November 20, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT COMPANY	)	Docket No. 50-400-LA
	)	
(Shearon Harris Nuclear Power Plant)	)	ASLBP No. 99-762-02-LA

SUMMARY OF FACTS, DATA, AND ARGUMENTS ON WHICH APPLICANT PROPOSES TO RELY AT THE SUBPART K ORAL ARGUMENT REGARDING CONTENTION EC-6

VOLUME 1

EXHIBIT 1

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COMPANY	)	
(Shearon Harris Nuclear Power Plant)	)	ASLBP No. 99-762-02-LA

AFFIDAVIT OF EDWARD T. BURNS, Ph.D.

COUNTY OF DUPAGE)
) ss:
STATE OF ILLINOIS)

I, Edward T. Burns, being sworn, do on oath depose and say:

1. I am a resident of the State of California. I am employed by ERIN Engineering and Research, Inc. ("ERIN") as Vice President and General Manager of BWR Technology. My business address is Suite 350, 2105 S. Bascom Avenue, Campbell, California 95008.

2. I was graduated from Rensselaer Polytechnic Institute in 1967 with a Bachelor of Science in Engineering Science, in 1968 with a Masters of Science in Nuclear Engineering, and in 1971 with a Ph.D. in Nuclear Engineering. Since graduation, I have been employed by the United States Department of Energy, Naval Reactors Division; Science Applications, Inc.; TENERA, L.P.; and ERIN Engineering and Research, Inc. During my tenure at ERIN, I have served as a manager and lead technical analyst for preparation and review of Level 1 and Level 2 Individual Plant Examinations ("IPE") using probabilistic risk

assessment techniques for numerous United States nuclear facilities. My resume and a list of publications over the last ten years are provided as Attachment A to this affidavit.

3. The purpose of this affidavit is to describe the extensive probabilistic analysis and review effort performed by ERIN, under the direction of Carolina Power & Light Company ("CP&L"), to determine the best estimate of the overall probability of the postulated sequence set forth in the chain of seven events as described on page 13 of the Board's Memorandum and Order dated August 7, 2000 ("Board's Order"), and the applicability of NUREG-1353 to this effort. First, I describe my role in preparing a response to the Board's questions, the task assigned to ERIN by CP&L, and the team I assembled to perform that task. Second, I describe generally the bases of probabilistic risk assessment, the advances in techniques and knowledge since initial applications, and the quality of the existing Harris Nuclear Plant ("Harris Plant" or "Harris") IPE and updated Probabilistic Safety Assessment ("PSA"). Third, I discuss the methodology and results, including uncertainty, of the ERIN analyses. Finally, I describe my conclusions.

BACKGROUND

4. ERIN is an industry leader in risk management and applying reliability and performance-based technologies to decision processes for clients in power, process, and manufacturing industries worldwide. ERIN has extensive experience in the application of risk and reliability analysis techniques to various situations and activities at nuclear power plants. ERIN personnel have been involved in numerous risk analysis projects performed since WASH-1400, "The Reactor Safety Study," in 1975. ERIN personnel managed, directed and performed the first commercial risk assessment project submitted to the NRC after WASH-1400. ERIN personnel developed the IPE methodology for boiling water reactor ("BWR") plants and assisted the Electric Power Research Institute in demonstrating this

method. ERIN personnel have worked with all of the vendor owners' groups to develop the PRA Peer Review Programs and have participated in essentially all of the PRA Peer Reviews which have used the NEI PRA Peer Review Process Guidelines (or their predecessor, the BWROG PRA Peer Review Certification Guidelines) that have been completed or scheduled to date at United States nuclear power plants. ERIN recently performed a Probabilistic Risk Assessment ("PRA") study for the Nuclear Energy Institute ("NEI") of a spectrum of spent fuel pool accident sequences as part of the NEI effort to participate in the Nuclear Regulatory Commission's ("NRC") consideration of spent fuel storage at decommissioned nuclear plants. ERIN is actively involved in the ASME Committees which are developing the PRA standard.

5. ERIN was retained by CP&L's counsel to provide a best-estimate probabilistic assessment analysis of the sequence of events described in the Board's Order (the "Postulated Sequence"). This analysis was to include not only internal events as modeled in the Harris updated Probabilistic Safety Assessment ("PSA") model, but also sensitivity analyses of the scenario frequency to other possible initiating events, including postulated internal fires and seismic events. The analysis was also to consider the sensitivity of the results to core damage events during shutdown conditions. As part of this evaluation, ERIN was asked to perform an independent peer review of the existing Harris updated Level 1 and Level 2 PSA for internal events.

6. My role was to lead and manage a qualified team to perform a best estimate risk assessment analysis of the Postulated Sequence. In this role, I formed a team of experts to examine the spectrum of potential severe accident challenges that could result in core damage and a containment failure or bypass. The analysis was then extended to examine

the impact of these challenges on spent fuel pool cooling. This evaluation used a probabilistic framework that relied on extensive deterministic calculations performed by CP&L and ERIN personnel to characterize equipment survivability, personnel access, and accident sequence timing. The results of the tasks were to determine the frequency of accident sequences that result in uncovering spent fuel in spent fuel pools C and D at the Harris Plant. I also gave sworn testimony in the form of a deposition in this proceeding on October 20, 2000.

7. ERIN was retained to perform this task in mid-August and was given a deadline to complete the analysis and prepare a final report in time to support the November 20, 2000, deadline for a submittal to the Board. To meet this schedule, ERIN dedicated 13 professionals to assist in the work required to perform the analysis. Key members of the team included Karl Fleming, ERIN Vice President PSA Technology, who has over 30 years experience in nuclear safety and PSA, and Douglas True, ERIN Senior Vice President Safety and Reliability Services, who has been an industry leader in the application of PSA technology to practical issues. Mr. Fleming and a small team of expert PSA analysts performed the independent peer review of the Harris Level 1 and Level 2 PSA. Mr. Fleming and Mr. True provided a peer review of the best-estimate risk assessment analysis and conclusions for the spent fuel pool probabilistic analysis. The project manager for the engagement was Jeff Gabor, ERIN Manager, Operations and Technical Solutions, and an expert in thermal hydraulic analyses, who modeled the postulated radionuclide releases from the initiating severe reactor accidents with containment failure or bypass. The resumes of the ERIN team members, including Messrs. Fleming, True and Gabor, are included as Attachment B. The total effort by

ERIN personnel dedicated to this analysis during the period between mid-August and mid-November exceeded 2000 hours of professional time.

8. CP&L personnel provided invaluable assistance in connection with ERIN's analysis. CP&L staff provided detailed calculations (including the Level 1 and Level 2 Harris PSA), system descriptions, interviews with operating personnel, and procedure interpretations. ERIN personnel made multiple Harris site visits to confirm the as-built design of certain key Harris buildings, systems and components. CP&L personnel performed an owner's review of the draft analysis to ensure accuracy of the Harris site specific information.

9. The results of ERIN's best estimate risk assessment analysis of the Postulated Sequence are described in detail in a report entitled "Technical Input for Use in the Matter of Shearon Harris Spent Fuel Pool before the Atomic Safety and Licensing Board," dated November 15, 2000 ("ERIN Report"), which is Attachment C to my Affidavit. The ERIN Report describes the methodology that was used, the results of ERIN's review of the Harris PSA and IPEEE, the details, the results and sensitivities of the probabilistic assessment, and our conclusions. The information in the ERIN Report is true and accurate to the best of my knowledge and belief. This affidavit provides only a brief overview and context of the information and results set forth in the ERIN Report.

PROBABILISTIC SAFETY ASSESSMENT TECHNIQUES

10. The analytical methodologies chosen to determine the best estimate overall probability of the Postulated Sequence are based on PSA techniques that have been developed in the nuclear and aerospace industries to assess the frequency and risks of accidents. The methodology has significantly evolved over the past 10 years in the nuclear industry, building on the methods, data, and approaches used in the NRC's mandated IPE process.

The current PSA methods are judged to be significantly improved beyond those used in the IPE process. Updated plant PSA models, such as the Harris PSA, are more realistic than the IPE, having incorporated advances in technology, plant specific data, computer code improvements, and additional model level of detail. In recognition of these improvements in the technology, the NRC has undertaken an update of the regulatory process to make use of the risk information made available by these state-of-the-technology models. The NRC Safety Goal Policy Statement and Regulatory Guides 1.174 and 1.177 are all examples of this revised process for risk informed regulation. The methodology used by ERIN is described in detail in Section 2.0 of the ERIN Report.

11. The PSA technology used in the Harris PSA and the assessment of the Postulated Sequence has built on the methodology developed for WASH-1400, refined over the period 1975 to 1985 by the industry and the NRC, and further improved in the NRC sponsored application of risk assessment in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" in 1991. Therefore, there has been extensive development that has occurred to enhance the PSA technology. With the advent of risk-informed regulation, these improvements have been carried even beyond the state of the technology in the NUREG-1150 approaches. This can be seen in the level of detail being incorporated in the proposed ASME PRA Standard (Revision 12) which has been considered for public distribution and comment. Mr. Karl Fleming and I have been intimately involved in the development of the ASME PRA Standard and therefore are well aware of the state of the technology expected for PSA applications in the regulatory arena. The Harris PSA has been evaluated by Mr. Fleming's expert team of analysts using the same PRA Peer Review process cited by the NRC in Regulatory Guide

1.174 and in the Revision 12 of the ASME PRA standard. The results indicate that the Harris PSA is fully capable of providing a best estimate frequency for internal events or Steps 1 and 2, *i.e.*, input to the Postulated Sequence.

12. I have reviewed the probabilistic estimates contained in NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, 'Beyond Design Basis Accidents in Spent Fuel Pools'" (1989). It is my opinion that the mean value of $2x10^{-6}$ per reactor year as the frequency of a zircaloy cladding exothermic oxidation reaction resulting from the loss of water from a spent fuel pool (referred to by the Board) is not relevant to analyzing the Postulated Sequence. My conclusion is based on the differences in the initiators considered in NUREG-1353 versus those to be addressed in the Postulated Sequence (*e.g.*, structural failure of the pool due to missiles, aircraft crashes, and heavy load drops; reactor cavity and transfer gate pneumatic seal failures are outside of the Postulated Sequence, which specifies loss of water in the spent fuel pool by evaporation (Step 6)). In addition, NUREG-1353 concludes that seismic events contribute 90 to 95% of the spent fuel pool damage frequency. In the Postulated Sequence, the contribution of seismic events must be limited to only those events that cause a partial loss of pool water (*i.e.*, events that cause a loss of cooling and makeup, but that do not damage the pool sufficiently to cause complete drainage of the pool). While the NUREG-1353 best estimate value of 6.0×10^{-8} per reactor year for loss of spent fuel pool cooling and makeup due to seismic events is not inconsistent with the ERIN results, the NUREG-1353 value includes an unspecified contribution from beyond design basis seismic induced draining of the spent fuel pool, which is not applicable to the Postulated Sequence.

13. To the extent possible, site specific analyses and information from the Harris PSA and IPEEE were used for this probabilistic analysis. They were only a starting point because they do not address loss of spent fuel pool cooling nor a self-sustaining exothermic oxidation reaction in the spent fuel pool. The Harris PSA (Level 1 and Level 2 Internal Events) was subjected to an independent peer review process as part of this evaluation. The independent peer review determined that the Harris PSA is robust and has a significant level of detail that is fully supportive of the proposed application. The independent peer review also found that the Harris PSA is capable of quantifying core damage frequency and large early release frequency and reasonably reflects the as-built and as-operated plant. The Harris PSA is consistent with accepted PSA practices, in terms of the scope and level of detail for internal events. Its quantification is quite detailed and the results are consistent with those in PSAs of pressurized water reactors of similar designs. The Harris PSA demonstrates that the plant meets the NRC Safety Goals and their subsidiary objectives (*i.e.*, Core Damage Frequency and Large Early Release Frequency). In addition, there are no unusual contributors to core damage frequency or containment failure. It was noted, however, that the interfacing systems LOCA analysis in the Harris PSA was overly conservative and needed to be updated if a best estimate set of frequencies were to be used as part of the Postulated Sequence requested by the Board. This update was performed and the results included in the best estimate calculation of the Postulated Sequence. The Harris PSA and its reviews are described in Section 3.0 of the ERIN Report.

14. CP&L also had completed an Independent Plant External Events Evaluation ("IPEEE") pursuant to Generic Letter 88-20, Supplement 4, that has been accepted by the NRC. The

Harris IPEEE considered 1) seismic risk, 2) internal fire risk, and 3) risk from other external events (*e.g.*, high winds, tornadoes, and nearby facility accidents). On the basis of the IPEEE review, the NRC staff concluded that CP&L's IPEEE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities and, therefore, that the Harris IPEEE has met the intent of Generic Letter 88-20, Supplement 4. ERIN relied on certain aspects of the Harris-specific information in the Harris IPEEE in evaluating the frequency contributors from fire initiating events, seismic events, and in screening other external events. The use of the Harris IPEEE is described in Section 3.0 of the ERIN Report.

ANALYSIS OF THE POSTULATED SEQUENCE AT THE HARRIS PLANT

15. The Postulated Sequence begins with a very low probability, beyond design basis, degraded core severe accident event at the Harris reactor (Step 1) with failure of the large dry Harris containment or bypass of the containment (Step 2). These two steps were evaluated using probabilistic risk assessment techniques. For the internal events (*i.e.*, events initiated at Harris such as steam generator tube rupture, loss of coolant accident, station blackout, etc.), the contribution to steps 1 and 2 was taken from the Harris PSA plus the updated ISLOCA analysis that was used to obtain a best estimate of the ISLOCA contribution (*i.e.*, to be consistent with the best estimate frequencies obtained in other parts of the Harris PSA). A sensitivity analysis was also performed to evaluate the potential contribution from fire initiating events, seismic events, and shutdown (rather than at-power) events. The Harris IPEEE was used for Harris specific information regarding the fire events and seismic events, as well as screening other external events. Generic industry data developed by the NRC was used to evaluate the shutdown events.

The accident sequence frequency development for each of the contributors are described in Section 4.0 of the ERIN Report.

16. Step 3 of the Postulated Sequence requires the loss of spent fuel cooling and makeup systems to the Harris spent fuel pools. A probabilistic evaluation was performed of the loss of all cooling and makeup systems, which included Fuel Pool Cooling and Cleanup System ("FPCCS") cooling failures (random, human error, test/maintenance and common cause); FPCCS cooling support system failures, including support system failures that may have contributed to the core damage accident sequence initiating event; and consequential failures of FPCCS cooling or its support systems due to adverse environmental conditions caused by containment failure or bypass. The addition of a separate, redundant FPCCS for spent fuel pools C and D reduces the probability of this event. It also provides alternate makeup paths in the event the FPCCS cannot be restarted. The Harris Fuel Handling Building ("FHB"), spent fuel pools ("SFP"), FPCCS, and other associated systems are described in Appendix A of the ERIN Report. The probabilistic evaluation of the loss of all FPCCS and makeup systems is discussed in Section 4.0 and Appendices A, C, D and E of the ERIN Report.

17. Step 4 of the Postulated Sequence assumes extreme radiation doses precluding personnel access and Step 5 assumes an inability to restart any pool cooling or makeup systems due to extreme radiation doses. For all sequences identified in Steps 1 and 2, radiation levels were calculated for specific areas in which access would be necessary in order to respond to Step 3. Consideration of the adverse impacts of extreme radiation and extreme conditions of steam or heat from the containment failure, the containment bypass, or boiling of the spent fuel pools on both personnel access and equipment survivability was

included and modeled in the probabilistic assessment. An extensive effort was performed to characterize the plant conditions, especially in the critical buildings, the Reactor Auxiliary Building and the Fuel Handling Building -- *i.e.*, the areas containing critical equipment. A deterministic evaluation of the plant thermal hydraulic response and the transport of radionuclides was performed to characterize issues such as access, timing, and adverse conditions on equipment. The method applied utilized the Modular Accident Analysis Program ("MAAP") computer model (see ERIN Report, Appendix E) to model the transient flow conditions due to the postulated accident sequences and containment failure modes. MAAP is the most widely used severe accident analysis code and has been reviewed extensively by the NRC and its contractors in support of Generic Letter 88-20. MAAP includes best estimate models to represent accident progression beginning with normal operation and extending to potential radionuclide release to the environment. The Harris-specific MAAP calculations also yielded the fission product release, transport, and deposition effects in the RAB and FHB. These results provided the input to the CP&L dose assessment to calculate the dose rates for areas to assess equipment survivability and personnel access. This deterministic analysis and its use in the probabilistic assessment is described in the ERIN Report, Section 4.0 and Appendices A, C and E.

18. Step 6 requires the loss of most or all spent fuel pool water through evaporation and the inability to restart FPCCS cooling or inject makeup water before the fuel is uncovered in the spent fuel pools. To evaluate this step, a deterministic evaluation was performed that included a calculation by CP&L of the time to boil and evaporate the water in the spent fuel pool after loss of FPCCS cooling. (The results of that calculation are set forth in

Section 2.0 of the ERIN Report and in the Affidavit of R. Steven Edwards). With a worst case heat load in spent fuel pools A and B (immediately after refueling), CP&L calculated that it would take over 8 days after all FPCCS cooling and makeup is lost to uncover the fuel. (It would take almost 100 days for the water in spent fuel pools C and D to evaporate with the 1.0 MBTU heat load permitted by the license amendment request.) Based on the ability to restore spent fuel pool level and prevent uncovering of any spent fuel with the most limiting make up sources credited, ERIN conservatively assumed access to critical plant areas to restore FPCCS cooling or makeup to the spent fuel pools to be required within 96 hours of the loss of spent fuel pool cooling.

19. The Harris FHB was constructed to accommodate a four unit site. The size and compartmentalization of the FHB influences its accident response. In addition, there are a substantial number of systems and pathways for establishing water makeup to the spent fuel pools. The addition of a redundant FPCCS for spent fuel pools C and D provides additional pathways for injection of makeup water to the spent fuel pools. The various makeup water pathways are described in the ERIN Report, Appendix A, and in the Affidavit of Eric McCartney. ERIN determined that access to at least one makeup water lineup was possible within 96 hours for all of the initiating accident sequences with containment failure or bypass. *See* ERIN Report at Appendix E.

20. Step 7, initiation of an exothermic oxidation reaction in spent fuel pools C and D, was not evaluated. A rigorous probabilistic assessment would require the development of new thermal hydraulic models. There was insufficient time to undertake such development work. Furthermore, the probability of reaching Step 7 was exceedingly low in any event. In this regard, ERIN took the same approach as the NRC in NUREG-1353 and assumed

that the conditional probability of a self-sustaining exothermic oxidation reaction was 1.0 for purpose of the best estimate analysis of the probability of the Postulated Sequence. This appears to be a conservative assumption based on a review of the literature reporting on the critical cladding oxidation temperature for a self-sustaining exothermic oxidation reaction of the zircaloy fuel cladding, the age and heat rate of the spent fuel that will be stored in Harris spent fuel pools C and D, and the likely ability of air to remove decay heat from old, cold spent fuel. The results of the literature review and specific information on the spent fuel to be stored in Harris spent fuel pools C and D is described in the Affidavit of Robert K. Kunita.

21. The results of ERIN's probabilistic analysis are described in Section 5.0 of the ERIN Report and are summarized in Table 5-1, which for convenience is reprinted in this Affidavit. The first column in Table 5-1 expresses the results of the calculation of the annual core damage frequency for severe accident event initiators with containment failure or bypass. The second column provides the results of the probabilistic assessment of Steps 1 through 6 for each severe accident initiator. The results of the internal events initiated sequences indicate that the loss of effective spent fuel pool water cooling has a best estimate annual occurrence probability of 2.65E-08 (less than three chances in one hundred million). Assuming conservatively that the probability of a self-sustaining exothermic oxidation reaction with the loss of effective spent fuel cooling and water inventory is 1.0, the best estimate answer to the Board's question 1 is 2.65E-08. As Table 5-1 shows, the external events and shutdown events were also evaluated to determine whether these events alter the conclusion reached based on the internal events assessment. It is recognized that the uncertainties associated with these events are greater

Table 5-1

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

Event	Description of Events that Involve Initiators, Core Damage, and Containment Failure or Bypass	Input CDF from FT Quantification[1]	Output from SFPAET[2]
Internal Events			
ISLOCA	INTERFACING SYSTEMS LOCA	9.97E-09	7.44E-10
LG-SGTR	LARGE STEAM GENERATOR TUBE RUPTURE	1.57E-06	3.44E-09
SM-SGTR	SMALL STEAM GENERATOR TUBE RUPTURE	1.51E-06	3.31E-09
LG-ISOL	LARGE ISOLATION FAILURE	7.59E-08	9.77E-10
SM-ISOL	SMALL ISOLATION FAILURE	1.88E-07	2.59E-09
EARLY	EARLY CONTAINMENT FAILURE	3.14E-08	1.15E-09
LATE	LATE CONTAINMENT FAILURE	4.28E-06	1.43E-08
Total Internal Events Contribution		7.67E-06	2.65E-08
Fire Induced Events			
EARLY	EARLY CONTAINMENT FAILURE	2.95E-09	7.98E-11
LATE	LATE CONTAINMENT FAILURE	9.77E-07	2.86E-09
Total Fire Events Contribution		9.80E-07	2.94E-09
Total Seismic Contribution		-	8.65E-08
Shutdown Events			
SHDN	SHUTDOWN WITH CONTAINMENT BYPASS	5.00E-07	1.45E-08

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

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

than those in the dominant internal events analyses. Consequently, several conservatisms were incorporated into the modeling, which produced inflated point estimate values. As indicated in Table 5-1, the point estimate annualized probability for the total fire events contribution was 2.94E-09 (or an order of magnitude less than the total internal events contribution). The total seismic contribution was based on data with large uncertainties, an approximate model, and greater conservatisms. Furthermore, it was difficult to analyze in the context of the Postulated Sequence because a seismic event less than the design basis earthquake cannot be an initiator of Steps 1 and 2, and a seismic event sufficient to cause breach of the spent fuel pools is outside of the Postulated Sequence (because the loss of cooling to the spent fuel must be by evaporation (Step 6) and not draindown of the spent fuel pools from a breach of pool integrity). While the point estimate annualized probability contribution due to seismic initiated events of 8.65E-08 is higher than for internal events, it is judged not to alter the conclusions reached based on the internal events analysis. Finally, the core damage frequency associated with internal events during shutdown refueling outages was estimated to be on the same order of magnitude as that calculated for power operation. This determination was based on generic studies rather than site specific PSA, because shutdown internal events are not included in the Harris PSA. In any event, the generic results for pressurized water reactors are judged applicable to Harris. The use of these core damage results and an assessment of the containment failure or bypass led to an assessment of the spent fuel pool Postulated Sequence that is consistent with the estimate of the probability reached for the dominant internal events.

22. As requested by the Board, the analysis performed was a best estimate analysis using the best available technical information representative of Harris. The best estimate is used for decision making because the use of upper bounds (or lower bounds) may introduce biases into the decision making process that are not properly characterized, *i.e.*, the biases may be unevenly applied (widely varying levels of conservatism) with the resulting upper bound yielding a distortion of the importance of individual components of the analysis and potentially of the overall results. Such biases could then lead to improper decisions regarding the importance of individual elements of the analysis. It may also lead to the improper allocation of resources to address conditions or postulated events that have been "conservatively" treated in an upper bound evaluation. The best estimate of the Postulated Sequence can be further understood in the context of the uncertainties surrounding the quantification.

23. The NRC, its contractors, and the industry have committed substantial efforts to the understanding of uncertainties in nuclear power plant risk analyses. These efforts have led to methods development, understanding of the contributors to the uncertainty distributions, and the identification of alternative ways to provide decision makers with effective ways of characterizing the risk spectrum. The evolving consensus in the industry on the treatment of uncertainties is that the use of focused sensitivity evaluations to characterize the change in the results as a function of changes in the inputs provides a physically meaningful method of conveying the degree of uncertainty associated with the analysis. Therefore, sensitivity cases were developed in connection with this analysis that portray the changes in the Postulated Sequence frequency if input variations occur. The results of these sensitivity studies are described in Section 5.0 of the ERIN Report.

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

- A substantial fraction of the containment does not interface with the RAB. However, the dominant failure modes for containment appear to be at locations where RAB impacts cannot be ruled out. Therefore, all containment failures are assumed to impact the RAB environment.

- The spent fuel pool boil off time is taken to be the minimum it can be, given the plant configuration and the times at which freshly discharged spent fuel could be introduced into the spent fuel pools.

- The seismic evaluation is subject to large uncertainty and is believed to be a conservative bound because of the assumptions of :

  -- Loss of site power with no opportunity for recovery

  -- Complete dependence of failures of similar components

  -- The early containment failure probability used in the seismic evaluation is the worst case found for any plant damage state. This is likely too conservative when applied to the seismic initiated sequences involving station blackout.

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

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

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

- The treatment of containment isolation failures into the RAB in the base model assumes that access to the RAB and FHB operating deck (286' Elevation) is not available. This is conservative relative to the deterministic calculations performed to support accessibility. The deterministic calculations indicate that the FHB is not affected by the Containment Isolation failure.

- The probability of a self-sustaining exothermic oxidation reaction in the event of fuel uncovery (Step 7) was assumed to be 1.0. A best estimate probability would require a detailed heat balance evaluation of the spent fuel pool. The qualitative analysis of the temperatures that might be reached in SFPs C and D given the heat rates of the fuel that would be stored there (particularly if limited to 1.0 MBTU/hr) was performed by CP&L. These assessments by CP&L suggest that the conditional probability of Step 7 would be less than 1.0.

CONCLUSIONS

25. I conclude that the Postulated Sequence of seven events described in the Board's Order has a best estimate overall annualized probability of occurrence at Harris of 2.65E-08. The bases for my conclusion and my confidence in the results are: (1) the quality of the Harris PSA and IPEEE; (2) the quantity of Harris-specific information incorporated in the analyses; (3) the breadth, qualifications, and technical skills of the team performing the work; (4) the quality and capabilities of the technical tools employed; (5) the quality and extent of internal, owner, and independent reviews; (6) the degree of correlation with similar analyses; and (7) the extensive set of sensitivity studies used to explore the uncertainty bands associated with the quantification. Indeed, the analysis still has a number of conservatisms which suggest that a true best estimate analysis would result in a probability that is even lower. For all these reasons, it is my professional opinion that the Postulated Sequence is so unlikely that it would not be reasonable to consider it further in decision-making regarding risks posed by the Harris spent fuel pools. The annual occurrence probability of the Postulated Sequence is, for example, considerably less than the probability of the recurrence of the ice age or the probability of a meteor strike creating world-wide havoc. (*See* ERIN Report, Section 6.0 and Appendix B).

I declare under penalty of perjury that the foregoing is true and correct.

Executed on November 15, 2000.

E T Burns

Edward T. Burns

Subscribed and sworn to before me
this 15 day of November 2000.

Maryellen D Long

My Commission expires: 4-20-2002

OFFICIAL SEAL
MARYELLEN D LONG
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 04/20/02

Document #: 1039223 v.3

Dr. Edward T. Burns

Vice President, BWR Risk and Reliability

AREAS OF EXPERTISE

- ***Severe Accident Mitigation Evaluation***
- ***PRA Senior Consultant***
- ***Consequence Analysis***
- ***Emergency Procedure Consultant***
- ***Shutdown Operations Analyst***
- ***IPE Methodology Developer***
- ***BWR PSA Peer Review Certification Developer***

EDUCATION

Ph.D., Nuclear Engineering, Rensselaer Polytechnic Institute, Troy, New York

M.S., Nuclear Engineering, Rensselaer Polytechnic Institute, Troy, New York

B.S., Engineering Science, Rensselaer Polytechnic Institute, Troy, New York

WORK EXPERIENCE SUMMARY

Dr. Burns, Vice President and General Manager of BWR Technology for ERIN Engineering, is a nuclear engineer with considerable experience in the application of probabilistic risk assessment technology to the solution of engineering problems. Dr. Burns has over 25 years of experience in the field of probabilistic risk assessment, severe accident analysis, and emergency procedure examination. Dr. Burns has also assisted the BWROG, Commonwealth Edison Company, Philadelphia Electric Company, Long Island Lighting, and Duane Arnold Energy Center in the application of probabilistic risk assessment (PRA) for efficient and safe implementation of hardware and procedure changes.

WORK EXPERIENCE

Vice President and General Manager of BWR Technology at **ERIN Engineering and Research, Inc.** Dr. Burns continues to work closely with utilities to provide workable engineering solutions to current problems. One valuable tool useful in the approach has been PRA techniques. The following are some of his recent activities:

- Manager and lead technical analyst for the **LaSalle** and **Quad Cities** PSA CAFTA update (1998-2000)
- Manager and lead analyst for the HRA development for **LaSalle**, **Quad Cities**, and **Dresden** (1999-2000)
- Manager and lead analyst for the PSA Application to Relax the Diesel Generator AOT to 14 days (2000)
- Manager and lead analyst for the Internal Flood analysis for **LaSalle** (2000)
- Manager and lead analyst in the risk assessment of a decommissioning plant (1999)
- Manager of SAMA projects for 2 BWRs and 1 PWR
- Led or participated in 19 BWR PSA Peer Review Certifications in 1996 - 2000
- Develop System Notebooks for PSA Applications at three BWRs using design basis information
- Extensive experience on the procedures and strategies involved in the Severe Accident Guidelines (SAG) developed for the BWROG
- Manager and lead technical analyst of the **Duane Arnold** Level 1 and 2 PRA Technical Support for response to the Severe Accident Policy Statement (1991 - 1995)
- Manager and lead technical analyst of the **Fermi** Mark I Level 2 IPE using RISKMAN for response to the Severe Accident Policy Statement (1993)

- Manager and lead technical analyst of the **Limerick** Mark II Level 2 IPE for response to the Severe Accident Policy Statement (1992)
- Manager and lead technical analyst of the **Nine Mile Point Unit 1** and **Unit 2** Level 2 IPE for response to the Severe Accident Policy Statement (1992 - 1994)
- Manager and lead technical analyst of the **Peach Bottom** (Mark I) Level 2 IPE for response to the Severe Accident Policy Statement (1991)
- Manager and lead technical analyst of the **Vermont Yankee** (Mark I) Level 2 IPE for response to the Severe Accident Policy Statement (1992 - 1994)
- Chief analyst of the **Limerick** and **Peach Bottom** HRA evaluation to support both Level 1 and 2IPEs for response to the Severe Accident Policy Statement (1991 - 1994)
- Chief consultant analyst of the **Vermont Yankee** HRA evaluation to support the Level 1 and Level 2 IPEs for response to the Severe Accident Policy Statement
- Advisor to the BWR Shutdown Risk Program for **EPRI** (1993)
- Analyst in assisting **BWROG/NUMARC/EPRI** in the development of accident management guidance (1991 - 1997)
- Review of Garona Level 1 PSA (1996)
- Developed the BWROG PSA Peer Review Certification Process and Guidelines under the auspices of the BWROG
- Technical Reviewer of Level 1 IPEs for:
  - Vermont Yankee
  - Monticello
  - Peach Bottom
  - Limerick
  - Brunswick
  - Hope Creek
  - Nine Mile Point Unit 2
  - Pilgrim
  - Cooper
  - River Bend
  - Grand Gulf
  - Oyster Creek
  - Dresden
  - Perry
  - Cooper
  - WNP2
  - Millstone Pt. 1
  - LaSalle
- Technical Reviewer of the **Perry** Level 2 IPE
- BWROG review of the EPRI Technical Basis Report for BWR Severe Accidents
- Manager and lead analyst for the Human Reliability Assessment
  - Peach Bottom
  - Limerick
  - Duane Arnold
  - Nine Mile Point 2
  - Vermont Yankee

- BWROG Developer (along with S. Taggart Rogers (OEI)) of the BWROG Accident Management Guidance and Associated EPG changes.

- Developer of the EPRI ISLOCA Evaluation Methodology which included operating experience reviews of related events

- Manager of ISLOCA applications to **Trojan** and at **Hope Creek** plants

- Assisted BWROG and GE in categorizing insights from the Severe Accident Applicability Report of Rev. 4 EPGs.

- Senior Engineering advisor to EPRI ALWR Program on:
  - Source term evaluation
  - Containment failure modes
  - Procedural impacts.

Director of BWR Technology for **TENERA, L.P.** (formerly **Delian** Corporation).

- BWROG - Review of the Emergency Procedure Guidelines from a severe accident perspective to assure maximum effectiveness of the procedures for accident management

- BWROG - Mark I Containment Safety Assessment

- **The BWR Owners' Group** - Review of PRA applications in the Industry Degraded Core Rulemaking (IDCOR)

- **Yankee Atomic Electric Corporation** - Consultant to **YAEC** and **Vermont Yankee** on their containment safety study

- **TVA** - Lead engineer on the containment safety study of the **Browns Ferry** Plant and technical reviewer of the **BFN** IPE

- **Commonwealth Edison** - Principal technical reviewer of the **Dresden** IPE

- **Northern States Power Company** - Principal technical reviewer of the **Monticello** IPE

- **BWR Owners' Group** - Principal investigator and project manager of the Severe Accident Applicability Review of the BWROG Emergency Procedures Guidelines (Revision 4)

- **Northeast Utilities** - Reviewer and consultant to **NUSCo**. on the **Millstone Point I** PRA and its application to the Integrated Safety Assessment Program (ISAP)

- **Boston Edison** - Lead technical engineer in developing the first detailed Mark I containment safety study using probabilistic techniques

- **BWROG** - Technical advisor for the **BWROG** generic Mark I containment integrity study

- **Hope Creek** - Technical advisor to **PSE&G** on the **Hope Creek** Level 1 PRA.

- Probabilistic evaluation of the effectiveness of containment venting for a Mark I and a Mark III in Spain

- BWROG - Developed responses to NRC questions regarding the efficacy of including containment venting in the emergency procedures

- IDCOR - The development of an Individual Plant Evaluation Method (IPEM) for BWRs to respond to the NRC Severe Accident Policy Statement

Assistant to the Vice President at **Science Applications, Inc.** Primary activities included:

- Lead engineer in the Severe Accident Mitigation Design Assessment (SAMDA) to support Limerick licensing in ACRS and NRC interaction

- Manager of the Shoreham Nuclear Power Station PRA for Long Island Lighting Company

- Lead engineer on the Limerick Generating Station PRA for Philadelphia Electric Company

- Lead engineer on the evaluation of risk reduction potential due to ATWS mitigation features for LWR owners group.

- Long Island Lighting Company - Application of PRA to the Shoreham facility. Provided both the project management and technical lead on the PRA for Shoreham

- **Philadelphia Electric Company** - Application of PRA to the **Limerick** and **Peach Bottom** facilities. Provided the technical lead for the 1981 **Limerick** PRA and a peer review role to the Level 1 **Peach Bottom** IPE

Engineer at the Department of Energy, Naval Reactors Division. Responsibilities included:

- Responsible for detailed review of the core mechanical design, balancing the thermal performance and lifetime versus the mechanical design, and establishing mechanical and hydraulic test programs

- Responsible for design review of laboratory thermal hydraulic testing to support qualification of computer design codes for reactor cores and the research development for the minimization of flow-induced vibrations.

SECURITY CLEARANCE:

U.S. Citizen

LICENSES/REGISTRATIONS/PROFESSIONAL SOCIETIES

American Nuclear Society

PUBLICATIONS

Available Upon Request

Dr. Burns' engineering experience was gained through employment with the following companies:

- ERIN Engineering and Research, Inc.
- TENERA, L.P.
- Science Applications, Inc.
- Department of Energy

List of Publications Authored by Dr. Edward T. Burns
Within Proceeding Ten Years

1. E.T. Burns and L.K. Lee., "Uncertainty: Can Risk Informed Regulation Survive the Challenge", PSA '96 Proceedings, American Nuclear Society Transactions, La Grange Park, IL, 1996.

2. E.T. Burns, G.A. Krueger, and R.A. Hill, " Assessment of PRA Quality", PSA '99, American Nuclear Society Transactions, Washington, D.C., August 1999, pg. 217.

3. E.T. Burns, T.P. Mairs, J.R. Gabor, et al., "BWR Accident Management Insights for Containment Flooding", PSA '93, American Nuclear Society Transactions, La Grange Park, IL, January 1993.

4. E.T. Burns, V.M. Andersen, J.R. Gabor, et al., "Level 2 Individual Plant Examination", PSA '93, American Nuclear Society Transactions, La Grange Park, IL, January 1993.

5. E.T. Burns, J.R. Gabor, T.P. Mairs, et al., "Accident Management Guidance Process for Technical Support Center (TSC)", PSA '93, American Nuclear Society Transactions, La Grange Park, IL, January 1993.

6. E.T. Burns, C.D. Sellers, and G.A. Krueger, "Risk Informed Decisions: 1st Intervals Using a Blended Approach", PSA '96 Proceedings, American Nuclear Society Transactions, LA Grange Park, IL, 1996.

7. E. T. Burns , D.E. Macleod, and L.K. Lee, "HRA Tailored for Risk Informed Decisions for Shutdown Safety", PSA '96 Proceedings, American Nuclear Society Transactions, La Grange Park, IL, 1996.

8. E.T. Burns and V.M. Andersen, "Risk Informed Decisions: Incorporating IPEEE Analyses into the Living PSA", PSA '96 Proceedings, American Nuclear Society Transactions, La Grange Park, IL, 1996.

9. E.T. Burns and V.M. Andersen, "Rational Approach and Radionuclide Release Characterization" 1994 Annual Meeting, Vol. 70 (pg.266), American Nuclear Society Transactions, New Orleans, Louisiana, June 1994.

10. E.T. Burns, J.R. Gabor, and T.P. Mairs, "Strategies for Operator Response in Mitigating Loss of Containment Heat Removal Accident Scenarios", Vol. 68, Part A (pg.281), American Nuclear Society Transactions, Inc., La Grange Park, IL, June 1993.

List of Publications Authored by Dr. Edward T. Burns Within Proceeding Ten Years (continued)

11. E.T. Burns, D.E. True, K.N. Fleming, et al., "The Importance of Utilizing a Blended Approach in Regulatory Applications of Probabilistic Safety Assessment", PSA '96, American Nuclear Society Transactions, La Grange Park, IL, 1996.

12. E.T. Burns, L.K. Lee, and R.F. Kirchner, " Evaluation of Nine Mile Point Unit 2 RFO 3 Using Riskman and ORAM", Safety of Operating Reactors Proceedings, American Nuclear Society Transactions, Seattle, WA, 1995, pg.385.

13. E.T. Burns and V.M. Andersen, "Approach to Enhancing PSA Pedigree", Safety of Operating Reactors Proceedings, American Nuclear Society Transactions, Seattle, WA, 1995, pg. 579.

14. E.T. Burns, L.K. Lee, and H. Ahn, "Reliability and Risk Assessment", 1994 Annual Meetings, Vol. 70 (pg. 228), American Nuclear Society Transactions, New Orleans, Louisiana, June 1994.

15. E.T. Burns, et al., "BWROG PSA Peer Review Certification Process", BWROG 97-01, January 1997.

16. E.T. Burns, et al., "A Review of Draft NRC Staff Report: Draft Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants", NEI, September 1999.

17. E.T. Burns, et al., "Quad Cities PRA", 1999.

ERIN Team Members

Karl N. Fleming

Douglas E. True

Jeff R. Gabor

Vincent M. Andersen

David A. Bidwell

Thomas A. Daniels

Jan F. Grobbelaar

Lawrence K. Lee

Bengt O. Y. Lydell

Donald E. MacLeod

Leo B. Shanley III

Donald E. Vanover

Karl N. Fleming

Vice President, PSA Technology

AREAS OF EXPERTISE

- ***Level 1, 2, and 3 PSA***
- ***Risk Informed Inservice Inspection***
- ***Common Cause Failure Analysis***
- ***Technical Specification Optimization***
- ***External Events and Internal Fire PRA***
- ***Plant Availability and Reliability Evaluations***
- ***Project and Resource Management***
- ***PSA Applications***

MAJOR PSA PROJECTS ON

- ***Beznau***
- ***Gösgen***
- ***Beaver Valley 1 and 2***
- ***Seabrook Station***
- ***South Texas Project 1 and 2***
- ***Salem PSAs***
- ***Technical Specifications***
- ***Piping In-Service Inspection***

WORK EXPERIENCE SUMMARY

Mr. Fleming is Vice President of ERIN's San Diego Office which is playing a key role in the Safety and Reliability services business area. Mr. Fleming is widely recognized as an expert in probabilistic risk, reliability, and safety evaluations of industrial and nuclear facilities. In his 30 years of experience in nuclear safety and Probabilistic Risk Assessment (PRA), he has directed more than a dozen large scale and full scope Probabilistic Safety Assessment (PSA) projects in the U.S., Western Europe, and Eastern Asia which were responsible for resolving major safety issues and large cost savings for his clients. He is deeply knowledgeable of all facets of Level 1, 2, and 3 PSAs and has made major contributions to the development of PSA technology and the expansion of this technology to the treatment of external events and accidents initiated in shutdown modes. Mr. Fleming is well known for his contributions to the state of the art in the PSA evaluations of common cause failures, internal fires, interfacing system LOCAs, technical specification optimization, emergency planning, accident management, and maintenance program prioritization. He was the principal author of the industry standard for common cause analysis (NUREG/CR-4780) and a contributing author to the EPRI PSA Applications Guide. He has made important contributions to the development and applications of state of the art reliability and availability assessment techniques for piping systems and power plants.

WORK EXPERIENCE

Mr. Fleming is Vice President of **ERIN Engineering and Research, Inc.** in charge of the San Diego Office which performs risk and reliability evaluations of nuclear and non-nuclear plants in the U.S., Europe, and other international markets. He was the co-author of the EPRI Risk Informed Inservice Inspection (RI-ISI) Topical Report (TR-112657) and played a key role in developing the risk aspects of this approach to RI-ISI that were essential to obtaining NRC acceptance of the EPRI RI-ISI method. Mr. Fleming led the team who developed the EPRI Markov Model for piping system reliability assessment, described in EPRI TR-110161, and for developing the latest industry estimates for pipe failure rates and rupture frequencies, as described in EPRI TR-111880. Currently, Mr. Fleming is the principal investigator of the Commonwealth Edison RISI project involving ten units at five different sites (Braidwood-1/2, Byron-1/2, Dresden-2/3, LaSalle-1/2 and Quad Cities-1/2).

At ERIN, Mr. Fleming was responsible for applying PSA Technology to nuclear property damage insurance. He was the principal investigator of major ERIN projects in applying PSA technology to risk based in-service inspection, risk based component testing prioritization, and extension of PSA models to assess the risk and availability impact of balance of plant system performance. He was the principal investigator of a comprehensive risk informed inservice inspection evaluation for Class 1 and 2 piping systems at all 10 reactors operated by Commonwealth Edison Co. Mr. Fleming was the project manager of a major PRA update project for the Byron and Braidwood PWR units, and for an integrated reliability assessment for the Lungmen ABWR units in Taiwan. He developed innovative methods for the extension of fault tree analysis to model plant availability and capacity factors. ERIN's

PLANTFORMA™ software for plant availability and reliability modeling and evaluation is based on Mr. Fleming's technical innovations. He was also responsible for developing a new approach for piping system reliability assessment for risk informed in-service inspection programs as well as practical applications of this method for evaluating the risk impacts of changes in the inservice inspection program.

In his most recent position with PLG Inc., he was a Vice President in charge of Nuclear Energy Services. This business unit that was involved in a number of major risk assessment projects for the nuclear utility industry. There, he was responsible for the performance of all risk and safety projects and served as project manager and principal investigator on many specific projects. He directed all business development activities in the areas of risk and safety.

Mr. Fleming was the project manager of the **Beznau**, **Gösgen**, **Beaver Valley**, **Seabrook Station**, **South Texas Project**, and **Salem** PSAs; manager of several applied risk management projects for utilities to enhance design, improve technical specifications, and optimize emergency planning. Principal author of the Seabrook PRA report and management plan. He was also the project manager of several major projects for the **Electric Power Research Institute** (EPRI) on common cause failures, with internationally recognized expertise in this area. He made significant contributions to the development and application of PSA methods for shutdown risk assessment and treatment of phenomenological probabilities in Level 2 PRAs. Made major contributions to the development of the modularized event tree linking method of modeling functional dependencies in PRAs event sequence models. Developed para-metric models for system-level common cause failure analysis, including the beta factor, multiple Greek letter, and basic parameter models. Author of the American Nuclear Society/Institute of Electrical and Electronics Engineers ANS/IEEE PRA Procedures Guide sections on dependent events, fires, and floods, and the NRC/EPRI Procedures Guide on common cause analysis (NUREG/CR-4780).

Responsible for technical review of dependent and external events analysis in the risk methods integration and evaluation program (RMIEP). As Manager, Safety and Reliability Branch of General Atomic Company, was engaged in assessing risk and evaluating reliability of light water reactor systems, radwaste storage facilities, fusion and hybrid (fission-fusion) reactors, and synfuels plants. Evaluated one of the first PRAs of the financial risk of accidents, a study for American Nuclear Insurers on the **Three Mile Island** Unit 2 cleanup operations. In prior positions at General Atomic, was principal investigator of the high temperature gas-cooled reactor risk assessment study. Made major contributions to the development of PRA methodology. Developed the beta factor method of common cause failure analysis and the first risk assessment of accidents caused by internal fires. Author of the STADIC computer program for Monte Carlo error propagation used in PRA uncertainty analysis.

He taught reactor physics, basic physics, and mathematics to nuclear power plant operators, nuclear power plant engineering staff, and officers in charge of training programs at Nuclear Power Plant Operators School, U.S. Army Reactors Group, Fort Belvoir, Virginia.

EDUCATION

M.S. Nuclear Science and Engineering, Carnegie-Mellon University in 1974

B.S. Physics at Penn State University in 1969

SECURITY CLEARANCE

U.S. Citizen

Active DOE "L" Clearance

Inactive DOD Secret Clearance

LICENSES and PROFESSIONAL SOCIETIES

ANS

ASME

ASME BNCS Task Group on PRA

Phi Kappa Phi

Tai Mu Epsilon

Who's Who in Frontier Science and Technology

Who's Who in Frontier Science and Technology (first edition)

Dudley Memorial Scholarship, Pennsylvania State University

Phi Kappa Phi, National Scholastic Honors Society

Sigma Pi Sigma, National Physics Honor Society

Pi Mu Epsilon, National Mathematics Honor Society

American Society of Mechanical Engineers

American Nuclear Society

American Association for the Advancement of Science

Society of Risk Analysis

Referee, IEEE Transactions on Reliability

Author ANS/IPEEE PRA Procedures Guide, NUREG/CR-2300

Co-author of EPRI PSA Applications Guide

Co-author ASME PRA Standard

Session Chairman, Probabilistic Risk Assessment, ANS Winter Meeting, Washington, D.C., 1983

Session Chairman, Dependent Events - Applications and Case Studies, American Nuclear Society Topical Meeting on Probabilistic Risk Assessment, San Francisco, 1985

Session Chairman, Common Cause Failures, Society of Risk Analysis Meeting (PSAM), Beverly Hills, California, February 1991

Member, RMIEP Quality Assurance Review Team 1985-1986

Chairman, Department of Energy Committee for Peer Review of Nuclear Grade Graphite Stress Criteria

Team Leader, U.S. Contribution to International Common Cause Failure-Reliability Benchmark Exercise 1985-1986

Member NUREG-1150 Expert Panel for Front End Issues

Member, U.S. Department of Energy Review Teams for N-Reactor and Savannah River PRAs

Session Chairman, IAEA Workshop on PSA Applications, Budapest, Hungary, September 7-11, 1992

Session Chairman, and Technical Program Committee, PSA '93, Clearwater, Florida, January 1993

Member IAEA Mission on PSA Applications at Nuclear Power Plant, Taejon, Korea, July 10-14, 1995

Member IAEA Mission on NPP Maintenance Optimization, Ljubljana, Slovenia, October 20-24, 1995

Session Chairman and Technical Program Committee, PSA '96, Park City, Utah, September 29-October 3, 1996

Leader, Third party Review Team, EPRI Risk Informed In-service Inspection.

Facilitator, BWROG PSA Certification Review Teams for Fermi 2 and Monticello Plants, Member PSA Certification Review Team for Duane Arnold. Millstone, Diablo Canyon

Technical Program Committee PSA'93, PSA'96, and PSAM4 Conferences

Member ASME BNCS Task Group for Probabilistic Risk Assessment Standards and Lead Author for Sections on Initiating Events, Accident Sequence Definition, Success Criteria and Internal Flooding

Invited Speaker, Advisory Committee on Reactor Safeguards Retreat, Clearwater Florida, January 2000 on the topic of risk informed regulation

PUBLICATIONS

Fleming, Karl N., et al., "Risk Informed Inservice Inspection Evaluation, Final Report, Byron Nuclear Power Plants Units 1 and 2," prepared for ComEd, July, 2000.

Fleming, Karl N., et al., "Risk Informed Inservice Inspection Evaluation, Final Report, Braidwood Nuclear Power Plants Units 1 and 2," prepared for ComEd, June, 2000.

Fleming, Karl N. and Michael V. Frank, "Methods for Treatment of Uncertainties and Dependencies in NASA PRA Applications," Advanced Course in PRA Methods and Applications for NASA Johnson Space Center, Houston, TX, May 10-12, 2000.

Fleming, Karl N., et al., "Risk-Informed Safety Management of Japanese Nuclear Power Plants," prepared for Institute of Nuclear Safety System and Computer Software Development, April, 2000.

Fleming, Karl N. and Jeff Mitman, "Quantitative Assessment of a Risk Informed Inspection Strategy for BWR Weld Overlays," Proceedings of ICONE 8, Baltimore, MD, April 2-6, 2000.

Fleming, Karl N., "PRA Quality in Risk Informed Applications," personal views presented to U.S. Nuclear Regulatory Commission Advisory Committee on Reactor Safeguards Retreat, Clearwater, FL, January 2000.

Fleming, Karl N., "Braidwood Nuclear Power Station Probabilistic Risk Assessment, October 1999 Upgrade Summary Document," prepared for ComEd Nuclear Generation Group, Risk Management Group, January, 2000.

Fleming, Karl N., "Risk Informed Inservice Inspection (RI-ISI)," ERIN Engineering and Research, Inc., Carlsbad, CA, December, 1999.

Fleming, Karl N., et al., "Taiwan Power Company, Lungmen Project Fourth Nuclear Power Plant Units 1 and 2, Plant Reliability and Availability Analysis Report," prepared for GE Nuclear Energy, November, 1999.

Fleming, K. and Carl R. Grantom, "A New Scenario Based Approach for Predictive Reliability Performance Assessment for Electric Power Plants and Other Production Facilities," Proceedings for the 7th Annual Conference of SMRP, Denver, CO, October 3-6, 1999.

Fleming, Karl N., "Byron Nuclear Power Station Probabilistic Safety Assessment Update, 1999 Summary Document," prepared for ComEd Nuclear Generation Group, Risk Management Group, August, 1999.

Fleming, Karl N., "Technical Approach for Integrated Reliability Analysis of Electric Generating Facilities and Application of *PLANFORMA™* Software" ERIN Engineering and Research, Inc., Carlsbad, CA, August 7, 1999. *ERIN Proprietary Information*

Fleming, Karl N., "Technical Issues in the Treatment of Dependence in Seismic Risk Analysis," presented to OECD – NEA Workshop on Seismic Risk, Tokyo, Japan, August, 1999.

Karl N. Fleming

PUBLICATIONS
(continued)

Fleming, Karl N., David A. Bidwell, and Mark A. Melnicoff, "Application of Piping Reliability Models and Service Data to Evaluate Large Service Water System Pipe Break Frequencies," Proceedings of the Probabilistic Safety Assessment, International Topical Meeting (PSA99), Washington, D.C., August 22-26, 1999.

Fleming, K., Steve Gosselin, and Jeffrey Mitman, "Application of Markov Models and Service Data to Evaluate the Influence of Inspection on Pipe Rupture Frequencies," Proceedings for 1999 ASME Pressure Vessels and Piping Conference, Boston, MA, August, 1999.

Fleming, Karl N. and Thomas J. Mikschl, "Technical Review of Draft Report INEEL/EXT-99-00613 *Common Cause Failure Insights Volume 2: Emergency Diesel Generators*," prepared for INEEL Lockheed Martin Idaho Technologies Company, Idaho Falls, ID, July 20, 1999.

Fleming, Karl N., F. A. Silady, and D. E. True, "Risk-Informed Safety Management of Japanese Nuclear Power Plants, Final Phase 2 Report" prepared for Institute of Nuclear Safety System and Computer Software Development, April, 1999.

Fleming, K., et al., "Revised Risk-Informed Inservice Inspection Evaluation Procedure," EPRI Report TR-112657 by ERIN Engineering and Research, Inc., April, 1999.

Fleming, K., et al., "Probabilistic Risk Assessment Risk Management Program Operating Guidelines (RMOG), Volume 4: Technical Approach for Performing and Updating ComEd Plant Specific PSAs," Draft Report, March 31, 1999.

Mikschl, Thomas J., et al., "Piping System Failure Rates and Rupture Frequencies for Use in Risk Informed In-service Inspection Applications," EPRI Report TR-111880 by ERIN Engineering and Research, Inc., March, 1999.

Fleming, K., et al., "Quantitative Assessment of Risk Impacts of Proposed Inspection Strategy for BWR Weld Overlays," Draft Report prepared for EPRI NDE Center, Charlotte, NC, March, 1999.

Bidwell, David A. and Karl N. Fleming, "Estimation of the Essential Service Watery System (SX) Piping Rupture Frequency at Commonwealth Edison's Byron and Braidwood Stations," prepared for Commonwealth Edison, December, 1998.

Fleming, Karl N., et al., "Application of Markovian Technique to Modeling Influences of Inspection on Pipe Rupture Frequencies," Proceedings of PSAM4, New York City, New York, September, 1998.

Mikschl, Thomas J. and Karl N. Fleming, "Estimation of Pipe Failure Rates from Service Experience to Support Risk Informed In-Service Inspection Programs," Proceedings of PSAM4, New York City, New York, September, 1998.

Rodgers, Shawn S. and Karl N. Fleming, "Property Damage Risk Assessment for Nuclear Power Plants," Proceedings of PSAM4, New York City, New York, September, 1998.

Karl N. Fleming

PUBLICATIONS *(continued)*

Fleming, Karl N., "Scenario Based Approach for Plant Reliability, Availability and Capacity Factor Assessment," Proceedings of PSAM4, New York City, New York, September, 1998.

Fleming, K., et al., "Piping System Reliability Models and Database for Use in Risk Informed In-Service Inspection Applications," EPRI Report TR-110161 by ERIN Engineering and Research, Inc., June, 1998.

Fleming, Karl N., Doug E. True and Thomas J. Mikschl, "Independent Review of Calvert Cliffs Probabilistic Risk Assessment," prepared for Baltimore Gas and Electric Company, May, 1998.

Aoi, S., D. E. True and K. N. Fleming, "Decision Criteria for Risk-Based Management of Japanese Nuclear Power Plants," prepared for Institute of Nuclear Safety System, February 24, 1998.

Fleming, Karl N. and Douglas E. True, "Regulatory Enhancements through Application of Probabilistic Safety Assessment Technology and Insights," prepared for Atomic Energy Control Board, Ottawa, Canada, October, 1997.

Fleming, Karl N., "Validation of PSAs for Use in Risk Monitoring Applications," ASME PVP Conference published in PVP - Vol. 358 *Risk Informed Decision Making,* Book No. G01070 - 1997.

Fleming and S. Gosselin, "Application of Markovian Technique to Modeling Influences of Inspection on Pipe Rupture Frequencies," Proceedings of the Seminar on Piping Reliability, Swedish Nuclear Power Inspectorate, Sigtuna Sweden, September 30-October 1, 1997.

Gosselin, Stephen R. and Karl N. Fleming, "Evaluation of Pipe Failure Potential Via Degradation Mechanism Assessment," Proceedings of ICONE 5, 5th International Conference on Nuclear Engineering, May 26-30, 1997, Nice, France.

Fleming, K. N., et al., "Property Damage Risk Assessment Scoping Study For South Texas Project Generating Electric Station," Electric Power Research Institute, November, 1996.

Fleming, K. N., et al., "Survey On The Use Of Configuration Risk And Safety Management Tools At Nuclear Power Plants," Électricité De France, November, 1996.

Fleming, K. N., et al., "Independent Review EPRI Risk-Informed Inservice Inspection Procedure - Final Report," Electric Power Research Institute, September, 1996.

Fleming, et al., "Independent Review EPRI Risk Informed In-Service Inspection Procedure," published by ERIN Engineering and Research, Inc. for EPRI, July, 1996.

Fleming, K. N., "Developing Useful Insights And Avoiding Misleading Conclusions From Risk Importance Measures In PSA Applications," PSA '96, Park City, Utah, June, 1996.

PUBLICATIONS (continued)

Contributing Author to:
Bidwell, D. A., C. R. Grantom, K. N. Fleming, "A PSA Application At The South Texas Project Electric Generating Station: Generic Letter 89-10 MOV Prioritization," PSA '96, Park City, Utah, June, 1996.

Contributing Author to:
True, D. E., K. N. Fleming, E. T. Burns, C. D. Sellers, "The Importance Of Utilizing A Blended Approach In Regulatory Applications Of Probabilistic Safety Assessment," PSA '96, Park City, Utah, June, 1996.

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Garrick, B. John and Karl N. Fleming, "A Progress Report on the Status of Fruitful Applications of PRA in the U.S. Nuclear Power Industry," Proceedings of PSA'95, Seoul, Korea, November, 1995.

Garrick, B. John, Karl N. Fleming and Jürg Landolt, "Engineering Insights into Safety Features of an Advanced Light Water Reactor Built in 1979," Proceedings of PSA'95, Seoul, Korea, November, 1995.

Fleming, K. N., et al., "Gösgen Probabilistic Safety Assessment," PLG-0870 Modules I-X, prepared by PLG for Kernkraftwerk Gösgen-Däniken AG, 1994.

Fleming, K. N., and J. Landolt, "Engineering Insights into Safety Features Derived from the Gösgen PSA," presented at 4th TÜV-Workshop on Living PSA Application, Hamburg, Germany, May 3, 1994.

Rao, S. B., G. A. Tinsley, and K. N. Fleming, "Common Cause Event Database for Risk and Reliability Evaluations," presented at 4th TÜV-Workshop on Living PSA Application, Hamburg, Germany, May 3, 1994.

Dykes, A. A., C. R. Grantom, K. N. Fleming, J. M. Oddo, F. J. Rahn, and D. H. Johnson, "U.S. Nuclear Industry Efforts in Utilizing PSA for Technical Specifications Changes," presented at IAEA Technical Committee Meeting on Procedures for Use of PSA for Optimizing NPP Operational Limits and Conditions, Barcelona, Spain, September 20-23, 1993.

Fleming, K. N., D. C. Bley, and J. H. Moody, "PSA Methods for Potential Accidents Initiated at Shutdown," *Proceedings of PSA '93* Probabilistic Safety Assessment International Topical Meeting, Clearwater Beach, Florida, pp. 91-96, January 26-29, 1993.

Fleming, K. N., and F. W. Etzel, "Assessment and Interpretation of Risk Importance Measures from Beaver Valley Level 2 PRA," Proceedings of *PSA '93* Probabilistic Safety Assessment International Topical Meeting, Clearwater Beach, Florida, pp. 116-122, January 26-29, 1993.

Read, J. W., and K. N. Fleming, "Electric Power Recovery Models," *Proceedings of PSA '93* Probabilistic Safety Assessment International Topical Meeting, Clearwater Beach, Florida, pp. 631-640, January 26-29, 1993.

PUBLICATIONS
(continued)

Emerson, M. A., K. N. Fleming, D. J. Wakefield, and S. A. Epstein, "RISKMAN® - A System for PSA," *Proceedings of PSA '93* Probabilistic Safety Assessment International Topical Meeting, Clearwater Beach, Florida, pp. 722-729, January 26-29, 1993.

Rao, S. B., G. A. Tinsley, and K. N. Fleming, "Common Cause Event Database for Risk and Reliability Evaluations," *Proceedings of PSA '93* Probabilistic Safety Assessment International Topical Meeting, Clearwater Beach, Florida, pp. 797-803, January 26-29, 1993.

Ho, V. S., and K. N. Fleming, "Impact of Plant Redundancy Level on Risk Contribution of Fires to Core Melt Frequency," *Proceedings of PSA '93* Probabilistic Safety Assessment International Topical Meeting, Clearwater Beach, Florida, pp. 913-917, January 26-29, 1993.

Deremer, R. K., K. N. Fleming, and D. J. Wakefield, "Accident Management Insights Obtained during the Beaver Valley Unit 2 Individual Plant Examination Process," *Proceedings of PSA '93* Probabilistic Safety Assessment International Topical Meeting, Clearwater Beach, Florida, pp. 1049-1053, January 26-29, 1993.

Fleming, K. N., and R. P. Murphy, "Lessons Learned in Applying PSA Methods to Technical Specification Optimization," presented at Technical Committee Meeting on Advances in Reliability Analysis and Probabilistic Safety Assessment, International Atomic Energy Agency, Budapest, Hungary, September 7-11, 1992.

Fleming, K. N., and D. C. Bley, "Risk Management Applications of Probabilistic Safety Assessment at PLG," presented at Third Workshop on Living PSA Applications, Hamburg, Germany, May 11-12, 1992.

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Karl N. Fleming

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Karl N. Fleming
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Fleming, K. N., et al., "HTGR Accident Initiation and Progression Analysis Status Report," General Atomic Company, GA-A13617, Vol. II, October, 1975.

Douglas E. True

Senior Vice President, Safety and Reliability Services

AREAS OF EXPERTISE

- ***Computer Program Development and Deployment***
- ***PRA***
- ***Regulatory Compliance***
- ***Operational Safety***
- ***Emergency Response***
- ***Radiation Protection***
- ***SAR Preparation***
- ***Radioactive Material Handling***
- ***Procedural Compliance***
- ***Consequence Analysis***

WORK EXPERIENCE SUMMARY

Mr. True is Senior Vice President of ERIN Engineering's Safety and Reliability services. He has significant experience leading the development of computer applications and complex integrated decision support programs. His technical background includes engineering, safety analysis and operations of Department of Energy facilities, making him uniquely qualified for safety analysis work. He has served as the technical director of numerous large scale safety analysis projects. Prior to joining ERIN, Mr. True worked for a DOE contractor serving as the facility manager of a plutonium oxide production facility and as a process engineering manager for a fuel reprocessing facility.

WORK EXPERIENCE

Senior Vice President, Safety and Reliability Services at **ERIN Engineering and Research, Inc.** Responsible for information technology and support services as well as providing consulting services in the areas of safety analysis, probabilistic risk assessment, hazard assessment, risk management and chemical process safety management. The scope of services supported has spanned from DOE facilities, to major chemical manufacturers, to nuclear power plants.

Mr. True has participated in an lead the development of a wide range of ERIN Engineering computer programs and applications. He has directly lead the development of the REBECA computer program and managed the overall development of several computer programs which enhance the ability to perform risk applications and integrate risk information with other data to support risk informed decisions. He has played a major role in the definition and development of both ORAM and SENTINEL and their deployment. He has support the development of PERMON and PUMA and directs the overall integration of data in both the LYNX workstation development and related applications. He has a strong technical background in decision analysis and information display. His operator background provides excellent input to support user applications in the field. He is responsible for the overall operation of the ERIN Engineering Information Technology Business Area.

Mr. True is also actively involved in the risk and reliability aspects of the nuclear power industry. He served as the technical director of five probabilistic risk assessments (PRAs) for nuclear power plants (**Trojan, San Onofre 1, San Onofre 2/3, Fermi 2, and Watts Bar**). He has supported the **Electric Power Research Institute (EPRI)** and the **Nuclear Management and Resources Council (NUMARC)** on numerous technical and policy issues ranging from graded approaches to risk management to safety goal policy implementation to development of reliability programs.

Additionally, served as Technical Director on numerous other risk related projects including:

- A comprehensive assessment of the current U.S. accident management capabilities
- Risk assessment/alternative design studies for system modifications

Douglas E. True
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EDUCATION

B.S. Chemical Engineering, University of California at Berkeley

SECURITY CLEARANCE

U.S. Citizen

In-Active Department of Energy "Q" Clearance

LICENSES/REGISTRATIONS/ PROFESSIONAL SOCIETIES

American Nuclear Society

PUBLICATIONS

Available Upon Request

- Support to NUMARC in resolving station blackout issues involving emergency diesel generator reliability

- Support to NUMARC in rebutting the NRC regulatory analysis and safety assessment of the proposed Maintenance Rule

- Expert witness support in administrative hearings on the risk impacts of power plants.

Processing Engineering manager at **Rockwell Hanford Operations**. Responsibilities included technical direction and management of an engineering staff, direct responsibility for support of plant operation and maintenance, process optimization, process and equipment problem resolution and safety system analysis. Performed a comprehensive review of plant safety systems to identify active components whose failure could compromise plant safety, identification resolution and correction of problems leading to high radioactive effluent release and process waste volume minimization.

Served as an operations manager for a plutonium dioxide production facility. Managed a group of operating personnel (operators and shift supervisors) to successfully startup and operate this facility. Responsible for all phases of operation, maintenance and testing of the facility and served as the prime technical interface between operations and engineering for plutonium processing.

Engineer at **General Atomics and Cygna Energy Services**. Performed probabilistic risk assessments and systems performance analysis of various high temperature gas cooled reactor concepts. These evaluations included system performance evaluations of alternative reactor heat uses and risk evaluations of potential interactions between reactor systems and associated explosion hazards for a combined reactor/chemical processing plant concept. This analysis involved a probabilistic evaluation of explosion hazards resulting from the release of methane gas from an adjacent chemical plant, including gaussian plume analysis, ignition source identification and meteorological considerations. This experience allowed him to be a contributing author of a comprehensive text on probabilistic risk assessment prepared and presented in a short course to a midwest utility.

Jeff R. Gabor

General Manager Operations and Technical Solutions

AREAS OF EXPERTISE

- ***Plant Thermal-Hydraulic Response***
- ***Severe Accident Analysis***
- ***Severe Accident Management***
- ***Plant Modeling***
- ***Thermal-Hydraulic and Severe Accident Training***

WORK EXPERIENCE SUMMARY

Mr. Gabor, a General Manager of Operations and Technical Solutions for ERIN Engineering and Research, is a Mechanical Engineer with considerable experience in the field of nuclear plant thermal-hydraulic and severe accident analysis. Mr. Gabor has over 18 years experience in Nuclear Power Plant Safety.

WORK EXPERIENCE

Mr. Gabor is currently involved in several Level 2 PSA updates and continues to support a variety of severe accident management and thermal-hydraulic activities at numerous utilities. The following are some of his recent activities:

- Containment response analysis to support EQ evaluation at Cooper Nuclear Station. Work included detailed thermal-lag analysis of key components.
- Level 2 PSA updates at Quad Cities, LaSalle, Vermont Yankee, and Browns Ferry.
- Development of **Technical Support Guidelines** at Clinton Power Station, Duane Arnold Energy Center, WNP2 and Fermi.
- Lead technical analyst for the **BWR Accident Scenario Template Development.**
- Manager and lead technical analyst for implementation of **MAAP4** at Nine Mile Point Unit 2, River Bend, and Cooper.
- Manager and lead technical analyst for **MAAP4** analysis in support of EdF PWR Level 2 PSA.
- **Severe Accident Training** at DAEC, WNP2, and Cofrentes.
- Lead **Thermal-hydraulic** analyst in support of Quad Cities PSSA

As an associate with Dames & Moore, Mr. Gabor was responsible for resource development, strategic planning, and technical oversight for all nuclear activities carried out in the Westmont, Illinois office. He worked with nuclear utilities in addressing issues related to plant thermal-hydraulic response.

- Lead technical analyst for the Garona Level 2 PRA for Nuclenor (Spain).
- Lead technical analyst for the Cofrentes Level 2 PRA for Iberdrola (Spain).
- Technical support to the Consumers Power Big Rock Point Nuclear Plant on issues related to plant thermal-hydraulic response, severe accident analysis, and equipment qualification.

Jeff R. Gabor

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EDUCATION

M.S., Mechanical Engineering,
University of Cincinnati
Cincinnati, Ohio

B.S., Nuclear Engineering,
University of Cincinnati
Cincinnati, Ohio

SECURITY CLEARANCE

U.S. Citizen

PUBLICATIONS

Furnished Upon Request

- MAAP4 parameter file development and severe accident training for Cooper Nuclear Station.
- Implementation of BWROG Technical Support Guidelines for the Cofrentes Nuclear Station.
- Development and Implementation of Remote Monitoring for soil vapor extraction remediation systems.
- Technical analyst for severe accident investigations in support of certification of advanced light water reactor designs. Development of computer simulation tools and presentations to USNRC and the ACRS concerning severe accident behavior. This work was performed under contract with the Department of Energy.

Member of a GE design review committee for the evaluation of the impact of Noble Metal Chemical Addition on the containment atmospheric monitoring systems.

Vice President and Co-founder of Gabor, Kenton & Associates, Inc.

- Technical support for Level 2 PRA on the General Electric Advanced Boiling Water Reactors. Included numerous technical presentations to USNRC, ACRS, and USDOE.
- Lead technical analyst for severe accident response on a number of BWR Level 2 PRAs:
  - Millstone Unit 1
  - Duane Arnold
  - Pilgrim
  - Nine Mile Point Units 1 and 2
  - Fermi
  - Vermont Yankee
  - Cofrentes
  - Browns Ferry
  - Cooper
- Lead BWR analyst for EPRI sponsored MAAP 3.0B Thermal-Hydraulic Qualification Study
- Independent Review of MAAP 3.0B and MAAP 4 maintenance activities
- Developed and managed Gabor, Kenton & Associates Quality Assurance Program
- Provided technical support for the containment vent evaluation of the Cofrentes and Garona plants (Spain). Performed vent sizing calculations and on-site radiation dose assessment.

Jeff R. Gabor

Manager of Plant Analysis and Special Projects for Fauske & Associates, Inc.

- Principal author of the BWR Modular Accident Analysis Program (MAAP), developed as part of the nuclear industry-sponsored degraded core rulemaking program (IDCOR)
- Severe accident evaluations of Grand Gulf and Peach Bottom in support of IDCOR
- Pilgrim Safety Enhancement Program
- BWR Owners Group Mark I Evaluation
- Caorso Severe Accident Analysis
- Mark I shell melt-through analysis and experiments
- Swedish Reactor Accident Mitigation Analysis (RAMA)
- Shoreham PRA
- Vermont Yankee 60 Day Study
- Empire State Electric Energy Research Corporation LWR Code Comparison
- Managed and participated in MAAP and severe accident phenomenology training courses for nuclear industry along with numerous presentations to the NRC and ACRS on severe accident phenomenology
- Author of the BWR Individual Plant Examination Methodology (IPEM) for source term
- Designated Westinghouse Expert Engineer in severe accident thermal hydraulic transient analysis

System Engineer for Cincinnati Gas and Electric Company

- Implementation of post-TMI design changes
- Pre-operational testing program
- Completed RETRAN training program

Resident Student Associate at Argonne National Laboratory

- Experimentation on the transition from film boiling to nucleate boiling on a flat plate
- Computer analysis of LMFBR core design

Jeff R. Gabor

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Mr. Gabor's engineering experience was gained through employment with the following companies:

- ERIN Engineering and Research, Inc.
- Dames & Moore
- Gabor, Kenton & Associates, Inc.
- Fauske & Associates, Inc.
- Cincinnati Gas and Electric Company
- Argonne National Laboratory

Vincent M. Andersen

Supervising Engineer, Probabilistic Safety Assessment and Reliability

AREAS OF EXPERTISE

- *Severe Accident Analysis*
- *Fault Tree Analysis*
- *Event Tree Analysis*
- *PSA Data Analysis*
- *PSA Applications*
- *External Events*
- *Shutdown Risk*
- *Emergency Operating Procedures*

WORK EXPERIENCE SUMMARY

Mr. Andersen is a supervisor experienced in nuclear systems engineering. He has a degree in Mechanical Engineering. He has over fifteen years experience in the risk assessment area. Mr. Andersen has contributed to and reviewed numerous Level 1 and Level 2 PSAs, as well as numerous other risk related projects.

WORK EXPERIENCE

As a Supervising Engineer at ERIN Engineering and Research, Inc., Mr. Andersen uses risk assessment and engineering methods to assist nuclear utility clients in responding to internal and regulatory issues. The following are highlights of Mr. Andersen's work experience:

- Pilgrim PRA Peer Review to support BWROG PRA Certification
- Development of plant-specific Significance Determination Process (SDP) models and guidelines for various BWRs (e.g., Peach Bottom, Limerick, WNP2).
- Assessment of the incremental plant risk at ComEd plants associated with seismic induced failure of RCS-connected piping that are blanketed with lead shielding during mode 5 maintenance activities.
- Update of the LaSalle and Quad Cities PSAs. These projects involved update and documentation of numerous supporting analyses, such as system notebooks, IE and data analyses, HRA, CCF dependencies and LERF.
- Development of responses to NRC Request for Additional Information (RAIs) for the DAEC, Fermi AND Vermont Yankee IPEEE Submittals.
- Update of the Cooper Nuclear Station Level 2 PSA (developed using the EVENTRE code).
- Assessment of the impact on the Browns Ferry Maintenance Rule Program due to a proposed conversion to a 24 month fuel cycle.
- Review of the Cofrentes (Mark III BWR in Spain) Level 2 PSA..
- Review and support for the Lungmen Severe Accident Analysis, Design Options, and PRA (specifically seismic analysis and interfaces and dependencies). Lungmen is an advanced BWR being designed by General Electric for Taiwan Power.
- Technical lead for Severe Accident Analysis and PSA activities at the Duane Arnold Energy Center (DAEC), including: Maintenance Rule risk characterization support, support for the SENTINEL models, IPEEE modeling and NRC Submittal preparation, PSA pedigree process, and PSA model/ documentation update.
- Fire modeling support to Baltimore Gas & Electric for the Calvert Cliffs IPEEE as part of the response to Severe Accident Policy Statement closeout. This support included teaching BG&E personnel the application of FIVE and EPRI Fire PRA Implementation Guide deterministic fire modeling techniques.
- Removed conservatisms and screening approaches from the DAEC fire IPEEE models, developed seismic models (which were not created for the DAEC IPEEE) and merged these models into the DAEC Living PSA models in preparation for use in risk informed

Vincent M. Andersen

EDUCATION

B.S.M.E., Mechanical Engineering San Jose State University

M.B.A., Master of Business, San Jose State University

SECURITY CLEARANCE

U.S. Citizen

decision making.

- Review and performance of severe accident analysis and PSA modeling for a number of plants in support of the NRC Individual Plant Examination Program. These plants include: San Onofre, Cooper, Duane Arnold, Peach Bottom, Limerick, Fermi, Nine Mile Point, and Vermont Yankee. For all of the plants listed, except San Onofre and Cooper, Mr. Andersen was involved in scoping, developing, quantifying and documenting the Level 2 analyses. In the case of Duane Arnold, Mr. Andersen was involved in both the Level 1 and Level 2 analyses and the IPE Submittal documentation. In the case of Cooper, Mr. Andersen was involved in the peer review process of the Cooper Level 1 and Level 2 PSA models and documentation over a many year period.
- IPEEE fire modeling and FIVE screening analyses for the Fermi plant in support of the Fermi IPEEE response. Due to Fermi specific plant configuration and also conservative screening techniques in the FIVE methodology, this effort required detailed deterministic fire modeling of the 4160V switchgear equipment and the Auxiliary Building in general.
- External event risk analyses for the Consolidated Tritium Facilities (CTF) at the DOE Savannah River Site. This effort involved the development of Scenario Analysis Notebooks for design basis earthquakes, high winds, and internal fires. Techniques used in the analyses were specified by DOE documents (e.g., UCRL-15910, Design and Evaluation Guidelines for DOE Facilities Subjected to Natural Phenomena Hazards).
- PSA pedigree process developed and pilot implemented for the Duane Arnold Energy Center. The project was a tailored collaboration between IES Utilities and EPRI. The process design and implementation involved the plant QA department, development of PSA procedural and technical guidelines, and modifications to the models and documentation. The project was documented in a published EPRI report.
- Study of risk management activities at U.S. nuclear utilities. This study was performed in collaboration with GE and the BWROG.
- Risk significance evaluations for the Cooper (CNS) Technical Specifications proposed by CNS for relocation to plant-controlled documents.
- Interfacing Systems LOCA (ISLOCA) evaluations in support of the EPRI project to supply ISLOCA PSA guidelines to utilities. This effort involved developing a process by which to identify dominant ISLOCA pathways and quantify associated dominant sequences. This project was documented in a published EPRI report.
- Mr. Andersen has also participated in a number of shutdown risk studies for various plants. These plants include Grand Gulf, Peach Bottom, Perry, WNP-2, Quad Cities, Fermi, and Duane Arnold. The ORAM code was used in all cases. In the case of Duane Arnold, the first shutdown models were developed using the ERIN multi-purpose PSA code, REBECA; the REBECA models were later converted into ORAM. These efforts typically involved development of time to boil curves, shutdown human error probabilities, shutdown initiating event frequencies, and shutdown event trees. In some cases, the effort included the development of Risk Management Guidelines and Safety

LICENSES/REGISTRATIONS/ PROFESSIONAL SOCIETIES

American Nuclear Society

Eagle Alliance

PUBLICATIONS

Available Upon Request

Function Assessment Trees.

- Mr. Andersen has provided PSA training courses to various plant groups over the years. In addition, Mr. Andersen developed, coordinated, and participated in the presentation of a one-week PSA training course that invited personnel from across the nuclear industry.

Prior to joining ERIN in 1990 and as an Engineer at Tenera, L.P. (formerly Delian Corporation), Mr. Andersen participated in the following projects:

- Study of charcoal adsorbers use in ALWR design and develop a control room heatup code in support of the ALWR Requirements Document.
- Risk evaluation for several facilities belonging to the Chemical Technology Division of Oak Ridge National Laboratory (ORNL). This effort focused on hazard identification and documentation for a number of processes and storage areas at the site. This project involved touring the ORNL facilities and discussing issues with cognizant individuals.
- Level 2 PSA containment venting study for two Spanish nuclear plants, Garoña and Cofrentes.
- Study of the impact of uncertainty on severe accident policy statement decision making. This study defined types of uncertainty, identified contributors to uncertainty, summarized past uncertainty evaluations, investigated uncertainty in past PSAs, and provided a qualitative method for treating uncertainty. This effort was performed for the Industry Degraded Core rulemaking body (IDCOR).
- Support systems modeling, event tree development, and data compilation in support of the Dresden and Quad Cities Individual Plant Evaluations. In both cases this involved plant visits to gather plant-specific data and the development of plant-specific component failure probabilities and initiating event frequencies.
- Development, quantification, and documentation of Level 1 and Level 2 PSA analyses to support the startup of the Shoreham nuclear power plant.
- System sensitivity analyses for the Monticello and Pilgrim IPEs.
- System modeling in support of the Pilgrim and WNP-2 IPEs.
- Review of the BWR Owners Group Emergency Procedures Guidelines (EPGs) as they pertain to the mitigation of accidents post core damage.

David A. Bidwell

Lead Senior Engineer

AREAS OF EXPERTISE

- ***NRC Significance Determination Process***
- ***Revisions to PWR and BWR IPEs***
- ***PWR IPE Development and Update***
- ***Maintenance Rule***
- ***Common Cause Modeling***
- ***Failure rate data, equipment maintenance data, equipment demand success data***
- ***Generic database development***
- ***IPEEEs***
- ***Plant Operations***
- ***RMPPs***

WORK EXPERIENCE SUMMARY

Mr. Bidwell has 12 years experience in power and shutdown Probabilistic Risk Analysis (PRA) for numerous U.S. and European utilities. Served as a member of the Nuclear Regulatory Commission (NRC) Senior Review Board for IPEEEs. Was a member of the U.S. NRC mandated oversight team at Sequoyah Fuels processing facility. Recent experience includes PRA systems and data update, Maintenance Rule, and industry support of the implementation of the NRC Significance Determination Process.

WORK EXPERIENCE

Lead Senior Engineer at **ERIN Engineering and Research, Inc.** Mr. Bidwell has recently participated in the review of the NRC's Significance Determination Process worksheets for TVA's Sequoyah, Watts Bar and Browns Ferry plants. Following the review, he participated in the meetings with the NRC that took place at each of the plants. Mr. Bidwell has created similar worksheets for each of the sites based on the most recent PRA revision, rather than the IPE which the NRC has used.

Other recent experience includes key support of the revision of the Browns Ferry and Watts Bar PRA models. His work included Bayesian updates to component failure rates, planned and unplanned maintenance terms, and common cause terms. He also supported event tree model development and debugging. Finally, his support included systems analysis and authoring of reports on final results and insights.

For Commonwealth Edison's Byron and Braidwood stations, Mr. Bidwell updated the essential service water system initiating event frequency caused by passive component ruptures. He also performed a plant specific common cause data update including the incorporation of plant to plant variability in the common cause parameters. He also participated in the update of the PRA component failure rate database, and initiating event frequencies including a special treatment of a dual unit loss of offsite power for Byron and Braidwood.

Mr. Bidwell also performed a data intensive re-analysis of piping failure and rupture rates by failure mechanism. The results of which are to be used by the Electric Power Research Institute (EPRI) risk-based in-service inspection program of plant piping. The complete set of results has recently been published by EPRI.

Maintenance Rule experience includes a long-term assignment at Southern California Edison's San Onofre Nuclear Generating Station. For a year and a half, Mr. Bidwell developed performance criteria for the low risk significant systems, provided scoping documentation on the high risk significant systems, and created definitive components lists for all systems within the scope of the Maintenance Rule. This facilitated the transition of the program from system-based to component-based. A by-product of the effort was a custom, Microsoft ACCESS based, relational database of all components and documentation.

David A. Bidwell
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EDUCATION

B.S. Applied Physics, Columbia University, New York

SECURITY CLEARANCE

U.S. Citizen

LICENSES/REGISTRATIONS/ PROFESSIONAL SOCIETIES

Air Force, Army, and Navy ROTC Scholarships

New York State Regents Scholarship

For Omaha Public Power District's Fort Calhoun Station, Mr. Bidwell has twice managed the PRA data update, including component failure rates, maintenance unavailability, durations, and many initiating events. These two updates corresponded to the conclusion of successive refueling cycles. Each effort was performed with the support of co-op students under his direction.

Mr. Bidwell has assisted in a major Probabilistic Safety Assessment (PSA) enhancement project at Houston Lighting and Power (HL&P) South Texas Project in which the balance of plant systems will be explicitly incorporated into the full power model. The project will model any secondary system whose failure or degradation will result in an initiating event or plant transient. In addition, the project will develop a plant reliability and availability predictive tool.

He has performed risk based prioritization of MOV testing in response to Generic Letter 89-10 for South Texas Project. This task entailed the decomposition of the plant model into its basic events and then ranking their risk significance using measures of Fussell-Vesely and Risk Achievement Worth. In addition, the common cause factors for key plant equipment were re-screened and updated accounting for the MOV testing that had already taken place. The plant at power model was updated and requantified.

Also for HL&P, Mr. Bidwell has performed an analysis of the risk trade-off of moving an important shutdown test to full power operation. The analysis compared the expected risk increase against criteria set forth in the PSA Applications Guide, and compared that against the expected decrease in plant risk realized by removing the test from shutdown operations. The analysis then evaluated the economic trade-off of a shortened refueling outage against the increased risk of reactor trip at power.

Prior to joining ERIN, Mr. Bidwell was an Engineering Consultant to PLG, Inc. He was a member of the U.S. NRC mandated oversight team at Sequoyah Fuels. This task entailed the oversight of the day to day facility operations and intra-departmental communications of the facility. Later, he helped to develop and update the site's licensing documents. He also contributed to the writing of the site Safety Analysis Report for the EG&G Mound facility.

Additionally, he has participated in hazard and operability studies (HAZOP) for several petrochemical facilities and contributed to the writing of RMPPs for those sites as required by the state of California.

He participated in the development of a shutdown PRA for HL&P. This work included developing event tree models (developing rules for split fraction assignment) and shutdown specific systems models based upon plant procedures, Technical Specifications, P&IDs, and input from HL&P personnel. Prior to that, Mr. Bidwell assisted HL&P in using the IPE model to justify extending the Technical Specification diesel generator allowed outage time. Concurrent with that activity was the incorporation of further refinements and updates to the at power model. In cooperation with HL&P personnel, Mr. Bidwell helped gather plant specific data in support of an update of the plant specific database. This task included site data collection, statistical analysis of failure rate data, equipment maintenance data development, and equipment demand success data development.

David A. Bidwell

Mr. Bidwell's experience also includes the development of analytical models and documentation in support of IPEEEs. His IPEEE experience includes other External Events (high winds, tornadoes, floods, and lightning) analysis for the plants Hatch, Farley, and Maine Yankee. In addition, he served as a member of the NRC's Senior Review board for IPEEEs. In this capacity, he reviewed Florida Power and Light's High Winds, Floods, Other External Events (HFO) portion of the IPEEE submittal for the Turkey Point plant. He participated in the reviews and discussions of the HFO submittals of Diablo Canyon, Catawba, Haddam Neck, McGuire, and St. Lucie.

He participated in the development of a multi-unit PRA for the Tennessee Valley Authority's Browns Ferry site. This study evaluated the total risk to the site of power operation by more than one unit in several combinations.

In addition, he developed the electric power model and documentation for Browns Ferry, Sequoyah, and Watts Bar IPE submittals. Lastly, he developed a time dependent off-site power and diesel generator recovery model for the Browns Ferry site.

Mr. Bidwell participated in PRA and updates of other plants including Seabrook, Watts Bar, Sequoyah, Maine Yankee, and the Swiss plants Beznau and Gösgen. For Gösgen, Mr. Bidwell was one of the team members visiting the plant for the initial visit to gather and review the documentation necessary to develop the PRA. Mr. Bidwell also participated in the development of systems analyses in support of a shutdown PRA for the Gösgen plant.

Mr. Bidwell was a trained reactor operator for Southern California Edison at SONGS Unit 1. His responsibilities included the manipulation of both primary and secondary plant systems. He also coordinated plant operations with chemistry, engineering, and technical testing departments. Prior to his work at SONGS Unit 1, he was an equipment operator at SONGS Units 2 and 3.

PUBLICATIONS

Bidwell, D. A., C. R. Grantom, and K. N. Fleming, "A PSA Application at the South Texas Project Electric Generating Station: Generic Letter 89-10 MOV Prioritization," published in *Probabilistic Safety Assessment - Moving Toward Risk-Based Regulation*, International Topical Meeting on Probabilistic Safety, sponsored by the American Nuclear Society, Park City, Utah, September 29 - October 3, 1996.

Read, J. W., and D. A. Bidwell, "Electric Power Recovery Actions for Browns Ferry Nuclear Plant Unit 2," prepared for Tennessee Valley Authority, PLG-0986, March 1994.

Bley, D. C., D. A. Bidwell, D. R. Buttemer, Y. M. Hou, D. H. Johnson, T. J. McIntyre, S. R. Medhekar, and S. L. Thompson, "HVAC Systems and Nuclear Plant Safety," PLG, Inc. prepared for Electric Power Research Institute, PLG-0871, April 1992.

Dykes, A. A., J. A. Mundis, and D. A. Bidwell, "Application of a Bayesian Aging Model to Predict Steam Generator Plugging Rates," published in *Probabilistic Safety Assessment and Management*, Proceedings of the International Conference on Probabilistic Safety Assessment and Management, sponsored by the Society for Risk Assessment, Beverly Hills, California, February 4-7, 1991.

Thomas A. Daniels

Lead Senior Engineer

AREAS OF EXPERTISE

- *Probabilistic Safety Assessment*
- *Systems Analyses*
- *Maintenance Rule*
- *Regulatory Compliance*
- *Program Management*
- *Project Management*
- *Information Management*
- *Technical Training*
- *Technical Communications*

EDUCATION

B.S. Mechanical Engineering (Nuclear Engineering) With Distinction - 1980
Worcester Polytechnic

WORK EXPERIENCE SUMMARY

Mr. Daniels is a Lead Senior Engineer with over twenty years of experience. His technical background includes probabilistic safety assessment (PSA), systems analysis, program management, project management, information management and regulatory compliance. He also has professional experience in print journalism, television and industrial public relations. Prior to joining ERIN in August of 1997, Mr. Daniels supervised a PSA group at a BWR utility, where he also served as the Maintenance Rule coordinator. He has also been the PSA program lead for a PWR utility

WORK EXPERIENCE

Mr. Daniels is a Lead Senior Engineer at **ERIN Engineering and Research, Inc.**. He is responsible for providing consulting services in probabilistic safety assessment, project / program management, information management and areas involving the Maintenance Rule.

Mr. Daniels has supported Carolina Power and Light in their efforts to license two additional spent fuel pools at the Shearon Harris Nuclear Power Plant. He worked as part of a high-level legal / technical team to prepare risk-related materials in support of CP&L's impending appearance in front of a panel of administrative law judges appointed by the United States Nuclear Regulatory Commission's Atomic Safety and Licensing Board.

Mr. Daniels was on the ERIN / General Electric team that recently completed the first of a kind ATLAS™ design bases accidents and transients model for **FirstEnergy's** Perry Nuclear Power Plant. He is currently supporting Exelon in their efforts to build Atlas™ models for Quad Cities and Dresden. He also recently conducted a detailed review of the Shearon

SECURITY CLEARANCE

U.S. Citizen

Harris PSA for **Carolina Power & Light** to asses the potential imapct of a planned power uprate, steam generator replacement and Technical Specifications setpoints optimization. He is working with **Entergy** at Arkansas Nuclear One, River Bend, Waterford and Grand Gulf to produce plant-specific information notebooks in support of the NRC's Significance Determination program.

Mr. Daniels supported **Nebraska Public Power District** in the Cooper Nuclear Station environmental qualification (EQ) recovery program in the spring of 2000. He assisted in identifying potential pre-startup field work scope reduction, in organizing and analyzing design information and field data and in developing station-specific, location-specific LOCA initiating event probabilities.

Mr. Daniels has done extensive work on restructuring 10 CFR 50.65 Maintenance Rule programs for **Rochester Gas and Electric** and **Consolidated Edison of New York**. At Indian Point 2, he led a team of 14 professionals (including ERIN personnel, contract staff and Edison employees) that completely restructured the program, trained the IP-2 staff and shepherded the new program through an NRC baseline inspection.

Mr. Daniels spent eighteen months working with **Ontario Power Generation** to develop an environmental qualification program for their Bruce Nuclear Power Development. His work in Canada included business process development, process modeling and extensive information management activities. He designed, built and managed an extensive system of linked Oracle and Microsoft Access databases and performance metrics that are being used to manage and monitor the Bruce EQ program

Previously, Mr. Daniels was Acting Engineering Programs Supervisor at **Energy Northwest** (formerly the **Washington Public Power Supply System**). He was responsible for supervising six engineers and one non-exempt engineering assistant in the Performance Monitoring Group at the Columbia Generating Station (formerly WNP-2). This group had complete responsibility for developing, implementing and maintaining performance metrics, PSA models and system / component performance tracking programs for the Columbia Generating Station (formerly WNP-2).

As a Principal Engineer at **Energy Northwest** Mr. Daniels' responsibilities included scoping, determination of risk significance, development of performance criteria for structures, systems, trains and components, for collection, analysis and distribution of all plant data and development and implementation of training for 10 CFR 50.65, *Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants* (Maintenance Rule). He was also responsible for developing an implementation plan for integration of an on-line safety monitoring program (EPRI Sentinel) into plant operations, scheduling and work control organizations, as well as, evaluation of risk impact of voluntary entry into technical specification action statements.

Mr. Daniels was employed by **Rochester Gas and Electric Corporation** as a Nuclear Engineer. He was Project manager for a Level 2 probabilistic risk assessment (PRA) in response to United States Nuclear Regulatory Commission (USNRC) Generic Letter 88-20, *Individual Plant Examination For Severe Accident Vulnerabilities - [10 CFR 50.54(f)]*, for the R. E. Ginna Nuclear Power Plant. Mr. Daniels responsibilities included establishment of an RG&E PRA team; preparation of request for proposal, evaluation of bids, interview of candidates, and selection of PRA contractors; preparation, presentation, and maintenance of project budgets and schedules; preparation and maintenance of project engineering procedures to ensure compliance with 10

Thomas A. Daniels
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CFR 50 Appendix B quality assurance requirements; direct supervisory responsibility for one professional and indirect, project responsibilities for remainder of the PRA team. This team included professional, hourly support staff, and contractors; extensive direct involvement with systems analysis, fault tree construction, quantification, recovery analyses, internal flooding analyses and fire analyses; author of USNRC submittal. He was also responsible for other risk-related licensing questions and analysis for Ginna and for RG&E response to NRC rulemaking on *Loss Of All Alternating Current Power* [10 CFR 50.63]; he was RG&E's representative to the Station Blackout Clearinghouse. Mr. Daniels was the Controlled Computer Software Coordinator for the Nuclear Safety & Licensing Group. He was the Nuclear Engineering Services Department representative to the RG&E Software Quality Assurance Task Force 1991-1992 and the Nuclear Engineering Services Department Software Quality Assurance Coordinator 1992-1992. Mr. Daniels was also a Member, Expert Panel, for scoping and risk significance determination for all systems, structures and components per 10 CFR 50.65.

As a Design Engineer I at **Duke Power Company**, Mr. Daniels was the senior technical systems analyst for an in-house Level 3 probabilistic risk assessment of Oconee Nuclear Station Unit 3. He was responsible for implementation, debugging, improvements, and upkeep of Electric Power Research Institute's Computer Aided Fault Tree Analysis (CAFTA) PRA work station software during the Oconee project. Mr. Daniels was a senior technical analyst for IDCOR Task 86.2OC, *Verification Of Individual Plant Evaluation (IPE) For Oconee Unit 3*, as part of demonstration of the IPE PWR IDCOR methodology.

Mr. Daniels was an Associate Engineer at **Babcock and Wilcox Company**. He was responsible for steam generator tube rupture event tree analysis for the Anticipated Transient Operating Guidelines

(ATOG) program. Mr. Daniels was a Task Engineer for Fluid and Transient Analysis Unit work on Washington Public Power Supply System analyses where he utilized transient analysis codes such as TRAP, RELAP5, and CONTEMPT-LT.

Jan F. Grobbelaar

Lead Senior Engineer, Probabilistic Safety Assessment and Reliability

AREAS OF EXPERTISE

- *BWR Systems*
- *PWR Systems*
- *Definition of Plant Operational States*
- *Initiating Event Analysis*
- *Event Tree Analysis*
- *Fault Tree Analysis*
- *Common Cause Analysis*
- *Human Reliability Analysis*
- *Risk Management Systems*
- *Relational Database Development*
- *PRA Software Development*

EDUCATION

B.Sc. Nuclear Engineering, University of Tennessee

B. Comm., University of South Africa

Diploma in Datametrics, University of South Africa

WORK EXPERIENCE SUMMARY

Mr. Grobbelaar is a nuclear engineer with 15 years' experience. Twelve years of his experience is in Boiling Water Reactor (BWR) and Pressurized Water Reactor (PWR) Probabilistic Risk Assessment (PRA).

WORK EXPERIENCE

Mr. Grobbelaar uses risk assessment and engineering methods to assist nuclear utility clients in responding to internal and regulatory issues. The following are highlights of Mr. Grobbelaar's work experience:

- Developed risk information reference to support the NRC Significance Determination Process (SDP) for WNP2.
- Developed risk information reference to support the NRC SDP for Limerick.
- Developed risk information reference to support the NRC SDP for Peach Bottom.
- Developed risk information reference to support the NRC SDP for Quad Cities.
- Determined offsite AC power non-recovery probabilities for WNP-2.
- Determined offsite AC power non-recovery probabilities for Quad Cities.
- Determined offsite AC power non-recovery probabilities for LaSalle.
- Analyzed human reliability for LaSalle.
- Developed system notebooks for LaSalle PRA.
- Analyzed dependencies between systems for LaSalle.
- Reviewed various LaSalle PRA fault trees.
- Modeled common cause failures in various LaSalle PRA fault trees.
- Developed fault trees for various system failures at various plants.

PRA Consultant for PGBI Engineers and Constructors, South Africa, 1998:

- Defined power operational states for Koeberg Nuclear Power Station (KNPS).
- Defined shutdown operational states for KNPS.
- Identified and quantified initiating events for KNPS at power operational states.
- Identified and quantified initiating events for KNPS at shutdown operational states.
- Proposed modifications to use Residual Heat Removal System to back-up Spent Fuel Pool Cooling System at KNPS.

Chief Consultant, ESKOM Nuclear Safety Division, South Africa, 1996 to

Jan F. Grobbelaar

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SECURITY CLEARANCE

Legal alien (H1B Visa)

LICENSES/REGISTRATIONS/ PROFESSIONAL SOCIETIES

American Nuclear Society

Professional Engineer (Engineering Council of South Africa)

PUBLICATIONS

Available Upon Request

1997:

- Developed methodology and software to support a Risk Management System for KNPS.
- Developed database to track plant state and configuration with interface in KNPS control room for input by operating staff.
- Developed post-processing software to support quantification of PRA model conditional on plant configuration.
- Managed the Nuclear Safety Division's budget (R 5 000 000).

Senior Engineer, ESKOM Nuclear Safety Division, South Africa, 1992 to 1995:

- Participated in the fire risk analysis of the Krško Nuclear Power Station in Slovenia.
- Reanalyzed the security risk and evaluated several modifications to physical security measures at KNPS after democratization of South Africa in 1994.
- Developed a methodology for quantifying the real-time risk associated with KNPS.
- Developed a Level 1 Security PRA for KNPS and determined the risk associated with security related initiating events like sabotage.
- Contributed to the initial development of Severe Accident Management Guidelines for KNPS.
- Determined risk associated with Loss of Ultimate Heat Sink for Fessenheim Nuclear Power Station in France.
- Developed a risk based operating regime for a gas turbine power station.
- Determined risk associated with road transportation of spent nuclear fuel.
- Managed the Nuclear Safety Division's budget (R 3 000 000).
- Developed system to manage nuclear safety concerns.
- Developed configuration management system for KNPS Level 1 PRA.
- Assessed fire risk at KNPS.
- Managed group of 8 people while acting as head (5/92 to 12/92).
- Moderated thermodynamics exam papers of students at Witwatersrand Technicon.

Senior Engineer, Koeberg Nuclear Power Station, South Africa, 1991:

- Established PRA site office.
- Determined risk associated with proposed modifications, operating and maintenance activities on a day to day basis.
- Trained site staff in PRA.

Senior Engineer, Koeberg Nuclear Power Station, South Africa, 1991:

(cont'd)

- Interpreted PRA results for site management.
- Coordinated site review of Level 1 PRA.
- Assisted Reliability Centered Maintenance Group in establishing reliability data collection program.

Engineer, ESKOM Nuclear Engineering Division, South Africa, 1988 to 1990:

- Developed pilot software program for fault and event tree analyses that proved viability of computerizing PRA. *Was highly commended for excellence in improvement and innovation in 1989 and 1990.*
- Instrumental in the establishment of the Nuclear Safety Analysis Section and the development of an in-house PRA capability. *Received management award for team building.*
- Developed the fault trees and event trees, which formed the basis of the computerized KNPS Level 1 PRA.

Assistant Engineer, Engineer in Training, ESKOM Nuclear Engineering Division, South Africa, 1987

- Determined the shielding requirements and cask type (in terms of the IAEA Safety Series 6 regulations) for a radioactive sample transport cask.
- Assisted in project management of high density fuel storage rack installation at KNPS.
- Supervised fuel loading in spent fuel building at KNPS (1984).

Lawrence K. Lee

Lead Senior Engineer, Probabilistic Safety Assessment and Reliability

AREAS OF EXPERTISE

- *Probabilistic Safety Assessment*
- *Maintenance Rule*
- *Shutdown Safety*
- *On-line Maintenance*
- *Fault Tree Analysis*
- *Event Tree Analysis*
- *Severe Accident Analysis*
- *PSA Compliance*
- *Equipment Survivability*

EDUCATION

B.S. Mechanical Engineering, University of California, Berkeley

WORK EXPERIENCE SUMMARY

Mr. Lee is employed as a Senior Engineer with ERIN. Mr. Lee has over 8 years experience in the nuclear field specializing in Probabilistic Safety Assessment. Mr. Lee has experience in providing support for Individual Plant Examinations (internal and external events), Maintenance Rule implementation, shutdown safety assessment, On-line Maintenance, MOV prioritization, and utility response to NRC compliance using PSA techniques.

WORK EXPERIENCE

Mr. Lee holds a Bachelor of Science degree in Mechanical Engineering from the University of California, Berkeley. He is responsible for providing support in the areas of Probabilistic Safety Assessment (PSA), Maintenance Rule implementation, Shutdown Safety Assessment, On-line Maintenance, and Level 2 Individual Plant Evaluations (IPE).

While at ERIN, Mr. Lee has participated in PSA projects involving fault tree and event tree analysis (linked fault tree methods and RISKMAN methods), thermal-hydraulic evaluations using the Modular Accident Analysis Program (MAAP) code, and containment safety studies during severe accident conditions. Mr. Lee's PSA experience includes contributions to the Peach Bottom, Limerick, Nine Mile Point Units 1 and 2, Vermont Yankee, and Duane Arnold Level 2 IPE projects.

Mr. Lee participated in the Update of the Quad Cities, Dresden and LaSalle PSAs. These projects included update and documentation of system models, accident sequence analysis, system notebooks to incorporate plant specific and BWR design basis data.

Mr. Lee has experience in applying the EPRI methodology for risk-informed evaluation of piping systems at the Quad Cities, Dresden and LaSalle stations. These projects included using PRA techniques and insights to identify risk important piping segments, define the elements that are to be inspected within this risk important piping, evaluate the risk impacts of proposed changes to the inspection program, and identify appropriate inspection methods.

Mr. Lee has extensive experience in using PSA techniques to comply with NRC requirements. Mr. Lee has modified plant specific PSA models in support of utility response to GL 89-10 MOV prioritization, the In-Service Testing Program, and the Maintenance Rule.

Mr. Lee has experience in the development of risk rankings for plant System, Structures, and Components (SSCs) for Maintenance Rule Expert Panel evaluations. In addition, Mr. Lee has experience in reviewing Maintenance Rule Performance Criteria and assessing their impact on plant PSA models for the Duane Arnold Energy Center and Diablo Canyon plants.

Lawrence K. Lee
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SECURITY CLEARANCE

U.S. Citizen

LICENSES/REGISTRATIONS/ PROFESSIONAL SOCIETIES

American Society of Mechanical Engineers

American Nuclear Society

Mr. Lee has experience in using PSA techniques to support On-line Maintenance safety evaluations for the Duane Arnold, WNP-2, and Fitzpatrick On-line Maintenance Programs. In addition, Mr. Lee has converted the fault tree/event tree based PSA models for the Duane Arnold and WNP-2 plants into large fault tree models to facilitate rapid solution times for supporting On-line Maintenance safety evaluations.

Mr. Lee has experience working with the Atomic Energy Control Board (AECB) in performing an independent review of the Pickering A Risk Assessment (PARA). This review included an evaluation of the PARA quantification methodology, which used the SETS and CAFTA codes to calculate the risk of fuel damage for the Pickering A CANDU reactor design.

Mr. Lee has experience in Probabilistic Shutdown Safety Assessment (PSSA). Mr. Lee developed fault tree and event tree models for the safety analysis of Duane Arnold refueling outage RFO 12. In addition, Mr. Lee has experience using the Outage Risk Assessment and Management (ORAM) Software for the Nine Mile Point Unit 2, LaSalle, Duane Arnold, Quad Cities, and Fermi 2 Shutdown Safety Assessment projects.

Mr. Lee has extensive experience in reviewing plant operating procedures as part of various IPE, IPEEE, ORAM and SENTINEL projects. As a result of these reviews, Mr. Lee has provided input to improvements in plant procedures, technical specifications and supplementary training plans.

The SENTINEL model development is used to support both the probabilistic and defense-in-depth evaluation required by the Maintenance Rule.

Mr. Lee has performed the quantitative evaluation of the Limerick (BWR/4 Mark II) and Peach Bottom (BWR/4 Mark I) plants using the REBECA event tree and fault tree code. This quantification involved linking the entire Level 1 cutsets to the entire Level 2 event tree/fault tree model and creating binned sequences by release category.

Mr. Lee has experience in developing radionuclide release bin rules using the RISKMAN code for the Nine Mile Point Unit 1 and Vermont Yankee Level 2 IPE projects.

Mr. Lee has performed the qualitative interpretation of containment failure modes performed by CB&I to obtain information usable in the probabilistic assessment of containment survivability.

Mr. Lee has developed a method of identifying accident release timing to the Emergency Action Levels. Mr. Lee has specialized in the assessment of equipment survivability under severe accident conditions.

Bengt O. Y. Lydell

Supervisor

WORK EXPERIENCE SUMMARY

Mr. Lydell has over 25 years experience years of focused risk and reliability analysis experience and is a Risk Spectrum PSA license holder. He supports European and U.S. chemical process, refining, offshore and energy industries with reliability and risk analysis services. Mr. Lydell's specialties include piping reliability, system reliability analysis, human reliability analysis, fault tree analysis, root cause analysis, reliability data analysis, quantitative risk assessment, and risk management.

AREAS OF EXPERTISE

- ***Probabilistic Safety Assessment***
- ***Risk Spectrum PSA***
- ***Human Reliability Analysis***
- ***Chemical Process Safety***
- ***Oil Refinery Risk & Reliability Analysis***
- ***System Reliability Analysis***
- ***Piping Reliability***
- ***Fault Tree Analysis***
- ***Root Cause Analysis***
- ***Reliability Data Analysis***
- ***Quantitative Risk Assessment***
- ***Risk Management***

WORK EXPERIENCE

Supervisor at **ERIN® Engineering and Research, Inc.** Currently, Mr. Lydell is the technical lead on the data analysis and human reliability analysis (HRA) tasks of the Watts Bar Ferry Nuclear Plant PSA Update. For TVA, he performed the 1999-2000 Browns Ferry HRA update. For Commonwealth Edison he supported the 1999 Byron and Braidwood PSA update. Member of the ERIN piping reliability analysis team. In February 2000, Mr. Lydell prepared a draft Technical Document (TECDOC) on passive component reliability data for the International Atomic Energy Agency.

Prior to joining ERIN, Mr. Lydell worked as a private consultant serving clients in the U.S. and Europe. With financial support from the Swedish Government (via the Swedish Nuclear Power Inspectorate, SKI), Mr. Lydell has been the principal investigator of a major research project on piping reliability. The list of clients (1993-1998) included:

- Ultramar Wilmington Refinery
- Universal Foods
- The International Atomic Energy Agency (IAEA, Vienna)
- Swedish Nuclear Power Inspectorate
- Barsebäck Kraft AB
- Ringhals AB (Ringhals Nuclear Power Plant, Sweden)
- Nordic Liaison Committee for Atomic Energy
- ABB Reaktor G.m.b.H., Mannheim (Germany)
- Paks Nuclear Power Plant, Hungary
- Vereinigung der Großkraftwerksbetreiber e.V. (VGB), Germany
- Slovenské Elektrárne a.s, Slovakia
- Nuclear Power Plant Dukovany, Czech Rėpublic
- NUPEC, Institute of Human Factors, Japan

Mr. Lydell developed an extensive database on the service experience with ASME Class 1, 2 and 3 piping systems in commercial nuclear power plants worldwide. Currently (July, 2000), this database includes over 3,700 significant piping degradations and failures encompassing 6400 reactor operation years. In addition to major, catastrophic failures (i.e., ruptures), this database also includes data on significant degradations (i.e., cracking in the through-wall direction, wall thinning), and small-to-large leaks.

Bengt O. Y. Lydell

EDUCATION

Chalmers University of Technology, Gothenburg, Sweden, Postgraduate studies in Probabilistic Mechanical Design under Prof. E.B. Haugen.

Chalmers University of Technology, Gothenburg, Sweden, MS Mechanical Engineering with Majors in Nuclear Engineering, Energy Conversion and Thermodynamics.

SECURITY CLEARANCE

Swedish Citizen

Permanent U.S. Resident

LICENSES/REGISTRATIONS/ PROFESSIONAL SOCIETIES

American Society for Quality (ASQ); including ASQ Reliability Division

Scandinavian Reliability Engineers (ScRE) - Co-founder

Specifically developed to meet the requirements of practical applications involving PSA applications (e.g., risk-based/risk-informed inservice inspection), the database was developed in MS-Access.

A framework for interpreting and analyzing service data builds on the concepts of piping reliability *attributes* and piping reliability *influence factors*. During the fall of 1997, the R&D was peer reviewed by Prof. Roger Cooke, Technical University of Delft, the Netherlands. A pilot project to estimate plant-specific piping reliability parameters (frequency of rupture and large leaks) for ASME Class 1 & 2 piping in Barsebäck-1 was completed in April 1999.

Mr. Lydell has a detailed working knowledge of the risk-based/risk-informed nuclear and onshore/offshore regulatory regimes of Denmark, Finland, France, the Netherlands, Norway, the United Kingdom, and the United States. He is currently finalizing the manuscript to major text on the 'Quality Assurance of Risk & Reliability Analysis' (Springer Verlag, Heidelberg, Germany). This text includes extensive coverage of the risk regulations and their impact on the analysis of risk and reliability of industrial facilities; e.g., how do we verify and validate the results to be applied in a safety case? Mr. Lydell was a contributor to a new guide on the Quality Assurance of Probabilistic Safety Assessment published by the IAEA in August 1999.

Halliburton NUS Corporation, Energy Group (1984 - 1993) - Senior Executive Engineer in system reliability, human reliability, and risk analysis. Involved in applied systems reliability and risk analysis for the nuclear and chemical process industries. Also involved in the development & application, and technology transfer of advanced PC-based software for reliability and risk analysis. Instrumental in the development of three commercial software products (CHEM-FT, NUSSAR-II and NUPRA); these products were based on RELTREE/Risk Spectrum.

Pickard, Lowe and Garrick, Inc. (1982 - 1984) - Consultant in risk and reliability analysis for U.S. and European nuclear industries; R&D as well as practical applications. Participated in the development of one of the earliest PSAs for the low power and shutdown modes of operations (published as NSAC-84). Assisted with business development in Scandinavia and Switzerland.

Swedish Nuclear Power Inspectorate (1980 - 1982) - Staff Engineer with responsibilities for reliability analysis, methods development and incident investigation. Provided internal training of nuclear inspectors in risk and reliability methods. Initiated and monitored research projects in risk and reliability analysis. Actively involved with the development and implementation of the 'ASAR' risk management program; ASAR preceded the conceptually similar U.S. Individual Plant Examination program by 7 years.

Chalmers University of Technology (1975 - 1980), Assistant Professor. Lecturer in nuclear engineering and systems reliability. Performed research on dependent failure analysis and systems reliability optimization with grants from the Swedish Nuclear Power Inspectorate and Swedish Energy Research Foundation. For the School of Mechanical Engineering, developed and implemented a graduate course in system reliability engineering.

Royal Dutch Shell (1973), at the central laboratory in Amsterdam, performed radiative heat transfer research on a vertical test furnace. This work resulted in the development of an analytical model for heat transfer in the corners of industrial furnaces (rectangular cross sections).

Selected Nuclear Probabilistic Safety Assessment (PSA) Experience

In 1997, for the European Commission (EC DG 1A), and in cooperation with KEMA and WESE, member of the project team for the Bohunice and Dukovany 'Low Power and Shutdown PSAs'; period of performance is July 1997 to September 1998. In 1994, for Swedish Nuclear Power Inspectorate, supported an independent peer review of the Ringhals 1 PSA. During 1992 - 1993, provided on-site support to the Mülheim-Kärlich Probabilistic Safety Assessment Project (Level 1+) in Mannheim, Germany. This support included system reliability analysis (fault tree analysis of low-head, high-head, recirculation and residual heat removal systems) using the Risk Spectrum PSA software, internal flooding analysis, accident sequence quantification (small, medium & large LOCA), and review.

In 1991, for Southern California Edison, provided PSA application services including support to the High Energy Line Break PSA for SONGS-1. In 1990, for Arizona Public Service, validated a computer code for tornado missile analysis. In 1987, for the Swedish Nuclear Power Inspectorate, performed a survey and evaluation of initiating events in ten U.S. PSA-studies in support of the Swedish "SUPER-ASAR" project. In 1986, system analyst on the Caorso Probabilistic Safety Study project. For Southern California Edison, participated in the Systematic Evaluation Plan (SEP) related ECCS reliability study for SONGS-1. For Japan NUS Co., performed severe accident R&D surveys. In 1985, task leader on the EPRI-funded seismic source term project. For Brunswick BWR plant, provided analytical support to a technical specification re-evaluation of a diesel generator system using the FRANTIC-III software (test interval optimization).

During 1982 - 1984, for EPRI/NSAC, participated in a plant specific PSA on cold shutdown operations (NSAC-84), and was responsible for the data analysis and the initial accident sequence quantification. For Consumers Power, developed a complete electric power system model (AC,DC and emergency power) using GO methodology for inclusion in the LIMCOM technical specification software. For EPRI, participated in the development of common cause failure data on pumps and motor operated valves (EPRI NP-3967).

In 1983, for the Swedish Nuclear Power Inspectorate, performed a review of the Ringhals-2 PSA. During 1980 - 1982, member of the steering committee for the safety analysis of FILTRA (a filtered vented containment concept developed for the Swedish Barsebäck nuclear power plant).

General Reliability Engineering Experience

In 1993, for Korea Power Engineering Company, Inc. (KOPEC), supported the Ulchin 3 & 4 reliability engineering program with development of a data base for a reliability critical items list (RCIL). In 1990, for Korea Electric Power Operating Service Company, provided training in basic reliability theory and in fault tree analysis as part of an eight-week training program. As a subcontractor, for the U.S. Air Force, performed reliability analysis of an electric power system.

In 1984, for Southern California Edison, helped prepare a "Performance Incentive Procedure Study" that involved analysis of plant and system-level availability data for U.S. PWR plants with NSSS by Combustion Engineering; this study also addressed the Balance-of-Plant (BOP) systems (especially English Electric turbine generator operating experience).

In 1982-1981, led the Swedish contribution in the first European Reliability Benchmark Exercise. Member of the steering committee for the development of the Swedish Reliability Data Book. Developed a probabilistic concept for the evaluation of licensee event reports (PSA-based event analysis).

In 1981, served on a review panel for the Nordic Research Project (LIT) on human reliability. Lecturer in systems reliability at the Royal Institute of Technology in Stockholm (Sweden). In 1979, co-founded the Scandinavian Reliability Engineers (ScRE), a professional society that actively promotes the risk and reliability disciplines in the Nordic countries. During the period 1976-1979, performed theoretical research on dependent failure modeling with grants from the Swedish Energy Commission. During the period 1974 - 1976, worked on the development of computerized work order systems for the Swedish Nuclear Industry. Maintained a computerized nuclear plant availability tracking system.

Human Reliability Analysis (HRA) Experience

In 1999-2000, HRA task leader on the Browns Ferry PSA Update Project. In 1998-99, HRA task leader on the Byron/Braidwood PSA Update Project. On behalf of the Institute of Human Factors (operated by NUPEC, Japan), as a member of the human factors research advisory group developed HRA R&D recommendations. For the Swedish Nuclear Power Inspectorate, performed a study on undetected latent errors in safety systems. In 1997-98, HRA task leader on the Bohunice and Dukovany shutdown PSAs. In 1994, performed an HRA of operator response to accidental hydrogen fluoride releases for a U.S. oil refinery. In 1992, provided the Swedish Nuclear Power Inspectorate with HRA support including transfer of insights from performing HRAs in support of the U.S. Individual Plant Examination program.

During 1990 - 1991, provided Southern California Edison with HRA services to resolve licensing issues, and support of PSA-projects. During 1989 - 1991, task leader for human reliability analysis (HRA) in the Peach Bottom, Surry, North Anna, Perry and Indian Point 2 Individual Plant Examinations (IPEs). For the Surry IPE, developed simulator experiments to generate data on crew responses to LOCA and ATWS scenarios.

In 1990, was HRA task advisor for the Borssele PSA project. Contributor to the EPRI-sponsored HRA-procedures (EPRI NP-6560-L), and the EPRI project "Accident Sequences for Training" (RP3050-1), directed to the development of guidelines for PSA-based simulator training plans. For the EPRI Nuclear Power Division, prepared a human reliability perspective on cold shutdown operations.

In 1989, served on the NUCLARR human reliability review panel. In 1988, was the project manager for the HRA-portion of the Ringhals 2 PSA update. Provided technology transfer to the Swedish State Power Board. The Ringhals-2 HRA included consideration of the then new Severe Accident Management Guidelines (SAMGs). For Japan NUS Co., provided surveys of the state-of-art in HRA including reviews of the then new Swedish SAMGs as implemented at the Ringhals Nuclear Power Plant (this station comprises one ABB-Atom BWR and three 3-loop Westinghouse PWRs). Also in 1988 and with emphasis on operator actions in response to ATWS sequences performed HRAs in support of the Limerick Generating Station PSA update.

In 1987, for the Central Electricity Generating Board (CEGB) in England performed an hierarchical task analysis (human factors evaluation) of the Sizewell 'B' nuclear power plant. For the Swedish Nuclear Power Inspectorate analyzed human reliability aspects of the pressurized thermal shock (PTS) issue. For Philadelphia Electric Company, provided HRA-support to the Limerick PSA update. Further, in the early part of 1987 provided HRA-support to the Industrial Power Company in Finland and its TVO-I/II Level 1 PSA project, including on-site technology transfer. Was a member of the U.S. team participating in the European Human Factors Reliability Benchmark Exercise (HF-RBE). Member of the EPRI/ORE team collecting operator performance data using full-scale training simulators.

In 1986, for the Swedish Nuclear Power Inspectorate, performed a detailed HRA of the 'back-flush' operations (a unique form of ECCS recirculation operation) in a 2nd generation BWR plant (Ringhals-1) designed by ABB-Atom.

In 1985, contributed to the development of the HRA training program prepared for the Pennsylvania Power & Light Company. For EPRI, reviewed the HRA-portions of Oconee, Seabrook and Shoreham PRA studies. In 1984, for the Swedish Nuclear Power Inspectorate prepared a state-of-the-art survey of HRA techniques in U.S. PSAs.

Selected Oil Refinery & Chemical Process Industry Risk & Reliability Analysis Experience

Maintains a database on mechanical equipment reliability that includes over 3,700 failure reports; including data on pumps, compressors, ball valves, piping. In 1994-95, for a Ultramar Wilmington Refinery, performed a risk assessment of three HF Acid Isolation & Evacuation System concepts. This study included an innovative, source-term oriented approach to initiating event characterization and quantification, and incident response model development. Also included was a detailed pipe segment-by-segment model of the entire HF alkylation processing unit. The piping reliability estimation process utilized data on refinery pipe inspection histories.

In 1993, also for Ultramar Wilmington Refinery, performed Butamer compressor reliability predictions and FCC equipment reliability assessments. Further, performed an update of the HF Alkylation Unit QRA for the triennial review of the HF Risk Management and Prevention Plan. Provided training in modern incident investigation and root cause analysis.

In 1996, for Universal Foods (Baltimore, MD) performed an ammonia system process hazard assessment. In 1993, for San Diego Refrigerated Services, performed a risk-benefit evaluation of HAZOP-based recommended action items. In 1992, for Exxon Production Research Company, Houston (TX), supported a pilot project to develop a data base for 'Fatal Injury Frequency Rate' using incident reports for offshore and land-based operations.

In 1991, for Center for Chemical Process Safety (CCPS) of the American Institute of Chemical Engineers (AIChE), co-author of the "Guidelines for Investigating Chemical Process Incidents". These guidelines were published in 1992 by the American Institute of Chemical Engineers (AIChE, ISBN 0-8169-055-X). Also for CCPS, developed and presented a tutorial on "Process

Safety Incident Investigation", with emphasis on methods for root cause analysis. For the Union Carbide, Health Safety and Environmental Technology Group, South Charleston (WV), developed and delivered a three-day training course in fault tree analysis. For EG&G Idaho, Inc. and Rockwell-INEL, Idaho National Engineering Laboratory, instructor in HAZOP Leader training courses.

In 1990, for Amoco Production Company, Houston (TX), performed process hazards analyses (PHAs) of the Floating Production Storage and Offloading Tanker (FPSO) and the Inde Gas Dehydration Platform. For IT Corporation, provided training in fault tree analysis. For Manville Sales Corporation, supported the HAZOP of the ammonia and sulfuric acid circuits.

In 1989, for Exxon Company, U.S.A., member of the HAZOP-team for the Santa Ynez Unit (SYU) Expansion Project. For Exxon Production Research Company, prepared an overview of the development and application of qualitative and quantitative acceptance criteria for risk analysis. For Systech, performed a human factors review of operating and emergency procedures for a new pyrolyzer facility, and performed an on-site safety audit of the facility at Fredonia (KS).

In 1988, developed and reviewed accident scenarios and fault trees as part of the NUS-certification of the RMPP for the Chevron-operated Point Arguello Oil and Gas Processing Plant and adjoining pipelines.

PUBLICATIONS

Selected Recent Papers and Reports

"Pipe Failure Probability – The 'Thomas Paper' Revisited," *Reliability Engineering and System Safety*, **68/3**:207-217 (July 2000).

Fleming, K.N. and B.O.Y. Lydell, *Evaluation of Turbine Building Pipe Rupture Frequencies and Inspection Strategies at the LaSalle Nuclear Generating Station*, prepared for Commonwealth Edison, Downers Grove, IL (June 2000).

Browns Ferry Nuclear Plant Human Reliability Analysis, prepared for Tennessee Valley Authority, Athens, AL (May 2000).

PUBLICATIONS (cont'd)

Byron and Braidwood Human Reliability Analysis - Validation of HRA Assumptions, prepared for Commonwealth Edison, Downers Grove, IL (April 2000).

Reliability Data for Passive Components, IAEA-99CT11858, International Atomic Energy Agency, Vienna, Austria (February 2000).

Failure Rates in Barsebäck-1 Reactor Coolant Pressure Boundary Piping. SKI 98:30 Appendix H: Barsebäck-1 Piping Reliability Database, RSA-R-99-01P, prepared for BKAB, Sweden (September 1999).

A Framework for a Quality Assurance Programme for PSA, IAEA-TECDOC-1101, International Atomic Energy Agency, Vienna Austria (August 1999) (contributor to the 2nd draft).

Leak & Rupture Frequencies in Barsebäck-1 Reactor Coolant Pressure Boundary Piping. Results & Insights from a R&D Project to Derive LOCA Frequencies Applicable to a 3rd Design Generation ABB-Atom BWR," _Proc. PSA '99_, American Nuclear Society, LaGrange Park, IL (August 1999).

Failure Rates in Barsebäck-1 Reactor Coolant Pressure Boundary Piping, SKI Report 98:30[1], Swedish Nuclear Power Inspectorate, Stockholm, Sweden, (May 1999).

Independent Review of Report GES-138/98: Ringhals-1. Dynamic Effects of Postulated Pipe Breaks Inside Containment. Description of Study Method, RSA-R-99-07, prepared for Ringhals AB, Väröbacka, Sweden (April 1999).

"Systematic Evaluation of Service Data on Piping: A Framework for Effective Aging Management," Paper #6303, _Proc. ICONE-6: 6th International Conference on Nuclear Engineering_, American Society of Mechanical Engineers, New York, NY (May 1998).

Some Views on the Role of Human Factors Empirical Studies in Probabilistic Safety Assessment, RSA-R-98-01, prepared for Nuclear Power Engineering Corporation (NUPEC), Tokyo, Japan, January 1998.

Reliability of Piping System Components: Framework for Estimating Failure Parameters from Service Data, SKI Report 97:26, Swedish Nuclear Power Inspectorate, Stockholm, Sweden (December 1997).

"A Practitioner's View on the State of HRA Methodology," _Reliability Engineering and System Safety_, **55**:257-260.

"On the Estimation of Piping System Component Reliability Parameters from Operating Experience," _Proc. 14th International Conference on Structural Mechanics in Reactor Technology (SmiRT-14)_, Lyon, France, August 17-22, 1997, **10**:1-8.

"Operational Data for Human Reliability Analysis," _Development in PSA Data in Eastern and Central Europe_, IAEA-RU-6964 (TC Project RER/9/046), International Atomic Energy Agency, Vienna, Austria (April 1997), pp 254-262.

[1] Available on the Internet at: www.ski.se

PUBLICATIONS (cont'd)

"Incorporation of Piping System Failures in PSA: Some Basic Analysis Considerations," *Proc. PSA '96*, American Nuclear Society, (October 1996), pp 1881-1887.

Reliability of Piping System Components. Vol. 4: The Pipe Failure Event Database, SKI Report 95:61, Swedish Nuclear Power Inspectorate, Stockholm, Sweden (July 1996).

Reliability of Piping System Components. Vol. 1: A Resource Document for PSA Applications, SKI Report 95:61, Swedish Nuclear Power Inspectorate, Stockholm, Sweden (December 1995).

Full list of publications and work products provided on request by contacting Mr. Lydell via e-mail: ***bolydell@erineng.com***.

Donald E. MacLeod

Senior Engineer, Probabilistic Safety Assessment and Reliability

AREAS OF EXPERTISE

- ***Human Reliability Analysis***
- ***On-Line Maintenance***
- ***Probabilistic Safety Assessment***
- ***Shutdown Safety***
- ***Fault Tree Analysis***
- ***Event Tree Analysis***

WORK EXPERIENCE SUMMARY

Mr. MacLeod is employed as a Senior Engineer with ERIN. He has approximately 6 years of experience in the nuclear field specializing in Probabilistic Safety Assessment. Specific experience includes Human Reliability Analysis for full power and shutdown conditions, Probabilistic Shutdown Safety Assessment development, and systems analysis.

WORK EXPERIENCE

Mr. MacLeod holds a Bachelor of Science degree in Nuclear Engineering from Rensselaer Polytechnic Institute. Relevant work experience includes the following:

Served as the lead analyst in the development of the revised Quad Cities and Dresden HRAs as part of the PSA enhancement projects.

Co-developed the Human Reliability Analysis for the Lungmen PRA. This project required detailed analysis of plant functions, design, procedures, the PRA model, and interface with system engineers for the Lungmen plant.

Aided in the evaluation of the effects of Procedural modifications on the Limerick HRA.

Co-Author of "HRA Tailored for Risk Informed Decisions for Shutdown Safety". This paper, written in conjunction with Dr. E.T. Burns and L.K. Lee, documents the adaptation of the EPRI Cause-Based HRA Method for use in Probabilistic Shutdown Safety Analyses.

Performed shutdown HRAs for the following sites; LaSalle, Duane Arnold, Quad Cities, Zion, Dresden, Cooper, Peach Bottom, and Fermi 2. These analyses included an assessment of the clarity and completeness of the procedures, quantification of HEPs, and identification of any potential procedural improvements.

Update and revision of the NMP-2 data analysis as part of the PRA update program. Work included compilation of a plant specific failure database, recalculation of component failure probabilities, recalculation of system maintenance unavailabilities, and requantification of the common cause failure analysis using INEL-94/0064.

Assistance in various stages of the DAEC IPEEE program including the internal fire evaluation and "other" external events. In addition, work was done to revise the turbine missile ejection frequency for GE turbines in order to reflect the conclusions of an NRC study.

Co-developed the initial NMP-2 shutdown model using the Scientech software code "Safety Monitor".

Donald E. MacLeod

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EDUCATION

B.S. Nuclear Engineering, Rensselaer Polytechnic Institute

SECURITY CLEARANCE

U.S. Citizen

LICENSES/REGISTRATIONS/PROFESSIONAL SOCIETIES

American Nuclear Society

Participation in the development of Probabilistic Shutdown Safety Assessments (PSSAs) for the LaSalle, Duane Arnold, Quad Cities, Zion, Dresden, Cooper, Peach Bottom, and Fermi 2 plants using the Outage Risk Assessment and Management (ORAM) code. Assistance in the adaptation and application of the EPRI Cause-Based method for use in PSSA Human Reliability Analysis.

Development of detailed fault trees to support the ORAM evaluation of Quad Cities and Cooper.

Extensive quantification of event trees and fault trees for DAEC.

In summary, expert in the PRA field in the following areas:

- data synthesis
- data evaluation
- fault tree development
- quantification
- HRA
- Shutdown Model Development (ORAM)
- On-Line Maintenance Model Quantification (SENTINEL and Large Fault Tree Fast Solver)

Leo B. Shanley III

Supervisor

WORK EXPERIENCE SUMMARY

Mr. Shanley is a supervising nuclear professional with over nine years experience in the commercial power industry, principally in the area of nuclear power plant risk assessment. Mr. Shanley is currently the project manager for ORAM-SENTINEL implementation projects at the five Commonwealth Edison sites and the ORAM-SENTINEL v3.4 upgrade project. Mr. Shanley has extensive experience with software project management.

AREAS OF EXPERTISE

- ***Probabilistic Safety Assessment***
- ***ORAM-SENTINEL Modeling***
- ***Data Collection and Analysis***
- ***Shutdown Risk Assessment***
- ***System Reliability***
- ***Maintenance Rule***
- ***Human Reliability Analysis***
- ***Software Management***

WORK EXPERIENCE

Mr. Shanley is the Supervisor of the Operational Solutions Group. He is presently the project manager for the ORAM-SENTINEL implementation project for all five ComEd sites. These projects involve the integration of probabilistic (PSA) and deterministic evaluations into an overall on-line and shutdown risk assessment, incorporating outage and at-power maintenance scheduling practices.

Mr. Shanley has significant experience with PWR and BWR outage risk assessments. He has been the project manager or supported ORAM model development and enhancement projects at Sizewell B (current project), NPP Krško (Slovenia), Three Mile Island, Watts Bar, D.C. Cook, Sequoyah, Indian Point 3, Virginia Power (North Anna and Surry), Cooper, Peach Bottom, Diablo Canyon, and Calvert Cliffs.

Mr. Shanley has reviewed outage schedules and performed shutdown safety evaluations (both deterministic and probabilistic) for several plants, including Fort Calhoun Station, Calvert Cliffs and NPP Krško. Mr. Shanley is co-author of the EPRI Shutdown Initiating Event Analysis Report (TR-113051).

Mr. Shanley has led or supported on-line ORAM-SENTINEL implementation at Diablo Canyon, Watts Bar, Browns Ferry, Calvert Cliffs Projects and Indian Point 3. These projects involved integrating the sites PRA with the ORAM-SENTINEL model.

In addition to his experience in ORAM-SENTINEL model development, Mr. Shanley has significant experience in software management principally in the ORAM-SENTINEL software development effort. Mr. Shanley was the Project Manager for ORAM-SENTINEL version 3.3 and is currently leading the effort for Maintenance Rule (a)(4) enhancements to ORAM-SENTINEL (v3.4). He is a principle contributor to the ORAM-SENTINEL Validation and Verification Test Plan and has considerable experience beta-testing the software.

Mr. Shanley has been involved in data and systems analysis for several PRA updates while at ERIN, including PECO's Limerick Generating Station and ComEd's Quad Cities. Prior to joining ERIN, Mr. Shanley was the project manager for the Calvert Cliffs IPE, and performed plant specific data collection and analysis in support of that project.

As Project Manager for the Calvert Cliffs Maintenance Rule System Scoping and Performance Criteria Review Project, Mr. Shanley led the effort to review the adequacy of the Calvert Cliffs Maintenance Rule risk significant system scoping and performance criteria development.

Leo B. Shanley III

EDUCATION

B.S. Materials Engineering, Cornell University

U.S. Navy Nuclear Power School and Prototype

SECURITY CLEARANCE

U.S. Citizen

Mr. Shanley was the lead engineer in a project to upgrade the Calvert Cliffs PRA. This involved review of over twenty System Analyses, incorporation of Human Action dependencies and restructuring of the Plant Model using RISKMAN software.

Mr. Shanley previously held the position of Senior Engineer with Baltimore Gas & Electric (BGE) at the Calvert Cliffs Nuclear Power Plant. During his five years at Calvert Cliffs, Mr. Shanley was assigned to the Reliability Engineering Unit, where he attained the position of Work Group Leader for the PRA Work Group.

At BGE, Mr. Shanley was the Project Manager for the Calvert Cliffs Seismic Probabilistic Risk Assessment (PRA). Duties included development of critical component list, walk-down coordination and incorporation of fragility results into the Calvert Cliffs PRA model. In this capacity, Mr. Shanley managed the computer simulation program development.

Mr. Shanley was the Project Manager for the Calvert Cliffs Individual Plant Examination (IPE), responsible for the closeout and documentation of the project. He supervised four engineers in finalizing the results of the IPE, including the technical review of most aspects of the IPE. Mr. Shanley was a principal contributor to the two volume summary report.

Mr. Shanley has been involved in various risk analyses in support of licensing and operations issues at Calvert Cliffs. These include Shutdown Risk Assessments using ORAM software, development of system Performance Indicators in support of the Maintenance Rule, on-line risk assessments, tornado risk analysis, and risk-based Allowed Outage Time extensions.

Mr. Shanley has extensive experience in data collection and analysis. At BGE, Mr. Shanley was responsible for all aspects of data collection for the Calvert Cliffs PRA. This includes industry and plant specific failure, success, initiating event, unavailability, and common cause data. He gained proficiency in the use of INPO's Nuclear Plant Reliability Data System (NPRDS). Mr. Shanley was also involved in developing methods and processes for on-going data collection to support the living PRA and Maintenance Rule.

As a member of the Combustion Engineering Owner's Group - Probabilistic Safety Assessment Working Group, Mr. Shanley provided significant contributions to the group's efforts in developing risk-based regulatory applications.

Prior to his commercial nuclear positions, Mr. Shanley spent seven years in the U.S. Navy. During his Naval career, Mr. Shanley held various positions of increasing responsibility on two nuclear powered surface ships. He was principally involved in the operation and maintenance of nuclear propulsion plants and other engineering systems. Mr. Shanley completed qualifications to be certified as the Engineering Department Head for a Naval Nuclear Propulsion Plant. During new construction, he served as Shift Officer and Shift Senior Supervisory Watch for all phases of systems and reactor testing.

Donald E. Vanover

Manager

WORK EXPERIENCE SUMMARY

Don Vanover is a Manager with nearly fifteen years of experience serving the commercial nuclear power industry. Mr. Vanover was extensively involved in the completion of numerous Individual Plant Examinations (IPEs) both in the U.S. and in Spain. Prior to that, he assisted with the testing, development, and benchmarking of many of the models contained in the Modular Accident Analysis Program (MAAP). Since joining ERIN, Mr. Vanover has completed a variety of tasks including numerous at-power PSA model updates and analyses, as well as Probabilistic Shutdown Safety Assessments for both BWR and PWR Plants. Additionally, he has acted as project manager for several Software Development projects.

AREAS OF EXPERTISE

- ***Probabilistic Safety Assessment***
- ***Thermal-Hydraulic Analysis***
- ***System Response***
- ***ORAM-SENTINEL Modeling***
- ***MAAP Modeling (BWR and PWR)***

WORK EXPERIENCE

Mr. Vanover is presently the Manager of Technical Solutions at **ERIN Engineering and Research, Inc.** in the Philadelphia area office. He has extensive knowledge of systems and procedures for both BWR and PWR reactors and has the ability to apply his hands-on and theoretical background to understand and analyze system response. Since joining ERIN, Mr. Vanover has contributed to the successful completion of several PSA-related projects including major updates to the Peach Bottom PSA model and the completion of several shutdown PSA models. He also performed the fire risk analysis portion of the Peach Bottom IPEEE, and has been extensively involved in the use the Peach Bottom and Limerick PSA models to support on-line maintenance activities. Mr. Vanover has also been a major contributor and project manager on several ORAM-SENTINEL plant-specific modeling projects.

While with **Gabor, Kenton, and Associates, Inc.**, Mr. Vanover performed, documented, and supplied technology transfer of engineering and thermal-hydraulic analysis for numerous nuclear utilities for their Individual Plant Examination submittals to the NRC (or equivalent). He also served as an instructor for severe accident phenomenology and Modular Accident Analysis Program (MAAP) training courses both in the U.S. and abroad.

Mr. Vanover conducted thermal-hydraulic analyses for use in the industry sponsored initiative for Steam Generator Alternate Repair Criteria and provided station blackout analyses for the Salem plant in support of PSE&G's response to NUMARC 87-00. Mr. Vanover developed MAAP source code modifications for the Trillo and Zorita plants in Spain. Additionally, he was responsible for evaluating available experimental data to perform a multitude of comparisons and sensitivity runs with both the BWR and PWR versions of MAAP.

At **Fauske & Associates, Inc.**, Mr. Vanover successfully headed the development of new models and organized an entire code to provide a severe accident analysis tool for CANDU reactors. Prior to that, he performed benchmark calculations for thermal-hydraulic transient analysis using the MAAP BWR and PWR codes.

Donald E. Vanover

EDUCATION

M.S. Mechanical Engineering, University of Delaware

B.S. Mechanical Engineering, University of Delaware

SECURITY CLEARANCE

U.S. Citizen

LICENSES/REGISTRATIONS/ PROFESSIONAL SOCIETIES

American Nuclear Society

PUBLICATIONS

Selected Papers and Reports Listed

Before attending graduate school, Mr. Vanover worked as a Mechanical Maintenance Supervisor for **Bethlehem Steel Corporation** at the Sparrows Point, Maryland plant. There, he assigned daily tasks to millwrights, pipefitters, and welders, and was responsible for maintaining pumps, compressors, condensers, heaters, and turbines for an on-site power plant.

Selected Papers and Reports

Experimental Study of Mixed Convection in a Horizontal Porous Annulus, D.E. Vanover and F.A. Kulacki, ASME Winter Annual Meeting, Boston, December 1987.

Simulation of the Semiscale Mod-2C SBLOCA Using MAAP-DOE, D.E. Vanover, C.D. Wu, M.A. Kenton, and R.J. Hammersley, ANS/ENS International Conference Session on Severe Accident Thermal-Hydraulics, Washington, D.C., November 1988.

Station Blackout MAAP Analysis for the Salem Generating Station - Units 1 & 2 in Support of NUMARC 87-00, GKA/91-2, April 1991.

MAAP Modifications to Represent the Zorita and Trillo Plants, GKA/91-3, December 1991.

MAAP Thermal-Hydraulic Qualification Studies, J.Gabor, J.Healzer, M.Kenton, G. Lellouche, and D. Vanover, Final Report, EPRI TR-100741, June 1992.

BWR Accident Management Insights for Containment Flooding, E.T. Burns, J.R. Gabor, T.P. Mairs, and D.E. Vanover, PSA International Topical Meeting, Clearwater Beach, January 1993.

Millstone Unit 1 Plant Vulnerabilities During Postulated Severe Nuclear Accidents, Y.F. Khalil, J.R. Gabor, and D.E. Vanover, ASME International Conference on Nuclear Engineering, San Francisco, March 1993.

Development of Accident Management Guidance, E.T. Burns, D.E. Vanover, J.R. Gabor, and T.P. Mairs, ASME International Conference on Nuclear Engineering, San Francisco, March 1993.

Implications of the NRC Sponsored MAAP3.0B Code Evaluation as Applicable to the Nine Mile Point 1 IPE, D.E. Vanover, J.R. Gabor, and E.T. Burns, GKA/93-1, November 1993.

Insights on the Use of a Large Cut Set Equation to Quantify Risk Associated with Different On-Line Maintenance Configurations, G.A. Krueger, D.E. Vanover, PSA International Topical Meeting, Park City, Utah, October 1996.

Practical Uses of PSA in Support of the Maintenance Rule, S. Hess, D. Vanover, G. Krueger, Conference on Probabilistic Safety Assessment and Management, PSAM 4, New York City, September 1998.

Automated Shutdown PSA Model Development for the Peach Bottom Atomic Power Station, J.T. Wilson, D.E.MacLeod, L.B.Shanley, and D.E. Vanover, PSA International Topical Meeting, Washington, D.C, August 1999.

TECHNICAL INPUT FOR USE IN THE MATTER OF SHEARON HARRIS SPENT FUEL POOL BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

(DOCKET No. 50-400-LA)

NOVEMBER 2000

TECHNICAL INPUT FOR USE IN THE MATTER OF SHEARON HARRIS SPENT FUEL POOL BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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Prepared by: E.T. Burns J.R. Gabor Date: 11/16/00 11/16/00
E.T. Burns & J.R. Gabor
ERIN Engineering and Research, Inc.

Reviewed by: Karl N. Fleming Date: 11/16/00
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Approved by: Date: 11/16/00
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Accepted by: Date: 11-16-00
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0	Original Issue			

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EXECUTIVE SUMMARY

Overview

A Probabilistic Safety Assessment (PSA) for the Shearon Harris Nuclear Power Plant (SHNPP) has been performed by ERIN to address a question posed by the Atomic Safety and Licensing Board (ASLB) in a Memorandum and Order dated August 7, 2000 (ASLB Order) in connection with Carolina Power & Light Company's (CP&L) license amendment request to expand spent fuel storage at SHNPP by placing spent fuel pools C and D in service. ERIN was asked by CP&L to determine the best estimate of the overall probability of the postulated sequence set forth in the following chain of seven events (referred to herein as the Postulated Sequence):

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

The analytical methodologies chosen by ERIN to determine the best estimate overall probability of the Postulated Sequence are characteristic of existing nuclear power plant PSAs (also referred to as probabilistic risk assessments (PRAs)). It has drawn on the available site specific results from the SHNPP Level 1 and Level 2 PSA and has been extended for the purpose of addressing the impact of severe accidents (steps 1 and 2) on the SHNPP spent fuel pools (SFPs). This analysis required the incorporation of the unique features of the SHNPP design, including the size and location of the Fuel Handling Building (FHB) and the multitude of SFP makeup systems and makeup pathways. Where site specific information was not available, applicable generic studies were used as appropriate.

The effort to determine the best estimate overall probability of the Postulated Sequence involved the formation of an analysis team (13 team members) and direct links to key CP&L staff. The CP&L staff provided both detailed calculations (including the Level 1 and 2 SHNPP PSA), system descriptions, interviews with operating personnel, on-site dose calculations, and procedure interpretations. The team effort included:

- Multiple SHNPP site visits to confirm the as-built design and crew response.
- An independent peer review of the inputs to the evaluation including the Level 1 and 2 SHNPP PSA.
- An independent review of the analysis report.

The total effort by ERIN personnel dedicated to the analysis exceeded one person-year of professional time during the period August through the date of this report in November, 2000.

Methodology

Important aspects of the PSA methodology in performing the analysis include the following actions:

- Provide a comprehensive examination of potential contributors to the Postulated Sequence. The methods used to characterize the severe accident frequencies vary with the type of challenge and the current state of PSA technology:

Internal Events	-	Full PSA methodology
Fire	-	Full PSA methodology for dominant IPEEE accident sequences
Seismic	-	Approximate method

Shutdown - Generic assessment based on similar PWR input frequencies

Other - Negligible contribution

- Calculate the plant response (adverse effects of radiation and steam-temperature) to severe accident conditions.
- Ensure that adverse conditions on-site are adequately addressed as they affect human performance and equipment survivability.
- Calculate the times available for actions to be taken in response to the challenges.
- Ensure that the characterization of the human performance addresses the critical performance shaping factors, which include:
  - Stress
  - Environment
  - Procedural adequacy
  - Access
  - Timing
- Characterize within the probabilistic framework the systems available to provide makeup to the SFP or SFP cooling under the Postulated Sequence.
- Incorporate CP&L direction to assume a conditional probability of step seven (exothermic oxidation reaction of the spent fuel to be stored in spent fuel pools C and D) equal to 1.0 because of uncertainties in the available analytical tools to model the projected heat balance in the spent fuel pools. CP&L chose to address the conservative nature of an assumed conditional probability of 1.0 for step seven of the Postulated Sequence.

<u>SHNPP PSA Quality</u>

The SHNPP PSA (Level 1 and 2 Internal Events) was subjected to an independent peer review as part of this evaluation. The independent peer review determined that the SHNPP PSA was robust, comprehensive, and consistent with the state-of-the-technology for such probabilistic assessments in the industry. The SHNPP PSA is fully supportive of risk-informed applications.

The SHNPP PSA for internal events demonstrates that the plant meets the NRC Safety Goals and their subsidiary objectives (i.e., Core Damage Frequency and Large Early Release Frequency). In addition, there are no unusual contributors to core damage frequency or containment failure.

Unique SHNPP Features

The Shearon Harris Fuel Handling Building (FHB) was constructed to accommodate a four unit site. The size and compartmentalization of the building enhances its accident response. These SHNPP FHB features have been explicitly represented in the deterministic calculation of post containment failure accident sequences. In addition, there are a substantial number of alternate systems and pathways for establishing water makeup to the SHNPP spent fuel pools.

Conclusions

The conclusions of the PSA to determine the best estimate overall probability of the Postulated Sequence can be summarized in a qualitative fashion based on the quantitative results and the sensitivity evaluations:

- The Postulated Sequence begins with severe accidents which are beyond the SHNPP Design Basis and are of low frequency.
- The design of the large SHNPP FHB, the multiple makeup water pathways, and multiple means of access to the FHB result in a high probability of recovery from a loss of spent fuel pool cooling before the spent fuel is uncovered.
- The best estimate frequency of the Postulated Sequence is considered extremely low and below what is reasonably considered "remote and speculative" or an acceptable societal risk.

- The addition of spent fuel pools C and D to SHNPP does not increase the frequency of the contributors to the Postulated Sequence. To the contrary, the plant modifications associated with placing spent fuel pools C and D in service actually decreases the frequency of spent fuel uncovering. This is related to the addition of alternate viable makeup pathways under nearly all postulated accidents with the installation of the redundant Spent Fuel Pool Cooling and Cleanup System (SFPCCS) for spent fuel pools C and D.

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

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

The point estimate contribution due to fire related initiating events was an order of magnitude less (2.94E-9/yr) than for internal events (2.65E-8/yr).

While the point estimate contribution due to seismic initiated events (8.65E-8/yr) is higher than for internal events (2.65E-8/yr), it is judged not to alter the conclusions reached based on the internal events analysis. Seismic initiated events are difficult to analyze for the Postulated Sequence because a seismic event less than the design basis earthquake cannot be an initiator of Steps 1 and 2, and a seismic event sufficient to cause a breach of the spent fuel pools is outside of the Postulated Sequence (because the loss of cooling to the spent fuel must be by evaporation (Step 6) and not a draining of the spent fuel pools due to a breach).

The annualized core damage probability associated with internal events during shutdown or refueling outages has been estimated to be the same order of magnitude as that associated with power operation. This analysis was based on generic studies rather than a site specific shutdown PSA, because shutdown internal events are not included in the SHNPP PSA.

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

A series of events with a frequency that is calculated to be on the order of 3E-8/yr. (i.e., a few chances in one hundred million per year) is not considered worthy of societal concern.

ACRONYMS AND INITIALS

ASLB	Atomic Safety and Licensing Board
BWR	Boiling Water Reactor
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CCW	Component Cooling Water
CDF	Core Damage Frequency
CDFM	Conservative Deterministic Failure Margin Method
CEUS	Central and Eastern United States
CS	Containment Spray
DFP	Diesel Fire Pump
DOE	U.S. Department of Energy
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EOPs/AOPs	Emergency Operating Procedures/Abnormal Operating Procedures
EPRI	Electric Power Research Institute
EQE	EQE Risk Management Company
ESW	Emergency Service Water
FHB	Fuel Handling Building
GIP	Generic Implementation Procedure
HCLPF	High Confidence of Low Probability of Failure
HVAC	Heating, Ventilation, And Air Conditioning
I&C	Instrumentation and Control
IE	Initiating Event
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
ISLOCA	Interfacing Systems Loss Of Coolant Accident
JPMs	Job Performance Measures

ACRONYMS AND INITIALS (Cont'd)

LERF	Large Early Release Frequency
LOCA	Loss Of Coolant Accident
LOSP/LOOP	Loss Of Offsite Power
MAAP	Modular Accident Analysis Program
MMI	Modified Mercalli Intensity
MOV	Motor Operated Valve
NEI	Nuclear Energy Institute
NRC	United States Nuclear Regulatory Commission
NSW	Normal Service Water
OBE	Operating Basis Earthquake
ORAM	ORAM-SENTINEL™ Computer Program
OSC	Operations Support Center
PCS	Power Conversion System
Pga	Peak Ground Acceleration
PMF	Probable Maximum Flood
PMWS	Primary Makeup Water System
POS	Plant Operating States
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PSHA	Probabilistic Seismic Hazard Analysis
PWR	Pressurized Water Reactor
QA	Quality Assurance
RAB	Reactor Auxiliary Building
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RLE	Review Level Earthquake
RPV	Reactor Pressure Vessel

ACRONYMS AND INITIALS (Cont'd)

RWST	Refueling Water Storage Tank
SAR	Safety Analysis Report
SEL	Seismic Equipment List
SFP-AET	Spent Fuel Pool Assessment Event Tree
SFPCCS	Spent Fuel Pool Cooling and Cleanup System
SFPs	Spent Fuel Pools
SGTR	Steam Generator Tube Rupture
SHNPP	Shearon Harris Nuclear Power Plant
SMA	Seismic Margin Assessment
SPLD	Success Path Logic Diagram
SPSA	Seismic Probabilistic Safety Assessment
SRO	Senior Reactor Operator
SSC	Structure, System, or Component
SSE	Safe Shutdown Earthquake
SSEL	Safe Shutdown Equipment List
SSHAC	Senior Seismic Hazard Analysis Committee
SSI	Soil Structure Interaction
SW	Service Water
THERP	Technique For Human Error Rate Prediction (see NUREG/CR-1278)
TS	Technical Specifications
TSC	Technical Support Center
UHS	Uniform Hazard Response Spectrum
ZR	Zircaloy

DEFINITIONS OF TERMS

Accident conditions — Conditions resulting from deleterious environmental effects or degraded equipment, components, or systems, occurring during events that are not expected in the course of plant operation, but are postulated by design or analysis.

Accident consequences — The extent of plant damage or the radiological release and health effects to the public or the economic costs of a core damage accident.

Accident sequence — A combination of events, beginning with an initiating event, that challenges safety systems and resulting in an undesired consequence (such as core damage or large early release). An accident sequence may contain many unique variations of events (cut sets) that are similar.

Accident sequence analysis — The process to determine the combinations of initiating events, safety functions, and system failures and successes that may lead to core damage or large early release.

Aleatory uncertainty — The uncertainty inherent in a non-deterministic (stochastic, random) phenomenon. Aleatory uncertainty is reflected by modeling the phenomenon in terms of a probabilistic model (which also must treat epistemic uncertainty.) In principle, aleatory uncertainty cannot be reduced by the accumulation of more data or additional information. (Sometimes called "randomness").

At power — Those plant operating states characterized by the reactor being critical and producing power, with automatic actuation of critical safety systems not blocked and with essential support systems aligned in their normal power operation configuration.

Availability — The fraction of time that a test or maintenance activity does not disable a system or component (see unavailability).

DEFINITIONS OF TERMS (Cont'd)

Available time — The time from which an indication is given that the human action is needed to when the action must be performed to advert core damage. Estimates of the overall system time available in a specific accident sequence is determined from engineering analyses which are intimately related to the accident sequence development and success criteria. Includes the point at which operators receive relevant cue indications in determining available time.

Basic event — An event in a fault tree model that requires no further development, because the appropriate limit of resolution has been reached.

CDFM method — Refers to the Conservative Deterministic Failure Margin (CDFM) method as described in EPRI NP-6041 (EPRI, 1991) wherein the seismic margin of the component is calculated using a set of deterministic rules that are more realistic than the design procedures.

Common cause failure (CCF) — A failure of two or more components during a short period of time as a result of a shared cause.

Component — An item in a nuclear power plant, such as a vessel, pump, valve, or a circuit breaker.

Composite variability — The composite variability includes the randomness variability and the uncertainty. The logarithmic standard deviation of composite variability, β_c, is expressed as $(\beta_R^2 + \beta_U^2)^{1/2}$.

Containment analysis — The process to evaluate the failure thresholds or leakage rates of the containment.

Containment bypass — An event that opens a direct or indirect flow path that may allow the release of radioactive material directly to the environment bypassing the containment.

Containment failure — Loss of integrity of the containment pressure boundary that results in unacceptable leakage to the environment.

Containment performance — A measure of the response of a nuclear plant containment to severe accident conditions.

DEFINITIONS OF TERMS (Cont'd)

Core damage	Uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage is anticipated representing the onset of gap release of radionuclides.
Core damage frequency (CDF)	Mean frequency of core damage per unit of time.
Core melt	Severe damage to the reactor fuel and core internal structures that includes the melting and relocation of core materials.
Cumulative distribution function	Integral of the probability density function; it gives the probability of a parameter of being less than or equal to a specified value.
Deaggregation	Determination of the functional contribution of each magnitude-distance pair to the total seismic hazard. To accomplish this, a set of magnitude and distance bins are selected and the annual probability of exceeding selected ground motion parameters from each magnitude-distance pair is computed and divided by the total probability.
Dependency	Requirement external to an item and upon which its function depends.
Diagnosis	Examination and evaluation of data to determine either the condition of a SSC or the cause of the condition.
Distribution system	Piping, raceway, duct, or tubing that carries or conducts fluids, electricity, or signals from one point to another.
Dominant contributor	A component, a system, an accident class, or an accident sequence that has a major impact on the CDF or on the LERF.
End state	The set of conditions at the end of an accident sequence that characterizes the impact of the sequence on the plant or the environment. In most PSAs, end states typically include: success states (i.e., those states with negligible impact), plant damage states for Level 1 sequences, and release categories for Level 2 sequences.

DEFINITIONS OF TERMS (Cont'd)

Epistemic Uncertainty The uncertainty attributable to incomplete knowledge about a phenomenon that affects our ability to model it. Epistemic uncertainty is reflected in a range of viable models, the level of model detail, multiple expert interpretations, and statistical confidence. In principle, epistemic uncertainty can be reduced by the accumulation of additional information. (Also called "modeling uncertainty").

Event tree A quantifiable, logical network that begins with an initiating event or condition and progresses through a series of branches that represent expected system or operator performance that either succeeds or fails and arrives at either a successful or failed end state.

Event tree top event The conditions (i.e., system behavior or operability, human actions, or phenomenological events) that are considered at each branch point in an event tree.

External event An initiating event originating outside a nuclear power plant that, in combination with safety system failures, operator errors, or both, may lead to core damage or large early release. Events such as earthquakes, tornadoes, and floods from sources outside the plant and fires from sources inside or outside the plant are considered external events (see also internal event). By convention, loss of offsite power and internal fires are considered to be "internal events."

Failure mechanism A physical explanation of why a failure has occurred. It can be characterized in many different ways, for example by the type of agent causing the failure (e.g., chemical mechanical, physical, thermal, human error) or by the physical process (e.g., vibration, corrosion).

Failure mode A specific functional manifestation of a failure, i.e., the means by which an observer can determine that a failure has occurred (e.g., fails to start, fails to run, leaks).

Failure probability The expected number of failures per demand expressed as the ratio of the number of failures to the number of type of actions requested (demands).

DEFINITIONS OF TERMS (Cont'd)

Failure rate	Expected number of failures per unit of time expressed as the ratio of the number of failures to a selected unit of time.
Fault tree	A deductive logic diagram that depicts how a particular undesired event can occur as a logical combination of other undesired events.
Fractile hazard curves	A set of hazard curves used to reflect the uncertainties associated with estimating seismic hazard. A common family of hazard curves used in describing the results of a PSHA is curves of fractiles of the probability distributions of estimated seismic hazard as a function of the level of ground motion parameter.
Fragility	Fragility of a system, structure or component is the conditional probability of its failure at a given hazard input level. The input could be earthquake motion, wind speed, or flood level. The fragility model used in seismic PSA is known as a double lognormal model with three parameters, A_m, β_R and β_U which are respectively, the median acceleration capacity, logarithmic standard deviation of randomness in capacity and logarithmic standard deviation of the uncertainty in the median capacity.
Fussell-Vesely (FV) importance measure	For a specified basic event, Fussell-Vesely importance is the fractional contribution to any figure of merit for all accident sequences containing that basic event.
Ground acceleration	Acceleration at the ground surface produced by seismic waves, typically expressed in units of g, the acceleration of gravity at the earth's surface.
Hazard	The physical effects of a natural phenomenon such as flooding, tornado, or earthquake that can pose potential danger (for example, the physical effects such as ground shaking, faulting, landsliding, and liquefaction that underlie an earthquake's potential danger).

DEFINITIONS OF TERMS (Cont'd)

Hazard (as used in probabilistic hazard assessment)
Represents the estimate of expected frequency of exceedance (over some specified time interval) of various levels of some characteristic measure of a natural phenomenon (for example, peak ground acceleration to characterize ground shaking from earthquakes). The time period of interest is often taken as one year, in which case the estimate is called the annual frequency of exceedance.

HCLPF capacity
Refers to the High Confidence of Low Probability of Failure capacity, which is a measure of seismic margin. In seismic PSA, this is defined as the earthquake motion level at which there is a high (about 95%) confidence of a low (at most 5%) probability of failure. Using the lognormal fragility model, the HCLPF capacity is expressed as A_m exp $[-1.65 (\beta_R + \beta_U)]$. When the logarithmic standard deviation of composite variability β_c is used, the HCLPF capacity could be approximated as the ground motion level at which the composite probability of failure is at most 1%. In this case, HCLPF capacity is expressed as A_m exp $[-2.33\ \beta_c]$. In deterministic seismic margin assessments, the HCLPF capacity is calculated using the CDFM method.

High winds
Tornadoes, hurricanes (or cyclones or typhoons as they are known outside the US), extra-tropical (thunderstorm) winds, and other wind phenomena depending on the site location.

Human error (HE)
Any member of a set of human actions that exceeds some limit of acceptability including inaction where required, excluding malevolent behavior.

Human error probability (HEP)
A measure of the likelihood that the operator will fail to initiate the correct, required, or specified action or response needed to allow the continuous or correct function of equipment, a component, or system, or by commission performs the wrong action that adversely effects the continuous or correct function of these same items.

Human reliability analysis (HRA)
A structured approach used to identify potential human errors and to systematically estimate the probability of those errors using data, models, or expert judgment.

DEFINITIONS OF TERMS (Cont'd)

Initiating event	Any event either internal or external to the plant that perturbs the steady state operation of the plant, if operating, thereby initiating an abnormal event such as transient or LOCA within the plant. Initiating events trigger sequences of events that challenge plant control and safety systems potentially leading to core damage or large early release.
Intensity	A measure of the observable effects of an earthquake at a particular place. Commonly used scales to specify intensity are the Modified Mercalli Intensity, Rossi-Forel, MSK, and JMA scales.
Interfacing systems LOCA (ISLOCA)	A LOCA when a breach occurs in a system that interfaces with the RCS, where isolation between the breached system and the RCS fails. An ISLOCA is usually characterized by the over-pressurization of a low pressure system when subjected to RCS pressure and can result in containment bypass.
Internal event	An event originating within a nuclear power plant that, in combination with safety system failures, operator errors, or both, can effect the operability of plant systems and may lead to core damage or large early release. By convention, loss of offsite power is considered to be an internal event, and internal fire is considered to be an external event.
Internal flooding event	An event located within plant buildings leading to equipment failure by the intrusion of water into equipment through submergence, spray, dripping, or splashing.
Large early release	The rapid, unscrubbed release of airborne fission products from the containment to the environment occurring before the effective implementation of off-site emergency response and protective actions.
Large early release frequency (LERF)	Mean frequency of a large early release per unit of time.
Level 1 analysis	Identification and quantification of the sequences of events leading to the onset of core damage.

DEFINITIONS OF TERMS (Cont'd)

Level 2 analysis	Evaluation of containment response to severe accident challenges and quantification of the mechanisms, amounts, and probabilities of subsequent radioactive material releases from the containment.
Level of detail	Different levels of logic modeling used in a PSA. A failure event in a fault tree analysis can address various levels of detail, depending on how much useful information is available concerning the contributors to the failure event.
Magnitude	A measure of the size of an earthquake. It is related to the energy released in the form of seismic waves. Magnitude means the numerical value on a standardized scale such as but not limited to Moment Magnitude, Surface Wave Magnitude, Body Wave Magnitude, or Richter Magnitude scale.
Minimal cut set (MCS)	Minimum combination of events in a fault tree that, if they occur, will result in an undesired event such as the failure of a system or the failure of a safety function.
Mission time	The time that a system or component is required to operate in order to successfully perform its function.
Model	An approximate mathematical representation that simulates the behavior of a process, item, or concept (such as failure rate).
Peak ground acceleration	Maximum value of acceleration displayed on an accelerogram; the largest ground acceleration produced by an earthquake at a site.
Performance shaping factor (PSF)	A factor that influences human error probabilities as considered in a PSA's human reliability analysis and includes such items as level of training, quality/availability of procedural guidance, time available to perform a action, etc.
Plant	A general term used to refer to a nuclear power facility (for example, plant could be used to refer to a single unit or multi-unit site).
Plant-specific data	Data consisting of observed sample data from the plant being analyzed.

DEFINITIONS OF TERMS (Cont'd)

Point estimate	Estimate of a parameter in the form of a single number.
Post-initiator human failure events	Human errors committed during actions performed in response to an accident initiator.
PSA application	A documented analysis influenced by a plant-specific PSA that affects the design, operation, or maintenance of a nuclear power plant.
PSA configuration control plan	The process and document used by the owner of the PSA to define the PSA technical elements that are to be periodically updated and to document the methods and strategies for maintenance of those PSA technical elements.
Pre-initiator human failure events	Human errors committed during actions performed prior to the initiation of an accident, for example, during maintenance or calibration procedures.
Probabilistic Safety Assessment (PSA)	A qualitative and quantitative assessment of the risk associated with plant operation and maintenance that is measured in terms of either risk or frequency of occurrence of risk metrics, such as core damage or a radioactive material release and its effects on the health of the public.
Probability of exceedance (as used in seismic hazard analysis)	The probability that a specified level of ground motion for at least one earthquake will be exceeded at a site or in a region during a specified exposure time.
Randomness (as used in seismic-fragility analysis)	The variability in seismic capacity arising from the randomness of the earthquake characteristics for the same acceleration and to the structural response parameters that relate to these characteristics.
Recovery	A general term describing restoration and repair acts required to change the state the initial or current state of a system or component into a position or condition needed to accomplish a desired function for a given plant state.
Repair	To restore a function, system or component by replacing a part or putting together what is torn or broken.

DEFINITIONS OF TERMS (Cont'd)

Required time — The time that is needed by operators to successfully perform and complete an action. Estimates of required time are derived from actual time measurements based on walk-throughs and simulator observations.

Respond — To react in response to a cue for action in initiating or recovering a desired function.

Response spectrum — A curve calculated from an earthquake accelerogram that gives the value of peak response in terms of acceleration, velocity, or displacement of a damped linear oscillator (with a given damping ratio) as a function of its period (or frequency).

Restore — To put back into a former or desired state.

Review level earthquake (RLE) — An earthquake larger than the plant SSE and is chosen in SMA for initial screening purposes. Typically, the RLE is defined in terms of a ground motion spectrum. [Note: A majority of plants in the Eastern and Midwestern United States have conducted SMA reviews for an RLE of 0.3g pga anchored to a median NUREG/CR-0098 spectrum (Newmark and Hall, 1978).]

Risk — Probability and consequences of an event, as expressed by the "risk triplet" that is the answer to the following three questions: (1) What can go wrong? (2) How likely is it? and (3) What are the consequences if it occurs?

Safe shutdown equipment list (SSEL) — The list of all SSCs that require evaluation in the seismic-fragilities task of an SMA (seismic margin assessment). Note that this list can be different from the Seismic Equipment List used in an seismic Probabilistic Safety Assessment.

Safety function — Function that must be performed to control the sources of energy in the plant and radiation hazards.

Safety systems — Those systems that are designed to prevent or mitigate a design-basis accident.

DEFINITIONS OF TERMS (Cont'd)

Safety-related — Structures, systems, and components that are relied upon to remain functional during and following design basis events to assure: (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shut down condition; or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable exposures established by the regulatory authority.

Screening analysis — An analysis that eliminates items from further consideration based on their negligible contribution to the frequency of a significant accident or its consequences.

Screening criteria — The values and conditions used to screen results to determine whether an item is a negligible contributor to the frequency of an accident sequence or its consequences.

Seismic equipment list (SEL) — The list of all SSCs that require evaluation in the seismic-fragilities task of an seismic Probabilistic Safety Assessment. Note that this list can be different from the Safe Shutdown Equipment List used in an seismic margin assessment.

Seismic margin — Seismic margin is expressed in terms of the earthquake motion level that compromises plant safety, specifically leading to severe core damage. The margin concept can also be extended to any particular structure, function, system, equipment item, or component for which "compromising safety" means sufficient loss of safety function to contribute to core damage either independently or in combination with other failures.

Seismic margin assessment — The process or activity to estimate the seismic margin of the plant and to identify any seismic vulnerabilities in the plant.

DEFINITIONS OF TERMS (Cont'd)

Seismic source	A general term referring to both seismogenic sources and capable tectonic sources. A seismogenic source is a portion of the earth assumed to have a uniform earthquake potential (same expected maximum earthquake and recurrence frequency), distinct from the seismicity of the surrounding regions. A capable tectonic source is a tectonic structure that can generate both vibratory ground motion and tectonic surface deformation such as faulting or folding at or near the earth's surface. In a PSHA, all seismic sources in the site region with a potential to contribute to the frequency of ground motions (i.e., the hazard) are considered.
Seismic spatial interaction	An interaction that could cause an equipment item to fail to perform its intended safety function. It is the physical interaction of a structure, pipe, distribution system, or other equipment item with a nearby item of safety equipment caused by relative motions from an earthquake. The interactions of concern are (1) proximity effects, (2) structural failure and falling, and (3) flexibility of attached lines and cables.
Severe accident	An accident that usually involves extensive core damage and fission product release into the reactor vessel, containment, or the environment.
Spectral acceleration	Pseudo-absolute response spectral acceleration, given as a function of period or frequency and damping ratio (typically 5%). It is equal to the peak relative displacement of a linear oscillator of frequency f attached to the ground, times the quantity $(2\pi f)^2$. It is expressed in g or cm/s_2.
Station blackout	Loss of all on-site and off-site AC power at a nuclear power plant.
Success criteria	Criteria for the establishing the minimum number or combinations of systems or components required to operate, or minimum levels of performance per component during a specific period of time, to ensure that their safety functions are satisfied within the limits of the acceptance criteria.

DEFINITIONS OF TERMS (Cont'd)

Success path (as used in Seismic Margin Assessments; see Section 3.5)	A set of components that can be used to bring the plant to a stable hot or cold condition and maintain this condition for at least 72 hours.
Support system	A system that provides a support function (e.g., electric power, control power, or cooling) for one or more other systems.
System failure	Termination of the ability of a system to perform any one of its designed functions. Note: Failure of a line/train within a system may occur in such a way that the system retains its ability to perform all its required functions; in this case, the system has not failed.
Truncation limits	The numerical cutoff value of probability or frequency below which results are not retained in the quantitative PSA model or used in subsequent calculations (such limits can apply to accident sequences/cut sets, system level cut sets, and sequence/cut set database retention).
Unavailability	The fraction of time that a test or maintenance activity disables a system or component (see availability); also the average unreliability of a system or component over a defined time period.
Uncertainty	A representation (usually numerical) of the state of knowledge about data, a model, or process, usually associated with random variability of a parameter, lack of knowledge about data, a model, or process, or imprecision in the model or process.
Uncertainty (as used in seismic-fragility analysis)	The variability in the median seismic capacity arising from imperfect knowledge about the models and model parameters used to calculate the median capacity.
Uniform hazard response spectrum	A plot of a ground response parameter (for example, spectral acceleration or spectral velocity) that has an equal likelihood of exceedance at different frequencies.

DEFINITIONS OF TERMS (Cont'd)

Walkdown

Inspection of local areas in a nuclear power plant where structures, systems, and components are physically located in order to ensure accuracy of procedures and drawings, equipment location, operating status, and environmental effects or system interaction effects on the equipment which could occur during accident conditions. For seismic-PSA and seismic-margin-assessment reviews, the walkdown is explicitly used to confirm preliminary screening and to collect additional information for fragility or margin calculations.

Section 1
INTRODUCTION

1.1 STATEMENT OF QUESTION ADDRESSED

A Probabilistic Safety Assessment (PSA) for the Shearon Harris Nuclear Power Plant (SHNPP) has been performed by ERIN to address a question posed by the Atomic Safety and Licensing Board (ASLB) in a Memorandum and Order dated August 7, 2000 (ASLB Order) in connection with Carolina Power & Light Company's (CP&L) license amendment request to expand spent fuel storage at SHNPP by placing spent fuel pools C and D in service. ERIN was asked by CP&L to determine the best estimate of the overall probability of the postulated sequence set forth in the following chain of seven events (Postulated Sequence):

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

1.2 SCOPE

The scope of this analysis is to directly respond to the ASLB Order and to determine the best estimate of the overall probability of the Postulated Sequence. Potential risk contributors outside the specific Postulated Sequence of events were not quantified.

No off-site consequence evaluation or calculation of public health effects were performed.

Degraded core conditions and degraded core conditions with containment failure or bypass could result from a number of different postulated accident scenarios. These degraded core conditions have similar characteristics for many of the postulated conditions despite their different initial plant conditions. These can be discussed under the following general risk contributing categories of events differentiated by mode of operation:

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

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

B. <u>Shutdown</u>

- Shutdown

The quantitative assessment of risk in nuclear power plants has proceeded from the methods and techniques developed in WASH-1400 up to the present day. The most emphasis and resources have been applied to the quantitative assessment of risk due to internal events. Other potential risk contributors have generally been treated using bounding or screening approaches which avoid explicit quantification or which treat the risk contributor in a conservative manner. Therefore, the industry does not have the same level of experience or degree of sophistication in the quantification of the risk associated with the other potential contributors to the risk profile, e.g., seismic, fire, shutdown events. This difference in level of experience and degree of sophistication in quantification methods will be addressed in evaluating uncertainties associated with the calculated event frequencies of different contributors to the Postulated Sequence.

Degraded core conditions are beyond the plant design basis. Both plant specific analyses and generic evaluations can be used to demonstrate the fact that the frequency of a degraded core event is very low. In addition, for many of the postulated degraded core event cases, the containment remains intact and radionuclide releases are considered low and would not cause on-site doses or adverse conditions that would significantly affect local operator actions to restore or provide backup sources of spent fuel pool cooling.

For a large fraction of degraded core events, the SHNPP large dry PWR containment would remain intact for a substantial period of time. Thus, there is a substantial amount of time available for operating crew or Technical Support Center (TSC) / Operations Support Center (OSC) actions to prestage equipment and establish backup cooling to the SFP if required. In a small fraction of postulated degraded core events, the containment may be: (1) open (e.g., during shutdown conditions); (2) failed; or (3) bypassed early in a core damage sequence resulting in relatively early radionuclide releases on-site without substantial benefit of containment to prevent or delay radionuclide releases.

The core damage event may also produce adverse conditions of radiation, high temperature, and steam in: (a) the area of the turbine (e.g., SGTR); or, (b) Reactor Auxiliary Building (RAB) and the connected Fuel Handling Building (FHB) (e.g., ISLOCA or containment failure). The combination of increased temperatures and steam environment could cause equipment failures in the local area that could adversely impact long term core melt mitigation and/or the ability to maintain SFP cooling. The radiation in the RAB or the FHB could result in a prohibitive environment for local manual actions for alternative SFP cooling alignments. This condition could require either early alignment actions prior to containment failure or late actions after radiation levels subside.

In this assessment, the potential for dependent failures due to pre-existing failures, sequence dependent failures, and spatial effects of the various accident scenarios are incorporated to address the potential for successful continued SFP cooling.

Best Estimate (Realistic Evaluation)

For the purpose of responding to the ASLB's Order, a realistic or best estimate evaluation is desired both because it was requested by the ASLB and because introducing biases into the analysis could result in an apparent "conservative" calculation for one purpose, but for other purposes may actually be affected in a non-conservative direction. Sensitivities are used to assess critical aspects of the analysis for which particularly large uncertainties may exist.

1.3 PLANT CONFIGURATION

Key aspects of SHNPP that influence the assessment of the Postulated Sequence are discussed in Appendix A of the report and in the SHNPP PSA. The following discussion provides some of the highlights of Appendix A.

1.3.1 Assumed Plant Configuration

The SHNPP FHB was constructed to accommodate a four unit site. The size and compartmentalization of the building influences its accident response. These features of the SHNPP FHB have been explicitly represented in the deterministic calculation of post containment failure accident sequences. In addition, there are a substantial number of alternate systems and pathways for establishing water makeup to the SHNPP spent fuel pools which are also included in this analysis.

Spent Fuel Pools

Spent fuel pools A and B (those currently in operation and licensed) are connected 99% of the time with gates removed from the connections to their common transfer canal.

Spent fuel pools C and D (those proposed for operation) are connected 99% of the time with gates removed from the connections to their common transfer canal.

All SFPs are assumed filled to their capacity with spent fuel for purposes of timing estimation.

The SFPCCS cooling pumps are assumed to trip for all postulated severe accidents. The SFPCCS cooling pumps may be energized from the emergency diesel generators. This action can be accomplished by the operators from the Control Room.

There are no automatic trips on the purification pumps, however, offsite AC power is required for their operation.

Fuel Pool Gates

SFP bulkhead gates are explicitly included in the model. The model for each gate includes a basic event to represent the probability that a gate is installed and its seals are inflated. The model also includes a basic event for each gate to represent the probability that the operators would deflate that gate's seals. For this analysis, no credit was given for the benefit associated with deflating the seals to increase communication among SFPs.

1.3.2 Future Configuration

Procedures for the C and D SFPs and their associated SFPCCS cooling pumps are not currently in place. Therefore, the PSA has been performed using procedures that are believed appropriate. This analysis assumes that the current modification to add two SFPCCS pump and cooling systems to support SFP C and D are installed and operational. It is also assumed that appropriate procedures for operating the C and D SFPs are in place, i.e., and generally consistent with those that exist for SFPs A and B.

Appendix A provides a description of the physical plant and its arrangement. This includes the critical systems affecting the ability to maintain adequate cooling of the fuel in the SFPs.

Section 2
METHODOLOGY

2.1 METHODOLOGY

The analytical methodologies chosen to determine the best estimate overall probability of the Postulated Sequence are based on Probabilistic Safety Assessment (PSA) techniques that have been developed in the nuclear and aerospace industries to assess the frequency and risks of accidents. The methodology has significantly evolved over the past 10 years in the nuclear industry, building on the methods, data, and approaches used in the NRC's mandated Individual Plant Examination (IPE) process. The current PSA methods are judged to be significantly improved beyond those used in the IPE process. Updated and expanded PSA such as the SHNPP PSA, are more realistic than the previous IPEs, which were limited to a search for severe accident vulnerabilities.

The purpose of this SHNPP PSA is to determine the best estimate of the overall probability of the postulated sequence set forth in the following chain of seven events (Postulated Sequence):

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

Figure 2-1 is a top level description of the process used in the quantification of the associated event frequency for the Postulated Sequence at SHNPP.

Steps 1 and 2 were evaluated using probabilistic techniques. For the internal events contribution to these steps, the SHNPP Level 1 and 2 PSA model was used. The dominant fire initiating events from the SHNPP IPEEE were added to the SHNPP Level 1 and 2 PSA model to estimate the frequency of accident sequences due to fire initiating events. Seismic contributions used the SHNPP hazard curve plus component fragility generic information within a seismic PSA framework. The frequency of shutdown core damage used generic PWR estimates of potential core damage frequency. Risk from other external events was judged negligible based on the SHNPP IPEEE.

Step 3 utilized probabilistic techniques as well. A fault tree model of the SFP cooling and makeup systems was used to assess the ability to preserve SFP cooling or makeup.

Steps 4 and 5 utilized deterministic methods to calculate conditions affecting whether personnel access to restore cooling or provide make-up to the SFPs was precluded.

Steps 6 and 7 were analyzed deterministically as follows: It was assumed that, given a loss of SFP cooling and make-up, evaporation would lead, over time, to loss of water in the pools. Industry experience and expert judgement indicates that the exothermic reaction for the low decay heat fuel that would be in SFPs C and D would be a low probability event. However, the probabilistic analysis conservatively assumes a 1.0 failure probability.

There are strong interfaces within the analysis that require multiple inputs from different sources. These inputs are discussed in detail in their specific section or Appendix and are integrated into the overall analysis in Section 4, the accident sequence evaluation. Some of the critical inputs are identified here for ease of reference:

- Accident Sequence types to be evaluated -- Section 2.
- Deterministic inputs describing the plant conditions during the accident sequences -- Appendices E and F.
- Plant configuration and description of mitigation methods -- Appendix A.
- Containment failure modes to consider -- Section 2
- Model for mitigation assessment -- Appendix D
- Human Reliability Analysis summary -- Appendix C

The following subsections describe in overview fashion the methods used in the evaluation of various contributors to the event frequency profile for the Postulated Sequence. The details of the implementation of these methods are described in Section 4.

The effort to determine the best estimate overall probability of the Postulated Sequence involved the formation of an analysis team (13 team members) and direct links to key CP&L staff. The CP&L staff provided both detailed calculations (including the Level 1 and 2 SHNPP PSA), system descriptions, interviews with operating personnel, and procedure interpretations. The team effort included:

- Multiple SHNPP site visits to confirm the as-built design and crew response
- An independent peer review of the inputs to the evaluation including the SHNPP Level 1 and 2 PSA for internal events
- An independent review of the analysis report

The total effort by ERIN personnel dedicated to the analysis exceeded one person year of professional time during the period August through the date of this report in November, 2000.

Sensitivity Cases were performed as part of the probabilistic evaluation in order to determine the impact of a change in plant configuration, changes in assumptions, or the impact of phenomenological probability ranges.

Severe Accident and Containment Failure or Bypass →	Loss of all SFP Cooling and Makeup Systems →	Radiation Impact on Personnel Access →	Evaporation and Exothermic Reaction
Steps 1 and 2[1]	**Step 3[1]**	**Steps 4 and 5[1]**	**Steps 6 and 7[1]**
Potential Contributors to Risk Profile	Probabilistic Evaluation	Deterministic Evaluation	Deterministic Evaluation
Probabilistic Evaluation of: • Internal Events • Seismic Events • Fire Events • Shutdown Events • Other Events	• Normal SFP Cooling Methodology • Normal SFP Inventory Makeup • Alternate SFP Inventory Makeup	• RAB and FHB Thermal Hydraulic Response to Accident Conditions • Radiological Environments in RAB and FHB • Impacts on personnel and equipment	• Boil-off Calculation • Exothermic Oxidation Reaction in C and D (not analyzed)

Figure 2-1 Process Used in Analysis of Postulated Sequence

[1] Steps as identified in Postulated Sequence.

2.2 OVERVIEW

Postulated severe accidents with containment failure or containment bypass at SHNPP are of low frequency and meet all NRC Safety Goals. The operation of the SFPs following beyond design basis accidents is not within the design bases mandated for the SFP facility by the Nuclear Regulatory Commission. Therefore, imposing severe accidents with containment failure on the continued safe operation of the SFP is a stringent demand.

SFPs C and D will have spent fuel with significantly lower decay heat levels than SFPs A or B (i.e., greater than 5 years since power operation). SFPs C and D will have a negligible or very small incremental contribution to the consequences of a severe release above the radionuclide releases already theoretically possible from SFPs A and B. In addition, the frequency of cooling and makeup failure at SFPs C and D will be indistinguishable from the potential for the frequency of cooling and makeup failure for SFPs A and B, which are already licensed. Therefore, the conclusion is that the incremental risk from the operation of SFPs A, B, C and D is very small when compared to the already licensed risk of operating SFPs A and B.

This analysis addresses the frequency of accident sequences associated with the postulated release of radionuclides from SFPs C and D, caused by a severe accident with containment failure or bypass at Unit 1. The frequency of a postulated release from SFPs A and B was determined to be essentially identical.

The probabilistic model framework was structured such that the event tree quantification was tied directly to the containment failure mode (e.g., bypassed, late failure, early failure). For internal events and fires, the system dependencies were explicitly treated by inputting the cutsets into the event tree and explicitly treating the support system failures in the top events of the SFP event tree. This approach for internal events and fire initiated events included the following steps:

1) Sum frequencies or cutsets for each containment failure mode. The computer code, CAFTA[1], was used to perform the Boolean algebra calculations for the plant logic models. The minimum failure combinations created by the model are called cutsets. Cutsets carry with them the system and support system dependencies.

2) Input SHNPP Level 1 and 2 PSA cutsets directly to SFP Event Tree. (This uses a CAFTA utility to create a fault tree from the cutsets.)

3) The SFP Event Tree explicitly accounts for the adverse impacts of specific containment failure modes through events such as:

   - Adverse environment causes equipment failure

   - Radionuclide release or high temperatures preclude local access and manipulation of valves:

     a) in RAB due to external cloud
     b) in RAB due to RAB radiation environment
     c) in FHB due to external cloud
     d) in FHB due to FHB radiation environment
     e) in FHB due to SFP boiling and consequential high temperature

For contributors to the event frequency profile other than internal events and fire initiated events, the best estimate analysis relied on available SHNPP information in a quantitative risk format to estimate the above items 1) and 2).

Human error dependencies were then addressed by examining the nodes, their inputs, and outputs to ensure the actions modeled adequately represent the potential for common cause failures among nodes.

[1] CAFTA is a widely used computer tool for the probabilistic assessment of complex logic models.

The first step in the PSA method is the identification of applicable initiating events. The next section addressees these initiating events.

2.3 RISK ANALYSES: INITIATORS, SEQUENCES, DETERMINISTIC MODELING

2.3.1 Risk Contributing Initiators

A crucial step in PSA methodologies is the identification of the initiating event or events. In the case at hand, the ASLB Order provides a very specific sequence of events that are to be considered within the analysis, i.e., the Postulated Sequence. This, in turn, allows determination of appropriate initiating events.

"Initiating events" that affect only the spent fuel pool are not the subject of the Postulated Sequence set forth in the ASLB Order. The initiator must impact the reactor core and containment first. Therefore, accident initiators under consideration are those events that could cause core damage and containment failure or bypass.

The initiators include the following:

A. At-Power

- Internal Events[1]
- Internal Flood
- Seismic Induced
- Fire Induced
- Other Events that could cause Core Damage

[1] Generally taken to be events that originate within the plant (e.g., turbine trip), but also includes loss of offsite power. It does exclude fire initiated events, which are treated separately.

B. Shutdown

- Shutdown Events

The evaluation of different contributors to core damage frequency (CDF) and containment failure uses probabilistic techniques. However, the degree of uncertainty in the calculated estimates may vary substantially. The degree of realism at which these PSA quantifications are performed can be significantly different because of the lack of experience and data in describing phenomena and the associated modeling uncertainties. The potential risk contributors derived from different sources can be characterized qualitatively. The following is a characterization of the degree of realism expected for each potential contributor.

Event Frequency Contributor	Qualitative Characterization of the Realistic Nature of the Model
Internal Events	Best Estimate Realistic Calculation
Seismic Events	Conservative Plant Specific Estimate
Fire Event	Conservative Plant Specific Estimate
Shutdown Events	Conservative Generic Estimate
Other Events	Realistic Estimate

2.3.2 Accident Sequences

The accident sequences evaluated in this assessment were developed from the SHNPP Level 1 and Level 2 PSA for internal events or were separately derived if no existing model was available. Section 4 describes this process in more detail.

2.3.3 Deterministic Calculations

An extensive effort was performed to characterize the plant conditions, especially in the critical buildings: the Reactor Auxiliary Building and the Fuel Handling Building -- i.e., the areas containing critical equipment. A deterministic evaluation of the plant thermal hydraulic response and the transport of radionuclides was performed to characterize issues such as access, timing, and adverse conditions on equipment. In addition to the effects due to severe accident core melt progression, the analysis also addressed the potential for SFP boiling and its consequential effects on accessibility.

The method applied utilized a Modular Accident Analysis Program (MAAP) computer model (see Appendix E) to estimate the transient flow conditions due to the postulated accident sequences and containment failure modes.

MAAP is the most widely used Severe Accident Analysis code and has been reviewed extensively by the NRC and its contractors in support of Generic Letter 88-20. The Electric Power Research Institute (EPRI) has continued to support MAAP through programs to address benchmarking of the code, peer reviews, comparisons to data and other code results, along with a very active users group. MAAP includes best estimate models to represent accident progression beginning with normal operation and extending to potential radionuclide release to the environment.

The SHNPP-specific MAAP calculations also yielded the fission product release and accumulation effects in the RAB and FHB. These results provide the input to the CP&L dose assessment to calculate the dose rates for areas to assess equipment survivability and personnel access.

Access

Accessibility to areas requiring operator actions has been addressed to incorporate the following important considerations:

- Radiation and other harsh environments in the local areas
- Radiation and other harsh environments precluding pathways to the local areas
- Time windows when the actions could be required compared with the time when actions could be performed
- Status of doors/locks under the postulated conditions

The first two environmental factors and the time available were addressed using the deterministic computer code, MAAP (see Appendix E). The last item has been reviewed to ensure that the accident sequence does not render the FHB doors inoperable. For station blackout (SBO) conditions with the security diesel also failed and batteries depleted, the FHB doors can be opened with keys carried by the security force and auxiliary operators(1). Therefore, for extended times, security personnel or auxiliary operators with keys would be available to provide access to the FHB even under SBO conditions.

Permissible operator accessibility is based on receiving a maximum dose of 25 rem which is the emergency dose limit as provided in plant procedure PEP-330,

[1] Correspondence from Eric McCarney (CP&L) to E.T. Burns (ERIN), November 10, 2000.

"Radiological Consequences," and further described in the Affidavit of Benjamin W. Morgen. For the ISLOCA sequence, the dose assessment concludes that within the 4 day window from the accident initiation access could be made to the FHB 216' N Elevation. The higher calculated doses in this region were partially due to shine through the equipment hatch from deposited radionuclides in FHB 236' El. The effect of limiting access is treated in sensitivity cases.

Section 2.4 provides a description of each of the release sequences along with the associated timeline. Additional details on the accident progression can be found in Appendix E.

Alignments outside the RAB and FHB could also be affected by radionuclide releases outside the buildings. This occurs for all containment failure and bypass sequences. The "Wind Rose" [4-9] for SHNPP is used to assess the probability that the wind could carry the radiation in the direction of the multiple remote line-up areas, e.g., the Water Treatment Building (WTB) (Southwest from the containment--SW), the intake structures, (South from the Containment—S) or the cooling tower basin (East from the containment--E).

The conditional probability that radionuclide releases cause on-site doses which limit worker access on-site can be estimated using the combination of the following two factors:

a) The conditional probability that the prevailing winds are such that a release would carry the radionuclides to the diesel fire pump (DFP) and demineralized water stations and cooling tower basin.

<u>AND</u>

b) The conditional probability that, given the radionuclide release is carried to that location, access would be effectively prevented. This is a function of the radionuclide release magnitude and the effective dose at approximately 4 days.

The probability that the wind directs the release towards that area of the site that houses the DFP and demineralized water pumps is determined from the FSAR Wind Rose. [4-9]

Therefore, the probability that the wind is either calm or blowing in the WTB direction is:

5.6%	Calm
8.0%	NNE to SSW
7.0%	NE to SW
20.6%	Total

Probability that the wind does not carry radiation to the WTB is 79.4%.

The conditional probability of the wind blowing radionuclides from containment toward the plant areas for which access is required (includes stagnant air cases also) is approximately 0.2 for each of the 3 critical locations. This results in a combined probability of 0.05 based on the stagnant air case dominating the adverse effect on all locations. Even if the wind carries the radiation in the WTB direction, the probability that the location would become uninhabitable for more than 1 day is judged to be less than 10%, and for 4 days to be less than 0.1%. This means the probability that actions cannot be taken locally within 4 days near the intake i.e., the location of the diesel fire pump (DFP), the demineralized water, or the cooling tower basin pathway is 0.05% (P_f = 0.0005).

<u>Time</u>

The accident characteristics associated with SFP evaporation events are significantly different than those for the at-power evaluation in that the time available for effective operator responses is generally significantly longer than for most of the operator actions included in an at-power PSA. This extended length of time means that far more resources would be available to assist in the performance of the actions than for the case under at-power conditions. This represents a substantial change in the treatment of recovery actions because at-power PSAs generally include no or modest probabilities of repair and recovery. This analysis includes modest credit for the potential substantial increase in resource availability, but does not increase the state of the technology by including credit substantially beyond that which is typically justified in PSAs.

SFP Boiling

In order to assess the effects associated with SFP boiling on building conditions, the MAAP code was used. A MAAP analysis assuming the maximum pool boiling rate was performed. This deterministic assessment resulted in the following insights:

- Railway Door would be opened by the FHB pressurization and would yield an escape pathway for steam, diverting it from the lower elevations.
- The FHB Operating Deck environment would reach temperatures of approximately190°F.

The onset of SFP boiling was considered in conjunction with the conditions imposed by the severe core damage accident.

CP&L has extensive fire brigade training. The results of this training and associated data indicates that entry into an environment of ~ 190°F (FHB operating deck with SFP boiling) can be performed by personnel equipped with available protective gear. This allows access of personnel to the FHB operating deck between the time of SFP initial boiling and the time at which the SFP water level is close to the top of the spent fuel (i.e., within approximately 3 ft). This latter time is approximately 5 to 6 days under the highest assumed SFP heat loads, however, no credit is assumed in the analysis beyond

4 days. Limited personnel access under these conditions is possible and is credited for the FHB 286' El. under SFP boiling conditions[1].

In each case, the time used in the model to represent accessibility to the FHB El. 286' was selected to be the time of the conditions imposed by the core damage event.

Deterministic Calculations to Support Accessibility

The thermal hydraulic calculations were then used to characterize the following:

- Timing of key events (See Section 2.4)
- Operator accessibility to take actions (See Appendix C and Table 2.3-1)
- Survivability of equipment (See Table 2.3-2)

Table 2.3-1 is derived from the MAAP results described in Appendix E and the detailed dose assessment performed by CP&L. Radionuclide concentrations in the critical RAB and FHB compartments were calculated using MAAP and provided as input to the CP&L dose analysis. An assessment was also made on access from outside the buildings to address recovery actions requiring operator action in the RAB and FHB. Table 2.3-1 identifies each critical location as either "Accessible" or "Not Accessible". For a compartment to be judged to be accessible, the dose over the time needed to perform an alternate lineup to provide makeup water to the SFPs must be maintained below 25 rem. A representative time period of 15 minutes was used to allow sufficient time for an operator to enter a region and perform the required valve manipulations. In general, the MAAP results would indicate that if the doors and hatches leading into a particular region remained closed and intact, that compartment would remain accessible for operator actions. The CP&L dose assessment also included the effect of "shine" from adjacent compartments as part of the overall dose estimate. To demonstrate the

[1] Correspondence from Davis MacCaffey (CP&L) to E.T. Burns (ERIN), November 10, 2000.

method used for developing Table 2.3-1, the Early Containment Failure case shows that, due to containment failure, the RAB along with the operating deck of the FHB (El. 286') would not be accessible for operator actions. This is due to the environment created as a direct result of containment failures causing discharge of radionuclides and forcing doors open between the RAB and FHB at El. 261'. Other compartments identified in the MAAP analysis as accessible for this sequence are marked as "A" in the Table 2.3-1. The SGTR and Late Containment failure sequences result in a release outside of the RAB/FHB (SGTR) and, potentially, a delayed release (Late Containment Failure). These conditions allow access to the RAB and FHB for an extended period until the containment fails late in the event. This would provide ample time to pre-stage any required valve lineups prior to exceeding dose limits in the buildings. A designator of "A/X" is denoted in Table 2.3-1 to represent these situations.

One final note on Table 2.3-1 is related to personnel accessibility. The MAAP analysis described in Appendix E for the Containment Isolation failure case shows that the door leading to the FHB from the RAB on the 261' elevation would not open. This same door was calculated to open in both the early and late containment failure cases. The doorway does not open in this case due to a very small variation in the calculated pressure difference across this doorway. To account for possible uncertainties in the door failure pressure a conservative assumption was made to fail the doorway from RAB 261' to FHB 261'. This results in assigning a "not accessible" condition for the FHB operating deck (286' El.) for the containment isolation event.

Table 2.3-2 is similar to Table 2.3-1 in that it establishes conditions for equipment survivability in response to the various severe accidents. The thermal hydraulic evaluation was used to determine the compartment conditions and to determine if the equipment would survive. As in Table 2.3-1, the RAB and FHB compartments are identified with either an "A" to denote that the equipment is expected to survive the conditions or an "X" if the thermal conditions are expected to challenge the operability of the equipment. This assessment utilizes typical qualification data to estimate the survivability. In most cases, if the area is exposed to flow from the containment breach or bypass event, then the equipment is assumed NOT to survive. In this respect, information in Table 2.3-2 is found to be consistent with that of Table 2.3-2.

Access to the RAB and FHB from outside may be necessary in some cases. CP&L assessed the doses at the following locations for each case resulting in a radionuclide release to the environment:

- Exclusion Area Boundary (EAB) representing entry to the site
- Entrance to the Power Block representing entry to the buildings
- Water Treatment Building
- Cooling Tower Basin

Table 2.3-3 has been created to summarize the results of the CP&L evaluation outside of the buildings. The results are tabulated for two situations. First, it is estimated that outside work to establish an alternate makeup source may require up to 2 hours to complete. Locations included for on-site work include the areas of the water treatment building, cooling tower basin, and the intake structure. The second column in Table 2.3-3 indicates if this work can be performed and still maintain a total individual exposure below 25 rem. The third column provides a similar indication for access into plant buildings which is assumed to require 15 minutes exposure. The dose levels at this location tend to be higher due to the assumption of a ground level release.

The outside radiation exposure analysis performed by CP&L uses a conservative atmospheric dispersion model and does not include an assessment of wind direction. The wind rose data [4-9] included in the Harris FSAR indicates prevailing winds from the west. In particular, the maximum wind speeds are found to be from the N, SW, and SSW directions. Access to the site can be accomplished from the northwest, generally upwind of the prevailing plume direction. Also, access to the FHB can be made at the northwest corner of the power block, also upwind of the prevailing release pathway. There are also multiple entrances to the "K" Building (Safety Meeting Room). Therefore, for cases where the CP&L outside dose assessment indicated limited access, the prevailing winds combined with the relative location for entry to the plant buildings make it possible for access. Table 2.3-3 shows that for most of the cases analyzed, access to the plant buildings from outside would be within a period of 4 days. This is the time available to establish inventory makeup to the SFP. Even when the dose levels exceeded the total exposure limit of 25 rem, limited access would be possible depending on the wind direction. Outside work to establish alternate makeup to the spent fuel pools would also be acceptable given the 25 rem limit within the 4 day period. Given that the operators would have a long time available to establish alternate spent fuel pool makeup (at least 4 days), sufficient time would exist to allow the radiation plume to disperse.

Table 2.3-1

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

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

LEGEND

A - Accessible

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

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

Table 2.3-2

SUMMARY OF EQUIPMENT SURVIVABILITY AS A FUNCTION OF SEVERE ACCIDENT CONDITIONS

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

LEGEND

A - Pumps are considered to have survived the environment.

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

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

Table 2.3-3

EX-BUILDING DOSE SUMMARY

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

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

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

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

Survivability

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

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

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

2.4 CONTAINMENT FAILURE MODES AND CRITICAL TIMES

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

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

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

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

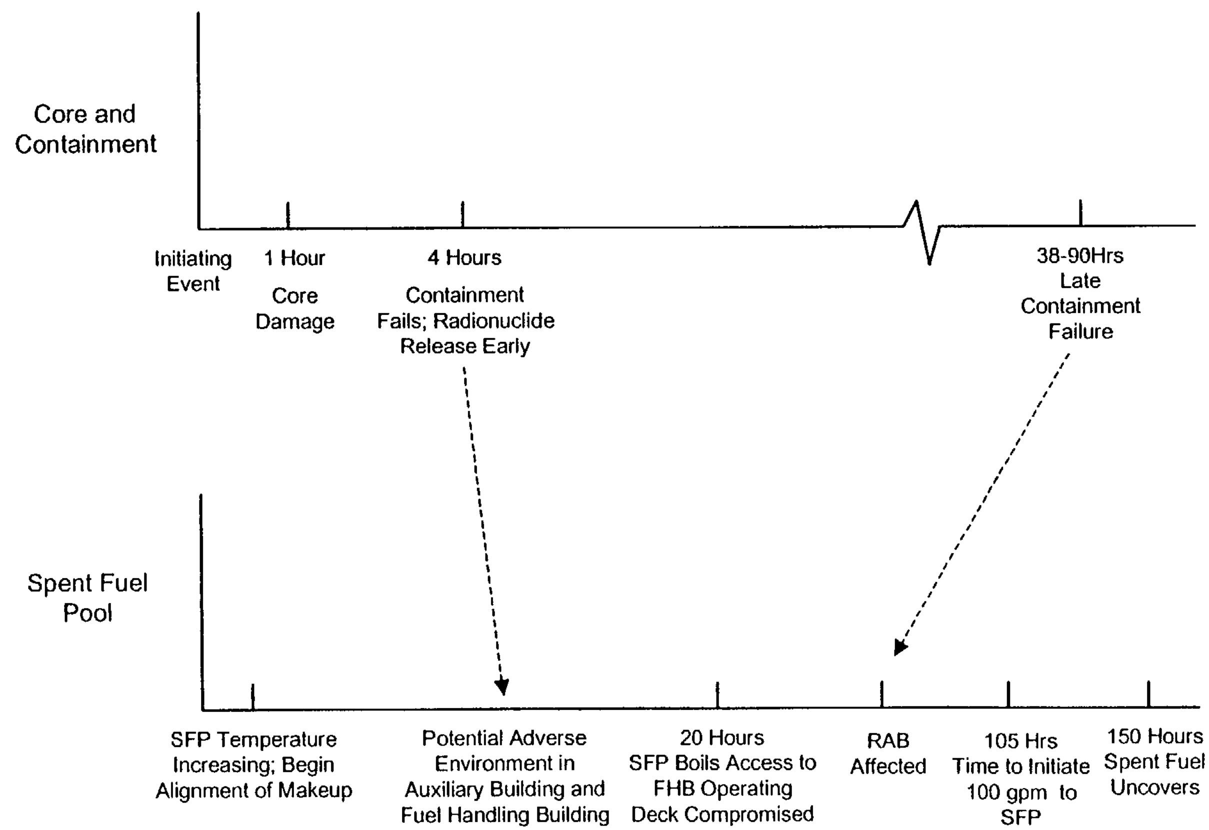


Figure 2.4-0 Comparison of Critical Times Associated with: (a) Core Damage plus Early and Late Containment Failures; and (b) Spent Fuel Pool Evaluation

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

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

2.4.1 Early Containment Failure

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

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

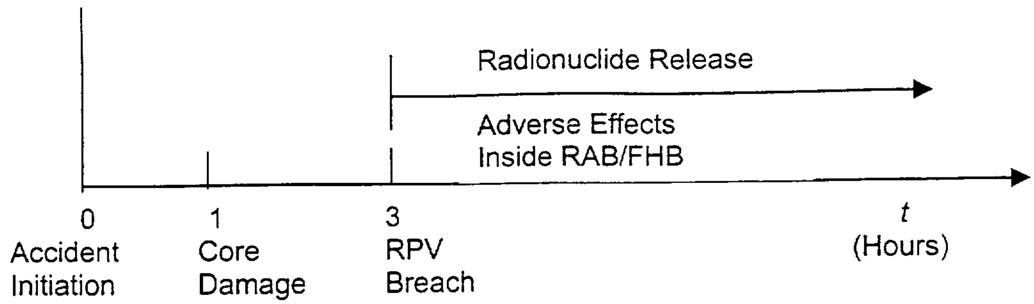


Figure 2.4-1 Typical Time Line for Effects Associated with Postulated Early Containment Failure

2.4.2 Containment Bypass

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

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

2.4.2.1 SGTR

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

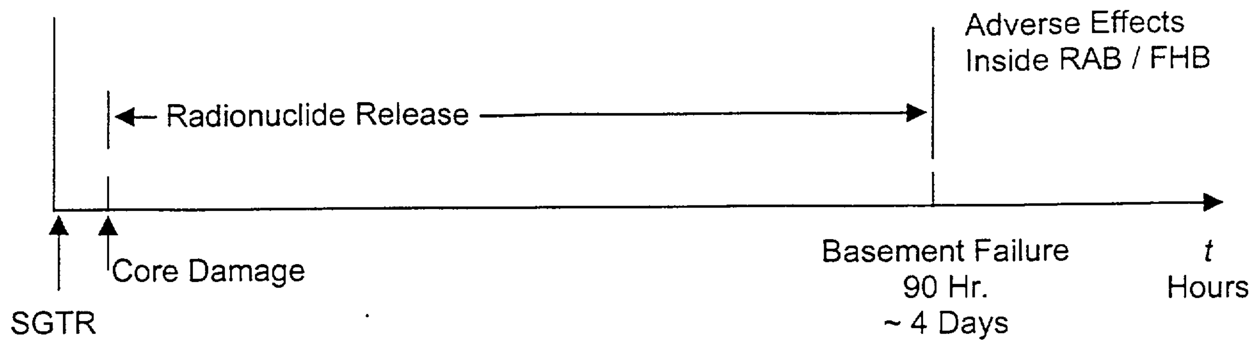
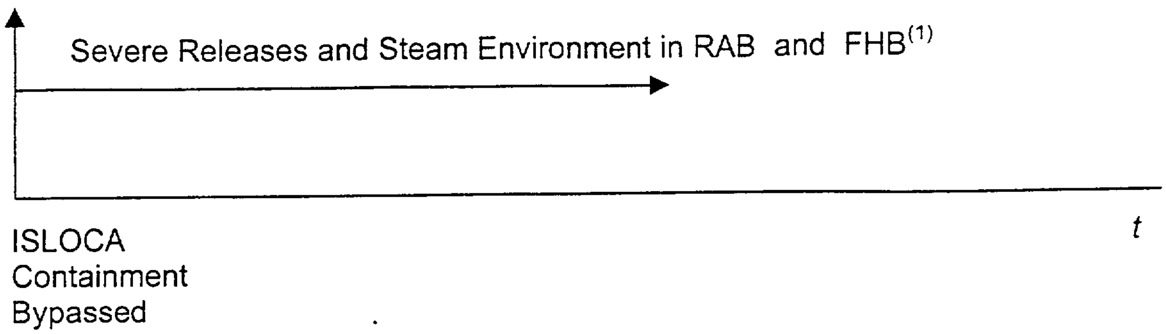


Figure 2.4-2 Approximate Time Line for SGTR

2.4.2.2 ISLOCA

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

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



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

Figure 2.4-3 Approximate ISLOCA Time Line

2.4.3 Containment Isolation Failure

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

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

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

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

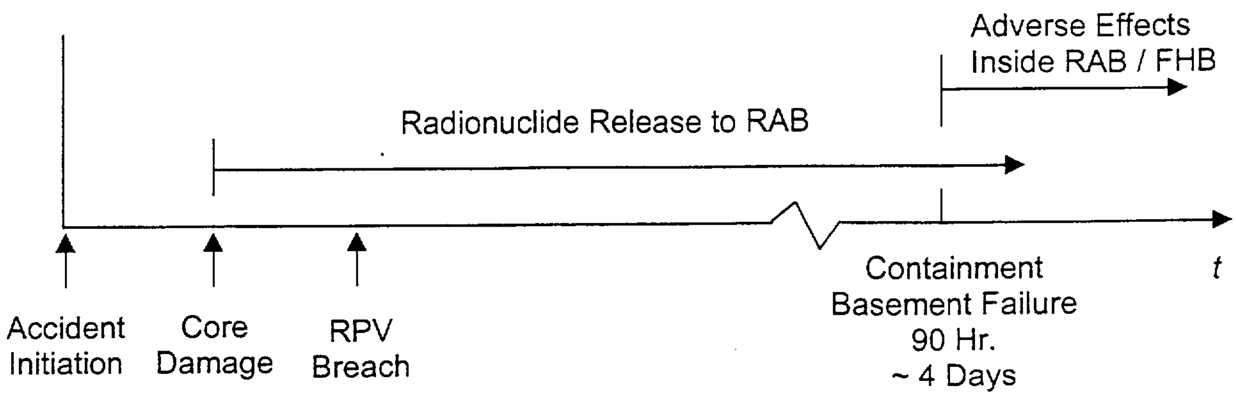


Figure 2.4-4 Approximate Time Line for the Effects Associated with the Containment Isolation Failure

2.4.4 Late Containment Failures

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

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

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

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

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

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

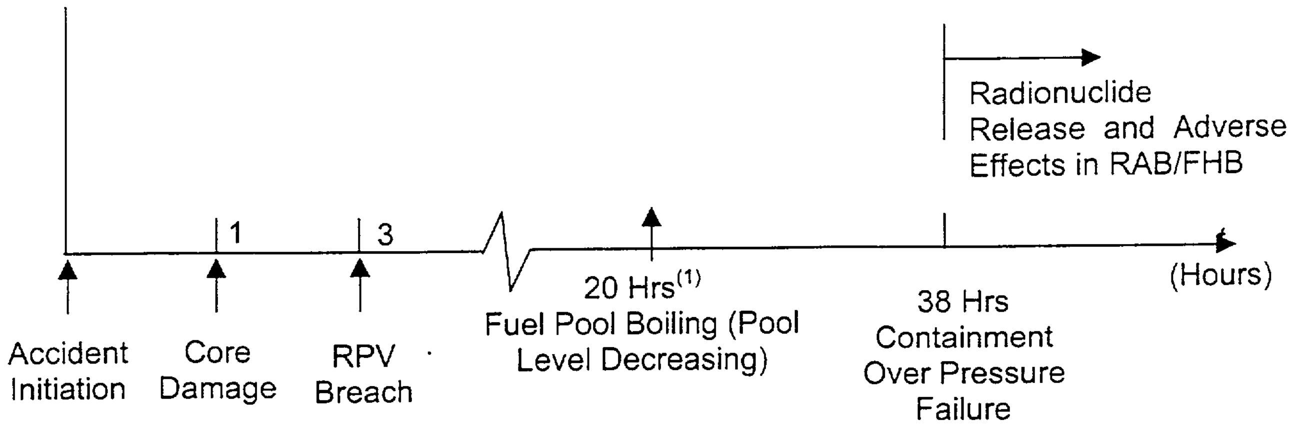


Figure 2.4-5 Approximate Time Line for Late Containment Failures

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

Table 2.4-1

SUMMARY OF IMPACTS ASSOCIATED WITH POSTULATED CONTAINMENT FAILURE MODES

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

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

Table 2.4-1

SUMMARY OF IMPACTS ASSOCIATED WITH POSTULATED CONTAINMENT FAILURE MODES

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

2.4.5 Summary of Critical Times

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

Table 2.4-2

CRITICAL TIMES

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

2.5 SCOPE, KEY ASSUMPTIONS, AND GROUNDRULES

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

2.5.1 Success Criteria

Time Available for Spent Fuel Pool Cooling

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Effect of Spent Fuel Pool Boiling

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

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

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

Makeup Success Criteria

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

2.5.2 Mission Time

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

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

2.5.3 Maintenance Unavailability

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

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

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

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

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

Shutdown: 1.0

2.5.4 Adverse Environmental Impacts

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

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

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

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

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

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

- With containment sprays
- Without containment sprays

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

2.5.5 Structural Analysis

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

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

Containment Structural Analysis

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

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

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

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

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

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

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

<u>Spent Fuel Pool Structural Analysis</u>

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

Reactor Auxiliary Building

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

Seismic Capability

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

Section 3
SHNPP PSA STATUS AND QUALITY

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

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

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

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

3.1 INTERNAL EVENTS

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

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

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

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

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

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

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

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

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

3.2 SEISMIC

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

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

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

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

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

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

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

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

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

3.3 FIRE

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

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

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

3.4 OTHER EXTERNAL EVENTS

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

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

3.5 SHUTDOWN

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

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

3.6 SUMMARY

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

Table 3-1

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

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

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

Section 4
SPENT FUEL POOL COOLING ANALYSIS

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

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

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

4.1 INTERNAL EVENTS

4.1.1 Accident Sequence Development

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

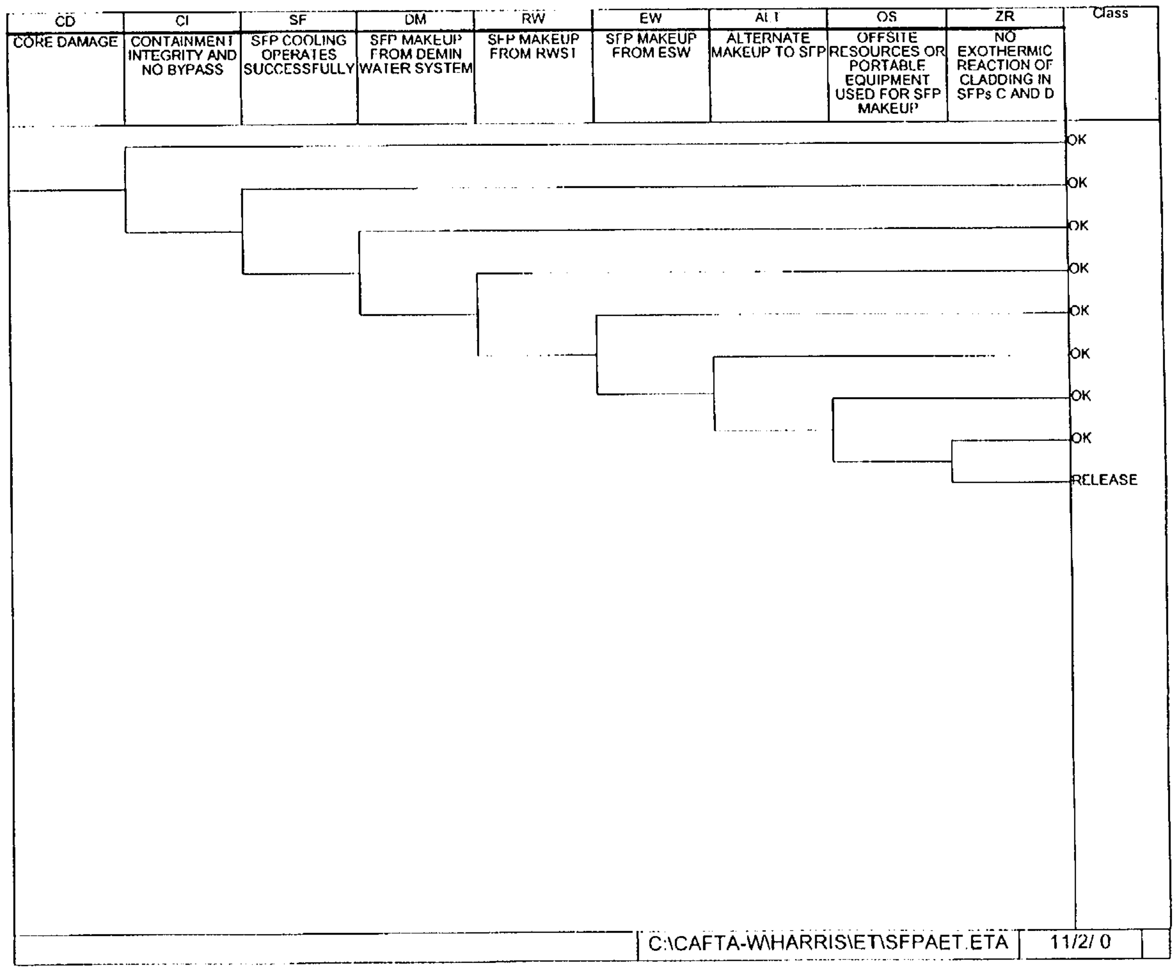
The approach for internal events was to take the results of the Level 1 and Level 2 SHNPP PSA in the form of individual cutsets and input these cutsets to the assessment of the SFP. Figure 4.1-1 summarizes the overall approach.

Typical PSA Accident Initiators	Level 1 PSA Core Damage Events	Level 2 Containment Failure or Bypass	ASLB Order: Spent Fuel Pool Analysis
Internal Events	Yes	Yes	Yes[*]
		No	None Required
	No	NA	None Required
ISLOCA & Steam Generator Tube Ruptute	Yes	Yes	Yes[*]
	No	No	None Required
Seismic	Yes	Yes	Yes[*]
		No	None Required
	No	NA	None Required
Fire	Yes	Yes	Yes[*]
		No	None Required
	No	NA	None Required
Shutdown	Yes	Yes	Yes[*]
		No	None Required
	No	NA	None Required

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

[*] See Page 2 of 2 Figure 4.0-1

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



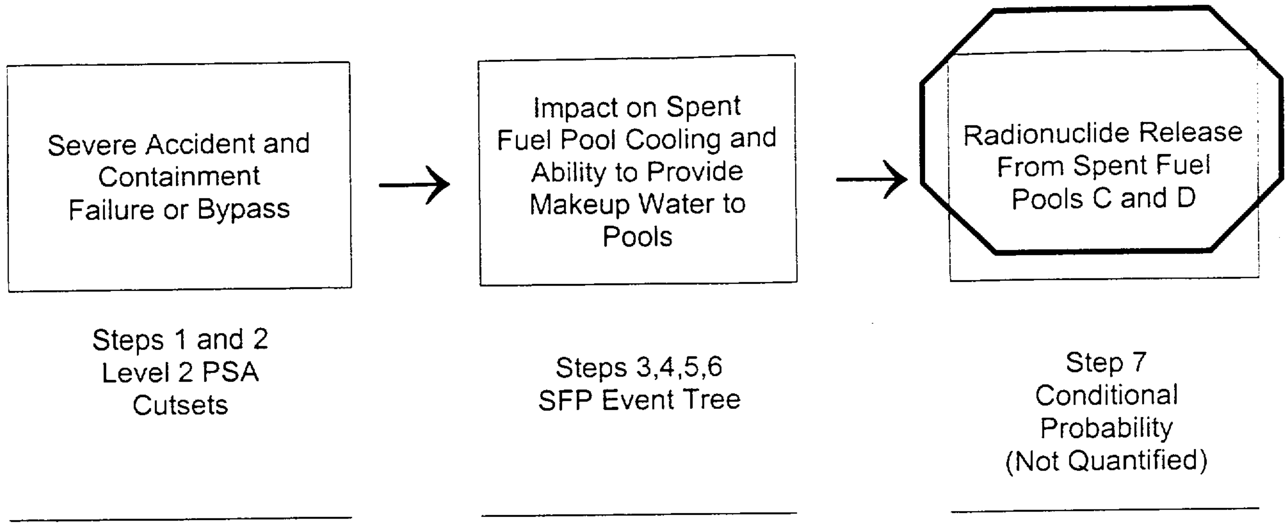


Figure 4.1-1 Internal Event Analysis Approach for Spent Fuel Pool Evaluation According to ASLB Order

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

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

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

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

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

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

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

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

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

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

CORE MELT PROGRESSION AND CONTAINMENT FAILURE MODE
• Containment Bypass (Large) (Includes ISLOCA)
• Containment Bypass (SGTR)
• Containment Isolation Failure
• Early Containment Failure
• Late Containment Failure
- Basemat Failure
- Overpressure Failure
• In Vessel Recovery and Containment Failure

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

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

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

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

Thermal Hydraulic Analysis

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

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

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

Systems

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

Data

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

HRA

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

Examples of the positive aspects are:

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

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

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

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

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

The model considered multiple questions regarding each operator action:

- How is the action diagnosed and by whom?

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

- How is the action carried out?

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

- How does accessibility play a role?

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

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

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

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

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

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

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

Dependencies Among Operator Actions

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

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

Dependencies

The treatment of dependencies included the following:

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

Structural

See Thermal Hydraulic Analysis.

Quantification

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

Level 2

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

Table 4.1-1

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

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

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

Table 4.1-1

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

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

Table 4.1-1

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

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

Table 4.1-1

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

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

Table 4.1-1

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

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

Table 4.1-1

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

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

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

Table 4.1-1

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

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

4.2 SEISMIC EVENTS

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

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

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

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

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

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

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

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

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

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

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

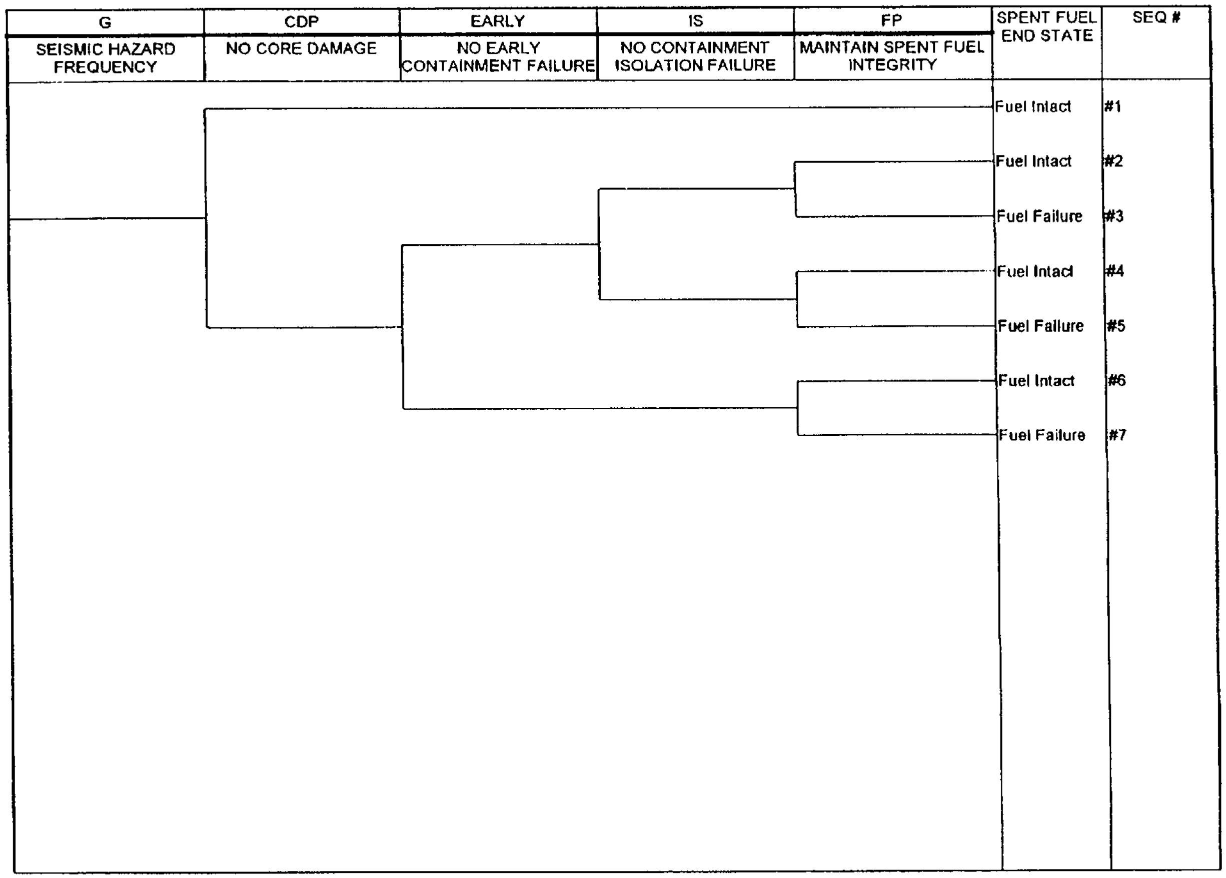


Figure 4.2-1 GRAPHICAL REPRESENTATION OF SEISMIC ANALYSIS

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

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

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

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

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

4.2.1 Seismic Hazard Frequency

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

Background

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

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

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

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

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

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

Frequency Estimation

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

Division of Seismic Hazard Curve

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

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

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

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

Table 4.2-1

SEISMIC HAZARD ESTIMATES FOR SHEARON HARRIS

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

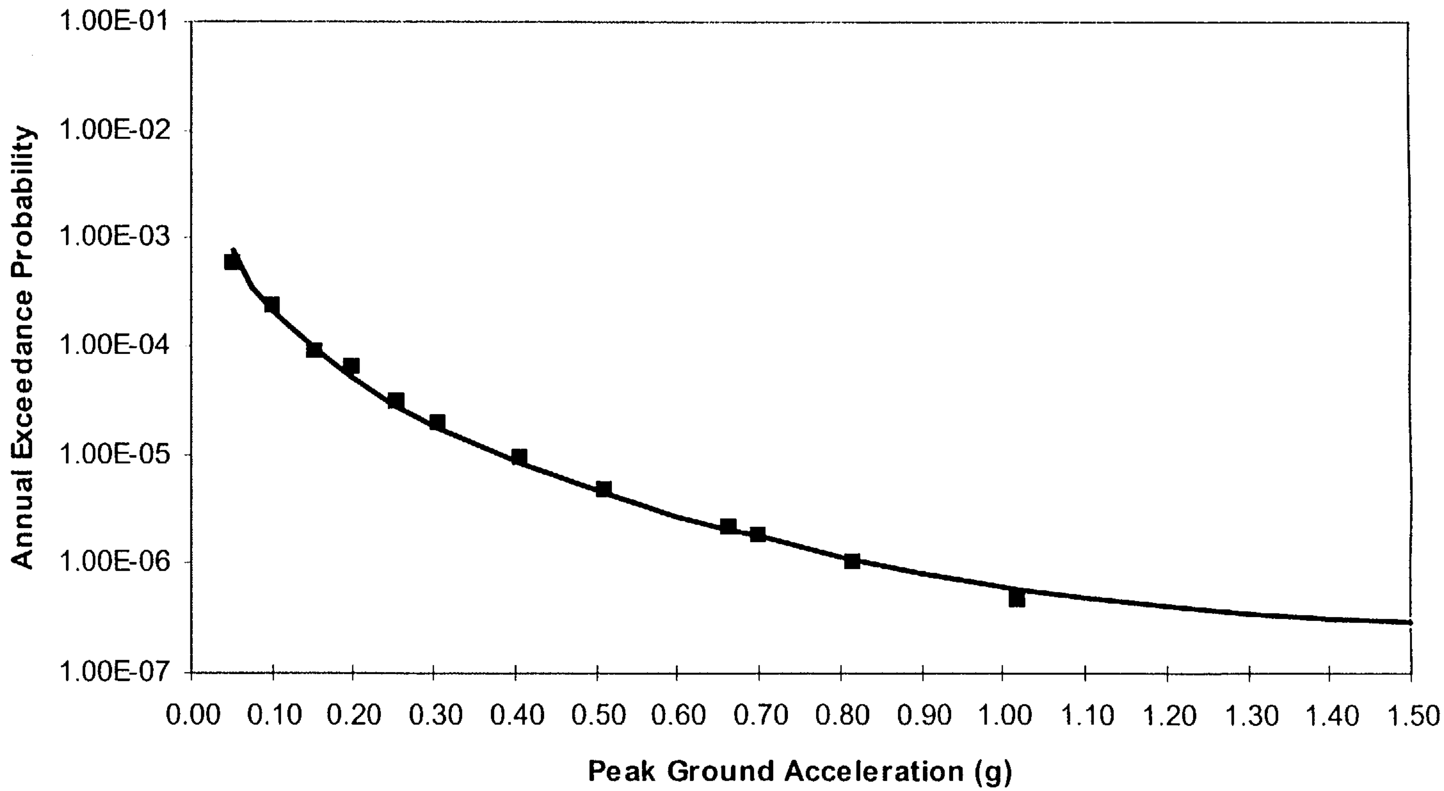
Notes:

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

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

Figure 4.2-2

SHEARON HARRIS SEISMIC HAZARD CURVE



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

4.2.2 Seismic Fragility Assessment

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

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

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

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

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

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

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

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

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

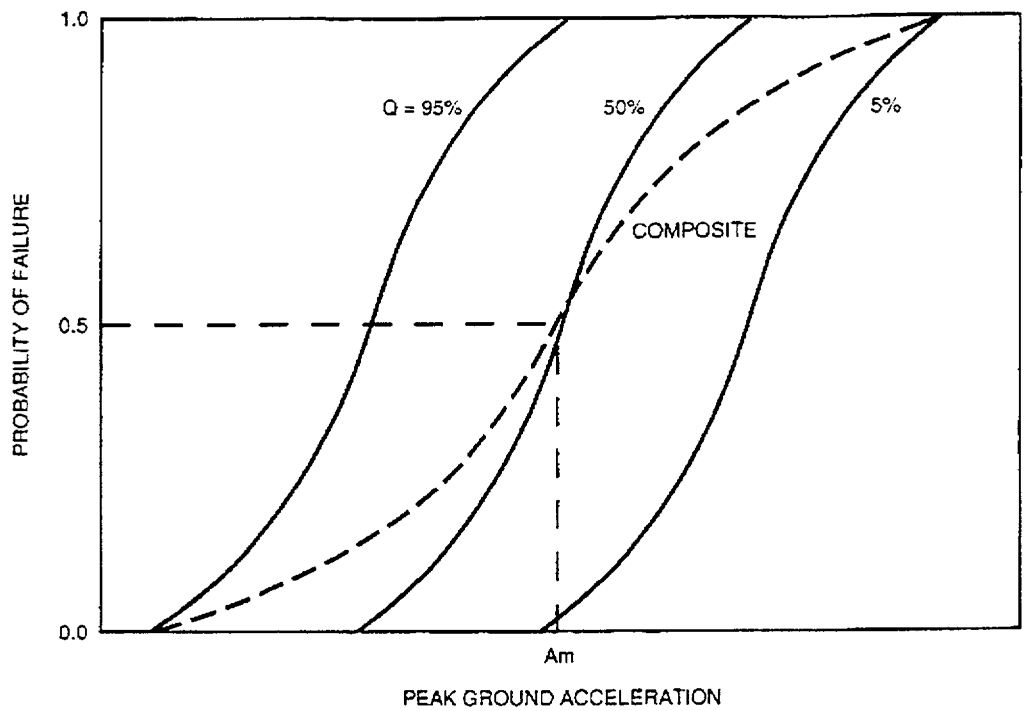


Figure 4.2-3 Typical Fragility Curves

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

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

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

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

Median Seismic Capacities

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

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

[1] Includes roads, bridges, communication systems.

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

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

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

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

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

Table 4.2-2

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

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

Notes to Table 4.2-2:

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

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

Seismic Fragilities

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

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

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

4.2.3 Seismic-Induced Core Damage

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

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

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

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

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

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

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

Table 4.2-3

SHNPP MEDIAN SEISMIC CAPACITIES

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

Notes to Table 4.2-3:

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

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

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

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

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

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

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

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

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

Table 4.2-4

SHNPP SEISMIC FRAGILITIES

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

4.2.4 Early Containment Failure Assessment

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

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

4.2.5 Containment Isolation Failure Assessment

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

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

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

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

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

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

Manual containment isolation was assigned a failure probability of 1.0.

4.2.6 Maintenance of Spent Fuel Pool Coolant Inventory

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

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

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

- Radiation environment
- Temperature environment

- Steam environment
- Door accessibility

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

Diesel Fire Pump Pathway

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

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

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

Demineralized Water System Pathway

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

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

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

Pumping Sources

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

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

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

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

Quantification of Alternate Alignment

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

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

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

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

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

Seismic Walkdown Insights

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

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

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

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

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

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

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

4.2.7 Seismic Quantification Summary

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

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

Constraints of the ASLB Order

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

Seismic Assessment Sensitivity Analysis

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Table 4.2-6

SUMMARY OF SEISMIC ASSESSMENT QUANTITATIVE SENSITIVITY CASES

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

NOTES
(1) Shaded cells indicate parameter changes with respect to the BASE Case
(2) Ten seismic hazard intervals between 0.0 and 1.0g, and one interval for >1.0g

Table 4.2-5

SPENT FUEL FAILURE DUE TO SEISMIC-INDUCED CORE DAMAGE SCENARIOS

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

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

4.3 FIRE INITIATED ACCIDENT SEQUENCES

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

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

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

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

Table 4.3-1

IPEEE DOMINANT FIRE INITIATORS

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

4.3.1 Fire Model

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

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

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

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

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

Table 4.3-2

FIRE MODEL

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

4.3.2 Quantification

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

Dependencies Among Operator Actions

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

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

4.3.3 Results

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

TABLE 4.3-3

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

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

4.4 AN ANALYSIS OF PWR SHUTDOWN RISK

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

4.4.1 Core Damage Frequency

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

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

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

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

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

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

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

4.4.1.1 Surry Data from NUREG-6144

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

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

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

4.4.1.2 Low Power Shutdown Workshop Information [4-7]

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

EPRI Perspective

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

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

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

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

South Texas Project (STP) Experience

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

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

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

Seabrook Shutdown PSA

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

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

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

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

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

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

Scientech Safety Monitor Experience

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

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

Westinghouse Experience

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

Additional insights are:

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

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

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

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

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

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

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

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

NUREG/CR-5015

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

NSAC 84

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

4.4.1.4 Industry Experience

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

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

Table 4.4-1

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

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

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

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

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

Table 4.4-2

SUMMARY OF CCDP FOR PWR REFUEL OUTAGES

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

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

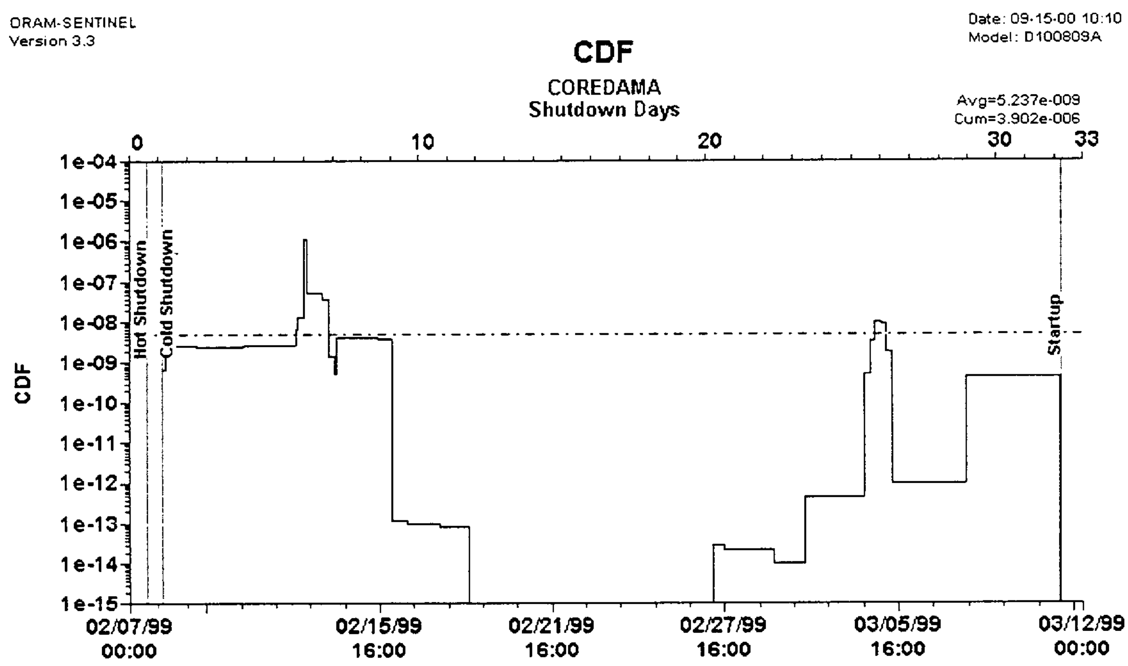


Figure 4.4-1 Typical PWR Refuel Outage CDF Profile

4.4.1.5 Summary of CDF Associated with Refueling Operations

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

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

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

4.4.2 Containment Integrity During Refuel Operations

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

4.4.2.1 Overview of Containment Integrity During Refuel Operations

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

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

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

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

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

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

4.4.2.2 Methodology

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

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

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

4.4.2.3 Evaluation of The Likelihood Of Containment Integrity

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

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

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

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

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

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

Containment Closure

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

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

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

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

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

   c. *Within the time to core boiling.*

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

Table 4.4-3
REPRESENTATIVE TIME AVAILABLE FOR ACTIONS

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

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

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

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

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

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

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

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

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

4.4.2.5 Containment Integrity Human Error Likelihood (Normal RCS Level)

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

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

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

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

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

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

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

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

4.4.3 Summary of Quantitative Results

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

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

Table 4.4.3-1

SUMMARY OF QUANTITATIVE RESULTS

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

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

4.5 OTHER EXTERNAL EVENTS

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

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