CASE SCHEDULED FOR ORAL ARGUMENT SEPTEMBER 5, 2002

In the
United States Court of Appeals
For the District of Columbia Circuit

Nos. 01-1073 and 01-1246 (Consolidated)

ORANGE COUNTY, NORTH CAROLINA, *Petitioner*
v.
UNITED STATES NUCLEAR REGULATORY COMMISSION
And the UNITED STATES OF AMERICA, *Respondents*
CAROLINA POWER & LIGHT COMPANY, *Intervenor-Respondent*

PETITION TO REVIEW A FINAL DECISION OF THE
U.S. NUCLEAR REGULATORY COMMISSION

JOINT APPENDIX VOLUME II: pages 542 through 1167

Diane Curran
Anne Spielberg
Harmon, Curran, Spielberg & Eisenberg, L.L.P.
1726 M Street N.W., Suite 600
Washington, D.C. 20036
202/328-3500

John H. O'Neill
David J. Cynamon
Douglas J. Rosinski
SHAWPITTMAN, L.L.P.
2300 N Street N.W.
Washington, D.C. 20037

Steve Carr, of Counsel
Progress Energy Service Co.
411 Fayetteville Street Mall
Raleigh, NC 27602-1551
919/546-4161

Karen D. Cyr
John F. Cordes, Jr.
E. Leo Slaggie
Charles E. Mullins
U.S. N.R.C.
Washington, D.C.
301/415-1606

Thomas L. Samsonetti
Assistant Attorney General

Ronald Spritzer
Attorney, Appellate Section
Environment and Natural Resources
U.S. Department of Justice
Washington, D.C. 20530
202/514-3977

Dated: June 4, 2002

JOINT APPENDIX

TABLE OF CONTENTS

Vol. II

Record

Summary of Facts, Data and Arguments on Which Applicant Proposes to Rely at the Subpart K Oral Argument Regarding Contention EC-6 and Volume 1 of Exhibits (November 20,2000).....544

Note regarding omissions from the Joint Appendix: Due to their volume, counsel for CP&L has agreed to omit the following materials from the Joint Appendix that were included in the record submittal of CP&L's evidentiary presentation of November 20, 2000. The Court is respectfully directed to the record to review the omitted materials.

Attachment D to Exhibit 5, Harris Plant Operating Manual, Vol. 6, Part 2, Fuel Pool Cooling and Clean-Up Systems (Exhibit 5 is the Affidavit of Eric A. McCartney)

Attachment E to Exhibit 5, Simplified Flow Diagram, Fuel Pools Clean-up Systems

Attachment F to Exhibit 5, Harris Plant Operating Manual, Vol. 6 Part 2, Demineralized Water System

Attachment G to Exhibit 5, Simplified Flow Diagram, Potable and Demineralized Water System

Attachment H to Exhibit 5, Harris Plant Operating Manual, Containment Spray System

Attachment I to Exhibit 5, Simplified Flow Diagram, Containment Spray System

Attachment J to Exhibit 5, Harris Plant Operating Manual, Service Water System

Attachment K to Exhibit 5, Simplified Flow Diagram, Circulating and Service Water System

Attachment L to Exhibit 5, Simplified Flow Diagram, Circulating and Service Water Systems

Attachment N to Exhibit 5, Simplified Flow Diagram, Demineralizer Water System

Attachment O to Exhibit 5, Harris Plant Operating Manual, Vol. 6 Part 2, Fire Protection/Detection Systems

Attachment P to Exhibit 5, Simplified Flow Diagram, Fire Protection System

Attachment Q to Exhibit 5, Harris Plant Operating Manual, Vol. 6 Part 2, Plant Lighting

Attachment R to Exhibit 5, Harris Plant Operating Manual,Vol. 3 Part 2, Fuel Pool Cooling

Attachment B to Exhibit 7, Harris Plant Operating Manual, Vol. 2, Part 5, Plant Emergency Procedure PEP-330) (Exhibit 7 is the Affidavit of Benjamin W. Morgan)

Exhibit 9, Deposition of Gareth W. Parry, Ph.D. (October 19, 2000).

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

Of Counsel:
Steven Carr
Legal Department
CAROLINA POWER & LIGHT COMPANY
411 Fayetteville Street Mall
Post Office Box 1551 – CPB 13A2
Raleigh, North Carolina 27602-1551
(919) 546-4161

John H. O'Neill, Jr.
Douglas J. Rosinski
SHAW PITTMAN
2300 N Street, N.W.
Washington, D.C. 20037-1128
(202) 663-8000
Counsel For CAROLINA POWER & LIGHT COMPANY

TABLE OF CONTENTS

TABLE OF AUTHORITIES iv

TABLE OF EXHIBITS AND ATTACHMENTS vii

I. INTRODUCTION 1

II. BCOC CANNOT SUSTAIN ITS BURDEN TO DEMONSTRATE THAT AN ADJUDICATORY HEARING MUST BE HELD TO RESOLVE CONTENTION EC-6 11

A. Contention EC-6 and the Questions Posed by the Board. 11

B. Congress Created Special Procedures For Spent Fuel Storage Expansion License Amendments. 14

C. The Purpose of Subpart K is to Expedite Resolution of Spent Fuel Licensing Issues. 15

D. Adjudicatory Hearings are Reserved for Genuine and Substantial Disputes of Material Facts That Cannot Be Resolved Without a Hearing. 17

E. BCOC Does Not Intend to Submit Facts or Data on Which to Base a Genuine and Substantial Dispute, Nor Has It Retained Experts Capable of Addressing the Board's Questions. 19

F. BCOC Cannot Sustain its Burden to Demonstrate an Adjudicatory Hearing is Required in this Proceeding. 28

III. THE NATIONAL ENVIRONMENTAL POLICY ACT DOES NOT REQUIRE PREPARATION OF AN EIS TO ADDRESS THE CONSEQUENCES OF BCOC'S POSTULATED SCENARIO 29

A. National Environmental Policy Act Requirements Are Well-Established. 29

B. The NRC Staff's Decision Not to Prepare an EIS Was Supported by Overwhelming Evidence that the Additional Environmental Impacts of the License Amendment Are Insignificant. 33

C. A Determination That BCOC's Postulated Scenario is Remote and Speculative is Consistent With Qualitative Guidelines, Commission Precedent, and Controlling Legal Authority. 37

D. A Frequency of Occurrence of One-in-a-Million Per Year Is a Reasonable Quantitative Threshold For Consideration of Remote and Speculative Events 46

IV. A STATE-OF-THE-TECHNOLOGY PROBABILISTIC ANALYSIS ESTABLISHES THAT THE FREQUENCY OF OCCURRENCE OF BCOC'S POSTULATED SCENARIO AT HARRIS IS SO LOW THAT IT IS HIGHLY REMOTE AND SPECULATIVE 50

A. Answer to Board Question 1: The Best Estimate Probability of BCOC's Postulated Scenario is on the Order of a Few Chances in One Hundred Million. 50

B. The Methodology Employed and Expertise Brought to Bear in Addressing Board Question 1 Was State-of-the-Technology and Relied Heavily on Harris-Specific Information. 51

C. The Probability of Initiating Events – A Severe Reactor Degraded Core Accident with Containment Bypass, Loss of Spent Fuel Pool Cooling and a Large Early Release of Fission Products Outside of Containment – Is Extraordinarily Low and Beyond the Harris Design Basis. 56

D. The Probability of Recovery of Spent Fuel Pool Cooling at Harris Before Evaporation Uncovers the Spent Fuel, After the Highly Unlikely Initiating Events Required By BCOC's Postulated Scenario, is Quite High Due to the Unique and Robust Design of the Harris Fuel Handling Building and the Multiple Alternate Sources of Makeup Water. 60

E. The Probability of a Self-Sustaining Exothermic Oxidation Reaction of Zircaloy Cladding of the Old, Cold Spent Fuel to be Stored in Harris Spent Fuel Pools C and D is Highly Unlikely in Any Event. 65

000540

F. While Applicant Attempted to Provide a "Best Estimate" Probability, the Resulting Analysis Still Contains Conservatisms That Tend to Overstate the Probability of BCOC's Postulated Scenario. 68

V. THE NUREG-1353 ESTIMATED VALUES ARE NOT RELEVANT TO DETERMINING THE FREQUENCY OF OCCURRENCE OF THE POSTULATED SCENARIO AT THE HARRIS PLANT 72

A. Answer to Board Question 2: The Probability Value of 2 x 10^{-6} Per Year Set Forth in the Executive Summary of NUREG-1353 is Not Relevant to BCOC's Postulated Scenario; In Any Event, the Assumed Conditional Probability for a Self-Sustaining Exothermic Reaction Cannot be Higher. 72

B. A Literature Review Does Not Suggest Changes to NUREG-1353 Values to the Extent They Are Relevant Here. 73

C. The Concerns Expressed in the April 13, 2000 ACRS Letter Do Not Suggest That the Probabilities of Individual Elements of the Postulated Scenario Are Greater Than Previously Analyzed. 77

VI. NEPA REQUIRES NO FURTHER ANALYSES 78

Answer to Board Question 3: The NRC Staff Does Not Have to Prepare Additional Environmental Impact Analyses Even If the Board Should Decide a Probability of Occurrence on the Order of a Few Chances in One Hundred Million Per Year is Not Sufficient to Classify BCOC's Postulated Scenario as Remote and Speculative 78

VII. ACTIONS REQUESTED OF THE BOARD 82

000547

TABLE OF AUTHORITIES

CASES

Baltimore Gas & Electric Co. v. NRDC, 462 U.S. 87 (1983). 29, 30, 33, 36, 79

Carmel By-The-Sea v. DOT, 95 F.3d 892 (9th Cir. 1996). 31, 35

Carolina Environmental Study Group v. U.S., 510 F.2d 796 (D.C. Cir. 1975).....31

Carolina Power & Light Co. (Shearon Harris Nuclear Plant), LBP-00-19 NRC ___, slip op. (August 7, 2000).....*Passim*

Dubois v. U.S. Dept. of Agric., 102 F.3d 1273 (1st Cir. 1996).....30, 31, 35

Duke Energy Corp. (Oconee Nuclear Station, Units 1, 2, and 3), CLI-99-11, 49 NRC 328 (1999).....37

Environmental Defense Fund v. Hoffman, 566 F.2d 1060 (8th Cir. 1977).....31

Idaho Sporting Congress v. Thomas, 137 F.3d 1146 (9th Cir. 1998).....31, 35

Kelley v. Selin, 42 F.3d 1501 (1995).....35

Kleppe v. Sierra Club, 427 U.S. 390 (1976).....29, 30, 79

Limerick Ecology Action v. NRC, 869 F.2d 719 (3rd Cir. 1989). 49, 50

Marsh v. Oregon Natural Resources Council, 490 U.S. 360 (1989). 35

Northeast Nuclear Energy Co. (Millstone Nuclear Power Station), LBP-00-2, 51 NRC 25 (2000).....37

Public Service Electric & Gas Co. (Hope Creek Generating Station, Units 1 and 2), ALAB-518, 9 NRC 14 (1979).....40, 41, 42

Robertson v. Methow Valley Citizens Council, 490 U.S. 346 (1989).....29, 30, 32, 33, 79

San Luis Obispo Mothers for Peace v. NRC, 751 F.2d 1287 (D.C. Cir. 1984).....30, 31

San Luis Obispo Mothers for Peace v. NRC, 789 F.2d 26 (D.C. Cir. 1986).....2, 41

Sierra Club v. Marsh, 976 F.2d 763 (1st Cir. 1992). 30

Vermont Yankee Nuclear Power Corp v. NRDC, 435 U.S. 519 (1978). 30, 31, 32, 42

Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), CLI-90-07, 32 NRC 129 (1990).....39

Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), CLI-90-04, 31 NRC 333 (1990).31, 32, 37, 38, 39, 40

Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), ALAB-919, 30 NRC 29 (1989). 36

Yankee Atomic Electric Co. (Yankee Nuclear Power Station), LBP-96-2, 43 NRC 61 (1996) 32

STATUTES AND REGULATIONS

10 C.F.R. § 2.714(e) (1997) 18
10 C.F.R. § 2.749 (1997) 19
10 C.F.R. § 2.1115(a)(2) (1997) 16
10 C.F.R. § 2.1115(b) (1997) 16
10 C.F.R. § 51.14 (1997) 31
40 C.F.R. § 1508.9 (1997) 31
40 C.F.R. § 1508.13 (1997) 31
42 U.S.C. §§ 4321-4347 (2000) 29
42 U.S.C. § 4332(2)(C) (2000) 29
42 U.S.C. § 10101 et seq. (2000) 14
42 U.S.C. § 10154(a) (2000) 14
42 U.S.C. § 10154(b) (2000) 15
50 Fed. Reg. 41,662 (1985) 15, 16
51 Fed. Reg. 28,044 (1986) 42, 43, 44
64 Fed. Reg. 71,514 (1999) 33

MISCELLANEOUS

128 Cong. Rec. S15,644 (daily ed. Dec. 20, 1982) 15
128 Cong. Rec. S4155 (daily ed. Apr. 28, 1982) 15
Carolina Power & Light Company Docket No. 50-400 Shearon Harris Nuclear Power Plant, Unit 1 Environmental Assessment and Finding of No Significant Impact (1999) 33, 34, 80
Commission Voting Record – SECY-00-0077, Comments of Chairman Meserve (June 27, 2000) 46
H.R. Rep. No. 97-785 (1982) 14, 15
Letter from Dana A. Powers to Richard A. Meserve, "Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," (April 13, 2000) 77

Letter from T.S. Kress to Shirley A. Jackson, "Risk-Informed, Performance-Based Regulation and Related Matters" (Aug. 15, 1996).....45

Licensing Board Memorandum and Order (Ruling on Standing and Contentions), slip op. (July 12, 1999).....4

Licensing Board Memorandum and Order (Subpart K Oral Argument Procedures), slip op. (Jan. 13, 2000).....17

Licensing Board Memorandum and Order (Ruling on Designation of Issues for an Evidentiary Hearing), slip op. (May 5, 2000).....4, 20

Memorandum from Annette L. Viette-Cook to William D. Travers, "Staff Requirements – SECY-00-0077 – Modifications to the Reactor Safety Goal Policy Statement" (June 27, 2000).....45

Memorandum from Samuel J. Chilk to James M. Taylor, "SECY-89-102 – Implementation of the Safety Goals" (June 15, 1990).....44

Orange County's Request for Admission to Late-Filed Environment Contentions (January 31, 2000).....27

NUREG-0972, "Final Environmental Statement Related to the Operation of Shearon Harris Nuclear Power Plant, Units 1 and 2" (1983).....2, 80

NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, 'Beyond Design Basis Accidents in Spent Fuel Pools'" (1989).....73, 74

NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants," Supplement 2 (1999).....49

"Policy on Conduct of Adjudicatory Proceedings; Policy Statement," 48 NRC 18 (1998).....16, 17

SECY-00-0077, "Modifications to the Reactor Safety Goal Policy Statement" (March 30, 2000).....45, 46

SECY-98-231, "Authorization of the Trojan Reactor Vessel Package for One-Time Shipment for Disposal" (October 2, 1998).....49

Shearon Harris Nuclear Power Plant Docket No. 50-400/License No. NPF-63 Request For License Amendment Spent Fuel Storage (Dec. 23, 1998).....2, 3

Solar and Heliospheric Observatory, Frequently Asked Questions, http://sohowww.nascom.nasa.gov/explore/faq/sun.htm#surface.....27

Staff Requirements Memorandum on SECY-00-0077.....46

000550

TABLE OF EXHIBITS AND ATTACHMENTS

Exhibit No.	Exhibit Title	Attachment No.	Attachment Title
1.	Affidavit of E.Burns	A	Resume with Publications
		B	ERIN Team Members
		C	ERIN Report – "Technical Input for use in the Matter of Shearon Harris Spent Fuel Pool Before the Atomic Safety and Licensing Board"
2.	Affidavit of R. Kunita	A	Resume
		B	References/Literature Survey
		C	Table of Reported Zircaloy Temperatures
		D	Figure 1 – Assembly Maximum Specific Heat (Shipments from Robinson - PWR Fuel)
		E	Figure 2 – Assembly Maximum Specific Heat (Shipments from Brunswick – BWR Fuel)
		F	Figure 3 – Assembly Specific Heat (PWR Fuel in Harris Pools A and B)
		G	Figure 4 – Assembly Specific Heat (BWR Fuel in Harris Pools)
		H	Figure 5 – Assembly Specific Heat (Harris High Burnup PWR Fuel)
		I	Figure 6 – Ruthenium Radioactivity
		J	Documents Evaluated to Determine Impact on NUREG-1353

000551

Exhibit No.	Exhibit Title	Attachment No.	Attachment Title
3.	Affidavit of S. Laur	A	Resume
		B	Peer Review of Shearon Harris PSA
		C	Summary of Reviews to Shearon Harris Nuclear Plant PSA, IPE, and IPEEE
		D	Plant-Specific Information Provided to ERIN
4.	Affidavit of S. Edwards	A	Resume
		B	Diagram Illustrating HNP Spent Fuel Storage Pools, Transfer Canals, and Current Bulkhead Gate Configuration
		C	Diagram Illustrating Anticipated Bulkhead Gate Configuration in the HNP Spent Fuel Pools Subsequent to Operational Use of C and D Pools
		D	Description of the Key Steps in the Spent Fuel Pool Heatup Calculations
		E	Data Sources for Input Values and Initial Conditions
		F	Summary Results of Heatup Calculations for Analyzed Scenarios
		G	Calculations to Determine Time Required to Reach Boiling Temperature and Additional Time Required to Boil Water to Top of Spent Fuel Racks
5.	Affidavit of E. McCartney	A	Resume
		B	Diagram Illustrating HNP Spent Fuel Storage Pools, Transfer Canals, and Current Bulkhead Gate Configuration

Exhibit No.	Exhibit Title	Attachment No.	Attachment Title
		C	Diagram Illustrating Anticipated Bulkhead Gate Configuration in the HNP Spent Fuel Pools Subsequent to Operational Use of C and D Pools
		D	SD-116, Fuel Pool Cooling and Clean-Up System Description
		E	Simplified Schematic of Spent Fuel Pool Cooling and Cleanup System (electronic copy not available)
		F	SD-143.03, Demineralized Water System Description
		G	Simplified Schematic of Demineralized Water System (electronic copy not available)
		H	SD-112, Containment Spray System Description
		I	Simplified Schematic of Refueling Water Storage Tanks and Connecting Systems (electronic copy not available)
		J	SD-139, Service Water System Description
		K	Simplified Schematic of Normal Service Water System (electronic copy not available)
		L	Simplified Schematic of Emergency Service Water System (electronic copy not available)
		M	SD-102, Primary Makeup System Description
		N	Simplified Schematic of Reactor Makeup Water Storage Tank and Connecting Systems (electronic copy not available)

Exhibit No.	Exhibit Title	Attachment No.	Attachment Title
		O	SD-149, Fire Protection/Detection Systems Description
		P	Simplified Schematic of Fire Protection System (electronic copy not available)
		Q	SD-158, Plant Lighting System Description
		R	OP-116, Fuel Pool Cooling and Cleanup Operating Procedure
6.	Affidavit of M. DeVoe	A	Resume
7.	Affidavit of B. Morgan	A	Resume
		B	In-Plant Dose Calculation Results
		C	Environmental Dose Calculation Results
		D	PEP-330, Radiological Consequences Plant Emergency Procedure
8.	Deposition Transcript of G. Thompson		
9.	Deposition Transcript of G. Parry		

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

I. INTRODUCTION

Pursuant to the Board's Memorandum and Order (Ruling on Late-Filed Environmental Contentions) dated August 7, 2000,[1] Applicant Carolina Power & Light Company ("CP&L") submits its "Summary of Facts, Data, and Arguments on which Applicant Proposes to Rely at the Subpart K Oral Argument Regarding Contention EC-6" ("Applicant's Summary"). As required by 10 C.F.R. § 2.1113(a), attached as exhibits to Applicant's Summary are supporting facts and data in the form of sworn written affidavits.

---

1 Carolina Power & Light Co. (Shearon Harris Nuclear Plant), LBP-00-19 NRC ___, slip op. (August 7, 2000) (hereinafter "Order").

000555

This proceeding relates to CP&L's December 23, 1998 application for a license amendment to place spent fuel pools C and D in service at CP&L's Harris Nuclear Plant ("Harris Plant," or "Harris").[2] Harris was originally planned as a four nuclear unit site (Harris Units 1, 2, 3 and 4). In order to accommodate four units, the Harris fuel handling building was designed and constructed with four separate pools capable of storing spent fuel. Spent fuel pools A and B were originally intended to support Harris Units 1 and 4. Spent fuel pools C and D were originally intended to support Harris Units 2 and 3.

Harris Units 3 and 4 were canceled in late 1981. Harris Unit 2 was canceled in late 1983. Spent fuel pools A, B, C and D and the spent fuel pool cooling and cleanup system ("SFPCCS") for spent fuel pools A and B were completed as part of the fuel handling building, are described in the Harris Final Safety Analysis Report ("FSAR"), and are licensed as part of Harris. Construction on the SFPCCS for spent fuel pools C and D was discontinued after Harris Unit 2 was canceled. By that time, all four spent fuel pools had been constructed, concrete had been poured, and the SFPCCS piping was installed, welded in place and embedded in reinforced concrete.

The Final Environmental Statement[3] supported the issuance of the Operating License for Harris Unit 1 alone, as Harris Unit 2 had been cancelled. The FES, however, considered two-unit operation and bounded the environmental impacts for single unit

---

2 Shearon Harris Nuclear Power Plant Docket No. 50-400/License No. NPF-63 Request For License Amendment Spent Fuel Storage (Dec. 23, 1998) (hereinafter "License Amendment Application").

3 NUREG-0972, "Final Environmental Statement Related to the Operation of Shearon Harris Nuclear Power Plant, Units 1 and 2" (1983) (hereinafter "FES").

operation. In fact, the maximum number of fuel assemblies contemplated at the time of the FES, for two-unit operation with all four spent fuel pools, exceeds the maximum number of fuel assemblies that will be stored pursuant to the instant License Amendment Application, because of the 1.0 MBTU/hr limit on total heat generation in spent fuel pools C and D.[4]

Harris Unit 1 began commercial operations in 1987. In addition, Harris was licensed to accept spent fuel for storage from CP&L's other nuclear plants, H. B. Robinson Unit 2, and Brunswick Units 1 and 2. Beginning in 1989, spent fuel assemblies from Robinson and Brunswick with cooling time greater than five years have been regularly shipped to Harris and are stored in spent fuel pools A and B.

The December 23, 1998 License Amendment Application and the need to expand spent fuel storage at Harris result from the failure of the U.S. Department of Energy ("DOE") to begin taking delivery of spent fuel in 1998, as required by the contract between DOE and CP&L and by the Nuclear Waste Policy Act of 1982, as amended.

---

4 The Applicant's License Amendment Application includes the addition of Technical Specification 5.6.3.d to the Harris operating license, which requires that "[t]he heat load from fuel stored in Pools 'C' and 'D' shall not exceed 1.0 MBtu/hr." Lic. Amend. App., Encl. 5 at 5-7a. Pursuant to the 1.0 MBTU/hr Technical Specification limit, Applicant does not currently intend to load any fuel in spent fuel pool D under this license amendment. See Lic. Amend. App., Encl. 1 at 4 (pool D is not scheduled for use until 2016). The total number of assemblies in pools A, B and C combined, even if pool C was loaded to its maximum capacity, is less than the total number of assemblies that was considered in the FES. Compare Lic. Amend. App. Enc. 1 at 1 (Harris originally licensed for up to 7640 assemblies), with id. at 3 (pools A, B and C combined would store 7359 assemblies). See also Lic. Amend. App., Encl. 5 at 5-7, Technical Specification 5.6.3.

000557

CP&L had requested that the license amendment to allow placement of spent fuel in spent fuel pools C and D be issued no later than December 31, 1999. CP&L originally planned to begin loading spent fuel in pool C in 2000. Further delays could adversely impact CP&L's ability to maintain adequate spent fuel storage capacity and, with the loss of full core discharge capability at one or more of CP&L's nuclear plants, could lead to a forced shutdown of the CP&L nuclear units.

Applicant invoked the Subpart K Procedures after the Board admitted Technical Contentions 2 and 3 proffered by intervenor Board of Commissioners of Orange County ("BCOC").[5] On January 21, 2000, the Board heard oral argument on whether to designate either of the two admitted issues for an evidentiary hearing. The Board determined that BCOC had failed to show that there was a genuine and substantial dispute of fact or law that could only be resolved by an evidentiary hearing, and disposed of both contentions in CP&L's favor.[6]

The Board admitted Contention EC-6 for litigation on August 7, 2000. The parties conducted discovery pursuant to the Board's schedule, which required completion of discovery by October 20, 2000.[7]

---

5 Licensing Board Memorandum and Order (Ruling on Standing and Contentions), slip op. (July 12, 1999).

6 Licensing Board Memorandum and Order (Ruling on Designation of Issues for an Evidentiary Hearing) slip op. at 88-89 (May 5, 2000).

7 Order at 19. During the discovery period, counsel for Applicant deposed BCOC's sole proffered expert, Dr. Gordon Thompson; BCOC's counsel deposed CP&L's experts Dr. Edwards Burns and Mr. Robert Kunita, CP&L's Manager of Environmental & Radiation Control, Mr. Ed Wills, and NRC Staff experts Dr. Gareth Parry, Robert Palla, and Stephen LaVie. In addition, Applicant provided

Footnote continued on next page

000558

This Applicant's Summary presents the facts, data, and arguments on which Applicant proposes to rely at the oral argument with regard to Contention EC-6.

Part II of Applicant's Summary describes the strict standards for an adjudicatory hearing required by 10 C.F.R. Part 50, Subpart K and the burden of proof that BCOC cannot possibly sustain.

Part III discusses the law applicable to determining whether consideration of the consequences of BCOC's postulated scenario involving a sequence of seven events, which begins with a postulated severe reactor accident with containment failure or bypass and a release of radionuclides (the "postulated scenario"), is required pursuant to the National Environmental Policy Act.

Part IV answers the Board's first question as set forth in the Order, and discusses Applicant's best estimate of the overall probability of the postulated scenario at Harris.

Part V answers the Board's second question and discusses whether any recent developments or new data or models suggest modification of the probability value determined in NUREG-1353 and whether any of the concerns expressed in the Advisory Committee on Reactor Safeguards' ("ACRS") letter dated April 13, 2000, are applicable to the postulated scenario. We also discuss the relevance of NUREG-1353 to the postulated scenario.

Footnote continued from previous page

BCOC's counsel and Dr. Gordon Thompson a guided tour of the Harris Plant and took photographs of plant features requested by BCOC. The parties responded to interrogatories and produced documents in response to requests for relevant documents.

000559

Part VI answers the Board's third question and discusses why no additional environmental impact analysis by the NRC Staff is required under any circumstance.

Part VII states the actions requested of the Board by Applicant at the conclusion of oral argument.

Applicant's Summary is supported by seven sworn statements in the form of affidavits with supporting attachments. We introduce each affidavit and its purpose below.

Exhibit 1 is the Affidavit of Dr. Edward T. Burns ("Burns Affidavit"). Dr. Burns is employed by ERIN Engineering and Research, Inc. ("ERIN") as Vice President and General Manager of BWR Technology. ERIN is the industry leader in risk management and application of risk and reliability analysis techniques to various situations and activities at nuclear power plants. Dr. Burns' affidavit describes the extensive probabilistic analysis and review effort performed by ERIN to determine the best estimate of the overall probability of the postulated scenario. First, Dr. Burns describes his role in preparing a response to the Board's questions, the tasks assigned to ERIN by CP&L, and the team he assembled to perform those tasks. Second, he describes generally the bases of probabilistic risk assessment, the advances in techniques and knowledge since initial applications, and the quality of the existing Harris Individual Plant Examinations ("IPE") and updated Probabilistic Safety Assessment ("PSA"). Third, he discusses the methodology and results, including uncertainty, of the ERIN analyses. Dr. Burns concludes that the postulated scenario has a best estimate overall

annualized probability of occurrence at Harris of less than three in one hundred million. ERIN's comprehensive technical report is Attachment C to Exhibit 1 ("ERIN Report").

Exhibit 2 is the Affidavit of Robert K. Kunita ("Kunita Affidavit"). Mr. Kunita has been employed by CP&L since 1973 and is currently a Principal Engineer, Spent Fuel Management. Mr. Kunita's affidavit evaluates the likelihood of the occurrence of a self-sustaining exothermic oxidation reaction of zircaloy spent fuel cladding in Harris spent fuel pools C and D following a postulated evaporation of water uncovering the spent fuel (i.e., "Step 7" in the postulated scenario). First, he describes the principles of a self-sustaining exothermic oxidation reaction of zircaloy spent fuel cladding. Second, Mr. Kunita discusses the literature survey he conducted to research the likelihood of a self-sustaining exothermic oxidation reaction of zircaloy spent fuel cladding. Third, he describes the application of the information obtained in his literature survey to the specific spent fuel to be stored in Harris spent fuel pools C and D and the analyses he performed to establish that a self-sustaining exothermic oxidation reaction of zircaloy spent fuel cladding is highly unlikely at Harris. Finally, Mr. Kunita concludes that the old, cold fuel to be stored in Harris spent fuel pools C and D, is highly unlikely to undergo such a self-sustaining exothermic oxidation reaction even if evaporation of the pool water occurs.

Exhibit 3 is the Affidavit of Steven A. Laur, P.E. ("Laur Affidavit"). Mr. Laur is the CP&L Superintendent of the Probabilistic Safety Assessment Unit. The purpose of Mr. Laur's affidavit is to describe the scope of engagement and Harris-specific information that was provided to ERIN for performance of ERIN's analysis of the

postulated scenario. First, Mr. Laur describes the documents, including the Harris PSA and the Harris Individual Plant Examination of External Events ("IPEEE"), that were used to perform the ERIN probabilistic analysis. Second, he discusses the specific steps, including an independent peer review of the Harris PSA, that CP&L took to ensure that the ERIN analysis was consistent with the Harris-specific attributes. Finally, Mr. Laur concludes that the ERIN analysis is of high quality and appropriately uses the Harris updated PSA model, the Harris IPEEE analysis and other Harris-specific information.

Exhibit 4 is the Affidavit of R. Steven Edwards ("Edwards Affidavit"). Mr. Edwards has been employed by CP&L since 1982 and is presently the Supervisor, Spent Fuel Project, responsible for commissioning and placing into service Harris spent fuel pools C and D. The purpose of his affidavit is to set forth the data and calculations on which CP&L relies in establishing the time to heat up the Harris spent fuel pools to boiling, and after boiling has started, the additional time necessary to boil the coolant level down to the top of the spent fuel racks. First, Mr. Edwards summarizes the background of the License Amendment Application and the information submitted in support of the application. Second, he describes the Harris spent fuel pool physical arrangement and associated equipment. Third, Mr. Edwards discusses the heatup calculations and their applicability to the Harris spent fuel pools. Fourth, he discusses the data and assumptions used in calculations. Finally, he describes the results of the time to heat-up and time to boil calculations. Mr. Edwards calculates that the time available to restore makeup water to the spent fuel pools is over a week under worst case assumptions.

Exhibit 5 is the Affidavit of Eric A. McCartney ("McCartney Affidavit"). Mr. McCartney is the Supervisor, Licensing/Regulatory Programs, responsible for managing regulatory interfaces for Harris. The purpose of his affidavit is to describe the numerous, diverse sources of water and methods of delivery which exist for establishing makeup to the Harris spent fuel pools. First, Mr. McCartney describes the Harris spent fuel pool physical arrangement, systems configurations, and plant equipment associated with normal and alternate makeup to the spent fuel pools. Second, he discusses the methods available for supplying makeup water to the Harris spent fuel pools and identifies the Harris procedures, controls, conditions, and equipment that establish the viability of each method. Third, Mr. McCartney describes the Technical Support Center ("TSC"), its functions and personnel, and how the Severe Accident Management Guidelines ("SAMGs") are used to assist the operating staff in responding to emergency conditions outside of existing procedures. Finally, he concludes that there are numerous, diverse methods for providing cooling and makeup water to the Harris spent fuel pools following a loss of normal cooling, that Harris operators are trained and capable of performing the actions necessary to initiate one or more of these methods under emergency conditions, and that the necessary tools and equipment are available to perform the required actions.

Exhibit 6 is the Affidavit of Michael J. DeVoe ("DeVoe Affidavit"). Mr. DeVoe is a nuclear engineer, employed by CP&L since 1984. He presently works in the Nuclear Fuel Services Unit of CP&L's Nuclear Fuel Management & Safety Analysis Section. The purpose of his affidavit is to describe the reactor core radioisotope inventory utilized in the dose rate calculations for the postulated scenario. First, Mr. DeVoe describes the

key assumptions and methodology used to develop the reactor core radioisotope inventory used in analyzing the postulated scenario. Second, he describes the CP&L owner's reviews performed on the reactor core radioisotope inventory calculation. Third, Mr. DeVoe describes the information provided to other CP&L personnel for use in performing the dose calculations. Finally, Mr. DeVoe concludes that the use of the calculated reactor core isotope inventory is appropriate for calculating dose rates resulting from the postulated scenario.

Exhibit 7 is the Affidavit of Benjamin W. Morgan, C.H.P. ("Morgan Affidavit"). The purpose of his affidavit is to describe the process he employed in performing the dose rate calculations to enable ERIN to determine the accessibility of Harris buildings and external areas following releases of radionuclides from the postulated scenario. First, Mr. Morgan describes the information he used as input to his calculations. Second, he discusses the methodology and assumptions he used in evaluating the dose rates at various locations resulting from the postulated scenario. Third, Mr. Morgan describes the methodology and assumptions he used to determine potential access restrictions and the information provided to ERIN. Fourth, he discusses the conservatisms in the dose rate calculations. Finally, he concludes that his dose calculations accurately represent a conservative estimate of conditions expected following the postulated scenario based on accepted industry analysis methodologies and Harris-specific information. He also concludes with a high degree of confidence that his dose calculation results demonstrate that certain internal and external areas at Harris are sufficiently accessible within

96 hours of the postulated scenario to allow personnel entrance to mitigate a postulated loss of spent fuel pool cooling and makeup.

Two other exhibits are attached for the convenience of the Board:

Exhibit 8 is the transcript of the sworn deposition of BCOC's designated expert Dr. Gordon Thompson ("Thompson Deposition").

Exhibit 9 is the transcript of the sworn deposition of the NRC Staff's expert on probabilistic risk assessment Dr. Gareth W. Parry ("Parry Deposition").

II. BCOC CANNOT SUSTAIN ITS BURDEN TO DEMONSTRATE THAT AN ADJUDICATORY HEARING MUST BE HELD TO RESOLVE CONTENTION EC-6

A. Contention EC-6 and the Questions Posed by the Board.

BCOC Contention EC-6, "Environmental Impact Statement Required," reads:

> In the Environmental Assessment ("EA") for CP&L's December 23, 1998, license amendment application, the NRC Staff concludes that the proposed expansion of spent fuel storage capacity at the Shearon Harris nuclear power plant will not have a significant effect on the quality of the human environment. Environmental Assessment and Finding of No Significant Impact Related to Expanding the Spent Fuel Pool Stage Capacity at the Shearon Harris Nuclear Power Plant (TAC No. MA4432) at 10 (December 15, 2000). Therefore, the Staff has decided not to prepare an Environmental Impact Statement ("EIS") for the proposed license amendment. The Staff's decision not to prepare an EIS violates the National Environmental Policy Act ("NEPA") and NRC's implementing regulations, because the Finding of No Significant Impact ("FONSI") is erroneous and arbitrary and capricious. In fact, the proposed expansion of spent fuel pool storage capacity at Harris would create accident risks that are significantly in excess of the risks identified in the EA, and significantly in excess of accident risks previously evaluated by the NRC Staff in the EIS for the Harris

> operating license. These accident risks would significantly affect the quality of the human environment, and therefore must be addressed in an EIS.
>
> There are two respects in which the proposed license amendment would significantly increase the risk of an accident at Harris:
>
> (1) CP&L proposes several substantial changes in the physical characteristics and mode of operation of the Harris plant. The effects of these changes on the accident risk posed by the Harris plant have not been accounted for in the Staff's EA. The changes would significantly increase, above present levels, the probability and consequences of potential accidents at the Harris plant.
>
> (2) During the period since the publication in 1979 of NUREG-0575, the NRC's Generic Environmental Impact Statement ("GEIS") on spent fuel storage, new information has become available regarding the risks of storing spent fuel in pools. This information shows that the proposed license amendment would significantly increase the probability and consequences of potential accidents at the Harris plant, above the levels indicated in the GEIS, the 1983 EIS for the Harris operating license, and the EA. The new information is not addressed in the EA or the 1983 EIS for the Harris operating license.
>
> Accordingly, the Staff must prepare an EIS that fully considers the environmental impacts of the proposed license amendment, including its effects on the probability and consequences of accidents at the Harris plant. As required by NEPA and Commission policy, the EIS should also examine the costs and benefits of the proposed action in comparison to various alternatives, including Severe Accident Mitigation Design Alternatives ("SAMDAs")and the alternative of dry storage.[8]

To support its contention, BCOC postulated the following seven-step chain of events ("postulated scenario"):

---

8 Order at 10-11 (internal footnote omitted).

000566

(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.[9]

In order to assess the significance of materials submitted in support of their positions, the Board asked the parties to address the following points:

1. What is the submitting party's best estimate of the overall probability of the sequence set forth in the chain of seven events in the CP&L and BCOC's filings, set forth on page 13 supra? The estimates should utilize plant-specific data where available and should utilize the best available generic data where generic data is relied upon.

2. The parties should take careful note of any recent developments in the estimation of the probabilities of the individual events in the sequence at issue. In particular, have new data or models suggested any modification of the estimate of 2×10^{-6} per year set forth in the executive summary of NUREG-1353, Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools (1989)? Further, do any of the concerns expressed in the ACRS's April 13, 2000 letter suggest that the probabilities of individual elements of the sequence are greater than those previously analyzed (e.g., is the chance of occurrence of sequence element seven, an exothermic reaction, greater than assumed in the decade-old NUREG-1353)?

---

9 Id. at 13.

000567

3. Assuming the Board should decide that the probability involved is of sufficient moment so as not to permit the postulated accident sequence to be classified as "remote and speculative," what would the overall scope of the environmental impact analysis the staff would be required to prepare (i.e., limited to the impacts of that accident sequence or a full blown EIS regarding the amended request)?[10]

B. Congress Created Special Procedures For Spent Fuel Storage Expansion License Amendments.

In the Nuclear Waste Policy Act of 1982,[11] Congress recognized that it would be many years before a permanent repository was ready to accept spent nuclear fuel. The Act provided special expedited licensing procedures designed "to encourage utilities to expand storage capacity at reactor sites."[12] The new procedures require written submissions and sworn testimony on any contentions, along with oral argument on the issues.[13] Following the oral argument, the Licensing Board must determine whether any of the contentions merits an adjudicatory hearing:

> (b) ADJUDICATORY HEARING. (1) At the conclusion of any oral argument . . . , the Commission shall designate any disputed question of *fact*, together with any remaining questions of law, for resolution in an adjudicatory hearing only if it determines that —
>
> > (A) there is a genuine and substantial dispute of *fact* which can only be resolved with sufficient accuracy by the introduction of evidence in an adjudicatory hearing; and

---

10 Id. at 17.

11 42 U.S.C. § 10101 et seq. (2000).

12 H.R. Rep. No. 97-785, at 39 (1982).

13 42 U.S.C. § 10154(a) (2000).

> (B) the decision of the Commission is likely to depend in whole or in part on the resolution of such dispute.[14]

Congress reasoned that by "scoping" the issues in this manner, the time and expense of adjudicatory hearings could be avoided unless the *factual* issues were truly significant and capable of accurate resolution only through full-blown adjudicatory proceedings.[15] It was recognized that the standards for an adjudicatory hearing were "extremely narrow."[16] Nevertheless, the narrow standards were judged necessary for a "streamlined regulatory process" that would "insure predictable and timely measures necessary to keep America's nuclear power plants in full operation without any threat of reduced operations or shutdown because of a failure by the Federal Government to provide for interim spent fuel management."[17]

C. The Purpose of Subpart K is to Expedite Resolution of Spent Fuel Licensing Issues.

The Nuclear Regulatory Commission implemented the Act's new procedures via a 1985 rulemaking that added Subpart K to the Commission's regulations.[18]

The regulations track the statutory language. Thus, an issue may be designated for an adjudicatory hearing *only* if (1) there is a genuine and substantial dispute of fact; *and* (2) the dispute can be resolved with sufficient accuracy *only* through introduction of

---

14 Id. § 10154(b) (emphasis added).

15 H.R. Rep. No. 97-785, at 39, 82.

16 128 Cong. Rec. S15,644 (daily ed. Dec. 20, 1982) (statement of Sen. Mitchell).

17 128 Cong. Rec. S4155 (daily ed. Apr. 28, 1982) (statement of Sen. McClure).

18 50 Fed. Reg. 41,662 (1985).

evidence at an adjudicatory hearing; *and* (3) the Commission's ultimate decision is likely to depend in whole or in part on the resolution of the dispute.[19] Any issues not meeting this test are to be disposed of by the Licensing Board promptly after the oral argument.[20]

Promptness, or the lack thereof, is an issue of significant weight in light of the two-year length of these proceedings and the associated burdens already placed upon Applicant. The Commission in adopting Subpart K acknowledged that the purpose of NWPA section 134 "is *to encourage and expedite the licensing of onsite spent fuel expansions and transshipments*."[21] Further, the Commission reiterated "its long-standing commitment to the expeditious completion of adjudicatory proceedings" only a few months before Applicant submitted the License Amendment Application at the focus of this proceeding.[22] An expedited resolution of this proceeding is required by the Commission's rules and policy.

---

19 10 C.F.R. § 2.1115(b) (1997) (emphasis added).

20 Id. § 2.1115(a)(2). The proposed rule would have required the Licensing Board to "decide" all issues not designated for an adjudicatory hearing. 48 Fed. Reg. 54,499, 54,505 (1983). The Edison Electric Institute and a group of interested utilities submitted comments challenging the proposed language requiring the Board to "decide" all issues, when in fact "dismiss" may be the more appropriate way to resolve certain issues. The NRC accommodated this comment in the final rule by using the term "dispose," which can include both "decide" and "dismiss."

21 50 Fed. Reg. at 41,665 (emphasis added).

22 "Policy on Conduct of Adjudicatory Proceedings; Policy Statement," 48 NRC 18, 24 (1998).

D. Adjudicatory Hearings are Reserved for Genuine and Substantial Disputes of Material Facts That Cannot Be Resolved Without a Hearing.

In adopting the Subpart K regulations, the Commission made it clear that the threshold for an adjudicatory hearing is strict:

> The Commission continues to believe that the statutory criteria are sufficient. As the Commission pointed out in connection with the proposed rules, the statutory criteria are *quite strict* and are designed to ensure that the hearing is focused exclusively on *real issues*. They are similar to the standards under the Commission's existing rule for determining whether summary disposition is warranted. *They go further, however, in requiring a finding that adjudication is necessary to resolution of the dispute and in placing the burden of demonstrating the existence of a genuine and substantial dispute of material fact on the party requesting adjudication.*[23]

The Board reminded the parties of BCOC's burden of demonstrating the existence of a genuine and substantial dispute of material fact in the Order directing the Subpart K proceeding.[24] Accordingly, as with its earlier, rejected contentions, BCOC again bears the burden of demonstrating that it is entitled to an adjudicatory hearing.

The Subpart K rules must be strictly applied to limit such hearings to real issues that can be decided only through formal adjudicatory procedures. First, there must be a dispute of fact. Pure questions of law obviously do not require an adjudicatory hearing

---

23 Id. at 41,667 (emphasis added).

24 Licensing Board Memorandum and Order (Subpart K Oral Argument Procedures), slip op. at 2 (Jan. 13, 2000).

and can be resolved by the Board on the briefs.[25] The only exceptions might be legal issues so interrelated with factual issues designated for a full hearing that they cannot be decided independent of the factual determination. Legal issues standing alone can never justify an adjudicatory hearing.

Second, the factual dispute must be genuine and substantial. If the dispute is genuine but peripheral or of secondary importance, then no hearing is warranted and the Board can resolve the issue on the basis of the sworn testimony and written submissions filed by the parties.

Third, even if the factual dispute is genuine and substantial, a hearing is still not warranted unless it is the type of dispute that can be accurately resolved only with traditional adjudicatory procedures, such as oral testimony from live witnesses subject to cross-examination. This might be the case, for example, if the issue turned primarily on the credibility of a particular witness. Most factual disputes, however, depend on technical or scientific issues that can be accurately decided on written submissions. Such issues are typically decided on the basis of plant records, scientific reports and other written materials that the Board itself can evaluate, drawing upon its own technical expertise. In this sense, the Subpart K rules go beyond the usual summary disposition procedures, as the Commission pointed out. Under the usual summary disposition

---

25 See 10 C.F.R. § 2.714(e) (1997) ("If the Commission or the presiding officer determines that any of the admitted contentions constitute pure issues of law, those contentions must be decided on the basis of briefs or oral argument according to a schedule determined by the Commission or presiding officer.").

procedures, any genuine issue of material fact requires a hearing.[26] Under Subpart K, by contrast, Licensing Boards must dispose of genuine factual issues without a hearing, if they are able to do so with sufficient accuracy.

Fourth, the resolution of the factual issue must be *central* to the ultimate decision in the case. In contrast, the summary disposition rules simply require the factual issue to be "material."[27] The Subpart K rules provide that a hearing may be held only if the Commission's decision "is likely to depend in whole or in part" on the resolution of the factual dispute. This is a stricter threshold than simple materiality. It implies that the factual issue must play a central role in the ultimate outcome of the case as a whole. Failing that, no adjudicatory hearing may be held.

E. BCOC Does Not Intend to Submit Facts or Data on Which to Base a Genuine and Substantial Dispute, Nor Has It Retained Experts Capable of Addressing the Board's Questions.

It may appear self-evident, but a genuine and substantial factual dispute requires the opposing parties to identify and argue relevant *facts*. Applicant, as discussed in detail below, has assembled data, analyses, and expert evaluations to support its position. The facts are presented in sworn affidavits by a team of individuals in various disciplines who have the relevant education, training, knowledge, experience, and access to identify and discuss relevant facts. These facts are interpreted by experts with the education, training, knowledge and experience to understand the facts, apply state-of-the-technology probabilistic assessment methodologies, and provide expert opinions necessary to answer

---

26 10 C.F.R. § 2.749 (1997).

27 Id. § 2.749(d).

the Board's questions. The NRC Staff, we understand, has also expended significant effort to do likewise.[28]

BCOC has not retained individuals who by education, training, knowledge or experience are capable of attesting to relevant facts or evaluating their significance. BCOC continues to rely solely on Dr. Gordon Thompson to attempt to address the wide range of technical issues involved in analyzing the complex accident scenario that he postulated.[29] This would be a daunting task for any one individual – even one with strong technical credentials. As the Board has found previously, however, Dr. Thompson's "expertise relative to reactor technical issues seems largely policy-oriented rather than operational."[30] Dr. Thompson's deposition during this phase of the proceeding once again confirmed that he lacks relevant knowledge and technical expertise to make a substantive contribution to an adjudicatory hearing.

While Dr. Thompson claims to be an expert "for purposes of this proceeding" capable of leading a team of experts on a multi-year research process to address the Board's questions on his own postulated scenario,[31] his answers to questions suggest otherwise:

---

28 Parry Dep. at 46-47 (describing the use of information from a variety of experts to answer the Board's questions).

29 Thompson Dep. at 28.

30 See Licensing Board Memorandum and Order (Ruling on Designation of Issues for an Evidentiary Hearing) slip op. at 51 n.9 (May 5, 2000); See Order at 9.

31 Thompson Dep. at 56.

Q Have you taken any specific seminars or other courses after your doctorate at Oxford that would include -- would be categorized as education on reactor accidents?

A No.

Q Have you performed any accident analyses using the codes that are accepted by the Nuclear Regulatory Commission or the regulatory commissions of any other country as appropriate to analyze reactor accidents?

A I have worked with consequence codes, but not codes that pertain to the incontainment aspects of reactor accidents.

. . .

Q I've been -- I've tried to be very careful to ask this question each time, and I want to just now sum. As I understand it, with respect to all of the studies in which you have mentioned, you have not yourself performed any original calculations or accident analyses using codes on reactor accidents? Is that true?

A That is correct, yes.

. . .

Q In connection with your deposition [i]n October, when asked the question are you licensed as a nuclear plant operator, you responded no. Is that still correct?

A That is still correct.

Q Have you been trained to operate a nuclear power plant, you answered no. Is that still correct?

A That is correct.

Q Have you been an engineer at a nuclear power plant, you said no, is that correct?

A That is correct.

Q Have you ever implemented procedures at a nuclear plant, and you stated no, is that correct?

A That is still correct.

Q Is that also true with respect to procedures for emergency planning at a nuclear power plant, have you ever done that? Implemented procedures?

A No, I have not been involved in implementing emergency response procedures.

Q Have you ever written procedures for a nuclear plant, you said no?

A That's correct.

Q Have you ever written emergency planning procedures for a nuclear plant?

A No, I have not.

Q Have you ever worked in any capacity at a nuclear power plant, you said no. Is that still correct?

A That is still correct.[32]

Turning specifically to his ability to address the Board's question 1 and provide the overall probability of the postulated scenario, Dr. Thompson again had nothing to offer:

Q Have you ever performed a PRA at a nuclear power plant?

---

32 Thompson Dep. at 10, 23, 63-64. Dr. Thompson has argued that he does not "have to be a qualified expert in a design or operational function" to provide meaningful information in this proceeding. Id. at 46. However, Dr. Thompson's answers to questions reveal his confusion due to lack of familiarity with nuclear plant operations. For example he described a steam generator tube rupture event as one where "flow out of the reactor coolant system *via a LOCA in one of the coolant pump seals*, will carry material from the core *to the point of rupture of the steam generator tubes.*" Id. at 39-40 (emphasis added).

A I have not.

Q Have you ever been on a team that performed a PRA at a nuclear power plant?

A I have not.

Q Have you ever done a peer review of a PRA for a nuclear power plant?

A By a peer review, I take it that you mean the sort of review that would be commissioned by the staff as a team effort involving an in-depth review, and I think the answer to that is no.[33]

Further, Dr. Thompson consistently admitted that he had not identified, nor was even planning to identify, *any* facts to present to the Board supporting BCOC's contention. For example, in exploring how Dr. Thompson would address the postulated inability to restart spent fuel pool cooling or makeup systems due to extreme radiation doses (Step 5 of his postulated scenario), he testified:

A *A definitive answer* to question five [Step 5 of the postulated scenario] is not – *cannot be provided by anyone*. The best that any individual or any group of experts can provide in answer to question five or issue number five, at the top of page 13, is a combination of analysis and judgement. That a – *the best that one could do* in addressing that issue would be to assemble a team of people with varying expertise. . . . And this team would

---

33 Thompson Dep. at 109 (emphasis added); see also id. at 114-119 (where Dr. Thompson demonstrated his lack of familiarity and understanding of even the basic vocabulary of PRA analyses or whether there was any industry standard for such PRAs in the nuclear industry). Dr. Thompson is also of the opinion "that the present state-of-the-art has not expanded substantially beyond NUREG-1150." Id. at 158. Both Dr. Burns and Dr. Parry disagree and have stated that significant improvements to the PRA process have occurred in the decade since NUREG-1150 was considered state-of-the-art. See Burns Aff. ¶ 11; Parry Dep. at 22.

conduct analysis and judgement and would come up with a *statement* about the inability to restart pool cooling.

. . .

Q You don't have a team. How are you going to do this?

. . .

A Now in order to support such a contention, *we do not need to perform the analysis* that I described, because it's - - I readily admit that this is [a] team effort that would require years of work and has never been done. And by definition, *BCOC cannot provide such an analysis* . . . All that is necessary is to show that the use of *a set of reasonable assumptions and supported by some scoping calculations* shows that there is a - - that the probability is characterized in some manner, and I will, in my brief, characterize the probability carefully, in such a manner that a preparation of an EIS is required.

Q But the Board asked us to answer a question. They didn't ask us to tell them how we couldn't answer it. Each party is asked to answer the question. Are you telling me that you are not going to answer this question because you are unable to come up with a best estimate of the overall probability of step five?

A *No party to this proceeding can provide a probability number or even a set of numbers with some uncertainty range in response to question five* that has a scientific quality to it. And whatever is said by any party will not meet the standards of science. It will involve assumptions and judgements. And *my brief will make statements* and may include in step five numerical statements in some bounding sense . . . [34]

Step 5 requires an ability to calculate internal and external doses in the Harris fuel handling building to determine personnel access for providing makeup water to the spent

[34] Thompson Dep. at 51, 56-58 (emphasis added).

000578

fuel pools. Dr. Thompson was unable to explain how he could address this issue other than with "scoping calculations:"[35]

> Q How will you calculate whether or not there will be contamination in the fuel handling building and in what levels, compartments in the fuel handling building?
>
> A *The most that I can provide in this brief will be scoping calculations.* I will not, as indicated previously, be running models to make such an estimation, nor do I believe that any party can provide credible modeling results in this time frame.
>
> Q What is the pressure that would be required to breach the access between the reactor auxiliary building and the fuel handling building?
>
> A In order to answer that question, you have to know all of the entire envelope of interface between the two buildings, and that's a very complicated envelope of interface. You have to follow all the ventilation ducts.
>
> Q Have you done that?
>
> A That's a major task.
>
> Q Do you intend to do it?
>
> A That sort of task is obviously beyond the scope of what I can do in this time frame.
>
> Q How can you make a *scoping calculation* if you don't know whether or not there is any credible scenario even with bypass that will get radioactive contamination into the fuel handling building?

---

35 His lack of any relevant education, training, knowledge or experience in this area may explain, in part, Dr. Thompson's misguided reliance on a 1983 figure issued by the *U.S. Department of Health and Human Services* as the *sole* basis for his conclusion that personnel will be precluded from site access in the event of the postulated scenario. Id. at 160, 182-3.

A Well, I repeat that a - - on this time frame, no party can provide such an analysis that is credible according to the standards that I set forth earlier for a team effort, which I repeat would take years of work and a lot of scientific debate to produce the best available scientific answer to this problem.

Q So what will be the basis for your assumptions if you can do no analysis?

A A *scoping calculation* is one in which you make a variety of simplifying assumptions, which you must state clearly if the scoping analysis is to have any value. And the results are to be regarded as indicative and not definitive. But the context for that is that no party can provide definitive answers.

. . .

Q Are you a health physicist?

A No.

Q Excuse me?

A No.

Q Do you have training in health physics?

A I do not.

Q Education in health physics?

A I do not.

Q Have you ever performed for a nuclear power plant or any other facility a calculation of doses that would occur at any point in a plant as a result of a release of radiation?

A I have not performed such a calculation. I am, however, familiar with the science involved. And I am

> qualified to make - - to perform a *scoping calculation* of that nature.[36]

However, Dr. Thompson's one attempt at performing useful "scoping calculations" strongly supports our position regarding his lack of competence. The single example of such a calculation in this proceeding is contained in his February 1999 report to BCOC, in which Dr. Thompson presents a "scoping analysis" to provide "insight" into the heat transfer pathways in the Harris spent fuel pools.[37] After considering decay heat output, upper bound of temperature rise, heat transfer by conduction, convective cooling by steam, and cooling by thermal radiation, Dr. Thompson calculated that when one-tenth of a spent fuel assembly is submerged, this "yields a T of 9,800 degrees C," where T is "the temperature of steam leaving the top of the fuel assembly."[38] This absurd result is remarkable because it is a steam temperature over one and a half times the temperature of *the surface of the sun.*[39]

---

36 Id. at 66-67, 71-72 (emphasis added). Merely being "familiar with the science involved" leads to uniformed "analyses" such as comparing the frequency of a boiling water reactor in-containment spent fuel pool boiling event with the core damage frequency from the Harris IPE, simply because the probabilities are "at a similar level." Id. at 178-79. Even though he admitted that this comparison "doesn't prove anything," Dr. Thompson still based his conclusion "that pool accidents could be a major contributor to risk at Harris" upon it. Id. at 179.

37 G. Thompson, "Risks and Alternative Options Associated With Spent Fuel Storage at the Shearon Harris Nuclear Power Plant," Appendix D, D-3 (February 1999); Orange County's Request for Admission of Late-Filed Environmental Contentions, Exhibit 3 (January 31, 2000).

38 Id. at D-4.

39 The temperature of the sun is approximately 6,000 degrees C. See Solar and Heliospheric Observatory, Frequently Asked Questions, http://sohowww.nascom.nasa.gov/explore/faq/sun.htm#surface.

Applicant submits that BCOC's burden to demonstrate the need for an adjudicatory hearing is more than asking a few questions about assumptions and providing dubious scoping calculations. BCOC must address the Board's questions with facts and its own defensible calculation of the probability of its postulated scenario. BCOC must also demonstrate that it would have something to contribute to an adjudicatory hearing. BCOC's only expert has confirmed that BCOC has not dedicated the resources to provide a meaningful response to the Board's questions, nor does BCOC's expert have the education, training, knowledge, or experience to address the issues.[40] An adjudicatory hearing is not required to respond to uninformed calculations that suggest that spent fuel temperatures could exceed those on the sun.

F. BCOC Cannot Sustain its Burden to Demonstrate an Adjudicatory Hearing is Required in this Proceeding.

In order to obtain an adjudicatory hearing on its Contention EC-6, Subpart K requires BCOC to place facts into evidence that are material and central to the ultimate decision in this case and that create a genuine and substantial dispute of fact with the evidence presented by Applicant and/or the NRC Staff. Congress explicitly reserved adjudicatory hearings on spent fuel storage expansion to disputes of material facts that *can only be* resolved with sufficient accuracy by the introduction of evidence in an adjudicatory hearing. Applicant submits that BCOC has demonstrated again that it does not possess the technical capability to establish a genuine and substantial dispute of fact.

---

40 Dr. Thompson had not completed any work to address the probability of any of the seven steps of his postulated scenario as of his deposition on October 16,

Footnote continued on next page

00058:

Further, Dr. Thompson is not in a position to make a meaningful contribution to any hearing. The Board is certainly capable of resolving any factual issues in dispute between the Applicant, NRC Staff and BCOC on the written record and oral argument. Of course, the legal questions would never require a hearing. Here, Subpart K presents an insurmountable burden to BCOC.

III. THE NATIONAL ENVIRONMENTAL POLICY ACT DOES NOT REQUIRE PREPARATION OF AN EIS TO ADDRESS THE CONSEQUENCES OF BCOC'S POSTULATED SCENARIO

A. National Environmental Policy Act Requirements Are Well-Established.

National Environmental Policy Act of 1969[41] ("NEPA") prescribes a process by which the federal government considers the environmental impacts of proposed actions. Federal agencies must prepare an Environmental Impact Statement ("EIS") for "major Federal actions *significantly* affecting the quality of the human environment." [42] NEPA forces an agency to take a "hard look" at environmental consequences and ensures that the agency has adequately considered and disclosed the environmental impacts of its actions.[43] It is well settled, however, that "NEPA itself does not mandate particular

---

Footnote continued from previous page

2000, and he had a relatively modest budget for additional work prior to the November 20, 2000, filing. See Thompson Dep. at 26-28, 149.

41 42 U.S.C. §§ 4321 - 4347 (2000).

42 Id. § 4332(2)(C) (emphasis added).

43 Robertson v. Methow Valley Citizens Council, 490 U.S. 346, 350 (1989); Baltimore Gas & Electric Co. v. NRDC, 462 U.S. 87, 97 (1983); Kleppe v. Sierra Club, 427 U.S. 390, 410 n.21 (1976).

results."[44] If "the adverse environmental effects of the proposed action are adequately identified and evaluated, the agency is not constrained by NEPA from deciding that other values outweigh the environmental costs."[45] The fundamental legal question in applying NEPA is, therefore, whether the cognizant federal agency "has adequately considered and disclosed the environmental impact of its actions."[46]

Not every possible environmental impact must be considered or included in an EIS. An "agency must allow all *significant* environmental risks to be factored into the decision whether to undertake a proposed action."[47] NEPA activities are subject to a "rule of reason," requiring consideration only of "reasonably foreseeable" environmental impacts.[48] Only impacts that are "likely," "foreseeable," or "reasonably foreseeable" need be discussed in an EIS.[49] A "reasonably foreseeable" environmental impact is one "sufficiently likely to occur that a person of ordinary prudence would take it into account in reaching a decision."[50] Under NEPA, an EIS need only provide "a reasonably

---

44 Robertson, 490 U.S. at 350; Vermont Yankee Nuclear Power Corp. v. NRDC, 435 U.S. 519, 558 (1978).

45 Robertson, 490 U.S. at 350; Kleppe, 427 U.S. at 410, n.21; Dubois v. U.S. Dept. of Agric., 102 F.3d 1273, 1284 (1st Cir. 1996).

46 Baltimore Gas & Electric Co., 462 U.S. at 98; see also Robertson, 490 U.S. at 350; Kleppe, 427 U.S. at 409-410.

47 Baltimore Gas & Electric Co., 462 U.S. at 100 (emphasis added); see also Vermont Yankee, 435 U.S. at 553; Dubois, 102 F.3d at 1285.

48 San Luis Obispo Mothers for Peace v. NRC, 751 F.2d 1287, 1300-01 (D.C. Cir. 1984), rehearing en banc granted on other grounds, 760 F.2d 1320 (D.C. Cir. 1985), aff'd en banc, 789 F.2d 26, cert. denied 479 U.S. 923 (1986).

49 Sierra Club v. Marsh, 976 F.2d 763, 767 (1st Cir. 1992).

50 Dubois, 102 F.3d at 1286; see also, Sierra Club, 976 F.2d at 767.

thorough discussion of the *significant* aspects of the probable environmental consequences."[51] An EIS is not required where there is no *substantial* question whether federal actions will significantly affect the quality of the human environment.[52] An environmental assessment may be prepared to determine whether an agency action will have a significant environmental effect, requiring an EIS, or whether it will not (in which case no EIS is required).[53]

It has long been established that an agency is not required to blindly evaluate every environmental risk contrived by opponents of an action. NEPA does *not* require consideration of "remote and speculative" impacts.[54] An agency "need not speculate about all conceivable impacts" of a proposed action.[55] "The requirement is not to explore every extreme possibility which might be conjectured."[56] The "rule of reason" governing

---

51 Carmel-By-The-Sea v. DOT, 95 F.3d 892, 899 (9th Cir. 1996) (emphasis added); see also Dubois, 102 F.3d at 1286; Sierra Club, 976 F.2d at 767; Environmental Defense Fund v. Hoffman, 566 F.2d 1060, 1067 (8th Cir. 1977).

52 Idaho Sporting Congress v. Thomas, 137 F.3d 1146, 1149-50 (9th Cir. 1998); 10 C.F.R. § 51.14 (1997) (NRC regulations defining "Finding of No Significant Impact"); 40 C.F.R. § 1508.13 (1997) (Council of Environmental Quality ("CEQ") regulations implementing NEPA).

53 10 C.F.R. § 51.14 (1997) (defining "Environmental Assessment" and "Finding of No Significant Impact"); 40 C.F.R. §§ 1508.9, 1508.13 (1997) (same).

54 Vermont Yankee, 435 U.S. at 551 (quoting NRDC v. Morton, 458 F.2d 827 837-38 (D.C. Cir. 1972)); San Luis Obispo, 751 F.2d at 1300; Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), CLI-90-04, 31 NRC 333, 335 (1990).

55 Dubois, 102 F.3d at 1286.

56 Carolina Environmental Study Group v. US, 510 F.2d 796, 801 (D.C. Cir. 1975).

NEPA interpretation provides that an agency need not consider "remote and speculative risks."[57] As the San Luis Obispo en banc court succinctly stated:

> At some point the probability of an occurrence becomes so infinitesimal that it would be absurd to say that a hearing about it is required. Thus, no one would argue, or so we would assume, that the Commission had to consider the possibility that a space satellite might fall on the [licensee's] plant. . . . It can be shown that the danger posited by [the opposition] here falls into the same range of improbability.[58]

This holding recognizes that to make an EIS "something more than an exercise in frivolous boilerplate" the extent of the required analyses "must be bounded by some notion of feasibility." [59]

Further, an EIS is also not required to include a "worst case analysis" of possible, but substantially uncertain, environmental impacts.[60] Indeed, as the Supreme Court has observed, including only reasonably foreseeable environmental impacts in an EIS promotes the purposes of NEPA by focusing on "those consequences of greatest concern to the public and of greatest relevance to the agency's decision."[61] Considering unlikely

---

57 Yankee Atomic Electric Co. (Yankee Nuclear Power Station), LBP-96-2, 43 NRC 61, 89 (1996) (citing Limerick Ecology Action v. NRC, 869 F.2d 719 (3rd Cir. 1989)).

58 San Luis Obispo Mothers For Peace v. NRC, 789 F.2d 26, 36 (D.C. Cir. 1986) (en banc).

59 Vermont Yankee, 435 U.S. at 551.

60 Robertson, 490 U.S. at 354-56; see also Vermont Yankee, 31 NRC at 334.

61 Id. at 356.

worst-case impacts "distort[s] the decision making process by *overemphasizing highly speculative harms*."[62]

Here we will show that the probability of the seven-step accident scenario postulated by BCOC falls into the same range of improbability as a space satellite falling on the Harris plant. As the Court held in San Luis Obispo, here it would be "absurd" to say a hearing is required. The postulated scenario is "remote and speculative" in the extreme and NEPA does not require consideration of such speculative consequences.

B. The NRC Staff's Decision Not to Prepare an EIS Was Supported by Overwhelming Evidence that the Additional Environmental Impacts of the License Amendment Are Insignificant.

Licensing Boards have consistently — and correctly — accepted NRC Staff determinations that license amendments related to storing spent fuel in fuel pools have no significant environmental impacts and, therefore, do not require an EIS. Here, the NRC Staff's Environmental Assessment ("EA") of the proposed spent fuel pool expansion found that amending the Harris license to allow use of spent fuel pools C and D will have no significant environmental impact.[63] The Staff's EA was in addition to the "hard look" that the Commission has given to this issue through generic rulemaking. As discussed above, NEPA requires nothing more.[64]

---

62 Id. (emphasis added).

63 Carolina Power & Light Company Docket No. 50-400 Shearon Harris Nuclear Power Plant, Unit 1 Environmental Assessment and Finding of No Significant Impact (December 15,1999); 64 Fed. Reg. 71,514 (1999) (hereinafter "EA").

64 See, e.g., Baltimore Gas & Elec., 462 U.S. at 101.

000587

The scope and depth of the Staff's EA was appropriate to the requested action. The Staff considered radioactive waste treatment, gaseous radioactive wastes, solid radioactive wastes, radiological impacts, accidents and alternatives.[65] With regard to accidents, the Staff considered design basis *and* beyond design basis events.[66] In particular, the Staff noted that in "the unlikely event of a total loss of the cooling system, makeup water sources are available to replace coolant lost through evaporation or boiling."[67] The Staff concluded that "the potential for environmental impact from severe accidents is negligible."[68] The Staff took a very "hard look" and appropriately found no significant impact from the proposed action. The facts clearly support the Staff's determination.

Despite this rigorous assessment of potential environmental impacts by the NRC Staff, BCOC insists that the proposed action is being "taken without a proper understanding of the phenomena that could occur."[69] Further, Dr. Thompson charges that

> [T]he staff has been has been *irresponsible* in licensing this and the industry has been irresponsible in doing it and applying for it, and the irresponsibility derives from the fact that neither side of -- neither industry nor the NRC has ever bothered to do a really thorough job of finding out what the implications are.[70]

---

65 EA at 3-9.

66 Id. at 5.

67 Id. at 6.

68 Id.

69 Thompson Dep. at 91.

70 Id. at 92 (emphasis added).

Dr. Thompson does not define what his concept of a 'thorough job' is, other than it "would be a complex enterprise that would take years to do properly."[71] This is not, however, required by NEPA.

Courts affirm an agency's decision not to prepare an EIS (or not to supplement an existing EIS) unless they find the decision was "arbitrary and capricious."[72] In deciding whether an agency decision was arbitrary and capricious, the court considers whether the agency based its decision on "the relevant factors and whether there has been a clear error of judgment."[73]

A court, however, may not substitute its judgment for the agency's, once the agency has considered the relevant factors.[74] Where the issue turns on expert opinion, and the experts disagree, an agency is entitled to "rely on the reasonable opinions of its own qualified experts even if, as an original matter, a court might find contrary views more persuasive."[75] Deference to the NRC's expertise is especially appropriate when, as

---

71 Id. at 56.

72 See, e.g., Marsh v. Oregon Natural Resources Council, 490 U.S. 360, 375-78 (1989) (affirming agency decision not to further supplement an EIS); Kelley v. Selin, 42 F.3d 1501, 1518 (1995) (affirming agency decision, based on an EA, not to prepare an EIS).

73 Marsh, 490 U.S. at 378. Accord Kelley, 42 F.3d at 1518-19.

74 Kelley, 42 F.3d at 1518.

75 Marsh, 490 U.S. at 378. For examples of agency decisions judged arbitrary and capricious, all conspicuously different from the Staff's decision here, see Carmel-By-The-Sea, 95 F.3d at 900 (agency ignored new wetlands, with rare grasses, pointed out to it by other agencies and relied on wetlands surveys that it knew were outdated); Dubois, 102 F.3d at 1292-93 (agency failed to supplement its EIS despite expanding a ski area, primarily outside the area considered in the EIS and outside the area of the existing permit; widening existing trails and

Footnote continued on next page

here, it is "making predictions, within its area of special expertise, at the frontiers of science."[76] To overturn the Staff's determination, and find that additional NEPA analysis may be required for the Harris spent fuel pool expansion, a court would have to find that the Staff has made a "clear error of judgment" in determining that BCOC's postulated scenario is "remote and speculative" *and* is significant enough to warrant consideration pursuant to NEPA *and* is not bounded by the consequences of other severe accident scenarios that have been addressed.

Licensing Board and Appeals Board decisions rejecting contentions that an EIS is required before licenses can be amended regarding storage of spent fuel are clearly correct. In the one case where the Licensing Board admitted a contention claiming that an EIS *was* required because of the possibility of the kind of zircaloy cladding reaction that BCOC relies on, the Appeal Board reversed the Licensing Board.[77] Although the Commission reversed and remanded the case back to the Appeal Board, the issue was

---

Footnote continued from previous page

eliminating buffers between them; developing new ski trails, access roads, and lifts on land previously designated as a woodland, and adding a 28,500 square foot lodge facility not previously considered); Idaho Sporting Congress, 137 F.3d at 1150-51 (agency's environmental assessment of water-quality impacts in one area with riparian buffers as narrow as 25 feet relied on a report for a different area with different characteristics; that report was premised on riparian buffers 100 feet wide was not to be applied to any other area). In contrast, the use of spent fuel pools here is "within the spectrum of alternatives" considered in the FES, and accordingly does not require a further analyses. Dubois, 102 F.3d at 1292-93.

76 Baltimore Gas, 462 U.S. at 103.

77 Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), ALAB-919, 30 NRC 29, 43-52 (1989).

000591

limited to providing an adequate basis for the decision.[78] Most recently, a Millstone Licensing Board rejected similar contentions, based on the same flawed February 1999 report prepared by Dr. Gordon Thompson, claiming that re-racking spent fuel at Millstone would significantly increase the probability of severe accidents and, therefore, required an EIS.[79] The mere postulation of an event, without supporting facts, was not sufficient to sustain a challenge to the Staff's determination that such a postulated event was not required to be considered in an EIS.[80]

C. A Determination That BCOC's Postulated Scenario is Remote and Speculative is Consistent With Qualitative Guidelines, Commission Precedent, and Controlling Legal Authority.

The NRC has not established a quantitative value for determining whether an occurrence is too remote and speculative to be considered in NEPA analyses. Licensing Boards and the Commission have, however, had occasion to review the issue, as discussed further below. The Commission has also developed a policy statement that contains *qualitative* safety goals for the operation of nuclear power plants. In our view, the Commission has explicitly, and properly, avoided establishing a bright line test for "remote and speculative." Taken together, and viewed in the light of the body of applicable federal NEPA case law, however, a frequency of occurrence value emerges – a

---

78 Vermont Yankee, 31 NRC at 336.

79 Northeast Nuclear Energy Co. (Millstone Nuclear Power Station), LBP-00-2, 51 NRC 25 (2000).

80 See also Duke Energy Corp. (Oconee Nuclear Station, Units 1, 2, and 3), CLI-99-11, 49 NRC 328 (1999).

value below which it is reasonable and appropriate to consider an event remote and speculative for NEPA purposes notwithstanding the postulated consequences.

In Vermont Yankee Nuclear Power Corporation (Vermont Yankee Nuclear Power Station),[81] the Commission reviewed an Appeal Board decision that a postulated accident[82] with a probability on the order of 10^{-4} per reactor year was remote and speculative and, therefore, did not require NEPA review. The Vermont Yankee intervenor's sought consideration, in a supplemental EIS, of a postulated accident with potential consequences greater than those previously evaluated by the NRC Staff in its NEPA review.[83] The intervenors had submitted documents implying an estimated upper limit probability of the postulated accident sequence as being on the order of 2.6×10^{-4} per reactor year. The Appeal Board determined that the postulated accident was too remote and speculative to consider. The Commission remanded the case and directed the Appeal Board to develop further "information on the plausibility or probability of" the postulated accident sequence:

> We are reluctant either to endorse or reject a holding that accidents of this probability should be considered remote and speculative, both because such a determination may be unnecessary here and because such a decision could have broader ramifications for the NRC's regulatory program that are better explored outside the scope of a particular case involving only a few parties. Therefore, to the extent that [the Appeal Board's decision] amounts to a holding

81 31 NRC 333 (1990).

82 The accident sequence proposed consisted of a spent fuel pool cladding fire caused by a failure of spent fuel pool cooling, with the cooling failure caused by combustion of hydrogen gas following a reactor accident. Id. at 334.

83 Id. at 334.

> that an accident with a probability on the order of 10^{-4} per reactor year is remote and speculative, we vacate that part of the Appeal Board's decision without prejudice to a later Commission determination on what the limits should be.[84]

The Commission had "difficulty" with relying on unsupported analyses as the bases for the "train of logic of the Appeal Board's decision" that the accident sequence of events was remote and speculative.[85] The Commission instructed the Appeal Board to obtain more fully developed information.

> The Appeal Board bridged the gap between the technical documents and the scenario in the contention by assuming, conservatively, that the probability of that scenario could be no greater than certain scenarios actually analyzed in the technical documents. If the scenarios in the technical documents were remote and speculative, then, a fortiori, the scenario in the contention must be remote and speculative as well. Our opinion makes clear that *future decisions that accident scenarios are remote and speculative must be more specific and more soundly based on the actual probabilities and accident scenarios being analyzed.*[86]

This clarification makes clear that the Commission did not reject the Appeal Board's determination that the accident sequence was remote and speculative because a frequency of 10^{-4} per reactor year was too high. Instead, the Commission remanded the issue because the Commissioners could not determine if 10^{-4} was the actual frequency value. The Commission explicitly reserved to itself, but did not reject, a determination that an accident probability of 10^{-4} per reactor year was remote and speculative.

---

84 Id. at 335.

85 Vermont Yankee Nuclear Power Corp. (Vermont Yankee Nuclear Power Station), CLI-90-07, 32 NRC 129, 131-32 (1990).

86 Id. at 132 (emphasis added) (internal citations omitted).

> If the Appeal Board finds that an accident probability on the order of 10^{-4} per reactor year is appropriate for the entire accident sequence postulated in this contention, the case should be returned to the Commission for further review. Otherwise, the Appeal Board should modify or confirm its judgment as to the remote and speculative nature of the accident on the basis of the accident probability derived on remand.[87]

Further, it is significant to the question before this Board that the Commission authorized the Vermont Yankee Appeal Board to itself determine the remote and speculative question if the probability was below 10^{-4} per year.[88]

Prior to the decision in Vermont Yankee, an Appeal Board had found that a calculated probability of 2.4×10^{-7} per year was sufficiently remote and speculative as to preclude NEPA consideration of the postulated occurrence.[89] In that case, the applicant was required to consider "the chain of events that would have to occur" for a postulated liquid natural gas ("LNG") cloud formed in a collision of a LNG tanker to move over the plant and ignite.[90] Following extensive calculations by the applicant, and detailed reviews by the Licensing Board and the Appeal Board, the Appeal Board found that the

---

87 Vermont Yankee, 31 NRC at 336.

88 The issue was not further resolved as the intervenors withdrew before final resolution of the matter. Id.

89 Pub. Serv. Elec. & Gas Co. (Hope Creek Generating Station, Units 1 and 2), ALAB-518, 9 N.R.C. 14 (1979).

90 Id. at 18.

000594

applicant could "show that this event is so unlikely that its environmental impact need not be considered."[91]

The federal courts have also found occurrences with a probability on the order of 10^{-6} per reactor year remote and speculative. An event with a probability of 3.575×10^{-7} per year is "extraordinarily low" and "so extremely low as to be, for any practical purpose, non-existent."[92] "At some point the probability of an occurrence becomes so infinitesimal that it would be absurd to say that a hearing is required."[93] In San Luis Obispo, the District of Columbia Circuit was asked to determine if the NRC was required to hold a hearing on the potential complicating effects of an earthquake on responses to a simultaneous but independently caused radiological accident at a nuclear power plant.[94] The court was clear that events with this, or a lower, probability were not required to be considered pursuant to the agency's emergency planning regulations:

> If the NRC is required to hold hearings on the emergency plans to deal with contingencies of that level of improbability, we can think of no speculative danger that would not require a hearing. Such a conclusion would serve no purpose other than to enable [opponents] to hold up licensing for many more years.[95]

---

91 Id. at 39 (citing New England Coalition on Nuclear Pollution v. NRC, 582 F.2d 87, 93-94 (1st Cir. 1978)).

92 San Luis Obispo, 789 F.2d at 40.

93 Even the dissent agreed with this conclusion. "I agree that by definition earthquakes greater than the SSE occur too infrequently to warrant consideration, since the SSE is the strongest earthquake that could ever be expected to hit the [plant] site." Id. at 51 (Wald, J., dissenting).

94 Id. at 28.

95 Id. at 40.

The court's logic is consistent with the Commission's discussion in Vermont Yankee and the Appeal Board decision in Pub. Serv. Elec. & Gas. Finally, all of these decisions are consistent with the Supreme Court's admonition in Vermont Yankee:

> Common sense also teaches us that the 'detailed statement of alternatives' cannot be found wanting simply because the agency failed to include every alternative device and thought conceivable by the mind of man. Time and resources are simply too limited to hold that an impact statement fails because the agency failed to ferret out every possible alternative, regardless of how uncommon or unknown that alternative may have been at the time the project was approved.[96]

Further illumination of consideration of a quantified remote and speculative probability value can be gleaned from the Commission's Safety Goal Policy.[97] The Policy contains two *qualitative* safety goals:

> Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.
>
> Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.[98]

---

96 Vermont Yankee, 435 U.S. at 551.

97 51 Fed. Reg. 28,044 (1986).

98 Id.

The Commission's intent with the first safety goal is to "require such a level of safety that individuals living or working near nuclear power plants should be able *to go about their daily lives without special concern* by virtue of their proximity to these plants."[99]

The second safety goal represents a decision that a limit should be placed on the "societal risks posed by nuclear power plant operation."[100] The Commission determined it "impractical to calibrate nuclear safety goals by comparing them with coal risks."[101] However, the Commission established "quantitative health effects objectives" to assure "that nuclear risks are *not a significant addition* to other societal risks."[102] The Commission adopted the following two quantitative health effects for measuring the success of the safety goals:

> The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.
>
> The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.[103]

---

99 Id. at 28,045 (emphasis added).

100 Id.

101 Id.

102 Id. (emphasis added).

103 Id. at 28,046.

The Commission approved the use of the qualitative safety goals and the qualitative health effects objectives "in the regulatory decisionmaking process."[104] In addition, the Commission proposed a general performance guideline for the NRC Staff in implementing the safety goals and health effects objectives:

> Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation.[105]

In 1990, the Commission provided further direction to the Staff on implementation of the Safety Goals in response to SECY-89-102.[106] In particular, the Commission stated that the Safety Goal Policy provides "a definition of 'how safe is safe enough' that should be seen as guidance on how far to go when proposing safety enhancements."[107] The Commission did, however, acknowledge that specifying the large early release frequency ("LERF") as an overall mean value "is inherently more conservative than either of the quantitative health effects objectives" but was "within an order of magnitude of the Commission's health objectives and provides a simple goal which has generally been accepted."[108]

---

104 Id. at 28,047.

105 Id.

106 Memorandum from Samuel J. Chilk to James M. Taylor, "SECY-89-102 – Implementation of the Safety Goals" (June 15, 1990).

107 Id. at 6.

108 Id. at 2.

The Advisory Committee on Reactor Safeguards ("ACRS"), by letter dated August 13, 1996, recommended, inter alia, that the safety goals should be used as a guide for plant-specific actions.

> We believe the safety goals and subsidiary objectives can and should be used to derive guidelines for plant-specific applications. It is, however, impractical to rely exclusively on the Quantitative Health Objectives (QHOs) for routine use on an individual plant basis. Criteria on core damage frequency (CDF) and large, early release frequency (LERF) focus more sharply on safety issues and can provide assurance that the QHOs are met. They should be used in developing detailed guidelines.[109]

The Chairman subsequently requested the Staff to study a number of ACRS recommendations related to the Safety Goals. This work culminated in a March 2000 report to the Commission on the Staff's recommendations for changes to the Safety Goal Policy.[110] The Commission approved the following key Staff recommendations on June 27, 2000:[111]

> Explicitly incorporated the statements that the 'Safety Goals establish a level of safety considered safe enough. They provide guidance on how far to go when proposing safety enhancements.'
>
> Changed the value of LERF to 1×10^{-5} from 1×10^{-6} per reactor year to be consistent with the QHO on early

---

109 Letter from T. S. Kress to Shirley A. Jackson, "Risk-Informed, Performance-Based Regulation and Related Matters" 1 (August 15, 1996).

110 SECY-00-0077, "Modifications to the Reactor Safety Goal Policy Statement" (March 30, 2000).

111 Memorandum from Annette L. Viette-Cook to William D. Travers, "Staff Requirements – SECY-00-0077 – Modifications to the Reactor Safety Goal Policy Statement" (June 27, 2000) (hereinafter, "Staff Requirements Memo on SECY-00-0077")

fatalities, the guidance in Regulatory Guide 1.174, and the Regulatory Analysis Guide for backfits.[112]

The Commission also *disapproved* adding a qualitative statement "that there be no adverse impact on the environment" from nuclear plant operation as a part of the Policy Statement.[113] As the Chairman observed, this statement is inconsistent with the concepts of risk and adequate protection, *since adverse impacts cannot always be completely eliminated.*"[114]

D. A Frequency of Occurrence of One-in-a-Million Per Year Is a Reasonable Quantitative Threshold For Consideration of Remote and Speculative Events

The Commission recognizes that nuclear plant safety cannot be guaranteed and not all adverse environmental impacts from the operation of nuclear power plants can be completely eliminated. The Commission, through the Safety Goal Policy, has provided qualitative guidance on what is "safe enough" and has assigned a quantitative value for the frequency of large radioactivity releases to the environment that satisfies the goal. The ACRS has weighed in on the safety goals, the LERF value, and its application to risk informing spent fuel pool safety decisions. The Commission, the NRC Staff and the ACRS all agree that 1×10^{-5} per reactor year is the appropriate value for the frequency of large early releases of radioactivity and prompt fatalities.

---

112 SECY-00-0077 at 5, 9.

113 Staff Requirements Memorandum on SECY-00-0077.

114 Commission Voting Record – SECY-00-0077, Comments of Chairman Meserve (June 27, 2000) (emphasis added).

It is Applicant's conclusion, therefore, that events with a best estimate probability value of 1×10^{-6} per reactor year or less can and should be considered too remote and speculative to require any consideration pursuant to NEPA. This value is conservatively within the bounds of values considered remote and speculative by Appeal Boards, the Commission, and federal courts. This value is an order of magnitude below the LERF value of 10^{-5}, agreed by the NRC Staff, the Commission, and the ACRS as protective of human and environmental safety from the impacts of nuclear power plant operation, and below which plant design does not need to be changed. It is also well and commonly understood (i.e., "one in a million chance") as unlikely and unnecessary to consider in the normal course of daily life.

From a practical standpoint, 1×10^{-6} provides an order of magnitude "margin" between the LERF, which defines what is "safe enough," and the point at which unlikely events do not have to be considered. Events with a best-estimate probability between 10^{-5} and 10^{-6} can be viewed as deserving a "hard look" to ensure that mitigation (e.g., design change) is not warranted under the circumstances. This marginal area, therefore, provides decision makers flexibility to address case specific concerns while establishing a reasonable limit on the extent of their discretion.[115]

---

115 Applicant recognizes that ACRS comments suggest a "decommissioned spent fuel pool LERF" of 10^{-7} (worst case) because of the potential consequences of multiple cores releasing ruthenium during a spent fuel fire. Ruthenium, however, decays with a half-life of approximately 1 year, so the conditions of concern (i.e., "elevated" risk) are only present for the initial few years following discharge from the reactor. Kunita Aff. ¶ 27. In the Harris case, spent fuel pools C and D will contain greater than five-year old fuel, so only a small, if any, amount of undecayed Ruthenium will remain. Id. ¶ 28.

Risk is the product of probability and consequences. However, a severe accident with release itself produces unacceptable consequences. If a severe accident satisfies the regulatory threshold of unacceptable consequence, then identifying the consequences beyond that which are unacceptable becomes an interesting theoretical exercise, but not one that provides useful information for a decisionmaker. Even if one considered consequences that were 10 times greater than that from a severe accident with a safety goal LERF of 1×10^{-5}, the acceptable probability of occurrence would simply be a factor of ten less or 1×10^{-6}. BCOC's own expert, with his extraordinary concern for the consequences of the postulated scenario, agreed with this proposition.[116]

Dr. Burns, who has participated in a high percentage of all nuclear plant PSA/PRAs, describes in the ERIN Report a "de minimus" point, or the point at which events may be so remote and speculative as to be below what can be rationally considered.[117] He has indicated that, for practical purposes, this point is a frequency of 1×10^{-6} per year. Risk reduction below the "de minimus" point might be accomplished by eliminating a product or service; however, in most cases society has decided that this is not suitable because it interferes with individual freedom and may in fact introduce new or competing risks that may be larger than the risks being "eliminated."[118] The ERIN Report concludes that events with frequencies below one in a million per year (1×10^{-6} per year) can be considered to be sufficiently low in frequency such that

---

116 Thompson Dep. at 191-93.

117 Burns Aff., Attach. C, App. B § B.3.

118 Id.

additional efforts by society to reduce the frequencies below this level are not considered warranted and these risks can be referred to as "remote and speculative."[119]

One in a million appears to be a cutoff for the Commission as well. The Commission approved a frequency of 1×10^{-6} as the cutoff for evaluating low risk accidents associated with the shipment of the Trojan reactor vessel. The Commission found this value was low enough to dismiss without further evaluation. No EIS was required.[120] In evaluating the environmental impacts from Oconee license renewal, the staff reviewed licensee's risk estimate for core damage frequency ("CDF") for internal and external events, which was 8.9×10^{-5} per year, (total external 6.3×10^{-5} and total internal 2.6×10^{-5}). In evaluating cutoff values for event analyses, the licensee used 4.5×10^{-7} for screening internal events and 8.5×10^{-7} per year for external events.[121] The NRC staff accepted these values, which are close to the proposed 1×10^{-6} cutoff, and events with lower probabilities were not included in the EIS.[122]

This conclusion is also consistent with the holding of Limerick Ecology Action v. NRC.[123] In that case, the court determined that consideration of the potential environmental effects of certain severe accidents was required because the Commission was not exempted from NEPA requirements by compliance with the Atomic Energy Act

---

119 Id. § B.4.

120 SECY-98-231, "Authorization of the Trojan Reactor Vessel Package for One-time Shipment for Disposal" (October 2, 1998).

121 NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Power Plants," Supplement 2 § 5.2.3.1 (1999).

122 Id. § 5.2.3.2.

("AEA") and could not exclude "consideration of design alternatives through a generic policy rather than through careful consideration."[124] The court was, therefore, "unwilling to conclude" that the Commission would have precluded consideration of the excluded design alternatives on the basis that the underlying risks were remote and speculative.[125] However, had the Commission properly (i.e., "through careful consideration") concluded that the risks were remote and speculative, the design alternatives at issue could have been excluded from NEPA analyses. In the instant case, and for any explicit Commission endorsement of a quantitative remote and speculative criterion, the Commission would certainly adopt a number arrived at "through careful consideration."

IV. A STATE-OF-THE-TECHNOLOGY PROBABILISTIC ANALYSIS ESTABLISHES THAT THE FREQUENCY OF OCCURRENCE OF BCOC'S POSTULATED SCENARIO AT HARRIS IS SO LOW THAT IT IS HIGHLY REMOTE AND SPECULATIVE

A. Answer to Board Question 1: The Best Estimate Probability of BCOC's Postulated Scenario is on the Order of a Few Chances in One Hundred Million.

In its Order, the Board first asked the parties to address the following issue:

> What is the submitting party's best estimate of the overall probability of the sequence set forth in the chain of seven events in the CP&L and BCOC's filings, set forth on page 13 supra? The estimates should utilize plant-specific data

---

Footnote continued from previous page

123 869 F.2d 719 (3rd Cir. 1989) (hereinafter "LEA").

124 Id. at 741.

125 Id.

000604

> where available and should utilize the best available generic data where generic data is relied upon.[126]

Applicant has determined that the best estimate overall probability of the postulated scenario is less than 3 in one hundred million (2.65×10^{-8}) per year. This value clearly supports the conclusion that the postulated scenario is remote and speculative. The remainder of this section will discuss how this probability was calculated, and the uncertainties, sensitivities, conservatisms, and confidence in the result.

B. The Methodology Employed and Expertise Brought to Bear in Addressing Board Question 1 Was State-of-the-Technology and Relied Heavily on Harris-Specific Information.

Applicant retained ERIN to perform a Harris-specific PSA to assist CP&L in addressing this question. ERIN is an industry leader in risk management and applying reliability and performance-based technologies 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's experience and that of the lead analyst for this project, Dr. Edward Burns, are unsurpassed in the industry. ERIN has developed many of the state-of-the-technology methods used in PSAs and is actively involved in the American Society of Mechanical Engineers ("ASME") Committees which are developing the PSA standard.[127]

---

126 Order at 17.

127 See Burns Aff. ¶¶ 2, 4, and Attach. A, B.

ERIN was tasked by CP&L to determine the best estimate of the overall probability of the postulated scenario occurring at Harris. This analysis was to include not only internal events (i.e., events initiated at Harris such as steam generator tube rupture, loss of coolant accident, station blackout, etc.) as modeled in the updated Harris PSA model, but also sensitivity analyses of the postulated scenario frequency to other possible initiating events, including postulated internal fires and seismic events. ERIN was also to consider the sensitivity of the results to core damage events during shutdown conditions.[128]

The updated Harris PSA is a probabilistic safety assessment model that was originally developed for the Harris IPE pursuant to NRC Generic Letter 88-20. CP&L maintains the updated Harris PSA in a quality manner under procedural controls.[129] The updated Harris PSA includes: (1) event trees that model core damage accident sequences and containment response following a core damage event; (2) fault trees that represent plant systems and failure modes; (3) initiating event, component failure, and human reliability data; and (4) special analyses, such as internal flooding and Interfacing Systems Loss of Coolant Accident ("ISLOCA"). The updated Harris PSA considers internal initiating events (except internal fires) and applies when the reactor is critical. The results of the updated Harris PSA include an estimated annualized CDF for initiating

128 Id. ¶ 5, Attach. C §§ 1.0 and 2.0. The total effort by ERIN personnel dedicated to analyzing the postulated scenario exceeded 2,000 hours of professional time during the period from August to November, 2000. A significant additional, but more difficult to quantify, effort was expended by CP&L personnel supporting the ERIN effort. Id. ¶ 7.

events. The analysis was performed pursuant to Generic Letter 88-20, Supplement 4. The 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).[130]

As part of the evaluation to respond to the Board's question, ERIN was asked to perform an independent peer review of the existing updated Harris Level 1 and Level 2 PSA for internal events. 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" in addressing the postulated scenario.[131]

The analytical methodologies chosen by ERIN to determine the best estimate overall probability of the postulated scenario are characteristic of past nuclear power plant PSAs (also referred to in the literature as probabilistic risk assessments ("PRAs")) and incorporate state-of-the-technology methods.[132] To the extent possible, site specific analyses and information from the updated Harris PSA and IPEEE were used for the

---

Footnote continued from previous page

129 Laur Aff. ¶ 7.

130 Laur Aff. ¶ 5. The pedigrees of the Harris PSA and Harris IPEEE are discussed in the Laur Affidavit ¶¶ 4-8 and Burns Affidavit ¶ 13 and Attach. C § 3.0.

131 Burns Aff. ¶ 13. The independent peer review report is found in Attachment B to the Laur Affidavit. ERIN personnel developed the peer review programs for the vendor owners' groups and have participated in essentially all of the PSA peer reviews completed or scheduled to date in U.S. nuclear power plants. Id. ¶ 4. Regarding the updated Harris PSA, the ERIN reviewers concluded: "On balance this PSA is viewed as one of the best-documented PSAs that the reviewers have seen." Laur Aff., Attach. B § 3.

132 PSA methodology has significantly evolved over the past ten years in the nuclear industry, building on the methods, data, and approaches used in the NRC's

Footnote continued on next page

probabilistic analysis performed by ERIN. The documents were only a starting point because they do not address loss of spent fuel pool cooling nor a self-sustaining exothermic oxidation reaction of zircaloy spent fuel cladding in the spent fuel pool, which are part of the postulated scenario.[133] The analysis required the incorporation of the unique Harris design features, including the size and location of the Harris fuel handling building and the multitude of spent fuel pool makeup systems and makeup pathways.[134] Where site specific information was not available, the best available generic studies were used as appropriate.

CP&L staff provided detailed calculations (including the Harris PSA), system descriptions, interviews with operating personnel, and procedure interpretations.[135] CP&L technical and operations personnel expended a great deal of effort researching and analyzing Harris-specific information in support of ERIN. In particular, Steven Edwards managed the efforts of a team of Harris engineers in performing the calculations establishing the time to heat up the Harris spent fuel pools to boiling, and after boiling

---

Footnote continued from previous page

mandated IPE process. The current PSA methods and technology are significantly improved beyond those used in the IPE process. Burns Aff. ¶ 11.

133 Id. ¶ 13. The methodology employed by ERIN is discussed in detail in Burns Aff., Attach. C § 2.0.

134 The Harris fuel handling building , spent fuel pools, and associated equipment are described in Edwards Affidavit ¶¶ 12 – 14 and McCartney Affidavit ¶¶ 4 – 21. The multitude of pathways for makeup water to the spent fuel pools is described in detail in McCartney Affidavit ¶¶ 22 – 34.

135 Burns Aff. ¶ 8; information provided by CP&L to ERIN is summarized in the Laur Affidavit; Edwards Affidavit; McCartney Affidavit; DeVoe Affidavit; and Morgan Affidavit.

000608

has started, the additional time necessary to boil the pool level to the top of the fuel racks.[136] Mr. Edwards also provided ERIN with the spent fuel pool gate alignments expected during operation with pools C and D in operation.[137]

CP&L staff provided ERIN information on the multitude of methods to provide cooling and makeup water to the Harris spent fuel pools under normal and emergency conditions.[138] Eric McCartney, an experienced senior reactor operator, also provided ERIN with Harris-specific information on the configuration and operation of doors, locks, emergency lighting, and protective equipment available to operators.[139] In addition, he established the best estimate times for operators to access and align the alternate spent fuel pool cooling and makeup flow paths.[140]

ERIN personnel made multiple Harris site visits to confirm the as-built design of certain key Harris buildings, systems and components.[141] CP&L personnel performed an owner's review of the draft probabilistic analysis to ensure accuracy of the Harris site specific information.[142]

In the following sections, we discuss in more detail the methodology for the calculation of the overall probability and the results.

---

136 Edwards Aff. ¶¶ 3, 15-18, 22.

137 Id. ¶ 13.

138 McCartney Aff. ¶¶ 25-34.

139 Id. ¶¶ 17-21.

140 Id. ¶¶ 26-34.

141 Burns Aff. ¶ 8.

142 Laur Aff. ¶ 9.

C. The Probability of Initiating Events – A Severe Reactor Degraded Core Accident with Containment Bypass, Loss of Spent Fuel Pool Cooling and a Large Early Release of Fission Products Outside of Containment – Is Extraordinarily Low and Beyond the Harris Design Basis.

The postulated scenario 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). ERIN evaluated these two steps using probabilistic safety assessment techniques. For the internal events (i.e., initiating events 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 updated 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).[143] ERIN also performed a sensitivity analysis 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 and seismic events, as well as screening other external events. Generic industry data developed by the NRC was used to evaluate the shutdown events.[144]

---

143 Burns Aff. ¶ 15. The independent peer reviewers had found the ISLOCA analysis in the Harris PSA overly conservative and it was updated to be useful in providing the best estimate calculation of the postulated scenario. Id. ¶ 13.

144 Id. ¶ 15. The accident sequence frequency development for each of the contributors are described in Burns Affidavit, Attach. C § 4.0.

Step 3 of the postulated scenario requires the loss of spent fuel cooling and makeup systems to the Harris spent fuel pools. ERIN performed a probabilistic evaluation of the loss of all spent fuel pool cooling and makeup systems, which included SFPCCS cooling failures (random, human error, test/maintenance and common cause); SFPCCS cooling support system failures, including support system failures that may have contributed to the core damage accident sequence initiating event; and consequential failures of SFPCCS cooling or its support systems due to adverse environmental conditions caused by containment failure or bypass.[145] The addition of a separate, redundant SFPCCS for spent fuel pools C and D provides alternate makeup paths in the event the SFPCCS cannot be restarted. One of the conclusions reached by ERIN was that overall probability of the first six steps of the postulated scenario is somewhat less with the addition of the SFPCCS for spent fuel pools C and D providing a redundant cooling system and alternate makeup water pathways for the spent fuel pools.[146]

Step 4 of the postulated scenario 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,

---

145 Burns Aff. ¶ 16.

146 Id. See McCartney Aff. ¶¶ 24 - 29 for a discussion of the additional makeup water pathways created by the addition of the redundant SFPCCS for spent fuel pools C and D.

CP&L calculated radiation levels for the specific areas in which access would be necessary in order to respond to Step 3.[147]

To determine these levels, Michael DeVoe, a CP&L engineer with over 21 years of design and safety analyses experience, calculated a best estimate reactor core radioisotope inventory to support the ERIN analyses.[148] Mr. DeVoe provided his results to Ben Morgan of CP&L.[149] Mr. Morgan combined the best estimate core inventory and the fractions of the core inventory released in each event obtained from ERIN to calculate the expected dose rates in the specified areas.[150] Using these dose rates, Mr. Morgan determined access restrictions applicable to each analyzed event, which were provided to ERIN.[151]

ERIN probabilistically considered and modeled 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. ERIN made an extensive effort to characterize plant conditions, especially in the reactor auxiliary building and the fuel handling building (i.e., the areas containing critical equipment). ERIN performed a deterministic evaluation of the plant

---

147 Burns Aff. ¶ 17; see also Morgan Aff. ¶¶ 8, 15, 17, 18, 19. The probabilistic evaluation of the loss of all SFPCCS and makeup systems for the spent fuel pools is discussed in Burns Affidavit. ¶ 16, Attach. C § 4.0, and Apps. A, C, D and E.

148 DeVoe Aff. ¶ 6.

149 Id. Aff. ¶ 12.

150 Id. ¶ 5. A more complete description of the dose calculations, including the widely accepted standards and methods used, can be found in Morgan Affidavit ¶¶ 6-16.

thermal hydraulic response and the transport of radionuclides to characterize issues such as access, timing, and adverse conditions on equipment.[152]

ERIN utilized the Modular Accident Analysis Program ("MAAP") computer model to analyze 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 NRC 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 reactor auxiliary building and fuel handling building. These results provided one input to the CP&L dose calculations used to assess personnel access to specific areas and to ERIN's assessment of equipment survivability.[153]

The annual frequency contributions of each of the internal events is summarized in Table 5-1 of the ERIN Report (which is reprinted in the Burns Affidavit at 14). The total internal events contribution is calculated to be 7.67×10^{-6}. The sensitivity analyses for the annual frequency contribution from fire induced events is calculated at 9.80×10^{-7}

Footnote continued from previous page

151 Id. ¶¶ 17-19.

152 Burns Aff. ¶ 17.

153 Id. ¶ 17, Attach. C § 4.0 and App. A, C, E; Morgan Aff. ¶ 5.

and the estimate of shutdown events is 5.00×10^{-7}.[154] The annual frequencies for steps 1 through 5 of the postulated scenario are exceedingly low, already lower than the safety goal for LERF and, of course, beyond the Harris design basis. The requirement of yet another improbable event by the postulated scenario in step 6 only highlights the extremely remote probability of occurrence of this event at Harris.[155]

D. The Probability of Recovery of Spent Fuel Pool Cooling at Harris Before Evaporation Uncovers the Spent Fuel, After the Highly Unlikely Initiating Events Required By BCOC's Postulated Scenario, is Quite High Due to the Unique and Robust Design of the Harris Fuel Handling Building and the Multiple Alternate Sources of Makeup Water.

Step 6 of the postulated scenario requires the loss of most or all spent fuel pool water through evaporation and the inability to restart spent fuel cooling or add makeup water to the spent fuel pool before the spent fuel is uncovered. To evaluate this step, ERIN performed a deterministic evaluation that included a calculation by CP&L of the time to boil and evaporate the water in the spent fuel pool after loss of all cooling.[156] With a worst case heat load in spent fuel pools A and B (i.e., immediately after

---

154 The sensitivity analysis for seismic contribution was not broken down in the same manner as for the internal events, fire and shutdown internal events. The sensitivity analyses will be discussed in greater detail infra § IV.D.

155 As Dr. Parry of the NRC Staff stated in his deposition, "The first step in this scenario is a degraded core accident. The second is a containment failure. The probabilities are so - - or the frequency of those events has been assessed, and it is judged that the frequency is sufficient to meet the agency's safety goals, which, and I think in terms of those safety goals, if you look at [NUREG-]1150, they certainly demonstrate a degree of prudence associated with the frequencies of such accidents." Parry Dep. at 83.

156 The results of that calculation are set forth in Section 2.0 of the ERIN Report, Burns Affidavit, Attach. C § 2.0 and Edwards Affidavit ¶¶ 15 - 25.

refueling), CP&L calculated that it would take over eight (8) days after all SFPCCS cooling and makeup is lost to uncover the fuel. (It would take over 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 Application.)[157] Based on the ability to restore spent fuel pool level and prevent uncovering of any spent fuel with the most limiting makeup sources credited, ERIN conservatively assumed access to critical plant areas to restore SFPCCS cooling or makeup to the spent fuel pools to be required within 96 hours.[158]

The size and compartmentalization of the Harris fuel handling building 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 SFPCCS for spent fuel pools C and D provides additional pathways for injection of makeup water to the spent fuel pools. 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.[159]

The results of ERIN's probabilistic analysis are described in Section 5.0 of the ERIN Report and are summarized in Table 5-1.[160] 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 (discussed in the previous

---

157 Edwards Aff. ¶ 22.

158 Burns Aff. ¶ 18.

159 Id. ¶ 18, Attach. C, App. E. The various makeup water pathways are described in Burns Affidavit, Attach. C, App. A and McCartney Affidavit at ¶¶ 25-34.

160 Burns Aff. ¶ 21 and Attach. C § 5.0.

section). The second column provides the results of the probabilistic assessment of Steps 1 through 6 for each severe accident initiator, taking into account the probability that Harris personnel could restore spent fuel pool makeup within 96 hours. The cumulative results of the internal events initiated sequences indicate that the loss of effective spent fuel pool cooling has a best estimate annual occurrence probability of 2.65×10^{-8}. This value is the best estimate answer to Question 1.[161]

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. CP&L and ERIN recognized that the uncertainties associated with these events are greater 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 2.94×10^{-9} (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 scenario 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 scenario (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

[161] Id. ¶ 21.

of pool integrity). While the point estimate annualized probability contribution due to seismic initiated events of 8.65×10^{-8} is higher than for internal events, it was judged by ERIN not to alter the conclusions reached based on the internal events analysis.[162]

Finally, the CDF associated with internal events during shutdown refueling outages was estimated by ERIN to be on the same order of magnitude as that calculated for power operation. This determination was based on generic studies rather than Harris-specific PSA, because shutdown internal events are not included in the Harris PSA. In any case, the generic results for pressurized water reactors were judged by ERIN to be 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 postulated scenario that is consistent with the estimate of the probability reached for the dominant internal events.[163]

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 decisionmaking 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

---

162 Id. Dr. Thompson agreed that seismic structural failure was not a contributor to the postulated scenario. Thompson Dep. at 127. In any event, in San Luis Obispo the court rejected consideration of the effects on a nuclear plant of earthquakes greater than the design basis safe shutdown earthquake – and that was in *California*. See note 92-93, supra.

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 scenario can be further understood in the context of the uncertainties surrounding the quantification.[164]

There are uncertainties surrounding any calculated probability. The NRC, its contractors, and the industry have made substantial efforts to understanding the 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, ERIN developed extensive sensitivity cases in connection with its analysis that portray the changes in the postulated scenario frequency if input variations occur. The results of the sensitivity analyses provide greater confidence in the validity of the best estimate results.[165]

Footnote continued from previous page

163 Burns Aff. ¶ 22.

164 Id.

165 Id. ¶ 24 and Attach. C § 5.0.

E. The Probability of a Self-Sustaining Exothermic Oxidation Reaction of Zircaloy Cladding of the Old, Cold Spent Fuel to be Stored in Harris Spent Fuel Pools C and D is Highly Unlikely in Any Event.

Step 7, initiation of a self-sustaining exothermic oxidation reaction in spent fuel pools C and D, was not evaluated by ERIN. A rigorous probabilistic assessment would have required the development of new thermal hydraulic models. There was insufficient time to undertake such development work. Furthermore, the probability of reaching Step 7 was calculated to be exceedingly low, as noted in the preceding section. 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 purposes of the best estimate analysis of the probability of the postulated scenario.[166]

This is considered to be a very conservative assumption. Actual spent fuel has been heated up in air to a temperature of approximately 800° C under controlled laboratory conditions without a self-sustaining exothermic oxidation reaction of zircaloy spent fuel cladding occurring.[167] Anecdotal evidence also exists that shows a self-sustaining exothermic oxidation reaction of zircaloy spent fuel cladding does not occur for air cooled spent nuclear fuel.[168] Between late 1977 and early 1981, CP&L shipped 290 PWR fuel assemblies from Robinson to Brunswick in over 40 shipments using air coolant in the shipping cask. At the time of shipment, this spent fuel had cooled between

---

166 Id. ¶ 20.

167 See Kunita Aff., Attach. B, reference 7.

168 Id. ¶ 26.

2.7 and 6.5 years. There is no evidence that there was anything unusual about these assemblies when they were unloaded after receipt at Brunswick.[169]

CP&L's Principal Engineer for Spent Fuel Management, Robert Kunita, undertook a review of the literature relating to the oxidation of zirconium and the potential for a self-sustaining exothermic oxidation reaction in the zircaloy cladding of the spent fuel to be stored in Harris spent fuel pools C and D in the event of evaporation of the pool water and uncovery of the spent fuel.[170] Mr. Kunita is an expert in the design, materials, performance, decay heat rate, storage and transportation of spent nuclear fuel. Mr. Kunita has been professionally responsible for matters involving nuclear fuel since 1966, when he joined the nuclear core design team for Admiral Hyman Rickover's Light Water Breeder Reactor Project, which subsequently ran successfully at the Shippingport Reactor. Mr. Kunita has been employed by CP&L for 27 years.[171]

Mr. Kunita determined that the literature contains a limit (3 kilowatts per metric ton) for use in determining whether a self-sustaining exothermic oxidation reaction is likely for spent fuel with a particular decay heat rate.[172] For spent fuel with heat outputs less than 3 kilowatts per metric ton, no self-sustaining zircaloy exothermic oxidation reaction will occur even if cooling is lost because the available energy is insufficient to initiate and sustain the reaction. For spent fuel with a heat output above 3 kilowatts per

---

169 Id.

170 Id. ¶¶ 14-16.

171 Id., Attach. A.

172 Id. ¶ 19.

00062

metric ton, it is unclear whether an uncontrolled exothermic oxidation reaction will occur, because the limit is very conservative.[173]

Mr. Kunita determined that spent fuel planned for storage in Harris spent fuel pools C and D has too low a decay heat rate to raise the zircaloy cladding to the critical cladding oxidation temperature and is, therefore, highly unlikely to undergo a self-sustaining exothermic oxidation reaction.[174] The primary contribution to heat generation rate in spent fuel is the radioactive decay of material in the fuel, referred to as decay heat. Decay heat is primarily a function of the combination of the burnup of the spent fuel, in megawatt-days per metric ton of uranium (MwD/Mtu), and the age (or "decay time") of the fuel. The decay heat rate drops rapidly with time after the spent fuel is discharged from the reactor and after approximately five years the decay heat is only a small fraction of when the spent fuel was first removed from the reactor.

Mr. Kunita concluded that because of the low heat load in the old, cold spent fuel to be stored in Harris spent fuel pools C and D, it is highly unlikely that the spent fuel in pools C and D could sustain a zircaloy cladding exothermic oxidation reaction, even if a loss of most or all pool water through evaporation occurred.[175] Thus, while for purposes

---

173 Id. ¶ 34.

174 Id. ¶ 35.

175 Id. Interestingly, BCOC's expert conceded in his deposition that this result might be the case:

> Q Look at the seven-step scenario again on page 13. Is it possible that the best estimate of a probability of that scenario is zero? That is, one of the steps itself might be zero.

Footnote continued on next page

of the probabilistic assessment of the best-estimate annual frequency of the postulated scenario Applicant assumed a conditional probability of 1.0 for a self-sustaining exothermic oxidation reaction of zircaloy spent fuel cladding, the more realistic probability might well be much closer to 0.0. As discussed in the next section, this is one of a number of conservatisms in the analysis.

F. While Applicant Attempted to Provide a "Best Estimate" Probability, the Resulting Analysis Still Contains Conservatisms That Tend to Overstate the Probability of BCOC's Postulated Scenario.

As requested by the Board, the analysis performed was a best estimate analysis using the best available technical information representative of Harris. Despite all prudent attempts to create a best estimate evaluation, there remain some potential residual conservatisms in the quantification. In addition to the highly unlikely assumption that the conditional probability of a self-sustaining exothermic oxidation reaction in zircaloy spent fuel cladding is 1.0 (as discussed supra), among these conservatisms are:

- A substantial fraction of the containment does not interface with the reactor auxiliary building. However, the dominant failure modes for containment appear to be at locations where reactor auxiliary building impacts cannot be ruled out. Therefore, all containment failures are assumed to impact the reactor auxiliary building environment. (This overstates the probability of a harsh or radioactive environment in the

---

Footnote continued from previous page

> A That's conceivable, *yes*. If the *fuel* were of an *age* or a spacing such that when drained, ignition would not occur, then the *probability* of the scenario *would be zero*. In fact, that pertained in the early years of nuclear plant operation when low density open racks were used.

Thompson Dep. at 152-3 (emphasis added).

reactor auxiliary building and fuel handling building which could preclude personnel access to restore makeup water to the spent fuel pools.)[176]

- The spent fuel pool boil off time is taken to be the minimum it can be (8 days), given the plant configuration and the times at which freshly discharged spent fuel could be introduced into spent fuel pools A or B. Only half of that time is allowed for recovery of makeup water to the spent fuel pools.[177]
- 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.[178]
- A conservative approach was taken by assuming that components fail if the room temperature exceeds the manufacturer's 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.[179]
- 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 component cooling water ("CCW") and emergency service water ("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.[180]
- The treatment of containment isolation failures into the reactor auxiliary building in the base model assumes that access to the reactor auxiliary building and fuel handling building operating deck (286' Elevation) is not

---

176 Burns Aff. ¶ 24.

177 Id.

178 Id.

179 Id.

180 Id.

available. This is conservative relative to the deterministic calculations performed to support accessibility. The deterministic calculations indicate that the fuel handling building is not affected by the containment isolation failure.[181]

- Several conservative assumptions are incorporated in the heatup calculations, including: (a) water volume in the cask unloading pool was not considered; (b) no credit is taken for heat transfer to the pool liners, concrete structure, or atmosphere; (c) no credit is taken for any makeup water addition after the initiation of the heatup.[182]

The dose calculations also contain a number of conservatisms.

- The derivation of the in-plant airborne dose factors using MicroShield, modeled each plant area as a rectangular box and calculated the dose rate at the highest dose point, (i.e., the geometric center of the box). This method produced conservative results because it does not account for support structures, installed equipment, and internal walls that shield an individual from some portion of the calculated airborne activity. Also, in some areas, the geometric center of the volume is above head height, so that the actual dose rate to an individual would be lower than calculated. Actual dose rates would also be lower than calculated in plant areas with lower ceilings in part of a space, because an individual would be exposed to less activity from overhead than calculated.[183]

- Access times in areas affected by environmental releases assume that all areas are downwind, (i.e., all entrances to the power block, the water treatment building, and the cooling tower basin are affected by the same release). This is extremely unlikely to occur because of the actual physical separation of these areas and the diverse directions from the release points. As a result, the calculated dose rates in one or more of these locations is very likely conservatively high.[184]

- Deposition was not assumed to remove any activity from the plume and the activity was not decayed during the time it would take the activity to travel from the release point to the location of interest. These assumptions both increase the conservatism of the calculated plume dose rates. The

---

181 Id.

182 Edwards Aff. ¶ 20.

183 Morgan Aff. ¶ 21.

184 Id. ¶ 23.

lack of decay during travel time also adds conservatism to the calculated deposition dose rates.[185]

- The dose calculations assumed radioactivity releases from a single point. This resulted in higher calculated doses than would result if the release occurred from multiple locations, as would likely be the case for all scenarios except Steam Generator Tube Rupture ("SGTR").[186]
- Conservative values were used for steam flow rate and ambient temperature in calculating the effective release height for SGTR. This resulted in a lower calculated release height and, therefore, higher calculated dose rates from both shine and plume immersion.[187]
- Dose consequences for personnel on the ground from radioactivity released from the plant vent are not as significant as the dose consequences from radioactivity released through the fuel handing building railroad door. Use of a ground level release for scenarios other than SGTR, therefore, produces dose values more conservative than actually expected because, although some activity could be released from the fuel handling building railroad door, most of the activity released would be from the plant vent.[188]

These conservatisms still inherent in the probabilistic assessment provide additional confidence that the calculated best estimate overall annualized probability of occurrence at Harris for the postulated scenario is 2.65×10^{-8} or less. In addition, Dr. Burns states that his confidence in the results are based on: (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

---

185 Id. ¶ 24.

186 Id. ¶ 25.

187 Id. ¶ 26.

188 Id. ¶ 27.

similar analyses; and (7) the extensive set of sensitivity studies used to explore the uncertainty bands associated with the quantification. For all these reasons, it is Dr. Burns and ERIN's professional opinion, and Applicant's position, that the postulated scenario is so unlikely that it would not be reasonable to consider it further in decision-making for NEPA regarding postulated risks posed by the Harris spent fuel pools. The annual occurrence probability of the postulated scenario 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.[189]

V. THE NUREG-1353 ESTIMATED VALUES ARE NOT RELEVANT TO DETERMINING THE FREQUENCY OF OCCURRENCE OF THE POSTULATED SCENARIO AT THE HARRIS PLANT

A. Answer to Board Question 2: The Probability Value of 2×10^{-6} Per Year Set Forth in the Executive Summary of NUREG-1353 is Not Relevant to BCOC's Postulated Scenario; In Any Event, the Assumed Conditional Probability for a Self-Sustaining Exothermic Reaction Cannot be Higher.

The Board asked the parties to address the following second point:

> The parties should take careful note of any recent developments in the estimation of the probabilities of the individual events in the sequence at issue. In particular, have new data or models suggested any modification of the estimate of 2×10^{-6} per year set forth in the executive summary of NUREG-1353, Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools (1989)? Further, do any of the concerns expressed in the ACRS's April 13, 2000 letter suggest that the probabilities of individual elements of the sequence are greater than those previously analyzed (e.g., is the chance of occurrence of sequence element seven, an

189 Burns Aff. ¶ 25, Attach. C § 6.0 and App. B.

exothermic reaction, greater than assumed in the decade-old NUREG-1353)?[190]

Based upon the assumptions and methodologies used in NUREG-1353,[191] and an extensive review of available literature, CP&L has concluded that the probability values estimated in NUREG-1353 are not applicable to the postulated scenario. To the extent that the NUREG-1353 probability value for a self-sustaining exothermic oxidation reaction of zircaloy spent fuel cladding applies to postulated scenario step 7, the NUREG-1353 conditional probability for PWR spent fuel elements is 1.0, which obviously cannot be increased. CP&L also concludes that recent literature does not contain sufficient information to evaluate the conditional probability value for BWR spent fuel elements of 0.25 specified in NUREG-1353. As described above, however, CP&L has used a conservative conditional probability value of 1.0 for step 7 in its analyses responding to the Board's questions in this proceeding.

B. A Literature Review Does Not Suggest Changes to NUREG-1353 Values to the Extent They Are Relevant Here.

To address the Board's points, CP&L directed Robert Kunita to conduct a literature review to identify any developments since 1989 (i.e., the publication date of NUREG-1353,) in the estimation of the probabilities of the individual events in the postulated scenario. Mr. Kunita reviewed an extensive list of documents, identified in Exhibit 2, Attachment D, to evaluate their impact on the estimates contained in NUREG-

190 Order at 17.

191 NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, 'Beyond Design Basis Accidents in Spent Fuel Pools'" (1989) (hereinafter "NUREG-1353").

1353. Specifically, the review was to identify any new models or data that could suggest a modification of the 2×10^{-6} per year value for the overall probability of a self-sustaining exothermic oxidation reaction of zircaloy spent fuel cladding resulting from a loss of water from a spent fuel pool and whether the probabilities of the individual elements of the postulated scenario could be greater than previously analyzed.[192]

To the extent that any NUREG-1353 estimated value is applicable to the postulated scenario, Mr. Kunita concluded that the data and models that have been reported since the publication of NUREG-1353 do not suggest any substantive modification of those values.[193] Mr. Kunita is also of the opinion, however, that, with the possible exception of the probability of a loss of spent fuel pool cooling, the estimated values in NUREG-1353 do not appear applicable to the postulated scenario.[194] The NUREG-1353 quantification of accident sequences in spent fuel pools includes structural failures due to: missiles, aircraft crashes, heavy load drops, and beyond design basis earthquakes, reactor cavity and transfer gate pneumatic seal failures, and inadvertent draining.[195] The postulated scenario specifically excludes these initiators (i.e., the postulated scenario only includes initiators that result in the loss of pool water due to evaporation). The frequency of spent fuel damage values resulting from the accident

---

192 Kunita Aff. ¶ 29.

193 Id. ¶ 30.

194 Id.

195 NUREG-1353, at 4-13 to 4-28.

sequences postulated in NUREG-1353, therefore, reflect conditions that are not applicable to the Board's questions.

Dr. Burns also reviewed NUREG-1353 in the process of preparing the ERIN report. Dr. Burns noted that while the NUREG-1353 best estimate value of 6.0×10^{-8} per reactor year for loss of spent fuel cooling and makeup due to seismic events is not inconsistent with the ERIN results, the value contains an unspecified beyond design basis contribution, which limits its usefulness.[196] Dr. Burns arrived at the same conclusion as Mr. Kunita: the mean value of 2×10^{-6} per reactor year estimated value in NUREG-1353 is not relevant to analyzing the postulated sequence.[197]

BCOC's expert apparently reached the same conclusion. Dr. Thompson stated in his deposition:

> Q Look on page 17 of Exhibit 2, the second question, for a moment. It says[, t]he parties should take careful note of any recent developments in the estimation of the probabilities of the individual events and the sequence at issue. In particular, have new data or models suggested any modification of the estimate of two-times-ten-to-the-minus-six per year, set forth in the Executive Summary of NUREG-1353, regulation analysis for the resolution of Generic Issue 82, beyond design basis accidents in spent fuel pools, 1989.
>
> What's your answer to that question?
>
> A In my brief, I will certainly respond in every particular to what the Board requests. My recollection at the moment of NUREG-1353 is that it did not address the scenario that's at issue here.

---

196 Burns Aff. ¶ 12.

197 Id.

. . .

Q . . . If you look at table 4.7.1, let's look under structural failures. Is it fair to say that missiles, aircraft crashes and heavy load drops are outside the scope of the seven-step sequence that we are about?

A By Board ruling, yes.

Q And, also, by Board ruling, what about pneumatic seal failures?

A Likewise.

Q Inadvertent drainage.

A Likewise.

Q How about loss of cooling makeup?

A As shown by the footnote, that includes seismically induced loss of cooling and makeup. My recollection of this document is that the initiating events for loss of cooling and makeup do not include a degraded core reactor accident.

Q That's your understanding.

A That's my recollection of this document, yes.

Q Okay. Seismic structural failure would not be included either under the Board's scenario, is that correct?

A That's correct, yes.[198]

There appears to be agreement that NUREG-1353 probability values are not applicable to determining the best estimate probability of the postulated scenario.

198 Thompson Dep. at 124-5, 126-7.

C. The Concerns Expressed in the April 13, 2000 ACRS Letter Do Not Suggest That the Probabilities of Individual Elements of the Postulated Scenario Are Greater Than Previously Analyzed.

The ACRS has speculated that the presence of zirconium hydrides in spent fuel cladding may lower the critical cladding oxidation temperature.[199] Mr. Kunita, however, did not identify any analysis that indicated zirconium hydrides would lower the onset temperature of a self-sustaining exothermic oxidation reaction of zircaloy spent fuel cladding below 800°C. Without such information or analysis, Mr. Kunita's heat balance calculations provide the most accurate analyses of the potential for a self-sustaining exothermic oxidation reaction of zircaloy spent fuel cladding in the Harris spent fuel pools C and D.

In any event, the conditional probability of an exothermic oxidation reaction cannot be greater than that assumed in NUREG-1353 for PWR spent fuel, as NUREG-1353 assumes a conditional probability of 1.0 for this event.[200] Further, as discussed in detail in section IV.E supra, Applicant's literature survey did not identify any analysis that reported a critical cladding oxidation temperature any lower than 800°C. The literature survey did identify several studies that report the critical cladding oxidation temperature for a self-sustaining exothermic oxidation reaction of zircaloy spent fuel cladding at about 900°C, with NUREG/CR-5597 showing the onset of rapid zircaloy

---

199 Letter from Dana A. Powers to Richard A. Meserve, "Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," (April 13, 2000).

200 Kunita Aff. ¶ 31.

oxidation at 1500°K (1227°C).[201] Based on this data, and the specific parameters of the spent fuel to be stored, Mr. Kunita concluded that a self-sustaining exothermic oxidation reaction of zircaloy spent fuel cladding is highly unlikely in Harris spent fuel pools C and D, despite the NUREG-1353 conditional probability estimate.

VI. NEPA REQUIRES NO FURTHER ANALYSES

Answer to Board Question 3: The NRC Staff Does Not Have to Prepare Additional Environmental Impact Analyses Even If the Board Should Decide a Probability of Occurrence on the Order of a Few Chances in One Hundred Million Per Year is Not Sufficient to Classify BCOC's Postulated Scenario as Remote and Speculative.

As a final point, the Board asked the parties to address the following issue:

> Assuming the Board should decide that the probability involved is of sufficient moment so as not to permit the postulated accident sequence to be classified as "remote and speculative," what would the overall scope of the environmental impact analysis the staff would be required to prepare (i.e., limited to the impacts of that accident sequence or a full blown EIS regarding the amended request)?[202]

In light of the infinitesimal probability of the postulated scenario, existing case law support for much higher probabilities as "remote and speculative," and the arguments in favor of one-in-a-million as a threshold, this question appears moot. However, CP&L has reviewed this matter carefully and has concluded that, under the circumstances, the NRC Staff would not have to prepare additional environmental impact analyses even assuming the Board should decide a probability on the order of a few chances in one

201 Id. ¶ 16.

hundred million per year is not sufficient to classify the postulated scenario as "remote and speculative."

The fundamental legal question in applying NEPA is, as discussed supra, whether the cognizant federal agency "has adequately considered and disclosed the environmental impact of its actions."[203] This has been done with respect to the potential environmental impacts of spent fuel storage generically, specifically at Harris, and at every other nuclear plant in the country. As discussed in Section III.B. supra, NEPA requires nothing more than a "hard look." The Commission has given the potential environmental impacts of wet, dry, short-term, and long-term storage of spent nuclear fuel a very, very hard look for operating plants, decommissioning plants, and independent spent fuel storage facilities. In considering what emergency planning should remain in place for a shutdown, decommissioning plant, the Commission continues to this day to take a hard look at potential accidents and consequences of the long-term storage of spent fuel.

With respect to Harris, the FES issued at the time of the operating license considered the environmental impacts of operation of all four spent fuel pools (for what was understood at the time to be two operating units, with additional storage of spent fuel shipped from CP&L's other nuclear units). The quantity of spent fuel assumed to be stored at the time of the Harris FES exceeds the quantity of spent fuel that can be stored

---

Footnote continued from previous page

202 Order at 17.

203 Baltimore Gas & Electric Co., 462 U.S. at 98; see also Robertson, 490 U.S. at 350; Kleppe, 427 U.S. at 409-410.

pursuant to the License Amendment Application presently before this Board.[204] Further, in its 1999 Environmental Assessment, the NRC Staff explicitly stated that the license amendment "does not involve the use of any resources not previously considered" in the FES.[205]

The "environmental risks" of BCOC's postulated scenario at Harris are bounded by the existing NEPA analyses.

> The environmental impacts that have been considered include potential radiation exposures to individuals and to the population as a whole, the risk of near- and long-term adverse health effects that such exposures could entail, and the potential economic and societal consequences of accidental contamination of the environment. These impacts could be severe, but the likelihood of their occurrence is judged to be small. . . . *The overall assessment of environmental risk of accidents*, assuming protective action, shows that *it is on the same order as the risk from normal operation*, although accidents have a potential for early fatalities and economic costs that cannot arise from normal operations. *The risks* of early fatality from potential accidents at the site *are small in comparison with risks of accidental deaths from other human activities* in a comparably sized population.[206]

Indeed, the theoretical consequences and limiting time to restore cooling or makeup water to the spent fuel pools at Harris are driven by the higher heat load of the fuel stored in spent fuel pools A and B. Spent fuel pools C and D add almost no potential risk

---

204 See Note 4, supra. However, the postulated scenario was also analyzed for the assumed maximum 15.6 MBTU/hr heat rate in spent fuel pools C and D that could be achieved in the future after modifications to cooling systems. Burns Aff. ¶ 18; Edwards Aff. ¶¶ 20 - 22.

205 EA at 9.

206 FES § 5.9.4.6 (emphasis added).

because, under the postulated scenario, it would take over 100 days to evaporate the water in those pools[207] and, because of the low heat rate, the probability of a self-sustaining exothermic oxidation reaction of zircaloy spent fuel cladding is highly unlikely, perhaps impossible.[208] The environmental risks of the proposed activity in the License Amendment Application are, therefore, bounded by the already licensed activity in spent fuel pools A and B.

CP&L has established that the best estimate probability of occurrence of the postulated scenario is on the order of 2.65×10^{-8} per year, which is nearly *three orders of magnitude* (i.e., a factor of 1,000) below the LERF.[209] BCOC's worst case scenario involves a *complete* release of radioactivity from a fire involving *all* the fuel elements in *all* Harris spent fuel pools when they are *completely* filled with freshly discharged fuel from approximately 30 reactor cores. Using the generally accepted definition of risk (i.e., probability times consequences), the environmental risk of BCOC's worst case postulated environmental impact is, therefore, *still one to two orders of magnitude less* than the risk the NRC Staff already considered in the FES.[210] There is nothing significant or relevant

---

207 Edwards Aff. ¶ 22.

208 Kunita Aff. ¶ 35.

209 See discussion section III.C, supra.

210 Dr. Thompson conceded during his deposition that if LERF were an acceptable safety goal, and an accident had a potential consequence an order of magnitude greater than the LERF accident, but also had a annual probability of occurrence an order of magnitude less, then the risk was equivalent. Thompson Dep. at 191-93.

about such an unlikely occurrence and there is no basis to order the NRC Staff to *further* analyze consequences that are dwarfed by those already considered.

In summary, the potential consequences of the seven-step postulated scenario have received all the consideration and analysis required by NEPA, whether or not it is deemed "remote and speculative." The case law is absolutely clear that an agency must prepare a supplement to an EIS only if there are *significant new* circumstances or information *relevant* to environmental concerns from the proposed action or its impacts. BCOC has failed to demonstrate that the postulated scenario is significant or adds new information relevant to the environmental impacts from Harris. Further, CP&L has shown that the environmental risks of the postulated scenario, even under worst case conditions and assuming consequences greater than those from a severe degraded core accident, are bounded by the existing Harris FES. There is no reason to require further analyses and NEPA does not so require.

VII. ACTIONS REQUESTED OF THE BOARD

Applicant CP&L respectfully submits that, at the conclusion of oral argument, the Board should, pursuant to 10 C.F.R. § 2.1115, "promptly by written order":

1. Determine that no issue of law or fact shall be designated for resolution in an adjudicatory hearing.

2. Dispose of Contention EC-6. The License Amendment Application to permit commissioning of spent fuel pools C and D for storage of up to 1.0 MBTU/hr of spent nuclear fuel increases neither the probability nor potential consequences of accidents at Harris. In fact, the addition of a redundant spent fuel pool cooling and cleanup system for spent fuel pools C and D provides alternative makeup water paths to the spent fuel pools and reduces the probability of the postulated scenario. The postulated scenario is highly "remote and speculative" and the environmental risk is

insignificant and is bounded by the existing environmental risk of the licensed activity and by existing NEPA analyses.

3. Find as a matter of fact and conclude as a matter of law that the NRC Staff has satisfied its obligations pursuant to NEPA and need not prepare a supplemental environmental assessment or environmental impact statement.

4. Dismiss this proceeding.

Respectfully submitted,

John H. O'Neill, Jr.
Douglas J. Rosinski
SHAW PITTMAN
2300 N Street, N.W.
Washington, D.C. 20037-1128
(202) 663-8000
Counsel For CAROLINA POWER & LIGHT COMPANY

Of Counsel:
Steven Carr
Legal Department
CAROLINA POWER & LIGHT COMPANY
411 Fayetteville Street Mall
Post Office Box 1551 – CPB 13A2
Raleigh, North Carolina 27602-1551
(919) 546-4161

Dated: November 20, 2000

1035326

000637

November 20, 2000

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
CAROLINA POWER & LIGHT	)	Docket No. 50-400-LA
COMPANY	)	
(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

EXHIBIT 1

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

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

000641

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.

000647

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

000651

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.

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

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.

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

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

Dr. Edward T. Burns

- **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. Edward T. Burns

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

Karl N. Fleming

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

Karl N. Fleming

Page 3

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

Karl N. Fleming

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
Page 5

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.

Karl N. Fleming

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.

Fleming, K. N., et al., "Risk and Performance Based Prioritization of Motor Operated Valve Testing - A PSA Application for the South Texas Electric Generating Station," Houston Lighting and Power Company, Wadsworth, Texas, January, 1996.

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.

Karl N. Fleming

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.

Fleming, K. N., S. B. Rao, G. A. Tinsley, A. Mosleh, and A. Afzali, "A Database of Common Cause Events for Risk and Reliability Evaluations," PLG, Inc., prepared for Electric Power Research Institute, PLG-0866, March, 1992.

Contributing Author to:
Pickard, Lowe and Garrick, Inc., and Stone & Webster Engineering Corporation, "Beaver Valley Unit 2 Probabilistic Risk Assessment," prepared for Duquesne Light Company, PLG-0730, December, 1989.

Fleming, K. N., "Recent Trends in Evaluation of Large, Early Release Frequency in PWR Plants," *Transactions of ANS Winter Meeting*, San Francisco, California, November, 1989.

Garrick, B. J., A. Torri, K. N. Fleming, R. K. Deremer, and J. E. Metcalf, "Individual Plant Examination (IPE) Containment Analysis," Short Course, Pickard, Lowe and Garrick, Inc., PLG-0711, Newport Beach, California, June 19-22, 1989.

Frank, M. V., and K. N. Fleming, "Risk-Significant Dependencies in PWRs," *Proceedings of PSA '89 International Topical Meeting Probability, Reliability, and Safety Assessment*, Pittsburgh, Pennsylvania, pp. 245-252, April 2-7, 1989.

Karl N. Fleming

PUBLICATIONS (continued)

Mosleh, A., K. N. Fleming, G. W Parry, H. M. Paula, D. H. Worledge, and D. M. Rasmuson, "Methodological Advancements in Procedures for Common Cause Failure Analysis."

Fleming, K. N., and K. L. Kiper, "Treatment of Uncertainties in a PRA Evaluation of Events Initiated at Shutdown," *Proceedings of PSA '89 International Topical Meeting Probability, Reliability, and Safety Assessment*, Pittsburgh, Pennsylvania, pp. 913-918, April 2-7, 1989.

Fleming, K. N., and A. Torri, "Accident Management To Ensure Containment Integrity at Seabrook Station," *Proceedings of PSA '89 International Topical Meeting Probability, Reliability, and Safety Assessment*, Pittsburgh, Pennsylvania, pp. 999-1003, April 2-7, 1989.

Fleming, K. N., "Parametric Models for Common Cause Failure Analysis," *Proceedings of Advanced Seminar on Common Cause Failure Analysis in Probabilistic Safety Assessment*, Ispra, Italy, pp. 159-174, November 16-19, 1988.

Kaplan, S., and K. N. Fleming, "On the Use of the Cause Table in Handling Common Cause Events in Systems Analysis," *Risk Analysis 7*, No. 4, December, 1987.

Fleming, K. N., "Parametric Models for Common Cause Failure Analysis," presented to Advanced Seminar on Common Cause Failure Analysis, Joint Research Centre, Ispra, Italy, November 16-19, 1987.

Kaplan, S., and K. N. Fleming, "On the Use of the Cause Table in Handling Common Cause Events in System Analysis," presented at PSA '87 International Topical Conference on Probabilistic Safety Assessment and Risk Management, Zurich, Switzerland, August 30-September 4, 1987.

Fleming, K. N., R. K. Deremer, and J. H. Moody, "Realistic Assessment of Interfacing System LOCA Scenarios in PWR Plants," presented at PSA '87 International Topical Conference on Probabilistic Safety Assessment and Risk Management, Zurich, Switzerland, August 30-September 4, 1987.

Moody, J. H., and K. N. Fleming, "Application of Seabrook Station PSA in Evaluation of Core Melt, Emergency Planning Strategies and Decision Making," presented at PSA '87 International Topical Conference on Probabilistic Safety Assessment and Risk Management, Zurich, Switzerland, August 30-September 4, 1987.

Fleming, K. N., A. Mosleh, and D. H. Worledge, "Development of a Systematic Approach for the Analysis of System-Level Dependent Failures," presented at PSA '87 International Topical Meeting Conference on Probabilistic Safety Assessment and Risk Management, Zurich, Switzerland, August 30-September 4, 1987.

Fleming, K. N., et al., "Risk Management Actions To Assure Containment Effectiveness at Seabrook Station," Pickard, Lowe and Garrick, Inc., prepared for New Hampshire Yankee Division, Public Service of New Hampshire, Seabrook, New Hampshire, PLG-0550, July, 1987.

Karl N. Fleming

PUBLICATIONS (continued)

Mosleh, A., K. N. Fleming, et al., "Procedures for Treating Common Cause Failures in Safety and Reliability Studies," Pickard, Lowe and Garrick, Inc., prepared for U.S. Nuclear Regulatory Commission and Electric Power Research Institute, NUREG/CR-4780, April, 1987.

Fleming, K. N., and P. Doerre, "Possible Pitfalls in CCG Event Data Evaluation," 1986.

Fleming, K. N., et al., "Seabrook Station Emergency Planning Sensitivity Study," Pickard, Lowe and Garrick, Inc., prepared for Public Service of New Hampshire, PLG-0465, April, 1986.

Fleming, K. N., A. Mosleh, and R. K. Deremer, "A Systematic Procedure for the Incorporation of Common Cause Events into Risk and Reliability Models," *Nuclear Engineering and Design 93*, Elsevier Science Publishers B.V., pp. 245-273, 1986.

Fleming, K. N., et al., "PRA Procedures for Dependent Events Analysis: Vol. I: Plant-Level Analysis; Vol. II: System-Level Analysis," Pickard, Lowe and Garrick, Inc., prepared for Electric Power Research Institute, PLG-0453, December, 1985.

Fleming, K. N., A. Torri, K. Woodard, and R. K. Deremer, "Seabrook Station Risk Management and Emergency Planning Study," Pickard, Lowe and Garrick, Inc., prepared for Public Service of New Hampshire, New Hampshire Yankee Division, PLG-0432, December, 1985.

Fleming, K. N., "PRA Procedures for Dependent Events Analysis - An Industry Perspective," presented at 13th Water Reactor Safety Research Meeting, National Bureau of Standards, Gaithersburg, Maryland, October 22-25, 1985.

Bier, V. M., K. N. Fleming, and L. Wredenberg, "Analysis of Steam Generator Reliability Improvement Options for the Ringhals 2 Nuclear Power Plant," presented at American Society for Metals International Conference on Nuclear Power Plant Aging, Availability Factor, and Reliability Analysis, San Diego, California, July 8-12, 1985.

Fleming, K. N., J. H. Moody, and K. Kiper, "The Seabrook PRA Viewed from Three Perspectives," *Proceedings of American Nuclear Topical Meeting on PRA*, San Francisco, California, February 24-28, 1985.

Fleming, K. N., and A. Mosleh, "Common Cause Data Analysis and Applications in System Modeling," *Proceedings of American Nuclear Society Topical Meeting on PRA*, San Francisco, California, February 24-28, 1985.

Fleming, K. N., "Systems Analysis of Common Cause Events and the Incorporation of Experience Data," Power Plant Risk and Availability Management, University of California Extension, Los Angeles, California, January 14-18, 1985.

Stampelos, J. G., J. K. Liming, and K. N. Fleming, "Benchmark of Systematic Human Action Reliability Procedure (SHARP)," Pickard, Lowe and Garrick, Inc., prepared for Electric Power Research Institute, EPRI NP-3583, PLG-0382, October 18, 1984.

Karl N. Fleming
Page 11

PUBLICATIONS
(continued)

Bier, V. M., and K. N. Fleming, "Analysis of Steam Generator Replacement and Improvement Options for the Ringhals 2 Plant," Pickard, Lowe and Garrick, Inc., prepared for Swedish State Power Board, PLG-0383, October, 1984.

Fleming, K. N., "The Seabrook Station Probabilistic Safety Assessment and the Issue of Completeness," *Proceedings of Society of Risk Analysis Annual Meeting*, Knoxville, Tennessee, September 30-October 3, 1984.

Fleming, K. N., A. Mosleh, D. L. Acey, and D. H. Worledge, "Event Classification and Systems of Common Cause Failures," presented at American Nuclear Society 1984 Annual Meeting, New Orleans, Louisiana, June 3-8, 1984.

Garrick, B. J., K. N. Fleming, and A. Torri, "Seabrook Station Probabilistic Safety Assessment Technical Summary Report," Pickard, Lowe and Garrick, Inc., prepared for Public Service Company of New Hampshire and Yankee Atomic Electric Company, PLG-0365, June, 1984.

Fleming, K. N., "Baseline Study Project Plan: South Texas Project, Energy Generating Station Plant Risk Model Development Program for the STP Probabilistic Risk Assessment," Pickard, Lowe and Garrick, Inc., prepared for Houston Lighting & Power Company, PLG-0361, April, 1984.

Contributing Author to:
Pickard, Lowe and Garrick, Inc., "Applications of PRA Methods to the Systems Interaction Issue," prepared for Electric Power Research Institute, PLG-0284, April, 1984.

Contributing Author to:
Pickard, Lowe and Garrick, Inc., "Classification and Analysis of Reactor Operating Experience Involving Dependent Events," prepared for Electric Power Research Institute, PLG-0334, January, 1984.

Kazarians, M., and K. N. Fleming, "Internal Flood Hazard Model," *American Nuclear Society Transactions 45*, TANSAO 45, 1-884, 1983.

Fleming, K. N., A. Mosleh, and A. P. Kelley, Jr., "On the Analysis of Dependent Failure in Risk Assessment and Reliability Evaluation," *Nuclear Safety 24*, No. 5, October, 1983.

Lydell, B. O. Y., K. N. Fleming, D. J. Wakefield, F. R. Hubbard, and A. Mosleh, "Preliminary Review of the Ringhals 2 Probabilistic Safety Study," Pickard, Lowe and Garrick, Inc., prepared for Swedish Nuclear Power Inspectorate, PLG-0304, October, 1983.

Fleming, K. N., and A. M. Kalinowski, "An Extension of the Beta Factor Method to Systems with High Levels of Redundancy," Pickard, Lowe and Garrick, Inc., prepared for Swedish Nuclear Power Inspectorate, PLG-0289, June, 1983.

Fleming, K. N., and J. G. Stampelos, "Analysis of Dependent Failures on Contemporary Risk Studies in the United States," *Proceedings of Workshop on Dependent Failure Analysis*, Vasteras, Sweden, April 27-28, 1983.

Karl N. Fleming

PUBLICATIONS (continued)

Contributing Author to:
Pickard, Lowe and Garrick, Inc., "Seabrook Station Probabilistic Safety Assessment," prepared for Public Service Company of New Hampshire and Yankee Atomic Electric Company, PLG-0300, December, 1983.

Kazarians, M., and K. N. Fleming, "Internal and External Flood Failure Model," presented at Impact of Natural Hazards, Fire and Flood on Nuclear Power Facilities, ACTA Inc., Palo Alto, California, April 13-15, 1983.

Heising, C. D., A. W. Barsell, K. N. Fleming, S. Kaplan, and B. J. Garrick, "A Comparison of Recent Nuclear Plant Risk Assessments," presented at Congres Annuel 1982, SFRP, La Comparison des Risques Associes aux Grandes Activites Humanines, Avignon, France, October 18-22, 1982.

Fleming, K. N., "Program Definition Report and Implementation Plan for Probabilistic Risk Assessment," Pickard, Lowe and Garrick, Inc., prepared for U.S. Department of Energy, PLG-0249, December, 1982.

Heising, C. D., and K. N. Fleming, "Development of Unavailability Expressions for One and Two Component Systems with Periodic Testing and Common Cause Failures," October, 1982.

Contributing Author to:
Pickard, Lowe and Garrick, Inc., "Seabrook Station Probabilistic Safety Assessment: Phase I Preliminary Risk Model Development," prepared for Public Service Company of New Hampshire and Yankee Atomic Electric Company, PLG-0242, August, 1982.

Garrick, B. J., G. Apostolakis, and K. N. Fleming, "Comments on NUREG/CR-2815 National Reliability Evaluation Program (NREP) Procedures Guide," June 12, 1982.

Fleming, K. N., S. Kaplan, and B. J. Garrick, "Seabrook Probabilistic Safety Assessment Management Plan," prepared for Public Service Company of New Hampshire, Pickard, Lowe and Garrick, Inc., PLG-0239, June, 1982.

Fleming, K. N., and M. K. Ravindra, "Seismic, Fire, and Flood Analyses as Applied to Risk Studies," International ANS/ENS Topical Meeting on Probabilistic Risk Assessment, Port Chester, New York, September 20-24, 1981.

Fleming, K. N., "Common Cause Methods in PRA and Implications for Licensing," presented to Advisory Committee for Reactor Safeguards Subcommittee on Reliability and Probabilistic Risk Assessment, Los Angeles, California, July 28, 1981.

Frank, M. V., D. D. Orvis, and K. N. Fleming, "Three Aspects of Reliability for a Geological Nuclear Waste Repository," *Transactions of ANS Annual Meeting*, Miami, Florida, June 7-11, 1981.

Fleming, K. N., W. J. Houghton, G. W. Hannaman, and V. Joksimovic, "Probabilistic Risk Assessment of HTGRs," *Reliability Engineering 2*, No. 1, March, 1981.

Karl N. Fleming

PUBLICATIONS (continued)

Fleming, K. N., "Risk Assessment of Major Fires in an HTGR Plant," *Proceedings of American Nuclear Society International Meeting on Thermal Reactor Safety*, Knoxville, Tennessee, April 11-18, 1980.

Fleming, K. N., G. W. Hannaman, and F. A. Silady, "Treatment of Operator Actions in HTGR Risk Assessment Study," *Transactions of American Nuclear Society*, San Francisco, California, 1979.

Fleming, K. N., W. J. Houghton, and F. P. Scaletta, "A Methodology for Risk Assessment of Major Fires and Its Applications to an HTGR Plant," General Atomic Company, prepared for U.S. Department of Energy, GA-A15402, 1979.

Fleming, K. N., and P. H. Raabe, "A Comparison of Three Methods for the Quantitative Analysis of Common Cause Failures," *Proceedings of American Nuclear Society Meeting on Probabilistic Safety Analysis*, Los Angeles, California, 1978.

Joksimovic, V., and K. N. Fleming, "Applications of Probabilistic Risk Assessment in the Development of HTGR Technology," *Proceedings of American Nuclear Society Thermal Reactor Safety Meeting*, Sun Valley, California, 1977.

Cairns, J. J., and K. N. Fleming, "STADIC A Computer Code for Combining Probability Distributions," General Atomic Company, prepared for U.S. Department of Energy, GA-A14055, 1977.

Fleming, K. N., "A Reliability Model for Common Mode Failures in Redundant Safety Systems," *Proceedings of Sixth Annual Conference on Modeling and Simulation*, Pittsburgh, Pennsylvania, April, 1975.

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

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

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

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

Vincent M. Andersen

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

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.

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

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

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

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

Institute

Thomas A. Daniels

Page 2

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

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

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
Page Two

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

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 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.

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 Republic
- 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

Page 2

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

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 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.

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

000716

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

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
K.N. Fleming

Approved by: Date: 11/16/00
D.E. True

Accepted by: Date: 11-16-00
CP&L (NGG)

Revisions:

Rev.	Description	Preparer/Date	Reviewer/Date	Approver/Date
0	Original Issue			

TABLE OF CONTENTS

Section		Page
EXECUTIVE SUMMARY		iii
ACRONYMS, INITIALS, DEFINITIONS		ix
1.0	INTRODUCTION	1-1
1.1	Statement of Question Addressed	1-1
1.2	Scope	1-1
1.3	Plant Configuration	1-4
2.0	METHODOLOGY	2-1
2.1	Methodology	2-1
2.2	Overview	2-6
2.3	Risk Analysis: Initiators, Sequences, Deterministic Modeling	2-8
2.4	Containment Failure Modes and Critical Times	2-23
2.5	Scope, Key Assumptions, and Groundrules	2-35
3.0	PSA STATUS AND QUALITY	3-1
3.1	Internal Events	3-1
3.2	Seismic	3-3
3.3	Fire	3-4
3.4	Other External Events	3-5
3.5	Shutdown	3-5
3.6	Summary	3-5
4.0	SPENT FUEL POOL COOLING ANALYSIS	4-1
4.1	Internal Events	4-1
4.2	Seismic Events	4-19
4.3	Fire Initiated Accident Sequences	4-54
4.4	An Analysis of PWR Shutdown Risk	4-58
4.5	Other External Events	4-76

TABLE OF CONTENTS (Cont'd)

Section		Page
5.0	RESULTS AND SENSITIVITIES	5-1
5.1	Introduction	5-1
5.2	Overview of Uncertainty	5-1
5.3	Sensitivity Case	5-6
5.4	Sensitivity Evaluation	5-7
5.5	Sensitivity Results	5-21
6.0	CONCLUSIONS	6-1
6.1	Overview	6-1
6.2	Conclusions	6-4
6.3	Conservatisms	6-7
7.0	REFERENCES	7-1

Appendix A SPENT FUEL POOLS AND ASSOCIATED EQUIPMENT

Appendix B DISCUSSION OF REMOTE AND SPECULATIVE

Appendix C HUMAN RELIABILITY ANALYSIS

Appendix D SPENT FUEL POOL ASSESSMENT EVENT TREE (SFP-AET)

Appendix E DETERMINISTIC ANALYSIS

Appendix F WALKDOWN OF THE SHEARON HARRIS REACTOR AUXILIARY AND FUEL HANDLING BUILDINGS

Appendix G SEISMIC ANALYSIS QUANTIFICATION DETAILS

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.

SHNPP PSA Quality

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.

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

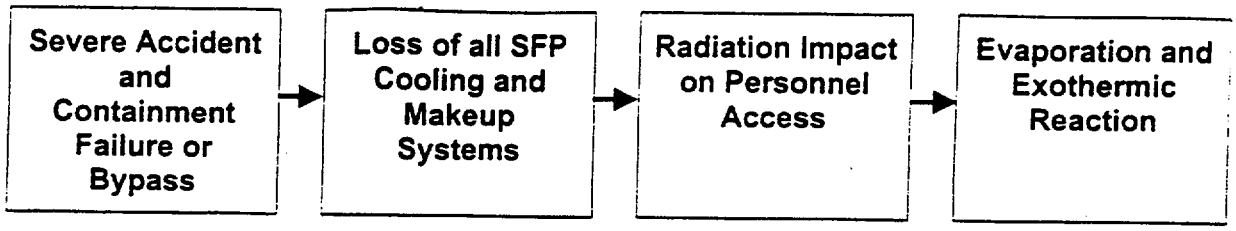
The following subsections describe in overview fashion the methods used in the evaluation of various contributors to the event frequency profile for the Postulated Sequence. The details of the implementation of these methods are described in Section 4.

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.



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).

Time

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.