCS.*

   c. *Within the time to core boiling.*

In general, the time to core boiling remains the most restrictive time when in mid-loop or reduced inventory conditions. Times to boil have been estimated in various literature sources. Table 4.4-3 illustrates the time to core boiling as well as the time to uncover the core based on a sampling of industry data.

Table 4.4-3
REPRESENTATIVE TIME AVAILABLE FOR ACTIONS

Shutdown Condition	Time to Boil	Time to Uncover Core
Normal RCS Water Level	0.5 hrs	6.5 hrs
Mid-loop Operation	0.2 hrs	1.2 hrs
Cavity Flooded	10 hrs	100 hrs

* Representative data based on TMI, STP, and Diablo Canyon shutdown evaluations.

Other sources of data [4-9, 4-21] have indicated approximately the same duration to core boiling for mid-loop operation ranging from a low of 9 minutes to a high of 24 minutes with an average of 15 minutes, also based on industry experience. However, the most important time is the time to uncover the core which is assumed to be equivalent to the time of adverse consequence. (This may be conservative.)

From Table 4.4-3 it can be assumed that approximately 15 minutes are available before bulk core boiling and an additional 60 minutes before the onset of adverse consequences during reduced inventory or mid-loop operation.

Various indications are available following the loss of RCS cooling during mid-loop conditions. The indications are generally dependent on the type of loss of RCS heat removal. However, these indications generally include control room indication of a failed pump and system temperature alarms (e.g., RHR, CCW or ESW), increased humidity and temperature in the primary containment, and visual verification of bulk boiling inside the reactor vessel.

Actions which would precede or are concurrent with attempts to restore primary containment integrity include those actions associated with the restoration of heat removal and/or RCS inventory makeup.

It can be assumed that sufficient personnel are available to perform the required action. This assumption is based on the procedural guidance that requires dedicated personnel for each containment penetration that is open during reduced inventory or mid-loop operation. In addition, refueling outages generally have outage command centers or work control centers which can provide additional personnel support should the need arise.

The quantification of this human action is divided into two phases. The first phase involves the diagnosis of the off normal event. The second phase of the quantification involves the quantification of error rates associated with the actual performance of the actions required. A detailed description of the quantification methodology is available in Reference [4-3].

The human error probability associated with the failure to successfully restore containment integrity during mid-loop operation was determined to be 1.1×10^{-2} per demand. This is a relatively high failure probability given the explicit guidance and the required ability to close the containment within a very short time.

4.4.2.5 Containment Integrity Human Error Likelihood (Normal RCS Level)

During normal RCS level or reactor cavity flooded conditions additional time is available for plant staff to restore containment integrity. However, at the same time the number of containment penetrations which are open is generally greater than during mid-loop operation. In addition, it can be assumed for analysis purposes that plant staff may not be as vigilant to the RCS conditions as in the case in mid-loop or reduced inventory conditions.

From Table 4.4-3, approximately 30 minutes are available before core boiling and an additional 6 hours before uncovery of the core (representative data taken from TMI, STP and Diablo Canyon shutdown evaluations).

As in the case with mid-loop or reduced inventory conditions, various indications are available following the loss of RCS cooling. The indications are generally dependent on the type of loss of RCS heat removal. However, these indications generally include control room indication of a failed pump and system temperature alarms (e.g., RHR, CCW or ESW), increased humidity and temperature in the primary containment, and visual verification of bulk boiling inside the reactor vessel.

Actions which would precede or are concurrent with attempts to restore primary containment integrity include those actions associated with the restoration of heat removal and/or RCS inventory makeup.

It can be assumed that due to the workload and command centers generally present during outages, as well as procedural guidance containing staffing requirements, sufficient dedicated personnel are available for the performance of the action.

As in the case with the mid-loop condition evaluation, the quantification of the restoration of containment integrity during normal RCS level error probability is divided into two phases. In the first phase the diagnosis of the off normal event is evaluated and in the second phase the actual performance of the action is evaluated. A detailed description of the human error probability evaluation method is contained in Reference [4-3].

The human error probability associated with the failure to successfully restore containment integrity during normal RCS level was determined to be 1.6×10^{-2} per demand.

The basis for the higher value during normal RCS level conditions are the assumptions contained in the detailed evaluation. In the normal RCS level condition, additional penetrations are assumed to be open; and therefore, although there is more time to perform the required actions, there is also a larger potential for error.

4.4.3 Summary of Quantitative Results

The quantitative results of this generic assessment identify a generic estimate of CDF of 2.5E-5/yr based on a 2 yr refuel cycle. This leads to the cases identified in Table 4.4.3-1 where 85% of the risk is associated with 6 days (including the 1 day of mid-loop operation. The CDF is developed using the configuration specific CDF (on a per-hour basis); then, multiplied by the number of hours encountered over a two-year period; and finally treated in the analysis as an annualized probability or a frequency per reactor year. Because mid-loop operation occurs for a much shorter time duration, the annualized CDP (or CDF) is less than that for the other activities occurring early in the refuel outage.

The containment isolation failure probability is the conditional probability of the failure to reclose the containment given a shutdown event is in progress that requires containment isolation. These conditional failure probabilities are dominated by the Human Error Probability calculated for these actions.

Table 4.4.3-1

SUMMARY OF QUANTITATIVE RESULTS

Condition	CDF[1] (per Rx yr)	Containment Isolation conditional Failure Probability	Core Damage with Containment Isolation Failure (per yr)
Normal RCS Level (early in outage)	1.8E-5	1.6E-2	2.9E-7
Mid LOOP Operation	3.5E-6	1.1E-2	3.9E-8
Cavity Flooded	Negligible	1.6E-2	Negligible
"Other" Draindown	3.8E-7	0.9	3.4E-7
"Other" Non-Draindown	3.4E-6	1.6E-2	5.4E-8
Total Core Damage with Containment Isolation Failure			7.2E-7

[1] A higher CDF than observed as the "average" is chosen. This may introduce some conservatism in the evaluation of the shutdown related SFP boiling and fuel exposure.

4.5 OTHER EXTERNAL EVENTS

The SHNPP IPEEE analysis of the impact of external events - other than fire and seismic - concluded that there are no other significant events that need to be quantified. A comprehensive screening analysis of the external hazards identified in the PSA Procedures Guide confirmed the NUREG-1407 conclusion that only high winds, external floods, transportation and nearby facility accidents had to be reviewed in detail. This review considered high winds, tornadoes, hurricanes, external floods, aircraft impact, road and rail accidents, fixed industrial facility accidents, fixed military facility accidents and pipeline accidents. For all these cases, the review concluded that the SHNPP design is conservative by a substantial margin and capable of withstanding all credible hazards associated with these other external events.

The "other" external events are not judged not to have a substantially different character than those already accounted for in the spectrum of severe accident challenges quantitatively assessed in this report. None of these external events is judged to have a significant contribution to either CDF or containment failure. Therefore, if quantified, based on the substantial margins of safety at SHNPP, these contributors are judged to contribute less than 1% of the risk calculated for the other contributors.

Section 5
RESULTS AND SENSITIVITIES

5.1 INTRODUCTION

This section discusses the results of the SHNPP spent fuel pool (SFP) best estimate probabilistic analysis of the seven step Postulated Sequence admitted as a contention in the SHNPP license amendment proceeding. However, in addition, it is judged vital to the decision-makers to provide a characterization of the uncertainty associated with the Base Case evaluation. Therefore, this section also addresses how the uncertainty should be characterized.

5.2 OVERVIEW OF UNCERTAINTY

The Best Estimate is used for decision making because the use of upper bounds (or lower bounds) may introduce biases into the decision making process that are not properly characterized, i.e., the biases may be unevenly applied (widely varying levels of conservatism) with the resulting upper bound yielding a distortion of the importance of individual components of the analysis and potentially of the overall results. Such biases could then lead to improper decisions regarding the importance of individual elements of the analysis. It may also lead to the improper allocation of resources to address conditions or postulated events that have been "conservatively" treated in an upper bound evaluation. Therefore, all prudent evaluations have been included to achieve the Best Estimate characterization.

This Best Estimate analysis is provided in the enclosed evaluation. It is noted, however, that there remain inherent conservatisms in the deterministic calculations, the models, and the assumptions. These "conservatisms" are not able to be extricated from the analysis because the current state of technology is not sufficient to remove them. For example, the assumption that the probability of an exothermic reaction in the SFP is 1.0 is considered to be a default estimate, recognizing both the current state of the technology for calculating the probability of such an SFP exothermic reaction and the low probabilities of the six steps leading to uncovering the spent fuel in SFPs C and D. In light of the information provided by CP&L relating to the "age" of the spent fuel after discharge from the reactor that is to be stored in SFPs C and D, the assumption that an SFP exothermic reaction will occur with a probability of 1.0 is judged to be a conservative assumption. CP&L has addressed qualitatively how unlikely such an exothermic oxidation reaction would be in SFPs C and D. (See Affidavit of Robert K. Kunita.)

The NRC, its contractors, and the industry have committed substantial efforts to the understanding of uncertainties in nuclear power plant risk analyses. These efforts have led to methods development, understanding of the contributors to the uncertainty distributions, and the identification of alternative ways to provide decision makers with effective ways of characterizing the risk spectrum.

There are several sources of uncertainty and several viable ways of categorizing these sources. A simple three category approach is used here [4-22, 4-23]. Each category is then further developed to illustrate more specifically those sources of uncertainty assigned to each category.

The three types or categories of uncertainties are generally considered to be the following:

- <u>Quantification:</u> The related contributors to the so-called "quantification" uncertainties include the following:
  - Failure rate models
  - Applicability of data
  - Statistical variation of parameters
  - Processing simplifications or truncations

- Logic Modeling: The related contributions to logic modeling uncertainties include the following:
  - Adequacy of details
    - Hardware, including instrumentation
    - Human interaction
    - Environmental/spatial
    - Equipment wear out
    - Applicability of data
  - Logic correctness
    - Success criteria
    - Event sequences
    - Systems analysis
  - Dependencies (initiating events, intercomponent, intersystem, functional, environmental, human, and physical similarity)

  Analysis of this category of uncertainties evaluates whether, given the scope of the evaluation, the implementation resulted in models capable of supporting the results, conclusions, and expected use in the support of decisions.

- Scope and Completeness: The considerations include the following:
  - Initial plant conditions (e.g., configurations)
  - End states
  - Inter-unit connections
  - Initiating events
  - Success criteria
  - Event sequence
  - Systems analysis
  - Failure modes and causes
  - Human interaction and errors of commission
  - Data
  - Design deficiencies

Analysis of this category of uncertainties evaluates whether the specific scope is sufficient to support the types of conclusions and decisions reached, and how scope limitations affect the results, conclusions, and decisions that can be supported.

Folded into each of the categories are a set of attributes. These attributes can affect the evaluation of the uncertainty and include the following:

- Plant-Specific: .

  Plants vary in hardware, personnel, procedures, organizations, management, training, etc. These major factors modify the uncertainty associated with accident sequences in each category.

- Time-Varying:

  A specific plant's characteristics will change as a function of plant life due to changes in plant hardware, training, procedures, management, equipment degradation, and aging.

- Sequence-Specific:

  Each accident sequence has unique characteristics that can profoundly affect the ability to quantify the likelihood of such sequences. The sequences vary in the complexity of operator actions, the specific hardware failures, etc.

There are several principles regarding the treatment of uncertainties in probabilistic analyses which have some consensus in the industry. They are identified here to provide a foundation for the scope of this uncertainty evaluation. These principles are as follows:

- The purpose of the uncertainty evaluation is to focus attention on important assumptions.

- Establishing a risk framework for the discussion of point estimate values and their uncertainties provides decision makers additional input.

- The uncertainty process should be usable as an engineering tool to enhance the confidence in the conclusions.

- Attempts to provide a quantitative perspective on uncertainty that is very costly and does not fully support the real objectives of establishing the validity of the conclusions of the assessment or application should be avoided.

- A reasonable, credible range in which the actual value will be found (90 percent degree of belief) is a desirable quantitative measure.

- A Probabilistic Safety Assessment (PSA) process is an engineering applications tool. Therefore, the uncertainty evaluation should be structured in a similar fashion to take maximum advantage of the available engineering insights and to add to those insights. The structure of the approach need not be a rigid formalism, but can, rather, borrow its justification from other published discussions such as the use of a subjectivist approach in risk assessment.

The conclusion from this overview is that the use of focused sensitivity evaluations to characterize the change in the results as a function of changes in the inputs provides a physically meaningful method of conveying the degree of uncertainty associated with the analysis. Therefore, sensitivity cases were developed that portray the changes in the Postulated Scenario frequency as posed by the ASLB, if input variations occur.

The key variations in the 21 sensitivity cases examined address the three categories of uncertainties cited above and adhere to the principles of an effective uncertainty evaluation:

- Quantification: Vary the input accident sequence frequencies and system configuration – See Cases A.1, A..2, A.4, and seismic cases 5.1 through 5.10.

- Logic Modeling: Vary success criteria, human interaction effectiveness, environmental factors, system reliability and dependency effects – See Cases A.3, B.1, B.2, B.3, B.4, B.5, and B.6.

- Completeness: Vary phenomenological effects – See Case C.1.

5.3 SENSITIVITY CASE

The measure of risk used in these analyses is the frequency of the Postulated Scenario (steps 1 through 6). All tables in this section use this parameter to characterize the risk.

The best estimate of the frequency of the loss of effective cooling to the spent fuel has been constructed within the current state of the technology. There are some assumptions that have been included in the model construction and quantification that may introduce some conservatisms. These have been discussed in Section 2.5 and are summarized in the conclusions, Section 6.

The quantitative results are properly considered in two groups: (1) internal events and (2) external events and shutdown events. For internal events, there is high confidence in the models and the evaluation of the SHNPP SFP response to the Postulated Sequence. Most of the effort focused on assessing the impact of the internal events because they are the most studied and lead to the highest frequency of core damage. The results of the internal events initiated sequences indicate that the loss of effective SFP water cooling occurs at a best estimate frequency of 2.65E-8/yr.

The external events and shutdown events were also evaluated to determine whether these events alter the conclusion determined based on the internal events assessment. It is recognized that the uncertainties associated with these sequences are greater than those in the internal events analyses. Consequently, several conservativisms were incorporated in the modeling, which produced inflated point estimate values. Thus, these results are not entirely a "best estimate" because of the conservatisms found in the existing models and generic studies.

Thus, the calculated best estimate annualized probability of the Postulated Sequence based on the internal events analysis is 2.65E-8. This "best estimate" includes the conservative assumption that the conditional probability of step 7 is 1.0. There are also other conservatisms included in the analysis because of the difficulty of removing embedded conservatisms from existing analyses. For example, the time to recover from the loss of cooling to the spent fuel pools was assumed to be four days, based on the maximum heat load in spent fuel pool A after discharge of fuel during refueling. A best estimate calculation could have integrated the reduction in decay heat load over the length of a normal fuel cycle. However, the probability of the Postulated Sequence was already so low, even with numerous conservatisms, that further analysis to refine the calculation was not justified.

The analysis from Section 4 is summarized in Table 5-1, indicating the probability of the Postulated Sequence from internal, fire-induced, seismic and shutdown events. Although this analysis concluded that the best estimate of the probability of the Postulated Sequence is represented by the contribution of internal events only, a composite case was created for the purpose of performing sensitivity analyses. This composite case, Case A, includes the best estimate probability as well as the contribution from the other identified contributors to severe accidents. Results from the sensitivity analyses can then be compared to Case A to determine the relative impact that variations in input parameters have on the overall estimate of the frequency of the Postulated Sequence.

5.4 SENSITIVITY EVALUATION

There are uncertainties associated with any probabilistic model. The purpose of this section is to address selected uncertainties that may have a substantial impact on the calculated frequency of SFP cooling under the postulated scenario. The sensitivity cases are used to explore those quantitative inputs, modeling, or completeness issues that could vary substantially and influence the results.

The general topics for the sensitivity evaluation include the following:

- Level 1 and 2 Severe Accident Frequencies
- System capabilities during severe accidents
- Plant Configuration
- Operator Actions during severe accidents
- Seismic response capabilities
- Exothermic reactions probability

The sensitivity cases related to each of these are discussed in the following text. It is noted that although the seismic accident sequence sensitivities are discussed last in this section, they are used in the evaluation of each of the other sensitivity cases identified above.

<u>Level 1 and 2 Severe Accident Frequencies (Cases A.1 and A.2)</u>

The frequency of a severe accident (core damage) caused by internal events that can lead to core damage and containment failure or bypass has an uncertainty associated with it. The calculated core damage frequency for SHNPP has an estimated uncertainty characterized by a lognormal distribution with an Error Factor of approximately 6 based on comparison with the NRC analysis in NUREG-1150.

This is characterized as follows:

Characterization	Internal Events Frequency (per yr)
95% Upper Bound	2.5E-5
Mean(1)	7.66E-6
Median	4.22E-6
5% Lower Bound	7.02E-7

Lowed monte carlo analysis propagate through analysis 6-10 error bnd. no. n

Two sensitivity studies are used to demonstrate the impact of considering variations in the quantitative inputs to the SFP analysis by using the 5% and 95% bounds for these inputs. These two sensitivity cases are discussed below.

Varying the accident sequence frequencies for Steps 1 and 2 of the ASLB Order can be performed by changing the frequencies to their 5% (Case A.1) or 95% (Case A.2) bounds. See Tables 5-2 and 5-3 for the lower and upper bound evaluation results, respectively. Note an exception to the above characterization of the uncertainty range is for an ISLOCA. The ISLOCA frequency upper bound has been estimated at approximately 50 times its point estimate value as an upper bound rather than approximately 3 for other sequences.

Monte carlo calc. single worse event

System Capabilities During Severe Accidents (Case A.3)

The performance of systems during severe accidents can be degraded by the adverse environmental conditions. For the Base Case evaluation, the systems exposed to adverse environments have had their performances adversely impacted in most sequences. In one protected area, equipment is assigned a high probability of reliable operation. The one area is the 6.9KV switchgear rooms to provide offsite power to the demineralized water pumps. If a pessimistic modeling of the 6.9KV switchgear is included in the probabilistic analysis, then an estimate of the impact can be made in Case A.3. (see Table 5-4).

Plant Configuration (Case A.4)

The plant configuration that is not explicitly modeled in the probabilistic model is the possibility that gates either between A and B SFPs or between C and D SFPs are in place.

The Base Case evaluation is performed with the specified SFP configuration. In particular, the probability that the gates are installed in their normal configurations as described in Appendix A is assigned a value of 1.0. However, there is a small probability that maintenance could be required that would result in installation of Gates 3 or 4 for the A and B SFPs or Gates 7 or 9 for the C and D SFPs.

The effects of these configuration changes are to isolate the following:

- SFP A from SFP B - Gate 3 or 4.
- SFP C from SFP D - Gate 7 or 9.

However, the probability of these configurations is estimated to be no larger than 1% of the time for each gate. A sensitivity can be performed to demonstrate the effect of having the gates installed for the maximum of 1% of the time. The sensitivity inputs are:

- Gate 3 or 4 installed 1% of the time.
- Gate 7 or 9 installed 1% of the time.

(1) Mean frequency of core damage and containment failure or bypass calculated in the SHNPP Level 1 and 2 PSA for internal events.

- The time to boil (SFP A) in the worst case is reduced from 20 hours to 6 hours in the worst case.
- The time to uncover fuel (SFP A) in the worst case could be reduced from 6 days to approximately 2 days.
- The HEP for action to align the makeup systems could become higher because of the reduced time available to take effective action. Upon reviewing the HRA, it is found that the HEP increases by a factor of less than 1.25 for each of critical actions (or 1.56 for coupled actions).

The result of these changes can be compared with the Base Model. The Base Model calculation was for the frequency of a radionuclide release from the SFPs with the subject gates always removed; i.e., the frequency of radionuclide release for the 2% of the time that the gates are in place is not increased.

Base Case

- $F^{B}_{Release} = 0.98 * X + 0.02 * X = 1.0X$

Where X = the calculated frequency of radionuclide release with the Base Case configuration (Gates Out)

Sensitivity Case with Gates In for 1% of Time in A and B and 1% of Time in C and D

- $F^{S}_{Release} = 0.98 * X + 0.01 * Z + 0.01 * Y$

Where:

Z = the calculated frequency of radionuclide release with the Gate configuration such that A and B are isolated from each other Z = 1.56 * X, based on increased human error probabilities due to decreased time available to respond effectively.

Y = the calculated frequency of radionuclide release with the Gate configuration such that C and D are isolated from each other Y = 1.56 * X, based on increased human error probabilities due to decreased time available to respond effectively.

- $F^{S}_{Release} = 0.98 * X + 0.01 * 1.56\ X + 0.01 * 1.56\ X$
- $F^{S}_{Release} = 1.01\ X$

This indicates that explicit treatment of the gates in the model would result in approximately a 1% increase in the calculated frequency of the SFP fuel being uncovered. The increase is so small because of the small probability of the configuration being present and the relatively small impact on the calculated operating crew and TSC response.

Operator Actions During Severe Accidents (Cases B.1, B.2, B.3, B.4, B.5, B.6)

The human action portion of the analysis is crucial to the Best Estimate characterization of SFP cooling following the postulated severe accidents. This is because human intervention is required to prevent evaporation from the SFP's. In order to address this crucial area of the analysis, there are a series of sensitivity cases that are performed to characterize the human interface. These include the following:

- Explicit TSC Guidance - Case B.1
- Access Compromised for ISLOCA, but with explicit TSC Guidance - Case B.2
- Access Compromised for ISLOCA and Upper Bound ISLOCA frequency, but with explicit TSC Guidance - Case B.3
- All human actions included at pessimistic failure probabilities - Case B.4
- Reasonable probability estimates of human actions - Case B.5
- Pessimistic impacts of the on-site radionuclides - Case B.6

Table 5-5 provides the operator action HEP's for cases B.1, B.2, and B.3. These human interface sensitivity cases are described in more detail as follows:

- Case B.1: The use of Best Estimate operator responses given the condition that explicit guidance for the TSC exists to support the alignment of makeup sources at an early time frame. There is some uncertainty regarding the timing and cues that would trigger the use of non-proceduralized and proceduralized actions in aligning makeup to the SFPs. The largest impacts are those associated with the internal events analysis. Overall a reduction of a factor of two in the calculated frequency of uncovering spent fuels is found if more explicit guidance is provided to the TSC than currently exists. [Table 5-6 provides the results.]
- Case B.2: This is the same as Case B.1, except an additional consideration is included that prohibits access to the 216' El North of the FHB due to radiation levels under ISLOCA conditions. The ISLOCA is one of the severe accidents that is being explicitly quantified consistent with the postulated sequence in the Board's Order. The ISLOCA sequence is calculated to be of low frequency and have potentially high offsite consequences. It also has severe

CC0888

effects on the RAB and FHB environments. These severe effects include adverse effects on personnel access and equipment operability which in this sensitivity case preclude the successful mitigation of the event by access to the FHB within 96 hours.

The sensitivity indicates that if the ISLOCA causes a sufficiently high dose to preclude access to the FHB 216'El North, it results in a 30% increase in the internal events contribution to the loss of effective spent fuel makeup. [Table 5-6 provides the results.]

- Case B.3: The same as Case B.2, except that the frequency of the ISLOCA core damage sequences uses the upper bound estimate of ISLOCA frequency which is slightly larger than the older (out of date) IPE analysis. The frequency of ISLOCA has a noteworthy impact on the frequency of the interruption of effective spent fuel cooling. The increase in ISLOCA frequency by a factor of 50 (upper bound) coupled with the limited access to the FHB assumption will lead to a total frequency of loss of SFP cooling and makeup of approximately 4.8E-7/yr. This means that the ISLOCA frequency and its effect on personnel access are some of the key inputs to the quantitative assessment of risk. [Table 5-6 provides the results.]

- Case B.4: All the human actions included in the post containment failure time frame for SFP boiling mitigation are set to 0.1 (or to 1.0 if they are 1.0 in the Base Case). This does not apply to responses where the containment has not failed. Table 5-7 summarizes the HEP's that are used in this sensitivity case. Table 5-8 provides the results of this sensitivity case.

- Case B.5: All the human actions included in the post containment failure time frame for SFP boiling mitigation are set to IE-3 (or to 1.0 if they are 1.0 in the Base Case). Table 5-9 summarizes the HEP's that are used in this sensitivity case. Table 5-10 provides the results of this sensitivity case.

- Case B.6: This sensitivity case represents a pessimistic evaluation of the radionuclide release from the containment. It includes the following:

Accident Type/ Containment Failure Mode	Probability		Site Access for Restoration of Makeup (OPERZOFFST)
	No Access to FHB 286'El.	No Access to FHB 216'El.N.	
SGTR	1.0	0.0	0.5
ISLOCA	1.0	1.0	0.5
Containment Isolation Failure	1.0	0.0	0.5
Early Containment Failure	1.0	1.0	0.5
Late Containment Failure	0.0	0.0	0.5

The purpose of this sensitivity case is to examine under pessimistic meteorological conditions and conservative plume modeling whether effective actions can be taken to provide mitigation. The results indicate that inhibiting access to critical areas of the FHB, the intake structure, and the cooling tower basin due to external plume effects could result in an increase in the frequency of the SFP evaporation and uncovering of the spent fuel by a factor of 4.7. Table 5-11 provides the results of this sensitivity case.

Exothermic Reaction Probabilities (Case C.1)

- Case C.1: A Best Estimate analysis would treat the SFP exothermic reaction in Pools C and D in a way that minimizes the maximum error that can occur given our current state of knowledge for this event. Analytic evidence indicates the possibility of such a reaction under high decay heat and high burnup. Spent fuel in SFP C and D, however, is not consistent with these preconditions. Therefore, the probability of 0.5 would be justified because it will minimize the maximum error that can be made.

Table 5-12 summarizes the results of this evaluation using the Case A characterization of Steps 1-6.

Seismic Response Capabilities

There are also a number of seismic related sensitivities performed to demonstrate the approximate uncertainty bounds on the seismic accident sequences.

Section 4.2 has identified the sensitivity cases to be discussed here. They are summarized in Table 5-13 and are discussed individually regarding their seismic contribution and also how they relate to the other sensitivity cases, A.1 to A.4, B.1 to B.6, and C.1.

The initial statement regarding seismic uncertainties is that the seismic hazard function and the equipment fragilities have substantial uncertainties. This model uses a curve fit to the mean hazard curve (the basis of the best estimate analysis) developed by Lawrence Livermore National Laboratory. Because of the lognormal uncertainty distribution, the mean hazard curve results in the best estimate being close to the upper bound. The lower bound is substantially below the mean. The upper bound hazard curve ranges from a factor of 1.9 times higher than the mean curve for low magnitude seismic events to a factor of 1.7 for high magnitude seismic events. Increasing only the seismic hazard frequency accordingly in each seismic interval results in a seismic induced frequency of spent fuel uncovery of 1.48E-7/yr. Therefore, even with the upper bound hazard curve the sequence frequency does not increase substantially from the best estimate.

On the other hand, the lower bound hazard curve ranges from a factor of 0.15 times lower than the mean curve for low magnitude seismic events to a factor of 0.01 for high magnitude seismic events. Using the lower bound seismic hazard frequency accordingly in each seismic interval results in a spent fuel uncovery frequency of 2.29E-9/yr. Therefore, the use of the lower bound hazard curve produces a substantial reduction in the sequence frequency (more than a factor of 35) compared with the Base Case seismic evaluation.

In addition to the variations in the hazard curve, ten separate seismic sensitivity cases were defined and quantified. The base case seismic assessment and seismic sensitivity case results are summarized in Table 5-13. Each of the ten sensitivity cases are described below.

- (Sensitivity Case S.1) Finer Division of Seismic Hazard Curve: This sensitivity case divides the SHNPP seismic hazard curve into 16 intervals (15 intervals between 0 and 1.5g, and one interval for >1.5g) instead of the Base Case 7 intervals. This sensitivity case tests the impact on the quantitative results from the analysis approach of dividing the seismic hazard curve into discrete intervals, quantifying the risk of each magnitude interval, and then integrating the results. Seismic PSAs typically divide the seismic hazard curve into approximately a half dozen intervals – the approach taken in the Seismic Base Case. Sixteen intervals is a comparatively fine division of the curve. The first fifteen intervals are 0.1g wide (e.g., 0 – 0.1, 0.1 – 0.2, 0.2 – 0.3, etc.) and the final interval is defined as >1.5g.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 7.42E-8/yr (a 15% reduction in frequency compared to the Seismic Base Case). This reduction is not unexpected; the coarser the division of the seismic hazard curve, the more conservative will be the final integrated results.

- (Sensitivity Case S.2) No Extrapolation Beyond NUREG-1488 Hazard Curve: This sensitivity case defines the final seismic magnitude range as >1.0g instead of the Seismic Base Case >1.5g. In the Seismic Base Case, the point at which the FHB is assumed to structurally fail given the seismic shock (and, thus, fall outside the bounds of this analysis) is 1.5g. However, NUREG-1488 only supplies frequency estimates for seismic events up to 1.0g; as such, a case may be made for defining >1.0g as the final magnitude range and assuming that seismic events beyond this are very low likelihood and highly likely to result in FHB failure.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 5.14E8-/yr (a 40% reduction in frequency compared to the Seismic Base Case). This reduction is not unexpected; high magnitude seismic events, although low in frequency, impact the quantitative results due to high component and structural fragilities at such g levels.

- <u>(Sensitivity Case S.3) Less Conservative Uncertainty Distribution for Seismic Fragilities</u>: This sensitivity case employs less conservative randomness and uncertainty parameters (0.30 and 0.30); respectively in the fragility calculations instead of the Base Case values of 0.40 and 0.40. This sensitivity case tests the impact on the quantitative results from the estimated randomness and uncertainty in the component and structural fragility calculations. Randomness and uncertainty parameters used in seismic PSAs are typically in the 0.20 to 0.40 range. In certain cases, values as low as 0.10 – 0.20 (e.g., offsite power transformers) and as high as 0.50 – 0.70 (e.g., relay chatter failures) are used. The Seismic Base Case employs 0.40 and 0.40 as a suitably conservative set of values. This sensitivity case uses 0.30 and 0.30 to represent a less conservative set of values.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 5.40E-8/yr (a 37% reduction in seismic induced accident sequence frequency compared to the Seismic Base Case). This reduction is not unexpected; all other issues being equal, the tighter the assumed uncertainty around the estimated seismic capacities, the lower are the calculated fragilities.

- <u>(Sensitivity Case S.4) Seismic Capacities Increased Approximately 25%</u>: This sensitivity case employs higher component and structural seismic capacities than used in the Seismic Base Case. The Seismic Base Case uses component and structural capacities estimated based on review of similar components in other seismic PSAs and knowledge of the SHNPP plant. This sensitivity case tests the impact on the quantitative results given the possibility that the selected capacities used in the assessment are conservative. A factor of approximately 1.25 was assumed in this sensitivity to indicate the comparative level of conservatism existing in the selected capacities of the Seismic Base Case.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 3.65E-8/yr (a 58% reduction in frequency compared to the Seismic Base Case). This reduction is not unexpected; all other issues being equal, the higher the estimated seismic capacities, the lower are the calculated fragilities.

- <u>(Sensitivity Case S.5) Seismic Capacities Decreased Approximately 25%</u>: This sensitivity case employs lower component and structural seismic capacities than used in the Seismic Base Case. The Seismic Base Case uses component and structural capacities estimated based on review of similar components in other seismic PSAs and knowledge of the SHNPP plant. This sensitivity case tests the impact on the quantitative results given the possibility that the selected capacities used in the assessment are non-conservative. A factor of approximately 0.75 was assumed in this sensitivity to indicate a comparative level of non-conservatism that may be postulated to exist in the selected capacities of the Seismic Base Case.

As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 1.62E-7/yr (1.9 times the Seismic Base Case). This increase Is not unexpected; all other issues being equal, the lower the estimated seismic capacities, the higher are the calculated fragilities.

- <u>(Sensitivity Case S.6) More Conservative Early Containment Failure Probability</u>: This sensitivity case employs a higher early containment failure probability than used in the Seismic Base Case. The Seismic Base Case uses a conditional (upon core damage) early containment failure probability of 3.76E-2 based on review of the current SHNPP PSA results. The 3.76E-2 value is the most conservative value of the assessed core damage scenarios. This sensitivity case tests the impact on the quantitative results from a higher early containment failure probability. An approximate factor of 3 is applied to the Seismic Base Case value, resulting in a nominal early containment failure probability of 0.10 for use in this sensitivity case.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 1.12E-7/yr (a 30% increase in frequency compared to the Seismic Base Case). This increase is not unexpected because early containment failure directly impacts the human error probabilities associated with providing cooling to the SFPs.

- <u>(Sensitivity Case S.7) More Conservative Human Error Probabilities</u>: This sensitivity case employs higher human error probabilities than used in the Seismic Base Case. The Seismic Base Case generally employs conservative human error probabilities (e.g., 1.0AC power recovery failure probability, 1.0 manual containment isolation failure probability). This sensitivity case applies a conservative element across the board to all human errors. Human error probabilities less than 0.1 are set to 0.1, and human error probabilities greater than or equal to 0.1 are left at the Seismic Base Case value.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 1.46E-7/yr (1.7 times the Seismic Base Case).

This increase is not unexpected; human error probabilities play a key role in the assessed spent fuel failure frequency.

- <u>(Sensitivity Case S.8) Less Conservative Human Error Probabilities</u>: This sensitivity case employs less conservative human error probabilities for selected human interfaces in the Seismic Base Case. The Seismic Base Case generally employs conservative human error probabilities (e.g., 1.0 AC power recovery failure probability, 1.0 manual containment isolation failure probability). This sensitivity case reduces the 1.0 failure probabilities to 0.5 for the following selected actions:

  - AC Power Recovery Failure
  - Containment Manual Isolation Failure
  - Fire Hose Alignment Failure Given Early Containment Failure
  - Fire Hose Alignment Failure Given Containment Isolation Failure

  All other human error probabilities are left at the Seismic Base Case value.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 3.86E-8/yr (a 55% decrease in frequency compared to the Seismic Base Case). This decrease is not unexpected; human error probabilities play a key role in the assessed spent fuel failure frequency.

- <u>(Sensitivity Case S.9) Overall Pessimistic Case</u>: This sensitivity case employs all the attributes of Sensitivity Cases 5, 6, and 7. This sensitivity case is aptly described as the overall pessimistic case.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 3.43E-7/yr (4 times the Seismic Base Case).

- <u>(Sensitivity Case S.10) Overall Optimistic Case</u>: This sensitivity case employs all the attributes of Sensitivity Cases 1, 2, 3, 4 and 8. This sensitivity case is aptly described as the overall optimistic case.

  As can be seen from Table 5-13, this sensitivity case resulted in a total frequency of 2.06E-9/yr (a 97% decrease in frequency compared to the Seismic Base Case).

5.5 SENSITIVITY RESULTS

Table 5-14 summarizes the results of the sensitivity cases performed to characterize the degree of uncertainty in the quantitative evaluation of the Postulated Sequence. As discussed in Section 5.3, the best estimate of the probability of the Postulated Sequence is best represented by the probability calculated for internal events alone. This is due to the level of uncertainty associated with the state of the technology for the calculation of external event and shutdown contributions. The sensitivity of the analysis to various input parameters, is shown relative to a composite Base Case, Case A. The sensitivity cases then used a composite frequency as well, and are compared to Case A to demonstrate the sensitivity of the probability estimate to the various input parameters. The results, therefore, include the contributions to the Postulated Sequence from internal, seismic, fire and shutdown events. The results make use of the appropriate seismic sensitivity cases.

Figure 5-1 provides a histogram comparison of the sensitivity results using the composite totals from internal, seismic, fire, and shutdown events. This figure also compares the results with the NRC surrogate safety goal for severe accidents leading to core damage (i.e., 1E-4/reactor year). In addition, the frequency cited in Appendix B of this report as "remote and speculative" is also shown for reference (i.e., 1E-6/year).

Figure 5-1 includes estimated upper and lower bounds on the evaluation based on the comparison of the sensitivity cases. These bounds should be interpreted to represent an approximation to the 90% confidence interval within which the frequency may lie.

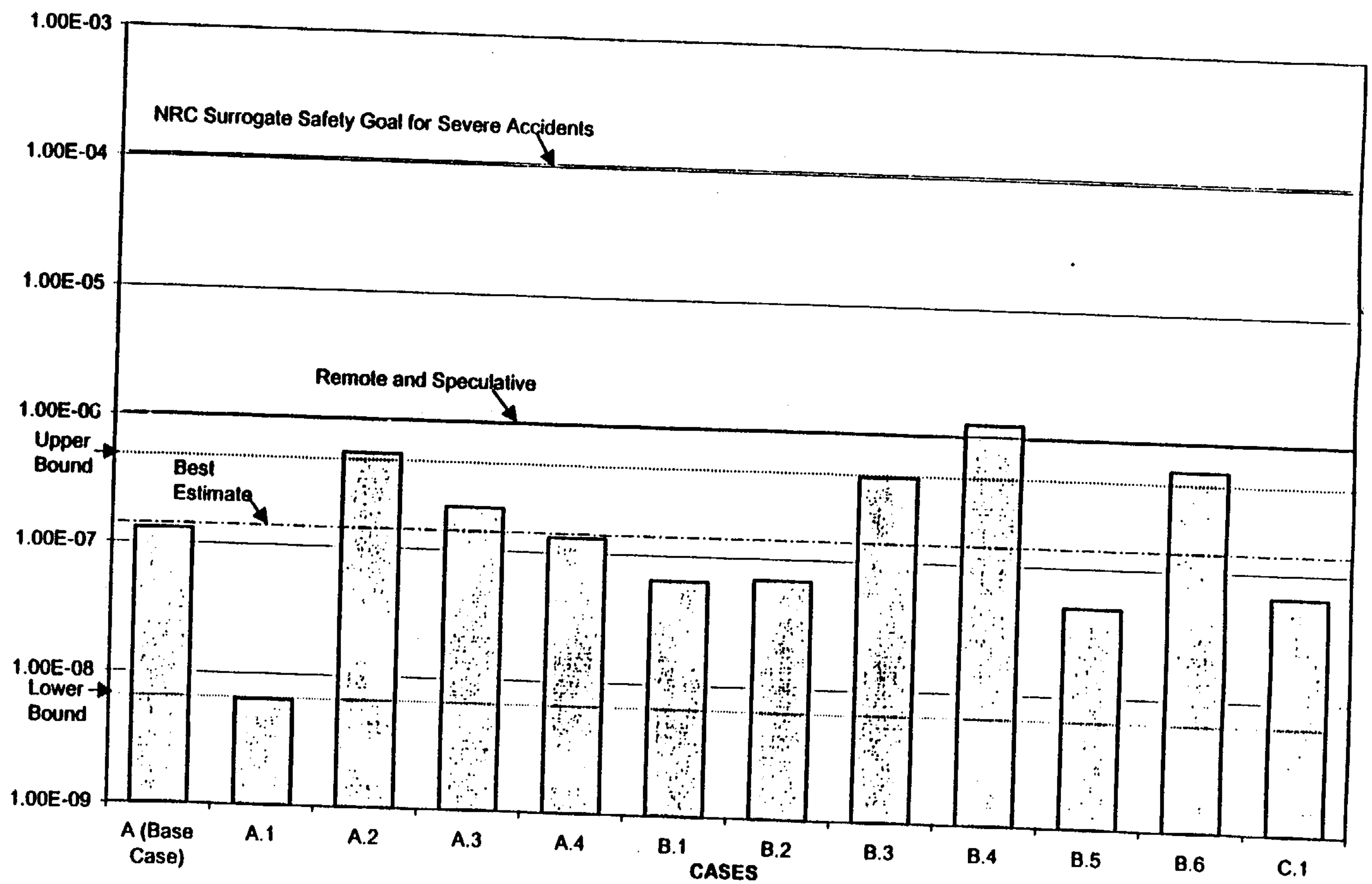


Figure 5-1 Summary of Sensitivity Cases to Demonstrate the Range of Uncertainty

Table 5-1

SHNPP SFPAET RESULTS
BEST ESTIMATE ACCIDENT SEQUENCE FREQUENCIES

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from Level 1 and 2 Quantification[(1)]	Output from SFPAET[(2)]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	9.97E-9	7.44E-10
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.57E-06	3.44E-09
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.51E-06	3.31E-09
LG-ISOL	LARGE ISOLATION FAILURE	7.59E-08	9.77E-10
SM-ISOL	SMALL ISOLATION FAILURE	1.88E-07	2.59E-09
EARLY	EARLY CONTAINMENT FAILURE	3.14E-08	1.15E-09
LATE	LATE CONTAINMENT FAILURE	4.28E-06	1.43E-08
Total Internal Events Contribution		7.67E-06	2.65E-08
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-09	7.98E-11
LATE	LATE CONTAINMENT FAILURE	9.77E-07	2.86E-09
Total Fire Events Contribution		9.80E-07	2.94E-09
Total Seismic Contribution		-	8.65E-08
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	7.2E-07	1.45E-08

(1) CDF with containment failure, bypass, or containment isolation failure (per year).

(2) Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-2

SHNPP SFPAET RESULT LOWER BOUND ACCIDENT SEQUENCE FREQUENCIES (CASE A.1)

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from Level 1 and 2 Quantification[1]	Output from SFPAET[2]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	0.0	0.0
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.4E-07	3.16E-10
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.4E-07	3.07E-10
LG-ISOL	LARGE ISOLATION FAILURE	7.0E-09	9.01E-11
SM-ISOL	SMALL ISOLATION FAILURE	1.7E-08	2.34E-10
EARLY	EARLY CONTAINMENT FAILURE	2.9E-09	2.89E-10
LATE	LATE CONTAINMENT FAILURE	3.9E-07	1.30E-09
Total Internal Events Contribution		7.0E-07	2.54E-09
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-10	7.98E-12
LATE	LATE CONTAINMENT FAILURE	9.77E-08	2.86E-10
Total Fire Events Contribution		9.80E-08	2.94E-10
Total Seismic Contribution (Case S.10)			2.1E-09
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	5.0E-08	1.45E-09

(1) CDF with containment failure, bypass, or containment isolation failure (per year).

(2) Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-3

SHNPP SFPAET RESULTS UPPER BOUND
ACCIDENT SEQUENCE FREQUENCIES (CASE A.2)

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from Level 1 and 2 Quantification[1]	Output from SFPAET[2]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	5.0E-7	3.73E-08
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	5.1E-06	1.12E-08
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	4.9E-06	1.07E-08
LG-ISOL	LARGE ISOLATION FAILURE	2.5 E-07	3.22E-09
SM-ISOL	SMALL ISOLATION FAILURE	6.1E-07	8.40E-09
EARLY	EARLY CONTAINMENT FAILURE	1.0E-07	3.66E-09
LATE	LATE CONTAINMENT FAILURE	1.4E-05	4.68E-08
Total Internal Events Contribution		2.55E-05	1.21E-07
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-.08	7.98E-10
LATE	LATE CONTAINMENT FAILURE	9.77E-06	2.86E-08
Total Fire Events Contribution		9.80E-06	2.94E-08
Total Seismic Contribution (Case S.9)		--	3.4E-7
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	2.0E-06	5.80E-08

(1) CDF with containment failure, bypass, or containment isolation failure (per year).
(2) Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-4

SHNPP SFPAET RESULTS FOR PESSIMISTIC MODELING OF 6.9KV SWITCHGEAR SURVIVABILITY[1] (CASE A.3)

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from Level 1 and 2 Quantification[2]	Output from SFPAET[3]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	9.97E-09	4.8E-09
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.57E-06	1.05E-08
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.51E-06	1.01E-08
LG-ISOL	LARGE ISOLATION FAILURE	7.59E-08	3.08E-09
SM-ISOL	SMALL ISOLATION FAILURE	1.88E-07	8.06E-09
EARLY	EARLY CONTAINMENT FAILURE	3.14E-08	2.67E-09
LATE	LATE CONTAINMENT FAILURE	4.28E-06	3.47E-08
Total Internal Events Contribution		7.67E-06	7.4E-08
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-09	2.19E-10
LATE	LATE CONTAINMENT FAILURE	9.77E-07	6.75E-09
Total Fire Events Contribution		9.80E-07	6.97E-09
Total Seismic Contribution (Base Case)[4]		-	8.65E-08
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	7.2E-07	5.38E-08

(1) Set the Demineralized Water Pumps to 1.0
(2) CDF with containment failure, bypass, or containment isolation failure (per year).
(3) Frequency of the loss of effective water cooling to the spent fuel (per year).
(4) Seismic event involves Loss of Offsite Power; therefore no effect of the Normal 6.9KV Power Switchgear.

Table 5-5

SHNPP SFPAET SENSITIVITY RESULTS

Basic Event	Description	Base Case	Case B.1, B.2, B.3
OPERDALNPB	Operators Fail To Align DW To The Unit 1 or Unit 2 FPCCS Cleanup Subsystem	1.90E-02	9.5E-3
OPER-TSC-E	TSC Fails to Take Pre-emptive Action for Early Failures	4.6E-03	2.4E-3
OPERPALNN1	Operators Fail To Use Water From The FHB Fire Header To Makeup To The SFPs	6.2E-2	1.1E-3
OPERPALNN2	Operators Fail To Use Water From The 19 FHB DM Stations To Makeup To The SFPs	1.00E+00	2.5E-1
OPER-TSC-L	TSC fails to take PRE-emptive Action for Late Failures	2.4E-3	1.4E-3

Table 5-6

SHNPP SFPAET SENSITIVITY RESULTS: CASE B.1, B.2, B.3

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Base Output[1] from SFPAET	Sensitivity Case B1[1]	Sensitivity Case B2[1]	Sensitivity Case B3[1]
Internal Events					
ISLOCA	INTERFACING SYSTEMS LOCA	7.44E-10	7.44E-10	9.0E-09	4.03E-07
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	3.44E-09	1.57E-09	1.57E-09	1.57E-09
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	3.31E-09	1.51E-09	1.51E-09	1.51E-09
LG-ISOL	LARGE ISOLATION FAILURE	9.77E-10	7.99E-10	7.99E-10	7.99E-10
SM-ISOL	SMALL ISOLATION FAILURE	2.59E-09	2.16E-09	2.16E-09	2.16E-09
EARLY	EARLY CONTAINMENT FAILURE	1.15E-09	1.15E-09	1.15E-09	1.15E-09
LATE	LATE CONTAINMENT FAILURE	1.43E-08	8.12E-09	8.12E-09	8.12E-09
Total Internal Events Contribution		2.65E-08	1.60E-08	2.43E-08	4.18E-07
Fire Induced Events					
EARLY	EARLY CONTAINMENT FAILURE	7.98E-11	8.35E-11	8.35E-11	8.35E-11
LATE	LATE CONTAINMENT FAILURE	2.86E-09	1.30E-09	1.30E-09	1.30E-09
Total Fire Events Contribution		2.94E-09	1.38E-09	1.38E-09	1.38E-09
Total Seismic Contribution (Case S.8)		8.65E-08	3.88E-08	3.88E-08	3.88E-08
Shutdown Events					
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	1.45E-08	7.62E-09	7.62E-09	7.62E-09

[1] Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-7
SHNPP SFP MAKEUP OPERATOR ACTION EVENTS: PESSIMISTIC HEP'S

New Basic Event	Case B.4	Description	OP-116 Step
OPERDALNPB	0.1	Operators Fail To Align DW To The Unit 1 FPCCS Cleanup Subsystem	8.4
OPERDALNPB	0.1	Operators Fail To Align DW To The Unit 2 FPCCS Cleanup Subsystem.	8.4
OPER-1CLBA	0.1	Operators Fail To Cross Tie Unit 1 FPCCS Pump Train B To Heat Exchanger A	N/A
OPER-2CLBA	0.1	Operators Fail To Cross Tie Unit 2 FPCCS Pump Train B To Heat Exchanger A	N/A
OPERPALNN1	0.1	Operators Fail To Use Water From The FHB Fire Header To Makeup To The SFPs	N/A
OPER-GATE1	1	Operators Fail To Deflate Gate 1 Seals	N/A
OPER-GATE2	1	Operators Fail To Deflate Gate 2 Seals	N/A
OPER-GATE3	1	Operators Fail To Deflate Gate 3Seals	N/A
OPER-GATE4	1	Operators Fail To Deflate Gate 4 Seals	N/A
OPER-GATE5	1	Operators Fail To Deflate Gate 5 Seals	N/A
OPER-GATE6	1	Operators Fail To Deflate Gate 6 Seals	N/A
OPER-GATE7	1	Operators Fail To Deflate Gate 7 Seals	N/A
OPER-GATE9	1	Operators Fail To Deflate Gate 9 Seals	N/A
OPER-GATES	1	Operators Fail To Remove Bulkhead Gates	8.27
OPERPALNN2	1.0	Operators Fail To Use Water From The 19 FHB DM Stations To Makeup To The SFPs	N/A
OPERPALNN3	1	Operators Fail To Use Water From The NSW System In The WPB To Makeup To The SFP	N/A
OPER-OFFST	0.1	Operators Fail To Use Portable / Off-Site Resources For Makeup To The SFPs	N/A
OPER-PROCD	0.1	Procedures To Maintain SFP Inventory Are Inadequate	All

Table 5-7
SHNPP SFP MAKEUP OPERATOR ACTION EVENTS: PESSIMISTIC HEP'S

New Basic Event	Case B.4	Description	OP-116 Step
OPERRALNPC	1	Operators Fail To Align The FPCCS Purification Subsystem To The RWST	8.5
OPER-LOLVL	0.1	Operators Fail To Diagnose Low SFP Levels And / Or Perform Recovery	All
OPER-ESW	0.1	Operators Fail To Open ESW Manual Valves	8.13
OPER-TSC-E	0.1	TSC Fails to Take Pre-emptive Action for Early Failures	NA
OPER-TSC-L	0.1	TSC Fails to Take Pre-emptive Action for Late Failures	NA
OPER-SKIMR	1	Operators Fail To Open The Crosstie Between Units 1 and 4 and 2 and 3 FPCCS Skimmers	NA
OPER-DWXTM	1	Operators Fail To Open DM Crosstie Valve 1SF-203	NA
OPER-START	0.1	OPERATORS FAIL TO MANUALLY START FPCS MOTOR-DRIVEN PUMP	NA
OPERZOFFST	0.1	Operator Fails to Align Offsite Resources to Previously Established Paths	NA
CI-CASE 1	1.1 E-2	Operator Fails to Restore Primary Containment Given Mid Level Operation (Shutdown only)	Tech specs
CI-CASE 2	1.6 E-2	Operator Fails to Restore Primary Containment Given Normal Level Operation (Shutdown only)	Tech specs
OPERATOR ACTIONS GIVEN NO CREDIT IN ANALYSIS			
OPEREALNPA	1	Operator Fails to Align and Initiate ESW to FPCC for Makeup	8.13
OPERMALNPD	1	Operator Fails to Align and Initiate RMWST to FPCC for Makeup	8.26
OPERDALNPE	1	Operator Fails to Align and Initiate Demin Water to FPCC Skimmer for Makeup	8.6
OPERRALNPF	1	Operator Fails to Align and Initiate RWST to FPCCS Cooling Pump for Makeup	8.5
OPERDALNPG	1	Operator Fails to Align and Initiate Demin Water to FPCC Cleanup for Makeup	8.5
OPER-IN-FA	1	Operator Fails to Initiate FPCC Cooling to Pools A and B	N/A
OPER-IN-FC	1	Operator Fails to Initiate FPCC Cooling to Pools C and D	N/A

C00003

Table 5-8

SHNPP SFPAET RESULTS (CASE B.4) PESSIMISTIC HEPs

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from level 1 and 2 Quantification[1]	Output from SFPAET[2]
<u>Internal Events</u>			
ISLOCA	INTERFACING SYSTEMS LOCA	9.97E-9	3.99E-09
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.57E-06	1.73E-07
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.51E-06	1.66E-07
LG-ISOL	LARGE ISOLATION FAILURE	7.59E-08	8.46E-09
SM-ISOL	SMALL ISOLATION FAILURE	1.88E-07	2.22E-08
EARLY	EARLY CONTAINMENT FAILURE	3.14E-08	8.17E-09
LATE	LATE CONTAINMENT FAILURE	4.28E-06	4.98E-07
Total Internal Events Contribution		7.67E-06	9.98E-07
<u>Fire Induced Events</u>			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-09	6.87E-10
LATE	LATE CONTAINMENT FAILURE	9.77E-07	1.66E-07
Total Fire Events Contribution		9.80E-07	1.17E-07
Total Seismic Contribution (Case S.7)			1.46E-07
<u>Shutdown Events</u>			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	7.2E-07	1.44E-07

(1) CDF with containment failure, bypass, or containment isolation failure (per year).

(2) Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-9

SHNPP SFP MAKEUP OPERATOR ACTION EVENTS: REASONABLE HEP's

New Basic Event	BASE case	Description	OP-116 Step
OPERDALNPB	IE-03	Operators Fail To Align DW To The Unit 1 FPCCS Cleanup Subsystem	8.4
OPERDALNPB	IE-03	Operators Fail To Align DW To The Unit 2 FPCCS Cleanup Subsystem	8.4
OPER-1CLBA	IE-03	Operators Fail To Cross Tie Unit 1 FPCCS Pump Train B To Heat Exchanger A	N/A
OPER-2CLBA	IE-03	Operators Fail To Cross Tie Unit 2 FPCCS Pump Train B To Heat Exchanger A	N/A
OPERPALNN1	IE-03	Operators Fail To Use Water From The FHB Fire Header To Makeup To The SFPs	N/A
OPER-GATE1	1	Operators Fail To Deflate Gate 1 Seals	N/A
OPER-GATE2	1	Operators Fail To Deflate Gate 2 Seals	N/A
OPER-GATE3	1	Operators Fail To Deflate Gate 3Seals	N/A
OPER-GATE4	1	Operators Fail To Deflate Gate 4 Seals	N/A
OPER-GATE5	1	Operators Fail To Deflate Gate 5 Seals	N/A
OPER-GATE6	1	Operators Fail To Deflate Gate 6 Seals	N/A
OPER-GATE7	1	Operators Fail To Deflate Gate 7 Seals	N/A
OPER-GATE9	1	Operators Fail To Deflate Gate 9 Seals	N/A
OPER-GATES	1	Operators Fail To Remove Bulkhead Gates	8.27
OPERPALNN2	1	Operators Fail To Use Water From The 19 FHB DM Stations To Makeup To The SFPs	N/A
OPERPALNN3	1	Operators Fail To Use Water From The NSW System In The WPB To Makeup To The SFP	N/A
OPER-OFFST	1.00E-03	Operators Fail To Use Portable / Off-Site Resources For Makeup To The SFPs	N/A
OPER-PROCD	1.00E-03	Procedures To Maintain SFP Inventory Are Inadequate	All
OPERRALNPC	1	Operators Fail To Align The FPCCS Purification Subsystem To The RWST	8.5

Table 5-9

SHNPP SFP MAKEUP OPERATOR ACTION EVENTS: REASONABLE HEP's

New Basic Event	BASE case	Description	OP-116 Step
OPER-LOLVL	1.00E-03	Operators Fail To Diagnose Low SFP Levels And / Or Perform Recovery	All
OPER-ESW	1.00E-03	Operators Fail To Open ESW Manual Valves	8.13
OPER-TSC-E	1.00E-03	TSC Fails to Take Pre-emptive Action for Early Failures	NA
OPER-TSC-L	1.00E-03	TSC Fails to Take Pre-emptive Action for Late Failures	NA
OPER-SKIMR	1	Operators Fail To Open The Crosstie Between Units 1 and 4 and 2 and 3 FPCCS Skimmers	NA
OPER-DWXTM	1	Operators Fail To Open DM Crosstie Valve 1SF-203	NA
OPER-START	2.00E-05	OPERATORS FAIL TO MANUALLY START FPCS MOTOR-DRIVEN PUMP	NA
OPERZOFFST	1.00E-03	Operator Fails to Align Offsite Resources to Previously Established Paths	NA
OPERATOR ACTIONS CURRENTLY MODELED AS GUARANTEED FAILURE			
CI-CASE 1	1.1 E-2	Operator Fails to Restore Primary Containment Given Mid Level Operation (Shutdown only)	Tech specs
CI-CASE 2	1.6 E-2	Operator Fails to Restore Primary Containment Given Normal Level Operation (Shutdown only)	Tech specs
OPERMALNPD	1	Operator Fails to Align and Initiate RMWST to FPCC for Makeup	8.26
OPERDALNPE	1	Operator Fails to Align and Initiate Demin Water to FPCC Skimmer for Makeup	8.6
OPERRALNPF	1	Operator Fails to Align and Initiate RWST to FPCCS Cooling Pump for Makeup	8.5
OPERDALNPG	1	Operator Fails to Align and Initiate Demin Water to FPCC Cleanup for Makeup	8.5
OPER-IN-FA	1	Operator Fails to Initiate FPCC Cooling to Pools A and B	N/A
OPER-IN-FC	1	Operator Fails to Initiate FPCC Cooling to Pools C and D	N/A

Table 5-10

SHNPP SFPAET RESULTS (CASE B.5): REASONABLE HEPs

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input CDF from Level 1 and 2 Quantification[1]	Output from SFPAET[2]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	9.97E-9	3.99E-11
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.57E-06	1.57E-09
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.51E-06	1.51E-09
LG-ISOL	LARGE ISOLATION FAILURE	7.59E-08	8.45E-11
SM-ISOL	SMALL ISOLATION FAILURE	1.88E-07	2.22E-10
EARLY	EARLY CONTAINMENT FAILURE	3.14E-08	7.13E-11
LATE	LATE CONTAINMENT FAILURE	4.28E-06	4.27E-09
Total Internal Events Contribution		7.67E-06	7.77E-09
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-09	5.63E-12
LATE	LATE CONTAINMENT FAILURE	9.77E-07	9.88E-10
Total Fire Events Contribution		9.80E-07	9.94E-10
Total Seismic Contribution (Case S.8)			3.90E-08
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	7.2E-07	1.44E-09

(1) CDF with containment failure, bypass, or containment isolation failure (per year).

(2) Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-11

SHNPP SFPAET RESULT FOR HIGH ON-SITE RADIATION DUE TO CONSERVATIVE CHI/Q ACCIDENT SEQUENCE FREQUENCIES (CASE B.6)

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from Level 1 and 2 Quantification[1]	Output from SFPAET[2]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	9.97E-09	9.97E-09
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.57E-06	3.36E-08
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.51E-06	3.24E-08
LG-ISOL	LARGE ISOLATION FAILURE	7.59E-08	6.51E-09
SM-ISOL	SMALL ISOLATION FAILURE	1.88E-07	1.81E-08
EARLY	EARLY CONTAINMENT FAILURE	3.14E-08	3.14E-08
LATE	LATE CONTAINMENT FAILURE	4.28E-06	1.03E-07
Total Internal Events Contribution		7.67E-06	2.51E-07
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-10	2.95E-09
LATE	LATE CONTAINMENT FAILURE	9.77E-08	1.69E-08
Total Fire Events Contribution		9.80E-08	1.99E-08
Total Seismic Contribution (Case S.9)			3.40E-07
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	7.2E-07	1.60E-08

(1) CDF with containment failure, bypass, or containment isolation failure (per year).
(2) Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-12

SHNPP SFPAET RESULTS FOR ASSESSMENT OF SENSITIVITY TO EXOTHERMIC REACTION PROBABILITY ACCIDENT SEQUENCE FREQUENCIES (CASE C.1)

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from Level 1 and 2 Quantification[1]	Output from SFPAET[2]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	9.97E-09	3.70E-10
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.57E-06	1.70E-09
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.51E-06	1.70E-09
LG-ISOL	LARGE ISOLATION FAILURE	7.59E-08	4.90E-10
SM-ISOL	SMALL ISOLATION FAILURE	1.88E-07	1.30E-09
EARLY	EARLY CONTAINMENT FAILURE	3.14E-08	5.80E-10
LATE	LATE CONTAINMENT FAILURE	4.28E-06	7.20E-09
Total Internal Events Contribution		7.67E-06	1.37E-08
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-09	4.00E-11
LATE	LATE CONTAINMENT FAILURE	9.77E-07	1.40E-09
Total Fire Events Contribution		9.80E-07	1.50E-09
Total Seismic Contribution (Special Case)		-	4.30E-08
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	7.2E-07	7.30E-09

(1) CDF with containment failure, bypass, or containment isolation failure (per year).
(2) Frequency of the loss of effective water cooling to the spent fuel (per year).

Table 5-13
SUMMARY OF SEISMIC ASSESSMENT QUANTITATIVE SENSITIVITY CASES

Sensitivity Case	Case Description (1)	Seismic Hazard Curve: # Seis. Mag. Intervals	Seismic Hazard Curve: Magnitude of Final Seismic Range	Seismic Fragility Parameters: BETA(r), BETA(u)	Seismic Fragility Parameters: EDG Am	Seismic Fragility Parameters: Ess. SWGR Am	Seismic Fragility Parameters: PCIV Am	Seismic Fragility Parameters: DFP Am	Seismic Fragility Parameters: FHB Flooding Am	Seismic Fragility Parameters: Offsite Infrastructure Am	Early Containment Failure Probability	Human Interfaces: AC Recovery Failure Prob	Human Interfaces: PCIV Manual Isolation	Human Interfaces: Fire Hose Align HEP: Early Cont. Failure	Human Interfaces: Fire Hose Align HEP: Cont. Isol Failure	Human Interfaces: Fire Hose Align HEP: Late Cont. Failure	Human Interfaces: Demin Align HEP: Early Cont. Failure	Human Interfaces: Demin Align HEP: Cont. Isol Failure	Human Interfaces: Demin Align HEP: Late Cont. Failure	Human Interfaces: Fire Truck Hook-Up HEP	Human Interfaces: Portable Pump/Gen Hook-Up HEP	Spent Fuel Uncovery Frequency (1/yr)
0	BASE Case	7	>1.5g	0.4,0.4	1.25	1.31	2.00	1.25	1.25	1.00	3.76E-2	1.00	1.00	1.00	1.00	0.062	0.10	0.019	0.019	1.00	0.05	8.65E-08
1	Finer Division of Seismic Hazard Curve	16	>1.5g	0.4,0.4	1.25	1.31	2.00	1.25	1.25	1.00	3.76E-2	1.00	1.00	1.00	1.00	0.062	0.10	0.019	0.019	1.00	0.05	7.42E-00
2	No Extrapolation Beyond NUREG-1488 Hazard Curve	7	>1.0g	0.4,0.4	1.25	1.31	2.00	1.25	1.25	1.00	3.76E-2	1.00	1.00	1.00	1.00	0.062	0.10	0.019	0.019	1.00	0.05	5.14E-00
3	Less Conservative Uncertainty Distribution for Seismic Fragilities	7	>1.5g	0.3,0.3	1.25	1.31	2.00	1.25	1.25	1.00	3.76E-2	1.00	1.00	1.00	1.00	0.062	0.10	0.019	0.019	1.00	0.05	5.40E-08
4	Seismic Capacities Increased Approximately 25%	7	>1.5g	0.4,0.4	1.50	1.65	2.50	1.50	1.50	1.25	3.76E-2	1.00	1.00	1.00	1.00	0.062	0.10	0.019	0.019	1.00	0.05	3.65E-00
5	Seismic Capacities Decreased Approximately 25%	7	>1.5g	0.4,0.4	1.00	1.00	1.50	1.00	1.00	0.75	3.76E-2	1.00	1.00	1.00	1.00	0.062	0.10	0.019	0.019	1.00	0.05	1.62E-07
6	More Conservative Early Containment Failure Probability	7	>1.5g	0.4,0.4	1.25	1.31	2.00	1.25	1.25	1.00	1.00E-1	1.00	1.00	1.00	1.00	0.062	0.10	0.019	0.019	1.00	0.05	1.12E-07
7	More Conservative Human Error Probabilities	7	>1.5g	0.4,0.4	1.25	1.31	2.00	1.25	1.25	1.00	3.76E-2	1.00	1.00	1.00	1.00	0.10	0.10	0.10	0.10	1.00	0.10	1.46E-07
8	Less Conservative Human Error Probabilities	7	>1.5g	0.4,0.4	1.25	1.31	2.00	1.25	1.25	1.00	3.76E-2	0.60	0.60	0.60	0.60	0.062	0.10	0.019	0.019	1.00	0.05	3.86E-08
9	Overall Pessimistic Case	7	>1.5g	0.4,0.4	1.00	1.00	1.50	1.00	1.00	0.75	1.00E-1	1.00	1.00	1.00	1.00	0.10	0.10	0.10	0.10	1.00	0.10	3.43E-07
10	Overall Optimistic Case	11 (Note 2)	>1.0g	0.3,0.3	1.50	1.65	2.50	1.50	1.50	1.25	3.76E-2	0.60	0.60	0.50	0.60	0.062	0.10	0.019	0.019	1.00	0.05	2.06E-09

NOTES

(1) Shaded cells indicate parameter changes with respect to the BASE Case

(2) Ten seismic hazard intervals between 0.0 and 1.0g, and one interval for >1.0g

CC00912

Table 5-14

SENSITIVITY CASE RESULTS

Sensitivity		Factor of Change Compared with Case A	Comments on Results
Case No.	Description		
A	Case A	1.3E-7/yr	This includes the best estimate contributions to the probability of the Postulated Sequence from the internal, seismic, fire, and shutdown analyses.
A.1	Lower Bound for Accident Frequencies (Steps 1 and 2) (Uses Case S.10 for seismic)	20 Reduction	Lower Bound estimate on the input accident frequency state in turn results in a substantial decrease in the SFP undesirable end state frequency estimates.
A.2	Upper Bound for Accident Frequencies (Steps 1 and 2) (Uses Case S.9 for seismic)	4.27 increase	Use of Upper Bound estimates on the inputs lead to a factor of 4 increase in the frequency SFP undesirable end state frequency.
A.3	Pessimistic Assessment of 6.9KV Switchgear Survivability (Uses Base Case for seismic)	1.67 increase	The impact of switchgear survivability for use of offsite power affects the internal events, shutdown and fire contributions. The use of a pessimistic assumption leads to a modest increase in the frequency of the undesirable end state.
A.4	Upper Bound Estimate for Installation of Gates Between A and B or Between C and D	1.01 increase	Essentially no impact on the Base Case evaluation.

Table 5-14

SENSITIVITY CASE RESULTS

Sensitivity		Factor of Change Compared with Case A	Comments on Results
Case No.	Description		
B.1	Written TSC Guidance Provided (Uses Case S.8 for seismic)	2.0 reduction	Written guidance regarding actions to be taken under severe accident conditions is calculated to lead to a reduction of approximately a factor of 2 in the frequency of SFP undesirable end state.
B.2	Access During ISLOCA Precluded (Uses Case S.8 for seismic)	1.8 reduction	Access to the FHB under ISLOCA conditions are found to have minimal impact on the assessed frequency when the Best Estimate ISLOCA frequency is used. Results are dominated by the TSC Guidance addition.
B.3	B.2 Plus Higher ISLOCA Frequency (Uses Case S.8 for seismic)	3.58 increase	When the upper bound ISLOCA frequency AND no access to the FHB are included in the quantitative model, it is found that the frequency of the undesirable end state for the SFP is found to increase by a factor of 3.6.
B.4	Degraded Human Response for all POST Containment Failure Actions (Uses Case S.7 for seismic)	9.85 increase	Because of the strong interface with operating crew actions, the calculated end state frequency is sensitive to changes in the HEPs

Table 5-14

SENSITIVITY CASE RESULTS

Sensitivity		Factor of Change Compared with Case A	Comments on Results
Case No.	Description		
B.5	Human Errors Are Set to 1E-3 to characterize a reasonable response to severe accidents (Except Guaranteed Failure Cases) (Uses Case S.8 for seismic)	1.61 reduction	Further reductions in the post containment failure HEPs from those used in the Base model have a relatively small impact on the results.
B.6	Accessibility Based on Worst Case Site Deposition with Chi/Q model (Uses Case S.9 for seismic)	4.6 increase	Radionuclide releases that are postulated to contaminate the site under worst case assumptions could lead to a substantial increase in the frequency of the undesirable SFP condition.
C.1	Estimate of Exothermic Reaction in SFP if water has evaporated	2 reduction	The exothermic reaction conditional probability is essentially a straight multiplier on the results. Therefore, a conditional probability that minimizes the maximum error, 0.5, results in a reduction in the undesirable end state of a factor of 2.

Section 6
CONCLUSIONS

6.1 OVERVIEW

A comprehensive PSA has been performed in response to the Postulated Sequence of events contained in the ASLB's August 7, 2000 Memorandum and Order. The PSA establishes the best estimate, given the current state of knowledge and technology, of the overall probability of the chain of seven events (Postulated Sequence) at SHNPP following the commencement of SFP C and D operation. The chain of seven events in the Postulated Sequence are as follows:

1. A degraded core accident
2. Containment failure or bypass
3. Loss of all spent fuel cooling and makeup systems
4. Extreme radiation doses precluding personnel access
5. Inability to restart any pool cooling or makeup systems due to extreme radiation doses
6. Loss of most or all pool water through evaporation
7. Initiation of an exothermic oxidation reaction in pools C and D.

This analysis has directly responded to the ASLB Order and establishes the probability for the specific scenario outlined by this Postulated Sequence. Furthermore, because the Postulated Sequence is focused on the ability of plant personnel to respond to the outlined events, this analysis did not consider off-site consequences associated with the scenario.

The seven steps of the Postulated Sequence are described in the following text; some related steps are discussed together.

Steps 1 and 2

A degraded core accident occurs and containment fails or is bypassed. Core damage sequences for which the containment is failed or bypassed as a result of internal, seismic, fire, and shutdown events are addressed in the quantitative assessment. The best estimate evaluation is judged to be best characterized by the internal events contribution. (See Section 4)

Step 3

Loss of all spent fuel pool cooling and makeup systems were considered as a result of the accident sequence <u>and</u> probabilistically, due to random or human-induced failures. (See Section 4, Appendices A, C and E).

Steps 4 and 5

For all sequences identified in Steps 1 and 2, radiation levels were calculated for specific areas in which access would be necessary in order to respond to Step 3. Consideration of the adverse impacts of extreme radiation on both personnel access and equipment survivability were then included in the probabilistic assessment. In addition, adverse environments due to high temperature or high humidity were deterministically assessed and included in the probabilistic model. (See Section 4, Appendices, A, C and E).

Step 6

Loss of most or all pool water through evaporation were then considered. To assess the probability of this step, a comprehensive analysis of the SFPs was conducted. The analysis considered the specific characteristics of the SFPs at SHNPP, as well as the potential methods available for injection of water in the event of the Postulated Sequence. A probabilistic assessment of the potential for the loss of SFP water through evaporation due to the loss of cooling and makeup systems was included. (See Section 4)

Step 7

Initiation of an exothermic oxidation reaction in pools C and D was then evaluated to determine whether it could be estimated probabilistically. Determining a best estimate probability for this step in the Postulated Sequence was difficult, given the state of knowledge related to this phenomenon. With the limited time and resources available to respond to the Postulated Sequence, this analysis assumes that the initiation of a self sustaining exothermic oxidation reaction in SFPs C and D occurred with a probability of 1.0, if the previous six steps had led to the evaporation of water from the SFPs. CP&L has addressed qualitatively how unlikely such an exothermic oxidation reaction would be in SFPs C and D. (See Affidavit of Robert K. Kunita.) Therefore, the assigned conditional failure probability of 1.0 is conservative.

The effort to respond to the ASLB Order involved the formation of an analysis team (13 Team Members) and a direct link to key CP&L staff. The CP&L staff provided detailed calculations (including the Level 1 and 2 SHNPP PSA), system descriptions, interviews with operating personnel, and procedure interpretations. The team effort included:

- multiple SHNPP site visits to confirm the as-built design and crew response;
- an independent peer review of the inputs to the evaluation, including the Level 1 and 2 PSA; and,
- an independent review of this analysis.

The methods chosen to evaluate each of the seven steps and arrive at a best estimate of the overall probability are characteristic of methods that have been used to perform past nuclear power plant PSAs. Where possible, this analysis relied on the results from the SHNPP Level 1 and Level 2 PSA. The specific method employed for each type of potential severe accident contributor that was evaluated varied according to the type of event being considered and the current state of technology:

Potential Severe Accident Contributor		**Methodology Utilized**
Internal Events	-	Full PSA methodology
Fire	-	Full PSA methodology for dominant sequences
Seismic	-	Approximate method
Shutdown	-	Generic assessment based on similar PWRs
Other	-	Determined to have negligible contribution

The SHNPP PSA (Level 1 and 2 Internal Events) was subjected to an independent peer review process as part of this evaluation. The review determined that the SHNPP PSA was robust, comprehensive, and consistent with the state-of-the-technology for such probabilistic assessments in the industry. The SHNPP PSA for internal events is fully supportive of risk-informed applications, even in cases where the absolute frequency of the accident sequences is required to support the application. The peer review also confirmed the finding of the SHNPP PSA (Level 1 and 2 Internal Events) that the plant meets the NRC Safety Goals and their subsidiary objectives (i.e., Core Damage Frequency and Large Early Release Frequency). In addition, the peer review confirmed that there are no unusual contributors to core damage frequency or containment failure.

6.2 CONCLUSIONS

Determination of the type of severe accidents that could result in the chain of events in the Postulated Sequence was the first step in this analysis. The analysis concluded that degraded core conditions with containment failure or bypass could result from a number of different postulated accident scenarios, which can be discussed under the following general categories of events differentiated by mode of operation:

A. <u>At-Power</u>

- Internal Events
- Internal Flood
- Seismic Induced
- Fire Induced
- Other

B. <u>Shutdown</u>

- Shutdown

This conclusion led to the separation of these severe accidents into two main subgroups, 1) Internal Events and 2) External Events and Shutdown. As discussed earlier in this report, the state of knowledge regarding the quantitative assessment of risk at nuclear power plants is best developed for assessing the risk due to internal events. It was therefore concluded that the best estimate of probability of the Postulated Sequence would be best determined by consideration of internal events. Following the determination of the best estimate probability for internal events, external events and shutdown events were evaluated to determine whether these events alter the conclusion reached based on the internal events assessment. These sensitivity analyses demonstrated that the best estimate probability that was determined was reasonable.

The results of the best estimate assessment for sequences initiated by internal events indicated that the loss of effective SFP cooling has an annual occurrence probability of 2.65E-8. Compared with other rare and accepted risks in life, this can be considered remote and speculative. The annual occurrence probability of the Postulated Sequence is, for example, considerably less than the probability of the recurrence of the ice age or the probability of a meteor strike creating worldwide havoc. (See Appendix B).

The conclusion from the external events and shutdown analysis is that the uncertainties associated with these sequences are sufficiently large that several conservatisms have been incorporated in the modeling. These conservatisms potentially result in inflated point estimate calculations. Therefore, while the point estimate contribution due to seismic initiated events is higher than for internal events, it is judged not to alter the conclusions reached based on the internal events analysis, i.e., that the postulated sequences of events can be considered "remote and speculative."

Table 6-1 is a summary table of the analysis results for the best estimate of the annualized probability of evaporation of SFP water and the uncovering of spent fuel from internal events, fire induced events, seismic events and shutdown events. The frequency for each event type is listed in the "output" column of Table 6-1. The internal event contribution directly responds to the questions regarding the Postulated Sequence presented in the ASLB Order, except it treats the time during the evaporation of water below the top of the fuel as inconsequential to the analysis and treats the probability of an exothermic reaction as equal to 1.0.

Fire induced events and shutdown events have a probability even lower than that estimated for internal events, and thus support the conclusion that the probability of the Postulated Sequence is below regulatory significance. The seismic contribution was calculated to be somewhat higher than the probability calculated for internal events. However, the Postulated Sequence requires that such a seismic event would have to be large enough to cause core damage and containment failure or bypass, and yet not damage the SFPs so as to preclude Step 6. Thus, the seismic evaluation is considered a "conservative" estimate not a "Best Estimate" as specified in the ASLB Question.

There are three main conclusions that can be drawn from the PSA applied to the chain of seven steps , and they can be qualitatively summarized based on the quantitative results and sensitivity evaluations:

1. The postulated chain of events is beyond the plant design basis.
2. The frequency of the Postulated Sequence is considered extremely low and is "remote and speculative".
3. The addition of SFPs C and D to SHNPP does not increase the frequency of the scenario. In fact, the plant modifications associated with the commissioning of SFPs C and D actually decrease the frequency of uncovering spent fuel at SHNPP. This is related to the new plant configuration which adds a viable makeup pathway under nearly all postulated accidents.

6.3 CONSERVATISMS

Despite all prudent attempts to create a best estimate evaluation, there remain some potential residual conservatisms in the quantification. Among these conservatisms are the following:

- Containment basemat failure has been treated in a manner that always causes a release into the RAB. The exact basemat failure locations are not defined in the Level 2 PSA. Therefore, this assumption has been made because of the lack of adequate information.
- A substantial fraction of the containment does not interface with the RAB. However, the dominant failure modes for containment appear to be at locations where RAB impacts cannot be ruled out. Therefore, all containment failures are assumed to impact the RAB environment.
- The SFP boil off time is taken to be the minimum it can be, given the plant configuration and the times at which freshly discharged spent fuel could be introduced into the A and B SFPs.
- The seismic evaluation is subject to large uncertainty and is believed to be a conservative bound because of the assumptions of :
  - -- Loss of site power with no opportunity for recovery
  - -- Complete dependence of failures of similar components
  - -- The early containment failure probability used in the seismic evaluation is the worst case found for any plant damage state. This is likely too conservative when applied to the seismic initiated sequences involving station blackout.
- Many motor operated pumps are located in the RAB or the FHB and are exposed to various degrees of harsh conditions, depending on their spatial relationship to the location of the primary containment failure. These pumps may fail to operate if an adequate room environment is not maintained.

An increase in the ambient temperature, due to loss of room cooling or due to primary containment failure, is the main concern. A conservative approach is taken by assuming that components fail if the room temperature exceeds the manufacturer recommended value. However, in the case of pump motors, the failure is more a function of time at temperature rather than simply exceeding a temperature limit. Therefore, continued pump operation may be likely even for temperatures exceeding manufacturer specified warranty values.

The pump motors may also fail due to moisture intrusion. The humid environment in the pump areas following primary containment failure would likely result in moisture intrusion in the CCW and ESW Booster Pump motors that could potentially result in shorted or grounded circuits. The CCW and ESW Booster Pumps are not credited with continuous operability following containment failure scenarios.

- The treatment of containment isolation failures into the RAB in the base model assumes that access to the RAB and FHB operating deck (286' Elevation) is not available. This is conservative relative to the deterministic calculations performed to support accessibility. The deterministic calculations indicate that the FHB is not affected by the Containment Isolation failure. Therefore, there is a slight conservatism in the current model. This is a conservatism, but it does not substantially reduce the calculated frequency. It also does not change the conclusions of the study.

- Air cooling of spent fuel that has low decay heat levels may be an effective cooling method (based on existing NRC National Laboratory calculations). However, this mode of cooling was not quantitatively credited in this Base Case PSA and the probability of a self-sustaining exothermic oxidation reaction in the event of uncovering a substantial portion of the spent fuel (Step 7) was assumed to be 1.0. A best estimate probability would require a detailed heat balance evaluation of the SFP, which is beyond the scope of this evaluation. The qualitative analysis of the temperatures that might be reached in SFPs C and D recognizing the heat rates of the fuel that would be stored (particularly if limited to 1.0 MBTU per hour) that was performed by CP&L would suggest that the conditional probability of Step 7 would be considerably less than 1.0.

Table 6-1

SHNPP SFPAET RESULTS BASE CASE
ACCIDENT SEQUENCE FREQUENCIES (CASE A)

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input from Level 1&2 Quantification[1]	Output from SFPAET[2]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	9.97E-9	7.44E-10
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.57E-06	3.44E-09
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.51E-06	3.31E-09
LG-ISOL	LARGE ISOLATION FAILURE	7.59E-08	9.77E-10
SM-ISOL	SMALL ISOLATION FAILURE	1.88E-07	2.59E-09
EARLY	EARLY CONTAINMENT FAILURE	3.14E-08	1.15E-09
LATE	LATE CONTAINMENT FAILURE	4.28E-06	1.43E-08
Total Internal Events Contribution		7.67E-06	2.65E-08
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-09	7.98E-11
LATE	LATE CONTAINMENT FAILURE	9.77E-07	2.86E-09
Total Fire Events Contribution		9.80E-07	2.94E-09
Total Seismic Contribution		-	8.65E-08
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	7.2E-07	1.45E-08

(1) CDF with containment failure, bypass, or containment isolation failure(per yr).
(2) Frequency of the loss of effective water cooling to the spent fuel(per yr).

Section 7

REFERENCES

[2-1] Transmittal of Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Plants and Federal Register Notice Requesting Public Comments of Technical Study, from Richard F. Dudley, Jr., Senior Project Manager (RA), (DLPM NRR) dated February 15, 2000.

[2-2] Personal Communication, Steve Edwards (CP&L) to E.T. Burns (ERIN), September 26, 2000.

[2-3] Babrauskas, et. al, Toxic Potency Measurement for Fire Hazard Analysis, NIST Special Publication 827, December, 1991.

[2-4] Blockley, W.V., Taylor, C.L., Human Tolerance Limits for Extreme Heat, in Heating, Piping and Air Conditioning Journal, May 21, 1949, P.111-116.

[3-1] Probabilistic Risk Assessment (PRA),Peer Review Process Guidance, NEI 00-02, Rev. A3, Nuclear Energy Institute Risk-Based Applications Task Force, 2000.

[3-2] Independent Peer Review Process of the SHNPP PSA, Erin Engineering and Research, Inc., November 2000.

[3-3] USNRC Information Notice 2000-13: Review of Refueling Outage Risk, dated September 27, 2000.

[3-4] Interfacing Systems LOCA (ISLOCA) Analysis, Probabilistic Safety Assessment - Appendix H, Revision 2, October 2000.

[3-5] Professional Loss Control, Inc. Fire-Induced Vulnerability Evaluation (FIVE), TR-100370, EPRI Research Project 3000-41, Rev. 1, September 1993.

[4-1] Personal Communication, Eric McCartney (CP&L) to E.T. Burns (ERIN), October 4, 2000.

[4-2] Swain, A.D., Guttmann, H.E., Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications, NUREG/CR-1278, August 1983.

[4-3] Parry, G.W., An Approach to the Analysis of Operator Actions in Probabilistic Risk Assessment, EPRI TR-100259, June 1992

[4-4] Oliver, R.E., Howe, A.J., and et al., Individual Plant Examination for External Events Submittal, Carolina Power & Light Company, June 1995.

[4-5] Letter from NRC (JW Shea NRR) to D.A. Lochbaum, Susquehanna Steam Electric Station, Units 1 and 2, Draft Safety Evaluation Regarding Spent Fuel Pool Cooling Issues (TAC-No M85337), dated October 25, 1994.

[4-6] NUREG-6144, Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1, Vol. 2, Part 1A.

[4-7] SAND99-1815, Summary of Information Presented at an NRC-Sponsored Low-Power Shutdown Public Workshop, April 27, 1999, Rockville Marlyland.

[4-8] Staff Requirements – SECY-00-0007- Proposed Staff Plan for Low Power and Shutdown Risk Analysis Research to Support Risk-informed Regulatory Decision Making, March 31, 2000.

[4-9] An Analysis of Loss of Decay Heat Removal Trends and Initiating Event Frequencies (1989 – 1998), EPRI, Palo Alto, CA, EPRI TR-113051, 1999.

[4-10] Carolina Power and Light, Shearon Harris Nuclear Power Plant, Plant Operating Manual, "Outage Shutdown Risk Management", Outage Management Procedure, Volume 9, Part 1, OMP-003, Revision 12.

[4-11] Carolina Power and Light, Shearon Harris Nuclear Power Plant, Plant Operating Manual, "Control of Plant Activities During Reduced Inventory Conditions", Outage Management Procedure, Volume 9, Part 1, OMP-004, Revision 8.

[4-12] ERIN Engineering and Research, Incorporated, "An Analysis of PWR Shutdown Risk", October 2000.

[4-13] Nuclear Regulatory Commission, "NRC Information Notice 2000-13: Review of Refueling Outage Risk", September 27, 2000.

[4-14] Nuclear Regulatory Commission, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications", NUREG/CR-1278, August 1983.

[4-14] PECO Nuclear, "Peach Bottom Atomic Power Stations Units 2 and 3, Probabilistic Safety Assessment (PSA) Human Reliability Analysis (HRA), 1999 Update, U99AF-03.

[4-15] Sobel, P., "Revised Livermore Seismic Hazard estimates for sixty – nine Nuclear

Plant Sites East of the Rocky Mountains", NUREG – 1488, April 1994.

[4-16] Electric Power Research Institute, "Seismic Hazard Methodology for Central and Eastern United States," EPRI NP-4726, November 1988.

[4-17] Bernreuter, D.L. and et.al., "Seismic Hazard Characterization of the Eastern United States: Comparative Evaluation of the LLNL and EPRI Studies NUREG/CR-4885, May 1987.

[4-18] Budnitz, R.J. and et.al., "An approach to the Quantification of Seismic Margins in Nuclear Power Plants", NUREG/CR-4334, August 1985.

[4-19] SHNPP FSAR, Section 2.3, Wind Rose.

[4-20] Walkdown for Seismic items, Ronald Knott (CP&L) to Steve Laur (CP&L), dated October 25, 2000.

[4-21] USNRC Information Notice 2000-13: Review of Refueling Outage Risk, dated September 27, 2000.

[4-22] Personal Communication, Steve Laur (CP&L) and Bruce Morgen (CP&L) to E.T. Burns and J.R. Gabor (ERIN), dated October 24, 2000.

[4-23] E-mail Communication, Steve Laur (CP&L) to Larry Lee (ERIN), attached file hnpsum2000.xls, October 11, 2000.

[4-24] Shearon Harris Nuclear Power Plant Unit 1, Individual Plant examination for External Events (IPEEE) Submittal, June 1995.

Appendix A

SPENT FUEL POOLS AND ASSOCIATED EQUIPMENT

Appendix A

SPENT FUEL POOLS AND ASSOCIATED EQUIPMENT

This Appendix provides a description of the key features of the Shearon Harris fuel handling building (FHB) and spent fuel pools (SFPs) and the systems that perform important functions associated with the SFPs. The appendix includes the following:

- Description of the location of the SFPs in the FHB
- Description of the SFPs
- Description of the SFP cooling and support systems
- Description of makeup methods for adding water to the SFP
- Description of the instrumentation used to monitor the SFP and cue any operator actions to the maintenance of adequate fuel cooling

A.1 FUEL HANDLING BUILDING

The Harris Fuel Handling Building is atypical of many nuclear power plants because of its large size. The FHB was constructed to accommodate a four unit site. Therefore, the size and compartmentalization of the building makes its response to a loss of cooling potentially different than many other sites. This feature of the Harris FHB has been explicitly represented in the deterministic calculations of post containment failure accident sequences.

<u>Fuel Handling Building</u>

The Shearon Harris Nuclear Power Plant (SHNPP) FHB is situated to the east of the Unit 1 power block and to the north of the Waste Processing Building (WPB). Its south wall abuts the WPB. Its east wall abuts the Unit 1 Reactor Auxiliary Building (RAB). Its west wall abuts structures that were to have been the Unit 4 and Unit 3 RABs. Its north wall does not abut any structures.

Figures A-1 through A-4 show the various elevations of the FHB.

The FHB consists of four levels plus the roof:

- 337 ft elevation – Roof. Notable components located on the roof include the RAB / FHB HVAC exhaust stack. Access to the FHB roof is from the adjacent RAB roof.

- 286 ft elevation – Main operating floor and top of all SFPs and transfer canals. Notable components located on this elevation of the FHB include: the fuel handling bridge; Fuel Pool Cooling and Cleanup System (FPCCS) skimmer subsystem skimmers (23) floating on the surface of the SFPs and canals; demineralized water system manual valve stations (19) along the west and east walls; FPCCS skimmer subsystem manual valves located along the tops of the SFPs and canals in service valve boxes; seven fire hose stations, each containing a 1.5" fire hose; FHB control panels FP-9 and FP-10 along the east wall; and the FHB 10 ton auxiliary crane. In addition, two 480 VAC General Service Buses (1-4A102 and 1-4B102) are located in a separate room on the south end of this elevation; this room may only be entered from the outside, from doors located off of the WPB roof. The FHB operating floor may be accessed through doors D893 and D894 in the southwest wall from the WPB stairwell. "Tornado" door D892 leads into this same stairwell airlock area from the FHB roof. There are two stairwells and a freight elevator in the north end of the FHB. The elevator and one of the stairwells go to the railroad bay at elevation 261. The second stairwell provides access to rooms in the northern ends of the 261 ft elevation, the 236 ft elevation and the 216 ft elevation.

- 261 ft elevation (site grade level) – Fuel unloading area (rail access bay) on the north end and a ventilation equipment room (with an attached demineralizer room on its south end) on the south end. Notable components located in the ventilation equipment room (room FH6) on this elevation of the FHB include: normal FHB HVAC and emergency exhaust equipment; 480 VAC motor control centers MCC-1&4A33-SA and MCC-1&4B33-SB (in mechanical equipment sub-room FH7); 480 VAC motor control centers 1-4A1021, 1-4A1022, 1-4B1021 and 1-4B1022; and the FPCCS purification subsystem demineralizers. Access to the ventilation equipment room is through combination double doors / single door D119 in the east wall from the RAB 261 ft elevation. Access to the demineralizer room is either through an open passageway from the south end of the ventilation equipment room or directly through a single door in the east wall from the RAB 261 ft elevation. Access to the railroad bay is from the outside through a large, airtight sliding door on the north end; from a stairwell and an elevator from the 286 ft elevation of the FHB; from the outside through air-tight double man doors to the right of the railroad door; or, from the outside through "tornado" door D3312 in the east wall.

- 236 ft elevation – This elevation of the FHB is comprised of three distinct areas: A room at the south end of the building that does not contain any equipment considered in the SFP cooling or makeup analysis; an equipment area in the central portion of the building; and, a room at the north end of the building. Key components located on this elevation of the FHB in the central equipment room include: FPCCS skimmer subsystem pumps, filter, strainers and demineralizers; FHB control panels FP-7 and FP-8 and associated instrument racks; and FPCCS cooling subsystem pumps, heat exchangers (cooled by component cooling water) and strainers. Access to this room is through either double doors D6500 or adjacent single door D650 from the 236 ft elevation of the RAB in the east wall, or through a single "tornado" door in the west wall from the fabrication shop at the 236 ft elevation (an area that was to have been the Unit 3 RAB). The North 236 ft elevation contains access to that elevation from exterior to the FHB and also access to the North 216 ft elevation.

- 216 ft elevation – Two completely separated compartments (North and South) containing: four (4) FPCCS purification subsystem pumps; demineralized water cross-tie valves 1SF-201 (South 216 ft.) and 2SF-201 (North 216 ft.); FHB floor drains and equipment drains sumps and sump pumps (North and South); FHB HVAC condensate recirculation transfer pump and tank (South room only); FPCCS filter backwash pumps and tanks (North and South); and component cooling water system transfer pump and holdup tank (North room only).

  Access to the South room is through single door D725 in the East wall near the South end or a double door in the east wall near the north end from the 216 ft elevation of the RAB.

Access to the North room is from: (a) the FHB northeast stairway via the 286 ft elevation of the FHB; (b) down the same stairway after entering the North end of the FHB at the 236 ft elevation through "tornado" door D3312 from the safety meeting room in what was to have been the Unit 3 RAB; or, (c) from the 236 ft elevation North end area via a ladder stored at that location without requiring access to the stairwell.

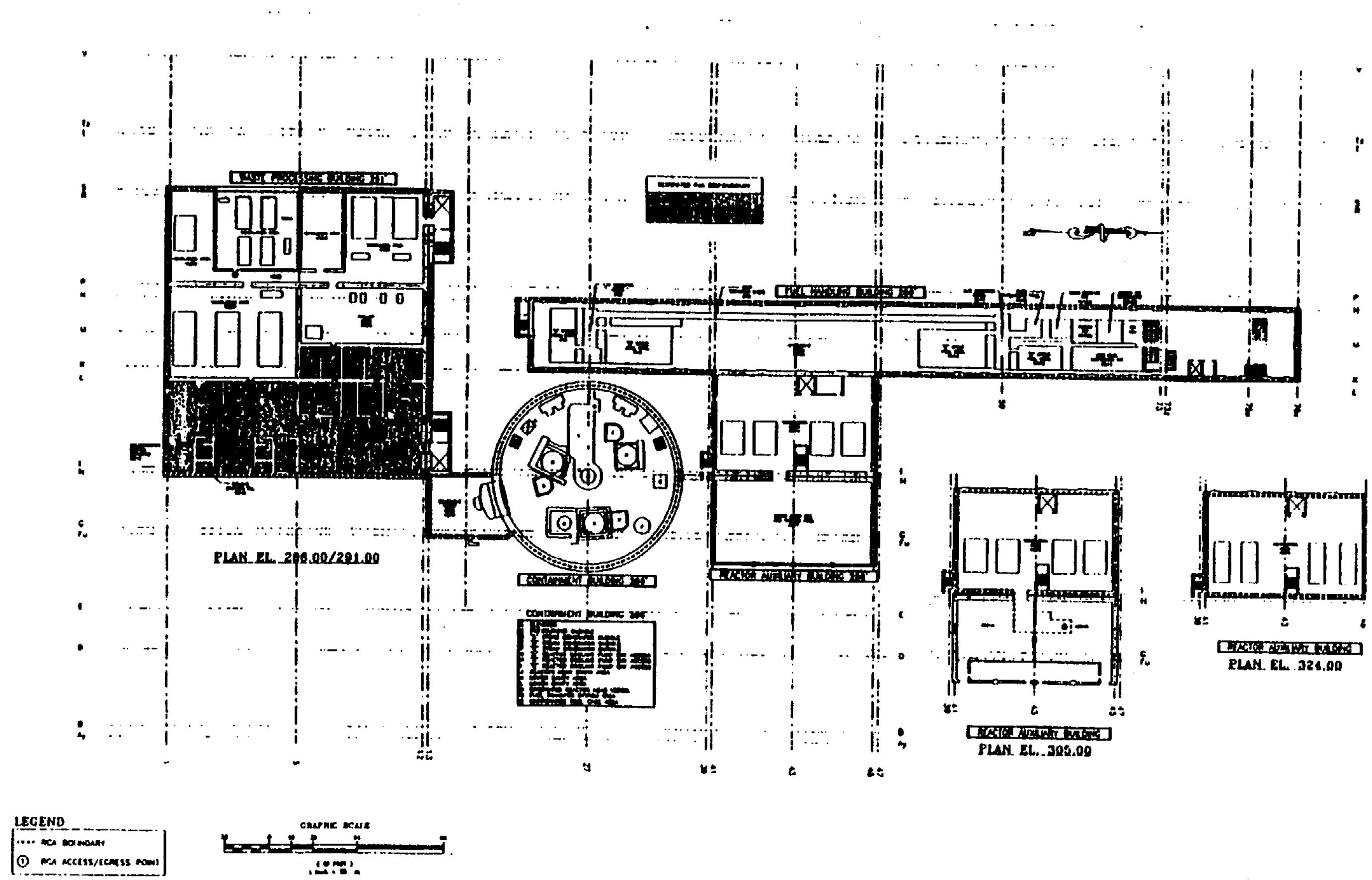


Figure A-1 Elevation 286' of RAB and Operating Deck Level of FHB

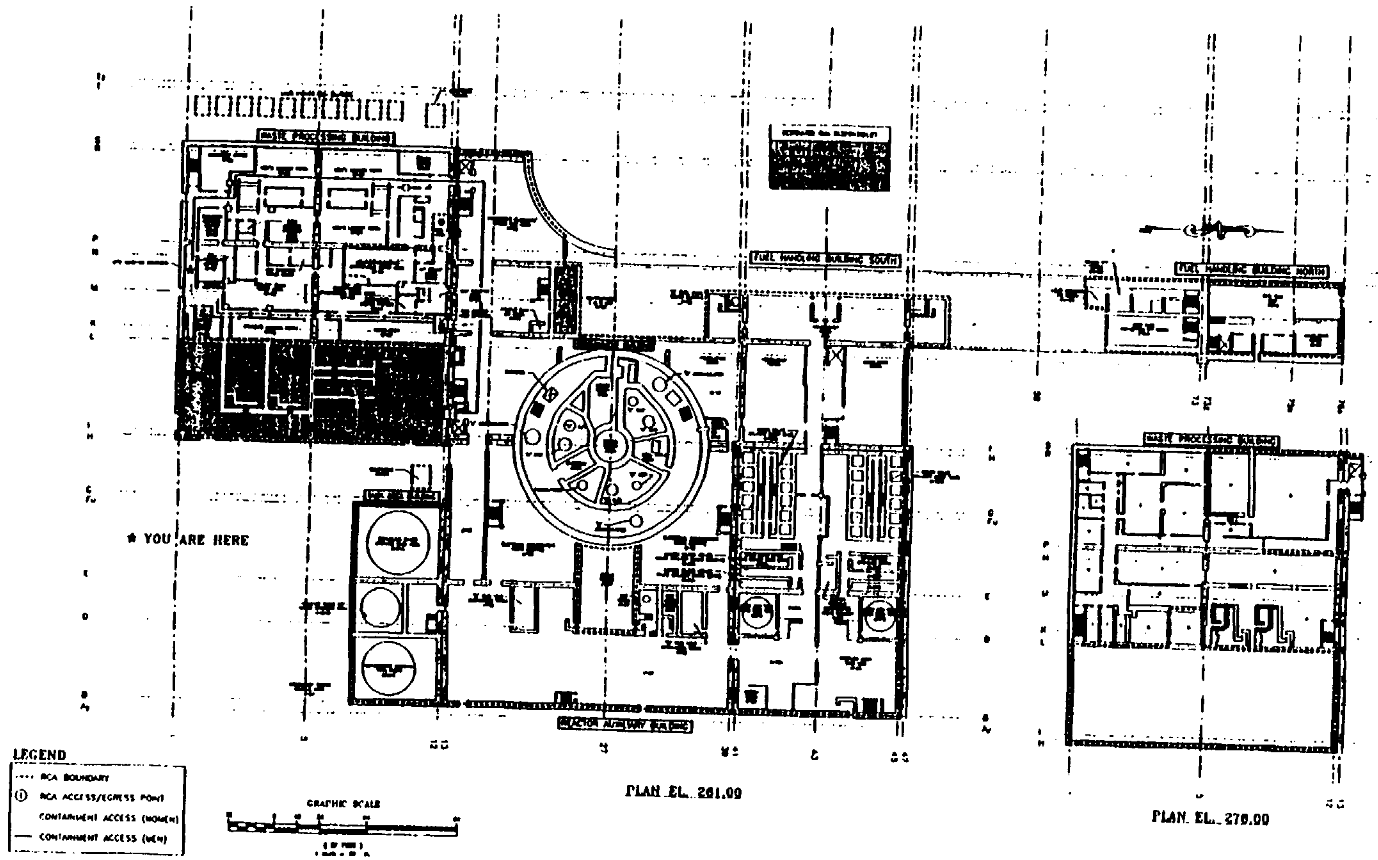


Figure A-2 Elevation 261' of FHB and RAB

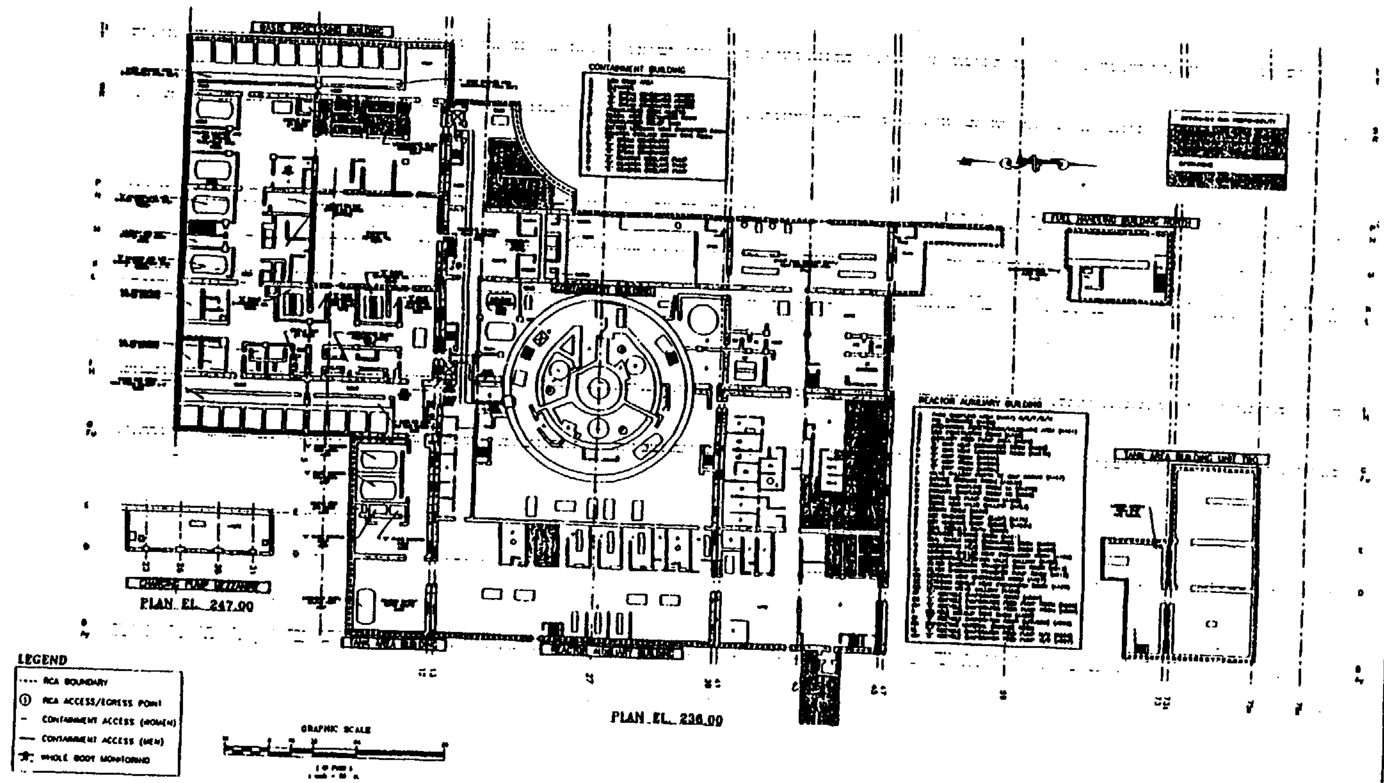


Figure A-3 Elevation 236' of RAB and FHB

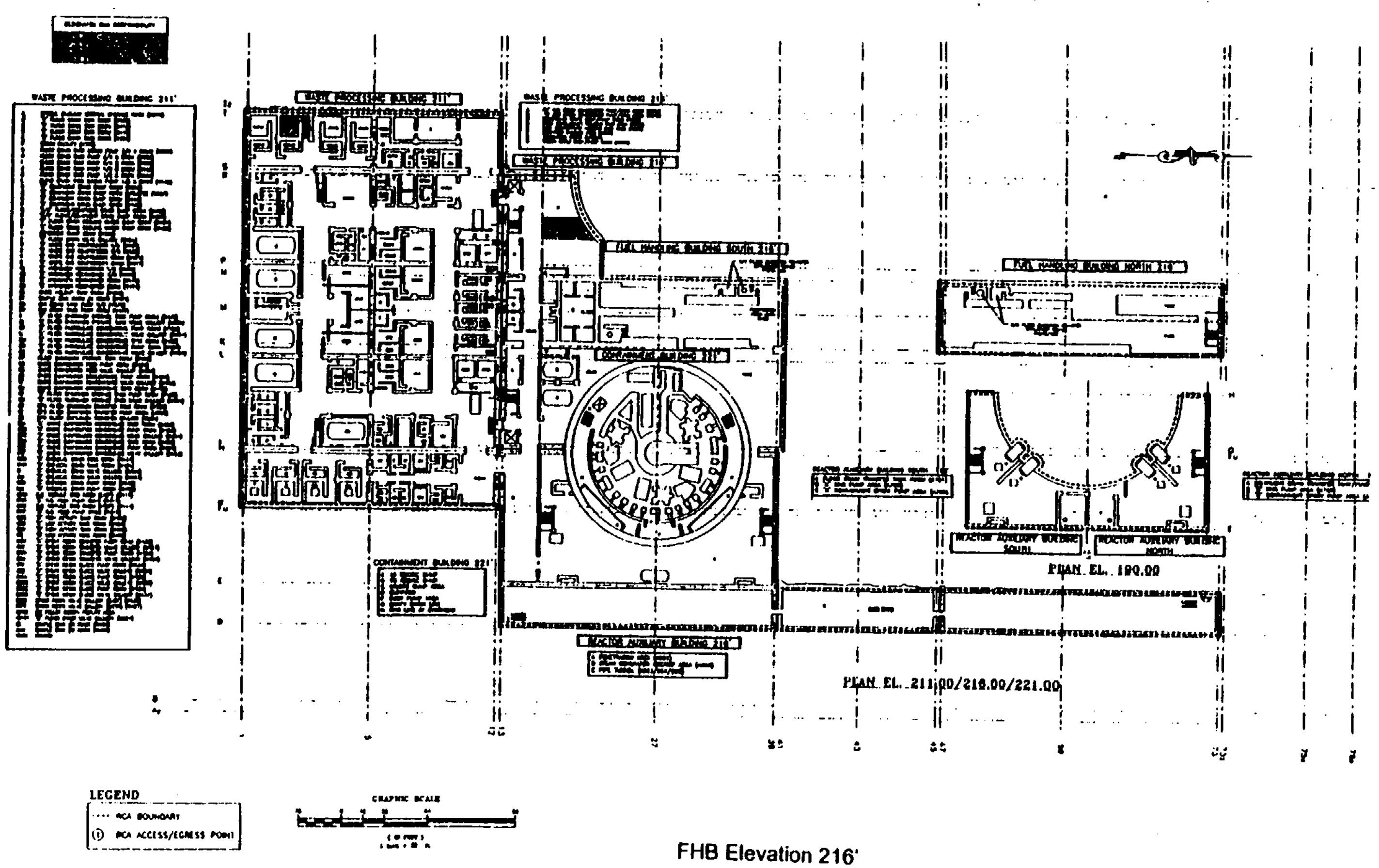


Figure A-4 FHB Elevation 216'

A.2 SPENT FUEL POOLS

A.2.1 Fuel Pools

The FHB contains five main pools. The south end of the FHB contains the new fuel pool (Pool "A") and a spent fuel pool (Pool "B"). The north end of the FHB contains two spent fuel pools (Pools "C" and "D") and the spent fuel shipping cask loading pool (Cask Loading Pool). These five pools are tied together by 3 interconnected canals: the Main Transfer Canal, the South Transfer Canal and the North Transfer Canal.

The four SFPs and the Cask Loading Pool are reinforced concrete structures with stainless steel liners. The bottoms of the four SFPs are at elevation 246.00 ft. Normal water level in the SFPs is maintained at 284.5 ft. The bottom of the Cask Loading Pool is at elevation 240.00 ft. Normal water level in this pool is maintained at 284.5 ft, consistent with the SFPs.

Draining or siphoning of the pools via piping or hose connections to the pools or the canals is precluded by the location of the penetrations, limitations on hose length, and the termination of piping penetrations flush with the liner. Main Control Room and local alarms are provided to alert operators to abnormal pool levels or high temperatures.

A.2.2 Main Transfer Canal

The Main Transfer Canal runs south to north (parallel to the west wall of the FHB) between the northwest corner of the South Transfer Canal and the southwest corner of the North Transfer Canal.

The Main Transfer Canal is a concrete structure with a stainless steel liner. The bottom of the Main Transfer Canal is at elevation 260.00 ft. Normal water level in the canal is maintained at 284.5 ft, consistent with the fuel pools.

A.2.3 South Transfer Canal

The South Transfer Canal runs west to east between Pools A and B. The Fuel Transfer Tube to the SHNPP Unit 1 Containment enters the east end of the South Transfer Canal. The South Transfer Canal is also connected by channels to Pools "A" and "B."

The South Transfer Canal is a concrete structure with a stainless steel liner. The bottom of the South Transfer Canal is at elevation 251.00 ft. Normal water level in the canal is maintained at 284.5 ft, consistent with the fuel pools.

A.2.4 North Transfer Canal

The North Transfer Canal runs west to east between Pool C and Pool D and the Cask Loading Pool. The North Transfer Canal is connected by channels to Pools "C" and "D" and the Cask Transfer Pool.

The North Transfer Canal is a concrete structure with a stainless steel liner. The bottom of the North Transfer Canal is at elevation 251.00 ft. Normal water level in the canal is maintained at 284.5 ft, consistent with the fuel pools.

A.2.5 Isolation Gates

Nine movable bulkhead gates may be used to isolate the five pools from each other:

- Gate 1 (1SF-E001) – Isolates the South Transfer Canal from the Main Transfer Canal.

- Gate 2 (1SF-E002) – Isolates the Main Transfer Canal from Pool "B."
- Gate 3 (1SF-E003) – Isolates the South Transfer Canal from Pool "B."
- Gate 4 (1SF-E004) – Isolates the South Transfer Canal from Pool "A."
- Gate 5 (1SF-E005) – Isolates the North Transfer Canal from the Main Transfer Canal.
- Gate 6 (1SF-E006) – Isolates the Main Transfer Canal form Pool "C."
- Gate 7 (1SF-E007) – Isolates the North Transfer Canal from Pool "C."
- Gate 8 (1SF-E008) – Isolates the North Transfer Canal from the Cask Loading Pool.
- Gate 9 (1SF-E009) – Isolates the North Transfer Canal from Pool "D."

The bulkhead gates are constructed of stainless steel plate and structural steel members. The sides and the bottoms fit into slots in the SFP's canal walls and floors. Inflatable rubber seals are installed in the sides of the bulkhead gates. The seals are inflated by Instrument Air (IA) once the gates are set in place. IA enters each installed gate's seals via a separate line attached with a quick disconnect plug at the top of the gate. Figure A.2-1 is a simplified schematic of the gate locations in the Spent Fuel Pools.

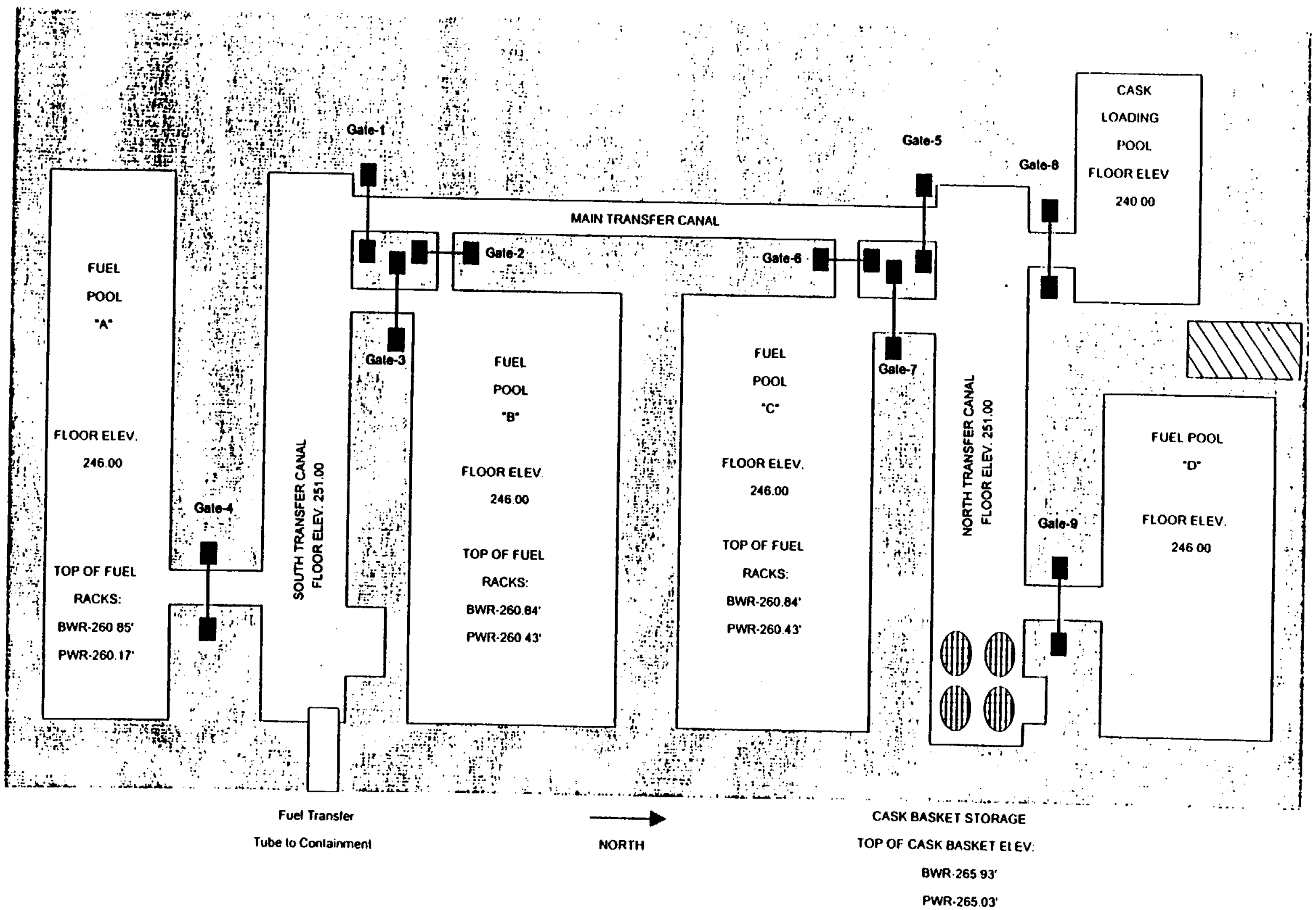


Figure A.2-1 Simplified Schematic of Gate Locations

00094

The gates are moved using the 12-ton FHB Auxiliary Crane (see SHNPP Operations Procedure OP-116 Section 8.27 and Attachment 7). The FHB Auxiliary Crane is powered from 480 VAC MCC 1-4B1022 (fed from General Service Bus 1-4B). The FHB Auxiliary Crane is not available in the event of a loss of off-site power. When they are not in use, the bulkhead gates are placed in dedicated storage areas in the Main Transfer Canal.

Gates (2 and 6) between the pools and the Main Transfer Canal are normally installed. Gates (3, 4, 7, and 9) between the SFPs and the North and South Transfer Canals are not normally installed. Installation and/or removal of gates during an emergency is estimated to require approximately 60 to 90 minutes per gate. Removal of gates in the event of a loss of SFP cooling is not procedurally required. In the case where makeup water from adjacent pools and transfer canals is needed to mitigate a loss of water inventory in a pool, removal of the gates is not required. The pneumatic seal on the gates can be deflated (within a period of minutes) via removal of a quick disconnect fitting or sufficient water can be injected to overflow the gates. Deflating a gate allows water to flow past the gate until an equilibrium water level condition is established. Under these conditions, the exchange and re-equilibration of water between the isolated pool (i.e., gate installed but deflated) and adjacent pools or canals is rapid, and typically occurs on a timescale of minutes. The model built for these analyses contains flag events that may be individually set for each gate; setting a gate's flag event to TRUE would represent that gate being installed.

The gates between SFPs A and B and those between SFPs C and D will be removed under most foreseeable circumstances. There is a very remote potential that maintenance could be required on the pools or transfer canal. This could necessitate installation of the gates for a very snort time. This is estimated to occur 1.0% of the time. [Eric McCartney, 9/29/00]. The percentage of time, on an annual basis, that the spent fuel pools would be operated with the gates removed is summarized as follows:

Estimated Percentage of Time, on an Annual Basis, the Bulkhead Gates Would be Normally Removed from the SHNPP Spent Fuel Pools Subsequent to Operational Use of C and D Pools	
Gate Number	Best Estimate Time Gates Removed [A-1]
1	99% [A-28]
2	1%
3	99%
4	99%
5	1%
6	1%
7	99%
8	99%
9*	99%

* The "normally open" configuration for gate 9 (gate removed 99% of the time) would apply subsequent to placing this pool in service, scheduled for early the next decade. Otherwise, this gate would remain normally closed.

The top of the pools and transfer canals (286 ft) is 10.5 inches above the top of the installed gates [A-2]; i.e., the tops of the installed gates are at an elevation of 285 feet 1 ½ inches (285.125 feet). The normal water level of the SFPs and the canals is 284.5 feet, which is 0.625 feet below the top of the installed gates.

A.2.6 Spent Fuel Pool Configurations

The SFP configuration is such that even with the SFP gates in place there would be communication among pools if makeup flow continues to flood a single pool. The water would overflow the gates, but not overflow out of the pools. This overflow would eventually flood all pools.

The boil off rate for the highest heat rate (SFPs A + B @ 25E+6 Btu/hr pool) is estimated at 52 gpm. Therefore, as long as the makeup exceeds this value all pools can be flooded.

The volume to flood the A + B South Canal + Main Transfer Canal pools from the low level point (284') to the overflow of the pools above the gates is 23,000 gal.

A.3 FUEL POOL COOLING AND HEATUP

A.3.1 Fuel Pool Cooling

The Fuel Pool Cooling and Cleanup System (FPCCS) has two primary purposes. It is designed to maintain water quality by removing particulate and dissolved fission and corrosion products resulting from the spent fuel stored in the pools; it is also designed to remove residual heat generated by the spent fuel stored in the pools and to maintain an adequate water inventory in the pools.

The FPCCS consists of the following three subsystems:

1. FPCCS Cooling Subsystem – Pools "A" and "B" are currently served by a two-loop FPCCS cooling subsystem. Major components in each of these loops include a pump, a heat exchanger and a strainer. The heat exchanger is cooled by the Component Cooling Water (CCW) system in the Reactor Auxiliary Building (RAB). Each of the 4560 gpm horizontal centrifugal pumps are able to be powered from the emergency diesel generators (EDGs) following a

loss of off-site power. Each loop of this cooling system is 100% capacity and is indépendent of the other loop. The pumps are locally controlled from panels FP-7 and FP-9 located in the FHB.

Pools "C" and "D" will be served by a two-loop FPCCS cooling subsystem identical to the system in pools "A" and "B". Installation of this subsystem is scheduled for completion by the end of 2000; it will, therefore, be fully operational prior to commissioning pools "C" and "D" for spent fuel storage. The proposed modification is adopted in this analysis as present when pools "C" and "D" are operational.

2. FPCCS Cleanup / Purification Subsystem – Pools "A" and "B" are currently served by a two-loop FPCCS cleanup subsystem. Major components in each of these loops include a fuel pool demineralizer, a fuel pool demineralizer filter, a fuel pool and refueling water purification filter and a 325 gpm pump. Each of these pumps is capable of taking suction from the canals, the pools, the Unit 1 refueling cavity in Containment and the RWST via the containment spray (CS) system. The system is operated only as needed.

   Pools "C" and "D" will be served by a two-loop FPCCS cleanup subsystem identical to the system in pools "A" and "B". Installation of this subsystem is scheduled for completion by the end of 2000; it will, therefore, be fully operational prior to commissioning pools "C" and "D" for spent fuel storage.

3. Fuel Pools Skimmer System - Pools "A" and "B" are currently served by a skimmer system that consists of a 385 gpm pump, a strainer and a filter. The system removes any floating debris from the surface of the pools and canals via 15 floating skimmers deployed as follows:

   - Pool "A" 3
   - Pool "B" 5
   - South Transfer Canal 2
   - Main Transfer Canal 2
   - North Transfer Canal 2
   - Cask Loading Pool 1

Pools "C" and "D" will be served by their own FPCCS skimmer subsystem identical to the system in pools "A" and "B". Five skimmers will serve pool "C"; three skimmers will serve pool "D". Installation of this subsystem is scheduled for completion by the end of 2000; it will, therefore, be fully operational prior to commissioning pools "C" and "D" for spent fuel storage. This analysis assumes that the modifications are in service when modeling the pools "C" and "D" FPCCS skimmer subsystem.

A.3.2 Fuel Pool Heatup

Calculations were performed by CP&L to determine the time required to reach boiling temperature and then the additional time required to boil the water to the top of the spent fuel racks for spent fuel pools A and B and for spent fuel pools C and D, with loss of spent fuel pool cooling and no operator action. The results of these calculations are summarized below.

The results of these calculations are summarized below:

Pools	Time to reach boiling temperature	Additional time for water level to reach top of racks	Total time	Makeup required to offset boiling
A and B (Beginning of cycle)	20.57 hours	7.21 days	8.07 days	53.70 gpm
A and B (End of cycle)	38.67 hours	13.56 days	15.17 days	28.57 gpm
C and D (1 MBTU/hr heat load)	384.66 hours	99.99 days	116.02 days	2.15 gpm
C and D (15.6 MBTU/hr heat load)	34.42 hours	8.80 days	10.23 days	33.64 gpm

These calculations did not take credit for any additional cooling or makeup that would be available to the pools.

The cases for which calculations have been performed include the following:

A & B (Beginning of cycle):	This represents a case which involves a fuel core off load into SFP "A". This represents the limiting or shortest time for a pool to boil.
A & B (End of cycle):	This represents a case which involves the condition at the end of a fuel cycle after a full core off load has decayed. This condition is less limiting than the BOC case.
C & D (1.0 MBTU/Hr):	This case represents a situation in which only a small amount of 5 year old fuel[(1)] is placed in the C pool.
C & D (15.6 MBTU/Hr):	This case represents a situation in which the C & D pools are filled with spent fuel, all of which is 5 years or older.

A.4 NORMAL WATER MAKEUP TO FUEL POOLS

Multiple water makeup sources to the A & B SFPs are available and proceduralized. This section discusses these proceduralized makeup methods, and Section A.5 discusses some non-proceduralized methods. Following the installation of plant modifications associated with SFPs C and D, a completely redundant SFP cooling system, purification system, and skimmer system will be installed in the North end of the FHB. This will provide redundant delivery locations for operators to align existing makeup water sources to SFPs C and D, transfer canals, and the cask loading pool. Operating procedures (OP-116) will be revised to reflect the redundant makeup water pathways to SFPs C and D prior to adding spent fuel to pool C.

Normal makeup to the pools and canals is accomplished by aligning the purification pumps to take suction from the demineralized water (DW) system. This is done by either opening locked closed manual valve 1SF-201 or 2SF-201 with the FPCCS Cleanup/Purification Subsystem in operation. These valves are located in the South and

(1) Fuel that has been removed from the RPV for more than 5 years.

North ends of the 216 ft Elevation of the FHB, respectively. Details of this lineup are contained in SHNPP Operating Procedure OP-116 Section 8.4.

CP&L [A-1] identified that the purification pumps are not required to run for success of this path. Demineralized water system pump operation is likely required. The flow paths for use of DW into the SFPs includes this method without the purification pumps running. Therefore, while the preferred and normal method of makeup is through the purification system pumps, the purification pumps need not to be running to obtain flow into the SFP through the normally open suction line up(1). [Eric McCartney, 9/29/00]. The source of water is the demineralized water storage tank, which has a capacity of 500,000 gallons. The flow rate is 100 gallons per minute. The operator can initiate this flow path in approximately five minutes, excluding any transit time.

Table A-1 is a summary of the normal and supplemental SFP makeup methods (See Section A.5 for discussion of the supplemental makeup methods). Table A-1 identifies the normal methods of SFP makeup to be from the DW system to the SFP via the locked closed manual valves on the 216' elevation of the FHB. This is labeled as method PB in Table A-1.

(1) Because the purification system is normally operating, the manual suction valves are open to at least one of the SFPs associated with the system. This is estimated at 99% by CP&L [Eric McCartney, 9/29/00]

In the following figures, the valve positions under normal operation are shown. The following indicates valve position:

- "Blackened" valve – normally closed
- "White" valve – normally open

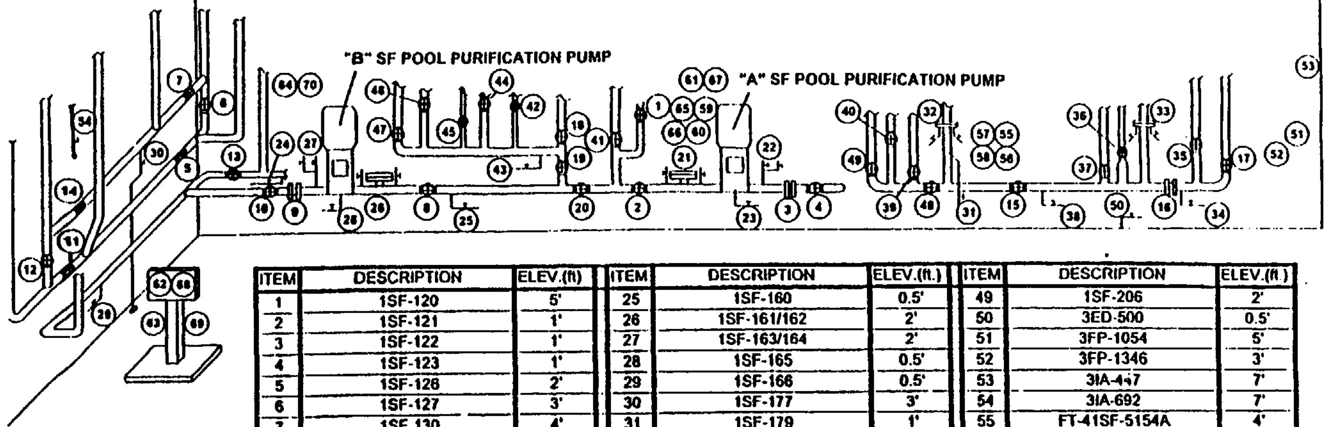
Figure A.4-2 shows the FHB South 216' Elevation and the specific locked closed manual valve that needs to be opened (1SF-201). This arrangement is similar to that in the North 216' Elevation.

Figure A.4-3 shows a simplified diagram of the flow path through 2FS201 and back through the suction of the clean up pumps.

Figure A.4-3 is similar to Figure A.4-2 except it shows the pathway into SFPs C&D through FPCCS when clean up is not in service. Manual valve 2SF-201 is required to be opened.

Figures A.4-4, A.4-5, and A.4-6 are simplified schematics for pathways when the FPCCS cleanup or skimmer pump is in service. These pathways are beneficial under most non-severe accident conditions. However, for the Postulated Sequence included in the ASLB Order, these line ups are not substantial benefits.

FHB 216' SOUTH - SPENT FUEL POOL PURIFICATION



ITEM	DESCRIPTION	ELEV.(ft)	ITEM	DESCRIPTION	ELEV.(ft.)	ITEM	DESCRIPTION	ELEV.(ft.)
1	1SF-120	5'	25	1SF-160	0.5'	49	1SF-206	2'
2	1SF-121	1'	26	1SF-161/162	2'	50	3ED-500	0.5'
3	1SF-122	1'	27	1SF-163/164	2'	51	3FP-1054	5'
4	1SF-123	1'	28	1SF-165	0.5'	52	3FP-1346	3'
5	1SF-126	2'	29	1SF-166	0.5'	53	3IA-447	7'
6	1SF-127	3'	30	1SF-177	3'	54	3IA-692	7'
7	1SF 130	4'	31	1SF-179	1'	55	FT-41SF-5154A	4'
8	1SF-131	1'	32	1SF-180/181	4'	56	FT-41SF-5154A-E/HD1/HD2/ HI1/HV1/LD1/LD2/LI1/LV1	1'-3'
9	1SF-132	1'	33	1SF-182/183	4'			
10	1SF-133	1'	34	1SF-184	0.5'	57	FT-41SF-5154B	4'
11	1SF-136	2'	35	1SF-187	3'	58	FT-41SF-5154B-E/HD1/HD2/ HI1/HV1/LD1/LD2/LI1/LV1	1'-3'
12	1SF-137	3'	36	1SF-188	3'			
13	1SF-138	1'	37	1SF-189	2'	59	PI-41SF-5190A	5'
14	1SF-139	4'	38	1SF-190	0.5'	60	PI-41SF-5190A-D1/D2/I1/V2	0.5'-4'
15	1SF-141	1'	39	1SF-191	2'	61	PI-41SF-5190A-V1	5'
16	1SF-142	1'	40	1SF-192	3'	62	PI-41SF-5190B	5'
17	1SF-143	2'	41	1SF-193	4'	63	PI-41SF-5190B-D1/D2/I1/V2	0.5'-3'
18	1SF-148	3'	42	1SF-194	4'	64	PI-41SF-5190B-V1	5'
19	1SF-149	2'	43	1SF-195	2'	65	PS-41SF-5190A	5'
20	1SF-150	1'	44	1SF-200	4'	66	PS-41SF-5190A-D1/D2/I1/V2	0.5'-4'
21	1SF-151/152	2'	45	1SF-201	3'	67	PS-41SF-5190A-V1	5'
22	1SF-153/154	2'	46	1SF-202	4'	68	PS-41SF-5190B	5'
23	1SF-155	0.5'	47	1SF-203	3'	69	PS-41SF-5190B-D1/D2/I1/V2	0.5'-3'
24	1SF-156	0.5'	48	1SF-205	1'	70	PS-41SF-5190B-V1	5'

FHB 216' South - Spent Fuel Pool Purification

Figure A.4-1 FHB 216' South - Spent Fuel Pool Purification

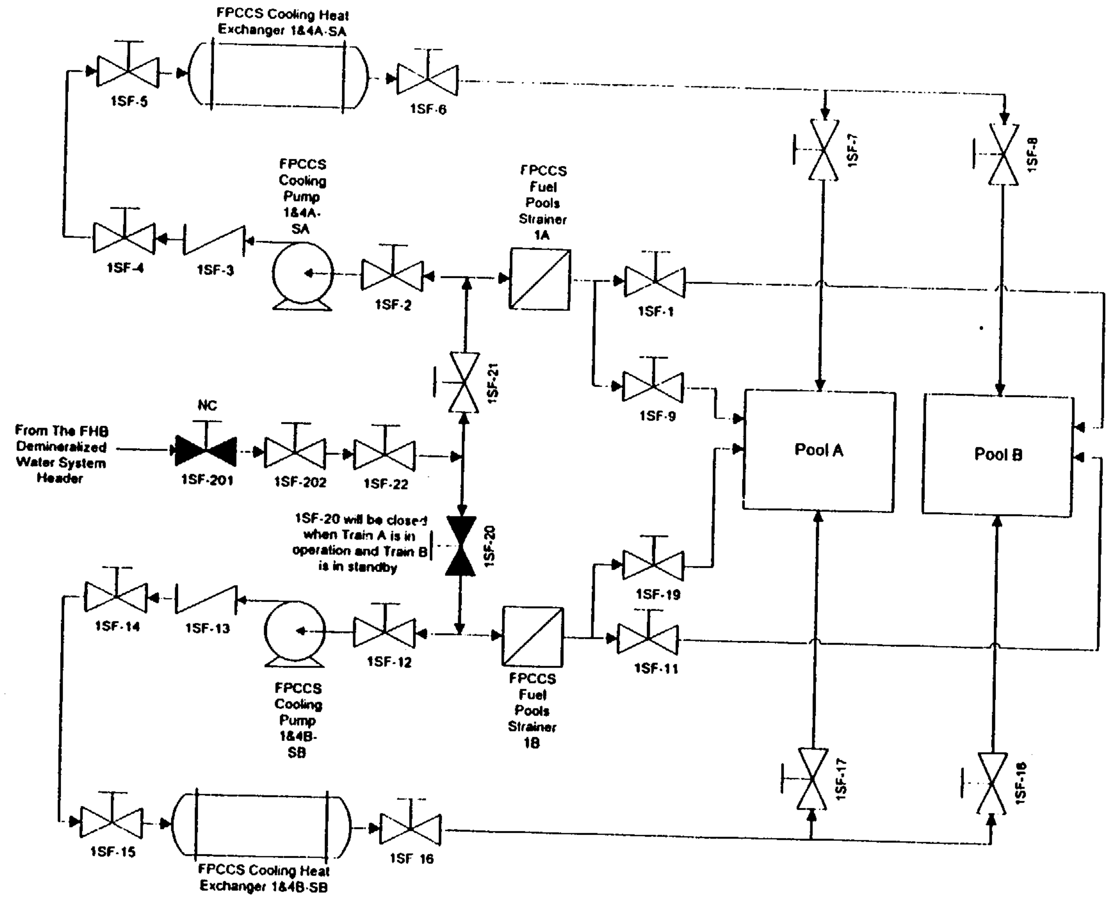


Figure A.4-2 Demineralized Water Makeup to Pools A and B with FPCCS Cleanup Not in Service

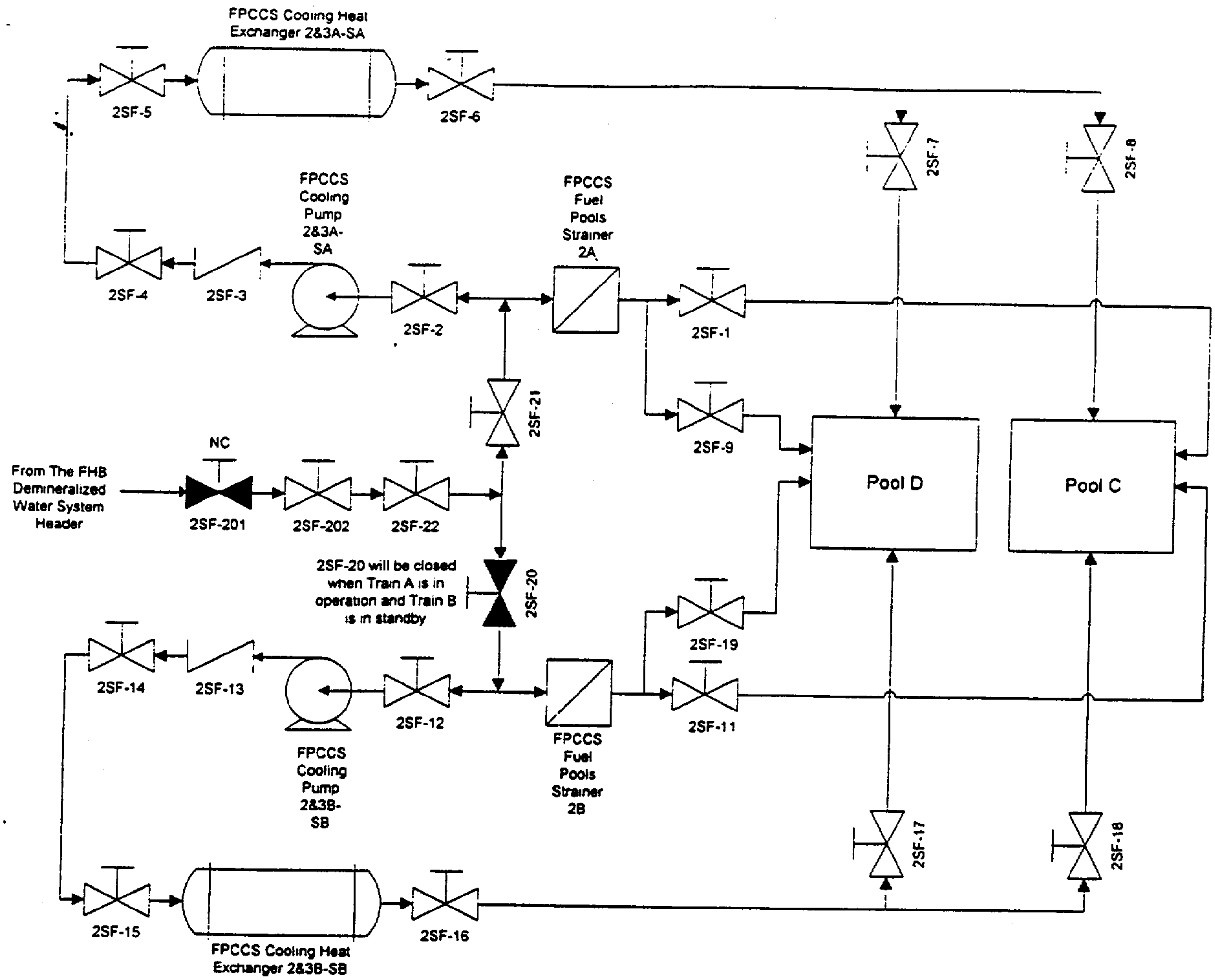


Figure A.4-3 Demineralized Water Makeup to Pools C and D with FPCCS Cleanup Not in Service

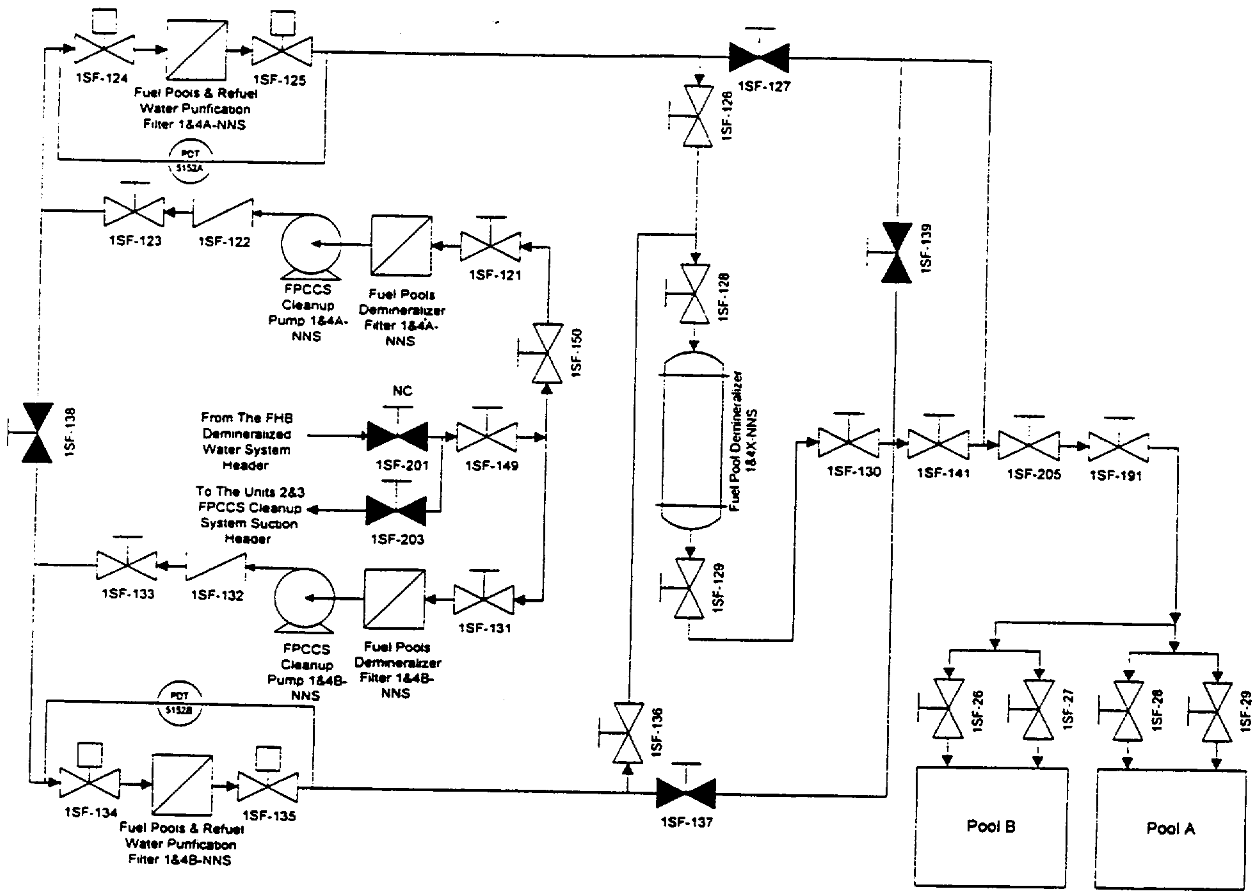


Figure A.4-4 Demineralized Water Makeup to Pools A and B with FPCCS Cleanup in Service

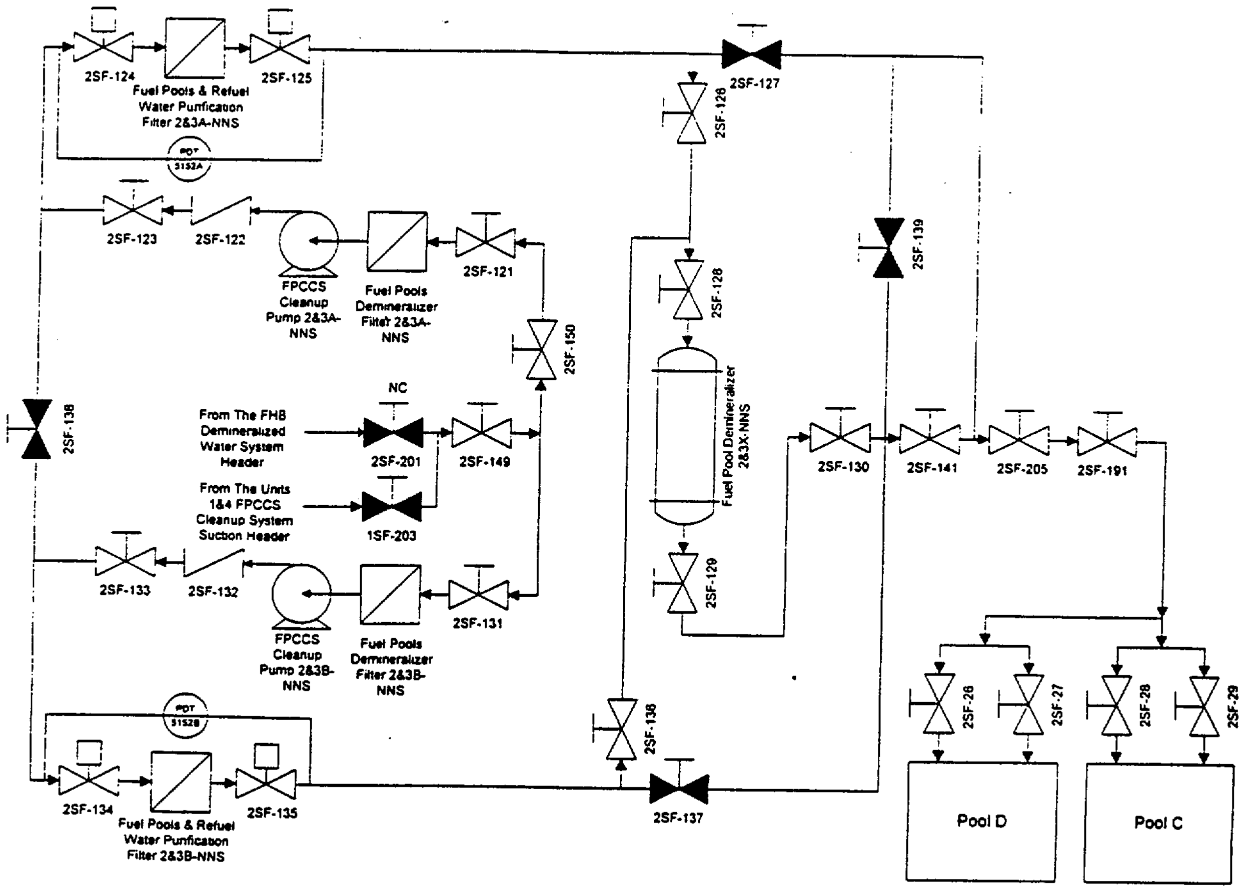


Figure A.4-5 Demineralized Water Makeup to Pools C and D with FPCCS Cleanup in Service

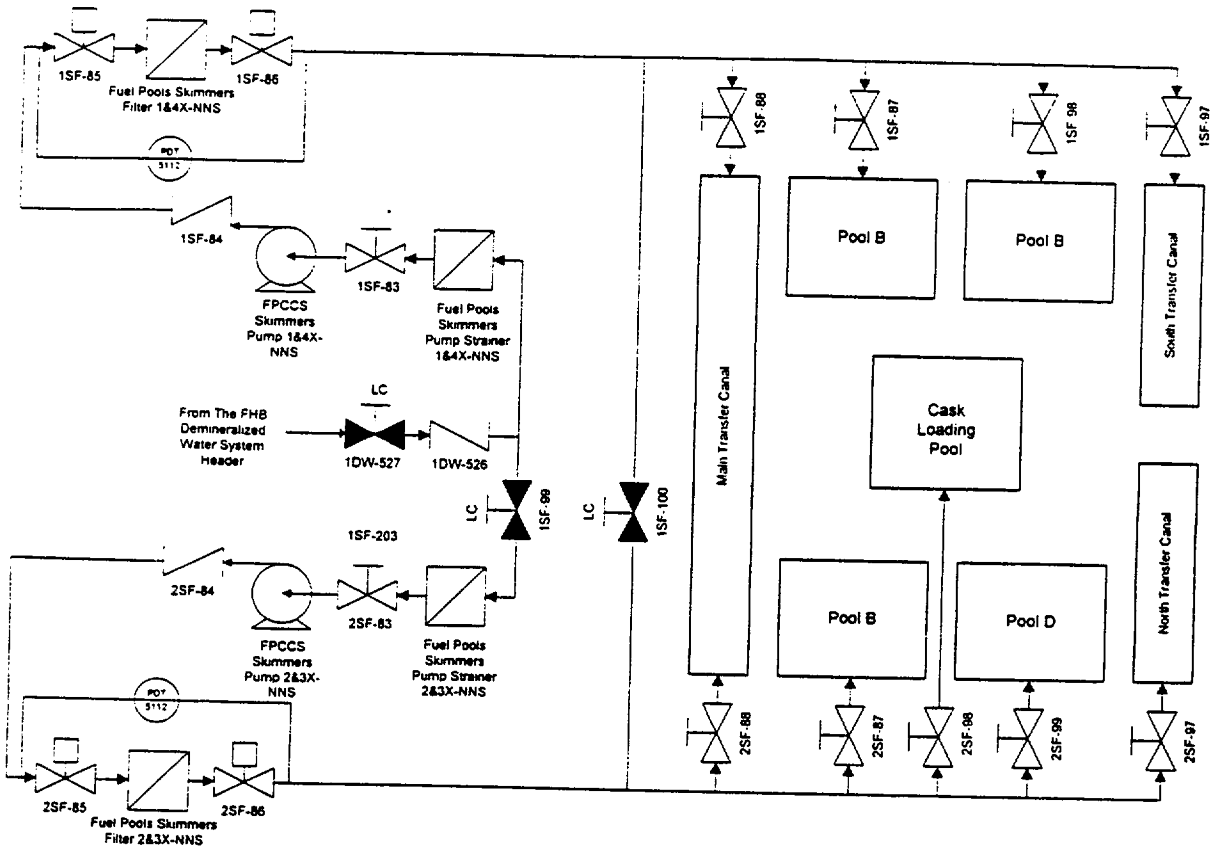


Figure A.4-6 Demineralized Water Makeup to Pools and Canals with FPCCS Skimmers in Service

A.5 SUPPLEMENTAL WATER MAKEUP TO FUEL POOLS

In the event of a loss of SFP water inventory, SFP low level alarms would be received in the Main Control Room at Auxiliary Equipment Panel Number 1. SHNPP annunciator panel procedure APP-ALB-023, *Auxiliary Equipment Panel No. 1*, directs the operators to initiate makeup to the SFPs per Plant Operating Manual Operating Procedure OP-116, *Fuel Pool Cooling and Cleanup*. Table A-1 summarizes the supplemental SFP makeup methods. These methods include both proceduralized and non-proceduralized methods. In the event that normal makeup from the demineralized water system through the FPCCS Cleanup / Purification Subsystem is not available, OP-116 gives the options provided in Table A-1.

Table A-1

SPENT FUEL POOL MAKEUP METHODS

Method		Procedure	Time	Access to Location Required	Pumps Required	Power	Water Source	Flow Rate (gpm)	Accessible Volume (gal)
Proceduralized Methods									
PA.	ESW (Alt. #5)[1]	OPP-116 (8.13)	30 min.[2] to 1 hr	FHB[3] 236' El. RAB 236' El.	ESW and ESW Booster	Div. I or II	Uniform Hazard Response System upper or lower reservoir	50 - 75 gpm	Large

(1) The alternate number references are those provided in the first interrogatory response to NRC issued September 26, 2000 regarding the ASLB order.

(2) Need to also have complement of people.

(3) Not required.

0C095C

Table A-1

SPENT FUEL POOL MAKEUP METHODS

Method		Procedure	Time	Access to Location Required	Pumps Required	Power	Water Source	Flow Rate (gpm)	Accessible Volume (gal)
PB*	Demin Water (Normal Makeup)	OPP-116 (8.4)	~ 30 min. (5 min. Excluding Transit Time)	FHB 216' El. North(4) or South(5) Valves 1SF 201 South(5) 2SF 201 North(4)	• Demin Pumps • Cleanup Pumps are part of procedure but not required(6)	Offsite Power(7) (AOVs not required)	Demin water tank	100 gpm with Demin pumps only (2" pipe)	500,000

* Normal Makeup Supply

(4) Makeup flow would be directed to the C & D Pools.

(5) Makeup flow would be directed to the A & B Pools.

(6) The normal operating range for the demin water system header pressure is 150 psig to 225 psig. Therefore, a minimum supplied head through 2SF-201 would conservatively be 100 psig (assuming a 50 # headloss through the piping) which would result in at least 100 gallons per minute. The status of the purification pump would have little or no impact on the delivery flow rate of demin water to the system. (Personal Communication Eric McCartney (CP&L) to E.T. Burns (ERIN), October 4, 2000)

(7) Emergency supply would require ad hoc alignment.

Table A-1

SPENT FUEL POOL MAKEUP METHODS

	Method	Procedure	Time	Access to Location Required	Pumps Required	Power	Water Source	Flow Rate (gpm)	Accessible Volume (gal)
PC	RWST (Alt #2)	OPP-116 (8.5)	30 min.	• FHB 216 ft. El. valve 1SF-193; and, • RAB 236 ft. El. Valve 1CT-23 • FHB 236 ft. El. <u>or</u> FHB 286 ft. El. for pump breaker	• N/A through suction path; or, • FPCCS Cleanup pumps through discharge path	N/A	RWST (Gravity Drain)	100 gpm	• 490,000 • May be unavailable because already discharged to containment
PD	RWMST (Alt #6)	OPP-116 (8.26)	30 min.	• RAB 236' • FHB 236'	Rx water M/U pumps <u>or</u> Gravity feed is feasible under certain conditions	Div. I & II	RWMST	75 - 100 gpm	80,000 (usually full)
PE	Demin to Fuel Pool Skimmer (Alt #3)	OPP-116 (8.6)	60 min. (Est.)	FHB 236' El. 1 valve	• Demin pumps • Skimmer pumps	Offsite Power	Demin water tank	100 gpm	500,000

Table A-1

SPENT FUEL POOL MAKEUP METHODS

Method		Procedure	Time	Access to Location Required	Pumps Required	Power	Water Source	Flow Rate (gpm)	Accessible Volume (gal)
PF	RWST to FPCC CLG pumps (Alt #4)	OPP-116 (8.12)	30 min.	FHB El. 236' FHP El. 216' RAB El. 236'	Gravity drain is adequate	• None for gravity drain • Div. I or II for pump operation	RWST	• 60 - 100 gpm by gravity • 5000 gpm with FPCC cooling pump operating	• 490,000 • May already be discharged to containment
PG	Demin Water to FPCC cleanup system (Alt #1)	OPP-116 (8.5)	30 min.	FHB El. 236' FHB El. 216' FHB El. 261' El. for pump breaker'	Cleanup pump	Offsite Power	Demin Water Tank	100 gpm with cleanup pumps running	500,000
PH	RWDT	OPP-116 (8.22)	More than 30 min.	FHB	Not Evaluated	Not Evaluated	RWDT during normal operation	Not estimated	Water not likely available during accident conditions
Non Proceduralized Methods									
N1	Fire Protection to hoses on 286' El. of FHB	None	30 min.	FHB 286' El.	Diesel Fire Pump or Electric Fire Pump	None	Upper Lake only (seismic guaranteed source)	~ 100 gpm per hose	Large

Table A-1

SPENT FUEL POOL MAKEUP METHODS

Method		Procedure	Time	Access to Location Required	Pumps Required	Power	Water Source	Flow Rate (gpm)	Accessible Volume (gal)
N2	Demin Water Quick Connect Options on 286' El.	None	30 min.	286' El. FHB	Demin Water	Offsite	Demin Water Tank	100 gpm (2" pipe)	500,000
N3	NSW	None[4]	> 60 min.	WPB	NSW	Offsite	Lake	> 100 gpm	Large

[4] 300 ft of hose would be required. This is currently not prestaged.

1. Emergency Service Water (ESW) System – The ESW system may be connected to dedicated FPCCS Cooling Subsystem emergency makeup connection vent valve 1SF-76 (located downstream of 1CT-23 at the 236 ft elevation of the RAB, column line E42 above the heat exchanger valve gallery) via approximately 50 feet of 1 inch rubber hose. This hose is stored in a gang box located in the stairwell opposite 1CT-23 (through door D605) at the 236 ft elevation of the RAB. The ESW valves are located in the overhead in the hallway just outside the hot machine shop (1SW-1239 for ESW train B) and in the overhead just inside the hot machine shop (1SW-269 for ESW train A) in the RAB at the 236 ft elevation, column line D43. The source of water is the Harris Lake, which provides a virtually unlimited source of water. The flow rate is approximately 50 to 75 gallons per minute. The operator can align this flow path within 30 minutes. Details of this lineup are contained in SHNPP Operating Procedure OP-116 Section 8.13. (Table A-1, Method PA)

2. RWST – Normally closed manual valves 1SF-193, located in the FHB at the 216 ft elevation (north) and 1CT-23, located in the RAB at the 236 ft elevation, column line E13 must be opened to align the FPCCS Cleanup / Purification Subsystem to the RWST. After aligning the valves, the operator turns on power supply breakers for the purification pumps and starts the pump from one of two locations, the 236-foot elevation FHB or the operating deck of the FHB. The source of this flow path is the RWST with a capacity of 490,000 gallons. The flow rate is 100 gallons per minute. The operator can align this flow path within 30 minutes. If the RWST is full, this flow path will result in gravity flow to the spent fuel pools, transfer canal, or cask loading pool without needing any pumps due the elevation difference between the RWST and the spent fuel pools. Details of this lineup are contained in SHNPP Operating Procedure OP-116 Section 8.5. (Table A-1, Method PC)

   The RWST is not filled during refuel operations with the cavity flooded; therefore, use of the RWST as a makeup water source to the SFP is precluded under those conditions. In addition, the RWST can be used for injection to containment during a severe accident, therefore it is likely not available for SFP makeup under the conditions postulated in the ASLB Order.

3. Primary Makeup Water System (PMWS) – Locked closed manual valve 7PM-V238-1 provides isolation between the FPCCS and the PMWS. This valve is located in the RAB on the 236 ft elevation. Opening this valve and aligning four manual valves in the FHB equipment room at the

236 ft elevation allows water from the 80,000 gallon Reactor Makeup Water Storage Tank (RMWST) to be used to fill the FHB pools and canals. The source of water is the RMWST with a capacity of 80,000 gallons. The flow rate is 75 to 100 gallons per minute. The operator can align this flow path within 30 minutes. Details of this lineup are contained in SHNPP Operating Procedure OP-116 Section 8.26. (Table A-1, Method PD)

4. Demineralized Water (DW) System – Normally locked closed manual valve 1DW-527, located in the FHB equipment room at the 236 ft elevation, may be opened when the FPCCS Skimmer is in service to slowly add DW to the pools and canals through their floating skimmers. The source of water is the demineralized water storage tank with a capacity of 500,000 gallons. The flow rate is approximately 100 gallons per minute. Details of this lineup are contained in SHNPP Operating Procedure OP-116 Section 8.6. (Table A-1, Method PE)

5. RWST to FPCC Cooling Pumps – To align the RWST to the suction of the FPCCS Cooling Subsystem pumps the operators must align eleven manual valves. This will deliver water to the South Transfer Canal, the Main Transfer Canal and the Cask Loading Pool. Eight of these valves are in the FHB equipment room at the 236 ft elevation, two valves are in the south end room of the FHB at the 216 ft elevation and 1CT-23 is located in the RAB at the 236 ft elevation, column line E13. If the RWST level is high, then the transfer canal or cask loading pool will fill due to gravity. The SFP cooling pump is then started from the Main Control Room. The source of water is the RWST with a capacity of 490,000 gallons. The flow rate is 5000 gallons per minute. The operator can align this flow path within 30 minutes. Details of this lineup are contained in SHNPP Operating Procedure OP-116 Section 8.12. (Table A-1, Method PF)

6. Demineralized Water System – To makeup water to SFPs "A" and / or "B," the operators must align four manual valves. (See OP 116 Section 8.5). Two are located in the FHB equipment room at the 236 ft elevation and two are located in the south end room at the FHB 216 ft elevation. To makeup water to SFPs "C" and / or "D," the operators must align two manual valves in the FHB equipment room at the 236 ft elevation and two additional manual valves located in the north end room at the FHB 216 ft elevation. Once the power supply is turned on, the operator turns on the purification pump at one of two locations, the operating deck of the FHB or the 236-foot elevation of the FHB. The source of water is the demineralized water storage tank with a capacity of 500,000 gallons.

The flow rate is 100 gallons per minute. The operator can initiate flow in approximately 30 minutes, excluding any transit time. Details of this lineup are contained in SHNPP Operating Procedure OP-116 Section 8.5. (Table A-1, Method PG)

7. RWDT – This method is considered viable during nominal operation for small quantities of makeup. It is not credited for larger volume during accidents. (Table A-1, Method PH)

There are several other potential sources of makeup to the SFPs that are not currently credited in SHNPP Operating Procedure OP-116. These non-procedural lineups may be attempted under the direction of the SHNPP Technical Support Center (TSC):

1. Fire System – The FHB is equipped with a fire header that runs along the east and west walls on the 286 ft elevation. There are three hose stations (each containing a 1.5" hose) along the west wall and four hose stations along the east wall on the 286 ft elevation operating floor connected to this header. Any or all of these hoses could be directed into the pools the canals to supply more than 100 gpm per hose. The fire protection system draws water from upper Harris Lake via a motor driven fire pump or a redundant diesel driven fire pump. (Table A-1, Method N1)

   It is noted that the Fire Protection System capability to provide SFP makeup may become more complicated under a seismic event. A seismic event may lead to the failure of the fire protection pumps (i.e., they are not seismic). However, the piping is seismic. The SHNPP method of supplying fire protection water is through the use of the ESW pumps, which are seismically qualified, through 2 manual cross connect valves located on 236' El. of RAB.

2. Demineralized Water (DM) System – There are 19 DM stations located along the east and south walls of the FHB operating deck at the 286 ft elevation. Each of these stations has a manual isolation valve and a standard quick disconnect fitting. Rubber hoses with matching fittings are readily available on the FHB operating deck at all times for routine work. Hoses could be quickly attached to any or all of these DM stations and directed into any of the pools and / or canals. (Table A-1, Method N2)

3. Normal Service Water (NSW) System – The NSW System extends into the Waste Processing Building (WPB) at the 261 ft elevation near the WPB stairwell that leads up to the south end of FHB 286 ft elevation. Approximately 300 feet of 1 inch rubber hose could be connected to any one of a number of 1 inch drain valves on the NSW lines in this area, run up the stairwell and directed into pool "A". (Table A-1, Method N3)

A.6 FUEL POOL INSTRUMENTATION

The critical levels in the SFPs are summarized in the following table:

Top of Pools/Canals	286.000 feet
Top of an installed gate	285.125 feet
HI Level Alarm in Main Control Room	284.900 feet
Normal water level	284.500 feet
LO Level Alarm in Main Control Room	284.000 feet
Technical Specification 3.9.11 Limit	283.790 feet
LO-LO Level Alarm in Main Control Room	282.000 feet
Top of BWR racks in Pools "B", "C" & "D"	261.250 feet
Top of PWR racks in Pools "B", "C" & "D"	260.480 feet
Top of PWR racks in Pool "A"	260.960 feet
Bottom of Main Transfer Canal	260.000 feet
Bottom of North / South Transfer Canals	251.000 feet
Bottom of fuel pools	246.000 feet
Bottom of Cask Loading Pool	240.000 feet

Monitoring capability of the SFPs at SHNPP can be summarized in the following table:

Monitoring Capability	Spent Fuel Pools			
	A	B	C[4]	D[4]
• Camera • Pool Level Indicator • Pool Level Alarm	None No Yes[2]	None No Yes[2]	None No Yes[2]	None No Yes[2]
• FPCCW Pump Flow (Lose Suction at -4 ft.)	No[1], [3]	No[1], [3]	No[1], [3]	No[1], [3]
• Temperature Alarm - Bistable Hi Level, - Lo Level - Lo-Lo Level	Control Room Indication	Control Room Indication	Control Room Indication	Control Room Indication
• Local Indications Level	Observation	Observation	Observation	Observation
• Radiation (.1 mr/hr - 10^3 mr/hr)	Local at 286' El. FHB	Local at 286' El. FHB	Local at 286' El. FHB	Local at 286' El. FHB

[1] Local flow and pressure drop indications in FHB are available

[2] 22 ft. above fuel

[3] Lose temperature and suction

[4] Equivalent instrumentation is projected to be available following activation of Pools C & D

REFERENCES

[A-1] Personal communication, Eric McCartney (CP&L) to E.T. Burns and T.A. Daniels (ERIN) on September 21, 2000.

[A-2] CP&L Drawing, Fuel Handling Building Bulkhead Gates & Details - Units 1 & 2, CAR-2168-G-125 Revision 4

[A-3] CP&L, SHNPP Plant Operating Manual, Volume 3, Part 6, Annunciator Panel Procedure, Auxiliary Equipment Panel No. 1, APP-ALB-023, Revision 19

[A-5] Carolina Power and Light Company (CP&L), Shearon Harris Nuclear Power Plant (SHNPP), Plant Operating Manual, Volume 6, Part 2, System Description, Fuel Pool Cooling and Cleanup System, SD-116, Revision 7

[A-6] CP&L, SHNPP Plant Operating Manual, Volume 3, Part 2, Operating Procedure, Fuel Pool Cooling and Cleanup, op-116, Revision 117

[A-7] CP&L, SHNPP Design Basis Document, Fuel Pool Cooling and Cleanup System, DBD-110, Revision 8

[A-8] CP&L, SHNPP Plant Operating Manual, Volume 3, Part 6, Annunciator Panel Procedure, Local Control Panel F-P7, APP-F-P7, Revision 4

[A-9] CP&L, SHNPP Plant Operating Manual, Volume 3, Part 6, Annunciator Panel Procedure, Local Control Panel F-P9, APP-F-P9, Revision 4

[A-10] CP&L, SHNPP ESR 95-00425, Drawing Mark-up, Simplified Flow Diagram - Fuel Pools Cooling System - Unit 1 (CPL-2165-S-0805), Page 6.1.1, Revision 0

[A-11] CP&L, SHNPP ESR 95-00425, Drawing Mark-up, Simplified Flow Diagram - Fuel Pools Cooling System - Unit 2 (CPL-2165-S-0807), Page 6.1.3, Revision 0

[A-12] CP&L, SHNPP ESR 95-00425, Drawing Mark-up, Simplified Flow Diagram - Fuel Pools Clean-Up Systems - Sheet 1 - Unit 1 (CPL-2165-S-0561), Page 6.1.5, Revision 0

[A-13] CP&L, SHNPP ESR 95-00425, Drawing mark-up, Simplified Flow Diagram - Fuel Pools Clean-Up Systems - Sheet 2 - Unit 1 (CPL-2165-S-0562), Page 6.1.6, Revision 0

REFERENCES (Cont'd)

[A-14] CP&L, SHNPP ESR 95-00425, Drawing Mark-up, Simplified Flow Diagram - Potable And Demineralized Water Systems - Unit 1 (CPL-2165-S-549S02), Page 6.1.14, Revision 0

[A-15] CP&L, SHNPP Drawing, Plot Plan, CAR-2165-G-002, Revision 20

[A-16] CP&L, SHNPP Drawing, General Arrangement - Reactor Auxiliary Building - Plan El. 190.00' & 216.00', CAR-2165-G-015, Revision 16

[A-17] CP&L, SHNPP Drawing, General Arrangement - Reactor Auxiliary Building - Plan El. 236.00', CAR-2165-G-016, Revision 20

[A-18] CP&L, SHNPP Drawing, General Arrangement - Reactor Auxiliary Building - Plan El. 261.00', CAR-2165-G-017, Revision 19

[A-19] CP&L, SHNPP Drawing, General Arrangement - Reactor Auxiliary Building - Plan El. 286.00', CAR-2165-G-018, Revision 20

[A-20] CP&L, SHNPP Drawing, General Arrangement - Reactor Auxiliary Building - Plan El. 305.00', CAR-2165-G-019, Revision 20

[A-21] CP&L, SHNPP Drawing, General Arrangement - Reactor Auxiliary Building - Section Sheet 1, CAR-2165-G-020, Revision 19

[A-22] CP&L, SHNPP Drawing, General Arrangement - Reactor Auxiliary Building - Sections Sheet 2, CAR-2165-G-022, Revision 21

[A-23] CP&L, SHNPP Drawing, General Arrangement - Fuel Handling Building - Plans - Sheet 1, CAR-2165-G-022, Revision 19

[A-24] CP&L, SHNPP Drawing, General Arrangement - Fuel Handling Building - Plans - Sheet 2, CAR-2165-G-023, Revision 10

[A-25] CP&L, SHNPP Drawing, General Arrangement - Fuel Handling Building - Sections - Sheet 1, CAR-2165-G-024, Revision 10

[A-26] CP&L, SHNPP Drawing, General Arrangement - Fuel Handling Building - Sections - Sheet 2, CAR-2165-G-025, Revision 15

REFERENCES (Cont'd)

[A-27] CP&L, SHNPP Drawing, General Arrangement - Fuel Handling Building - Sections - Sheet 3, CAR-2165-G-026, Revision 13

[A-28] Interrogatory update from CP&L. Also discussed in Steven Edwards Personal Communication to E.T. Burns (ERIN) on October 30, 2000.