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Loss of Spent Fuel Pool Cooling PRA: Model and Results

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ABSTRACT

This letter report documents models for quantifying the likelihood of loss of spent fuel pool cooling; models for identifying post-boiling scenarios that lead to core damage; qualitative and quantitative results generated for a selected plant that account for plant design and operational practices; a comparison of these results and those generated from earlier studies; and a review of available data on spent fuel pool accidents. The results of this study show that for a representative two-unit boiling water reactor, the annual probability of spent fuel pool boiling is 5×10^{-5} and the annual probability of flooding associated with loss of spent fuel pool cooling scenarios is 1×10^{-3}. Qualitative arguments are provided to show that the likelihood of core damage due to spent fuel pool boiling accidents is low for most U.S. commercial nuclear power plants. It is also shown that, depending on the design characteristics of a given plant, the likelihood of either: a) core damage due to spent fuel pool-associated flooding, or b) spent fuel damage due to pool dryout, may not be negligible.

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1. INTRODUCTION

1.1 Background

In 1975, the risk from spent fuel pools in nuclear power plants was analyzed using simple models and assessed to be very small (orders of magnitude lower in frequency) in comparison with the risk associated with core damage accident scenarios [1]. In the mid-1980's, changes in conditions (the onsite storage of fuel using high density storage racks) and new information concerning the possibility of cladding fires in drained spent fuel pools prompted a re-examination of spent fuel pool risk under Generic Issue 82. Based on value/impact and cost-benefit analyses, it was determined that no actions were required by the U.S. Nuclear Regulatory Commission (USNRC) [2]. In 1992, questions raised concerning a newly postulated accident scenario, in which boiling of the spent fuel pool leads to core damage, led to a new study, performed by the Pacific Northwest Laboratory (PNL) under the sponsorship of the USNRC, of the spent fuel pool risk at the Susquehanna Steam Electric Station (SSES) [3]. In a safety evaluation that referenced some of the results reported in Ref. 3, the USNRC staff concluded that "potential regulatory action based on safety concerns was not justified at the SSES". [4].

More recently, the USNRC Office for Analysis and Evaluation of Operational Data (AEOD) has initiated a broader investigation of safety issues associated with spent fuel pools. The AEOD study involves the collection and analysis of event data and plant-specific information (e.g., on configurations, procedures, and training). As part of this study, the Idaho National Engineering Laboratory (INEL) has been tasked with providing a risk perspective to the investigation[1]. The specific objectives of the INEL work are as follows:

- Assess the likelihood of the loss of spent fuel pool cooling for up to six different configurations.
- Determine if the implications of operating experience are consistent with available risk insights and critically evaluate substantive differences.
- Develop qualitative insights on risk associated with accident scenarios involving the loss of spent fuel pool cooling.

1.2 Objectives

The objectives of this letter report are to present:

a) the models used to quantify the likelihood of loss of spent fuel pool cooling;

[1] Note that "risk", as defined in Ref. 5, is a generic term treating both the likelihood and consequences of accident scenarios. The consequences (e.g., loss of spent fuel pool cooling) to be addressed in a particular study depend on the underlying purpose for the study.

b) the models used to delineate key post-loss of cooling scenarios and the qualitative impact of plant design and operational features on the likelihood of core damage associated with these scenarios;

c) qualitative and quantitative results generated for a selected plant design; and

d) a comparison of these results and those generated from earlier studies [2,3].

Note that an earlier letter report submitted to the NRC [6] documents a review of the models and results presented in Ref. 3: it also provides initial documentation of the models discussed in this report. This report provides more complete documentation of the models and the results generated from these models.

1.3 Report Outline

Section 2 of this report outlines the event tree/fault tree model used to estimate the likelihood of loss of spent fuel pool cooling and to delineate possible accident scenarios following loss of cooling. The purpose of the section is to provide a basis for interpreting the results in the following sections. The detailed model documentation is provided in Appendices A-D of the report.

Section 3 provides the estimated near-boiling frequency (NBF) results for a base case plant (the base case is based upon the Susquehanna plant). The section also discusses how sensitivity analyses treating variations in plant design and operational practices can be performed using the model documented in this report.

Section 4 presents a discussion on the factors that affect post-heatup accident progression. The discussion derives the conditional core damage probability (given a spent fuel pool accident) that must be exceeded for a spent fuel pool accident to be a significant contributor to core damage risk, and qualitatively addresses the effect of plant design features and operational practices which will contribute to this conditional core damage probability.

Finally, Section 5 presents risk assessment insights developed from a review of the AEOD database. These insights concern the degree to which the model documented in this report reflects actual operating experience. Insights concerning earlier spent fuel pool modeling efforts are also drawn.

Appendix A of the report documents the event tree/fault tree model developed in this study. The initiating events, event trees, success criteria, and fault trees are presented. Appendix B presents the approach used to quantify basic events not treated in the human reliability analysis (HRA) and the values obtained. Appendix C presents the HRA, and Appendix D lists the key modeling assumptions.

1.4 Summary of Results

The following conclusions regarding the likelihood of spent fuel pool boiling are based upon the calculations and analysis summarized in Section 3 of the report.

- For the base case plant studied (a 2-unit boiling water reactor), the annual probability of spent fuel pool (SFP) boiling events is 5×10^{-5}. The dominant contribution (56%) comes from scenarios initiated by a loss of offsite power (LOOP). The contribution from loss of SFP inventory events is also significant (31%).

- The instantaneous frequency of SFP boiling events during operation is 4×10^{-5}/yr. The instantaneous frequency during refueling is 1×10^{-4}/yr. The risk profile during operation is dominated by LOOP (66%); loss of inventory also contributes (28%). During refueling, the largest contribution comes from loss of inventory (45%). LOOP and loss of coolant accidents (LOCAs) also provide large contributions (25% and 22%, respectively).

- Major contributions to the likelihood of SFP boiling come from initiators involving: a) loss of inventory, and b) non-pipe break LOCAs during refueling. These initiators were not addressed in Ref. 3.

- The annual probability of flooding events associated with the SFP is 1×10^{-3}. The annual probability of flooding following a large seal failure is around 3×10^{-4}.

- The annual probability of SFP events involving a large loss of inventory and boiling (but not necessarily boil-off) of the remaining inventory is 6×10^{-6}. For single unit plants, credit cannot be taken for the operators or makeup systems of the second unit and the probability may be a factor of 7 higher.

Regarding the likelihood of core damage involving spent fuel pool initiators, note first that, assuming a base case (non-SFP associated) core damage frequency (CDF) of 5×10^{-6}/yr, the conditional probability of core damage given spent fuel pool boiling needs to be greater than 10^{-3} in order for the boiling scenario to be a visible (> 1%) contributor to core damage risk.[2] Similarly, the conditional probability of core damage due to flooding given an SFP event involving severe flooding needs to be greater than 2×10^{-4} in order for the flooding scenario to be a visible contributor to core damage risk. The following conclusions are based upon the discussion presented in Section 4 of this report.

- For most, if not all, nuclear power plants, the conditional probability of core damage given spent fuel pool boiling is likely to be smaller than 10^{-3}. A small probability value is expected due to: a) the spatial separation of emergency core cooling system (ECCS) equipment (which implies that steam/heat must be carried to several rooms to create a

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[2] The CDF value used is roughly correct for the Grand Gulf boiling water reactor plant. (The CDF from the Susquehanna individual plant examination — IPE — is not used in this comparison, due to significant differences in methodology and quantification between the IPE and this study.) For plants with a higher core damage frequency, spent fuel pool accidents must have an even higher conditional core damage probability to be risk significant.

serious challenge to core cooling), b) the spatial separation of the spent fuel pool area from the ECCS equipment areas, c) the robustness of most nuclear power plant components with respect to the temperatures associated with spent fuel pool boiling, and d) the length of time available to the operators to mitigate SFP boiling (e.g., by diverting the resulting steam). This conclusion may not be valid for plants with potentially sensitive equipment (e.g., solid-state protection cabinets).

- For plants with spent fuel pools housed in the same building (and above) the ECCS equipment areas, as is the situation in the base case plant, the conditional probability of core damage given SFP-associated flooding may be high enough to warrant additional investigation. In this case: a) the spatial separation may not be as effective as for the heat/steam transport scenario (the elevation boundaries may not be watertight), b) ECCS equipment are generally assumed to be vulnerable to immersion, c) the flooding times can be relatively rapid, especially in the case of large seal failures during refueling.

The above conclusions are the result of a limited scope study. Key modeling simplifications are listed in Section 2; these include the simplified treatment of seal failures (the model treats only seal failures that lead to a loss of spent fuel pool inventory, but does not distinguish between different seals), the use of a simplified human reliability analysis method and the lack of a recovery analysis for dominant sequences. Key simplifications during quantification are discussed in Section 3. These include the use of generic estimates for initiating event frequencies and basic event probabilities, the use of point-estimate calculations throughout, and the lack of any sensitivity analyses. It should also be noted that a number of initiating event frequency estimates are based on the events included in a June 13, 1996 version of the SFP database being developed by AEOD; events added to the database after this date are not reflected in the analysis.

2. MODEL SUMMARY

This section summarizes the loss of spent fuel pool cooling (SFPC) model developed in this study. The description of the modeling approach and key assumptions is intended to provide a basis for interpreting the results in Sections 3 and 4. The detailed model documentation is provided in Appendices A-D of the report. Note that the model is based largely on the Susquehanna plant, a two-unit boiling water reactor with two spent fuel pools, a SFPC system powered off of a non-safety bus and cooled by non-safety service water, residual heat removal (RHR) assist cooling as backup to the SFPC system, and a safety-related emergency service water system to provide makeup when normal pool makeup is unavailable or inadequate. The event tree models are generic enough to allow analyses of a variety of other plant configurations. (For example, they can treat plants where the fuel pools are not cross-tied, as they are at Susquehanna.) However, the success criteria and the fault trees developed are intended to represent Susquehanna.

2.1 Modeling Approach

The premise underlying the analysis is that events involving the loss of spent fuel pool cooling can, under some circumstances, lead to boiling of the pool. Furthermore, in a subset of these events, the consequences of these scenarios (e.g., steam release, water flooding) can affect emergency core cooling system (ECCS) equipment, and eventually lead to core damage. Even if core damage does not result, adverse consequences associated with the spent fuel are also potentially of interest [2].

This section summarizes the event trees and fault trees developed in this study. The starting point for model development was the model presented in Ref. 3. As discussed in Ref. 6, improvements have been made to improve the chronological representation of scenarios, to treat demands on operators and equipment due to core challenges during some scenarios (e.g., loss of offsite power), to increase the detail of the post-boiling analysis, to correct some errors in quantification, and to reflect lessons learned from AEOD's review of operating plant experience with spent fuel pool events.

It is useful to note that, with respect to scenarios that can affect the core, the modeling approach employed is analogous to that used in the analysis of the so-called external events (e.g., internal fires). This approach divides the accident scenario analysis into three portions: a) the quantitative hazard analysis (e.g., the frequency of fires of a given size in a given location), b) the equipment fragility analysis (e.g., the conditional probability of damage to a given set of equipment, given the fire), and c) the plant response analysis (e.g., the conditional probability of core damage, given the loss of the given set of equipment). In simplified mathematical form,

$$\mathrm{CDF} = \sum_{j} \lambda_j \phi_{ed|j} \phi_{cd|j,ed} \tag{2.1}$$

where λ_j is the frequency of hazard scenario j, $\phi_{ed|j}$ is the conditional probability of equipment damage, given hazard scenario j, and $\phi_{cd|j,ed}$ is the conditional probability of core damage, given equipment damage and hazard scenario j.

In this study, it can be seen that λ_j corresponds to the near boiling frequency (NBF) associated with a given scenario; this is assessed quantitatively. The term ϕ_{edlj}, on the other hand, is not quantified. (This term is highly dependent on the particular geometry, equipment layout, and ventilation conditions for the plant being analyzed. Furthermore, the analysis of heat and mass transport needed to support quantification is beyond the scope of this limited study.) Instead, qualitative issues affecting the likelihood of equipment damage are discussed. Note that the term $\phi_{cdlj,ed}$ can be quantified using the internal events model for the plant in question, as long as the likelihood of operator errors is not drastically affected by the spent fuel pool boiling event.

Regarding the particular analysis approach employed, the SAPHIRE [7] software package is used to implement a fault tree linking approach. The SAPHIRE database produced is not documented in this report.

2.2 Initiating Events and Cases

The initiating event categories and specific initiating events (acronyms in parentheses) treated in this study are as follows.

- Loss of Spent Fuel Pool Cooling System (LSFP1, LSFP2, LSFP3)
- Loss of Offsite Power (LP1, LP2, LP3)
- Loss of Spent Fuel Pool Inventory (LINVC, LINCS, LINVR, LINRS)
- Loss of Primary Coolant (PLOCA, PLOCR)
- Seismic Event (EQE)

The LSFP1 initiating event covers a loss of the SFPC system for Case 1 (both units operating, as defined below); LSFP2 and LSFP3 cover Cases 2 and 3 (also defined below). These events treat system loss due to hardware failures and human errors. They also include system loss due to loss of cooling to the SFPC heat exchangers and due to internal flooding and fires.

Note that in principle, the loss of heat exchanger cooling, internal flooding, and internal fires should be treated as separate initiating events, since these causes for loss of SFPC might also affect other parts of the plant. (At Susquehanna, heat exchanger cooling is normally provided by a non-safety service water system.) These events are intentionally grouped with direct losses of SFPC because of the limited scope of this study, and because the results of Ref. 3 indicate that, at least in the case of Susquehanna, the contributions to risk from the loss of service water and internal flooding initiators are relatively small.

The LP1, LP2, and LP3 initiating events treat loss of offsite power (LOOP) events for Cases 1, 2, and 3 (defined below). Note that unlike the model in Ref. 3, the analysis of station blackout (SBO) events is integrated in the LOOP model.

The LINVC, LINCS, LINVR, and LINRS events include losses of inventory from leaks/breaks from the piping (including misalignments) or gates/seals. (LINVC and LINCS treat large and small leaks, respectively, when all units are operating — Case 1; LINVR and LINRS treat large and small leaks, respectively, where one unit is refueling — Cases 2 and 3.) Only leaks/breaks for which the outgoing flow rate exceeds the normal makeup flow rate are considered. Losses of inventory due to structural failure of the spent fuel pool boundary (e.g., due to missiles, heavy load drops, thermal stresses) are not treated. This category of events may need to be re-examined, depending on the quantitative results of the models documented in this study.

The PLOCA and PLOCR events respectively cover primary system pipe break loss of coolant accidents (LOCAs) in an operating unit and non-pipe break LOCAs (e.g., maintenance-induced LOCAs) in a unit undergoing refueling. (PLOCA treats situations when all units are operating — Case 1; PLOCR treats situations where one unit is refueling — Cases 2 and 3.) These events are of potential concern because, depending upon plant design, a LOCA can lead to a trip of the SFPC system, and because it creates a demand for the RHR system, which serves as an alternate cooling system for the spent fuel pool. In the case of LOCAs during refueling, the event also provides a potential means for quickly draining the spent fuel pool down to the bottom of the transfer gate.

The EQE event covers seismically-induced losses of offsite power, SFPC piping integrity, and spent fuel pool boundary integrity. Two classes of earthquakes are treated: those with peak ground acceleration (PGA) between 0.2g and 0.6g, and those with PGA above 0.6g.

The response of the plant to an initiating event depends on the operational states of the reactor units. The following different plant configurations ("cases") are analyzed in this study:

- Case 1 - Both units operating.
- Case 2 - Unit 2 operating, Unit 1 refueling (1/3 core offload).
- Case 3 - Unit 2 operating, Unit 1 refueling (full core offload).

It should be recognized that although some of the event trees presented in the following section are used for a number of cases (e.g., the PLOCR event tree is common to Cases 2 and 3), a separate analysis may be performed for each initiating event/case combination. This allows for changes in top event success criteria and failure probabilities to represent differences between situations (e.g., reduced time to boil, increased presence of plant personnel on the refueling floor).

2.3 Event Trees

Event trees are used to represent the sequence of events following an initiating event. In general, the structure and level of detail of the NBF trees developed in this study (see Appendix A) are similar to those of the event trees presented in Ref. 3. The three key differences are as follows.

1) Those trees that model initiators with potential direct impacts on the core (LOOP, seismic, PLOCA) include a top event (UNREC) indicating if recovery is uncomplicated. Assuming

that operators are generally more concerned with the core than the spent fuel pool, a complicated recovery can inhibit the operators from devoting sufficient resources to deal with the spent fuel pool in a timely fashion. Appendix B provides the operational definition for complicated scenarios used in this analysis.

2) The trees explicitly allow for the possibility that operators will not respond to the initiating event until pool boiling occurs. This delay can be due to lack of awareness (e.g., failed instrumentation) or distraction (e.g., due to a complicated recovery). Note that the AEOD database includes a number of events in which operator response was delayed for many hours, although none were delayed to such an extent that pool boiling occurred.

3) The LOOP, seismic, and primary LOCA trees represent the possibility of "direct core damage" (i.e., core damage not due to the consequences of a spent fuel pool scenario) for complicated scenarios. The purpose of this treatment is to ensure that any final core damage frequency estimates developed from the results of this study do not double count risk contributing scenarios. (Thus, for example, station blackout scenarios which lead directly to core damage are not included in the NBF estimation, even though they could lead to pool boiling.)

As an example, part of the NBF tree for the LP1 initiator is shown in Figure 2.1. The event progression model underlying Figure 2.1 is presented in Appendix A. In addition to top events representing the success/failure of systems and key operator actions, the figure shows a "flag" (non-probabilistic) top event (FVPWR) used to model the plant design, a top event to represent the current plant status (GSTAT), a top event used to model the fraction of times a given scenario is not complicated (UNREC), and a top event used to model the fraction of times a complicated scenario does not lead directly to core damage (NCD). The top events are defined in success terms; per the usual convention, a "Yes" answer to a given top event question selects the upper path at the corresponding event tree branching point.

The post-heatup event trees (PHETs) are presented in Appendix A. These trees treat the progression of selected accident scenarios past pool heatup; one or more separate trees are developed for each non-successful endstate of the NBF trees. (Multiple trees are required for endstates where steaming and flooding effects are of potential concern.) They address the following issues: the spatial isolation of the spent fuel pool from other safety equipment, the vulnerability of exposed safety equipment to the hazards associated with the scenario (i.e., heat and humidity from pool boiling, water from losses of pool inventory), the ability of operators to divert steam/water away from the safety equipment, and the recoverability of safety equipment affected by the steam/water.

As an example, the FPIS1 event tree, whose entry condition involves the loss of spent fuel pool cooling (from the SFPC system, the RHR system, or any other alternate cooling system) and subsequent pool boiling, is shown in Figure 2.2. As in the LP1 tree, there is a flag event (FSPIS) modeling the plant design (in this case, the degree of isolation of key ECCS equipment with regard to heat and steam from the spent fuel pool). There is also a phenomenological top event (SSNV) which models the vulnerability of ECCS equipment to the heat and steam hazard, and two top events modeling the operator and system response.

The CDF event tree top events are not intended for fault tree analysis. Rather, they raise key questions whose answers can be used to identify entry states into an internal events tree. (Again, this is analogous to the approach used in the analysis of such external events as fires and floods. For example, a fire in a given location may lead to damage of different sets of components, depending on the result of the competition between growth and suppression processes. A major part of the fire analysis is to define the likelihood of damage of different sets of components; this information is then fed into a conventional event tree model.)

2.4 Success Criteria

The hardware success criteria developed for each event tree are presented in Appendix A. The success criteria for operator action top events are implicitly defined; the human reliability analysis (HRA) is described in Appendix C.

An example set of success criteria is shown in Table 2.1. These criteria, which apply to the LOOP scenario, Case 1, are based on the analysis of Ref. 3, and are appropriate to the Susquehanna plant. (In general, alternative success criteria will need to be developed when analyzing other plants.) It can be seen that the success criteria depend on the plant's electric power state during the scenario. While not shown (because the relevant portions of the event tree are not shown in Figure 2.1), the success criteria also depend on the status of the gates separating the two spent fuel pools.

2.5 Fault Trees

Fault trees have been developed to describe how each of the top events in the model can occur. (A number of these are trivial — they have only a single basic event.) The complete set of fault trees is provided in Appendix A.

For the non-trivial fault trees, e.g., for treating the unavailability of the SFPC system and the RHR system(s), the models supporting Ref. 3 have been used as a starting point. These trees have been modified using a modeling approach similar in spirit to that used in developing Accident Sequence Precursor (ASP) models (see for example Ref. 8). Using this approach, components on a single pipe segment are generally grouped into super-components. In some cases, entire trains of equipment are treated with a single super-component. Also, a number of low probability failure modes (e.g., normally closed manual valves transferring open during the scenario) are omitted. This simplified approach is judged to be adequate for treating spent fuel pool scenarios whose risk, as shown in Ref. 3, tends to be dominated by human error contributions.

An example fault tree for top event R1 (modeling the failure of the Unit 1 RHR system to start and run during a loss of spent fuel pool cooling system scenario, Case 1) is shown in Figure 2.3. (The simplified system piping and instrumentation diagram showing how the fault tree super-components are developed is shown in Figure 2.4.) It can be seen that the tree includes common cause failure and human error basic events. (The scenario-dependence of human error is treated during the accident sequence analysis; rule sets specifying which human error probability is used under which conditions are developed by the analyst and used by SAPHIRE during quantification.) Note that the tree also includes a number of basic events modeling the closing of

normally open manual valves. While not generally significant contributors to system unavailability, these failures are typically included in the ASP models.

From the accident sequence perspective, it is important to observe that only a single train of RHR is modeled. This is due to the assumption that one train of RHR is always reserved for standby core cooling.

2.6 Human Reliability Analysis

In keeping with the simple modeling approach used in other parts of the analysis, a simple human reliability analysis (HRA) technique is employed. This technique, documented in Ref. 9, is a worksheet-based approach developed for the ASP program. A sample worksheet for a single action is shown in Figure 2.5. The analyst evaluates the following performance shaping factors (PSFs) relevant to a given action and modifies base human error probabilities (HEPs) based on the evaluation.

- Complexity, stress, and workload
- Experience/training
- Procedures
- Ergonomics
- Fitness for duty
- Crew dynamics

The first four PSFs are of special interest to this study, due to the nature of the spent fuel pool accident scenarios hypothesized. For example, some of the modeled actions (e.g., placing RHR in a spent fuel pool assist cooling mode) can be fairly complex and time consuming; variations in scenario timing, e.g., due to different decay heat loads, can affect the time available (which affects workload); procedures may not be well developed for some spent fuel pool scenarios because they have not received as much attention as direct core damage scenarios; and some of the needed accident mitigation equipment may not be accessible during the scenario (e.g., elevated radiation levels near the pool during a severe draining event). (The last problem can be considered, in a broad sense, as an ergonomics issue. Another important ergonomics issue concerns the human-machine interface, as this affects how operators are informed of spent fuel pool conditions and how they manipulate components in response to their indications.)

The likelihood of failure of subsequent actions is treated using a second worksheet (see Figure 2.6). This worksheet addresses issues that could increase the dependency between actions. This study treats multiple unit actions (e.g., failure of operators at Unit 2 to restore spent fuel pool makeup using Unit 2 systems, given that operators at Unit 1 have failed using the Unit 1 systems) using the worksheet. (In general, the result is that there is a moderate level of dependency between actions.)

The base HEPs and the modification factors used in this procedure are derived from the widely used Technique for Human Error Rate Prediction (THERP) [10] methodology. Thus, the approach does not represent a fundamentally different approach to dealing with human errors;

rather, it is a consistent, psychology- and human factors-based compilation which allows relatively quick (if sometimes conservative) estimates of HEPs under a wide variety of conditions.

2.7 Basic Event Quantification

As indicated earlier, the fault trees used in this study are super-component based. The unavailability of a given super-component is approximated as the sum of the unavailabilities of the components contained in the super-component definition. The base component unavailabilities, in turn, are the same generic values used in the ASP models [11,12]. The basic events and associated unavailabilities used in this study (including a breakdown into components where relevant) are listed in Appendix B.

In some cases, the basic event values (e.g., for the relative frequency of SFPC system leaks versus SFP boundary leaks) are derived. The estimation process used for each of these values is presented in Appendix B.

2.8 Key Modeling Simplifications and Limitations

Because of the limited scope of this study, a simplified approach has been used in the modeling. Attempts have been made to ensure, where appropriate, that the simplifications have been applied uniformly across the initiating events analyzed, in order to avoid distortion of the computed risk profile. However, some distortion is inevitable.

Some of the key modeling simplifications are as follows. (Caveats due to simplifications in the quantification process are discussed in Section 3.)

- The loss of inventory models distinguish between losses of inventory from the spent fuel pool cooling system and those from the spent fuel pool (via seal failures), but do not treat the precise location of the leak. (For example, if a loss of inventory due to seal failure occurs, the model does not address which seal has failed.)

- Recovery analysis has not been performed for any of the dominant sequences or cutsets. (Such an analysis treats operator actions in restoring unavailable equipment.)

- A simplified HRA method (described in Section 2.6 above) has been used. This method does not require a formal task analysis, and is not sensitive to the detailed characteristics of available operating procedures.

- The same HRA models have been employed for the two refueling cases (Cases 2 and 3). The time windows used are appropriate for Case 3 (full core discharge).

- The PLOCR model, which treats a LOCA during refueling, treats all LOCAs as being relatively large (e.g., the equivalent diameter of an RHR pipe).

- It has been assumed that when plant recovery from a LOCA, LOOP, or earthquake is complicated, the operators will not deal with a loss of spent fuel pool cooling until near

boiling conditions develop. (A "complicated recovery" is defined as a plant recovery given one or more of the following conditions: offsite power is unavailable and one or more emergency diesel generators are unavailable; one or more relief valves is failed open or closed; high pressure coolant injection is unavailable; RHR is unavailable; an earthquake with peak ground acceleration greater than 0.2g has occurred. Details on the modeling of complicated recovery are provided in Appendix B.)

- In sequences where the operator response to a loss of spent fuel pool cooling is greatly delayed, it has been assumed that RHR assist cooling will not be employed because of the length of time required to establish this mode of cooling. This assumption is based on an estimate of 8 hours provided for Susquehanna; for other plants, the time required may be sufficiently less that RHR assist cooling could become a viable option.

It can be seen that some of these simplifying assumptions tend to make the model predictions more conservative. The degree of conservatism cannot be determined without more detailed study or at least some sensitivity studies. Both of these options were not pursued due to the limited scope of the project.

Table 2.1 - Success Criteria for LP1 (Case 1, Pools Cross-Connected)

DGs Available	Offsite Power	SR1O2	LS1O2	ALT-C
All	Early recovery	2 of 6 SFPC pumps or 1 train RHR in any unit	2 of 6 SFPC pumps	Any available alternate cooling system
All	Late recovery	1 train RHR in any unit	2 of 6 SFPC pumps	Any available alternate cooling system
All	None	1 train RHR in any unit	Not modeled	Any available alternate cooling system
Some	Early recovery	2 of 6 SFPC pumps or 1 train RHR in any unit	2 of 6 SFPC pumps	Any available alternate cooling system
Some	Late recovery	Not modeled	2 of 6 SFPC pumps	Any available alternate cooling system
Some	None	Not modeled	Not modeled	Any available alternate cooling system
None	Early recovery	Not modeled	2 of 6 SFPC pumps	Any available alternate cooling system

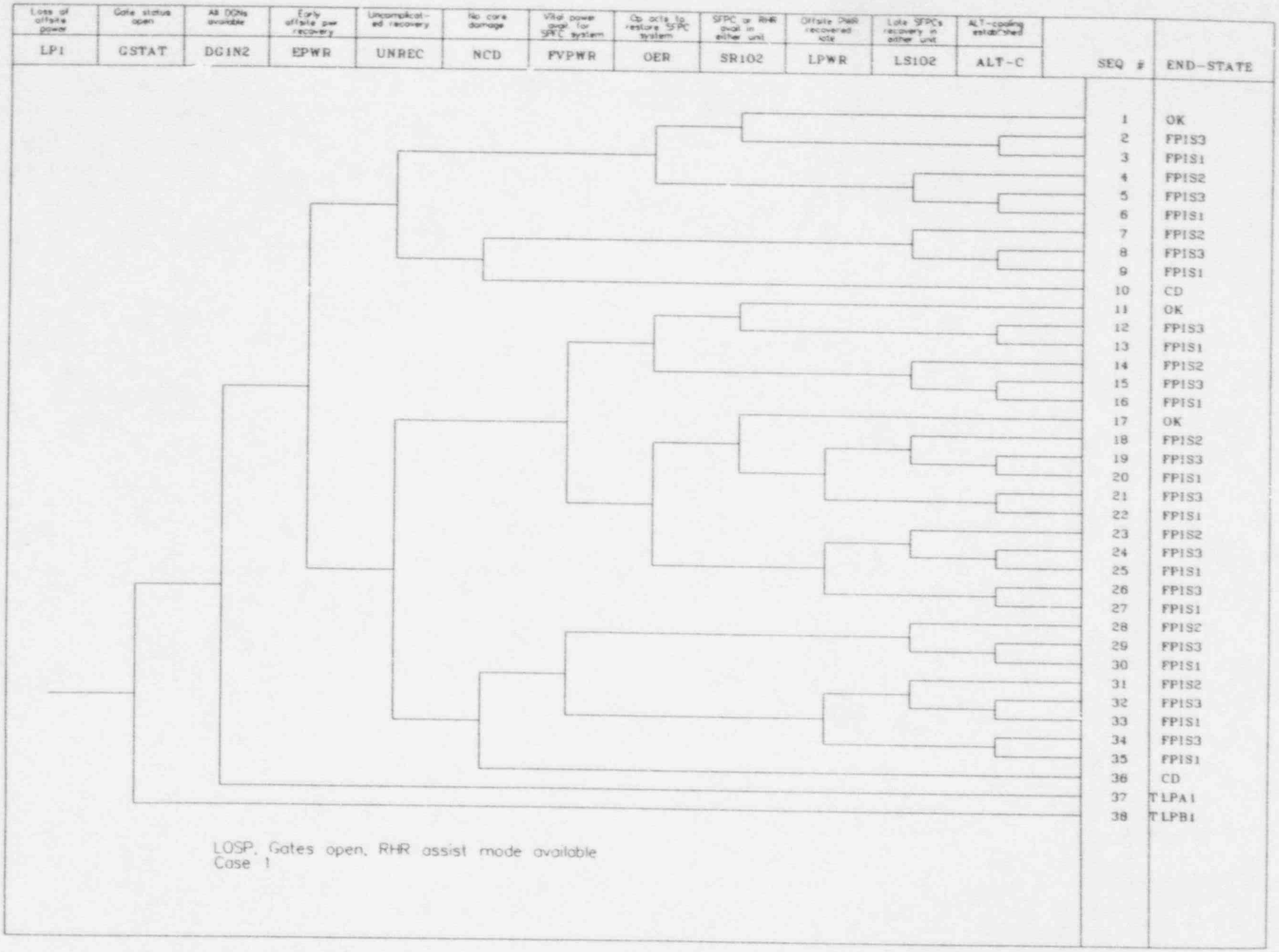

Figure 2.1 - LOOP SFP Heatup Event Tree (Partial)

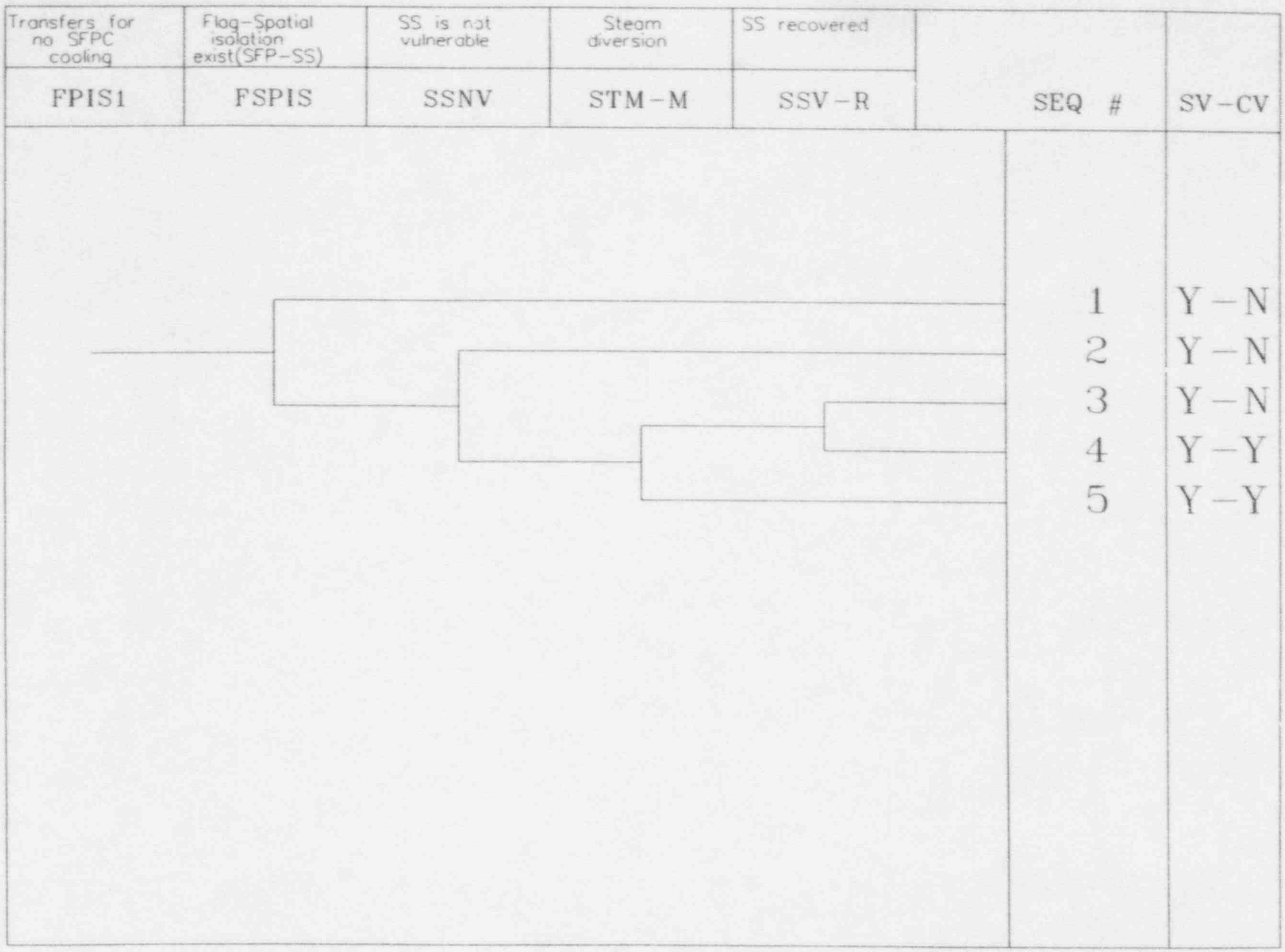

Figure 2.2 - FPIS1 Post -Heatup Event Tree

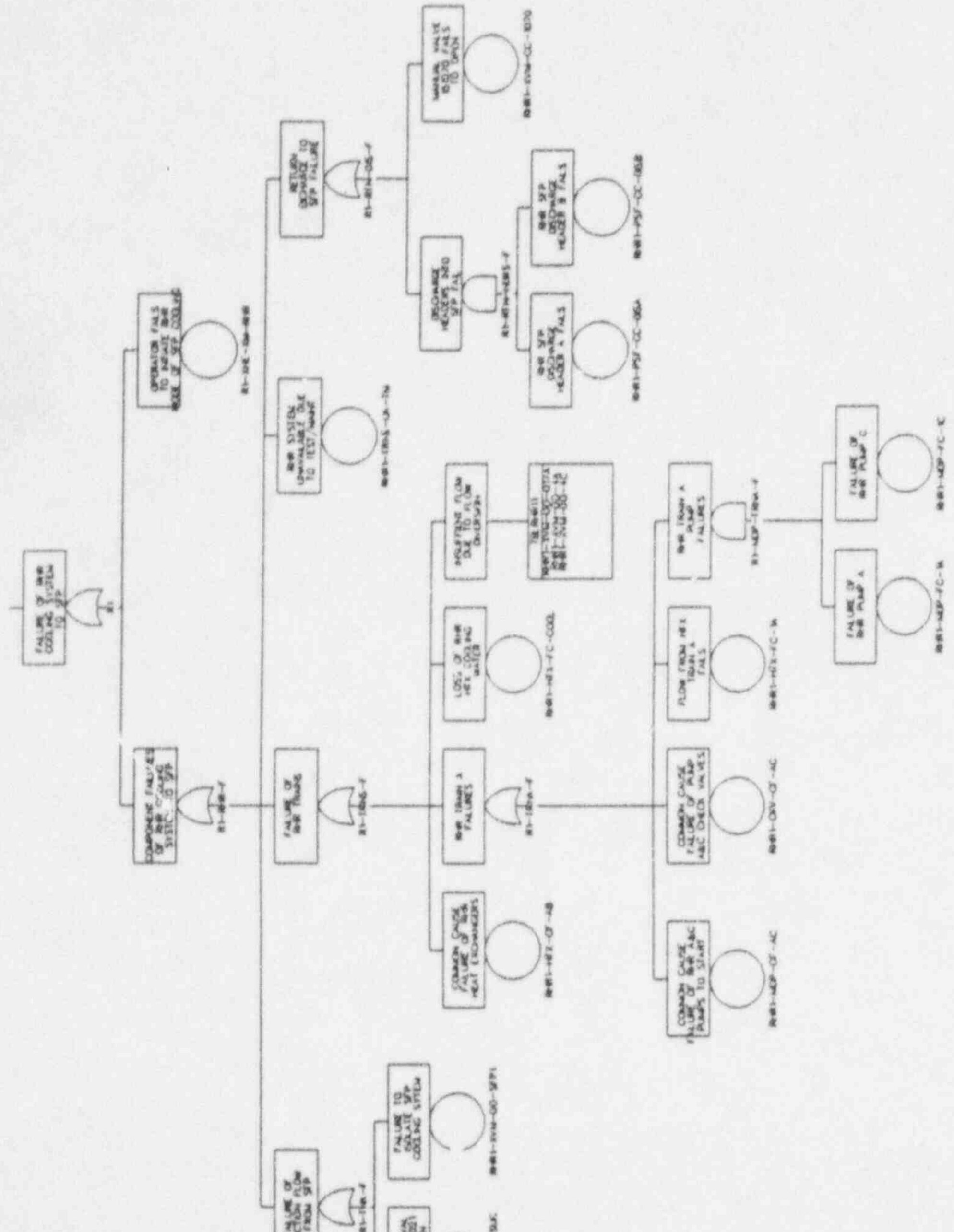

Figure 2.3 - R1 (RHR) Fault Tree

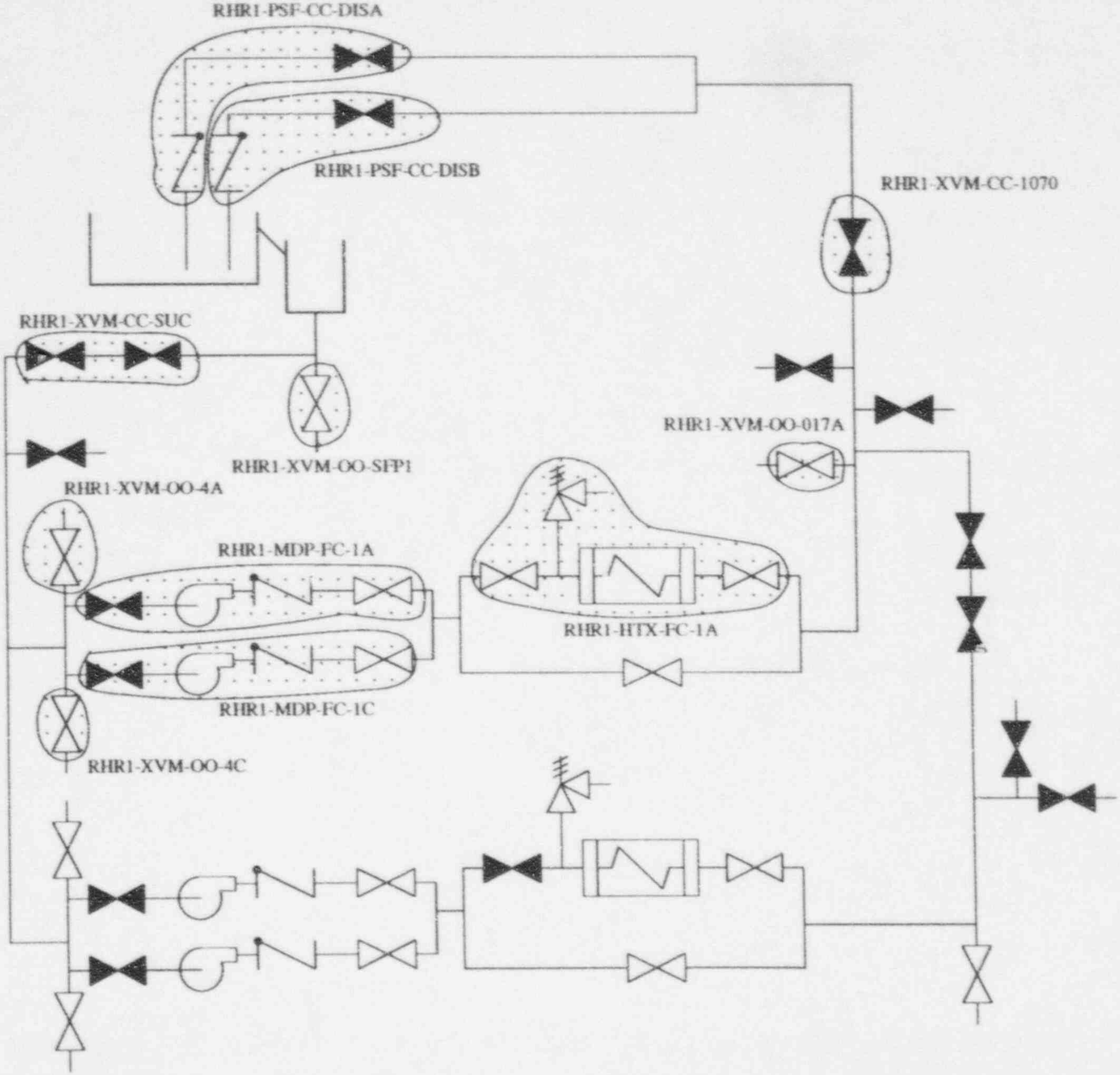

Figure 2.4 - Simplified Piping Diagram, RHR System

Plant:________ **Scenario:** _______ **Sequence Number:**______ **Cutset:**_________ ______

Task Error Description:____

1. Complexity, stress, and workload			Processing	Response
	high threat & stress	inadequate time	∞	∞
		adequate time	5	5
		expansive time	2	2
	low threat & stress	inadequate time	∞	∞
		adequate time	1	1
		expansive time	1	1

2. Experience/training			Processing	Response
	low experience	poor training	10	10
		good training	1	1
	high experience	poor training	5	5
		good training	0.5	0.5

3. Procedures			Processing	Response
	procedures absent		10	10
	procedures present	poor procedures	5	5
		good procedures	1	1

4. Ergonomics			Processing	Response
	old plant	poor ergonomics	5	5
		good ergonomics	1	1
	retrofit plant	poor ergonomics	3	3
		good ergonomics	0.7	0.7
	new plant	poor ergonomics	2	2
		good ergonomics	0.4	0.4

5. Fitness for duty		Processing	Response
	unfit	25	25
	fit	1	1

6. Crew dynamics		Processing	Response
	poor crew dynamics	10	10
	good crew dynamics	1	1

Task Portion		Complexity, stress, and workload	Experience/ training	Procedures	Ergonomics	Fitness for duty	Crew dynamics		
Processing:	10 E-2	x ______	x ______	x ______	x ______	x ______	x ______	=	______ Processing Failure Probability
Response:	10 E-3	x ______	x ______	x ______	x ______	x ______	x ______	=	+ ______ Response Failure Probability
									________ Task Failure Probability Without Formal Dependence

Figure 2.5 - ASP HRA Worksheet (Sheet 1 of 2)

DEPENDENCY CONDITION TABLE

Condition Number	Crew (same or different)	System (same or different)	Location (same or different)	Time (close in time or not close in time	Cues (additional or not additional)	Dependency	Number of Human Action Failures
1	s	s	s	c	–	complete	if this
2	s	s	s	nc	na	high	
3	s	s	s	nc	a	moderate	error is the
4	s	s	d	c	–	high	
5	s	s	d	nc	na	moderate	third error
6	s	s	d	nc	a	low	
7	s	d	s	c	–	moderate	in the
8	s	d	s	nc	na	low	
9	s	d	s	nc	a	low	sequence
10	s	d	d	c	–	moderate	
11	s	d	d	nc	na	low	then the
12	s	d	d	nc	a	low	
13	d	s	s	c	–	moderate	dependency
14	d	s	s	nc	na	low	
15	d	s	s	nc	a	zero	is moderate,
16	d	s	d	c	–	zero	
17	d	s	d	nc	na	zero	if it is the
18	d	s	d	nc	a	zero	
19	d	d	s	c	–	low	fourth error
20	d	d	s	nc	na	zero	
21	d	d	s	nc	a	zero	dependency
22	d	d	d	c	–	zero	
23	d	d	d	nc	na	zero	is high
24	d	d	d	nc	a	zero	

Using N=Task Failure Probability Without Formal Dependence (calculated on previous page):

For Complete Dependence the probability of failure is 1.
For High Dependence the probability of failure is (1+N)/2
For Moderate Dependence the probability of failure is (1+6N)/7
For Low Dependence the probability of failure is (1+19N)/20
For Zero Dependence the probability of failure is N

(1 + (___*_______))/___ = _________ Task Failure Probability With Formal Dependence

Figure 2.6 - ASP HRA Worksheet (page 2 of 2)

3. POOL HEATUP RESULTS

3.1 Pool Heatup: Instantaneous Frequencies

As shown in the example LP1 event tree shown in Figure 2.1, a loss of spent fuel pool cooling accident can lead to a number of different endstates, depending on the sequence of events comprising the accident. The endstates defined in this study are:

- FPIS1 - Spent fuel pool boiling
- FPIS2 - Spent fuel pool heatup (cooling restored late)
- FPIS3 - Spent fuel pool steaming (alternate cooling employed in "feed and boil")
- FPSF1 - Spent fuel pool boiling; flooding from loss of inventory or LOCA
- FPSF2 - Spent fuel pool heatup; flooding from loss of inventory or LOCA
- FPSF3 - Spent fuel pool steaming; flooding from loss of inventory or LOCA
- CD - Direct core damage (damage not caused by spent fuel pool heatup)
- OK - Success state

FPIS1 and FPSF1 are the most significant spent fuel pool-related endstates from the standpoint of SFP-heatup induced challenges to other plant systems; the other endstates are included in case a plant has a special vulnerability to steam or hot air (e.g., if solid state protection system cabinets can be exposed to the steam/hot air). From the standpoint of flooding following a loss of inventory or LOCA event, the FPSF2 and FPSF3 endstates also are potentially important.

Table 3.1 shows the initiating event frequencies used in this study; Table 3.2 shows the resulting frequencies for each of the endstates for Cases 1, 2, and 3. (Recall that in Case 1, both units are operating; in Case 2, Unit 1 is refueling and has offloaded 1/3 of its core; and in Case 3, Unit 1 is refueling and has offloaded its entire core.) It is important to note that even though the units of the frequencies shown are per year, these are instantaneous (and not annualized) frequencies. Thus, the Case j probability of Endstate i occurring over an arbitrary time interval [0,t] is given by:

$$P\{\text{Endstate i in } [0,t] \mid \text{Case j}\} = 1 - e^{-\lambda_{ij}t} \approx \lambda_{ij}t \tag{3.1}$$

where λ_{ij} is the appropriate frequency obtained from Table 3.2. (The approximation on the right hand side of Eq. 3.1 is reasonable as long as $\lambda_{ij}t < 0.1$.) In order to obtain the annual probability of endstate i, the contributions from each case are weighted and summed. The results of this weighting and summing are provided in the next section.

It should also be noted that the direct core damage (CD) endstate frequencies are underestimated for some scenarios. Because this endstate is being treated only to ensure that scenarios which lead to core damage before pool boiling are excluded from the analysis, the CD endstate was not modeled for some sequences where SFP cooling is assured. The CD endstate frequencies are provided for accounting purposes only, and should not be interpreted as representing the total core damage frequency.

Table 3.2 shows that for Case 1 (both units operating), three initiating events contribute almost 90% of the frequency of SFP boiling (i.e., endstates FPIS1 and FPSF1); they are loss of offsite power (LP1), small loss of inventory (LINCS), and large loss of inventory (LINVC). These contribute 66%, 17%, and 11%, respectively, to the frequency of boiling. For Cases 2 and 3 (one unit refueling), four initiating events contribute over 90% to the frequency of boiling: loss of offsite power (LP2 and LP3), large loss of inventory (LINVR), primary LOCA during refueling (PLOCR), and small loss of inventory (LINRS). The contributions to the pool boiling frequency from each of these initiators ranges from 20 to 25%. For all cases, the loss of the spent fuel pool cooling system (LSFP1, LSFP2, and LSFP3) initiator and the seismic (EQE) initiator are visible but not large contributors.

It can be seen in Table 3.2 that the relative contribution of LOOP to the pool boiling frequency is significantly higher for Case 1 than for Cases 2 and 3. This is not because the absolute frequency of LOOP-induced pool boiling changes, but rather because the instantaneous frequencies of the other contributing initiators (loss of inventory, LOCA) are significantly lower during operation than during refueling. (See Table 3.1.)

Note that the earthquake contribution is low because most earthquakes large enough to cause problems with the spent fuel pool are large enough to be a significant hazard to the rest of the plant. (The endstates FPIS1 and FPSF1 only include scenarios involving pool boiling but not direct core damage.) Similarly, a number of important PLOCR-initiated scenarios involve direct core damage, and do not contribute to the FPIS1 and FPSF1 endstates.

Note also that the results for the 1/3 core discharge (Case 2) and the full core discharge (Case 3) are only slightly different. This is due to the dominance of human error as a contributor to failure and a modeling simplification mentioned in Section 2.8: the Case 3 time windows used for the different human actions were employed for Case 2 as well. This simplification was made to reduce modeling effort; a deeper investigation of the different contributions of different refueling strategies should be conducted for this issue.

Tables 3.3a through 3.3d list the Case 1 and Case 3 cutsets for the FPIS1 and FPSF1 endstates which have instantaneous frequencies greater than 1.0 x 10^{-6}/yr. (The results for Case 2 are very similar to those for Case 3.) These tables show the instantaneous cutset frequency, the percent contribution of the cutset to the total endstate instantaneous frequency, the initiating event, the subsequent basic event successes and failures, and some notes on the cutsets.

For both endstates, it can be seen that very few hardware-related basic events are included in the top cutsets. Most of the basic events involve the failure of operators/plant personnel to perform a required action; also included are basic events modeling the failure to recover offsite power.

Table 3.3a lists the dominant FPIS1 cutsets for Case 1. Seven LOOP scenarios contribute over 50% of the total instantaneous FPIS1 frequency; these scenarios involve situations where the two spent fuel pools are isolated (so SFP cooling must be restored in both units) and where the operators fail to respond early enough to employ RHR in the SFP cooling assist mode. Two loss of inventory scenarios contribute more than 10% to the FPIS1 frequency. One scenario involves

the operators failing to respond early and failing to provide makeup late. The other scenario involves failure to provide makeup after the leak has been isolated.

Table 3.3b lists the dominant FPIS1 cutsets for Case 3. Two of the most important cutsets involve a small loss of inventory followed by failure of the operators to provide makeup; they contribute over 25% of the endstate frequency. The seven LOOP scenarios, which also contribute more than 25%, are similar to the scenarios discussed in Case 1; they involve failure of the operators to respond early.

The single dominant FPSF1 cutset for Case 1 is shown in Table 3.3c. The cutset involves failure to isolate a large leak in SFP cooling system and failure to provide makeup.

The two dominant FPSF1 cutsets for Case 3 are shown in Table 3.3d. Both cutsets involve failure of isolation and failure to provide makeup. It should be noted that, as shown in Appendix C, the human error probability used for these cutsets takes credit for the presence of a second unit and operating crew (the reduction factor is 0.14, or about 1/7). For a single unit plant, or for a multi-unit plant where the spent fuel pools are not cross-connected, such credit may not be taken. The instantaneous frequency of endstate FPSF1 during refueling may then be 2.6×10^{-4}/yr. Even after weighting the frequency to account for the fraction of time the plant is undergoing refueling, the resulting annual probability of endstate FPSF1 (which involves boiling in a drained pool) is on the order of 3×10^{-5}. This is higher than the total probability of pool draining employed in Ref. 2 (around 7×10^{-6}, as shown in Table 3.7). It therefore appears that for a number of plants, the probability of cladding fires may be higher than the value used in Ref. 2.

3.2 Pool Heatup: Annual Probability

Eq. (3.1) provides the probability of observing a particular endstate in a specified period of time when the plant is in a given configuration (as modeled by Cases 1 through 3). To estimate the probability of observing a particular endstate in one year, the following equation is used:

$$P\{\text{Endstate i in 1 year}\} \approx \lambda_{i1}(1-\phi_r) + \lambda_{ik}\phi_r \qquad (3.2)$$

where, as before, λ_{ij} is the frequency of Endstate i for Case j, k = 2 or 3 (depending on whether the plant performs 1/3 or full core offloads during refueling), and ϕ_r is the average fraction of time the plant is in a refueling outage.[1] Note that this model assumes that one unit is always operating over the year. It therefore does not treat situations when both units are in an outage, nor does it treat situations where one unit is in a non-refueling outage. It is not believed that these simplifications will greatly distort the results, due to the relatively small amount of time multi-unit plants would be undergoing simultaneous outages (under normal circumstances), and due to the lower heat loads associated with outages. However, as implied by the results of recent shutdown risk assessments [15, 16], the risk from these situations may not be negligible. Additional analysis is needed to determine the quantitative significance of these simplifications.

[1] To make the units consistent, the right hand side of Eq. (3.2) can be multiplied by 1 year.

The results for endstates FPIS1 and FPSF1 (where the SFP is boiling) obtained assuming an 18-month refueling cycle and a 2-month refueling outage are shown in Table 3.4. The results associated with a 1-month refueling outage are shown in Table 3.5. (These latter results assume that the model parameters, e.g., the failure rates, do not change significantly as the outage time decreases.) Not surprisingly, both annualized risk profiles closely resemble the risk profile for Case 1. However, the contribution for primary LOCA has become more important, due to its importance during refueling.

Only three initiators are modeled as being able to lead to flooding: a loss of inventory, a primary LOCA during refueling (i.e., a LOCA in a connected system — a "J LOCA", or a maintenance-induced LOCA — a "K LOCA"), or an earthquake. The annual probabilities for endstates involving flooding (i.e. FPSF1, FPSF2, and FPSF3) are shown in Table 3.6. Both the relatively high total probability (nearly 1 x 10^{-3}) and the risk profile (a large contribution from primary LOCAs) are notable.

3.3 Comparison With Earlier Studies

The results of an earlier NRC report on spent fuel pool risk (which focused on scenarios involving pool drainage and consequent zircaloy cladding fires) [2] and of an NRC-sponsored investigation of the risk at Susquehanna [3] are shown in Tables 3.7 through 3.10. (Table 3.9 provides the annualized frequencies obtained from Ref. 3; Table 3.10 presents the instantaneous frequencies, which can be compared with the results of this study shown in Table 3.2.)

Comparing Table 3.4 (this study) with the BWR "Best-Estimate" column in Table 3.7 (Ref. 2), it can be seen that the results of this study are higher by nearly an order of magnitude. (Note that Table 3.4 addresses spent fuel pool boiling; Table 3.7 addresses a complete loss of spent fuel pool inventory.) Perhaps more importantly, the dominant contributors are quite different.

In Ref. 2, the endstate frequency is dominated by seismic contributions. In this study, the earthquake contribution is relatively unimportant. This difference arises because: a) this study estimates much higher contributions for other initiators, as discussed below, and b) this study excludes the pool boiling contribution due to earthquakes that are severe enough to cause core damage, regardless of their impact on the spent fuel pool. Note that the sum of the combined CD, FPIS1, and FPSF1 frequencies in Table 3.2 is 7.3 x 10^{-6}/yr; this is quite comparable to the value of 6.7 x 10^{-6}/yr reported in Ref. 2.

Regarding the contribution from loss of inventory events, Ref. 2 reduces the empirical frequency of pneumatic seal failures (around 0.01/yr) by a factor of 1000 to arrive at an estimated frequency of severe pneumatic seal failures. As discussed in Section 5 of this report, there seems to be little evidence to support such a reduction; this report uses empirically estimated generic initiating event frequencies for losses of inventory, which include seal failures. (Details on the initiating event frequencies used in this study are provided in Appendix B.) Note that while this study's predicted annual probability of 4.6 x 10^{-6} for endstate FPSF1 (loss of inventory events only) is comparable to that of Ref. 2 (6.7 x 10^{-6}), the probability of endstate FPSF1 may increase by nearly an order of magnitude for single unit plants (see Section 3.1).

Regarding the contribution from LOOP events, this initiating event was not modeled explicitly in Ref. 2.

To compare the results of this study with those of Ref. 3, Table 3.4 can be compared with Table 3.9. It can be seen that the total frequencies of pool boiling differ only by about a factor of 2. Furthermore, the contributions from the different initiators are comparable. (Note that this study treats LOOP, Extended LOOP, and station blackout using a single initiating event; the combined NBF of 1.1 x 10^{-5}/yr in Table 3.9 compares reasonably well with the pool boiling frequency of 2.7 x 10^{-5} reported in Table 3.4 for LOOP events.) The most significant difference concerns the loss of inventory initiating event; this event is an important contributor in this study, but is not treated in Ref. 3. Another difference concerns the LOCA initiator; Ref. 3 does not treat special LOCAs during shutdown (the "J" LOCAs, i.e., LOCAs in connected systems, and the "K" LOCAS, i.e., maintenance-induced LOCAs). As a result, Ref. 3 underestimates the LOCA contribution to pool boiling frequency. A third significant difference concerns the LOCA with LOOP initiator. As discussed in Ref. 6, the treatment in Ref. 3 is incorrect and is overly conservative. This study does not treat the LOCA with LOOP initiator explicitly because of its very low probability.

It should be pointed out that while the overall pool boiling frequency results of this study are numerically comparable to those of Ref. 3, the modeling approaches employed are quite different. In Ref. 3, non-conservatisms in initiating event frequency estimates (e.g., for station blackout and for earthquakes) are balanced by quite conservative human error probabilities, by not deducting contributions from "direct core damage scenarios" (i.e., scenarios that lead to core damage directly regardless of the behavior of the spent fuel pool), and by not allowing credit for alternative cooling systems. This study eliminates the non-conservatisms in initiating event frequencies, but also employs more realistic human error probabilities, deducts contributions from direct core damage scenarios, and allows credit for alternative cooling systems. As it turns out, these modeling differences do not lead to significantly different bottom line results for the Susquehanna plant; however, they may lead to different results when applied to a different plant with different systems and operating practices.

It should also be noted that there is significant flooding potential associated with some of the scenarios treated in this study. This hazard is not treated in either Refs. 2 or 3. In principle, SFP-initiated flooding scenarios should be treated in standard internal flooding analyses. However, most of these analyses have been performed for operating units (i.e., for Case 1) only; the results of this study indicate that a significant flooding risk may arise during refueling, due to the higher (instantaneous) frequency of loss of inventory events. Note also that conventional flooding studies are not likely to address scenarios involving a combination of pool boiling and plant flooding (i.e., scenarios leading to endstate FPSF1).

3.4 Remarks

3.4.1 Caveats

The results presented in the preceding sections should be employed with caution for a number of reasons. First, the initiating event frequencies for loss of the SFPC system and for

losses of inventory are based on an early version of the AEOD database (dated June 13, 1996). A number of events have been added to the database since that date; the estimates for these initiating event frequencies may therefore be low.

Second, the results are based on generic data which may or may not be applicable to a specific plant being analyzed. In particular, the loss of inventory model employs industry-wide statistics to estimate the frequencies of SFPC system and SFP boundary failures during operation and refueling. However, there are wide variations in seal design; one design may allow large leaks on failure, while another may not. A rigorous, mechanistically based analysis of the severity-dependent likelihood of seal failures would be extremely helpful in addressing this issue.

Third, again in regard to the loss of inventory model, the available data for large seal failures are both sparse and uncertain. Different interpretations of the event narratives might lead to large changes in the estimated frequency of pool boiling and, perhaps more importantly, significant changes in the risk rankings of scenarios. A formal sensitivity analysis identifying key risk parameters (beyond basic events) and assumptions and determining the impact of changes in these parameters is outside the bounds of this limited scope study. Such an analysis should be performed before the results of this study are used in any decision support activities.

Finally, as a related point, the calculations performed in this analysis employ point estimates throughout. No attempt has been made to deal with uncertainties. An uncertainty analysis is a valuable tool for placing analysis results in context, as well as for identifying areas where additional modeling efforts may be useful. Again, an uncertainty analysis should be performed before the results of this study are used in any decision support activities.

3.4.2 Model Capability to Treat Other Plants

The results reported in Sections 3.1 and 3.2 are appropriate for plants whose design and operational practices are similar to those of the Susquehanna plant. However, the model used to generate these results is more general; with relatively simple changes in event tree top event success criteria, system fault trees, and/or basic event probabilities, a number of different plants can be modeled without a great deal of effort. Some of the different situations that can be treated are as follows.

1) A one-unit plant. Some of the modeling changes involved ensure that: a) the event tree top events modeling the fuel pool gate status are always failed, b) the top events modeling the cooling systems for the second unit are always failed, c) credit is not taken in the human reliability analysis for the actions of the second unit's crew (e.g., in establishing makeup during a loss of inventory scenario), and d) the initiating event PLOCA is not analyzed for Case 3 (since there is no fuel in the core).

2) Plants with a two-train SFPC system powered from safety buses. (Susquehanna has 3 trains powered from a non-safety bus.) The modeling changes here would involve: a) using fault trees appropriate for two-train systems, b) using common cause failure models appropriate for two-train systems, and c) ensuring that the event tree top event

FVPWR (a flag event modeling the availability of vital power to the SFPC system) is set to "true".

3) Plants with different reactor cavity seal designs. As discussed in the preceding section, a generic statistical analysis has been used to estimate the frequency of large seal failures. The results of a design-specific reliability analysis could be used to replace this generic estimate in the fault trees for top event LKSFP.

4) Plants which do not routinely leave their spent fuel pools cross-connected. (Susquehanna currently leaves its pools cross-connected.) Similar to the first sensitivity study, this involves modeling changes that ensure that the event tree top events modeling the fuel pool gate status are always failed.

5) Plants which have different maintenance policies for the diesel generators, SFPC system, and RHR system. For example, some plants may elect to perform all SFPC system maintenance prior to a refueling outage. Such a situation is simply modeled by increasing the maintenance unavailability of the SFPC system for Case 1 and reducing it for Cases 2 and 3.

6) Plants having potential instrumentation problems (e.g., vulnerability to hazards associated with spent fuel pool heatup, lack of redundancy, poor accessibility, poor readout design). This problem can be treated in the fault tree for top event OER by: a) adding basic events for instrumentation failures, and b) modifying the HRA to account for a poor human-machine interface (poor ergonomics).

7) Plants with different refueling policies (i.e., discharge of 1/3 of the core instead of the full core). As mentioned in Section 3.1, the current HRA model is not sensitive to the differences between the two policies. Case-dependent modifications of the time windows for operator actions are needed to make the human error probabilities case-sensitive.

Table 3.1 - Initiating Event Frequencies (Instantaneous)

Initiating Event	Description	Frequency (/yr)	Source
LSFP1	Loss of SFPC system, Case 1	2.4E-2	Data (see App. B)
LSFP2	Loss of SFPC system, Case 2	2.8E-1	Data (see App. B)
LSFP3	Loss of SFPC system, Case 3	2.8E-1	Data (see App. B)
LP1	Loss of offsite power, Case 1	8.0E-2	Ref. 14
LP2	Loss of offsite power, Case 2	8.0E-2	Ref. 14
LP3	Loss of offsite power, Case 3	8.0E-2	Ref. 14
LINVC	Large loss of inventory, Case 1	2.0E-3	Data (see App. B)
LINCS	Small loss of inventory, Case 1	5.0E-3	Data (see App. B)
LINVR	Large loss of inventory, Cases 2 and 3	2.0E-2	Data (see App. B)
LINRS	Small loss of inventory, Cases 2 and 3	3.0E-2	Data (see App. B)
PLOCA	Primary LOCA, Case 1	1.5E-2	Ref. 8
PLOCR	Primary LOCA, Cases 2 and 3	1.2E-1	Ref. 15, 16 (see App. B)
EQE	Seismic event ($0.2g < PGA \leq 0.6g$)[a]	1.2E-4	Ref. 17

[a]Earthquakes with PGA > 0.6g are for the purposes of this analysis, assumed to lead directly to core damage (with frequency 3.2E-6/yr).

Table 3.2 - Instantaneous Frequencies of Endstates (/yr)

	CD[a]	FPIS1	FPIS2	FPIS3	FPSF1	FPSF2	FPSF3	Total	Boil	% Boil
Case 1										
LSFP1		6.2E-07	8.6E-05	7.7E-06				9.4E-05	6.2E-07	1
LP1	1.4E-06	2.7E-05	3.6E-03	2.4E-04				3.9E-03	2.7E-05	66
LINVC		2.2E-06	9.7E-05	1.2E-06	2.2E-06	1.3E-04		2.3E-04	4.4E-06	11
LINCS		6.7E-06	3.1E-04	3.8E-06	7.4E-08	1.2E-05		3.3E-04	6.8E-06	17
PLOCA	8.7E-09	9.5E-07	1.3E-04	1.2E-05				1.4E-04	9.5E-07	2
EQE	6.1E-06	1.2E-06		5.5E-06	6.4E-08		1.1E-04	1.2E-04	1.3E-06	3
Total Case 1	7.5E-06	3.9E-05	4.2E-03	2.7E-04	2.3E-06	1.4E-04	1.1E-04	4.8E-03	4.1E-05	100
Case 2										
LSFP2		7.3E-06	1.0E-03	9.0E-05				1.1E-03	7.3E-06	7
LP2	1.5E-06	2.8E-05	3.7E-03	2.4E-04				4.0E-03	2.8E-05	25
LINVR		4.6E-06	1.6E-05	1.9E-07	2.3E-05	1.3E-03		1.3E-03	2.8E-05	25
LINRS		2.2E-05	1.5E-04	1.7E-06	8.5E-08	5.8E-05	1.7E-10	2.3E-04	2.2E-05	20
PLOCR	2.3E-05	1.1E-05	1.2E-02	1.6E-03	1.4E-05		5.4E-03	1.9E-02	2.5E-05	22
EQE	6.1E-06	1.2E-06		5.5E-06	6.4E-08		1.1E-04	1.2E-04	1.3E-06	1
Total Case 2	3.1E-05	7.4E-05	1.7E-02	1.9E-03	3.7E-05	1.4E-03	5.5E-03	2.6E-02	1.1E-04	100
Case 3										
LSFP3		8.3E-06	1.1E-03	1.0E-04				1.2E-03	8.3E-06	7
LP3	1.5E-06	2.9E-05	3.7E-03	2.4E-04				4.0E-03	2.9E-05	26
LINVR		4.6E-06	1.6E-05	1.9E-07	2.3E-05	1.3E-03		1.3E-03	2.8E-05	24
LINRS		2.2E-05	1.5E-04	1.7E-06	8.5E-08	5.8E-05	1.7E-10	2.3E-04	2.2E-05	20
PLOCR		1.1E-05	1.2E-02	1.6E-03	1.4E-05		5.4E-03	1.9E-02	2.5E-05	22
EQE	6.1E-06	1.2E-06		5.5E-06	6.4E-08		1.1E-04	1.2E-04	1.3E-06	1
Total Case 3	7.6E-06	7.6E-05	1.7E-02	1.9E-03	3.7E-05	1.4E-03	5.5E-03	2.6E-02	1.1E-04	100

[a]Endstate provided for accounting purposes only; frequencies listed do not provide total core damage frequency.

Table 3.3a - Dominant Cutsets, Endstate FPIS1 (SFP Boiling), Case 1

% Total	Frequency	IE	Cutset Elements (Successes and Failures)
8.20	3.19E-06	LP1	/DG1N2, /EPWR, ALT-XHE-XM-SFPL, SFP-OPEN-GATE, SFP-XHE-XE-LP, SFP2-XHE-XM-LSFP, /UNREC01
8.20	3.19E-06	LP1	/DG1N2, /EPWR, ALT-XHE-XM-SFPL, SFP-OPEN-GATE, SFP-XHE-XE-LP, SFP1-XHE-XM-LSFP, /UNREC01
8.15	3.17E-06	LP1	/GSTAT, /DG1N2, ALT-XHE-XM-SFPP, EPWR-XHE-EA-REC, LPWR-XHE-LA-REC, SFP-XHE-XE-LP, /UNREC02
7.92	3.08E-06	LINCS	/AISOL, LMKUP-XHE-XA-SFP, SFP-XHE-XE-LINVC, /LKSMC
7.34	2.86E-06	LP1	/DG1N2, /EPWR, ALT-XHE-XM-SFPL, /NCD-DGAL, SFP-OPEN-GATE, SFP2-XHE-XM-LSFP, UNREC-XHE-RECV-1
7.34	2.86E-06	LP1	/DG1N2, /EPWR, ALT-XHE-XM-SFPL, /NCD-DGAL, SFP-OPEN-GATE, SFP1-XHE-XM-LSFP, UNREC-XHE-RECV-1
7.30	2.84E-06	LP1	/GSTAT, /DG1N2, ALT-XHE-XM-SFPP, EPWR-XHE-EA-REC, LPWR-XHE-LA-REC, /NCD-DGAL, UNREC-XHE-RECV-2
4.78	1.86E-06	LINCS	/AISOL, /OERLINVC, MKP-XHE-XA-CSMIS, /LKSMC
4.17	1.62E-06	LP1	/DG1N2, ALT-XHE-XM-SFPP, EPWR-XHE-EA-REC, LPWR-XHE-LA-REC, RHR1-TRNS-UA-TM, /OER-LP, SFP-OPEN-GATE, /UNREC02
2.99	1.16E-06	EQE	ALT-XHE-XM-SFPL, /NDSFP-EQ, /NLEAK-EQ, /NCD-EQ, R1TRUE
2.77	1.08E-06	LINVC	/AISOL, LMKUP-XHE-XA-SFP, SFP-XHE-XE-LINVC, /LKLGC
69.19	(Total)		

Table 3.3b - Dominant Cutsets, Endstate FPIS1 (SFP Boiling), Case 3

% Total	Frequency	IE	Cutset Elements (Successes and Failures)
20.95	1.59E-05	LINRS	/AISOL, /OERLINVR, MKP-XHE-XA-RSMIS, /LKSMR
9.42	7.17E-06	LSFP3	/FGATE, ALT-XHE-XM-SFPL, SFP-OPEN-GATE, SFP-XHE-XE-UR, SFP1-XHE-XM-LSFP
7.88	6.00E-06	PLOCR	SFP-MKUP-ECCS-F, /NCDPLOCR, TGATE-STAT, MKP-XHE-XA-CSMIS
5.20	3.96E-06	LINRS	/MISLSFLE, /OERLINVR, MKP-XHE-XA-RSMIS, LKSMR
4.23	3.22E-06	LP3	/EPWR, ALT-XHE-XM-SFPL, SFP-OPEN-GATE, SFP-XHE-XE-LP, SFP2-XHE-XM-LSFP, /UNREC01
4.23	3.22E-06	LP3	/EPWR, ALT-XHE-XM-SFPL, SFP-OPEN-GATE, SFP-XHE-XE-LP, SFP1-XHE-XM-LSFP, /UNREC01
4.21	3.20E-06	LP3	/GSTAT, ALT-XHE-XM-SFPP, EPWR-XHE-EA-REC, LPWR-XHE-LA-REC, SFP-XHE-XE-LP, /UNREC02
3.79	2.88E-06	LP3	/EPWR, ALT-XHE-XM-SFPL, /NCD-DGAL, SFP-OPEN-GATE, SFP2-XHE-XM-LSFP, UNREC-XHE-RECV-1
3.79	2.88E-06	LP3	/EPWR, ALT-XHE-XM-SFPL, /NCD-DGAL, SFP-OPEN-GATE, SFP1-XHE-XM-LSFP, UNREC-XHE-RECV-1
3.77	2.87E-06	LP3	/GSTAT, ALT-XHE-XM-SFPP, EPWR-XHE-EA-REC, LPWR-XHE-LA-REC, /NCD-DGAL, UNREC-XHE-RECV-2
3.76	2.86E-06	LINVR	/OERLINVR, /MISLLGE, MKP-XHE-XA-RLGIS, LKLGR
2.15	1.64E-06	LP3	ALT-XHE-XM-SFPP, EPWR-XHE-EA-REC, LPWR-XHE-LA-REC, RHR1-TRNS-UA-TM, /OER-LP, SFP-OPEN-GATE, /UNREC02
1.96	1.49E-06	LINVR	/AISOL, /OERLINVR, MKP-XHE-XA-RLGIS, /LKLGR
1.85	1.41E-06	LINRS	/AISOL, MUES-XHE-XA-LSFP, SFP-XHE-XE-LINVR, /LKSMR
1.53	1.16E-06	EQE	ALT-XHE-XM-SFPL, /NDSFP-EQ, /NLEAK-EQ, /NCD-EQ, R1TRUE
1.44	1.09E-06	PLOCR	ALT-XHE-XM-SFP, RHR1-TRNS-UA-TM, SFP1-XHE-XM-SFP, /NCDPLOCR, TGATE-STAT

80.15 (Total)

Table 3.3c - Dominant Cutsets, Endstate FPSF1 (SFP Boiling + Flooding), Case 1

% Total	Frequency	IE	Cutset Elements (Successes and Failures)
76.78	1.81E-06	LINVC	/OERLINVC, MKP-XHA-XA-CLNIB, LKLGC, MISLLGE
76.78	(Total)		

Table 3.3d - Dominant Cutsets, Endstate FPSF1 (SFP Boiling + Flooding), Case 3

% Total	Frequency	IE	Cutset Elements (Successes and Failures)
61.45	2.26E-05	LINVR	/OERLINVR, MISLLGE, MKP-XHE-XA-RLNIB, LKLGR
36.72	1.35E-05	PLOCR	/NCDPLOCR, TGATE-STAT, MKP-XHA-XA-CLGNI, ILOC
98.17	(Total)		

Table 3.4 - Annual Probability of Spent Fuel Pool Boiling By Initiator
(18-month refueling cycle, 2-month refueling outage)

	Boiling (FPIS1)	Boiling + Flooding (FPSF1)	Total	% Total
Loss of SFP cooling system	1.4E-06	0.0E+00	1.4E-06	3
Loss of offsite power	2.7E-05	0.0E+00	2.7E-05	56
Loss of Rapid inventory (Large)	2.5E-06	4.5E-06	7.0E-06	14
Loss of Slow inventory (Small)	8.4E-06	7.5E-08	8.5E-06	17
Primary LOCA	2.1E-06	1.6E-06	3.6E-06	7
Earthquake	1.2E-06	6.4E-08	1.3E-06	3
Total	4.3E-05	6.2E-06	4.9E-05	100

Table 3.5 - Annual Probability of Spent Fuel Pool Boiling By Initiator
(18-month refueling cycle, 1-month refueling outage)

	Boiling (FPIS1)	Boiling + Flooding (FPSF1)	Total	% Total
Loss of SFP cooling system	9.9E-07	0.0E+00	9.9E-07	2
Loss of offsite power	2.7E-05	0.0E+00	2.7E-05	60
Loss of inventory (Large)	2.3E-06	3.4E-06	5.7E-06	13
Loss of inventory (Small)	7.6E-06	7.5E-08	7.6E-06	17
Primary LOCA	1.5E-06	7.8E-07	2.3E-06	5
Earthquake	1.2E-06	6.4E-08	1.3E-06	3
Total	4.1E-05	4.3E-06	4.5E-05	100

Table 3.6 - Annual Probability of Flooding Associated with SFP By Initiator
(18-month refueling cycle, 2-month refueling outage)

	Flooding (FPSF1, FPSF2, FPSF3)	% Total
Loss of SFP cooling system	0.0E+00	0
Loss of offsite power	0.0E+00	0
Loss of inventory (Large)	2.6E-04	27
Loss of inventory (Small)	1.7E-05	2
Primary LOCA	6.0E-04	60
Earthquake	1.1E-04	11
Total	9.9E-04	100

Table 3.7 - SFP Accident Frequencies, Generic Issue 82 Analysis [2]

	PWR		BWR	
Accident Sequence	Best Estimate	Upper Bound	Best Estimate	Upper Bound
Structural Failures				
Missiles	1.0E-8	1.0E-7	1.0E-8	1.0E-7
Aircraft crashes	6.0E-9	2.0E-8	6.0E-9	2.0E-8
Heavy load drop	3.1E-8	3.1E-7	3.1E-8	3.1E-7
Seismic	1.8E-6		6.7E-6	
Pneumatic Seal Failures	3.0E-8	5.0E-7	3.0E-8	5.0E-7
Inadvertent Drainage	1.2E-8	1.0E-7	1.2E-8	1.0E-7
Loss of Cooling and Makeup[c]	6.0E-8	1.4E-6	6.0E-8	1.4E-6
TOTAL	1.9E-6		6.8E-6	
Cladding Fire Probability	1.0		0.25	

[a]Adapted from Table 4.7.1, Ref. 2.

[b]All frequencies in events/yr.

[c]Includes beyond design basis seismic induced loss of cooling and makeup.

Table 3.8 - Susquehanna Annualized Initiating Event Frequencies (Ref. 3)

Initiating Event	Frequency (/yr)
Loss of SFPC	1.57E-4
LOOP	7.00E-2
Extended LOOP	7.00E-3
SBO	2.73E-8
LOCA	3.67E-3
Flooding	3.90E-3
Loss of SWS	2.00E-3
Pipe Break	3.40E-3
Seismic (PGA < 0.6g)	8.55E-6
Seismic (PGA ≥ 0.6g)	4.20E-7
LOCA w/LOOP	2.57E-4

Table 3.9 - Susquehanna Annualized NBF, As-Fixed Conditions (Ref. 3)

	Annualized Frequency (/yr)					
Initiating Event	Case 1	Case 2	Case 3	Case 4	Total	% Total
Loss of SFPC	1.1E-7	1.9E-8	5.0E-8	4.6E-8	2.3E-7	1.1
LOOP	5.5E-7	7.9E-8	8.5E-7	4.6E-7	1.9E-6	9.3
Extended LOOP	3.0E-6	4.0E-7	3.5E-6	2.1E-6	9.0E-6	43.2
SBO	4.0E-9	5.0E-10	1.1E-9	7.1E-10	6.2E-9	0.0
LOCA	1.5E-6	1.7E-7	1.6E-6	1.1E-6	4.3E-6	20.7
Flooding	2.8E-7	3.8E-8	3.8E-7	2.3E-7	9.3E-7	4.5
Loss of SWS	3.5E-8	5.0E-9	5.4E-8	2.9E-8	1.2E-7	0.6
Pipe Break	2.5E-7	3.3E-8	3.3E-7	2.0E-7	8.1E-7	3.9
Seismic (PGA < 0.6g)	1.2E-7	1.6E-8	6.9E-8	4.4E-8	2.5E-7	1.2
Seismic (PGA ≥ 0.6g)	3.1E-7	3.8E-8	4.6E-8	3.1E-8	4.2E-7	2.0
LOCA w/LOOP	1.6E-6	9.6E-8	6.9E-7	4.6E-7	2.8E-6	13.6
Total	7.7E-6	9.0E-7	7.6E-6	4.7E-6	2.1E-5	100.0

Case 1 = Both units operating

Case 2 = One unit operating, one unit shutdown, fuel not completely offloaded, RHR out of service part of the time

Case 3 = One unit operating, one unit refueling, 1 SFPC pump required to maintain T < 200°F

Case 4 = One unit operating, one unit refueling, 2 SFPC pumps required to maintain T < 200°F

Table 3.10 - Susquehanna Instantaneous NBF, As-Fixed Conditions (Ref. 3)

Initiating Event	Instantaneous Frequency (/yr)			
	Case 1	Case 2	Case 3	Case 4
Loss of SFPC	1.5E-07	2.1E-07	4.6E-07	6.3E-07
LOOP	7.6E-07	8.7E-07	7.8E-06	6.3E-06
Extended LOOP	4.1E-06	4.4E-06	3.2E-05	2.9E-05
SBO	5.5E-09	5.5E-09	1.0E-08	9.7E-09
LOCA	2.1E-06	1.9E-06	1.5E-05	1.5E-05
Flooding	3.9E-07	4.2E-07	3.5E-06	3.2E-06
Loss of SWS	4.8E-08	5.5E-08	4.9E-07	4.0E-07
Pipe Break	3.4E-07	3.6E-07	3.0E-06	2.7E-06
Seismic (PGA < 0.6g)	1.7E-07	1.8E-07	6.3E-07	6.0E-07
Seismic (PGA ≥ 0.6g)	4.3E-07	4.2E-07	4.2E-07	4.2E-07
LOCA w/LOOP	2.2E-06	1.1E-06	6.3E-06	6.3E-06
Total	1.1E-05	9.8E-06	6.9E-05	6.4E-05

Case 1 = Both units operating

Case 2 = One unit operating, one unit shutdown, fuel not completely offloaded, RHR out of service part of the time

Case 3 = One unit operating, one unit refueling, 1 SFPC pump required to maintain T < 200°F

Case 4 = One unit operating, one unit refueling, 2 SFPC pumps required to maintain T < 200°F

4. POST-HEATUP ACCIDENT PROGRESSION: DISCUSSION

4.1 Post-Heatup Hazards

During a loss of spent fuel pool cooling scenario, hazards to the rest of the plant can arise due to: the cause of the scenario (e.g., flooding water from a loss of inventory event); the plant response to the event (e.g., pool makeup water flowing through an unisolated leak); and the direct consequences of the scenario (e.g., heat and steam released during pool boiling). These hazards can be grouped into two categories based on their transport mechanisms: heat/steam and flooding. (Radiation, while a hazard to plant personnel attempting to mitigate the scenario, is a lesser hazard with respect to equipment performance during the course of the accident. Fires and their associated consequences are also not treated in this study; zircaloy cladding fires following a complete loss of spent fuel pool inventory are addressed in Ref. 2. Note that, as discussed in Section 3.1, the likelihood of these scenarios in single unit plants may be higher than estimated in Ref. 2.)

The following sections provide general qualitative discussions of the heat/steam and flooding hazards from the standpoint of their ability to initiate a core damage scenario, followed by discussions specific to the base case plant analyzed in this study. Note that a given hazard will be a potential problem to the core only if:

a) isolation is failed, i.e., there exists a pathway that allows movement of the hazard from the spent fuel area to vital safety equipment;

b) vital safety equipment functionality is lost due to the effects of the hazard;

c) operator hazard mitigation efforts fail, i.e., operators do not isolate or divert the hazard from the areas housing the vital safety equipment; and

d) operator efforts to recover failed equipment are unsuccessful.

It is important to recall that, per Eq. 2.1, a release of steam, heat, or water does not guarantee ECCS equipment damage. Even if some ECCS equipment are damaged, core damage is not guaranteed. Alternative equipment/systems and operator recovery actions may need to fail before core damage occurs.

To provide a risk perspective, typical at-power core damage frequencies estimated without consideration of scenarios involving the spent fuel pool generally fall in the range of 10^{-6} to 10^{-4} per reactor year for internal events; external events contributions can be of similar magnitude. The results of two shutdown risk analyses (internal events) indicate that the annualized frequency of core damage during shutdown may be lower than the at-power value. (NUREG/CR-6143 estimates an annual core damage frequency of 2×10^{-6}/yr for Grand Gulf, Unit 1 during a refueling outage [15]; NUREG/CR-6144 estimates an annual core damage frequency of 5×10^{-6}/yr for Surry, Unit 1 during mid-loop operations [16]; the comparable at-power values are 4×10^{-6}/yr and 4×10^{-5}/yr, respectively.) The instantaneous frequencies, obtained by dividing the shutdown

values by the fraction of time the plant is in shutdown, may be comparable to or greater than the at-power values.

The total pool boiling annual probabilities reported in Section 3 are around 5×10^{-5}. Assuming a baseline (non-SFP) core damage frequency of around 5×10^{-6}/yr, it can be seen that the combined failure probability of safety barriers (a)-(d) listed above *and* of the safety systems not damaged by the spent fuel pool scenario needs to be greater than around 10^{-3} in order for spent fuel pool-initiated scenarios to be visible (> 1%) contributors to core damage risk. Because the unavailability of a typical safety system is around 10^{-2}, this implies that the boiling associated with a severe spent fuel pool accident must damage most of the plant's ECCS equipment with high probability in order for the accident to affect core damage frequency.

Table 4.1 shows the impacts of the initiating events on the post-boiling safety barriers. (These are impacts beyond those associated with the heat/steam release from a postulated boiling spent fuel pool, e.g., limited room access.) It can be seen that for the dominant contributors to pool boiling (LOOP, loss of inventory), two of the safety barriers are unaffected. The third and fourth barriers can be affected, but, in the case of LOOP, some of the effects may actually be positive.

The preceding discussion applies to the risk associated with spent fuel pool accidents affecting core cooling. Scenarios involving spent fuel pool dryout and subsequent cladding fires do not require the failure of multiple engineered barriers to cause problems; they only require that pool boiling continue long enough that dryout conditions are reached. Because pool boil-off times are greatly reduced in cases where the pool has lost a large amount of inventory, the scenarios of special concern involve endstate FPSF1. Table 3.4 shows that the annual probability of FPSF1 is around 6×10^{-6}; this is comparable to the best-estimate probability for BWR pool draining reported in Ref. 2 (although the dominant contributors to this probability differ greatly from those reported in Ref. 2). Based on the value-impact analysis of Ref. 2, it might be argued therefore that the risk is not high enough to implement any of the alternatives identified in Ref. 2. However, as discussed in Section 3.1, the results of this study are based on a two-unit plant with a connected spent fuel pool. If credit is not taken for the makeup systems associated with the second unit, the estimated annual probability of endstate FPSF1 could be a factor of 7 higher (based on the HRA method used in this study). Additional investigation is needed to determine if the risk from this scenario is indeed significant.

4.2 Heat and Steam

4.2.1 Spatial Isolation

Heat and/or steam from the spent fuel pool has the potential to damage ECCS equipment through direct and indirect pathways. Two direct pathways are:

- transport through open air passages to the equipment; and
- transport through the plant heating, ventilation, and air-conditioning (HVAC) systems.

Given the compartmentalization typical of commercial nuclear power plant designs, the first mechanism is not likely to be a significant contributor to risk.[1] However, as seen in numerous external events analyses, plant layouts and communication paths between compartments (intended or otherwise) tend to be highly plant-specific. A plant walkdown is needed to confirm that this mechanism is not important for a given plant.

The second mechanism can potentially lead to widespread effects throughout the affected plant. Depending on the alignment of the ventilation system at the time of the accident, operator actions may be needed to isolate the spent fuel pool area from key equipment areas. (Tripping the ventilation system should greatly reduce the rate of heat/steam transport, but may not entirely prevent it.) Given the length of time available to the operators[2], the likelihood of failure is expected to be small. (For example, assuming that the need for action is obvious, the time available is expansive, stress is high, experience is low but the required actions are not especially difficult, procedures are available but not very specific, and the plant has a retrofit control system with good ergonomics, use of the worksheet shown in Figure 2.5 leads to a nominal estimate of 7×10^{-3}.) Again, due to plant-to-plant variations in design, a plant-specific analysis is needed to determine if this expectation is met for a given plant.

An indirect pathway for equipment loss due to heat/steam release from the spent fuel pool involves the plant's fire protection systems. As discussed in Ref. 18, steam can lead to the undesired actuation of a fire protection system through moisture intrusion into a fire protection system controller, activation of heat/smoke detectors, or melting of fusible links. Of 67 Licensee Event Reports on steam release events reviewed in Ref. 18, four involved actuations of the fire protection system. Three events involved smoke detector actuation in the same room as the steam release; the other event involved melting of a fusible link in the same room as the steam release, moisture ingress into a controller, and the actuation of the fire protection system in an adjacent room. (It is not clear if any of these four events involved steam of the quality that would be released from a spent fuel pool boiling at atmospheric pressure.) Ref. 18 estimates the generic probability of equipment damage, given actuation of a fire suppression system, for different suppression system types:

Water:	0.27/actuation
CO_2:	0.015/actuation
Halon:	0.0054/actuation

(The estimates for CO_2 and Halon are based on zero observed events in the database.)

Of course, the probability of equipment damage is expected to vary as a function of equipment type and design. If the probability of fire protection system actuation given exposure to a steam release is on the order of 10^{-1} and the probability of equipment damage is also on the order of 10^{-1}, then the combined probability of equipment damage due to a steam-initiated fire

[1] Intervening barriers, e.g., fire doors, can be failed of course, but the likelihood of such failures would have to be counted against the 10^{-3} margin discussed in the preceding section.

[2] Utility calculations for Susquehanna indicate that even if the HVAC system continues to run, environmental qualification temperatures — around 104°F — will not be exceeded for at least 6 hours after the start of boiling.

suppression system actuation will be around 10^{-2}. Because: a) this indirect pathway still requires the transport of heat/steam to a fire protection system controller, heat detector, or sprinkler head; b) multiple ECCS components must be damaged to seriously challenge core cooling; and c) these components are generally well separated and are not vulnerable to a single suppression system actuation, it is not expected that this pathway, by itself, will be a significant contributor to risk for most initiators. (The earthquake initiator may provide an exception, since severe earthquakes can trigger fire suppression systems [18]. However, this is not a spent fuel pool issue.)

4.2.2 Equipment Vulnerability

Most of the equipment in a nuclear power plant is relatively robust with respect to heated environments. For example, Ref. 19 reports the results of full-scale fire tests, in which electrical cables were exposed to a severe fire environment. The cable temperatures at which electrical failure occurred ranged from 420°F to 860°F. Ref. 20 notes that the standard design temperature of electrical cabinets is 120°C, i.e., about 250°F. These temperatures are above the temperature expected of steam leaving a boiling spent fuel pool.

Two groups of components that may be vulnerable to the temperatures associated with steam released from the spent fuel pool are:

- sensitive electronic equipment, e.g., solid-state components in advanced protection system cabinets
- equipment tripped or isolated by local high temperature signals (e.g., RCIC and HPCI turbines).

Long term exposures to elevated temperatures present possible problems for major components (e.g., large pumps). Extended losses of room cooling have been shown to be visible risk contributors in some risk studies. However, the likelihood of such extended exposures is not expected to be very high because of the time available for operators to mitigate the event. (Note also that in all of the scenarios contributing to the FPIS1 and FPSF1 endstates, either offsite or emergency diesel generator power is available.)

Aside from temperature issues, the vulnerability of equipment to a steam environment is less clear. Sensitive electronic equipment can be affected by elevated humidity levels, and it may be possible for steam condensation on electrical equipment (e.g., aging cables) to cause problems. Ref. 21 notes that the switchgear at Susquehanna are rated for 90% relative humidity, but argues that such a humidity level cannot be achieved within the switchgear cabinets, due to the elevated temperatures within the cabinets. Further investigation on the mechanism of failure is needed to validate this argument.

4.2.3 Hazard Mitigation

In the case of the heat/steam hazard, mitigation involves either the isolation of the spent fuel pool area from other key plant areas, or the active diversion of steam. Manual operator actions will be required; the likelihood of success depends on the same factors routinely considered in human

reliability analyses, e.g., accessibility and availability of equipment, quality of procedures and training, and time available. Note that the mitigation efforts can be affected by the characteristics of the initiating event, as shown in Table 4.1.

4.2.4 Equipment Recovery

If irreparable damage has not occurred, equipment lost during the scenario may be recovered in many cases simply by allowing the affected equipment to cool/dry off. (Equipment recovery is not an issue, of course, if hazard mitigation efforts are not successful.) For example, in the case of HPCI/RCIC room temperature isolation signals, the signal can be reset once the temperature level drops. As in the case of hazard mitigation, manual actions will be required. These actions can be affected by the initiating event, as shown in Table 4.1.

4.3 Flooding

The treatment of SFP-initiated flooding scenarios is similar to the treatment of other internal flooding scenarios in conventional analyses [22]. Thus, the discussion on safety barriers (a)-(d) identified in Section 4.1 is straightforward. Note that in principle, SFP-initiated floods should be treated in standard internal flooding analyses. However, most of these analyses have been performed for operating units (i.e., for Case 1) only; the results of this study indicate that a significant flooding risk may arise during refueling, due to the higher (instantaneous) frequency of loss of inventory events. Note also that conventional flooding studies are not likely to address scenarios involving a combination of pool boiling and plant flooding (i.e., scenarios leading to endstate FPSF1).

4.3.1 Spatial Isolation

For a flooding event to seriously challenge the ECCS function, there must be a path from the source of the flooding to multiple ECCS equipment areas. This path can include non-watertight doors (e.g., fire doors) and drainage systems (water may back up through drains if anti-backflow devices are not installed or functioning).

4.3.2 Equipment Vulnerability

It can be assumed that most electrical equipment is vulnerable to flooding, as long as the water level rises high enough to immerse key component parts. In addition, some equipment (e.g., electrical panels) may be vulnerable to water spray or dripping.

4.3.3 Hazard Mitigation

Unlike the heat/steam hazard, flooding may occur relatively rapidly, especially in the case of large seal failures. Noting that the FPSF1 endstate is only arrived at in scenarios where the operators have failed to isolate the flood, the likelihood of success of further mitigation efforts may be relatively low.

4.3.4 Equipment Recovery

In conventional flooding analyses, credit is not usually taken for recovery of equipment damaged by a flood.

4.4 Application to Base Case Plant

This section applies the preceding discussion points to the base case plant, which is largely based on the Susquehanna Steam Electric Station (SSES).[3] The information underlying the application is drawn from a limited walkdown[4] of the SSES and from discussions with Pennsylvania Power and Light Co. staff.

4.4.1 Heat and Steam Hazard

Noting that the conditional probability of core damage given spent fuel pool boiling must be on the order of 10^{-3} or higher for spent fuel pool boiling accidents to become important contributors to risk, it does not appear that spent fuel pool boiling accidents are important contributors at the base case plant. The reasons for this conclusion are as follows.

- While transport of heat and steam throughout key areas of the plant is possible (via the HVAC system in recirculation mode), the length of time required for significant room heatup (see Footnote 2) greatly reduces the likelihood of operator failure.

- Transport of significant amounts of heat and steam from the spent fuel pool area to areas housing ECCS equipment via open air passages (as opposed to transport through the HVAC system) appears to be very unlikely, due to the separation of the spent fuel pool area from other parts of the reactor building and the fact that the spent fuel pool area is significantly higher than the ECCS equipment areas.

- The only ECCS equipment with potential vulnerability to the elevated temperatures caused by the postulated pool boiling and steam transport appear to be the HPCI and RCIC turbines (which are isolated on high room temperature — 167°F — signals).

- The plant has provisions to divert steam through the standby gas treatment system, should pool boiling occur.

4.4.2 Flooding Hazard

In contrast with the heat/steam hazard, the flooding hazard associated with certain loss of spent fuel pool cooling accidents is more difficult to dismiss as a risk contributor.

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[3] The base case analysis employs generic estimates for initiating event frequencies, and does not necessarily reflect SSES-specific design or operational details.

[4] The walkdown covered major ECCS equipment and areas, but did not cover cable routings, the fire protection system or penetrations.

- The frequency of flooding is greater than the frequency of boiling. The annual flooding probability may be on the order of 10^{-3}, as shown in Table 3.6. Focusing on large seal failures alone, the cutset results for endstates FPSF1, FPSF2, and FPSF3 can be used to show that the annual probability of flooding is around 3×10^{-4}. Using this latter value, scenarios with a 2×10^{-4} conditional probability of core damage could be visible risk contributors.

- The boundaries between elevations in the reactor building are not necessarily watertight. As indicated by conversations with utility staff, it is probably fair to assume that, at least in the case of a failure of the inflatable seal between the reactor cavity and the reactor building during refueling, all flood water will eventually end up in the bottom floor of the reactor building (where the major ECCS equipment are housed). It is not clear that all flood water will end up in the reactor building sump (as assumed in Ref. 3). Note also that the room drains on the bottom floor employ normally closed manual valves instead of check valves. While the valve positions are checked every refueling outage, there is a possibility that the valves could be mispositioned, providing a flow path from the sump to each affected room.

- For large flooding events, the time available to take mitigative actions may be significantly less than for room heatup events.

The above reasons do not guarantee that the conditional core damage probability is greater than 2×10^{-4}, but they do indicate that further investigation of the flooding hazard may be needed. Note that the Individual Plant Examination for the SSES indicates that the core damage frequency due to flooding during normal operations is around 4% of the total core damage frequency [23]; it is not clear from available documentation that the analysis addresses floods involving the spent fuel pool.

Table 4.1 - Post-Boiling Impacts for Initiating Event Classes

	Post-Boiling Impacts				
Initiating Event Class	Spatial Isolation	Equipment Vulnerability	Hazard Mitigation	Equipment Recovery	Other
Loss of SFPC system	—	—	—	—	
Loss of offsite power	—	—	(1, 2)	(1)	
Loss of inventory	—	—	(3)	(3)	For unisolated leaks, low pool inventory reduces boil-off time; increases importance of cladding fire issue
Primary LOCA	—	—	(3)	(3)	For unisolated leaks, low pool inventory reduces boil-off time; increases importance of cladding fire issue
Earthquake	(4)	(5)	(1, 2, 3, 6)	(1, 3)	For unisolated leaks, low pool inventory reduces boil-off time; increases importance of cladding fire issue

NOTES:

1) Power may be unavailable to various systems. (Endstates include contributions from scenarios where either diesel generators are available or offsite power has been recovered. In all cases, some power is available; station blackout scenarios provide a direct challenge to the core, independent of the spent fuel pool.)
2) Loss of power may actually be beneficial from the standpoint of circulation of heat/steam by the ventilation systems.
3) Flooding may inhibit access to different areas of the plant.
4) Severe seismic events might affect barriers that ordinarily prevent the passage of steam/hot air or water to other areas of the plant.
5) Seismic events might distort or damage electrical cabinet boundaries and increase the vulnerability of electrical equipment enclosed in those cabinets.
6) Seismic events might fail equipment used to mitigate hazards (e.g., dampers, forced ventilation systems).

5. RISK ASSESSMENT INSIGHTS FROM OPERATIONAL DATA

In order to gain additional insights concerning the risk associated with spent fuel pools, operational data collected by the AEOD were reviewed. The primary objective of the review was to identify issues that should be addressed in the risk model (e.g., observed initiators, representative sequences of events). A secondary objective was to develop, where possible, quantitative insights relevant to the risk model. This section summarizes the results of the review.

5.1 Event Data

Table 5.1 provides a breakdown of reported spent fuel pool events included in the AEOD database by event type. (This breakdown is current as of June 13, 1996; it has been changed since then.) Both actual occurrences and potential events[1] are shown.

The AEOD database covers events reported over the period 1976 through June, 1996. Table 5.2 shows a breakdown of the events by year. (The total number shown is less than that shown in Table 5.1, since a number of events in Table 5.1 are included in multiple event type categories.) It can be seen that the annual number of events increases about halfway through the time period.

Two key classes of events of interest to this study involve loss of spent fuel pool cooling and loss of spent fuel pool inventory. The loss of cooling events for which severity information are available are listed in Table 5.3. The loss of inventory events for which severity information are available are listed in Table 5.4. Table 5.4 distinguishes between the events that involved the failure of inflatable seals from other losses of inventory.

The loss of cooling events primarily resulted from loss of electrical power to the SFPC pumps, and/or from engineered safety features (ESF) actuations causing the load shed of the pumps. In most cases, plant staff quickly restored SFPC. In those cases where the cooling loss was extended, lack of operator awareness was often the cause. In none of the reported events did the cooling loss result in pool temperatures exceeding the plant Technical Specification limits (where applicable) or approaching boiling conditions.

From the standpoint of a risk assessment, the Wolf Creek event (September 30, 1994) is one of the more notable loss of SFPC events In this event, the entire reactor core had been offloaded to the SFP and the plant staff calculated the time to boil to be about 5.8 hours. A significant portion of the 'A' train systems were out of service for maintenance, including the component cooling water (CCW) required for the 'A' train SFPC heat exchanger. The 'A' emergency diesel generator (EDG) was also out of service for testing. While in this condition, the operators discovered smoke emanating from the 'B' train SFPC pump inboard bearing, and were forced to secure the pump. In 18 minutes, the operators were able to restore the 'A' CCW train to

[1] Potential events involve either: a) conditions which degraded the ability of the plant to respond to an initiating event, or b) hypothesized scenarios which have become potentially important due to the acquisition of new knowledge (e.g., concerning accident phenomenology or plant conditions).

service and start the 'A' SFPC pump. Later, during subsequent post-maintenance testing of the 'A' train EDG, the diesel's exciter transformer failed and caught fire. With no backup power available to the only operable SFPC pump (the 'A' pump), the licensee began seeking ways to temporarily supply emergency power to the pump in case offsite power failed. Fortunately, this did not occur in the two days it took to restore the EDG.

The loss of inventory events primarily resulted from loss of inflatable seals (usually from loss of air) or from system alignment problems. Not all of the reports specified the amount or duration of the inventory loss. Of those that reported the amount of inventory lost, the scenarios ranged from small leaks to large rapid losses.

The largest loss of inventory event in the database is the well-known Haddam Neck event (August 21, 1984). During that event, the refueling cavity seal failed while the cavity was filled. Within 20 minutes, 200,000 gallons of water spilled through to the lower containment levels. Preparations had been underway for opening the fuel transfer tube, but the tube was shut at the time of the event. Had the transfer tube been open, the SFP could have been drained leaving spent fuel partially uncovered.

Comparing the event categorization of Table 5.1 with the INEL SFP model documented in Appendix A, it can be shown that almost all of the categories and subcategories are explicitly treated in the model (through the identification and quantification of initiating events). The exception is the "Other" category, which involves events outside the bounds of the study (e.g., boron dilution events in PWRs).

5.2 Mapping to Risk Model

The risk model deals with the failure of mitigating systems (the safety barriers) as well as with initiating events. To determine if the risk model addresses observed combinations of initiating event/safety barrier failures, the events in the AEOD database were reviewed to determine how they would map to the risk model. (This mapping procedure is similar in spirit to that performed in the accident sequence precursor — ASP — program for reactor incidents. The only difference is that the analysis performed in this study is qualitative; no effort is made to determine "how close" an event came to spent fuel pool boiling. Of course, the quantitative models developed in this study could be applied for this purpose.)

The results of the mapping analysis are shown in Tables 5.5 and 5.6. Table 5.5 shows the mapping of actual and potential events to the initiating events defined in Section 2.2. (The number in parentheses corresponds to events which occurred in the time period 1987 – June 1996.) It can be seen that the loss of spent fuel pool cooling and loss of inventory initiating events appear to be the most likely. However, it should be cautioned that the database is focused on spent fuel pool issues. Thus, for example, loss of offsite power events for which spent fuel pool impacts have not been reported are not included in the database.

Table 5.6 shows, for events which had impacts or potential impacts beyond the initiating event, the event tree top events potentially affected (but not usually failed). Comparing Tables 5.5 and 5.6, it can be s n that most of the event sequences did not progress beyond the initiating

event. Most events were quickly identified and corrected by plant staff prior to significant consequences to the spent fuel pool. Of those that progressed beyond the initiating event, most involved indication or operator awareness issues. Except for the previously mentioned Wolf Creek loss of spent fuel pool cooling event, all of the actual events (and most of the potential events) listed in Table 5.6 are explicitly treated in the risk model. The Wolf Creek event is not completely treated because the SFPC and RHR fault tree models developed in this report do not explicitly treat loss of AC power. While this is a weakness in the fault trees, it should be pointed out that the likelihood of a concurrent LOOP is generally lower by an order of magnitude or more than the human error and common cause failure probabilities included in the fault trees.

5.3 On Reductions in Seal Failure Frequency

In Ref. 13, it is estimated that "advances in seal design, increased awareness and surveillance" will lead to a factor of 10 reduction in the frequency of seal failure. However, Table 5.4 shows that 6 seal failure events have occurred in the time period (1976 – 1986) and 5 have occurred in the period (1987 – June, 1996). Counting the events for which severity data were not found, the respective numbers are 7 and 8. Since the number of operating reactors has not changed dramatically since 1976, there appears to be little support for the proposed reduction.

Ref. 13 also estimates the fraction of serious seal failures (i.e., failures with the potential to quickly drain the pool) to be about 0.01. The evidence provided by the AEOD database is less conclusive here. While one out of the 11 seal failure events in Table 5.4 was a rapid, large draining event (the Haddam Neck event), it has been argued that this was an event associated with a unique design. Without this event, the statistical evidence does not support large estimates for the frequency of severe seal failures. (Of course, it also does not provide strong evidence that the failure frequency is as small as 0.01.) A more thorough investigation of seal designs, operational practices (including inspection, testing, and maintenance), failure mechanisms, and failure recovery resources and practices is needed to provide a technically based estimate of severe seal failure frequencies.

5.4 Concluding Remarks

As a result of this review of the AEOD database, the following conclusions relevant to the construction and evaluation of the risk model can be drawn.

- While a number of events relevant to the loss of spent fuel pool cooling are included in the AEOD database, none of these events resulted in substantial heating of the pool or uncovering of the fuel. (There has been at least one "near miss" — the Haddam Neck seal failure event.) This implies that the frequency of significant loss of cooling should be less than 10^{-3}/ry.

- The INEL risk model documented in the report appendices explicitly treats most of the events included in the database.

    - Almost all of the categories and subcategories of events included in the AEOD database are explicitly treated in the model. The categories and subcategories not

treated are either not directly relevant to the loss of spent fuel pool cooling issue or appear to be probabilistically dominated by categories/subcategories with the same impact.

- All of the relevant events can be mapped to the initiating events treated in the risk model.

- Of the events which apparently affected safety barriers treated in the event trees, most can be directly treated by the model.

- The Wolf Creek event provides one exception, because LOOP events subsequent to a loss of spent fuel pool cooling have not been explicitly treated.

- For those events which correlate to one of the initiating events modeled in this report, very few of the safety barriers modeled by the event tree top events have been seriously degraded (or even challenged).

• The database appears to indicate that the frequency of reported seal failures has remained relatively constant over time. Also, while it does not preclude small-valued estimates (0.01 or less) for the frequency of serious seal failures, it does not support these estimates either.

Table 5.1 - Breakdown of AEOD Database Events

Type of Event	Counts[a] Actual	Potential	Total	Type #[b]
Loss of Water	46	47	93	
Gates Leak	10	2	12	1
Transfer Tube Leaks	0	0	0	2
Pool Leaks	9	1	10	3
Pool Seismically Damaged	0	8	8	4
Anti-Siphon Device Fails	1	1	2	5
Refuel Cavity Seal Leaks	2	0	2	6
Loss of Coolable Geometry	0	0	0	7
Drop of "Penetrating" Load	5	30	35	8
Pipe Problems	0	3	3	9
Alignment Problems	19	1	20	23
System Interaction	0	1	1	31
Loss of Makeup Capability	0	6	6	
Valve Problems	0	1	1	12
Pipe Problems	0	4	4	13
LOOP w/Failure to Restore	0	1	1	14
Loss of Cooling	58	36	94	
Pumps Stop (Loss Electric Power)	38	9	47	15
Pump Problems	5	15	20	33
Heat Exchangers Fail	0	1	1	16
Loss of HX Cooling	8	4	12	17
Pipe Problems	1	1	2	18
Alignment Problems	2	0	2	25
Inadequate Cooling	0	6	6	28
ESF Isolation Loss of Cooling	4	0	4	29
Loss of Non-SFP Equipment	4	2	6	
Boiling	0	0	0	20
Flooding	4	0	4	21
Susquehanna Effect	0	2	2	24
Other	16	45	61	
Ventilation	5	24	29	26
Criticality	0	18	18	27
Loss of Monitoring Capability	0	3	3	32
Other	11	0	11	30

[a]Count as of 6/13/96; revisions have been made since then.
[b]"Type" field coded in AEOD database.
[c]Total count = 260; total number of events = 245 (some events are in multiple categories).

Table 5.2 - Breakdown of AEOD Database Events By Year

Year	Number
not known	2
1976	1
1980	3
1981	7
1982	6
1983	5
1984	8
1985	11
1986	7
1987	15
1988	17
1989	20
1990	14
1991	15
1992	20
1993	20
1994	21
1995	36
1996[a]	17
Total	245

[a]As of June, 1996

Table 5.3 - Loss of SFP Cooling Events[a]

Plant	Date (y/m/d)	Duration (hr)	Heatup (°F)	Notes
Point Beach 1	81/01/02	0.03		
Haddam Neck	87/08/14	1.3	6	
Browns Ferry 1	87/12/04	0.10		
Peach Bottom 2	87/12/30	1+		
Haddam Neck	90/06/08	0.42		
Duane Arnold	90/07/09	minutes		
Turkey Point 4	91/03/13	1.5	3	
Indian Point 2	91/06/22	0.60		
Turkey Point 4	91/06/26	1	0	
Haddam Neck	91/07/17	8	3	
Turkey Point 3	91/07/24	0.32	1	
Haddam Neck	91/08/14	10.5	3	
Haddam Neck	91/09/20	6		
Haddam Neck	92/01/31	1.5		
Nine Mile 1	92/02/21	0.25		
Haddam Neck	92/02/22	1.67		
Comanche Peak 1	92/05/11	17	5	
Indian Point 2	92/06/19	0.17		
FitzPatrick	92/06/23	7.5	7	
Millstone 2	92/07/06	1.5	4	
Cooper	93/03/28	0.17		
Cooper	93/05/14	0.25	0	
South Texas 2	93/06/14	13	19	
Haddam Neck	93/06/22	0.05		
Haddam Neck	93/06/26	0.67	4	
Duane Arnold	93/08/13	6	7	Partial loss
LaSalle 2	93/09/14	5	5	
Farley 2	93/10/05	3	40	
Salem 1	93/10/21	0.08		
Haddam Neck	94/04/23	13+	7	
Seabrook 1	94/08/11	24	30	
Wolf Creek	94/09/30	0.30		Time to boil = 5.8 hr
Indian Point 3	95/02/27	2.25		
Diablo Canyon 1	95/10/21	8	20	
San Onofre 1	95/10/25	0.33	1	Estimated time
Limerick 1	96/02/20	0.87	2	
Haddam Neck	96/03/01	32	16	

[a]Includes only the 37 events for which severity information is available; database includes 56 loss of SFPC events.

Table 5.4 - Loss of SFP Inventory Events[a]

Plant	Date (y/m/d)	Duration (hr)	Amount (gal / ft)	Seal Fail?	Rflg Otg?[b]	Est Size[c]	Notes
Davis Besse	82/02/01				–	M	Below TSL[d]
Salem 1	82/02/01	0.33	23,000 / –		–	S	Est. 20 hrs drainage time[e], connected to Rx cavity
Trojan	82/06/10				Y	L	15" below TSL; fast
Wolf Creek	87/12/22				Y	S	22 ft above TAF[f]
Harris	89/01/17		– / 5		N	L	Sounds fast, easily fixed
Clinton	89/02/03		– / 0.42		Y	M	
Millstone 2	92/07/06		– / 1.2		N	L	Sounds fast, easily fixed
La Salle 2	93/01/01				Y	S	Small leak
Indian Point 1	94/05/18				N	S	Small leak
Hatch 1	94/12/28				N	S	Small leak
San Onofre 1	95/05/16				N	S	Small leak
Braidwood 1, 2	95/05/30		3,000 / 0.25		N	–	
Duane Arnold	95/06/14				N	S	5" below TSL
Cooper	95/10/31		10,000 / 0.1		Y	S	Est. 26 hrs drainage time, connected to Rx cavity
Arkansas 2	96/03/20		900 / 0.13		N	S	
Trojan	80/05/22			Y	Y	–	10" below TSL
Arkansas 2	81/05/15		– / 2	Y	–	–	
Haddam Neck	84/08/21	0.33	200,000 / –	Y	Y	L	Refueling cavity loss only
San Onofre 2	84/10/02	0.70	– / 1.7	Y	–	S	Est. 8 hrs drainage time; no fuel in SFP
Sequoyah 1,2	85/12/18	1	– / 2	Y	Y	–	
Hatch 1	86/12/02	24	– / 5.5	Y	N	S	
Surry 1	88/05/17		25,800 / –	Y	Y	–	
Wolf Creek	91/09/23	4		Y	Y	S	Below TSL
Comanche Peak 1	93/10/26		20,000 / –	Y	Y	L	
Hope Creek	94/04/13		20,000 / –	Y	Y	–	22 ft above TAF
Indian Point 2	95/01/20		– / 3+	Y	N	–	20 ft above TAF

[a]Includes only the 26 events for which severity information is available; database has 29 loss of inventory events. The database contains 4 additional seal failures for which severity information was not found: San Onofre (November 5, 1984), Surry (October 2, 1988), Point Beach (September 30, 1992), and San Onofre (April 12, 1995).

[b]Did event occur during a refueling outage?

[c]Estimates are based on interpretation of event narratives. Estimated size is large if operators have a small amount of time to isolate the leak and small if they have an expansive amount of time. Note that the rate of level drop during a small leak may still be great enough to dominate boiling as a mechanism for inventory loss, at least until the maximum level drop occurs.

[d]Technical Specification Limit

[e]Top of Active Fuel

Table 5.5 - Mapping of AEOD Database Events to Initiating Events

Initiating Event	Event Counts[a]		
	Actual	Potential	Total
Loss of spent fuel pool cooling (LOSFP)	56 (53)	23 (21)	79 (74)
Loss of offsite power (LP1, LP2, LP3)	4 (4)	14 (4)	18 (8)
Loss of inventory (LINVC, LINVR)	30 (20)	13 (6)	43 (26)
Primary LOCA (PLOCA, PLOCR)	0 (0)	1 (1)	1 (1)
Seismic (EQE)	0 (0)	10 (7)	10 (7)
Total	89 (77)	78 (66)	106 (85)

[a]Numbers in parentheses represent events occurring in the time period 1987 – June, 1996

Table 5.6 - Mapping of AEOD Database Events to Initiating Events and Top Events

Plant	Date (y/m/d)	Actual/ Potential	IE[a]	Top Events[b,c]	Notes
La Salle	93/01/01	A	LOINV	OER	Indication
Trojan	80/05/22	A	LOINV	OER	Indication
Trojan	82/06/10	A	LOINV	OER	Indication
Wolf Creek	87/12/22	A	LOINV	OER	Indication
Wolf Creek	91/09/23	A	LOINV	OER	Indication
Nine Mile	92/02/21	A	LOSFP	S1, R1, LS1	Ultimate heat sink, RHR OOS
Seabrook	94/08/11	A	LOSFP	OER	Awareness
South Texas	93/06/14	A	LOSFP	OER	Awareness
Wolf Creek	94/09/30	A	LOSFP	S1, R1, LS1	SFPC system (1 train), RHR OOS
WNP	93/04/28	P	LOCA	ALT-C, SSNV	
Oconee	96/01/08	P	LOINV	ERMUP, LTMUP	Makeup
Catawba	88/01/09	P	LOINV, LOSFP	ALT-C, ERMUP, LTMUP	Alternate cooling, makeup
Cook	96/04/14	P	LOOP	DG1N2, DG1O2	Diesel generators
Davis Besse	96/04/03	P	LOOP	DG1N2, DG1O2	Diesel generators
Diablo Canyon	91/02/13	P	LOOP	DG1N2, DG1O2	Diesel generators
Susquehanna	92/10/20	P			Susquehanna issues
Arkansas	96/05/22	P	LOSFP	S1, LS1	Heat load
Millstone	93/09/17	P	LOSFP	S1, LS1	Heat load
Nine Mile	96/03/28	P	LOSFP	S1, LS1	Heat load
Oyster Creek	83/12/23	P	LOSFP	S1, LS1	Spent fuel pool cooling
River Bend	91/04/15	P	LOSFP	OER	Appendix R, indication
San Onofre	90/04/24	P	LOSFP	S1, LS1	Heat load
WNP	90/20/24	P	LOSFP		Appendix R
Diablo Canyon	91/04/10	P	Seismic	LTMUP	Alternate cooling, makeup
Millstone	95/05/17	P	Seismic	LTMUP	Makeup
WNP	89/08/11	P	Seismic		Appendix R

[a]IE = initiating event, LOINV = loss of inventory, LOSFP = loss of spent fuel pool cooling system, LOCA = loss of coolant accident, LOOP = loss of offsite power
[b]See Appendix A for definition of top events
[c]Top events affected (but not necessarily failed)

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