NUREG-1742
Vol. 2

Perspectives Gained From the Individual Plant Examination of External Events (IPEEE) Program

Tables

Final Report

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Washington, DC 20555-0001

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ABSTRACT

On June 28, 1991, the U.S. Nuclear Regulatory Commission (NRC) issued Supplement 4 to Generic Letter (GL) 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, 10 CFR 50.54(f)," and NUREG-1407, "Procedure and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities." Specifically, the NRC requested that each licensee perform an IPEEE to identify and report to the NRC all plant-specific vulnerabilities to severe accidents caused by external events. The external events to be considered in the IPEEE included seismic events; internal fires; and high winds, floods, and other (HFO) external initiating events including accidents related to transportation or nearby facilities and plant-unique hazards. All of the currently operating U.S. nuclear power plants have completed their assessments and submitted their analyses for NRC review.

The objective of the NRC's IPEEE submittal reviews was to ascertain whether the licensees' IPEEE processes were capable of identifying severe accident vulnerabilities to such external events, and implementing cost-effective safety improvements to either eliminate or reduce the impact of those vulnerabilities. However, the reviews did not attempt to validate or verify the results of a licensee's IPEEE.

The primary purpose of this report is to document the perspectives gleaned from the technical reviews of the IPEEE submittals. These include a description of the overall IPEEE process and findings; conclusions regarding the dominant risk contributors for the major areas of evaluation (i.e., seismic events, fires, and HFO events); an overview of plant improvements made by licensees as a result of the IPEEE program; a description of the overall strengths and weaknesses in the licensees' implementation of the IPEEE evaluation methodologies; and an assessment of the overall effectiveness in meeting the IPEEE objectives, including the extent to which licensees have met the intent of Supplement 4 to Generic Letter 88-20. Volume 1 of this report includes general IPEEE perspectives while Volume 2 includes detailed tables with plant-specific information relevant to the IPEEE program.

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1. INTRODUCTION

On June 28, 1991, the U.S. Nuclear Regulatory Commission (NRC) issued Supplement 4 to Generic Letter (GL) 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, 10 CFR 50.54(f)." That supplement described the objectives and overall logistics of the Individual Plant Examination of External Events (IPEEE) program, which addresses externally initiated events. In particular, the external events considered in the IPEEE program include seismic events; internal fires; and high winds, floods, and other (HFO) external initiating events involving accidents related to transportation and nearby facilities. The IPEEE program was intended as a means for licensees to identify potential vulnerabilities to severe accidents initiated by external events, and to conceive cost-effective improvements to ensure that plants do not pose any undue risk to public health and safety.

Along with Supplement 4 to GL 88-20, the NRC issued NUREG-1407, "Procedure and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," in June 1991. In NUREG-1407, the NRC provided guidelines for conducting IPEEEs. Specifically, the guidance pertained to evaluations concerning the following external initiators: seismic events; internal fires; and high winds, floods, and other (HFO) external events, including accidents related to transportation or nearby facilities and plant-unique hazards. Subsequent to the publication of NUREG-1407, the NRC issued Supplement 5 to GL 88-20 on September 8, 1995, to notify licensees of modifications to the recommended scope of the seismic portion of the IPEEE for certain plant sites in the eastern United States (EUS).

The NRC received 70 IPEEE submittals covering all operating U.S. nuclear reactors. (Some submittals covered more than one unit at multi-unit sites with similar or almost identical plant designs.) The staff of the NRC's Office of Nuclear Regulatory Research completed 69 Staff Evaluation Reports (SERs) which document the staff's overall conclusions for each of the IPEEE reviews.[1] Additional details on the plant-specific IPEEE review findings are presented in Technical Evaluation Reports (TERs) for each of the 69 IPEEE submittals.[2] Each TER discusses the strengths and weaknesses of the licensee's IPEEE submittal, particularly with reference to the guidelines established in NUREG-1407. The TERs also typically present (1) an overview of the licensee's IPEEE process and insights; (2) the review process employed for evaluation of the seismic, fire, and HFO events; (3) the dominant contributors to core damage frequency for fire, seismic, and HFO events; (4) licensee-identified vulnerabilities; (5) plant improvements made or planned as a result of the licensee's IPEEE process; and (6) an overall evaluation of the strengths and weaknesses of the IPEEE submittal. This report provides insights and perspectives gleaned from the reviews of all of the licensees' submittals.

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[1] One plant, Haddam Neck, was permanently shut down, so the staff suspended work on reviewing that plant's IPEEE submittal.

[2] The TERs in the seismic and fire areas were written by NRC contractors (i.e., Energy Research, Inc. (ERI), Brookhaven National Laboratory (BNL), and Sandia National Laboratories (SNL)). TERs in the HFO area were written by ERI (26 submittals) and RES staff (the balance of the submittals). Readers interested in specific plants can obtain the plant-specific SERs and TERs through the NRC's Agencywide Documents Access and Management System (ADAMS). (Include the plant name and "IPEEE" in the Title Contains block of the ADAMS Find window.) SERs that were issued prior to November 1999 are available to the public, for a fee, by contacting the NRC Public Document Room (PDR) librarian at (800) 397-4209 or by e-mail to pdr@nrc.gov.

In developing this report, the staff sought to address each distinct, significant topic considered in NUREG-1407, including seismic events, fires, and HFOs, as well as the relevant IPEEE-related aspects of generic safety issues (GSIs) and unresolved safety issues (USIs). Volume 1 of this report includes general IPEEE perspectives, while Volume 2 includes detailed tables with plant-specific information relevant to the IPEEE program.

In Volume 1, Chapter 1 covers the general background and objectives of the IPEEE program, while Chapter 2 discusses the perspectives derived from the seismic portion of the IPEEE submittals, and includes comments regarding licensees' seismic probabilistic risk assessments (PRAs) and seismic margin assessments (SMAs). It also discusses information provided in seismic IPEEE submittals relevant to specific GSIs and USIs.

Chapter 3 discusses the perspectives derived from the fire portion of the IPEEE submittals, and includes comments regarding licensees' fire PRAs and fire-induced vulnerability evaluation (FIVE) studies. It also discusses fire-related findings concerning specific GSIs and USIs, as well as issues arising from the fire risk scoping study conducted by Sandia National Laboratories (SNL).

Chapter 4 presents findings derived from the HFO portion of the IPEEE submittals. Each major category of HFO initiator is discussed, including high winds and tornadoes, external floods, and accidents related to transportation or nearby facilities. It also discusses HFO-related findings concerning specific GSIs and USIs.

Chapters 2 through 4 each provide summaries of applicable walkdown findings, human action perspectives, containment performance perspectives, plant improvements, generic versus plant-specific perspectives, as well as observations of specific strengths and weaknesses relevant to the evaluation of each particular type of external initiator.

Chapter 5 describes each of the external-event related unresolved and generic safety issues and provides the staff's conclusions regarding the resolution of these issues for each plant.

The staff anticipates that this report will be used by readers with different backgrounds. Some terms used in this report may have different definitions depending on the technical context in which they are used. Therefore, a glossary is provided at the beginning of Volume 1 to aid the reader in understanding the specific meaning of each term used in this report.

Volume 2 of this report, which includes detailed plant-specific tables, is organized as follows: Section 2 covers seismic events; Section 3 fire; Section 4 high winds, floods, and other external events; and Section 5 IPEEE-related unresolved safety issues and generic safety issues.

2. SEISMIC TABLES

This section contains 13 plant-specific tables of summary information obtained from the seismic portions of the IPEEE. Table 2.1 contains the seismic review category and evaluation approach used. Tables 2.2 through 2.6 provide information about those plants that performed a seismic PRA. Table 2.2 contains the core damage frequency (CDF). Table 2.3 identifies the licensee-identified dominant risk contributors. Table 2.4 lists the licensee-identified plant improvements along with the screening used during the walkdown and walkdown findings. Table 2.5 shows the containment type and the results of the licensee's seismic containment evaluation, including quantitative and qualitative results and identified plant improvements.

Those licensees that did not perform a seismic PRA performed a seismic margin assessment. Tables 2.6 through 2.8 provide information about those plants that performed a seismic margin assessment. Table 2.6 identifies the format of the analysis, the basis for the earthquake spectral shape, and the licensee's identified high confidence of low probability of failure (HCLPF) value. Table 2.7 lists the licensee-identified plant improvements along with the screening used during the walkdown and findings. Table 2.8 shows the containment type and the results of the licensee's seismic containment evaluation, including quantitative and qualitative results and identified plant improvements.

Table 2.9 contains plant-specific information concerning the potential of low-ruggedness relays to chatter (open and close their contacts repeatedly) during a seismic event. This table describes the licensee's approach to evaluating relay chatter, implications of the relay chatter, and plant improvements.

Table 2.10 lists the soil and foundation characteristics for each plant site. Based on the plants' classification in NUREG-1407, licensees were to perform different levels of assessment of the earthquake effect on the ground (foundation) and plant facilities. The licensees' results are presented in this table. The earthquake effects on operator actions, as identified in the licensees' submittals, are shown in Table 2.11.

Other potential effects of earthquakes include seismically induced fires or floods. The results of the licensees' assessments of these two possibilities are shown in Table 2.12. The potential for an earthquake to cause the in-core flux mapping system to fail in such a manner as to result in a loss of coolant accident (LOCA) is applicable only to Westinghouse plants. The results of the licensees' evaluation of the flux mapping system are shown in Table 2.13.

Table 2.1: Seismic review categories and evaluation approaches

Plant	Seismic review category	Seismic IPEEE evaluation approach
Arkansas Nuclear One 1	3 (0.3g full-scope)	0.3g full-scope EPRI SMA
Arkansas Nuclear One 2	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Beaver Valley 1	2 (0.3g focused-scope)	Seismic PRA
Beaver Valley 2	2 (0.3g focused-scope)	Seismic PRA
Braidwood 1&2	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Browns Ferry 2&3	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Brunswick 1&2	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Byron 1&2	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Callaway	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Calvert Cliffs 1&2	2 (0.3g focused-scope)	Seismic PRA (using surrogate element at 0.3g)
Catawba 1&2	2 (0.3g focused-scope)	Existing Seismic PRA
Clinton	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Columbia Generating*	5 (0.5g full-scope)	Seismic PRA (using surrogate element at 0.5g)
Comanche Peak 1&2	1 (reduced-scope)	reduced-scope EPRI SMA
Cooper	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Crystal River 3	1 (reduced-scope)	reduced-scope EPRI SMA
D.C. Cook 1&2	2 (0.3g focused-scope)	Seismic PRA
Davis-Besse	2 (0.3g focused-scope)	reduced-scope EPRI SMA
Diablo Canyon 1&2	7 (Seismic PRA)	Existing Seismic PRA
Dresden 2&3	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Duane Arnold	1 (reduced-scope)	reduced-scope EPRI SMA
Farley 1&2	1 (reduced-scope)	reduced-scope EPRI SMA
Fermi 2	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
FitzPatrick	2 (0.3g focused-scope)	0.3g focused-scope NRC SMA
Fort Calhoun 1	2 (0.3g focused-scope)	0.3g focused-scope NRC SMA (using surrogate element at 0.5g)
Ginna	2 (0.3g focused-scope)	Modified focused-scope EPRI SMA

Table 2.1: Seismic review categories and evaluation approaches (Continued)

Plant	Seismic review category	Seismic IPEEE evaluation approach
Grand Gulf 1	1 (reduced-scope)	reduced-scope EPRI SMA
H.B. Robinson 2	3 (0.3g full-scope)	0.3g full-scope EPRI SMA
Haddam Neck	2 (0.3g focused-scope)	Seismic PRA (using surrogate element at 0.3g)
Hatch 1&2	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Hope Creek	2 (0.3g focused-scope)	Seismic PRA
Indian Point 2	3 (0.3g full-scope)	Update of Existing Seismic PRA (using surrogate element at 0.5g)
Indian Point 3	3 (0.3g full-scope)	Seismic PRA (using surrogate element at 0.3g)
Kewaunee	2 (0.3g focused-scope)	Seismic PRA (using surrogate element at 0.3g)
La Salle 1&2	2 (0.3g focused-scope)	Existing Simplified Seismic PRA (SSMRP)
Limerick 1&2	2 (0.3g focused-scope)	reduced-scope EPRI SMA
McGuire 1&2	2 (0.3g focused-scope)	Existing Seismic PRA
Millstone 2	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Millstone 3	2 (0.3g focused-scope)	Existing Seismic PRA
Monticello	2 (0.3g focused-scope)	Modified focused/Expanded reduced-scope EPRI SMA
Nine Mile Point 1	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Nine Mile Point 2	2 (0.3g focused-scope)	SPRA & focused EPRI SMA (using surrogate element at 0.5g)
North Anna 1&2	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Oconee 1,2,&3	3 (0.3g full-scope)	Seismic PRA (using surrogate element at 0.3g)
Oyster Creek	2 (0.3g focused-scope)	Seismic PRA
Palisades	2 (0.3g focused-scope)	Seismic PRA (using surrogate element at 0.5g)
Palo Verde 1,2,&3	5 (0.5g full-scope)	0.3g full-scope EPRI SMA
Peach Bottom 2&3	2 (0.3g focused-scope)	Modified focused-scope EPRI SMA
Perry 1	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Pilgrim 1	4 (Seismic PRA)	Seismic PRA (using surrogate element at

Table 2.1: Seismic review categories and evaluation approaches (Continued)

Plant	Seismic review category	Seismic IPEEE evaluation approach
		0.5g)
Point Beach 1&2	2 (0.3g focused-scope)	Seismic PRA (using surrogate element at 0.3g)
Prairie Island 1&2	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Quad Cities 1&2	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
River Bend	1 (reduced-scope)	reduced-scope EPRI SMA
Salem 1&2	2 (0.3g focused-scope)	Seismic PRA
San Onofre 2&3	7 (Seismic PRA)	New Seismic PRA
Seabrook	4 (Seismic PRA)	Existing Seismic PRA
Sequoyah 1&2	3 (0.3g full-scope)	0.3g full-scope EPRI SMA
Shearon Harris 1	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
South Texas Project 1&2	1 (reduced-scope)	Existing Seismic PRA
St. Lucie 1&2	1 (reduced-scope)	Site-specific approach
Summer	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Surry 1&2	2 (0.3g focused-scope)	Seismic PRA (using surrogate element at 0.3g)
Susquehanna 1&2	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
TMI 1	2 (0.3g focused-scope)	Seismic PRA
Turkey Point 3&4	1 (reduced-scope)	Site-specific approach
Vermont Yankee	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Vogtle 1&2	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Waterford 3	1 (reduced-scope)	reduced-scope EPRI SMA
Watts Bar 1	2 (0.3g focused-scope)	0.3g focused-scope EPRI SMA
Wolf Creek	2 (0.3g focused-scope)	reduced-scope EPRI SMA
Number of plants = 71		

* Formerly known as Washington Nuclear Project Number 2.

Table 2.2: Seismic CDF for plants performing a seismic PRA

Plant	Mean seismic CDF			HCLPF (g)	Spectrum shape	Surrogate element
	EPRI or other	Update	LLNL			
Beaver Valley 1	9.10E-06	1.29E-05	2.46E-05	--	1989 EPRI - UHS	No
Beaver Valley 2	5.53E-06	1.03E-05	2.33E-05	--	1989 EPRI - UHS	No
Calvert Cliffs 1&2			1.29E-05	--	LLNL - UHS	Yes
Catawba 1&2	1.60E-05			--	Sequoyah Spectra	No
Columbia*	2.10E-05			--	Site-specific	Yes
D.C. Cook 1&2	3.20E-06		1.00E-05	0.25	1989 LLNL - UHS	No
Diablo Canyon 1&2	4.20E-05			1.56	LTSP Site-specific	No
Haddam Neck	2.30E-04		1.50E-04	< 0.05	1989 EPRI - UHS	Yes
Hope Creek	1.06E-06		3.60E-06	--	1989 EPRI - UHS	No
Indian Point 2	1.30E-05		1.50E-05	--	1989 EPRI - UHS	Yes
Indian Point 3	5.90E-05		4.40E-05	0.13	LLNL - UHS	Yes
Kewaunee	1.10E-05		1.30E-05	0.23	1989 LLNL - UHS	Yes
La Salle 1&2	7.60E-07			--	Not Specified	No
McGuire 1&2	1.10E-05			--	NUREG/CR-0098	No
Millstone 3	9.10E-06			0.25	Site-specific	No
Nine Mile Point 2	2.50E-07		1.20E-06	0.50 (24 hr), 0.23 (72 hr)	NUREG/CR-0098	Yes
Oconee 1,2,&3	3.47E-05			--	1989 EPRI - UHS	Yes
Oyster Creek	3.62E-06	4.74E-06	6.36E-06	--	1989 EPRI - UHS	No
Palisades			8.90E-06	0.22	1993 LLNL - UHS	Yes
Pilgrim 1	5.80E-05		9.40E-05	0.25	1989 LLNL - UHS	Yes
Point Beach 1&2	1.40E-05		1.30E-05	0.16	1989 LLNL - UHS	Yes
Salem 1&2	4.70E-06		9.50E-06	--	1989 EPRI - UHS	No
San Onofre 2&3	1.70E-05			Approx. 0.67	Site-specific	No
Seabrook	1.20E-05		1.30E-04	--	RG 1.60, 0.25g	No

Table 2.2: Seismic CDF for plants performing a seismic PRA (Continued)

Plant	Mean seismic CDF: EPRI or other	Update	LLNL	HCLPF (g)	Spectrum shape	Surrogate element
South Texas Project 1&2	1.90E-07		2.20E-05	--	River Bend Hazard Spectra	No
Surry 1&2	8.20E-06			0.16	1989 EPRI - UHS	Yes
TMI 1	3.21E-05		8.43E-05	--	1989 EPRI - UHS	No
Number of plants = 27						

* Formerly known as Washington Nuclear Project Number 2.

Table 2.3: Dominant risk contributors reported in seismic PRAs

Plant	Seismic failures	Random failures	Operator actions
Beaver Valley 1	125V dc ERFS station battery (0.064 HCLPF), offsite power grid (0.119), auxiliary building (0.223), 125V dc block walls (0.193), containment instrument air (0.126), RW pump intake structure (0.245)	None reported.	RCS depressurization using atmospheric steam dump valves; Station crosstie connecting the 4 kV buses of BV-1 and BV-2.
Beaver Valley 2	125V ERFS station battery (0.064 HCLPF), offsite power grid (0.119), 125V dc station battery 2-5 (0.079), station air compressor 2SAS-C2 (0.126), turbine building block wall (0.126), auxiliary building (0.223), SWS pumps intake structure (0.245).	None reported.	RCS depressurization using atmospheric steam dump valves; Station crosstie connecting the 4 kV buses of BV-1 and BV-2.
Calvert Cliffs 1&2	Surrogate element (leading contributor), offsite power, service water system (leading to failure of 3 EDGs due to loss of cooling), self cooled emergency diesel generators.	None reported.	Alignment of the battery supplied vital 120V ac buses to their backup buses. Recovery from spurious auxiliary feed actuation system by opening the AFW block valves and locally controlling AFW flow. Manually open the steam admission valve that failed closed on loss of power at the 125Vdc bus. Local manual control of AFW flow. In a station blackout with the SG overfilled, drain the AFW steam supply header and start the turbine driven AFW pump.
Catawba 1&2	Offsite power DG battery chargers DG oil tanks ac switchgear Inverters ac and dc panels	Diesel generators	

Table 2.3: **Dominant risk contributors reported in seismic PRAs (Continued)**

Plant	Seismic failures	Random failures	Operator actions
Columbia Generating*	Offsite power, switchgear room cooling, diesel generator controls, surrogate element.	EDG, dc distribution system, HPI, long term heat removal.	Establish suppression pool cooling; Initiate LPCI injection after failure of automatic actuation.
D.C. Cook 1&2	Offsite Power auxiliary building Block Walls 250V dc Panels RPS Panels Ice Condenser (turbine building Pedestal) (4 kV Switchgear) (Cable Trays)	Turbine-Driven AFW Pump	
Diablo Canyon 1&2	Offsite Power 230 kV Transformer Station 4 kV Switchgear (Chatter) DG Control Panel	Diesel Generators Pressurized safety relief valves (SRVs) (Reclose)	Reduce component cooling water (CCW) heat Loads Crosstie Units 1 and 2 Switch Containment Sump Recirculation
Haddam Neck	AFW pipe Main feedwater (MFW) heaters Cont. Air recirc. Fans Battery bank		
Hope Creek	Offsite power, 1E 120V ac instrumentation panels, 1E 125V dc distribution panels, high pressure injection.	Reactor depressurization system, EDG.	Manual recovery action given failure of all four divisions of 1E 120V ac instrumentation distribution panels.
Indian Point 2	turbine building frame Unit 1 superheater stack CCW surge tank fuel storage building Cable trays 480V MCCs service water pumps CCW heat exchangers Intake Structure (sliding) Offsite Power	Emergency diesel generators	

Table 2.3: Dominant risk contributors reported in seismic PRAs (Continued)

Plant	Seismic failures	Random failures	Operator actions
Indian Point 3	Offsite Power, Switchgears for EDG, service water pumps, RHR heat Exahangers, Control Room supervisory panel, RHR pumps, CCW Surge tank and heat exchangers, and Surrogate element.	RHR shutdown cooling, Emergency Diesel Generators.	None reported.
Kewaunee	Offsite Power Surrogate Element		Switch CST to SW for AFW
La Salle 1&2	Offsite Power Condensate Storage tank	Diesel Generators	None
McGuire 1&2	Offsite Power 120V dc	Diesel Generators	Align SW to Pond
Millstone 3	Offsite Power Diesel Generator Oil Coolers (anchor bolts) Roof Diaphragm (Control Bldg) Wall Footing (EDG Bldg) Shear Wall (ESF Bldg) Pumphouse Sliding (Soil)	Diesel Generators AFW system	
Nine Mile Point 2	Surrogate Element Nitrogen Accumulators Offsite Power		
Oconee 1,2,&3	Offsite power, Jocassee Dam (0.15g HCLPF, leading to failure of SSF and other systems), Keowee Dam (0.20 HCLPF, leading to failure of emergency ac power from Keowee Hydro Unit and other systems), SSF components surrogates (0.3), auxiliary building components surrogates (0.24).	Standby Shutdown Facility (SSF).	Align the SSF ASW system for operation.

Table 2.3: Dominant risk contributors reported in seismic PRAs (Continued)

Plant	Seismic failures	Random failures	Operator actions
Oyster Creek	turbine building, Reactor Building, switchgear room fans, Condensate Storage tank, battery room fans, offsite power.	Isolation Condenser makeup, failure of EMRVs to close.	Alignment of fire protection system to isolation condenser makeup, offsite power recovery via combustion turbines.
Palisades	Diesel Fire Pump Day tanks Diesel Fire Pump Control Panel Station transformer Main Steam Isolation Valves (MSIVs), interaction DG Pump Oil tank Bus Undervoltage Relay	Diesel Generators, AFW Pump, Automatic Depressurization Valves (ADVs)	Initiate Once-Through Cooling Initiate AFW Make-up AFW Flow Control
Pilgrim 1	Motor control centers (MCCs) Bus Panels CCW Pumps Residual heat Removal (RHR) Pumps SW Pumps CCW Surge tanks Block Wall Control Rod Structural Failures CSTs, interaction		SBO Diesel Procedure Reset SBO-Related Relay Initiate Suppression Pool Cooling
Point Beach 1&2	Cable Trays Surrogate Element 4 kV Transformers 480V Load Centers Block Walls		Shutdown from Remote Panel Align SW to AFW Suction
Salem 1&2	Offsite power, service water system, battery train failure due to block wall failure, control room ceiling.	Emergency Diesel generators	Establish alternate ESF room ventilation.

Table 2.3: Dominant risk contributors reported in seismic PRAs (Continued)

Plant	Seismic failures	Random failures	Operator actions
San Onofre 2&3	Offsite Power Switchyard Relays (Chatter) 480V Switchgear MCCs auxiliary building Emergency Chillers Emergency Sump Valve Bellows Safety Equipment Building CCW heat exchangers SWC Valve Relays (Chatter) Primary Make-up tank SWC Discharge Gate CST	DG, DG Supply Fans, DG Fuel Transfer Pumps, Turbine-Driven AFW Pump, Battery Chargers, Motor-Driven AFW Pumps, Emergency Chillers, CCW heat exchangers, CCW Non-Crit Loop Isolation Valves, CCW Pumps, HPSI Pumps	Condensate Make-up T-D AFW Pump Control Valves Reset Relays
Seabrook	Offsite Power, 4.16 kV Switchgear (chatter), RWST, and EDG.	Emergency Diesel Generators (EDG).	Reset relays.
South Texas Project 1&2	Offsite Power Diesel fuel oil day tanks 4.16 kV Switchgear Large Chiller-tanks CCW surge tank AFW Storage tank Electrical Cabinets (inverters, chargers)		
Surry 1&2	Offsite power, turbine building, Condensate Storage tank, CCW surge tank, seismic induced lube oil fire in turbine building.	Emergency Diesel Generators.	Prevent intake canal draining; stop AFW pumps before cavitation; align RHR inside containment; depressurize RCS using steam dump valves.
TMI 1	Offsite power, Class 1E ac power, control room ceiling, emergency feedwater, EDG air start receiver.	Class 1E ac power train A.	Relay chatter recovery.
Number of plants = 27			

* Formerly known as Washington Nuclear Project Number 2.

Table 2.4: Seismic outliers and improvements for PRA plants

Plant	Walkdown screen level	Type(s) of findings	Description of findings	Plant improvements
Beaver Valley 1	0.5g HCLPF	Potential plant improvement	Potential for reinforcement of the block walls in the Emergency 125V dc Battery Room.	None. Improvement not implemented due to the low seismic CDF and because the block walls conform to resolution of both USI A-46 and IEB 80-11.
Beaver Valley 2	0.5g HCLPF	Potential plant improvement	Failure of the diesel building due to interaction with the emergency diesel generator.	None. No improvement on the diesel building will be made because its HCLPF of 0.28g is more than twice the SSE level along with a low contribution to total CDF.
Calvert Cliffs 1&2	0.3g HCLPF	None, except for one issue addressing a seismically induced fire scenario (smoke from a burning fuel oil tank).	No fundamental weakness or vulnerability found.	The issue of smoke from the fire being drawn into the control room ventilation system is addressed in one of the improvements for fire concerns (changes in operator procedures for the ventilation system).
Catawba 1&2	Structures: 2.5g median Equipment: 2.0g median	Outliers: Reactor Building Containment Internal Structures Anomalies: Minor spatial interaction concerns.	- Reactor Building and Containment Internal Structures could not be screened out. - Walkdown identified minor spatial interaction concerns.	Fixes were made to three minor spatial interaction concerns, and were deemed not be risk significant. - DG battery rack modifications. - Instrument relocated. - Valve replaced (Table 3-3 of IPEEE)
Columbia Generating*	0.5g HCLPF generic screening threshold used.	Items where plant improvements could be made.	Issues/outliers identified: missing anchorage nuts or washers in two air handling units in the Division 1 diesel generator room, inadequate connection between cabinets of E-SM-7 and E-SM-7/75/2, proximity of the hangers of three MCCs and two instrument racks, lack of restraint of the batteries for the diesel driven fire pumps, weak MCC base connections, lack of alternate switchgear room cooling.	Minor corrections, including replacing missing anchorage nuts or washers to design anchorage configuration, proper connection between cabinets and tie-down of batteries, the strengthening of MCC base connections, and procedures (to open door) for alternate switchgear room cooling.

Table 2.4: Seismic outliers and improvements for PRA plants (Continued)

Plant	Walkdown screen level	Type(s) of findings	Description of findings	Plant improvements
D.C. Cook 1&2	Not reported	Several anomalies and housekeeping concerns	Block walls; poor fire-extinguisher mountings; interaction with fire protection pilot lines; interaction of fluorescent lights in control room; missing/broken anchorages on some MCCs; questionable support of a 17-ton CO_2 tank; potential for earthquake-induced hydrazine spill.	Three design-related improvements (mounting/support of instrument rack, Halon bottles, and emergency service water (ESW) piping) and 13 housekeeping-related fixes (replacing or tightening nuts/bolts or clamps, rust protection, etc.) were made.
Diablo Canyon 1&2	Not reported	None.	None.	No IPEEE plant improvements. Earlier programs: - LTSP improvements - DC PRA-based improvements - Ongoing improvements.
Haddam Neck	0.3g HCLPF, 0.8 spectral acceleration	Identified vulnerabilities, including 30 seismic and 8 seismic-fire risk outliers	Numerous conditions, including poor anchorage/support, interaction concerns, housekeeping concerns, and relay chatter.	Numerous meaningful plant improvements have been proposed (Table 7.1-1 of IPEEE submittal).
Hope Creek	0.5g HCLPF or 1.5g median	A number of components could not be screened out (Table 3-4, mostly of interaction and anchorage-support concerns). About 100 LRRs were identified and later screened.	No fundamental weakness or vulnerability was found (Screened-in components were evaluated for seismic fragilities, table 3-5).	None.
Indian Point 2	0.5g PGA HCLPF, 1.5g PGA Median	Unscreened include: - 20 item categories identified from earlier SPRA, - 11 item categories from USI A-46, - 17 additional item categories from IPEEE walkdowns	Fragilities were calculated for unscreened components; 15 components were ultimately included in quantifying the seismic PRA model.	component cooling water Surge tank Anchor Bolts were strengthened.
Indian Point 3	0.8g peak spectral acceleration level, or 0.38g HCLPF	"Seismic vulnerabilities"	No unique plant vulnerabilities were found. However, the submittal does refer to a "seismic vulnerability" in which a seismic event may induce a spurious operation of the EDG room CO_2 system and subsequent shutdown of the EDG ventilation system. This has been addressed with a temporary modification.	Installation of new actuation control panel for CO_2 system suggested. The SRT identified several "seismic vulnerabilities" regarding seismically induced fire, but there was no discussion regarding their resolution.

Table 2.4: Seismic outliers and improvements for PRA plants (Continued)

Plant	Walkdown screen level	Type(s) of findings	Description of findings	Plant improvements
Kewaunee	0.3g HCLPF; 0.8g spectral acceleration	Open issues/anomalies; no vulnerabilities or risk-significant concerns.	Missing fasteners on DG excitation and control cabinets; poor anchorage of station service transformers; potential interaction of relay racks; other equipment anchorages concerns; mercoid switches	Resolution of USI A-46 concerns has led to some equipment enhancements, one procedural implementation, an administrative control, and several housekeeping improvements. (Submittal Table 3-4)
La Salle 1&2	Not reported	Outlier, anomalies not reported.	CST was found to be an outlier.	None, but the submittal notes that some plant improvements have been made since 1985.
McGuire 1&2	Structures: 2.5g median Equipment: 2.0g median	Anomalies: Minor spatial interaction concerns and maintenance concerns.	Walkdown identified 6 spatial interaction concerns, two equipment mounting/support concerns, and one maintenance concern.	- Spacers installed on DG batteries/racks. - Grating trimmed near steam vent valves. - MCCs bolted together. - Guidelines developed for movable equipment. - Panel modified to clear 8-in pipe. - Arc barriers tightened in main control boards. - Grout installed below saddle support of CCW heat exchanger. - Missing bolts installed on surge tank. - Corrosion on anchor bolts of AFW/CST cleaned and bolts recoated (Table 3-3 of IPEEE).
Millstone 3	Not reported	Outliers; anomalies not reported.	Diesel generator oil cooler bolts identified as an outlier.	None for IPEEE; however, diesel generator oil cooler bolts were previously replaced with stronger bolts.
Nine Mile Point 2	0.5g HCLPF, 1.2g spectral acceleration	Anomalies/open issues	Three concerns were cited: potential for overhead rack to impact an MOV; potential interaction of hoist assemblies mounted on electric cabinets; and fire water piping in control building.	Rack over a motor operated valve (MOV) was secured; rail stops were installed to prevent movement of hoist assemblies on electrical cabinets.

Table 2.4: Seismic outliers and improvements for PRA plants (Continued)

Plant	Walkdown screen level	Type(s) of findings	Description of findings	Plant improvements
Oconee 1,2,&3	0.3g HCLPF	Recommended enhancements	While seismic events are the most significant external event contributors to core damage risk, "there are no unduly significant sequences (vulnerabilities) from external events."	Plant improvements listed in Table 6-1, including 142 low-ruggedness relays (LRRs) & issues related to anchorage, support and restraint. Plans are to complete resolution of all outliers, including the relays identified in Table 3-2 of the 1997 Supplemental Report, by the end of 2002. It is the licensee's intention to assure the final fragilities for these relays to be at or above PRA modeled values by testing, analysis, or replacement modifications.
Oyster Creek	0.3g HCLPF.	Potential plant modifications	The seismic IPEEE identified no plant vulnerabilities (i.e., a failure will result in a CDF of 1.0E-6/ry). Liquefaction-induced failures were identified as most risk significant contributors.	Two potential plant modifications: check for tightness of bolts on Forked River Combustion Turbine tin-fan coolers, and provide battery spacers for Combustion Turbine battery compartments.
Palisades	0.3g HCLPF and 0.5g HCLPF	Outliers and anomalies.	Fifty-two (52) conditions were encountered, including instances of poor anchorage, unqualified (and unanalyzed) block walls, and interaction concerns.	None.
Pilgrim 1	1.0g median (EPRI NP-6041-SL SMA Column 2 screening criteria)	Outliers and anomalies.	Various concerns were identified in USI A-46/IPEEE evaluations, but not fully reported in IPEEE submittal.	Stiffening of SBO diesel muffler support; fix a seismic interaction hazard due to potential failure of a main transformer bushing and adjacent lightning arrester, and fix potential weakness of friction-clip restraints connecting A8 bus to its concrete foundation.
Point Beach 1&2	0.3g HCLPF and 0.5g HCLPF	Outliers	Various concerns were identified in USI A-46 evaluation, and apply to IPEEE. Weaknesses in RWST and CST encountered (and modeled in SPRA fragility analyses).	Fix anchorage deficiencies on cable trays and some equipment (for USI A-46); resolve concerns associated with Westinghouse Model ITH relays (for USI A-46); and add two diesel generators and their support systems (for IPE).
Salem 1&2	A median capacity of 1.5g and an HCLPF of 0.5g were used as part of the screening criteria.	Plant improvements	No definition of vulnerability was found. The submittal does state that as a result of the seismic PRA analysis, no vulnerabilities have been identified. However, a few plant improvements were assumed and credited in the risk quantification.	Replacement of low-ruggedness relays (LRRs) and reinforcement of block walls in switchgear room. Procedural change to ensure long term alternate ventilation for rooms in the auxiliary building.

Table 2.4: Seismic outliers and improvements for PRA plants (Continued)

Plant	Walkdown screen level	Type(s) of findings	Description of findings	Plant improvements
San Onofre 2&3	Components with median capacity greater than 10g S sub a (1-10 Hz) were screened out; components with median capacity between 8-10g S sub a (1-10 Hz) were screened out if their seismically induced failure rate was less than the 10E-7, and components with median capacity up to 8g S sub a (1-10 Hz) were included in the SPRA quantification process.	Approximately 30 anomalous conditions were observed, most of which were resolved through additional consideration.	Issues potentially affecting functionality (3 items). Anchorage anomalies (3 items). Load path anomalies (9 items). Seismic II/I interaction concerns (9 items). Commodity clearance concerns (6 items). Fragilities were computed for over 150 components.	- Improved reliability of cross-connecting EDGs between units. - Strengthened supports of an ammonia tank to eliminate a spill hazard. - Removed a floor grating surrounding AFW valve actuators to eliminate an interaction hazard. - Removed a concrete plug surrounding the Unit 2 DG fuel oil transfer piping to improve the seismic capacity of the pipe and to provide a consistent configuration among units. - Fastened together adjacent electrical cabinets/panels to help prevent interactions and relay chatter. - Stabilized light fixtures that may interact with electrical cabinets.
Seabrook	A median capacity of 2.0g PGA	None.	There are no fundamental weaknesses or vulnerabilities with regard to severe accident risk.	None.
South Texas Project 1&2	Not reported	Not reported	Not reported	None reported

Table 2.4: Seismic outliers and improvements for PRA plants (Continued)

Plant	Walkdown screen level	Type(s) of findings	Description of findings	Plant improvements
Surry 1&2	0.3g HCLPF	Outliers and low-ruggedness components	Issues/outliers identified include loose or missing fasteners, anchorage concerns, outliers for cable and conduit raceways, issues related to seismic induced fire/flood evaluations, housekeeping/conduct of maintenance issues, control room ceiling review, and items related to electrical and mechanical equipment.	Issues/outliers involving loose or missing fasteners, anchorage concerns, and outliers for cable and conduit raceways have been mostly resolved. About 60 items related to electrical and mechanical equipment have been resolved via field modifications. An operating procedure, 0-AP-12.01, was revised to require opening of the condenser waterbox vacuum breakers to conserve intake canal inventory. Issues/outliers on seismic induced fire/flood evaluations, housekeeping/conduct of maintenance, control room ceiling, a few minor modifications for cable tray and conduit supports, and further evaluation and enhancement for 36 mechanical and electrical components indicated in Table 6.1-1 will be resolved by the end of the refueling outage currently scheduled to commence in September 2000.
TMI 1	1.0g medium	Low-ruggedness relays and other issues/outliers for improvements (mostly related to anchorage/support/ restraint).	Identified issues/outliers including: some relays not being able to pass any seismic screening criteria; EDG air receivers seismic restraint; control room ceiling; restraining of the penetration pressurization tank PP-T-1A; Load Centers 1P, 1R, 1S, and 1T gusset weld; supports for the fuel oil tanks and batteries for the diesel-driven fire pumps; anchorage for the decay heat service heat exchangers.	Replacement of LRRs, Control Room ceiling modification, and fixing anchorage/restraint problems. Due to small contribution to CDF, improvements not planned for rewelding the load center gusset welds, upgrading the supports for the fuel oil tanks and batteries for the diesel-driven fire pumps, and upgrading decay heat service heat exchangers.

* Formerly known as Washington Nuclear Project Number 2.

Table 2.5: Seismic containment performance for PRA plants

Plant	Containment type	Quantitative findings	Qualitative findings	Plant improvements
Beaver Valley 1	Subatmospheric containment; steel-lined reinforced-concrete	Level 2 analysis, releases dominated by small release from early containment failure and bypass (88% of CDF).	Seismic-induced failures do not play a significant role in the results.	None.
Beaver Valley 2	Subatmospheric containment; steel-lined reinforced-concrete	Level 2 analysis, releases dominated by small release from early containment failure and bypass (58% of CDF).	Seismic-induced failures do not play a significant role in the results.	None.
Calvert Cliffs 1&2	Large dry type; steel-lined prestressed-concrete	Included containment isolation failure (due to seismic event) in the event tree. Containment structural failure was screened out. CDF for early large failure is 1.41E-6.	Containment penetrations and the isolation valves bound all other failures and were screened at 0.5g HCLPF.	None.
Catawba 1&2	Ice Condenser, Pressure Suppression Type, with Steel Primary Containment and a Reinforced Concrete Shield Building.	None.	- Reactor building, containment internal structures did not screen out. - Cabinets, panel boards, and MCCs (for containment isolation system) did not screen out. - No fragility analysis for ice condenser.	None.
Columbia Generating*	Mark II	None.	A walkdown was conducted to ensure containment performance function and the ability to isolate containment. No unique containment vulnerabilities were identified.	None.
D.C. Cook 1&2	Ice Condenser; steel-lined reinforced-concrete	Direct/structural containment failure leads to 1% of seismic CDF; frequencies of early releases otherwise not quantified.	No significant anomalies were cited.	None.
Diablo Canyon 1&2	Large dry type; steel-lined reinforced-concrete	Large early release frequency: 3% of seismic CDF. Small early release frequency: 16% of seismic CDF.	None.	No related IPEEE improvements.

Table 2.5: Seismic containment performance for PRA plants (Continued)

Plant	Containment type	Quantitative findings	Qualitative findings	Plant improvements
Haddam Neck	Large dry type; steel-lined reinforced-concrete	None.	Vulnerabilities found in Adams filter units and CAR fans, diesel fire pump batteries and diesel fire pump fuel oil tank, and exhaust penetration P39 (CP system).	Licensee's resolution to these items is unclear.
Hope Creek	Mark I; steel containment	Failure of instrument distribution panels 1A(B,C,D)J482 may lead to core damage and containment isolation failure (about 5% of total CDF).	Containment structural integrity, penetrations, and associated isolation valves, cables, etc., were screened out.	None.
Indian Point 2	Large dry type; steel-lined reinforced-concrete	About 65% of the seismic CDF results in plant damage states with initial loss of containment pressure suppression and heat removal functions. If these functions are not regained, long-term over pressure failure of the containment could result. None of these sequences leads directly to early containment failure or bypass.	Containment fan coolers did not screen out and were included in the SPRA model.	None.
Indian Point 3	Large dry type; steel-lined reinforced-concrete	Containment event tree was developed for containment failure frequencies (figure 3.1.6.1). Summary information was not provided. Results were said to be similar to those derived for IPE study.	A walkdown was performed to identify vulnerabilities that could result in early containment failure, and none were found.	None.
Kewaunee	Large dry type; free-standing steel containment vessel, surrounded by a reinforced-concrete shield building, with an annular space in between the two	Mean frequency of containment failure: 6.2E-6/ry (EPRI hazard). Containment HCLPF (large early failures): 0.3g.	No significant anomalies were cited.	None.

Table 2.5: Seismic containment performance for PRA plants (Continued)

Plant	Containment type	Quantitative findings	Qualitative findings	Plant improvements
La Salle 1&2	Mark II, with inerted, primary containment of post-tensioned reinforced concrete with steel liner; secondary containment is the reinforced concrete reactor building.	None reported.	None reported.	None.
McGuire 1&2	Ice Condenser; steel-lined reinforced-concrete	None.	None. (No fragility analysis was performed for ice condenser.)	None.
Millstone 3	Subatmospheric containment; steel-lined reinforced-concrete	Not reported.	Containment recirculating system heat exchangers are outliers.	None.
Nine Mile Point 2	Mark II; reinforced-concrete	Crediting operator actions to close valves outside containment, less than 2% of the CDF is associated with early containment failure or bypass.	No significant anomalies were cited.	None.
Oconee 1,2,&3	Large dry type; steel-lined prestressed-concrete	None.	Walkdown on containment performance was conducted. Equipment and structures required for containment performance and potential failure modes and consequence have been examined.	None.
Oyster Creek	Mark I; steel containment	None.	Seismic structural capacity of the drywell and the performance of containment isolation following a seismic event were evaluated and were found to be higher than 1.0g. Containment bypass was considered based on important bypass sequences from internal event IPE and was found to be of no concern.	None.
Palisades	Large dry type; pre-stressed, post-tensioned reinforced con-crete structure lined with a 1/4-inch carbon steel layer.	Dominant seismic containment failure mode was found to be relocation of core debris to the auxiliary building, having a frequency of 2.3E-6/ry.	No significant anomalies were cited.	None.

Table 2.5: Seismic containment performance for PRA plants (Continued)

Plant	Containment type	Quantitative findings	Qualitative findings	Plant improvements
Pilgrim 1	Mark I; pressure suppression type, steel containment	Frequency of early containment failure: 1.6E-5/ry (EPRI mean hazard), 3.2E-5/ry (1993 LLNL mean hazard).	No discussion provided.	None.
Point Beach 1&2	Large dry type, with pre-stressed, post-tensioned reinforced concrete structure.	Frequency of early large release: 1.3E-5/ry (the submittal relies on manual containment isolation to reduce this by a factor of 10.)	None, although the quantitative analysis indicates that the automatic containment isolation function has low seismic capability.	None.
Salem 1&2	Large dry type; steel-lined reinforced-concrete	None.	Walkdowns and capacity calculations were performed and no vulnerabilities were found regarding any aspect of containment performance.	None.
San Onofre 2&3	Large dry type, pre-stressed reinforced concrete.	- Success, no containment failure within 48 hours, less than 0.1% volatiles released: 9.1E-6/ry (53% of seismic CDF). - Late containment failure, release up to 0.1% volatiles: 7.5E-6/ry (43% of seismic CDF). - Containment bypassed, less than 0.1% volatiles released: 2.6E-7/ry (1.5% of seismic CDF). - Late containment failure, more than 10% volatiles released: 2.4E-8/ry (0.2% of seismic CDF). - Early/isolation failure, prior to or at time of vessel failure, up to 10% volatiles released: 3.9E-7/ry (2.3% of seismic CDF).	No significant anomalies pertaining to early containment failure or unique conditions for seismic events were cited.	None.
Seabrook	Large dry type; steel-lined reinforced-concrete	None documented.	No significant anomalies were cited.	None. (However, relevant IPE improvements have been cited.)

Table 2.5: Seismic containment performance for PRA plants (Continued)

Plant	Containment type	Quantitative findings	Qualitative findings	Plant improvements
South Texas Project 1&2	Large dry type, with a steel-lined, post-tensioned reinforced-concrete structure.	None.	None reported.	None, although earlier (as part of the Level-2 PSA effort) enhancements were made to selected containment isolation valves.
Surry 1&2	Subatmospheric containment; steel-lined reinforced-concrete	None.	Containment integrity, containment isolation, and containment cooling systems were examined and no concerns were noted with respect to containment performance.	.None.
TMI 1	Large dry type; steel-lined prestressed-concrete	None.	Evaluations were performed for containment structure seismic capacity (with the lowest median capacity estimated at greater than 11 g) and the fragility of containment isolation valves and signals (with ESAS relays with a median fragility of 0.89g the weakest component).	None.

* Formerly known as Washington Nuclear Project Number 2.

Table 2.6: Plant capacity results from IPEEE SMA

Plant	Format of SMA	HCLPF (g)	Spectral shape
Arkansas Nuclear One 1	0.3g full-scope EPRI SMA	0.3	NUREG/CR-0098, Rock
Arkansas Nuclear One 2	0.3g focused-scope EPRI SMA	0.3	NUREG/CR-0098, Rock
Braidwood 1&2	0.3g focused-scope EPRI SMA	0.3	NUREG/CR-0098, Rock
Browns Ferry 2&3	0.3g focused-scope EPRI SMA	0.26	NUREG/CR-0098, Rock
Brunswick 1&2	0.3g focused-scope EPRI SMA	0.30	NUREG/CR-0098, Soil
Byron 1&2	0.3g focused-scope EPRI SMA	0.3	NUREG/CR-0098, Rock
Callaway	0.3g focused-scope EPRI SMA	0.30	NUREG/CR-0098, Soil
Clinton	0.3g focused-scope EPRI SMA	0.3	Multiple Analysis Method (MAM), Soil
Comanche Peak 1&2	reduced-scope EPRI SMA	--	Plant SSE (RG 1.60), 0.12g, Rock
Cooper	0.3g focused-scope EPRI SMA	0.3	NUREG/CR-0098, Soil
Crystal River 3	reduced-scope EPRI SMA	--	Housner, Soil/Marshland
Davis-Besse	reduced-scope EPRI SMA	0.26	NUREG/CR-0098, Rock
Dresden 2&3	0.3g focused-scope EPRI SMA	0.2	NUREG/CR-0098, Rock
Duane Arnold	reduced-scope EPRI SMA	--	DBE Spectra, Rock/Soil
Farley 1&2	reduced-scope EPRI SMA	--	Plant SSE (NUREG/CR-0098), Soil
Fermi 2	0.3g focused-scope EPRI SMA	0.3	NUREG/CR-0098, Rock
FitzPatrick	0.3g focused-scope NRC SMA	0.22	NUREG/CR-0098, Rock
Fort Calhoun 1	0.3g focused-scope NRC SMA	0.25	NUREG/CR-0098, Soil
Ginna	0.3g focused-scope EPRI SMA	0.2	RG 1.60, 0.25g, Rock
Grand Gulf 1	reduced-scope EPRI SMA	--	Design Basis Response Spectra, Soil
H.B. Robinson 2	0.3g full-scope EPRI SMA	0.28	NUREG/CR-0098, Soil
Hatch 1&2	0.3g focused-scope EPRI SMA	0.30	NUREG/CR-0098, Soil
Limerick 1&2	reduced-scope EPRI SMA	0.15	Plant SSE (Newmark), 0.15g, Rock
Millstone 2	0.3g focused-scope EPRI SMA	0.25	NUREG/CR-0098, Rock
Monticello	Modified focused/Expanded reduced-scope EPRI SMA	0.12	NUREG/CR-0098 (for screening); Plant SSE (Housner, 0.12g) (for evaluation), Soil
Nine Mile Point 1	0.3g focused-scope EPRI SMA	0.27	NUREG/CR-0098, Rock
Nine Mile Point 2	SPRA & focused EPRI SMA	0.23	NUREG/CR-0098, Rock
North Anna 1&2	0.3g focused-scope EPRI SMA	0.16	NUREG/CR-0098, Rock/Soil
Palo Verde 1,2,&3	0.3g full-scope EPRI SMA	0.3	NUREG/CR-0098, Soil
Peach Bottom 2&3	Modified focused-scope EPRI SMA	0.2	NUREG/CR-0098, Rock
Perry 1	0.3g focused-scope EPRI SMA	0.30	NUREG/CR-0098, Rock/Class A backfill
Prairie Island 1&2	0.3g focused-scope EPRI SMA	0.28	NUREG/CR-0098, Soil

Table 2.6: Plant capacity results from IPEEE SMA (Continued)

Plant	Format of SMA	HCLPF (g)	Spectral shape
Quad Cities 1&2	0.3g focused-scope EPRI SMA	0.09 original/0.24 planned	NUREG/CR-0098, Rock
River Bend	reduced-scope EPRI SMA	--	Design Ground Response Spectra, Soil
Sequoyah 1&2	0.3g full-scope EPRI SMA	0.27	NUREG/CR-0098
Shearon Harris 1	0.3g focused-scope EPRI SMA	0.29	NUREG/CR-0098, Rock
St. Lucie 1&2	Site-specific approach	--	Plant SSE, 0.10g, Structural Fill
Summer	0.3g focused-scope EPRI SMA	0.22	NUREG/CR-0098, Rock/Soil
Susquehanna 1&2	0.3g focused-scope EPRI SMA	0.21	NUREG/CR-0098, Rock/Soil
Turkey Point 3&4	Site-specific approach	--	Plant SSE (Housner), 0.15g, Rock
Vermont Yankee	0.3g focused-scope EPRI SMA	0.25	NUREG/CR-0098, Rock
Vogtle 1&2	0.3g focused-scope EPRI SMA	0.3	NUREG/CR-0098, Soil
Waterford 3	reduced-scope EPRI SMA	--	Design Basis Response Spectra, Soil
Watts Bar 1	0.3g focused-scope EPRI SMA	0.3	NUREG/CR-0098, Rock/Soil
Wolf Creek	reduced-scope EPRI SMA	0.2	NUREG/CR-0098 Median, Rock (for screening); Plant SSE (for evaluation)
Number of plants = 45			

Table 2.7: Seismic outliers and improvements for SMA plants

Plant	Walkdown screen	Anomalies & outliers	Controlling outliers	HCLPF capacities (g)	Plant improvements
Arkansas Nuclear One 1	0.3g HCLPF	A significant number of outliers were identified and a list of "opportunities for plant improvement" involving spatial interaction and inadequate anchorage is provided.	Anchorage for the Emergency Diesel Generator Fuel tanks (0.2g, shear of bolt). Outliers associated with the Cable Tray & Conduit Raceway and Relay review portions are being tracked for resolution as part of the USI A-46 Program for ANO-1.	The licensee indicated that the 0.3g screening criteria would be met.	An A-46 plant. The IPEEE program identified some IPEEE-only improvements. "Opportunities for plant improvement" identified to resolve 10 spatial interactions and anchorage concerns (see Table 7-1 of IPEEE submittal). The method of resolution and schedule for implementation is not specifically identified. The licensee indicated that the 0.3g screening criteria would be met.
Arkansas Nuclear One 2	0.3g HCLPF	A significant number of outliers were identified and a list of "opportunities for plant improvement" involving spatial interaction and inadequate anchorage is provided.	Anchorage for the Emergency Diesel Generator Fuel tanks (0.2g, shear of bolt). Outliers associated with the Cable Tray & Conduit Raceway and Relay review portions are being tracked for resolution as part of the USI A-46 Program for ANO-2.	The licensee indicated that the 0.3g screening criteria would be met.	An A-46 plant. The IPEEE program identified some IPEEE-only improvements. "Opportunities for plant improvement" identified to resolve 10 spatial interactions and anchorage concerns (See Table 7-1 of IPEEE submittal). The method of resolution and schedule for implementation is not specifically identified. The licensee indicated that the 0.3g screening criteria would be met.
Braidwood 1&2	0.3g HCLPF	Outliers that are largely interaction concerns.	Control room ceiling diffusers (made of aluminum; if dislodged by a seismic event, may pose a personnel hazard); seismic interaction between closely spaced electrical cabinets which contain essential relays.	> 0.3g	Equipment maintenance and modifications (e.g., secure control room ceiling diffusers to T-bars, as appropriate).
Browns Ferry 2&3	0.3g HCLPF	Issues related to maintenance, housekeeping, and seismic interaction that required work orders to satisfy SRT field issues; items requiring repairs or modifications.	Two transformers in the diesel generator building for Units 1 and 2 (which will eventually be replaced as part of the long-term asbestos material removal program at BFN). Two Valve Operators exceeding the GIP Limit for height and weight.	0.26g. Any unresolved outliers as a result of the USI A-46 program will be modified; thus, the resulting configurations will have HCLPF capacities much greater than 0.3g.	An A-46 plant. Design modifications made to resolve two valve operators. Any unresolved outliers as a result of the USI A-46 program will be modified for the A-46 program.

Table 2.7: Seismic outliers and improvements for SMA plants (Continued)

Plant	Walkdown screen	Anomalies & outliers	Controlling outliers	HCLPF capacities (g)	Plant improvements
Brunswick 1&2	0.3g HCLPF	Joint USI A-46/IPEEE seismic evaluation effort has identified a number of housekeeping, maintenance, and interaction concerns, and equipment outliers.	Several outliers were identified, but none had calculated HCLPF capacities less than 0.3g.	All outliers have HCLPF capacities exceeding 0.3g.	Being made under USI A-46 resolution. Note: the IPEEE findings assume USI A-46 improvements (which are still to be resolved).
Byron 1&2	0.3g HCLPF	Outliers that are largely interaction concerns.	Control room ceiling diffusers (made of aluminum; if dislodged by seismic event, may pose a personnel hazard); miscellaneous interactions (with lights, bins, carts, etc.) for motor control centers (MCCs), switchgears, batteries, and inverters; valves with inadequate clearance; loose internal "shipping" bolts on transformers; Unanchored heat trace cabinet located in the vicinity of MCC; some MCCs, instrument and control cabinets, battery chargers, and breakers were not tied together posing an impact issue.	> 0.3g	Equipment maintenance or modifications. For example: the control room ceiling diffusers were secured to T-bars as appropriate; maintenance on the valves with inadequate clearance was requested; loose internal "shipping" bolts on transformers were tightened; anchored heat tracing cabinet was welded to foundation pad; MCCs, I&C cabinets, battery chargers, and breakers that were not tied together were tied together.
Callaway	0.3g HCLPF	21 anomalies/open issues were identified; some outliers were identified.	Outliers had calculated HCLPF capacities exceeding 0.3g.	All outliers have HCLPF capacities exceeding 0.3g.	- Remounted hand-held extinguishers. - Trimmed floor grating. - MCCs bolted to walls. - Missing shear pins installed on AFW pump. - Procedures and signs for storage of transient equipment. - Procedure for securing chain hoists.
Clinton	0.3g HCLPF	None.	None.	0.3g	No plant improvements are needed.
Comanche Peak 1&2	SSE	Some minor anomalies and maintenance concerns were identified.	No SSE outliers were identified.	Not applicable.	Follow-up actions to resolve: - Unanchored non-plant equipment near safety equipment in control room. - Insufficient clearance between an MCC and cable tray support.

Table 2.7: Seismic outliers and improvements for SMA plants (Continued)

Plant	Walkdown screen	Anomalies & outliers	Controlling outliers	HCLPF capacities (g)	Plant improvements
Cooper	0.3g HCLPF	Several outliers (mostly interaction issues) identified in combined A-46/IPEEE walkdown; some items with HCLPF less than 0.3g	For IPEEE: The vibration-isolated air handling systems (with an estimated HCLPF capacity of 0.21g, removed from SSEL after additional system analysis); Four seismic vulnerabilities in the fire suppression systems two electric drive pumps depend on offsite power, the fuel oil tank of the diesel driven pump have low seismic capacity, all pumps are housed in a block wall structure, and the water storage tanks are flat bottom tanks supported on a ring foundation).	At least 0.3g	A-46 plant. Most issues resolved under A-46. Upgrade of "seismic vulnerabilities" in fire suppression system two electric drive pumps, one diesel driven pump. Water storage tanks).
Crystal River 3	SSE	No additional outliers other than the existing USI A-46 outliers	None from IPEEE.	Not applicable.	All outliers were resolved and no additional improvements were made beyond the USI A-46 program.
Davis-Besse	0.2g	No plant vulnerabilities beyond the findings of the A-46 program were identified in the IPEEE.	Based on HCLPF calculated for a limited number of items: masonry walls near 480V essential MCC (0.26g); BWST (0.28g).	0.26g	A-46 plant. Outliers have been identified for resolution under the A-46 program. In addition, restraint of two flammable compressed gas bottles on auxiliary building.
Dresden 2&3	0.3g HCLPF	Besides A-46 outliers there are no significant concerns identified as a result of the seismic margins assessment.	Buckling of condensate storage tank (0.20g); Diesel Fuel Oil Storage Day tank (0.26g controlled by adjacent masonry wall); Torus Suppression Chambers (0.28g controlled by torus shell stress); and about 20 electrical equipment anchorage capacities are also listed between 0.20g and 0.30g PGA.	0.20g	An A-46 plant. No plan to make improvements beyond those required for resolution of A-46; no plan to improve items that meet or exceed the design basis requirement of 0.20g PGA (and thereby meet Dresden's intention to ensure that all IPEEE components have a seismic capacity that complies with design requirements).

Table 2.7: Seismic outliers and improvements for SMA plants (Continued)

Plant	Walkdown screen	Anomalies & outliers	Controlling outliers	HCLPF capacities (g)	Plant improvements
Duane Arnold	DBE	109 outliers were identified for combined A-46 and IPEEE	Seismic qualification of a masonry wall; potential fall of control room ceiling elements onto critical equipment; anchorage adequacy; two air handlers in the HPCI room which were identified as flood/spray outliers; and gas storage bottles that were inadequately restrained.	Not applicable.	Outliers that could not be resolved by calculation, e.g., anchorage outliers in the A-46 evaluation report, were resolved by plant maintenance action or modification. The concerns regarding the three gas storage bottles were resolved by providing adequate restraint or removing the bottles from the area.
Farley 1&2	SSE	A total of 117 outliers were identified in A-46/IPEEE	List of outliers is in Appendix A. Outliers include LRRs and anchor bolts.	Not applicable.	Actions of resolution involve: - installing restraining wires for overhead lights; - replacing anchor bolts; - bolting panels to walls or bolting cabinets together; - installing missing screws; and - performing additional detailed analysis.
Fermi 2	0.3g HCLPF	Minor hardware deficiencies (largely associated with maintenance activities)	The plant was found to be seismically rugged in that upon completion of the few plant modifications and corrective maintenance activities discussed below, all structures, systems, and components required for the two identified safe shutdown paths met the seismic capacity requirements of the 0.3g RLE.	0.3g after completion of a few plant modifications.	- Improved maintenance training to minimize minor hardware deficiencies; - improved operator training and simulator training to handle seismic-induced scenarios; - replacement of four (4) low-ruggedness relays; - bolting relay panels together to reduce chatter probability; - strengthening of seismic support for an air dryer; and - two (2) instrument panels.
FitzPatrick	0.3g HCLPF	Seismic-induced structure failures	The failure of emergency diesel generator building and electric bay block walls (0.17g); the failure of the containment atmosphere dilution (CAD) building with a HCLPF of 0.22g.; failure of the hydrogen line in the turbine building was identified.	0.17g	A-46 plant. Strengthen block walls in EDG building; and close hydrogen line in the event of earthquake (procedure).

Table 2.7: Seismic outliers and improvements for SMA plants (Continued)

Plant	Walkdown screen	Anomalies & outliers	Controlling outliers	HCLPF capacities (g)	Plant improvements
Fort Calhoun 1	0.3g HCLPF	Several outliers were identified.	Relays MCCs (anchorage) Service building Fire pumps Turbine building. DG air receiver RWS heat exchangers CST MCCs (anchorage) Liquefaction Transformer MCCs (anchorage) RWS pump	0.01g 0.05g 0.10g 0.10g 0.10g 0.17g 0.17g 0.17g 0.24g 0.25g 0.25g 0.27g 0.29g	Replacement of bad actor relays; improvement of MCC anchorages; raw water system tie-in to the emergency feedwater storage tank (EFWST); and others
Ginna	0.3g HCLPF	52 items of equipment could not be screened out; approximately 90 items of equipment were identified as being vulnerable to block walls; the Reactor Makeup Water tank and the Monitor tank, if failed, can cause the interruption of one or more of the systems selected for the second success path (for small LOCA).	All components meet their existing licensing basis.	0.2g.	None. No further work will be performed by RG&E with respect to seismic issues outside of those related to USI A-46 closeout. Under various programs (e.g., SEP), RG&E has conducted extensive reevaluations of, and made upgrades to, Ginna's structures, systems, and equipment, using a 0.2g Regulatory Guide (RG) 1.60 spectrum as seismic input.
Grand Gulf 1	SSE	One potential vulnerability to a seismic event was identified and corrected.	The grouted condition for the penetration of the Standby service water (SSW) piping in the Control Building, which had the potential to induce significantly high seismic stresses in the piping between the buildings, was not accounted for in the stress analysis of the piping systems.	Not required for reduced-scope.	Pipe support at penetration was modified to coincide with a design basis piping analysis assumption. A number of "design enhancements" were implemented, including issuance of a new standard, GGNS-08-17, to address seismic housekeeping problems; securing of "S" hooks on lighting fixtures; installation of missing clips and screws on several items; and revision to several design basis calculations.

Table 2.7: Seismic outliers and improvements for SMA plants (Continued)

Plant	Walkdown screen	Anomalies & outliers	Controlling outliers	HCLPF capacities (g)	Plant improvements
H.B. Robinson 2	0.3g HCLPF	33 issues/anomalies related to interactions, maintenance, or housekeeping were identified; 47 components were identified as outliers.	MOV RHR-750 MOV RHR-751	0.28g 0.28g	Concerns for 32 components were addressed by maintenance actions; enhancements for 34 components required repairs or modifications; 16 issues involving electrical raceways involved maintenance of modifications; many of these concerns are being resolved under USI A-46. Note: the IPEEE assumes USI A-46 improvements (which are still to be resolved).
Hatch 1&2	0.3g HCLPF	A number of outliers, mostly related to interaction and anchorage issue.	Outliers are listed in Table 2 of Appendix I (HCLPF would be at least 0.3g after modification).	0.3g after certain components were modified to raise their HCLPF capacities.	A-46 plant. A number of outliers were identified and resolved through modifications, repairs, or complete replacement in order to raise their HCLPF capacity to 0.3g. For example, control room light fixtures were tied up to prevent falling; anchorage of diesel generator relay panel motor control centers were modified.
Limerick 1&2	SSE (but the 0.3g HCLPF screening tables were mostly used)	Some maintenance and housekeeping anomalies were observed.	None identified.	Not evaluated.	Tracking of housekeeping and maintenance issues

Table 2.7: Seismic outliers and improvements for SMA plants (Continued)

Plant	Walkdown screen	Anomalies & outliers	Controlling outliers	HCLPF capacities (g)	Plant improvements
Millstone 2	0.3g HCLPF	"Vulnerabilities"/outliers. A list of opportunities for safety enhancements is in Table 7.1-1 of the submittal, and updated in Attachment 8 of RAI response.	Seismic capacity of turbine building (0.25g), 125 VDC emergency Bus (0.26g), Spent fuel Pool Cooling heat Exchanger (0.26g), 480V buses (0.28g), RBCCW heat Exchanger (0.29g), and equipment related to the open issues (inverter, 0.051g due to block wall failure; batteries, 0.13g; 125 Vac instrument panel, 0.17g; RSST Feeder Breaker, 0.19g; Chilled Water Surge tank, 0.22g.	0.25g for turbine building (after improvements for components with lower HCLPF values).	A-46 plant. Improvements made in order to close items include: modification of the RBCCW Surge tank support, repair of isolation control panel mount housing, modification to anchorage of some battery racks, and bolt modification of some instrument panels. Other items were resolved by verifying component adequacy by calculation or by correcting housekeeping problems. Open items include: the limiting anchorage for the RSST Feeder Breaker's enclosure expansion, the limiting anchorage of the Chilled Water Surge tank, and the unreinforced status of a block wall. In addition, the three issues associated with fire-seismic interaction remain open according to Attachment 8. These are: adequacy of the seismic capacity of the Unit 1 diesel fire pump fuel tank, seismic capacity of a long run of the fire header system piping, and the block wall construction of the fire pump house.
Monticello	0.3g HCLPF	21 categories of outliers (39 components, in total) were identified.	The controlling outliers were not identified.	Fragilities calculations were performed for 4 components, indicating HCLPF capacities exceeding 0.3g. Other outliers were assumed to have HCLPF capacities equal to the SSE.	Three USI A-46 improvements: Fastening of U-bolts on diesel generator starting air receivers. Eliminating the potential impact of an HVAC duct on a relay panel. Upgrading light fixtures in the control room to have a means of anchorage independent of the T-bar supports.

Table 2.7: Seismic outliers and improvements for SMA plants (Continued)

Plant	Walkdown screen	Anomalies & outliers	Controlling outliers	HCLPF capacities (g)	Plant improvements
Nine Mile Point 1	0.3g HCLPF	A number of outliers were identified (on anchor/support and interaction).	Battery boards 11 and 12 (0.27g); containment spray raw water pumps (0.29g); all others (0.3g or better).	0.27g	A-46 plant. A list of improvement initiatives to be resolved in refueling outage 14 was provided. It includes installation of control room panel top cross ties, and improvements of welds and anchoring of power boards and control room cabinets.
Nine Mile Point 2	0.5g HCLPF	See Table 2.4.	Nitrogen bottles HFA Model 154 Relay	0.23g 0.45g	See Table 2.4.
North Anna 1&2	0.3g HCLPF	A number of outliers - tank overturning moment capacity, anchorage, and relay capacity.	Emergency Condensate Storage tanks (0.16g); RWST tank (0.18g); 120 Vac bus (0.19g); Refueling Water Chemical Addition tank Unit 1 (0.19g), Unit 2 (0.24g); Boric Acid tanks (0.21g); Control Room air conditioners (0.21g); Sequence of Events Recorders (0.22g); 4kV Emergency Bus (0.23g); Reactor trip Breakers (0.24g); heating and ventilation chiller Units (0.27g); SG blowdown containment isolation valves (0.28g); CCW pumps (0.29g).	0.16g	A-46 plant. Problems with 58 items of electrical and mechanical equipment, including 3 tanks, have been resolved via field modification. In addition, several minor deficiencies for cable and conduit raceways were resolved. The remaining unresolved issues, consisting of seismic induced fire/flood evaluations, housekeeping issues, control room ceiling review, etc., are scheduled to be resolved by the end of the North Anna Unit 1 refueling outage, scheduled to start in April 2000.
Palo Verde 1,2,&3	0.3g HCLPF	None identified. However, the submittal notes that the walkdown identified a limited number of actions which need be taken to improve plant seismic capacity, but the submittal provides no listing of these actions.	None.	0.3g.	Some modifications were carried out to improve plant seismic capacity. For example, the anchorage on the bookshelves behind the control cabinets in Unit 3 (not 2) was improved.
Peach Bottom 2&3	0.3g HCLPF	At least 168 components or conditions could not be screened out, based on over 45 outlier issues/concerns identified.	RHR heat exchangers (anchorage) Block walls in reactor building HP SW pumps serving RHR HX (interaction) Fans 0AK32, 0BK32, and OCK32 tanks 2BE24, 2CE24, 3BE24, 3CE24, and (anchorage).	0.20g 0.20g 0.21g 0.23g 0.24g Five other component categories have HCLPF capacities less than 0.3g	Numerous USI A-46 and/or IPEEE improvements are planned to address anchorage, equipment support, and housekeeping and maintenance concerns.

Table 2.7: Seismic outliers and improvements for SMA plants (Continued)

Plant	Walkdown screen	Anomalies & outliers	Controlling outliers	HCLPF capacities (g)	Plant improvements
Perry 1	0.3g HCLPF	Four outliers were not screened out for further analysis (they were all spatial interaction issues, and good housekeeping was considered sufficient to resolve these concerns).	None.	0.3g.	Four outliers that did not screen out are 1) maintenance test bench, 2) operations electrical locker, 3) switchgear trolleys, and 4) control room furniture. They involve spatial interactions and are considered housekeeping items and no plant modifications were needed.
Prairie Island 1&2	0.3g HCLPF	Outlier requiring minor maintenance activity (This outcome takes credit for outlier resolution of 22 components under the A-46 program).	component cooling water heat exchangers (0.28g); all others (0.3g or greater).	0.28g	A-46 plant. Correction of spatial interactions involving unrestrained scaffolding and ladders were carried out (through maintenance activities).
Quad Cities 1&2	0.5g HCLPF and 0.3g HCLPF	107 items of equipment and 8 electrical raceway systems were identified as USI A-46 outliers.	Failure of cable tray systems (inadequate anchorage or frame capacity). Racks (anchorage). Switchgear (anchorage). Silencers (anchorage). Chargers (anchorage). Cubicle coolers (anchorage). MCC, Switchgear, Transformers (anchorage). In total, 24 categories of components (comprising about 58 items of equipment) were ultimately determined to have HCLPF capacities less than the 0.3g.	0.09g, 010g, 0.11g, 0.16g, 0.22g 0.11g 0.13g, 0.22g, 0.23g, 0.24g, 0.28g 0.18g, 0.22g 0.22g 0.23g 0.24g	An extensive number of plant improvements or other actions are being undertaken to resolve identified USI A-46 outliers. These improvements pertain primarily to enhancing anchorage/support capacity and reducing or eliminating the potential for adverse interactions.
River Bend	SSE	None identified.	None.	Not required for reduced-scope.	None.
Sequoyah 1&2	0.3g HCLPF	A design-related deficiency, four anomalous conditions, and were identified.	RHR heat exchangers	0.27g	- Replacement of MCC anchorages; - upgrade of RHR heat exchanger anchorages; and - corrective changes to eliminate interactions.

Table 2.7: Seismic outliers and improvements for SMA plants (Continued)

Plant	Walkdown screen	Anomalies & outliers	Controlling outliers	HCLPF capacities (g)	Plant improvements
Shearon Harris 1	0.3g HCLPF	13 issues related to maintenance, housekeeping and seismic interaction that required work orders; six items were found to require repairs or modifications; five instruments are powered from sources that may not be available after a seismic event; and an interaction issue was found regarding the potential for a first aid storage cabinet to fall on a Halon bottle located in the stairwell of the RAB 286 ft.	RHR heat exchangers (0.29g); all others (0.3g or greater).	0.29g	Repair and modification work was scheduled to be completed by the end of refueling outage RFO-7 (Tables 5.3 and 5.4 of the submittal). Alternative instruments or actions have been identified and site abnormal operating procedures are being updated to address the alternate instruments. A Halon bottle located in a stair well was relocated to avoid damage from a falling first aid storage cabinet.
St. Lucie 1&2	SSE	Unit 1: 11 anchorage concerns; low capacity of CCW surge tank platform; and three interaction concerns Unit 2: two interaction concerns; four maintenance issues	None reported having HCLPF capacity less than SSE		Several significant improvements to anchorages; maintenance actions; and implementation of a strict housekeeping policy
Summer	0.3g HCLPF	Outliers related to interaction concerns, missing lateral support, and the use of ceramic components in the neutral grounding resistor.	The plant HCLPF should be 0.3g or greater with the exception of the earth dams and embankment that have an HCLPF of 0.22g (and with no cost effective solutions to fix it).	0.3g for plant.	Bolting together adjacent electrical cabinets at 17 locations throughout the plant to remove interaction concerns, providing lateral support for the isolation valve where the support was missing, and performing analysis to show adequate HCLPF value for the neutral grounding resistor that use ceramic components.
Susquehanna 1&2	0.3g HCLPF	Some anomalies and maintenance concerns were noted; numerous outliers were identified.	Interactions: HPCI pump discharge valve Suppression pool inlet valve Automatic transfer switch MCC	 0.21g 0.21g 0.25g 0.26g	Trolleys removed from switchgear cabinets. (Two anomalies and three housekeeping concerns are being tracked.)
Turkey Point 3&4	SSE	26 anchorage/support concerns, 12 interaction hazards, two functional concerns, and some seismic housekeeping issues were identified	CST RWST Diesel Oil tank	0.11g 0.11g 0.21g Note: These capacities are for the upgraded condition.	Plant actions, analyses, or enhancements were undertaken to resolve all outliers as part of USI A-46.

Table 2.7: Seismic outliers and improvements for SMA plants (Continued)

Plant	Walkdown screen	Anomalies & outliers	Controlling outliers	HCLPF capacities (g)	Plant improvements
Vermont Yankee	0.3g HCLPF	A number of outliers were identified in A-46/IPEEE, mostly related to interaction and anchorage/support concerns, and the use of some old batteries (more than 10 years old).	Condensate Storage tank (0.25g); Diesel Fuel Oil Storage tank (0.29g). All others (will meet or exceed the 0.3g review-level earthquake upon resolution of A-46 outliers).	0.25g	A-46 plant. A-46/IPEEE outliers were resolved to meet A-46 criteria; reroute the fuel line tubing of the diesel fire pump fuel tank; enhance the support of the fire system standpipe. The CST will not be upgraded because there is no simple cost effective enhancement method.
Vogtle 1&2	0.3g HCLPF	Twenty-four open items for each unit, mostly interaction issues.	None.	0.3g	Open items were scheduled to be resolved by August 1, 1996. They included a gap between the battery rack end rails and batteries, potential interactions between the diesel generators and crane controller, etc.
Waterford 3	SSE	No outliers that are operability issues at the plant; three unresolved issues, not considered significant to seismic risk, but related to conforming with standard practice in seismic design (loose items in the Control Room; station air pipe not meeting clearance requirements; and storage of temporary equipment).	None.	Not required for reduced-scope.	All loose items in Control Room were corrected (removed or restrained the lockers and file cabinets in the control room, removed book shelves in the vicinity of safety-related cabinets, and relocated or restrained other loose items in the vicinity of safety-related cabinets). Inadequate clearance issues were resolved; reasons why the existing clearance is acceptable were documented. Procedure to prevent hazardous seismic interactions for transient combustibles was instituted.
Watts Bar 1	0.3g HCLPF	None, other than some minor maintenance and housekeeping issues which were dispositioned, and for which work requests were written as needed.	None.	0.3g or greater	No plant improvements related to the seismic analysis were identified or carried out.
Wolf Creek	0.3g HCLPF (a few components were screened at 0.5g HCLPF)	Five categories of equipment did not satisfy the screening criteria. Miscellaneous equipment installation and housekeeping concerns were identified.	RWST turbine building Four, 60-cell batteries and racks 12 LSELS/ESFAS cabinets Strainers and screens.	The outliers were all assigned an HCLPF capacity of 0.20g, based on judgment.	Resolutions of three housekeeping issues and four equipment installation concerns are planned or have been implemented. A performance improvement request related to placement of transient equipment was issued.

Table 2.8: Containment performance for SMA plants

Plant	Containment type	Walkdown findings/outliers	HCLPF capacity (g)	Plant improvements
Arkansas Nuclear One 1	Large dry type; steel-lined prestressed-concrete	No major vulnerabilities that will compromise containment performance for the RLE were identified.	Not reported.	Inadequate support for the RB Cooling units was identified and additional anchorage for these units has been installed.
Arkansas Nuclear One 2	Large dry type; steel-lined prestressed-concrete	No vulnerabilities that will compromise containment performance were identified.	Not reported.	None.
Braidwood 1&2	Large dry type; steel-lined prestressed-concrete	The walkdown did not identify any early containment failure vulnerabilities.	Nor reported.	None.
Browns Ferry 2&3	Mark I; steel containment	No vulnerabilities in the containment isolation system, relays, containment isolation valves, or containment penetrations due to an RLE event were identified.	Not reported.	None.
Brunswick 1&2	Mark I; reinforced-concrete, steel-lined wetwell	No outliers or anomalies were reported	HCLPF capacity against large early failure is at least 0.3g	None. (However, may be affected by USI A-46 plant improvements.)
Byron 1&2	Large dry type; steel-lined prestressed-concrete	The walkdown did not identify any early containment failure vulnerabilities.	Not reported.	None.
Callaway	Large dry type with steel-lined, post-tensioned reinforced concrete	No outliers or anomalies were reported	HCLPF capacity against early failures is at least 0.3g	None.
Clinton	Mark III; reinforced-concrete drywell, steel-lined reinforced-concrete wetwell	The containment and components are seismically rugged for the RLE.	Not reported.	None.
Comanche Peak 1&2	Large dry type, with steel-lined reinforced concrete structure.	No outliers or anomalies were reported.	Not applicable.	None.
Cooper	Mark I; steel containment	No concerns specifically pertaining to early containment failure were identified.	Components were included in the SSEL or screened for 0.3 g PGA RLE.	None.
Crystal River 3	Large dry type; steel-lined prestressed-concrete	The walkdown did not identify any early containment failure vulnerabilities.	Not reported.	None.
Davis-Besse	Large dry type; steel containment	No containment vulnerabilities were found	Not reported.	None.

Table 2.8: Containment performance for SMA plants (Continued)

Plant	Containment type	Walkdown findings/outliers	HCLPF capacity (g)	Plant improvements
Dresden 2&3	Mark I; steel containment	The walkdown did not identify any vulnerabilities associated with early containment failure due to a postulated seismic event.	Not reported.	None.
Duane Arnold	Mark I; steel containment	Important equipment essential to containment performance was included in the SSEL and reviewed by the SRT during the walkdown.	Not reported.	None.
Farley 1&2	Large dry type; steel-lined prestressed-concrete	No concerns specifically pertaining to early containment failure were identified.	Not reported.	None.
Fermi 2	Mark I; steel containment	No vulnerabilities in the containment isolation system, relays, containment isolation valves, or containment penetrations were identified.	Not reported.	None.
FitzPatrick	Mark I; steel containment	The licensee found no containment vulnerabilities.	HCLPF capacity against early failures is at least 0.3g	None.
Fort Calhoun 1	Large dry type; steel-lined prestressed-concrete	Not reported; a quantitative Level 2 analysis was performed, indicating that the conditional probability of large early release, given seismic core damage, is about 1%.	Not reported.	No improvements were made specifically to address containment performance.
Ginna	Large dry type; steel-lined prestressed-concrete	The review did not identify any features which would give rise to an early containment failure concern.	Not reported.	None.
Grand Gulf 1	Mark III; reinforced-concrete drywell, steel-lined reinforced-concrete wetwell	The submittal states that containment isolation was factored into the SSEL. However, no specific information is provided.	Not reported.	None.
H.B. Robinson 2	Large dry type, of prestressed concrete with a steel liner.	Potential interfacing systems LOCA (ISLOCA) inside containment due to MOV failures.	Reported HCLPF capacity against large-early failure of at least 0.3g.	None. The potential ISLOCA concern was evaluated as being adequate for the RLE.
Hatch 1&2	Mark I; steel containment	No outliers or anomalies were reported.	HCLPF capacity against early failures is at least 0.3g PGA.	None.
Limerick 1&2	Mark II; reinforced-concrete drywell, steel-lined reinforced-concrete wetwell	No outliers or anomalies were reported.	Not reported, although all components essentially screened at 0.3g HCLPF.	None.
Millstone 2	Large dry type; steel-lined prestressed-concrete	No safety-related concerns were found.	Not reported.	None.

Table 2.8: Containment performance for SMA plants (Continued)

Plant	Containment type	Walkdown findings/outliers	HCLPF capacity (g)	Plant improvements
Monticello	Mark I; steel containment	No concerns or additional seismic outliers were reported.	Not reported.	None.
Nine Mile Point 1	Mark I; steel containment	No containment vulnerabilities were found.	Not reported.	None.
Nine Mile Point 2	Mark II	See Table 2.5.	HCLPF capacity against large early release was not reported.	None.
North Anna 1&2	Subatmospheric containment; steel-lined reinforced-concrete	No vulnerabilities were noted in the containment walkdown.	Not reported.	None.
Palo Verde 1,2,&3	Large dry type; steel-lined prestressed-concrete	Only one minor concern with containment penetration was observed and resolved by analysis.	Not reported.	None.
Peach Bottom 2&3	Mark I; steel containment	None in addition to those encountered in evaluation of success paths (components to avert early containment failure were included in success paths).	Not reported.	No additional plant improvements beyond those already identified for success path equipment.
Perry 1	Mark III; reinforced-concrete drywell, steel wetwell	No vulnerabilities in the containment isolation system, relays, containment isolation valves, or containment penetrations due to an RLE event were identified in the IPEEE.	Not reported.	None.
Prairie Island 1&2	Large dry type; steel containment	No concerns specifically pertaining to early containment failure were identified.	Not reported.	None.
Quad Cities 1&2	Mark I; steel containment	No concerns specifically pertaining to early containment failure were identified.	None reported.	None, beyond those already identified for success path equipment.
River Bend	Mark III; reinforced-concrete drywell, steel wetwell	No concerns specifically pertaining to early containment failure were identified.	Not reported.	None.
Sequoyah 1&2	Ice Condenser; steel containment	Walkdown revealed no anomalies or outliers.	HCLPF capacity against large-early failure of at least 0.3g.	None.
Shearon Harris 1	Large dry type; steel-lined reinforced-concrete	An interaction concern was raised, which involved a platform in the equipment hatch at 286 ft elevation.	Not reported.	None. The interaction issue identified in plant walkdown was evaluated and determined not to be detrimental to the containment integrity.

Table 2.8: Containment performance for SMA plants (Continued)

Plant	Containment type	Walkdown findings/outliers	HCLPF capacity (g)	Plant improvements
St. Lucie 1&2	Steel vessel surrounded by a reinforced-concrete biological shield, with an annular space in between.	No evaluation was conducted.	No evaluation was conducted.	None.
Summer	Large dry type; steel-lined prestressed-concrete.	No concerns specifically pertaining to early containment failure were identified.	Not reported.	None.
Susquehanna 1&2	Mark II; reinforced-concrete drywell, steel-lined reinforced-concrete wetwell	None. (No comprehensive walkdown of containment safeguards; only piping/valves and containment structure were considered.)	Insufficient evaluation to determine.	None.
Turkey Point 3&4	Large dry type of steel-lined post-tensioned reinforced-concrete.	No evaluation was conducted.	No evaluation was conducted.	None.
Vermont Yankee	Mark I; steel containment	The containment performance evaluation does not identify any vulnerabilities associated with early containment failure due to a postulated seismic event.	Not reported.	None.
Vogtle 1&2	Large dry type; steel-lined prestressed-concrete	No concerns specifically pertaining to early containment failure were identified.	Not reported.	None.
Waterford 3	Large dry type; steel containment	No concerns specifically pertaining to early containment failure were identified.	Nor reported.	None.
Watts Bar 1	Ice condenser; steel containment	No vulnerabilities were noted in the containment walkdown.	Not reported.	None.
Wolf Creek	Large dry type, steel-lined, reinforced, post-tensioned concrete	Containment cooling and isolation systems were included in equipment list; no outliers were identified. Missing bolts on one seated beam connection (later found to be addressed in existing plant design documents). Instances were found where conduits interfere with the seismic isolation gap between the containment steel liner and the operating floors within the reactor building. These were judged to be sufficiently spaced so as not to be a significant concern.	HCLPF capacity against early failure of at least 0.3g.	None.

Table 2.9: Relay evaluation

Plant	Treatment	Low-ruggedness relays identified	Safety implications	Related plant improvements
Arkansas Nuclear One 1	USI A-46 and IPEEE relay evaluation performed simultaneously. full capacity versus demand screening, in accordance with the GIP, was performed on all relays.	There were 48 relay outliers as a result of the relay screening process. 92 relays have unknown capacities and require further research. No "bad actor" relays were identified.	None of the identified outliers represented any adverse operability issues.	Relay outliers are being tracked for resolution as part of the USI A-46 program.
Arkansas Nuclear One 2	USI A-46 and IPEEE relay evaluation performed simultaneously. full capacity versus demand screening, in accordance with the GIP, was performed on all relays.	There were 10 relay outliers as a result of the relay screening process. 198 relays have unknown capacities and require further research. No "bad actor" relays were identified.	None of the identified outliers represented any adverse operability issues.	Relay outliers are being tracked for resolution as part of the USI A-46 program.
Beaver Valley 1	Relays were not included in the PRA model. The licensee screened out from the analysis any relay actuating devices which depend on offsite power.	None in IPEEE.	None	Relays, including "bad actor" relays are resolved via the USI A-46 program.
Beaver Valley 2	Relays were not modeled in the PRA, as they were deemed to pass the 0.3g review-level earthquake criterion.	Two LRRs were found within the IPEEE scope.	The two LRRs will not impact the plant response following a seismic event.	None.
Braidwood 1&2	Relay chatter evaluation consists of identifying low-ruggedness relays that may affect SSEL equipment functions, using the list in EPRI-NP-7148-SL, Appendix E.	None of the relays were of the low-ruggedness type as listed in Appendix E of EPRI NP-7148-SL; a non-safety related mercury relay for the CO_2 fire protection system was identified as a poor performer.	None. Inadvertent actuation of the CO_2 fire protection system (using a mercury relay) does not impact the operation of the EDGs.	None.
Browns Ferry 2&3	The identification procedures followed the GIP for the A-46 relays and were expanded for IPEEE .	None identified for IPEEE. Relays which have not been screened will be labeled as outliers in the A-46 program, with additional review or resolution to be performed.	None for IPEEE. Problem with IPEEE-specific relays and their resolutions were not identified.	None.
Brunswick 1&2	USI A-46 relay evaluation, expanded to IPEEE-only circuitry.	Several identified, four in IPEEE-only circuitry (for containment performance).	The IPEEE-only relays were found acceptable based on consequence review; others are being addressed in USI A-46.	Concerns are being addressed under USI A-46.

Table 2.9: Relay evaluation (Continued)

Plant	Treatment	Low-ruggedness relays identified	Safety implications	Related plant improvements
Byron 1&2	The relay study focused on identifying low-ruggedness relays by looking through the database to find relays generically known to be of low-ruggedness (Westinghouse COM-5 and SSC-T relays).	None. The very few outliers identified, were found either not to affect safe shutdown or to have been modified/designed to conform to the higher acceleration level specifications.	None. Nonsafety relays were also included in the study. An example of nonsafety relays is the relays controlling the ventilation dampers (and also CO_2 discharge) in the diesel generator rooms. CO_2 discharge will not affect the operation of the diesel generators. The closing of the ventilation dampers will be dealt with by operator procedures created to provide instructions to the operators on how to restore ventilation to the diesel generators.	None.
Callaway	Documentation-based evaluation to identify low-ruggedness relays and determine consequences of chatter, spot-check of relay installations.	Some low-ruggedness relays were identified.	Relay chatter was determined to be acceptable with respect to safe shutdown of the plant.	None.
Calvert Cliffs 1&2	The relay chatter was not included in the seismic IPEEE analysis.	The A-46 program found no bad actor relays.	None	None
Catawba 1&2	Low-ruggedness evaluation; relay chatter and recovery actions modeled in SPRA.	One, in a diesel generator maintenance and testing circuit.	Modeled in seismic PRA.	None.
Clinton	EPRI-NP-7148-SL was used for the screening. A circuit analysis was then performed for relays that were screened in.	22 low-ruggedness relays: model numbers: GE CEH, GE HGA, W SSC.	None. Walkdowns were performed to verify seismic adequacy of the identified LRRs. In addition, the plant seismic qualification test records indicated that these relays are capable of withstanding the CPS SSE without compromise of structure or electrical function.	None.

Table 2.9: Relay evaluation (Continued)

Plant	Treatment	Low-ruggedness relays identified	Safety implications	Related plant improvements
Columbia Generating*	Relays were screened using a generic screening criterion that is consistent with the guidelines in Appendix Q of EPRI NP-6041.	None reported.	Relay chatter modeled in PRA, and no recovery was modeled in the logic model.	None.
Comanche Peak 1&2	None, and none required (non-USI A-46 reduced-scope plant).	None, not applicable.	Not applicable.	None.
Cooper	USI A-46 and IPEEE evaluation. The screening techniques used were similar to those utilized for the more strict USI A-46 relay review.	None identified.	None.	None.
Crystal River 3	Not required for reduced-scope IPEEE plant. Relay evaluation was performed as part of the plant's USI A-46 program, completed before the initiation of the IPEEE evaluation.	None for IPEEE.	None.	None.
D.C. Cook 1&2	USI A-46 relay evaluation, expanded to IPEEE-only circuitry.	A number of low-ruggedness relays were identified, none in IPEEE-only circuitry.	Being addressed under USI A-46.	Licensee plans to replace low-ruggedness relays affecting safety equipment.
Davis-Besse	Relay evaluation performed as part of A-46 program, expanded for IPEEE.	None for IPEEE. No relays beyond the ones identified in the A-46 program were identified as low-ruggedness.	None. In the case of the pump motor circuits, where low-ruggedness Westinghouse ITH relays are employed for ground fault detection in the 4 kV high pressure injection and makeup pump motor circuits, these relays are not of the lockout type and, therefore, the tripped pump can be simply restarted from the control room.	None.
Diablo Canyon 1&2	Relay evaluation in LTSP.	None in IPEEE.	Modeled in seismic PRA.	None.
Dresden 2&3	Joint A-46/IPEEE program.	All relays have been evaluated except for those associated with the Isolation Condenser system (approximately 65% of the relays).	Walkdown of a group of relays associated with the Isolation Condenser system is still needed, as the plant status permits.	Resolution of a group of relays associated with the Isolation Condenser system is still pending.

Table 2.9: Relay evaluation (Continued)

Plant	Treatment	Low-ruggedness relays identified	Safety implications	Related plant improvements
Duane Arnold	Not required for reduced-scope plant. USI A-46 evaluation.	None.	None.	None.
Farley 1&2	Not required for reduced-scope plant. A USI A-46 relay evaluation was performed per the SQUG GIP for Unit 1. Although not required, an evaluation of relay chatter is being performed for Unit 2 as a prudent measure.	None.	None.	None.
Fermi 2	The relay study focused on identifying low-ruggedness relays, in accordance with the guidance in NUREG-1407 for focused-scope plants not included in the USI A-46 program. The licensee reviewed all safety-related systems to identify low-ruggedness relays.	Six low-ruggedness relays were identified in the review of plant electrical systems. Of these, four were identified for replacement; two were evaluated to have no effect on the operability of systems. 214 low-ruggedness relays were identified in the review of plant control systems, and an evaluation of the consequences of relay chatter indicated that none of these relays would cause a control system malfunction and, therefore, none needed to be replaced.	During a severe seismic event, it is expected that many spurious alarms will be received in the control room due to low seismic ruggedness relay chatter. Although this may not have a direct effect on safe plant shutdown, it may cause some confusion in the control room. This item will also be included in the new seismic simulator training event.	Four LRRs for the plant electrical systems were identified for replacement, and a new seismic simulator training event has been established for spurious alarms in the control room due to low seismic ruggedness relay chatter.
FitzPatrick	USI A-46 relay evaluation expanded for IPEEE.	None for IPEEE. No "bad actor" relays were identified in the EDG building in the scope of the IPEEE program.	None for IPEEE. All of the USI A-46 relay outliers have already been resolved.	None.
Fort Calhoun 1	USI A-46 relay evaluation, expanded to IPEEE-only circuitry.	Six low-ruggedness relays in diesel generator lock-out circuitry were identified; no IPEEE-only low-ruggedness relays were found.	The low-capacity relays were assessed as limiting the plant HCLPF capacity to 0.01g.	The six low-ruggedness relays are being replaced as part of USI A-46 resolution.
Ginna	Relay chatter review was not performed in the seismic IPEEE because no low-ruggedness relays were found during the resolution of the USI A-46 relays.	None.	None.	None.
Grand Gulf 1	For reduced-scope plants, which are not included in the USI A-46 program, a relay chatter evaluation is not necessary per NUREG-1407.	Not applicable.	Not applicable.	Not applicable.

Table 2.9: Relay evaluation (Continued)

Plant	Treatment	Low-ruggedness relays identified	Safety implications	Related plant improvements
H.B. Robinson 2	USI A-46 relay evaluation, expanded to partially address IPEEE-only circuitry.	No low-ruggedness relays were identified affecting the SSEL.	All relays were found acceptable based on capacity screening and/or consequence assessment.	None.
Haddam Neck	USI A-46 relay evaluation, expanded to IPEEE-only circuitry.	Several installations of Westinghouse COM-5 relays; mercoid relays in actuation circuitry for fire protection systems.	Addressed in USI A-46 and SPRA model.	The submittal states that relay chatter is a risk outlier to be resolved; changes to abnormal operating procedures (AOPs) have been proposed.
Hatch 1&2	USI A-46 relay chatter evaluation expanded for IPEEE.	List is included in the submittal.	The identified low-ruggedness relays were resolved by determining that either malfunction of the relay is acceptable or operator actions can be used to reset relays or restore systems to operation. Therefore, all low-ruggedness relays identified as part of the USI A-46 or IPEEE evaluation for both units were resolved at a HCLPF level of at least 0.3g PGA.	None.
Hope Creek	Relay chatter was not incorporated into the PRA model. Screening review at the 0.3g level was performed.	Approximately 100 potentially low-ruggedness relays (LRR) were identified.	The identified potentially LRRs were all screened out (because they are not associated with safe shutdown or containment performance; they have high seismic capacity, or chatter is acceptable).	None.
Indian Point 2	USI A-46 relay evaluation, expanded to IPEEE-only circuitry.	In IPEEE-only circuitry, four Westinghouse SC over-current relays used for protection of the station auxiliary transformer.	Recoverable loss of offsite power. (A low capacity against seismically induced loss of offsite power is already assigned in the seismic PRA.)	None for IPEEE-only circuitry.

Table 2.9: Relay evaluation (Continued)

Plant	Treatment	Low-ruggedness relays identified	Safety implications	Related plant improvements
Indian Point 3	Based on IPE study and the A-46 program, as well as examination of the impact of chattering of each relay.	12 relays related to the emergency diesel generator system.	Modeled in the PRA model. Chatter was assumed regardless of the level of ground motion and recovery actions were not credited.	EDG room CO_2 system.
Kewaunee	USI A-46 relay evaluation, expanded to IPEEE-only circuitry.	12 (Westinghouse SC) low-ruggedness relays identified, none in IPEEE-only circuitry.	Being addressed in USI A-46.	Low-ruggedness relays are to be replaced or circuitry re-worked.
La Salle 1&2	Not documented.	None reported.	Not documented.	None reported.
Limerick 1&2	Relay evaluation following EPRI NP-6041-SL guidelines.	Five chatter-prone relays were identified.	The relays were evaluated and found to be acceptable.	None.
McGuire 1&2	Low-ruggedness evaluation; relay chatter and recovery actions modeled in SPRA.	Low-ruggedness relays found in alarm circuitry.	Low-ruggedness relays affect alarm circuitry only; other relay chatter effects are modeled in the seismic PRA.	None.
Millstone 2	USI A-46 program expanded to include IPEEE components.	No low-ruggedness relays were identified as part of the USI A-46 program.	None for IPEEE.	None.
Millstone 3	Potentially vulnerable relays were identified; relay chatter and recovery actions modeled in SPRA.	Not documented in IPEEE submittal report.	Modeled in SPRA.	The licensee updated AOPs to enhance recovery from earthquake-induced relay chatter.
Monticello	USI A-46 relay evaluation, expanded to IPEEE-only circuitry.	An extensive list of known low-ruggedness GE/HGA relays, and potential low-ruggedness relays, was reported. (The list includes both USI A-46 and IPEEE relays.)	All but four of the low-ruggedness relays were found acceptable based on configuration or functional analysis. The remaining four low-ruggedness relays are all within the scope of USI A-46.	Four low-ruggedness relays are being dispositioned under USI A-46.

Table 2.9: Relay evaluation (Continued)

Plant	Treatment	Low-ruggedness relays identified	Safety implications	Related plant improvements
Nine Mile Point 1	USI A-46/IPEEE evaluation.	LRRs were identified and discussed in the submittal individually.	LRRs were evaluated for impact on the safe shutdown of the plant. Some LRRs were found not to have any impact; others were found to have a very low probability of causing problems or to be easily restored from the control room, or they will be evaluated for either replacement or for improved operator procedures, if applicable.	Replace mercury relays and modify procedures (scheduled to be implemented by the end of RFO15).
Nine Mile Point 2	Detailed relay evaluation at 0.5g HCLPF.	All relays screened out at 0.5g, except one (HFA Model-154) which was determined to have an HCLPF capacity of 0.45g.	Based on a relay screening and consequence assessment, the licensee concludes relay chatter will not limit the plant HCLPF to be below 0.5g.	None.
North Anna 1&2	USI A-46 relay evaluation.	None. No relay chatter review performed because no low-ruggedness relays were found at NAPS during the evaluation of USI A-46 relays. A relay chatter evaluation was thus not performed in the seismic IPEEE.	None for IPEEE.	None.
Oconee 1,2,&3	USI A-46 and IPEEE evaluation.	The overhead power path relays were found to have low fragilities, and 142 other low-ruggedness relays were listed for further analysis or replacement.	Modeled in the PRA.	Capacity issues for 59 relays have been resolved by analysis and/or testing, six relays have been actually replaced with an additional 14 relays awaiting implementation. Several other relay modification design packages are in progress, and additional relay testing is being conducted. Complete resolution by 2002.

Table 2.9: Relay evaluation (Continued)

Plant	Treatment	Low-ruggedness relays identified	Safety implications	Related plant improvements
Oyster Creek	USI A-46 and IPEEE evaluation. Relay evaluation was performed per the guidelines of EPRI NP-7148-SL.	Low capacity relays which do not meet the USI A-46 requirements will be replaced.	Modeled in PRA. No recovery was modeled in the logic model.	None.
Palisades	USI A-46 relay evaluation, expanded to IPEEE-only circuitry.	A number of low-ruggedness relays were identified, none in IPEEE-only circuitry.	Being addressed in USI A-46; SPRA modeling is unclear.	Concerns are being dispositioned under USI A-46.
Palo Verde 1,2,&3	The relay capacities were examined using two sources: (1) GERS in EPRI-NP-7147-SL, and (2) the plant seismic qualification test records. The relay evaluation was documented in accordance with EPRI-NP-7148.	None identified. (There is a concern on the new Impell SASSI analysis. However, it may not have a significant effect on conclusion.)	None.	None.
Peach Bottom 2&3	USI A-46 relay evaluation, expanded to IPEEE-only circuitry.	A number of low-ruggedness relays were identified from the USI A-46 evaluation; no additional low-ruggedness relays were encountered in IPEEE-only circuitry. The seismic-fire interaction assessment identified mercoid switches in the fire water system, and potentially vulnerable relays in the Cardox system.	Being addressed in USI A-46.	Concerns are being dispositioned under USI A-46. To address seismic-fire interaction concerns, mercoid switches are being replaced, and procedural controls are being implemented to mitigate any effects of spurious relay operation in the Cardox system.
Perry 1	The relay evaluation consisted of locating low seismic ruggedness relays in accordance with Appendix E of EPRI NP-7148-SL.	Low-ruggedness relays are GE HFA relay type used in HPCS DG control circuitry and RPS Motor Generator Set control circuitry.	RPS system was screened out. Of the sixteen HPCS DG relays, 11 are chatter acceptable, five require operator action after the RLE. Four of the five relays can be reset in the HPCS DG room. Operator notified of a malfunction of the fifth relay by annunciators and can then manually reset tripped breaker. Also, at least 25 min. available to perform the required operators' actions per existing Alarm Response Instructions before RPV water at TAF.	None.

Table 2.9: Relay evaluation (Continued)

Plant	Treatment	Low-ruggedness relays identified	Safety implications	Related plant improvements
Pilgrim 1	USI A-46 relay evaluation, use of relay generic equipment ruggedness spectrum (GERS), and SPRA review.	Not specified.	Being addressed in USI A-46; SPRA modeling of relay chatter assumes USI A-46 resolution.	Concerns are being addressed under USI A-46.
Point Beach 1&2	USI A-46 relay evaluation, expanded to partially address IPEEE-only circuitry.	A number of Westinghouse ITH relays.	Being addressed in USI A-46; no SPRA modeling of relay chatter.	Concerns are being addressed under USI A-46.
Prairie Island 1&2	USI A-46 evaluation conducted following SQUG procedures, expanded for IPEEE.	None for IPEEE.	None for IPEEE.	None.
Quad Cities 1&2	USI A-46 relay evaluation, expanded to partially address IPEEE-only circuitry.	No low-ruggedness relays were reported as existing within the list of essential IPEEE-only relays.	None for IPEEE-only essential relays.	As part of USI A-46 outlier resolution, a "bad actor" mercoid switch (PE-1) is being replaced.
River Bend	Not required for IPEEE. A reduced-scope plant not included in USI A-46 program.	Not applicable.	Not applicable.	None.
Salem 1&2	The licensee did not incorporate relay chatter into the PRA model. Instead, a screening review at the 0.3g level (which was the plant's review level earthquake) was performed.	Approximately 100 potentially low-ruggedness relays were identified. Some of the identified LRRs have been replaced with higher seismic capacity relays, including the 4 kV Phase A/B/C diesel generator differential relays.	None. All relays were screened out because LRRs are not associated with safety shutdown or containment performance; relay chatter is acceptable; the LRRs have high seismic capacity.	None. Some of the identified LRRs have already been replaced with higher seismic capacity relays.
San Onofre 2&3	Rigorous treatment, consisting of (1) identification and classification of essential relays; (2) relay walkdown; and (3) SPRA modeling of relay chatter, including fragility evaluations and consideration of operator recovery actions.	Several lower capacity relays that whose chatter was unacceptable or required operator actions were identified for fragility calculations.	Relay chatter and associated operator recovery were modeled in the seismic PRA. Seismic fragilities for several relays were developed. Relay chatter has been identified as a dominant risk contributor. The licensee performed a cost benefit analysis, and found that changing the relays in DG circuitry was not cost effective.	No relays were replaced. Some cabinets are to be fastened together to reduce the potential for relay chatter.

Table 2.9: Relay evaluation (Continued)

Plant	Treatment	Low-ruggedness relays identified	Safety implications	Related plant improvements
Seabrook	Fragilities of electrical relays were addressed in original SSPSA, using generic test and analysis data. These fragilities were updated in 1986 based on actual component qualification reports.	88 relays that have a factor of safety less than four above the SSE (0.25g) were classified in two groups and included in the seismic PRA quantification.	Group 1 relays are protective relays on diesel generator control panels. The most significant impact of chatter of these relays is the shutdown of the diesel generators. Should this happen, the operators would immediately be alarmed and are expected to follow existing emergency operating procedures to manually restart the diesel. Group 2 relays have higher fragility values. The most significant impact of chatter of Group 2 relays is to trip off loads on the emergency buses. These are also considered recoverable, but the diagnosis and response to multiple relay chatter would be more difficult.	None.
Sequoyah 1&2	Full relay chatter evaluation, including capacity screening and consequence assessment.	Several low-ruggedness relays were identified, none of which were determined to cause malfunction of SSEL equipment.	Consequence analysis indicates no effects on SSEL.	None.
Shearon Harris 1	EPRI-NP-7148-SL was used for the relay screening.	Fifty-one relays were identified to be potentially low-ruggedness relays that required a chatter evaluation.	Forty-five relays, either rugged or non-essential. Inadvertent relay trip of the remaining six relays, all GE model 12PVD21B1A, is not a concern based on further relay chatter study.	None.
South Texas Project 1&2	None, and none required (non-USI A-46 reduced-scope plant); however, relay chatter is modeled in SPRA.	None, not applicable.	Modeled in SPRA.	None.

Table 2.9: Relay evaluation (Continued)

Plant	Treatment	Low-ruggedness relays identified	Safety implications	Related plant improvements
St. Lucie 1&2	No evaluation required for Unit 2 (non-USI A-46, reduced-scope plant); For Unit 1, USI A-46 evaluation searched for bad actor relays, verified mountings of relays.	Not reported.	Licensee concluded there were no deleterious effects of chatter of bad actor relays.	None.
Summer	The relay evaluation looked for "bad actor" relays from the IPEEE list of such relays.	Two bad actor relays were found to provide alarm functions only and, therefore, were of no concern.	None.	None.
Surry 1&2	Relay chatter was not considered in the seismic IPEEE because, according to the submittal; a detailed evaluation was performed in the A-46 program.	None for IPEEE. Identified under A-46 program: Westinghouse SV model relays, used in the EDG circuits as field flashing relays; and mercury relays, used in the fire protection circuit for the EDG rooms.	None for IPEEE. LRRs are being replaced; and the mercury relays would be able to withstand a spurious CO_2 release coincident with an EDG start.	None for IPEEE. Under A-46, the LRRs are planned to be replaced via station-approved Design Change 95-017.
Susquehanna 1&2	Identification of low-ruggedness relays; walkdown verification; evaluation of chatter effects.	Four locations of low-ruggedness relays were identified.	Effects of chatter deemed acceptable.	None.
TMI 1	Relays screened out or evaluated using criteria in EPRI NP-7147, "Generic Ruggedness of Relays" (GERS).	87 relays were evaluated for fragility evaluation using the generic data from GERS.	Relay chatter and recovery actions were modeled in the PRA.	All relays that cannot pass any seismic screening criteria will be replaced during upcoming refueling outages.
Turkey Point 3&4	USI A-46 evaluation searched for bad actor relays, verified mountings of relays.	Not reported.	Licensee concluded there were no deleterious effects of chatter of bad actor relays.	None.
Vermont Yankee	USI A-46 evaluation expanded for IPEEE.	Two low-ruggedness relays.	No effect on plant safe shutdown.	None.
Vogtle 1&2	The principal method used for screening was to review the VEGP Equipment Qualification Data Packages (EQDPs) which are design documents containing test results of the VEGP Equipment Qualification (EQ) program.	None.	None.	None.
Waterford 3	Not required for reduced-scope plant.	Not applicable.	Not applicable.	None.

Table 2.9: Relay evaluation (Continued)

Plant	Treatment	Low-ruggedness relays identified	Safety implications	Related plant improvements
Watts Bar 1	EPRI NP-7147-SL was used to establish the low-ruggedness relays at WBN.	None.	None.	None.
Wolf Creek	Computerized identification of low-ruggedness relays; review of electrical schematics; no walkdown verification.	One model of low-ruggedness relay (GE HGA) was found to exist in the safe shutdown equipment list (SSEL).	The relay model of concern is not used in a low-ruggedness configuration in SSEL equipment.	None.

* Formerly known as Washington Nuclear Project Number 2.

Table 2.10: Soil evaluation

Plant	Soil/foundation Characteristics	SSI or soil response analysis	Soil liquefaction	Other soil failure modes
Arkansas Nuclear One 1	Rock	Rock site. The rock-founded structures at ANO-1 were modeled with base springs representing the "soil" stiffness.	Not applicable.	None.
Arkansas Nuclear One 2	Rock	Structures founded on rock. The rock-founded structures at ANO-2 were modeled as fixed-based structures.	Not applicable.	None.
Beaver Valley 1	Soil/rock. Structures founded on mat foundations with approximately 60 to 100 ft of overburden soils (sands) to bedrock.	Scaling of existing design analysis results using the ratios of the medium UHS to the design spectrum (NUREG/CR-0098).	The minimum safety factors against liquefaction as developed in the original design analysis were reviewed to screen out the soil liquefaction following RLE.	None.
Beaver Valley 2	Soil/rock. Structures founded on mat foundations with approximately 60 to 100 ft of overburden soils (sands) to bedrock.	Scaling of existing design analysis results using the ratios of the medium UHS to the design spectrum (NUREG/CR-0098).	The minimum safety factors against liquefaction as developed in the original design analysis were reviewed to screen out the soil liquefaction following RLE.	None.
Braidwood 1&2	Founded either completely or partly on bedrock	Structures founded on rock. The seismic analysis is based on a direct generation technique using a random vibration approach.	No soil evaluation performed.	None.
Browns Ferry 2&3	Rock	Structures founded on rock. IRS for the IPEEE were developed by scaling the A-46 IRS, based on the guidelines in EPRI NP-6041.	Not applicable.	None.
Brunswick 1&2	76 ft soil; 50 ft of structural fill over 26 ft dense sands.	Results of design SSI analysis, scaled and modified for frequency shift.	No IPEEE evaluation.	No IPEEE evaluation.
Byron 1&2	Rock	Structures founded on rock. The seismic analysis is based on a direct generation technique using a random vibration approach.	Not applicable.	None.
Callaway	Some structures founded on rock; others are founded on structural fill or stabilized backfill having a depth of anywhere from 19 ft to 54 ft over bedrock.	FLUSH finite element SSI analysis for power block structures.	No IPEEE evaluation; however, the licensee concludes the fill materials are not susceptible to liquefaction.	Capability of buried piping (between power block and other structures) was determined to exceed the RLE.
Calvert Cliffs 1&2	Soil	Probabilistic SSI analyses were performed for the containment structure, auxiliary building, intake structure, turbine building, and new EDG building.	The soil liquefaction analyses were performed by S&A. The results concluded that, although liquefaction would occur for the new EDG building at a median PGA of 0.27g, it would not cause a realistic hazard. It is also concluded that the seismic-induced foundation settlement is negligibly small.	None.

Table 2.10: Soil evaluation (Continued)

Plant	Soil/foundation Characteristics	SSI or soil response analysis	Soil liquefaction	Other soil failure modes
Catawba 1&2	Category I structures: Rock or concrete fill extending to rock. Some components founded on, or buried in, soil.	None (deemed negligible).	No concerns identified.	No concerns identified.
Clinton	Soil	Using a so-called Multiple Analysis Method by EQE which includes direct comparisons of free-field motions, comparisons of deconvoluted motions to the structure foundation level, and simplified soil-structure interaction (SSI) analyses for comparisons of floor response spectra.	A focused-scope plant that is not required to perform soil evaluations.	None.
Columbia Generating*	Soil	New probabilistic SSI analyses were performed.	A soil liquefaction analysis and a seismic soil settlement analysis were performed. It is concluded that both the liquefaction potential and seismic incurred settlement are negligible at the site.	None.
Comanche Peak 1&2	Rock site predominantly.	Not reported.	No IPEEE evaluation required.	No IPEEE evaluation required.
Cooper	Soil. Bedrock elevation was taken as 822', ground surface as 902', and the top of the water table as 880'.	An SSI analysis was performed for the control and reactor buildings using a substructuring approach following the general procedure outlined in Appendix E of NP-6041.	No soil evaluation was conducted (based on Supplement 5 to GL 88-20).	None.
Crystal River 3	Soil (marshland)	A reduced-scope plant, using the SSE design basis ground spectra of the Housner type with a PGA of 0.1g as the IPEEE review level earthquake (RLE).	Soil evaluation not required (reduced-scope plant).	None.
D.C. Cook 1&2	Soil site; a slope (approximately 2:1) bounds the plant site to the east.	SSI margin factors developed.	No concerns identified.	Soil pressure failure found to dominate containment building fragility; no other concerns identified.
Davis-Besse	The site consists mostly of marshland. The station structures are located approximately in the center of the site and are built on a bedrock foundation.	A re-analysis of plant structures was conducted by EQE International, utilizing 3-D structural models and accounting for foundation embedment effects, for 0.3g PGA with a NUREG/CR-0098 median rock spectral shape. A scale factor of 0.697 was used for all spectral values.	No soil evaluation was performed (based on Supplement 5 to GL 88-20).	None.
Diablo Canyon 1&2	Rock sites; some components founded on, or buried in, soil.	None described.	No concerns identified.	No concerns identified; effects modeled in fragilities.

Table 2.10: Soil evaluation (Continued)

Plant	Soil/foundation Characteristics	SSI or soil response analysis	Soil liquefaction	Other soil failure modes
Dresden 2&3	Rock	Structures founded on rock. SME in-structure response spectra were developed using a 3-dimensional horizontal model of the major structures.	Not applicable.	None.
Duane Arnold	Rock/soil. Structures supported on bedrock or lean concrete over bedrock or on 30 to 50 feet of overburden soil or compacted fill soil.	DBE analysis (for reduced-scope plant).	Not required for reduced-scope plant.	A site with shallow soil conditions, accounted for in the development of the IRS.
Farley 1&2	Soil	SSE analysis (for reduced-scope plant)	Soil evaluation analyses not required for reduced-scope plant.	New SSI analyses conducted for the DG and SW intake structures supported on cast-in-place caissons.
Fermi 2	Rock. Major Category I structures supported on bedrock.	Structures founded on rock. Performed new dynamic analysis for IPEEE.	Not applicable.	None.
FitzPatrick	Rock	Buildings founded on bedrock. The structural response analyses were performed using the direct generation method.	Not applicable.	None.
Fort Calhoun 1	65-75 ft of sandy soil over bedrock; structures are supported on pipe piles.	Soil springs in lumped-mass model.	Liquefaction HCLPF=0.25g for soil outside the vicinity of Category I structures; controls capacity of diesel fuel oil storage tanks and raw water system piping.	Soil failures are dominated by liquefaction.
Ginna	Rock	Safety-related buildings founded on rock. Same spectra as those used in USI A-46. However, a safety factor of 1.5 was used to justify meeting the IPEEE seismic demands.	Not applicable.	None.
Grand Gulf 1	Soil	SSE analysis (reduced-scope plant).	Soil failure analyses are not necessary for reduced-scope plant.	None.
H.B. Robinson 2	Very deep (460 ft) soil site, dense below 50 ft depth; some structures are supported on piles to a depth of 50 ft; the circulating water intake structure is founded on 50-ft depth.	New, multiple SSI analyses using CLASSI conducted for five Class I structures.	Some data points indicated liquefaction at isolated location, which the licensee concluded was acceptable. (However, after a Step 2 review of the licensee's IPEEE, liquefaction remained as an issue requiring further investigation, which eliminated the concern.)	Embankment failure and wave-induced strains in buried piping, were considered and judged by the licensee not to be significant.
Haddam Neck	Predominantly a rock site; the new DG and switchgear buildings are founded on shallow soil; diesel fuel oil tanks and piping are buried.	SSI conducted for new DG and switchgear buildings.	No concerns identified.	No concerns identified.

Table 2.10: Soil evaluation (Continued)

Plant	Soil/foundation Characteristics	SSI or soil response analysis	Soil liquefaction	Other soil failure modes
Hatch 1&2	Soil. Plant site underlain by approximately 4000 ft of relatively unconsolidated Mesozoic and Cenozoic sands, gravel, clay marls, claystone, sandstones, and limestone.	New SSI analyses performed for Control building, Reactor buildings, Diesel Generator building, and Intake building. The substructuring approach was applied and uncertainties associated with soil properties were addressed.	The soil evaluation addressed issues related to liquefaction potential and ground settlement, and found that the RLE should be reduced to 0.28g to meet the requirement for the minimum values of the factor of safety. It would not have any detrimental effect because it would occur at a depth where HCLPF is controlled by the impact of ground settlement and differential settlement on buried structures and pipe penetrations.	Stability of the soil slopes in the river intake area was evaluated and was found unlikely to experience a serious stability problem following an RLE.
Hope Creek	Soil. Soil improvement conducted on the site by replacing the loose hydraulic fill with engineered backfill, underlying which is the Kirkwood formation consisting of fine to medium grained sands having blow counts ranging from 20 to 70 blow counts per foot.	A new SSI analysis using a probabilistic approach to account for the variabilities in soil and structural properties. The SSI effects and the spectral shape of the UHS are considered to be the main contributors for the building response reduction.	The liquefaction potential was assessed using a probabilistic approach. The lateral spreading due to liquefaction of slopes becomes significantly large at a peak acceleration of about 0.35g.	None.
Indian Point 2	Rock site; some piping is laid in trenches that were excavated in rock and backfilled.	Design-basis fixed-base structural models were used; SSI effects were deemed negligible.	A specific evaluation for liquefaction and slope failures was performed according to EPRI NP-6041. No concerns were identified.	Potential failure of diesel fuel oil tanks, due to hold-down strap failure and failure of grouted rock anchors, was modeled in the seismic PRA.
Indian Point 3	Rock. The site consists of a hard limestone formation which provides a solid bed for the plant foundation.	The structures are founded on bedrock. New floor response spectra were developed using the so-called direct generation method.	Not applicable.	None.
Kewaunee	Clay-sand soil deposited to a depth of 76 ft.	Elastic half-springs used to model soil behavior.	Assessed as being very unlikely.	Screened out based on high seismic capacity.
La Salle 1&2	No information provided.	No information provided.	No information provided.	No information provided.
Limerick 1&2	Rock site.	None reported.	None.	None.
McGuire 1&2	Category I structures: Rock or concrete fill extending to rock. Some components founded on soil/backfill.	None (deemed negligible).	No concerns identified.	No concerns identified.
Millstone 2	Rock	IRS generated using either a direct generation method or a scaling method to convert the SSE IRS to the RLE IRS.	Not applicable.	None.

Table 2.10: Soil evaluation (Continued)

Plant	Soil/foundation Characteristics	SSI or soil response analysis	Soil liquefaction	Other soil failure modes
Millstone 3	Soil site (specific characteristics were not described); beach and glacial outwash sands adjacent to SW pumphouse.	Not described.	No concerns identified for power block; soil adjacent to SW pumphouse assumed to fail, but determined not to impair function.	Included in fragility evaluations.
Monticello	Moderately deep soil site.	New SSI analyses of turbine and reactor building were performed for USI A-46, using existing structural dynamics models (with minor modifications). The results of these analyses were scaled from the SSE to RLE. For the control building, new SSI results were computed directly for the RLE. Three different soil profiles (best estimate, upper, and lower bound) were used.	No soil evaluation performed.	No soil evaluation performed.
Nine Mile Point 1	Rock	Safety-related buildings founded on rock. The RLE IRS were developed by scaling up the upgraded ground spectrum (which is a NUREG/CR-0098 50% spectral shape and is anchored to 0.13g) to 0.3g RLE using a scaling factor of 2.31 (i.e., 0.3/0.13).	Not applicable.	None.
Nine Mile Point 2	Rock site.	None.	No concerns identified.	None.
North Anna 1&2	Structures founded on rock: Containment structure and internals; reactor Safeguards Structure; Main Steam Valve House, Unit 1; AFW Pumphouse. Structures founded on soil: Main Steam Valve House, Unit 2; service water Pumphouse; service water Valvehouse; Auxiliary Bldg., Intake Structure. Also, Service/Turbine Bldg. is founded partially on rock and partially on soil.	Structures were modeled using lumped-mass beams and stiffness matrix elements with 6 degrees of freedom at each node. For structures founded on soil foundations, the building models were used together with the proper impedance and scattering functions for the soil-structure interaction (SSI) effects, and SSI analyses were performed for the best estimate and lower and upper bounds of soil properties.	NAPS is a focused-scope plant that is not required to perform soil evaluation.	None.

Table 2.10: Soil evaluation (Continued)

Plant	Soil/foundation Characteristics	SSI or soil response analysis	Soil liquefaction	Other soil failure modes
Oconee 1,2,&3	Rock/soil. The site has a shallow soil layer over bedrock. All major structures are founded on rock, except for the transformer CT-4, blockhouse, borated water storage tank, main startup transformer, condenser circulating water piping, relay house, and switchyard.	All major safety-related structures at Oconee are founded on bedrock, the shallow overburden soil was not modeled. Scaling of existing results using the UHS scaling factor (ratio of the UHS to NUREG/CR-0098 spectrum).	Soil liquefaction was addressed by reviewing the existing geotechnical studies, and concluded that no concerns were found for liquefaction.	None.
Oyster Creek	Soil	A new set of soil-structure interaction analyses was performed. The analyses are based on the SSMRP-type approach, and the variabilities in both the ground motions and structural properties are considered in multiple time history analyses.	The likelihood of soil liquefaction for varying water table conditions was evaluated and expressed in terms of probabilities of occurrence conditional on the occurrence of a given ground acceleration. It is estimated that soil liquefaction is expected to occur at a peak ground acceleration of 0.40 g at the locations of the EDGB and the fire protection piping.	None.
Palisades	150-160 ft of soil (dense fine sands, over very dense fine sands, over hard silty clay and glacial till) over shale bedrock.	New 3D nonlinear SSI analyses; also SHAKE computer code used for ground response analyses.	No concerns identified.	Screened out soil displacements and settlements.
Palo Verde 1,2,&3	Soil	A new set of soil-structure interaction (SSI) analyses was performed at PVNGS for the 0.3g RLE.	The potential for liquefaction of cohesionless soils that underlie the site was evaluated and the results of the analysis showed factors of safety against liquefaction of approximately 2.5 and higher for a peak ground acceleration of 0.3g. Earthquake-induced settlements were also addressed and found to be negligible	None.
Peach Bottom 2&3	Predominantly a rock site; there are some buried piping and equipment at the plant.	No SSI analyses; fixed-base, lumped-mass dynamic models were employed.	No concerns identified.	Failures of buried equipment and piping were screened out.
Perry 1	Safety-related buildings are founded on rock, except for the diesel generator building, which is founded on compacted Class A backfill.	The RLE In-structure Response Spectra (IRS) are generated by scaling the DBE IRS.	Exempt from performing soil evaluation.	None.
Pilgrim 1	30 to 50 feet of heavily compacted fill materials above 30 to 50 feet of very dense glacial outwash deposits underlain by bedrock.	New 3D SSI analyses.	No concerns identified.	Soil settlements and foundation rocking of CST were modeled in fragility calculations.

Table 2.10: Soil evaluation (Continued)

Plant	Soil/foundation Characteristics	SSI or soil response analysis	Soil liquefaction	Other soil failure modes
Point Beach 1&2	100 ft of soil (stiff to very stiff glacial deposits) over fractured dolomite bedrock.	New 3D nonlinear SSI analyses.	Assessed as being very unlikely.	Soil settlements and displacements affecting components were screened out.
Prairie Island 1&2	Soil	Soil-structure interaction (SSI) analysis performed using the computer programs CLASSI and SHAKE.	No soil evaluation was conducted (based on Supplement 5 to GL 88-20).	None.
Quad Cities 1&2	Predominantly a rock site; there are some buried piping and equipment at the plant.	No SSI analyses were performed.	No concerns identified.	Effects of differential movements were qualitatively screened out; retaining wall for intake/discharge building was screened out based on a factor of safety of 1.2 beyond the design basis.
River Bend	Soil	SSE analysis (for reduced-scope plant).	Soil failure analyses not necessary.	None.
Salem 1&2	The SGS is built on an artificial island, and most of the Seismic Category I structures are founded on a common lean concrete mat poured within the confines of a cellular coffer dam.	New SSI analyses were performed for the containment building including internal structures, auxiliary building, and the service water intake structure. Variabilities in stiffnesses and damping of both structures and soil were considered in the analyses based on a Latin Hypercube Simulation.	The liquefaction potential was assessed using a probabilistic approach. An HCLPF of 0.72 was estimated and used to evaluate the fragility of buried piping.	The liquefaction of slopes due to lateral spreading appears to be initiated at about a peak ground acceleration of 0.35g. It is not clear whether this information is used for evaluation of buried piping.
San Onofre 2&3	Very deep (over 900 ft) soil site, consisting of stiff, well-graded sands. During plant construction, about 70 ft of native soils (terrace deposits) were excavated, and plant structures were founded directly on the underlying stiff sand deposit/formation.	New soil-structure-interaction (SSI) dynamic response calculations, based on probabilistic characterization of soil properties, were performed using existing 3D dynamic stick models.	SSE analyses were used as basis to screen out (at an acceleration of 5.4g S-sub a (1-10 Hz)), potential liquefaction of filled cavities adjacent to, or beneath, structures. Potential development of blockages in offshore conduit caused by conduit separation and inflow of liquefied soils was screened out since water velocity in the conduits was determined to be sufficiently high to remove any potential sand blockage.	Potential ground failure of sands in the plant area was screened out based on consideration of soil properties. Results of slope response analyses were used to screen out concerns that failure of cut slopes in native terrace deposits could cause excessive ground movements at adjacent critical plant facilities.
Seabrook	Rock or concrete filled to rock	Structures are founded on either rock or concrete filled to rock. Both response spectrum and modal time-history analyses were performed for the Category I structures.	Since seismic Category I structures are founded on either rock or concrete filled to rock, the soil liquefaction potential is not an issue for Seabrook.	None.

Table 2.10: Soil evaluation (Continued)

Plant	Soil/foundation Characteristics	SSI or soil response analysis	Soil liquefaction	Other soil failure modes
Sequoyah 1&2	Rock site predominantly; some structures are founded on shallow soil (clays and silts over shale bedrock).	New, probabilistic evaluations of response, including SSI effects, using stick models.	Assessed as having low susceptibility.	Compaction/settlement and failures due to slope instability were considered, but not assessed as being important.
Shearon Harris 1	Rock	All seismic category I structures at SHNPP founded on rock. IRS scaled up from SSE to RLE, using the method for scaling IRS as outlined in EPRI NP-6041-SL	Not applicable.	Two dams, which are located in the Buckhorn Creek watershed to impound cooling water for SHNPP, were evaluated for RLE, and found to have an HCLPF of 0.31g.
South Texas Project 1&2	Very deep soil deposit.	SSI finite element analysis.	No analysis required (reduced-scope plant).	No analysis required (reduced-scope plant).
St. Lucie 1&2	Category I structures: founded on Category I fill, underlain by cemented sands and sandy limestones.	Soil modeled using translational and rotational springs.	No analysis required (reduced-scope plant).	No analysis required (reduced-scope plant).
Summer	Rock/soil.	Most of the structures are founded on rock. For structures and components founded on soil, the defined RLE spectrum was increased to an effective PGA of 0.5g by a factor of 1.67. Soil conditions were applied to the service water pump house (SWPH) and condensate storage tank (CST).	Summer is a focused-scope plant that is not required to perform soil evaluation.	A HCLPF calculation was performed for the earth dams and embankment that impound cooling water for the plant and are treated as Seismic Category I structures. The HCLPF capacity is 0.22g for the earth dams and embankment.
Surry 1&2	Soil. The site is bordered by the James River on either side of the peninsula, and characterized as a deep soil site.	New floor spectra were obtained by SSI analyses. The Latin Hypercube sampling technique was used to account for the uncertainties in frequencies, damping, and soil properties.	No soil evaluations were performed (based on Supplement 5 of GL 88-20).	None.
Susquehanna 1&2	All Category I structures are founded on rock, except the essential station service water (ESSW) pumphouse and the spray pond	SSI model used for ESSW pumphouse; flexible-base model used for reactor building; fixed-base model used for other structures.	No concerns identified.	No concerns identified with settlements, instability, sliding, or distortion of buried pipe.

Table 2.10: Soil evaluation (Continued)

Plant	Soil/foundation Characteristics	SSI or soil response analysis	Soil liquefaction	Other soil failure modes
TMI 1	Soil	Structural response analysis performed by EQE.	A soil liquefaction analysis and a seismic soil settlement analysis were performed. These analyses concluded that, should soil liquefaction occur, foundation-bearing strength failures are not expected, but ground settlement on the order of 0.5 to 1.0 inch could be expected. For the structures founded on compacted backfill, including the DG building, the borated water storage tank, the condensate storage tanks, and the turbine building, the analysis concludes that it is unlikely that such a small soil settlement would lead to significant structural damage.	None.
Turkey Point 3&4	Rock site predominantly.	None described.	No analysis required (reduced-scope plant).	No analysis required (reduced-scope plant).
Vermont Yankee	Rock. All of the Seismic Category I structures are founded either on mat foundations bearing directly on bedrock, or on a grillage of grade beams over a series of reinforced concrete piers on bedrock.	Structures founded on bedrock. The RLE In-structure Response Spectra (ISRS) were generated by scaling up the design basis ISRS.	Not applicable.	A stability analysis of the Vernon Dam showed a factor of safety equal to 1.12 for the overturning mode against 0.3g RLE.
Vogtle 1&2	Deep soil site; the depth of bedrock below the plant site is approximately 950 ft.	Scaling of the design basis earthquake analysis results.	Not required to perform soil evaluations. However, since the VEGP site has a deep soil profile (950 ft) and is situated on the bank of the Savannah River, liquefaction should be an important safety issue. The licensee has performed soil evaluations for the site, including liquefaction potential, stability of slopes, and ground settlement. The method proposed by EPRI for soil evaluations in EPRI NP-6041-SL was used. The results presented appear to be reasonable and the factor of safety against soil liquefaction potential at the RLE is approximately 1.5.	None.
Waterford 3	Soil	Design basis analysis for SSE (for reduced-scope plant).	Soil failure analyses are not necessary for reduced-scope plants.	None.
Watts Bar 1	Rock/Soil. All of Seismic Category I buildings at WBN are founded on rock, except for the Diesel Generator Building which is supported on a soil foundation.	Scaling of existing ARS to the RLE. The DGB is supported on soil, and its seismic response should involve the soil-structure interaction (SSI) effect. The submittal did not discuss how the SSI effect was considered in the spectral scaling.	Exempted from performing soil evaluation.	None.

Table 2.10: Soil evaluation (Continued)

Plant	Soil/foundation Characteristics	SSI or soil response analysis	Soil liquefaction	Other soil failure modes
Wolf Creek	Predominantly a rock site; Seismic Category I structures are founded on shallow soil columns over bedrock. The soil overburden is less than 16 feet.	FLUSH finite element analysis for power block structures.	No evaluation performed.	Relative displacement of buried piping between the power block and the emergency service water pumphouse was evaluated. A review of design documents led to the judgment that the interaction of the piping with the associated structures can be accommodated at the RLE.

* Formerly known as Washington Nuclear Project Number 2.

Table 2.11: Non-seismic failures and human actions

Plant	Treatment in systems modeling	Screening criteria	Impacts of ground motion of HEPs	Human actions unique to seismic events
Arkansas Nuclear One 1	Success path selection; safe shutdown systems chosen in the IPEEE are of minimal complexity and effort to operate, and are consistent with the normal ANO-1 Emergency Operating Procedures.	The overall core damage frequency (CDF) from a 0.3 PGA event is assessed and found to be insignificant, and a review of the equipment selected in the SPLDs does not show any concern of nonseismic failures.	The potential adverse environmental conditions, such as the potential for failure of plant structures and equipment, the potential for local failure of architectural features, and the potential for adverse seismic spatial interactions in the vicinity of safe shutdown equipment, are also considered.	Local manual operations related to the operation of the steam-driven EFW pump and those related to the local starting of the emergency diesel generator are credited in the IPEEE.
Arkansas Nuclear One 2	Success path selection. The safe shutdown systems chosen in the IPEEE are of minimal complexity and effort to operate, and are consistent with the normal ANO-2 Emergency Operating Procedures.	The overall core damage frequency (CDF) from a 0.3 PGA event is assessed and found to be insignificant. A review of the equipment selected in the SPLDs does not show any nonseismic failure concerns.	An assessment of the overall CDF from a 0.3g PGA event, including the consideration of potentially adverse environmental conditions, such as loss of lighting, show insignificant CDF.	Local manual operations credited in the IPEEE are those related to the operation of the steam-driven EFW pump and those related to the local starting of the emergency diesel generator.
Beaver Valley 1	IPE model	Not applicable.	The human error probability (HEP) values from the internal PRA model were used up to an acceleration level of 0.5g. Above an acceleration level of 0.5g the HEPs were set to 1.0.	None reported.
Beaver Valley 2	IPE model	Not applicable.	The human error probability (HEP) values from the internal PRA model were used up to an acceleration level of 0.5g. Above an acceleration level of 0.5g the HEPs were set to 1.0.	None reported.
Braidwood 1&2	Success path selection. The selected success paths utilized equipment and operator actions consistent with current plant operating procedures that were evaluated in the Braidwood station IPE.	A review of the success paths and systems selection does not reveal any concern with using single-train systems with recognized poor availability problems on the success path.	Important operator actions involved in the selected success paths are evaluated and found to be consistent with current plant procedures for which the operators are regularly trained.	The switchover of the AFW suction from the CST to the ESW.
Browns Ferry 2	Success path selection. To minimize the number of components to be evaluated, only low pressure injection systems are selected in the success paths.	None reported. The success path depends on a single train of the RHR system, which does not have any reported reliability problem.	Operator actions required for the success paths and their failure probabilities obtained from the plant probabilistic risk assessment are discussed.	Manual RPV depressurization and low pressure system initiation are required for the success paths selected in IPEEE.

Table 2.11: Non-seismic failures and human actions (Continued)

Plant	Treatment in systems modeling	Screening criteria	Impacts of ground motion of HEPs	Human actions unique to seismic events
Brunswick 1&2	Success path selection; location and timing of required operator actions were reported.	Random failures having probability exceeding 0.001 and existing in a significant cutset were screened in; no screening was performed for human actions, although the success paths were chosen to maximize operator familiarity and redundancy.	None modeled.	Chatter recovery.
Byron 1&2	Success path selection. The selected success paths utilized equipment and operator actions consistent with current plant operating procedures that were evaluated in the Byron Station IPE	A review of the success paths and systems selection does not reveal any concern with using single-train systems with recognized poor availability problem on the success path.	Important operator actions involved in the selected success paths are evaluated and found to be consistent with current plant procedures for which the operators are regularly trained.	The switch over of the AFW suction from the CST to the ESW.
Callaway	Success path selection.	Random or human failures having probability exceeding 0.001, if failure impacts multiple trains or systems, or 0.01, if failure impacts only a single train and system, were screened in.	None modeled.	Not documented.
Calvert Cliffs 1&2	IPE model.	Not applicable.	Considering how different performance shaping factors (PSFs) are affected at different g levels. Recovery actions were not modeled.	None reported.
Catawba 1&2	IPE model.	Not applicable.	None.	Relay chatter recovery.
Clinton	Success path selection.	None reported. There are sufficient redundancy and reliability in both success paths selected in the IPEEE.	There are no immediate operator actions required for the success paths. Operator actions can be performed in the main control room. They are not time critical and are proceduralized and trained upon.	None reported.
Columbia Generating*	IPE model.	Not applicable.	Human error was considered in the analysis by increasing the human error probability (HEP) values used in the IPE study by roughly a factor of 10 to account for the extreme stress during a seismic event. No credit was taken in any of the accident sequences for recovery actions.	None reported.
Comanche Peak 1&2	Success path selection.	Qualitative screening: success paths involve dual-train systems and actions are familiar to operators.	None.	None reported.

Table 2.11: Non-seismic failures and human actions (Continued)

Plant	Treatment in systems modeling	Screening criteria	Impacts of ground motion of HEPs	Human actions unique to seismic events
Cooper	Success path selection. Only low pressure systems are selected. High pressure injection systems are not included in the SSEL. This is based on the reasoning that depressurization may be operationally desirable during a postulated RLE scenario.	Nonseismic related failures were evaluated using the data from the internal event PSA.	A review of all actions in the PSA model was performed in the IPEEE to consider their importance to the expected plant post-seismic reliability. The increased level of stress to the operators and the potential disruption in the Control Room (e.g., falling ceiling tiles and items falling off of shelves) after an SME was taken into consideration.	Manual RCS depressurization is required for both success paths.
Crystal River 3	Success path selection.	Nonseismic failures are addressed by the selection of two success paths and the redundant components in the systems selected in the success paths.	Through a validation process, operators ensure that the plant procedure steps can be performed and are adequate to manage the emergency situation.	None reported.
D.C. Cook 1&2	IPE model.	Not applicable.	None documented.	None documented.
Davis-Besse	Success path selection.	The licensee states that according to the EPRI methodology, each success path should have an unavailability of less than 0.001.	There is no discussion of important human actions. There does not seem to be any credit given to any type of recovery action, except for the restarting of the HPI or the makeup pump, after it trips due to relay chatter.	The restarting of the HPI or the makeup pump, after it trips due to relay chatter.
Diablo Canyon 1&2	IPE model with unique seismic impacts introduced in seismic event tree.	Not applicable.	HEPs increased based on spectral acceleration.	Relay chatter recovery and others.
Dresden 2&3	Success path selection. Isolation condenser is used for decay heat removal because the cooling water to the LPCI heat exchangers is lost following a dam failure.	None reported. The selected success path for some of the safety functions relies on a single train of a safety system. It is not expected to cause a significant nonseismic failure concern because of the reliability of the LPCI system.	The success paths selected in the IPEEE are generally consistent with those the operators are likely to perform under accident conditions.	Operator actions will be required for the proposed seismically qualified/verified makeup path to the isolation condenser.
Duane Arnold	Success path selection. High pressure injection systems are not included as safe shutdown systems.	None reported.	Some discussion of operator actions is presented in the DAEC IPEEE, including actions that were disallowed.	None reported.
Farley 1&2	Success path selection.	There is sufficient diversity because both a small LOCA and a LOOP are considered in both paths, and feed-and-bleed cooling is considered as an alternative to the steam generator cooling.	Equipment which requires operator actions was included in the seismic evaluation. No credit was given for recovery of offsite power loss.	None reported.

Table 2.11: Non-seismic failures and human actions (Continued)

Plant	Treatment in systems modeling	Screening criteria	Impacts of ground motion of HEPs	Human actions unique to seismic events
Fermi 2	Success path selection. Both HPCI and RCIC systems are included in the preferred success path for RCS inventory control.	There is sufficient redundancy and diversity in systems/components selected for the success paths. Both HPCI and RCIC systems are included in the preferred success path for RCS inventory control.	The selection of the success paths takes into consideration plant procedures and training, as well as instrumentation and indication available following a seismic event.	None reported.
FitzPatrick	Success path selection. The initial component list was developed based on the recent IPE study and the A-46 component list.	The screening of nonseismic failures was conducted by assigning threshold values of 10E-2 to 10E-4, depending on the redundancy of the component.	A model is developed which correlates the human error probability (HEP) to the peak acceleration level.	None reported.
Fort Calhoun 1	IPE model.	Not applicable.	HEPs increased based on spectral acceleration.	Not reported.
Ginna	Success path selection.	Nonseismic failures are addressed by the redundant components in the systems selected in the success paths. All of the equipment relied upon is the normal equipment set used in the plant emergency operating procedures.	The effects of the potentially adverse environmental conditions during a seismic event on operator actions have been addressed.	None reported.
Grand Gulf 1	Success path selection. The selection was based on operational and systems considerations originally developed for the IPE.	The NUREG/CR-4826 screening approach for single-train/multiple-train systems is used.	The success paths are based on highly successful operational sequences.	None reported.
H.B. Robinson 2	Success path selection; location and timing of required operator actions were reported.	Random failures having probability exceeding 0.001 and existing in a significant cutset were screened in; no screening was performed for human actions, although the success paths were chosen to maximize operator familiarity and redundancy.	None modeled.	Chatter recovery.
Haddam Neck	IPE model.	Not applicable.	HEPs increased for seismic events; operator fragility curves were developed.	Response to seismic failures of upstream dams.
Hatch 1&2	Success path selection. HPCI is the only high pressure system included in the success path.	Nonseismic failures are addressed by the redundant components in the systems selected in the success paths.	Qualitative discussions on operator actions and event timing for some events are provided in the submittal.	None reported.
Hope Creek	IPE model.	Not applicable.	The human actions were modeled such that the internal PRA human error probabilities (HEPs) were raised by a factor of 10.	Recovery of 1E 120Vac instrumentation distribution panel; recovery of long-term cooling in the switchgear room; operator shutdown from the remote shutdown panel.

Table 2.11: Non-seismic failures and human actions (Continued)

Plant	Treatment in systems modeling	Screening criteria	Impacts of ground motion of HEPs	Human actions unique to seismic events
Indian Point 2	IPE model, modified for increased mission times and assumed unavailabilities due to the seismic event.	Not applicable.	Due to assumed unavailabilities, some IPE human actions were excluded from the model. IPE HEPs were used for actions required after 1 hour; for actions required within 1 hour, simple amplification factors were applied to IPE HEPs.	None.
Indian Point 3	IPE model with consideration of a 72-hour mission time.	Not applicable.	For post-initiator events, the human failure probabilities are assumed to have the same values as those used in the IPE for seismic levels less than or equal to the DBE, twice the IPE values for seismic hazard levels between 0.15g (DBE) and 0.5g, and 10 times the IPE values for a seismic hazard level at 0.5g. Beyond 0.5g, a failure probability of 0.1 is used for in-control-room human actions and 1.0 for actions outside the control room. Restoration of offsite power is not considered in the IPEEE.	None reported.
Kewaunee	IPE model.	Not applicable.	HEPs increased for seismic events, as based on simplified operator error fragilities.	None documented.
La Salle 1&2	IPE model.	Not applicable.	None documented.	None documented.
Limerick 1&2	Success path selection.	No screening; success paths were chosen considering redundancy and operator familiarity.	None.	None reported.
McGuire 1&2	IPE model.	Not applicable.	None.	Relay chatter recovery.
Millstone 2	Success path selection. The final SSEL includes almost all components modeled in the MP2 Internal Event PRA. The components in the SSEL that were determined to potentially not be able to meet the RLE with high confidence were also determined not to be required for safe shutdown.	Nonseismic failure is not an issue because of the availability of multiple success paths using redundant and diverse systems.	Requirements regarding nonseismic failures and human actions are consistent with the description of NUREG-1407.	None reported.
Millstone 3	IPE model.	Not applicable.	None documented.	Relay chatter recovery.

Table 2.11: Non-seismic failures and human actions (Continued)

Plant	Treatment in systems modeling	Screening criteria	Impacts of ground motion of HEPs	Human actions unique to seismic events
Monticello	Success path selection; location and timing of required operator actions were reported; random failure rates for key equipment were reported.	No screening was performed. Random failure of the Division I diesel generator and its supports is a single point of failure in one success path.	None modeled.	None documented.
Nine Mile Point 1	Success path selection. The success path chosen emphasizes simplicity in the frontline systems and minimizes the number of support systems and operator actions that needed to be evaluated. The two success paths consist of redundant trains of the same equipment.	The components in the success path with the highest nonseismic unreliability are the diesel generators. The combined unreliability of the diesel generators is still low (<5%).	Very few operator actions are credited, and most of them are long term (i.e., several hours).	Credit is given for some relay chatter recoveries. Otherwise, very few operator actions are credited, and most of them are long term (i.e., several hours). Examples are recovery of diesel generator room cooling, and intermittent operation (including raw water flow path alignment) of the containment sprays.
Nine Mile Point 2	IPE model.	None.	Not specified.	Not specified.
North Anna 1&2	Success path selection.	Components and systems that were identified in the IPE report as important for core damage risk reduction, significant for risk achievement, or with high probability of failure were walked down and evaluated.	The emergency and abnormal procedures include reliance on operator actions for safe shutdown following a seismic event.	Manual valve operations if offsite power is lost and procedural action by operators in case of possible loss of annunciator lights during strong motion of an earthquake.
Oconee 1,2,&3	IPE model.	Not applicable.	The effect of the earthquake on HEPs has been considered in the analysis.	Not reported.
Oyster Creek	IPE model.	Not applicable.	No recovery actions were credited except for the near offsite power recovery via the combustion turbines. Otherwise, the same human error probability (HEP) values used in the IPE were used here.	None reported.
Palisades	IPE model.	Not applicable.	HEPs increased for seismic events, as based on simplified operator error fragilities that account for location and timing of actions.	None documented.

Table 2.11: Non-seismic failures and human actions (Continued)

Plant	Treatment in systems modeling	Screening criteria	Impacts of ground motion of HEPs	Human actions unique to seismic events
Palo Verde 1,2,&3	Success path selection. Because of the lack of the PORVs for feed and bleed operation, the Palo Verde analysis considered an additional system, called the N Train AFW system, which is not seismically qualified in both success paths.	Nonseismic failures were addressed in the IPEEE (by the reliability demonstrated in the IPE of the systems selected in the success paths).	It was concluded in the IPEEE that the operators are trained and can be relied upon to achieve cold shutdown following a loss of offsite power event, and there is high confidence that these actions would be performed within the available time.	None reported.
Peach Bottom 2&3	Success path selection; location and timing of required operator actions were reported; random failure rates for key equipment were reported.	None applied. Success paths were chosen to ensure that any required human actions are familiar to the operators, and to ensure redundancy in equipment. Success paths were developed to rely upon procedures that (a) are available from the main control room. and (b) operators are trained in.	None modeled.	None documented.
Perry 1	Success path selection. The existing Perry probabilistic risk assessment (PRA) was used as the basis for identifying the systems and components for the IPEEE.	Nonseismic failures are addressed by the redundant components in the systems selected in the success paths.	Plant procedures and indications (e.g., annunciators in the control room) are available for required operator actions. There is also sufficient time for these actions.	The restoration of the HPCS DG should the low-ruggedness relay cause it to trip and the implementation of containment venting for success path; and manual depressurization of the RPV and implementation of containment over pressurization protection using either the RHR system or containment venting.
Pilgrim 1	IPE model.	Not applicable.	HEPs increased for seismic events, as based on simplified operator error fragilities that account for location of actions.	Relay chatter recovery.
Point Beach 1&2	IPE model, with unique seismic effects modeled in the entry seismic event tree.	Not applicable.	HEPs increased for seismic events, as based on simplified operator error fragilities that account for location of actions.	Some actions modeled, but none related to relay chatter recovery.
Prairie Island 1&2	Success path selection.	The success paths were chosen based on the screening criterion applied to nonseismic failures and human actions. Nonseismic failure probabilities of the systems selected in the SSEL are presented.	Operator actions required for the critical safety functions, the time in which the action must be completed, and the location in the plant in which the action must take place are discussed in the submittal.	Operator actions are required to reduce the system flow of the cooling water system to below the capacity of the emergency intake line if the normal path from the Mississippi River through the outer Screenhouse is blocked or if Lock/Dam # 3 fails.

Table 2.11: Non-seismic failures and human actions (Continued)

Plant	Treatment in systems modeling	Screening criteria	Impacts of ground motion of HEPs	Human actions unique to seismic events
Quad Cities 1&2	Success path selection; location, timing, and expected reliability of required operator actions were reported. The treatment of random equipment failures was not reported.	No quantitative criteria were applied.	None modeled.	None documented.
River Bend	Success path selection.	Both HPCS and RCIC systems are included in the success path for high pressure injection. Nonseismic failures are addressed by the redundant components in the systems selected in the success paths.	Operator actions required for the success paths are discussed in the submittal. Important operator actions include RCS depressurization using ADS and initiation of the SPC mode of the RHR system. These actions are included in the plant procedures and are likely to be performed by control room operators.	None reported.
Salem 1&2	IPE model.	Not applicable.	Most human error probabilities were kept at the same values as in the IPE model. No power recoveries were allowed within the first 24 hours.	None reported.
San Onofre 2&3	IPE model with unique seismic impacts introduced in seismic event tree.	Not applicable.	A severe earthquake ground motion was assumed in developing HEPs. Performance shaping factors were developed based on required timing of action.	Relay chatter recovery; start redundant SWC pump; align fire truck to CCW make-up, given failure of primary make-up tank; respond to high-temp alarm in the SWGR/distribution room; and open SWC emergency discharge line to seawall, given gate failure.
Seabrook	IPE model	Not applicable	Not specifically discussed in the submittal. No credit is given for recovery actions for loss of offsite power, diesel generators, or ATWS.	None reported.
Sequoyah 1&2	Success path selection.	No screening; success paths were chosen considering redundancy and operator familiarity.	None modeled.	Relay chatter recovery.

Table 2.11: Non-seismic failures and human actions (Continued)

Plant	Treatment in systems modeling	Screening criteria	Impacts of ground motion of HEPs	Human actions unique to seismic events
Shearon Harris 1	Success path selection.	The equipment train reliability is qualitatively considered and only the most reliable alternative is chosen for the systems selected in the success paths.	The development of the success path evolved from studying available plant equipment functions as well as the plant's normal and emergency operating procedures, and was reviewed and agreed upon by plant operations personnel.	None reported.
South Texas Project 1&2	IPE model.	Not applicable.	Not specified.	Relay chatter recovery.
St. Lucie 1&2	Operating procedures were reviewed in developing success paths.	None.	None modeled.	None considered.
Summer	Success path selection. The two success paths share many of the same systems.	Nonseismic failures were discussed, but not in sufficient detail. Diversity in the success path systems (between the two success paths) is limited.	The system selection process is consistent with plant operator utilization of procedures, training, and available instrumentation indicators to not affect the seismic margin evaluation.	None reported.
Surry 1&2	IPE model.	Not applicable.	Adjusted to account for additional stresses after an earthquake.	Conserve intake canal inventory, initiate RHR with loss of instrument air, initiate steam dump by opening the steam dump valves and stop the AFW pumps to prevent pump damage after the suction is dry following a seismic event.
Susquehanna 1&2	Success path selection.	No screening; failure probabilities are reported as being consistent with screening values used in the Maine Yankee SMA; HPCI and RCIC have a high combined failure probability of 0.0024 per demand; manual starting of residual heat removal service water (RHRSW) pumps is a key action.	None modeled.	None reported.
TMI 1	IPE model.	Not applicable.	Recovery is not allowed for systems failed by seismic causes. The effects of human action failure rates are evaluated in sensitivity studies.	Loss of onsite ac power due to relay chatter is considered in the seismic model, and recovery from relay chatter is added to the seismic model.
Turkey Point 3&4	Operating procedures were reviewed in developing success paths.	None.	None modeled.	None considered.

Table 2.11: Non-seismic failures and human actions (Continued)

Plant	Treatment in systems modeling	Screening criteria	Impacts of ground motion of HEPs	Human actions unique to seismic events
Vermont Yankee	Success path selection.	The selected success path for some of the safety functions relies on a single train of a safety system. There is not a significant nonseismic failure concern because of the reliability of the selected systems.	Existing plant operating and emergency procedures were used during the development of this SSEL. Every effort was made to minimize any actions or equipment use not covered by existing procedures.	None reported.
Vogtle 1&2	Success path selection.	Nonseismic failures are addressed by the redundant components in the systems selected in the success paths.	The selected success paths were reviewed by plant operations personnel to ensure that they are compatible with plant operations procedures and operator training.	None reported.
Waterford 3	Success path selection. There is a heavy reliance on secondary cooling for success, probably due to the limitation of the available systems. For example, feed and bleed cannot be performed at Waterford because of the lack of pressurizer PORVs, and the CVCS pumps do not have sufficient capacity to mitigate a small LOCA condition.	The selected success path for some of the safety functions relies on a single train of a safety system. There is not a significant nonseismic failure concern because of the reliability of the selected systems.	Required operator actions are likely to be carried out because the requirements of the systems for the safety functions are developed from a review of plant procedures. In addition, no out-of-control-room operator actions are required to accomplish a safe shutdown using the success paths.	None reported.
Watts Bar 1	Success path selection.	Nonseismic failures are addressed by the diversity and redundancy in the equipment selected in the SSEL for the success paths.	Operator actions required to achieve the success paths are those normally included in the operator training program and trained on by operators. This allows human error failures to be screened out using the guidelines identified in NUREG/CR-4826.	None reported.
Wolf Creek	Success path selection; location and timing of required operator actions were reported.	No screening; success paths were chosen to make use of high-reliability equipment; all operator actions are performed in the control room, except one (30 minutes are available to perform this action).	Qualitative assessment indicated that the reliability of one human action (required within 5 minutes) might be impaired by the RLE ground motion event. However, the safety impact was judged to be small.	None identified.

* Formerly known as Washington Nuclear Project Number 2.

Table 2.12: Seismic-fire interaction and seismic-flood interaction

Plant	Evaluation approach	Seismic-fire observations/outliers	Seismic-flood observations/outliers	Related plant improvements
Arkansas Nuclear One 1	Walkdown for seismic-fire and seismic-flood concerns.	Potential safety concerns with hydrogen pipe rupture (in the turbine generator and the makeup tank) and flammable liquids identified and dismissed.	Potential failure of dams evaluated and found adequately considered in the development of the Probable Maximum Flood, per the Standard Review Plan (SRP).	None.
Arkansas Nuclear One 2	Walkdown for seismic-fire and seismic-flood concerns.	Potential safety concerns with hydrogen pipe rupture (in turbine generator and volume control tank) and flammable liquids identified and dismissed.	Potential failure of dams evaluated and found adequately considered in the development of the Probable Maximum Flood, per the SRP.	None.
Beaver Valley 1	Seismic walkdown and frequency consideration.	Seismically induced fires screened out based on comparison with the frequency of initiation of internal fires. Fire suppression equipment not found to be a seismic concern by walkdown.	The failure of the Conemaugh Dam, considered the worst case scenario was evaluated and found not to be a problem for the site.	None.
Beaver Valley 2	Seismic walkdown and frequency consideration.	Seismically induced fires screened out based on comparison of equipment HCLPF with the frequency of initiation of internal fires. Fire suppression equipment not found to be a seismic concern by walkdown.	The failure of the Conemaugh Dam, considered the worst case scenario was evaluated and found not to be a problem for the site.	None.
Braidwood 1&2	Walkdown for seismic-fire and seismic-flood concerns.	Potential issues with respect to seismic-induced fire hazards, such as gas bottles with insufficient constraints and locations of flammable storage cabinets, were identified and resolved except for the issue related to "unanchored hydrogen local control panel."	No concerns identified.	None.

Table 2.12: Seismic-fire interaction and seismic-flood interaction (Continued)

Plant	Evaluation approach	Seismic-fire observations/outliers	Seismic-flood observations/outliers	Related plant improvements
Browns Ferry 2&3	Walkdowns to identify sources of combustion and possible interactions.	No concerns identified.	Potential outliers for seismic-induced spray and flooding hazards from non-Class I systems and components were identified and resolved (which principally included maintenance of deficient hardware and support modifications or new installations).	None for IPEEE.
Brunswick 1&2	Walkdown for seismic-fire and seismic-flood concerns.	Potential interactions involving water piping for the fire protection system, as well as mobile/cart mounted CO_2 cylinders.	Potential concerns with overhead water lines and CST were ultimately screened out.	Procedure to secure CO_2 cylinders when not in use; the submittal also cites several past improvements made to enhance fire protection system seismic capability.
Byron 1&2	Walkdown for seismic-fire and seismic-flood concerns.	No significant concerns. Potential issues were identified and resolved except for the following issues: overturning of storage cabinets for oil, grease, and lubricants; interaction between hydrogen piping and a clothing bin on wheels; and poorly restrained gas bottles (resolution not discussed).	No concerns identified.	None.
Callaway	Walkdown for seismic-fire and seismic-flood concerns.	No concerns identified.	Relay chatter effects on fire pumps, and sprinkler head breakage, could lead to localized flooding; but they were determined not to affect SSEL equipment.	None.

Table 2.12: Seismic-fire interaction and seismic-flood interaction (Continued)

Plant	Evaluation approach	Seismic-fire observations/outliers	Seismic-flood observations/outliers	Related plant improvements
Calvert Cliffs 1&2	Walkdown for seismic-flood concerns. Both fire and flood initiators are screened at 0.3g, with some fire-inducing components also screened at 0.5g.	No concerns identified.	No concerns identified.	None.
Catawba 1&2	Walkdown for seismic-fire and seismic-flood concerns.	None.	None.	None.
Clinton	Walkdown for seismic-fire and seismic-flood concerns.	No risks were found.	No risks were found.	None.
Columbia Generating*	Walkdown for seismic-fire concerns.	No unusual or unique vulnerabilities. However, some problems were identified and addressed (e.g., inadequate support for the batteries of the diesel driven fire pumps and the possibility of inadvertent Halon actuation).	Seismic-induced floods are screened out in the submittal. The external floods are screened out based on the worst case Grand Coulee Dam failure. The internal floods are screened out based on comparison of the effects and frequencies of the loss of offsite power scenarios, or based upon the ruggedness of the piping.	Actions were taken to address the support problem for the batteries of the diesel driven fire pumps.
Comanche Peak 1&2	Walkdown for seismic-fire and seismic-flood concerns.	None.	None.	None.
Cooper	Walkdown for seismic-fire and seismic-flood concerns.	Four "seismic vulnerabilities" were identified in the fire suppression systems (two electric driven pumps, the diesel driven pump, and the water storage tanks) and included in the IPEEE Issue Resolution Plan; however, no specific corrective action was identified.	Seismic-induced failures of upstream dams were addressed. No unacceptable conditions concerning seismically-induced flooding were noted.	None.

Table 2.12: Seismic-fire interaction and seismic-flood interaction (Continued)

Plant	Evaluation approach	Seismic-fire observations/outliers	Seismic-flood observations/outliers	Related plant improvements
Crystal River 3	Not addressed . The licensee states that the GL 88-20, Supplement 4, does not require that a seismic/fire interaction review be performed for a "reduced-scope" plant like CR-3.	Not addressed in IPEEE.	Not addressed in IPEEE.	None.
D.C. Cook 1&2	Walkdown for seismic-fire and seismic-flood concerns.	Potential breakage of glass fuses in pilot lines; subsequently screened out because no potential was identified for sprinkler head breaks.	Same as for seismic-fire.	None.
Davis-Besse	Walkdown for seismic-fire and seismic-flood concerns.	Two small flammable compressed gas bottles in the auxiliary building were found to have inadequate anchorage.	No concerns identified.	The anchorage problem of two small flammable compressed gas bottles in the auxiliary building was being resolved.
Diablo Canyon 1&2	Walkdown for seismic-fire and seismic-flood concerns.	None.	None.	Addressed earlier in LTSP and Seismically Induced Systems Interaction Program (SISIP).
Dresden 2&3	Walkdown for seismic-fire and seismic-flood concerns.	Potential issues with respect to seismic-induced fire hazards identified and resolved (e.g., the effect of the failure of the hydrogen seal oil panel and hydrogen monitors on the integrity of the hydrogen lines).	Some concerns were identified and resolved (e.g., tanks behind switchgear).	Resolutions to the potential problems identified in the evaluation are presented in Tables 3.3 and 3.4 of the submittal.

Table 2.12: Seismic-fire interaction and seismic-flood interaction (Continued)

Plant	Evaluation approach	Seismic-fire observations/outliers	Seismic-flood observations/outliers	Related plant improvements
Duane Arnold	Walkdown screening evaluations performed in conjunction with the IPEEE SSEL equipment walkdown.	Three additional outliers were identified for equipment having nearby gas storage bottles that were not adequately restrained for seismic loadings.	Two air handlers in the HPCI room were identified as flood/spray outliers because nearby piping could potentially impact fire protection sprinkler piping and break off the sprinkler heads, whose spray could damage the air handler motors.	The air handler concern was resolved by analysis which showed adequate clearance between sprinkler heads and other piping, and the bottle concern was resolved by providing adequate restraint or removing the bottles.
Farley 1&2	As part of the seismic capacity walkdown, all potential internal flooding sources, mainly piping and tanks, were evaluated by the SRT in areas containing SSEL equipment.	No seismic-fire interaction issues exist at a seismic capacity of at least SSE level .	No flooding concerns were identified because piping has a high seismic capacity, and all tanks were well anchored.	None.
Fermi 2	Walkdown for seismic-fire and seismic-flood concerns.	No concerns identified.	No concerns identified.	None.
FitzPatrick	Walkdown for seismic-fire and seismic-flood concerns.	A vulnerability to fire or explosion as a result of the seismic-induced failure of the hydrogen line in the turbine building was identified.	No concerns identified.	Procedure AOP-14, "Earthquake," was modified. A note was added to AOP-14 stating that the hydrogen supply piping in the turbine building is susceptible to failure during seismic events and that the piping can be isolated by closing 89A-H2HAS-1, the hydrogen supply isolation valve.

Table 2.12: Seismic-fire interaction and seismic-flood interaction (Continued)

Plant	Evaluation approach	Seismic-fire observations/outliers	Seismic-flood observations/outliers	Related plant improvements
Fort Calhoun	Walkdown for seismic-fire and seismic-flood concerns.	Various concerns identified in turbine building; in the intake building, a fuel oil tank supplying fire water pumps has low capacity (HCLPF of about 0.05g).	Low seismic capacity of shutdown heat exchangers; flooding of junction boxes in Room 23; external flooding due to seismic dam break.	Fuel oil tank to be adequately anchored; a sight glass tube is to be replaced; anchorage of storage cabinet; additional anchor bolts on shutdown heat exchangers; waterproofing of junction boxes; external flooding addressed by severe accident management guidance.
Ginna	Issues examined by the SRT during the seismic capability walkdown.	Several issues were identified. They are related to the lack of anchorage for the house heating boiler (which could shift and damage the attached natural gas line) and the failure of block walls (which are used as fire barriers throughout the plant). The two reactor coolant pump oil collecting tanks in the containment basement were not reviewed during the seismic walkdown because the containment was inaccessible.	A concern was system failure due to seismically induced flooding from failure of the Reactor Makeup Water tank and the Monitor tank. These tanks will be considered outliers and will be examined to determine the correct course of action.	The seismic-fire issues were resolved as a part of Ginna's IPEEE fire analysis by either design evaluations or design changes.
Grand Gulf 1	Included in the SSEL for the IPEEE.	No concerns identified.	No concerns identified.	None.
H.B. Robinson 2	Walkdown for seismic-fire and seismic-flood concerns.	Some issues pertaining to panel interactions and poorly anchored electrical cabinets were identified.	None reported.	None reported.

Table 2.12: Seismic-fire interaction and seismic-flood interaction (Continued)

Plant	Evaluation approach	Seismic-fire observations/outliers	Seismic-flood observations/outliers	Related plant improvements
Haddam Neck	Walkdown for seismic-fire and seismic-flood concerns; SPRA modeling of flooding due to dam failure.	Eight vulnerabilities or risk outliers were identified.	None reported.	Issues have been resolved or proposed for resolution (See Table 7.1-1 of IPEEE submittal).
Hatch 1&2	Issues reviewed as a part of the seismic walkdown.	No concerns identified.	No concerns identified.	None.
Hope Creek	Walkdown for seismic-fire concerns and analysis.	The only seismic capacity concerns are: (1) FPS water pump house - assumed to fail, and (2) FPS water tanks - no credit taken after a seismic event (median acceleration capacity of 0.73g, HCLPF of 0.26g).	There is no discussion of seismic-induced flooding concerns.	None.
Indian Point 2	Walkdown for seismic-fire and seismic-flood concerns; no modeling of seismically induced fire or flood sequences in seismic PRA.	Questionable anchorage of the reactor coolant pump lube oil collection tank; subsequently determined to be adequate. Concern with hydrogen bottles stored near alternate shutdown panel; no action taken because alternate shutdown panel is not credited in the seismic PRA.	None reported.	None.
Indian Point 3	Walkdown for seismic-fire and seismic-flood concerns.	The seismic "vulnerabilities" identified are: (1) the CO_2 system whose rupture poses little risk; (2) the low seismic fragility level of the two 350,000-gallon fire water tanks; (3) the availability of the FPS pumps which are housed in the FPS pump house with masonry block walls; and (4) the marginal lateral support capacity of the fuel tank for the diesel pump.	No concerns identified.	No discussion is provided in the submittal on improvements for the identified seismic-fire "vulnerabilities."

Table 2.12: Seismic-fire interaction and seismic-flood interaction (Continued)

Plant	Evaluation approach	Seismic-fire observations/outliers	Seismic-flood observations/outliers	Related plant improvements
Kewaunee	Walkdown for seismic-fire and seismic-flood concerns.	Potential damage to fire water capability and sprinklers/lines; mercoid fire pump jockey switches and Cardox pressure switches.	Same as for seismic-fire.	None.
La Salle 1&2	None documented.	None.	None.	None.
Limerick 1&2	Walkdown for seismic-fire and seismic-flood concerns.	Sight glass tubes on lube oil make-up tanks do not have isolation valves; mercoid switches in two fire protection systems. These concerns were determined not to be significant.	No additional.	None.
McGuire 1&2	Walkdown for seismic-fire and seismic-flood concerns.	None.	None.	None.
Millstone 2	Walkdown for seismic-fire and seismic-flood concerns.	Identified three issues: adequacy of the seismic capacity of the Unit 1 diesel fire pump fuel tank, seismic capacity of a long run of the fire header system piping, and the block wall construction of the fire pump house.	No concerns identified.	Resolutions of seismic-fire outliers include additional evaluation to ensure seismic adequacy or hardware modification.
Millstone 3	None documented.	None.	None.	None.
Monticello	Walkdown for seismic-fire and seismic-flood concerns.	Sliding of turbine lube oil tank located in MCC-133/feedwater pump area, was identified as a potential concern; but based on a qualitative assessment, the licensee judged that additional analysis was unwarranted.	Several non-safety tanks were found to have low seismic resistance, but these were determined to be isolated, or far from, success path equipment.	None.
Nine Mile Point 1	Walkdown for seismic-fire and seismic-flood concerns.	No concerns identified.	No concerns identified.	None.

Table 2.12: Seismic-fire interaction and seismic-flood interaction (Continued)

Plant	Evaluation approach	Seismic-fire observations/outliers	Seismic-flood observations/outliers	Related plant improvements
Nine Mile Point 2	Walkdown for seismic-fire and seismic-flood concerns.	None.	None.	None.
North Anna 1&2	Walkdown for seismic-fire and seismic-flood concerns.	A potential issue is seismic-induced fires from the lube oil heat exchanger, hydrogen piping, and hydrogen bottles.	Two issues identified: (1) inadequate support for feedwater heaters in turbine building, and (2) the flooding potential for the casing cooling tanks, located next to the auxiliary feedwater pump house.	Issues to be resolved by the end of the NAPS Unit 1 refueling outage currently scheduled to commence in April 2000.
Oconee 1,2,&3	Walkdown for seismic-fire and seismic-flood concerns. More detailed analysis for items not screened by walkdown.	Issues identified in seismic-fire review resulted in procedural and physical improvements.	The fault tree models used in the seismic analysis include the effect of both internal and external flooding sources.	Possible improvements, such as replacement of sprinkler heads, are discussed in Section 4.9.
Oyster Creek	The seismic-fire-flood interactions were not directly considered in the seismic PRA, but were qualitatively screened in the fire analysis section, with some ambiguity as to which earthquake levels were included in the walkdown and the evaluation.	The conclusion is that no sources of seismic induced fire initiation at "reasonable levels of earthquake beyond the design basis" were identified. Words such as "nominal" earthquake appear elsewhere in this section. It is not clear whether the licensee considered the same ground acceleration levels as in the seismic study for this evaluation. It appears that this was mostly a qualitative evaluation. In the area of inadvertent fire suppression actuation, it is noted that electrical equipment is usually well protected by shields or is sealed.	No discussion in the submittal is provided regarding any seismic-flood interactions, whether internal or external.	None.

Table 2.12: Seismic-fire interaction and seismic-flood interaction (Continued)

Plant	Evaluation approach	Seismic-fire observations/outliers	Seismic-flood observations/outliers	Related plant improvements
Palisades	Walkdown and SPRA modeling of seismic-fire and seismic-flood concerns.	Hydrogen piping through turbine building is not seismically designed and passes through block walls and cable trays which pose a rupture hazard; there exist a number of unanchored flammable liquid storage cabinets throughout the turbine building.	Seismic-induced flooding in the turbine building and screenhouse were identified for SPRA modeling; circulating water pipe failures in screenhouses were later screened out.	None.
Palo Verde 1,2,&3	Seismically-induced fires/floods addressed in the plant walkdown.	No concerns identified.	One flooding concern was found and this was judged not to be a problem by the seismic review team.	None.
Peach Bottom 2&3	Walkdown for seismic-fire and seismic-flood concerns.	Mercoid switches encountered in fire protection systems. Unanchored CO_2 tanks in Cardox room of the DG building. No spacers between batteries, and lack of end rails, on CO_2 battery racks.	No problems encountered other than potential inadvertent actuation of fire protection systems due to spurious behavior of mercoid switches.	- Replace four mercoid switches in fire water manual pull stations with non-mercoid switches. - Establish procedures to mitigate spurious relay operation in Cardox panels. - Add restraints to Cardox tank protecting diesel generator areas. - Evaluate the potential and effects of CO_2 release in the turbine building, due to failure of Cardox tanks.
Perry 1	Walkdown for seismic-flooding, seismic-fire concerns.	The one concern is the seismic capacity of the FPS diesel driven pump's fuel oil tank located in the ESW pumphouse was identified and dismissed by HCLPF evaluation.	No concerns identified.	None.

Table 2.12: Seismic-fire interaction and seismic-flood interaction (Continued)

Plant	Evaluation approach	Seismic-fire observations/outliers	Seismic-flood observations/outliers	Related plant improvements
Pilgrim 1	Walkdown for seismic-fire and seismic-flood concerns.	Truck lock in turbine building contains hydrogen and lube oil piping runs, and a hydrogen control station; switchgear room "B" also contains lengths of piping which contain lube oil.	Interaction potential between CST 105B and cryogenic nitrogen storage tank, modeled as leading to loss of CST as water source for HPCI and RCIC.	None, but the licensee stated that consideration should be given to isolation of combustible sources following an earthquake.
Point Beach 1&2	Walkdown for seismic-fire and seismic-flood concerns.	None.	RWST could fail and disable RHR pumps.	None.
Prairie Island	Seismic walkdown.	No concerns identified.	No concerns identified.	None.
Quad Cities 1&2	Walkdown for seismic-fire and seismic-flood concerns.	Several concerns pertaining to seismically-induced fires, inadvertent seismic actuation of fire suppression systems, and seismically induced failure of fire protection capability were noted.	Fire piping risers in the cable spreading room may break via interaction with adjacent multi-tier cable trays. Piping attached to cubicle coolers (located in corners of the reactor building) was observed to have inadequate flexibility to accommodate movement of the rod-hung units.	Six mercoid relays were replaced in the Cardox system protecting the emergency diesel generators; oxygen cylinders in the common turbine building mezzanine floor are now chained top and bottom to a newly installed cylinder rack. Cubicle coolers are being addressed under USI A-46, and HCLPF capacities for these components have been determined.
River Bend	Walkdown for seismic-fire and seismic-flood concerns.	No vulnerabilities were reported.	No vulnerabilities were identified.	None.
Salem 1&2	Walkdown for seismic-fire and seismic-flood concerns.	Several concerns were identified and dismissed after additional mitigating considerations.	No concerns identified. However, there was no discussion of any external flooding by river water, etc., caused by seismic events.	None.

Table 2.12: Seismic-fire interaction and seismic-flood interaction (Continued)

Plant	Evaluation approach	Seismic-fire observations/outliers	Seismic-flood observations/outliers	Related plant improvements
San Onofre 2&3	Walkdown for seismic-fire and seismic-flood concerns; detailed qualitative evaluation.	No concerns identified.	As a potential flooding source, CCW seals were not screened out, and were included in the core damage model. The licensee concluded that there is no significant risk of core damage due to seismically induced flooding.	The licensee's evaluation also examined the potential for seismically induced toxic material releases, and identified a plant improvement to the anchorage of an ammonia storage tank.
Seabrook	Walkdown for seismic-fire and seismic-flood concerns. A seismic-induced flooding analysis was conducted in 1991 as part of the update of the seismic PRA.	No concerns identified.	No seismically-induced flooding scenarios were identified that would have the potential to fail other risk-important equipment or systems.	None.
Sequoyah 1&2	Walkdown for seismic-fire and seismic-flood concerns.	Potential of four light transformers in the auxiliary building to impact SSEL-related cables. (Transformers were subsequently assessed as having an HCLPF capacity of 0.37g.)	Potential for sprinkler head breakage, but not in the vicinity of SSEL equipment.	None.
Shearon Harris 1	Walkdown for seismic-fire and seismic-flood concerns.	No risks were found.	No issues were identified.	None.
South Texas Project 1&2	None documented.	None.	None.	None.
St. Lucie 1&2	Documentation review; no discussion of a seismic-fire walkdown.	None.	None.	None.

Table 2.12: Seismic-fire interaction and seismic-flood interaction (Continued)

Plant	Evaluation approach	Seismic-fire observations/outliers	Seismic-flood observations/outliers	Related plant improvements
Summer	Walkdown for seismic-fire and seismic-flood concerns.	Identified some minor concerns, which were either resolved by evaluation (e.g., no impact on SSEL equipment) or by simple corrective actions (e.g., better housekeeping with regard to unsecured flammable gas bottles). Seismic actuation of fire suppression systems and seismic degradation of fire suppression systems were not evaluated in IPEEE.	No concerns were identified for seismically-induced external flooding. Although internal seismic-induced flooding hazard was not specifically discussed, it was considered adequately addressed by the original design basis of the plant and the individual plant examination program.	None.
Surry 1&2	Walkdown for seismic-fire and seismic-flood concerns and PRA modeling of seismic-induced fire concerns.	Some concerns identified and dismissed after further evaluation. The potential fire arising from a concern of anchorage of lube oil tanks in the turbine building, which may lead to the loss of plant service water, was modeled in the seismic event tree.	Some concerns were identified and dismissed after further evaluation. Some tanks in the turbine building were identified which could slide causing a severance of connections. The concern was dismissed because the resulting flooding scenario would be enveloped by the internal flooding analysis. The question of whether such scenarios which could lead to a large conditional core damage probability should be modeled above the RLE is not addressed.	None.
Susquehanna 1&2	Walkdown for seismic-fire and seismic-flood concerns.	Fire pumps in non-seismically-designed structure; CO_2 supply tank is not seismically anchored; batteries for fire pumps do not have spacers; unanchored small electrical cabinets.	Non-seismically-designed fire water system. The submittal notes that the potential for inadvertent actuation of fire water system is low.	None.

Table 2.12: Seismic-fire interaction and seismic-flood interaction (Continued)

Plant	Evaluation approach	Seismic-fire observations/outliers	Seismic-flood observations/outliers	Related plant improvements
TMI 1	Walkdown for seismic-fire and seismic-flood concerns.	No significant concerns identified. Some concerns regarding the availability of the fire protection system following a seismic event were identified.	Potential failure of the piping and heat exchangers in the heat exchanger vault area was identified and was dismissed because the walkdown team determined that the area annunciation was adequate to allow the plant operators to respond long before flooding became a concern.	None.
Turkey Point 3&4	Documentation review; no discussion of a seismic-fire walkdown.	None.	None.	None.
Vermont Yankee	Walkdown for seismic-fire and seismic-flood concerns.	Identified a few "improvement opportunities" related to seismic resistance of the H_2 piping in the turbine building, the lack of positive attachment between the diesel-driven fire pump fuel tank and its supports, the lack of anchorage of Buses 1 and 2 to the structure, and the support of the fire system northwest standpipe in the reactor building.	No concerns identified.	Modification to locally reroute the fuel line tubing for the diesel fire pump fuel tank; Improvement to enhance the support of the fire system standpipe.
Vogtle 1&2	Walkdown for seismic-fire and seismic-flood concerns.	No concerns identified.	No concerns identified.	None.
Waterford 3	Walkdown screening evaluations were performed in conjunction with the IPEEE SSEL walkdown. Potential seismic induced fire/flood sources were identified. Verification walkdowns were performed by the SRT.	As a result of the seismic-induced fire walkdown, no vulnerabilities were identified.	As a result of the seismic-induced flood walkdown, no vulnerabilities were identified.	None.

Table 2.12: Seismic-fire interaction and seismic-flood interaction (Continued)

Plant	Evaluation approach	Seismic-fire observations/outliers	Seismic-flood observations/outliers	Related plant improvements
Watts Bar 1	Walkdown for seismic-induced flooding and seismic-induced fire issues.	No safety issues were identified.	The submittal did not contain much discussion on the seismically induced flooding issues, except the mention that the potential for seismic-induced floods was evaluated by the SRT as a part of walkdown procedures.	None.
Wolf Creek	Addressed as part of the SSEL walkdown; no walkdown was performed to evaluate seismic-fire or seismic-flooding effects outside the direct influence on SSEL equipment.	None identified.	None identified.	None.

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Table 2.13: Flux mapping systems

Plant	GI-131 applicable	Previous upgrade	IPEEE findings	Related plant improvements
Arkansas Nuclear One 1	No	Not applicable	Not applicable	Not applicable
Arkansas Nuclear One 2	No	Not applicable	Not applicable	Not applicable
Beaver Valley 1	Yes	Analysis performed in 1989 for SSE	Adequacy of system verified by an SMA analysis with a review level earthquake of 0.3g.	None.
Beaver Valley 2	Yes	Issue resolved prior to initial plant startup	Plant is not vulnerable to a small LOCA from this issue. Its contribution to CDF was considered negligible and not quantified in IPEEE.	None.
Braidwood 1&2	No	Not applicable	Not applicable	Not applicable
Browns Ferry 2&3	No	Not applicable	Not applicable	Not applicable
Brunswick 1&2	No	Not applicable	Not applicable	Not applicable.
Byron 1&2	No	Not applicable	Not applicable	Not applicable
Callaway	Yes	In 1987, the hold-down assembly of the flux mapping system was upgraded by increasing the size and strength of bolts and plates of the assembly.	None; flux mapping system was inaccessible due to radioactive exposure concerns; no analysis for events that are beyond the design basis.	No additional improvements for IPEEE.
Calvert Cliffs 1&2	No	Not applicable	Not applicable	Not applicable
Catawba 1&2	Yes	Restraints added during construction.	No analysis for events that are beyond the design basis.	No additional improvements for IPEEE.
Clinton	No	Not applicable	Not applicable	Not applicable

Table 2.13: Flux mapping systems (Continued)

Plant	GI-131 applicable	Previous upgrade	IPEEE findings	Related plant improvements
Columbia*	No	Not applicable	Not applicable	Not applicable
Comanche Peak 1&2	Yes	None; however, in previous licensing spatial interaction program activities, it was determined that the flux mapping system was designed and constructed to preclude interactions at SSE loads.	None	None.
Cooper	No	Not applicable	Not applicable	Not applicable
Crystal River	No	Not applicable	Not applicable	Not applicable
D.C. Cook 1&2	Yes	Hold-down straps attached to the top of the cart were redesigned and modified; a lower lateral restraint to the flux mapping cart was installed at an elevation just above the seal table.	0.32g HCLPF capacity assessed based on walkdown and review of the modified configuration.	No additional improvements for IPEEE.
Davis-Besse	No	Not applicable	Not applicable	Not applicable
Diablo Canyon 1&2	Yes	SISIP-related modifications to improve the seismic structural integrity of the frame assemblies.	No analysis for events that are beyond the design basis.	No additional improvements for IPEEE.
Dresden 2&3	No	Not applicable	Not applicable	Not applicable
Duane Arnold	No	Not applicable	Not applicable	Not applicable
Farley 1&2	Yes	The flux mapping system cart hold-down bolts were replaced to comply with the Westinghouse recommendation.	No issue identified in IPEEE walkdown.	None.
Fermi 2	No	Not applicable	Not applicable	Not applicable

Table 2.13: Flux mapping systems (Continued)

Plant	GI-131 applicable	Previous upgrade	IPEEE findings	Related plant improvements
FitzPatrick	No	Not applicable	Not applicable	Not applicable
Fort Calhoun	No	Not applicable	Not applicable	Not applicable
Ginna	Yes	Not described in submittal.	System was examined by the A-46/IPEEE SRT during the containment walkdown, and was found not to be seismically vulnerable.	None.
Grand Gulf 1	No	Not applicable	Not applicable	Not applicable
H.B. Robinson 2	Yes	Four hold-down restraints were fabricated of steel angle, welded to the cart, and bolted to the structure.	Seismic review team determined the flux mapping system to be adequate for RLE loads.	No additional improvements for IPEEE.
Haddam Neck	Not directly, since cart is not movable.	None; flux mapping cart is already bolted to the platform.	Walkdown verified adequacy of configuration.	None.
Hatch 1&2	No	Not applicable	Not applicable	Not applicable
Hope Creek	No	Not applicable	Not applicable	Not applicable
Indian Point 2	Yes	The flux monitoring cart had previously been braced in two directions.	Screened out at 0.5g HCLPF.	No additional improvements for IPEEE.
Indian Point 3	Yes	Not discussed in submittal.	Support and restraint of the movable portion of the system were inspected during containment walkdown and seismic-induced damage to the seal table was judged not credible.	None.
Kewaunee	Not directly, since flux mapping cart is not movable.	None; lateral resistance of flux mapping system has already been determined to be adequate.	Walkdown found that a chain hoist above the flux mapping cart might interact with the 10 path assembly.	Administrative control was implemented to better secure chain hoist.

Table 2.13: Flux mapping systems (Continued)

Plant	GI-131 applicable	Previous upgrade	IPEEE findings	Related plant improvements
La Salle 1&2	No	Not applicable	Not applicable	Not applicable
Limerick 1&2	No	Not applicable	Not applicable	Not applicable
McGuire 1&2	Yes	Previous seismic analyses have indicated installed restraints are adequate.	No analysis for events that are beyond the design basis.	No additional improvements for IPEEE.
Millstone 2	No	Not applicable	Not applicable	Not applicable
Millstone 3	Yes	A modification was implemented to limit relative displacement between the flux mapping equipment and the seal table.	None described.	No additional improvements for IPEEE.
Monticello	No	Not applicable.	Not applicable	Not applicable
Nine Mile Point 1	No	Not applicable	Not applicable	Not applicable
Nine Mile Point 2	No	Not applicable.	Not applicable	Not applicable
North Anna 1&2	Yes	A provision for restraining this system during plant operation is outlined in the station procedure IMPC-C-1-IC-07. The equipment is restrained with floor-mounted brackets which are located on either side of the equipment.	The system was examined during walkdown. Review of the restraint/anchorage showed HCLPF value greater than 0.3g.	None.
Oconee 1,2,&3	No	Not applicable	Not applicable	Not applicable
Oyster Creek	No	Not applicable	Not applicable	Not applicable
Palisades	No	Not applicable	Not applicable	Not applicable
Palo Verde 1,2,&3	No	Not applicable	Not applicable	Not applicable

Table 2.13: Flux mapping systems (Continued)

Plant	GI-131 applicable	Previous upgrade	IPEEE findings	Related plant improvements
Peach Bottom 2&3	No	Not applicable	Not applicable	Not applicable
Perry 1	No	Not applicable	Not applicable	Not applicable
Pilgrim 1	No	Not applicable	Not applicable	Not applicable
Point Beach 1&2	Not directly, since flux mapping cart is not movable.	None.	IPEEE submittal notes that the flux mapping system is identical to Kewaunee's, which was found to be adequate.	None.
Prairie Island 1&2	Yes	Not discussed in submittal.	The moveable in-core flux mapping systems were "found to have sufficient seismic capacity to the SSE level of the USAR" by the licensee. No further information is provided.	None.
Quad Cities 1&2	No	Not applicable	Not applicable	Not applicable
River Bend	No	Not applicable	Not applicable	Not applicable
Salem 1&2	Yes	Not discussed in submittal.	No seismic vulnerabilities identified in walkdown.	None.
San Onofre 2&3	No	Not applicable	Not applicable	Not applicable
Seabrook	Yes	GSI-131 was addressed by installation of hold-down bolts for the flux mapping cart.	The presence of hold-down bolts was verified during the seismic walkdown. Therefore, GSI-131 is considered closed by the licensee.	None.
Sequoyah 1&2	Yes	Restraints have been installed on the flux mapping cart.	Seismic review team determine the flux mapping system to be adequate for RLE loads.	No IPEEE improvements.

Table 2.13: Flux mapping systems (Continued)

Plant	GI-131 applicable	Previous upgrade	IPEEE findings	Related plant improvements
Shearon Harris 1	Yes	Issue identified in 1984 and resolved by installing wheel stops on the Flux Mapping Control Trolley to prevent seismic interaction between the trolley and the ICFM tubing or fittings.	The installation was visually verified by using the Harris Surrogate Tour system, and the licensee stated that the installed wheel stops have sufficient design margin to be screened out for the RLE.	None.
South Texas Project 1&2	Yes	None described.	IPEEE submittal notes that there are no vulnerabilities or risk outliers associated with this issue.	None.
St. Lucie 1&2	No	Not applicable	Not applicable	Not applicable
Summer	Yes	A plant modification was carried out after a review of this issue in 1985. The modification restrained the 10-path rotary transfer and valve support assembly from seismic movement via four floor mounted steel tubes.	The submittal does not mention this issue, but states that the in-core detectors are generally "inherently rugged," that walkdown of this equipment is not required, but that seismic evaluation should be conducted. In response to an RAI the licensee simply stated that by the "rule of the box" the in-core flux mapping system is considered a part of the "reactor internals," and that according to Supplement 5 of GL 88-20, reactor internals are considered generically rugged. An email reply by the licensee to a subsequent communication resolved the issue.	None.

Table 2.13: Flux mapping systems (Continued)

Plant	GI-131 applicable	Previous upgrade	IPEEE findings	Related plant improvements
Surry 1&2	Yes	A station procedure already in place to restrain the system during the operation of the plant.	System examined through walkdowns. It was concluded that the restraint/anchorage for the system had a HCLPF capacity greater than 0.3g and this issue is considered to be closed by the licensee.	None.
Susquehanna 1&2	No	Not applicable	Not applicable	Not applicable
TMI 1	No	Not applicable	Not applicable	Not applicable
Turkey Point 3&4	Yes	In 1989, lateral restraint was added to the movable support assembly of the flux mapping system, and was evaluated as being adequate.	No evaluation.	None.
Vermont Yankee	No	Not applicable	Not applicable	Not applicable
Vogtle 1&2	Yes	VEGP has installed a stiffener and four anchor assemblies.	This system was walked down as part of the IPEEE evaluation and no issues were identified.	None.
Waterford 3	No	Not applicable	Not applicable	Not applicable
Watts Bar 1	Yes	Not discussed in submittal.	The flux monitoring cart was verified during the IPEEE walkdowns to have been restrained adequately and was determined to be capable of withstanding the RLE.	None.

Table 2.13: Flux mapping systems (Continued)

Plant	GI-131 applicable	Previous upgrade	IPEEE findings	Related plant improvements
Wolf Creek	Yes	None.	The seismic review team questioned the lateral restraint of wide-flange beams that support the frame containing the movable flux mapping system (excessive lateral movement of the beams could dislodge the frame), but noted that when the cart is positioned above the seal table, steel angles (that are welded to the beams) can be bolted to the movable frame, and hence, would prevent the frame from being dislodged. Assuming that the operators utilize this provision and secure the frame, the seismic review team screened out the flux mapping system at an HCLPF capacity of 0.3g.	None.

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3. FIRE TABLES

This section contains five tables of summary information obtained from the fire portions of the IPEEEs. Table 3.1 contains basic plant information for each plant. The table includes the plant type, plant class, containment class, present gross capacity, location (by State), and the dates the construction permit and operating license were issued.

Table 3.2 presents the plant-wide fire core damage frequency (CDF) information. The table includes the fire PRA methodology used at each plant, the CDF reported in the IPEEE, any revised CDFs after all RAIs were addressed, and the internal events CDF. Table 3.3 lists the fire areas that contributed to a plant's overall CDF and their corresponding CDFs.

Tables 3.4 and 3.5 list the vulnerabilities and improvements, respectively, identified by the various plants. Included in the list of vulnerabilities is the definition of "vulnerability" used by a plant. The improvements table includes whether the improvement was a result of the fire analysis, a seismic/fire analysis, or some other program or action. Information regarding the current status of the improvement is also provided.

Table 3.1: Summary of basic plant information

Plant name	Plant type	Plant class	Containment class	MWe	Plant location (State)	Date CP issued	Date OL issued
ANO 1	PWR	B&W	Large Dry	836	Arkansas	12/06/1968	05/21/1974
ANO 2	PWR	CE	Large Dry	858	Arkansas	12/06/1972	09/01/1978
Beaver Valley 1	PWR	West 3-loop	Subatmospheric	833	Pennsylvania	06/26/1970	07/02/1976
Beaver Valley 2	PWR	West 3-loop	Subatmospheric	833	Pennsylvania	05/03/1974	08/14/1987
Braidwood 1	PWR	West 4-loop	Large Dry	1120	Illinois	12/31/1975	07/02/1987
Braidwood 2	PWR	West 4-loop	Large Dry	1120	Illinois	12/31/1975	05/20/1988
Browns Ferry 1	BWR	BWR 3/4	Mark I	1065	Alabama	05/10/1967	12/20/1973
Browns Ferry 2	BWR	BWR 3/4	Mark I	1065	Alabama	05/10/1967	08/02/1974
Browns Ferry 3	BWR	BWR 3/4	Mark I	1065	Alabama	07/31/1968	08/18/1976
Brunswick 1	BWR	BWR 3/4	Mark I	821	North Carolina	02/07/1970	11/12/1976
Brunswick 2	BWR	BWR 3/4	Mark I	821	North Carolina	02/07/1970	12/27/1974
Byron 1	PWR	West 4-loop	Large Dry	1120	Illinois	12/31/1974	02/14/1985
Byron 2	PWR	West 4-loop	Large Dry	1120	Illinois	12/31/1974	01/30/1987
Callaway	PWR	West 4-loop	Large Dry	1150	Missouri	04/16/1976	10/18/1984
Calvert Cliffs 1	PWR	CE	Large Dry	865	Maryland	07/07/1969	07/31/1974
Calvert Cliffs 2	PWR	CE	Large Dry	865	Maryland	07/07/1969	11/30/1976
Catawba 1	PWR	West 4-loop	Ice Condenser	1129	South Carolina	08/07/1975	01/17/1985
Catawba 2	PWR	West 4-loop	Ice Condenser	1129	South Carolina	08/07/1975	05/15/1986
Clinton	BWR	BWR 5/6	Mark III	950	Illinois	02/24/1976	04/17/1987
Columbia*	BWR	BWR 5/6	Mark II	1100	Washington	03/19/1973	04/13/1984
Comanche Peak 1	PWR	West 4-loop	Large Dry	1150	Texas	12/19/1974	04/17/1990
Comanche Peak 2	PWR	West 4-loop	Large Dry	1150	Texas	12/19/1974	04/06/1993
Cook 1	PWR	West 4-loop	Ice Condenser	1020	Michigan	03/25/1969	10/25/1974
Cook 2	PWR	West 4-loop	Ice Condenser	1060	Michigan	03/25/1969	12/23/1977
Cooper	BWR	BWR 3/4	Mark I	778	Nebraska	06/04/1968	01/18/1974
Crystal River 3	PWR	B&W	Large Dry	821	Florida	09/25/1968	01/28/1977
Davis-Besse	PWR	B&W	Large Dry	906	Ohio	03/24/1971	04/22/1977
Diablo Canyon 1	PWR	West 4-loop	Large Dry	1073	California	04/23/1968	11/02/1984
Diablo Canyon 2	PWR	West 4-loop	Large Dry	1087	California	12/09/1970	08/26/1985
Dresden 2	BWR	BWR 2/3	Mark I	794	Illinois	01/10/1966	12/22/1969
Dresden 3	BWR	BWR 2/3	Mark I	794	Illinois	10/14/1966	03/02/1971
Duane Arnold	BWR	BWR 3/4	Mark I	538	Iowa	06/22/1970	02/22/1974
Farley 1	PWR	West 3-loop	Large Dry	829	Alabama	08/16/1972	06/25/1977
Farley 2	PWR	West 3-loop	Large Dry	829	Alabama	08/16/1972	03/31/1981
Fermi 2	BWR	BWR 3/4	Mark I	1093	Michigan	09/26/1972	07/15/1985
FitzPatrick	BWR	BWR 3/4	Mark I	816	New York	05/20/1970	10/17/1974
Fort Calhoun	PWR	CE	Large Dry	488	Nebraska	06/07/1968	08/09/1973
Ginna	PWR	West 2-loop	Large Dry	470	New York	04/25/1966	07/01/1970
Grand Gulf 1	BWR	BWR 5/6	Mark III	1250	Mississippi	09/04/1974	11/01/1984
Haddam Neck	PWR	West 4-loop	Large Dry	582	Connecticut	05/26/1964	06/30/1967
Hatch 1	BWR	BWR 3/4	Mark I	797	Georgia	09/30/1969	10/13/1974
Hatch 2	BWR	BWR 3/4	Mark I	806	Georgia	12/27/1972	06/13/1978
Hope Creek 1	BWR	BWR 3/4	Mark I	1031	New Jersey	11/04/1974	07/25/1986
Indian Point 2	PWR	West 4-loop	Large Dry	939	New York	10/14/1966	09/28/1973
Indian Point 3	PWR	West 4-loop	Large Dry	965	New York	08/13/1969	04/05/1976
Kewaunee	PWR	West 2-loop	Large Dry	540	Wisconsin	08/06/1968	12/21/1973
LaSalle 1	BWR	BWR 5/6	Mark II	1078	Illinois	09/10/1973	03/13/1982

Table 3.1: Summary of basic plant information (Continued)

Plant name	Plant type	Plant class	Containment class	MWe	Plant location (State)	Date CP issued	Date OL issued
LaSalle 2	BWR	BWR 5/6	Mark II	1078	Illinois	09/10/1973	03/23/1984
Limerick 1	BWR	BWR 3/4	Mark II	1055	Pennsylvania	06/19/1974	08/08/1985
Limerick 2	BWR	BWR 3/4	Mark II	1055	Pennsylvania	06/19/1974	08/25/1989
McGuire 1	PWR	West 4-loop	Ice Condenser	1129	North Carolina	02/23/1973	07/08/1981
McGuire 2	PWR	West 4-loop	Ice Condenser	1129	North Carolina	02/23/1973	05/27/1983
Millstone 2	PWR	CE	Large Dry	858	Connecticut	12/11/1970	09/30/1975
Millstone 3	PWR	West 4-loop	Subatmospheric	1150	Connecticut	08/09/1974	01/31/1986
Monticello	BWR	BWR 3/4	Mark I	542	Minnesota	06/19/1967	02/18/1971
Nine Mile Point 1	BWR	BWR 2/3	Mark I	613	New York	04/12/1965	08/22/1969
Nine Mile Point 2	BWR	BWR 5/6	Mark II	1062	New York	06/24/1974	07/02/1987
North Anna 1	PWR	West 3-loop	Subatmospheric	907	Virginia	02/19/1971	04/01/1978
North Anna 2	PWR	West 3-loop	Subatmospheric	907	Virginia	02/19/1971	08/21/1980
Oconee 1	PWR	B&W	Large Dry	846	South Carolina	11/06/1967	02/06/1973
Oconee 2	PWR	B&W	Large Dry	846	South Carolina	11/06/1967	10/06/1973
Oconee 3	PWR	B&W	Large Dry	846	South Carolina	11/06/1967	07/19/1974
Oyster Creek	BWR	BWR 2/3	Mark I	650	New Jersey	12/15/1964	12/01/1969
Palisades	PWR	CE	Large Dry	805	Michigan	03/14/1967	12/31/1969
Palo Verde 1	PWR	CE	Large Dry	1270	Arizona	05/25/1976	06/01/1985
Palo Verde 2	PWR	CE	Large Dry	1270	Arizona	05/25/1976	04/24/1986
Palo Verde 3	PWR	CE	Large Dry	1270	Arizona	05/25/1976	11/25/1987
Peach Bottom 2	BWR	BWR 3/4	Mark I	1055	Pennsylvania	01/31/1968	12/14/1973
Peach Bottom 3	BWR	BWR 3/4	Mark I	1035	Pennsylvania	01/31/1968	07/02/1974
Perry 1	BWR	BWR 5/6	Mark III	1205	Ohio	05/03/1977	11/13/1986
Pilgrim 1	BWR	BWR 3/4	Mark I	670	Massachusetts	08/26/1968	09/15/1972
Point Beach 1	PWR	West 2-loop	Large Dry	485	Wisconsin	07/19/1967	10/05/1970
Point Beach 2	PWR	West 2-loop	Large Dry	485	Wisconsin	07/25/1968	03/08/1973
Prairie Island 1	PWR	West 2-loop	Large Dry	503	Minnesota	06/25/1968	04/05/1974
Prairie Island 2	PWR	West 2-loop	Large Dry	500	Minnesota	06/25/1968	10/29/1974
Quad Cities 1	BWR	BWR 3/4	Mark I	789	Illinois	02/15/1967	12/14/1972
Quad Cities 2	BWR	BWR 3/4	Mark I	789	Illinois	02/15/1967	12/14/1972
River Bend 1	BWR	BWR 5/6	Mark III	936	Louisiana	03/25/1977	11/20/1985
Robinson 2	PWR	West 3-loop	Large Dry	718	South Carolina	04/13/1967	09/23/1970
Salem 1	PWR	West 4-loop	Large Dry	1106	New Jersey	09/25/1968	12/01/1976
Salem 2	PWR	West 4-loop	Large Dry	1106	New Jersey	09/25/1968	05/20/1981
San Onofre 2	PWR	CE	Large Dry	1070	California	10/18/1973	09/07/1982
San Onofre 3	PWR	CE	Large Dry	1080	California	10/18/1973	09/16/1983
Seabrook 1	PWR	West 4-loop	Large Dry	1148	New Hampshire	07/07/1976	03/15/1990
Sequoyah 1	PWR	West 4-loop	Ice Condenser	1141	Tennessee	05/27/1970	09/17/1980
Sequoyah 2	PWR	West 4-loop	Ice Condenser	1136	Tennessee	05/27/1970	09/15/1981
Shearon Harris 1	PWR	West 3-loop	Large Dry	900	North Carolina	01/27/1978	01/12/1987
South Texas 1	PWR	West 4-loop	Large Dry	1250	Texas	12/22/1975	03/22/1988
South Texas 2	PWR	West 4-loop	Large Dry	1250	Texas	12/22/1975	03/28/1989
St. Lucie 1	PWR	CE	Large Dry	839	Florida	07/01/1970	03/01/1976
St. Lucie 2	PWR	CE	Large Dry	839	Florida	05/02/1977	06/10/1983
Summer	PWR	West 3-loop	Large Dry	895	South Carolina	03/21/1973	11/12/1982
Surry 1	PWR	West 3-loop	Subatmospheric	788	Virginia	06/25/1968	05/25/1972
Surry 2	PWR	West 3-loop	Subatmospheric	788	Virginia	06/25/1968	01/29/1973

Table 3.1: Summary of basic plant information (Continued)

Plant name	Plant type	Plant class	Containment class	MWe	Plant location (State)	Date CP issued	Date OL issued
Susquehanna 1	BWR	BWR 3/4	Mark II	1050	Pennsylvania	11/02/1973	11/12/1982
Susquehanna 2	BWR	BWR 3/4	Mark II	1050	Pennsylvania	11/02/1973	06/27/1984
Three Mile Island	PWR	B&W	Large Dry	819	Pennsylvania	05/18/1968	04/19/1974
Turkey Point 3	PWR	West 3-loop	Large Dry	666	Florida	04/27/1967	07/19/1972
Turkey Point 4	PWR	West 3-loop	Large Dry	666	Florida	04/27/1967	04/10/1973
Vermont Yankee	BWR	BWR 3/4	Mark I	522	Vermont	12/11/1967	02/28/1973
Vogtle 1	PWR	West 4-loop	Large Dry	1158	Georgia	06/28/1974	03/16/1987
Vogtle 2	PWR	West 4-loop	Large Dry	1158	Georgia	06/28/1974	03/31/1989
Waterford 3	PWR	CE	Large Dry	1104	Louisiana	11/14/1974	03/16/1985
Watts Bar 1	PWR	West 4-loop	Ice Condenser	1154	Tennessee	01/23/1973	02/07/1996
Wolf Creek 1	PWR	West 4-loop	Large Dry	1150	Kansas	05/31/1977	06/04/1985

* Formerly known as Washington Nuclear Project Number 2.

Table 3.2: Summary of fire CDFs

Plant name	Methodology used	Fire CDF reported in the IPEEE	Fire CDF after RAIs completed	Internal event CDF
ANO 1	FIVE +	4.17E-05	4.42E-05	4.67E-05
ANO 2	FIVE +	not reported	4.51E-05	3.40E-05
Beaver Valley 1	PRA	1.75E-05	1.75E-05	2.14E-04
Beaver Valley 2	PRA	1.05E-05	1.05E-05	1.92E-04
Braidwood 1	FIVE, PRA	2.50E-06	3.90E-06	2.74E-05
Braidwood 2	FIVE, PRA	2.40E-06	3.80E-06	2.74E-05
Browns Ferry 1	not analyzed, see Unit 2			
Browns Ferry 2	FIVE, PRA	6.78E-06	9.80E-06	4.80E-05
Browns Ferry 3	FIVE, PRA	4.38E-06	7.40E-06	
Brunswick 1	not analyzed, see Unit 2			
Brunswick 2	FIVE, PRA	3.39E-05	3.62E-05	2.70E-05
Byron 1	FIVE, PRA	2.40E-06	4.20E-06	3.09E-05
Byron 2	FIVE, PRA	2.50E-06	5.30E-06	3.09E-05
Callaway	FIVE, PRA	8.88E-06	8.88E-06	5.85E-05
Calvert Cliffs 1	FIVE, PRA	7.30E-05	7.20E-05	2.40E-04
Calvert Cliffs 2	FIVE, PRA	1.10E-04	1.10E-04	2.40E-04
Catawba 1	PRA	4.70E-06	4.63E-06	5.80E-05
Catawba 2	not analyzed, see Unit 1			
Clinton	PRA	3.26E-06	3.64E-06	2.66E-05
Columbia Generating*	FIVE, PRA	1.76E-05	5.50E-05	1.75E-05
Comanche Peak 1	FIVE, PRA	2.09E-05	2.09E-05	5.72E-05
Comanche Peak 2	not analyzed, see Unit 1			
Cook 1	PRA	1.61E-07	3.76E-06	6.26E-05
Cook 2	not analyzed, see Unit 1			
Cooper	FIVE, PRA	6.87E-06	6.87E-06	7.97E-05
Crystal River 3	FIVE, PRA	4.19E-05	4.19E-05	1.53E-05
Davis-Besse	FIVE, PRA	1.67E-05	2.97E-05	6.60E-05
Diablo Canyon 1	PRA	2.73E-05	2.73E-05	8.80E-05
Diablo Canyon 2	not analyzed, see Unit 1			
Dresden 2	FIVE, PRA	1.69E-05	1.69E-05	1.85E-05
Dresden 3	FIVE, PRA	3.08E-05	3.08E-05	1.85E-05
Duane Arnold	FIVE +	1.05E-05	1.05E-05	7.84E-06
Farley 1	FIVE, PRA	1.66E-04	1.66E-04	1.30E-04
Farley 2	FIVE, PRA	1.28E-04	1.28E-04	1.30E-04
Fermi 2	FIVE, PRA	1.70E-05	2.15E-05	5.70E-06
FitzPatrick	FIVE, PRA	2.00E-05	2.56E-05	1.92E-06
Fort Calhoun	FIVE, PRA	2.74E-05	2.74E-05	1.36E-05
Ginna	FIVE, PRA	6.40E-05	3.34E-05	8.38E-05
Grand Gulf 1	FIVE, PRA	8.76E-06	8.89E-06	1.72E-05
Haddam Neck	FIVE +	6.08E-05	6.08E-05	1.90E-04
Hatch 1	PRA	7.50E-06	7.80E-06	2.23E-05
Hatch 2	PRA	5.40E-06	5.80E-06	2.36E-05
Hope Creek 1	FIVE +	8.10E-05	8.10E-05	4.63E-05

Table 3.2: Summary of fire CDFs (Continued)

Plant name	Methodology used	Fire CDF reported in the IPEEE	Fire CDF after RAIs completed	Internal event CDF
Indian Point 2	FIVE, PRA	1.84E-05	1.84E-05	3.13E-05
Indian Point 3	FIVE, PRA	5.64E-05	5.64E-05	4.40E-05
Kewaunee	FIVE, PRA	9.81E-05	1.80E-04	6.65E-05
LaSalle 1	not analyzed, see Unit 2			
LaSalle 2	PRA	3.21E-05/ry	3.21E-05/ry	4.74E-05
Limerick 1	FIVE	not reported	not reported	4.30E-06
Limerick 2	not analyzed, see Unit 1			
McGuire 1	PRA	2.32E-07	6.74E-07	4.00E-05
McGuire 2	not analyzed, see Unit 1			
Millstone 1	no IPEEE submittal			
Millstone 2	FIVE +	6.30E-06	6.30E-06	3.42E-05
Millstone 3	FIVE +	4.80E-06	4.80E-06	5.61E-05
Monticello	FIVE +	7.90E-06	7.90E-06	2.60E-05
Nine Mile Point 1	FIVE, PRA	2.00E-05	2.00E-05	5.50E-06
Nine Mile Point 2	FIVE, PRA	1.40E-06	1.40E-06	3.10E-05
North Anna 1	FIVE, PRA	3.91E-06	3.91E-06	7.16E-05
North Anna 2	FIVE, PRA	4.08E-06	4.08E-06	7.16E-05
Oconee 1	not analyzed, see Unit 3			
Oconee 2	not analyzed, see Unit 3			
Oconee 3	PRA	5.80E-06	5.96E-06	2.30E-05
Oyster Creek	FIVE, PRA	7.70E-06	1.56E-05	3.90E-06
Palisades	FIVE	3.31E-05	3.31E-05	5.07E-05
Palo Verde 1	FIVE+	8.67E-05	8.67E-05	9.00E-05
Palo Verde 2	not analyzed, see Unit 1			
Palo Verde 3	not analyzed, see Unit 1			
Peach Bottom 2	FIVE	not reported	not reported	5.53E-06
Peach Bottom 3	not analyzed, see Unit 2			
Perry 1	FIVE, PRA	3.27E-05	3.27E-05	1.30E-05
Pilgrim 1	FIVE, PRA	2.20E-05	2.20E-05	5.80E-05
Point Beach 1	FIVE +	5.11E-05	5.28E-05	1.15E-04
Point Beach 2	not analyzed, see Unit 1			
Prairie Island 1	FIVE, PRA	4.93E-05	4.93E-05	5.05E-05
Prairie Island 2	not analyzed, see Unit 1			
Quad Cities 1	FIVE, PRA	5.40E-03	6.60E-05	1.20E-06
Quad Cities 2	FIVE, PRA	5.20E-03	7.31E-05	1.20E-06
River Bend 1	PRA +	2.25E-05	2.25E-05	1.55E-05
Robinson 2	FIVE, PRA	2.22E-04	9.23E-05	3.20E-04
Salem 1	FIVE, PRA	2.30E-05	2.30E-05	5.20E-05
Salem 2	FIVE, PRA	2.40E-05	2.40E-05	5.50E-05
San Onofre 2	FIVE, PRA	1.60E-05	1.60E-05	3.00E-05
San Onofre 3	not analyzed - see unit 2			
Seabrook 1	FIVE	1.20E-05	1.20E-05	6.60E-05
Sequoyah 1	FIVE, PRA	not reported	1.56E-05	1.70E-04

Table 3.2: Summary of fire CDFs (Continued)

Plant name	Methodology used	Fire CDF reported in the IPEEE	Fire CDF after RAIs completed	Internal event CDF
Sequoyah 2	FIVE, PRA	not reported	1.56E-05	1.70E-04
Shearon Harris 1	PRA +	1.10E-05	1.30E-05	7.00E-05
South Texas Project 1	PRA	not reported	5.06E-07	4.30E-05
South Texas Project 2	not analyzed, see Unit 1			
St. Lucie 1	FIVE +	1.87E-04	1.87E-04	2.30E-05
St. Lucie 2	FIVE +	1.87E-04	1.87E-04	2.62E-05
Summer	FIVE, PRA	4.04E-04	8.52E-05	2.00E-04
Surry 1	FIVE, PRA	6.28E-06	6.28E-06	1.25E-04
Surry 2	FIVE, PRA	6.28E-06	6.28E-06	1.25E-04
Susquehanna 1	FIVE, PRA, qualitative	(est.) < 1E-9/cycle	3.60E-08	8.96E-08
Susquehanna 2	not analyzed, see Unit 1			
Three Mile Island 1	FIVE, PRA	2.16E-05	2.16E-05	4.49E-05
Turkey Point 3	FIVE	1.94E-04	1.94E-04	3.73E-04
Turkey Point 4	not analyzed in full; no unit fire CDF given; for most areas see Unit 3			
Vermont Yankee	FIVE, PRA	3.80E-05	5.60E-05	4.30E-06
Vogtle 1	PRA	1.01E-05	1.01E-05	4.90E-05
Vogtle 2	not analyzed, see Unit 1			
Waterford 3	FIVE, PRA	7.00E-06	7.00E-06	1.80E-05
Watts Bar 1	FIVE, PRA	not reported	6.90E-06	8.00E-05
Wolf Creek 1	FIVE, PRA	7.59E-06	7.59E-06	4.20E-05

* Formerly known as Washington Nuclear Project Number 2.

Table 3.3: Significant fire area CDFs

Plant	Significant fire areas	CDF
ANO 1	Turbine room/hall/building	8.92E-06
	South battery room	6.50E-06
	Emergency diesel generator corridor	6.43E-06
	Electrical equipment room/auxiliary relay room	3.88E-06
	Main control room/control room	3.81E-06
	South switchgear room	3.72E-06
	Cable spreading room	3.01E-06
	Pipe room	2.48E-06
	Lower south electrical penetration room	2.02E-06
	North switchgear room	1.99E-06
	Controlled access exit	1.45E-06
ANO 2	Turbine room/hall/building	1.77E-05
	Cable spreading room	5.94E-06
	Diesel corridor	5.90E-06
	Switchgear room	3.59E-06
	Electrical equipment room/auxiliary relay room	3.32E-06
	Intake structure	2.85E-06
	Lower south electrical/piping penetration room	2.72E-06
	Main control room/control room	1.88E-06
	auxiliary building EXT or super-compartment	1.16E-06
	Switchgear room	5.30E-07
Beaver Valley 1	Cable spreading room	4.70E-06
	Cable vault or cable tunnel	4.27E-06
	Primary auxiliary building general area E	2.44E-06
	Switchgear room	1.42E-06
	Control room, process instrument room	1.22E-06
	Control room, general area	9.58E-07
	Total contribution from scenarios screened during spatial interactions phase	8.10E-07
	Turbine building, general area	3.81E-07
	Control room, communication equipment and relay panel room	2.70E-07
	Turbine generator area	2.34E-07
	Emergency switchgear 1AE room	1.89E-07
	Reactor containment area	1.83E-07
	Emergency switchgear 1DF room	1.19E-07
	Primary auxiliary building general area D	1.05E-07
	West cable vault area	4.36E-08
	Auxiliary feed water pumps room	4.31E-08
	Control room, HVAC equipment room	4.31E-08
	East cable vault area	2.37E-08
	Pipe tunnel penetration A cubicle	1.01E-08
	Turbine oil reservoir, coolers, and oil conditioner	9.08E-09
	Hydrogen seal oil unit	7.69E-09
	Safeguard building pipe tunnel general area	5.18E-09
	Turbine building to pipe tunnel area	8.94E-10
Beaver Valley 2	Main control room/control room	1.86E-06
	Cable vault or cable tunnel	1.32E-06
	Normal switchgear room	1.10E-06
	West cable vault area, elevation 735'	6.54E-07

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	West cable vault area, elevation 755'	6.12E-07
	Total contribution from scenarios screened during spatial interactions phase	6.01E-07
	Switchgear room	5.40E-07
	Switchgear room	5.10E-07
	Communication, instrumentation, and relay room	4.75E-07
	North safeguards area (including RSS cubicles)	3.49E-07
	Diesel generator building No. 1	3.25E-07
	Diesel generator building No. 2	3.08E-07
	East cable vault area, elevation 735'	3.00E-07
	MCC 2-E03 cubicle, PAB	2.95E-07
	South safeguards area (including RSS cubicles)	2.62E-07
	Battery room 2-5	2.52E-07
	Primary auxiliary building general area, elevation 760'	2.09E-07
	Electrical equipment room/auxiliary relay room	1.94E-07
	Cable spreading room	1.28E-07
	East and west communications rooms	1.19E-07
	Primary auxiliary building general area, elevation 718', 735'	4.89E-08
	Battery room 2-2	2.68E-08
	Reactor containment area	1.91E-08
	Service building cable tray area	1.62E-08
	Pipe tunnel	7.52E-09
	Main steam valve room	1.51E-09
	Turbine room/hall/building	6.05E-10
Braidwood 1	Unit 1 LCSR nonsegregated bus duct area	1.52E-06
	Unit 1 auxiliary electrical equipment room	7.00E-07
	auxiliary building general area elevation 426'	6.10E-07
	Div 11 miscellaneous electrical equipment room	5.40E-07
	auxiliary building general area elevation 401'	2.70E-07
	Cable vault or cable tunnel	1.01E-07
	Main control room/control room	6.32E-08
	Switchgear room	5.90E-08
	Turbine room/hall/building	5.86E-08
	Switchgear room	1.60E-08
	Auxiliary building general area elevation 383'	1.90E-09
	Auxiliary building general area elevation 364'	1.30E-09
	Div 12 miscellaneous electrical equipment room	2.80E-10
	Auxiliary building general area elevation 346'	1.50E-10
Braidwood 2	Unit 2 LCSR nonsegregated bus duct area	1.60E-06
	Div 21 miscellaneous electrical equipment room	7.00E-07
	Unit 2 auxiliary electrical equipment room	7.00E-07
	Auxiliary building general area elevation 401'	4.00E-07
	Auxiliary building general area elevation 426'	2.00E-07
	Main control room/control room	6.32E-08
	Cable vault or cable tunnel	5.91E-08
	Switchgear room	5.90E-08
	Turbine room/hall/building	5.86E-08
	Switchgear room	1.60E-08

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Auxiliary building general area elevation 383'	4.20E-10
	Div 22 miscellaneous electrical equipment room	2.90E-10
	Auxiliary building general area elevation 346'	1.50E-10
	Auxiliary building general area elevation 364'	5.10E-11
Browns Ferry 1	Not analyzed	
Browns Ferry 2	Main control room/control room	3.05E-06
	Unit 2 reactor building	1.50E-06
	Yard	7.95E-07
	Turbine room/hall/building	7.30E-07
	Unit 2 battery, battery board rooms	5.53E-07
	4kV shutdown board, room B	4.97E-07
	Control bay 593'	4.73E-07
	Intake pump station	4.72E-07
	4kV shutdown board, room C and 250 V battery room	4.51E-07
	Cable spreading room	4.48E-07
	4kV shutdown board, room D	4.15E-07
	4kV shutdown board, room A and 250 V battery room	2.54E-07
	Unit 3 reactor building	1.06E-07
	Unit 1 reactor building	5.19E-08
	Pipe tunnel	1.00E-10
Browns Ferry 3	Main control room/control room	3.05E-06
	Yard	7.95E-07
	Turbine room/hall/building	7.30E-07
	Control bay, 593'	4.72E-07
	Intake pump station	4.50E-07
	Cable spreading room	4.48E-07
	Shutdown board room F	4.43E-07
	Unit 3 reactor building	3.64E-07
	Unit 1 reactor building	2.22E-07
	Shutdown board room E	1.99E-07
	4Kv shutdown board rooms 3EA, 3EB	1.24E-07
	Unit 2 reactor building	1.00E-07
	Pipe tunnel	1.00E-10
Brunswick 1	Not analyzed, see Unit 2	
Brunswick 2	Main control room/control room	1.93E-05
	North central area (20')	4.72E-06
	Cable spreading room	1.56E-06
	Unit 1 cable spreading room (23')	1.56E-06
	NW area (20')	1.28E-06
	Switchgear room	1.10E-06
	Switchgear room	1.07E-06
Byron 1	Auxiliary building general area 426'	8.43E-07
	Auxiliary building general area 346'	7.40E-07
	Auxiliary building general area 401'	6.36E-07
	Laundry room, auxiliary building	4.60E-07
	Switchgear room	3.70E-07
	Auxiliary building general area 383'	2.57E-07

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Switchgear room	2.10E-07
	Div. 11 miscellaneous electrical equipment room (MEER)	1.80E-07
	Unit 1 LCSR nonseq. bus duct area, cable spreading room	1.30E-07
	Cable vault or cable tunnel	1.13E-07
	Auxiliary building general area 364'	9.50E-08
	Main control room/control room	8.17E-08
	Div. 12 MEER	5.10E-08
	U-1 AEER, auxiliary building	1.10E-08
	U-1 & U-2 turbine building, elevation 451'	6.10E-09
	Turbine room/hall/building	3.90E-11
Byron 2	U-2 upper cable spreading room (UCSR) Rm 3.3B-2	1.30E-06
	Auxiliary building general area 426'	9.91E-07
	Auxiliary building general area 346'	7.40E-07
	Auxiliary building general area 401'	6.45E-07
	Switchgear room	5.40E-07
	Auxiliary building general area 383'	2.61E-07
	Switchgear room	1.70E-07
	Div. 21 MEER, auxiliary building	1.60E-07
	Cable vault or cable tunnel	1.15E-07
	U-2 LCSR nonseg. bus duct area, cable spreading room	1.08E-07
	Auxiliary building general area 364'	9.10E-08
	Main control room/control room	8.17E-08
	Div. 22 MEER	5.10E-08
	U-1 & U-2 turbine building, elevation 451'	6.10E-09
	U-2 AEER	4.90E-09
	Turbine room/hall/building	3.90E-11
Callaway	Main control room/control room	2.65E-06
	Switchgear room	2.26E-06
	Safety-related ac switchgear room (C-10)	1.29E-06
	Cable spreading room	6.78E-07
	Auxiliary building area, elevation 2000'	5.32E-07
	Control room ac units room	4.08E-07
	Turbine room/hall/building	3.13E-07
	Switchgear room	2.27E-07
	Auxiliary building area, elevation 1988'	1.51E-07
	Auxiliary building area, elevation 1974'	8.61E-08
	Auxiliary building area, elevation 2026'	7.59E-08
	Cable chase (C-30)	6.37E-08
	Cable chase (C-33)	6.02E-08
	Communications corridor lower level	5.58E-08
	Cable chase (C-23)	2.42E-08
	Service water valve area	1.00E-10
Calvert Cliffs 1	Main control room/control room	2.45E-05
	Turbine room/hall/building	1.66E-05
	Unit 1 cable spreading room	6.72E-06
	Switchgear room	4.28E-06

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Transformers, tanks, and Independent structures	3.53E-06
	Battery room 21	3.52E-06
	Battery room 11	2.98E-06
	5' Multi-compartment area (fire area 11)	2.83E-06
	Switchgear room	1.54E-06
	Cable chases 1A, 1B, 2A, 2B	9.67E-07
	Unit 1 east piping penetration rooms	6.82E-07
	Unit 2 cable spreading room	6.81E-07
	Unit 1 AFW pump room	4.76E-07
	Unit 1 ECCS pump rooms and recirculation tunnel	4.49E-07
	2A diesel generator room	4.00E-07
	Unit 2 27' switchgear room	2.22E-07
	Cask and equipment loading area-truck bay	1.86E-07
	1B diesel generator room	1.39E-07
	Cross-zone fire initiators	1.26E-07
	69' multi-compartment area (fire area 11)	1.22E-07
	Unit 2 45' switchgear room	1.19E-07
	Unit 2 service water, component cooling, and radiation exhaust rooms	1.13E-07
	Unit 2 purge air room	1.12E-07
	2B diesel generator room	1.02E-07
	Unit 1 69' electrical room	9.18E-08
	minus 10' / minus 15' hallways	8.58E-08
	Unit 1 east electric penetration room	7.10E-08
	Unit 1 west electric penetration room	6.78E-08
	Battery room 12	5.78E-08
	Battery room 22	5.18E-08
	27' multi-compartment area (fire area 11)	4.62E-08
	Unit 1 RWT pump room	4.39E-08
	Unit 2 ECCS pump rooms	4.22E-08
	Unit 1 charging pump rooms	3.48E-08
	Unit 1 service water pump room	3.47E-08
	Unit 2 west electric penetration room	3.43E-08
	Unit 1 main vent fan room	2.37E-08
	Hallways outside the control room	1.87E-08
	Reactor coolant waste evaporator room	1.56E-08
	Intake structure	1.22E-08
	Reactor coolant waste tank rooms	1.05E-08
	Unit 2 AFW pump room	8.88E-09
	Unit 1 radiation exhaust equipment room	3.99E-09
	Unit 2 main steam piping area	3.65E-09
	Unit 2 east electrical penetration room	3.27E-09
	Unit 1 west piping penetration rooms	2.62E-09
	auxiliary building stair tower AB-2	2.29E-09
	Control room HVAC room	1.91E-09
	Unit 1 purge air room	1.15E-09
Calvert Cliffs 2	No information provided.	

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
Catawba 1	"Short room" near KC pumps	3.74E-06
	Control room and cable room	9.40E-07
	Cable vault or cable tunnel	6.96E-07
	Main control room/control room	1.52E-07
	Switchgear room	2.28E-08
	Turbine room/hall/building	1.52E-08
	Diesel generator room A	1.00E-08
	Nuclear service water pump room	1.00E-08
	Switchgear room	1.00E-08
	Turbine driven auxiliary feedwater pump room	< 1.0E-08
	Vital I & C room	< 1.0E-08
Catawba 2	Not analyzed, see Unit 1	
Clinton	Main control room/control room	1.20E-06
	781' auxiliary building, Div 1 safety switchgear room	7.11E-07
	781' & 790' control, DC/UPS equipment area	4.84E-07
	678' & 699' screenhouse, general access and pipe tunnel areas	3.38E-07
	762' auxiliary building, Div 1 non-safety switchgear room	2.95E-07
	781' auxiliary building, Div 2 safety switchgear room	2.00E-07
	781' control, Div 3 switchgear area	1.36E-07
	762 auxiliary building, Div 2 non-safety switchgear room	1.27E-07
	762' control, component cooling water equipment area	6.85E-08
	800' control, operations kitchen/restroom/storage areas	3.72E-08
	719' control, entire level excluding stairwells	2.04E-08
	737' fuel building, general access area	1.11E-08
	781' radwaste, general access area	4.75E-09
	781' control, Div 2 cable spreading area	3.97E-09
	781' control, Div 1 cable spreading room	1.39E-09
	737' & 751' control, general access and lab HVAC areas	1.14E-09
	755' & 781' fuel building, entire area of both elevations	1.13E-09
	737' control, chemistry laboratory areas	5.36E-10
	712' fuel building, general access area	4.28E-10
	707'6" auxiliary building, hallway	3.25E-10
	737' auxiliary building, general access area	1.79E-10
Columbia Generating*	Electrical equipment room/auxiliary relay room	1.05E-05
	Main control room/control room	8.40E-06
	Div. 2 electrical equipment room	7.67E-06
	Remote shutdown room	6.67E-06
	Div. 2 battery room	5.06E-06
	Switchgear room	4.64E-06
	Turbine general corridor	3.67E-06
	Reactor building 501'	2.41E-06
	Reactor building 471'	2.40E-06
	Reactor building 522'	1.18E-06
	Div. 1/2 electrical/battery room corridor	1.14E-06
	Div. 1 battery room	7.98E-07
	NW reactor building 471'	4.87E-07
	Equipment hatch	3.77E-07

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Switchgear room (switchgear room #1)	3.36E-07
	NW reactor building 501'	2.90E-07
	Turbine generator building west 441'	1.06E-07
	NE reactor building 471'	7.35E-08
Comanche Peak 1	Main control room/control room	9.04E-06
	Train A electrical equipment area, safeguards building, elevation 810'	1.74E-06
	Train B electrical equipment area, safeguards building. elevation 852'	1.45E-06
	UPS & distribution room - train B, E, & C building. elevation 792'	1.34E-06
	810' safeguards building corridor	1.26E-06
	UPS & distribution room - train A, E, & C building. elevation 792'	1.25E-06
	AA021	7.99E-07
	Multi-compartment analysis	7.25E-07
	SB004	7.10E-07
	Cable spreading room	4.69E-07
	SG010	3.72E-07
	SI012	3.72E-07
	EA057	3.16E-07
	EQ149	2.50E-07
	ER150	2.49E-07
	AA153	2.33E-07
	SK017	1.87E-07
	SB015	7.32E-08
	SE016	5.36E-08
	EA043	1.04E-09
Comanche Peak 2	Not analyzed, see Unit 1	
Cook 1	44S - auxiliary building S - both units	3.80E-07
	16 - 1AB diesel generator room - U1	3.50E-07
	15 - 1 CD diesel generator room - U1	3.04E-07
	40B - 4 kV CD switchgear room	1.86E-07
	53 - U1 Control room	1.81E-07
	42D - EPS AB battery room	1.68E-07
	40A - 4 kV AB switchgear room	1.32E-07
	41 - engineering safety system & MCC room (& under floor) - U1	1.12E-07
	29B - ESW pump PP-1W - U1	1.07E-07
	29E - MCC for ESW pumps - U1	1.07E-07
	91 - turbine room SE portion - U1	1.02E-07
Cook 2	Not analyzed, see Unit 1	3.80E-07
Cooper	Switchgear room	2.72E-06
	Main control room/control room	1.71E-06
	Service water pump room	1.33E-06
	Switchgear room	1.11E-06
	Cable spreading room	8.23E-07
	Reactor building, elevation 903'6"	8.16E-07
	Div II dc switchgear room	7.90E-07
	Train B RPS room	7.30E-07
	Div. II battery room	6.73E-07
	RHRSW booster and service air compressor	5.58E-07

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Auxiliary relay room	3.66E-07
	Electrical equipment room/auxiliary relay room	3.66E-07
	Cable expansion room	3.45E-07
	Div. I dc switchgear room	3.36E-07
	Non-critical switchgear room	3.32E-07
	Reactor building, elevation 932'6"	2.73E-07
	Div. I battery room	1.77E-07
	Turbine room/hall/building	1.41E-07
	Train A RPS room	6.52E-08
	Seal water pump area	4.86E-08
Crystal River 3	Battery charger room 3A	1.49E-05
	Switchgear room	7.31E-06
	Switchgear room	6.79E-06
	480V ES switchgear room BUS 3A	3.79E-06
	Battery charger room 3B	2.72E-06
	Hallway and remote shutdown room	2.66E-06
	CRD and communication equipment room	1.58E-06
	Inverter room 3B	1.45E-06
	Main control room/control room	5.70E-07
	Central hallway	*3.86E-07*
	480V ES switchgear room BUS 3B	*1.76E-07*
	Cable spreading room	9.90E-08
	North hallway and nuclear sample room	*7.98E-09*
	Turbine building mezzanine floor	*4.96E-11*
	Turbine building basement floor	*4.84E-11*
	RWSW pump room	*4.21E-11*
	Turbine EFW, pump penetration area, fan room	*1.07E-11*
	North hallway	*1.05E-11*
Davis-Besse	High voltage switchgear room B	8.20E-06
	High voltage switchgear room A	6.46E-06
	Main control room/control room	6.40E-06
	Low voltage switchgear room	6.00E-06
	Liquid radwaste equipment area	2.60E-06
	Turbine room/hall/building	2.30E-07
Diablo Canyon 1	Cable spreading room	9.99E-06
	Main control room/control room	8.97E-06
	Fire initiators FS1, FS5, & FS6 which consist of various scenarios initiating in various fire zones.	8.30E-06
Diablo Canyon 2	Not analyzed, see Unit 1	
Dresden 2	Main control room/control room	7.15E-06
	Unit 2 north trackway/switchgear area	5.38E-06
	Unit 2 second floor reactor building	1.65E-06
	Unit 2 mezzanine	6.74E-07
	Control room backup ventilation	5.86E-07
	Electrical equipment room/auxiliary relay room	5.36E-07
	Unit 2/3 turbine building corridor	2.52E-07

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Cribhouse upper	2.45E-07
	Unit 3 west corridor and trackway	1.17E-07
	Unit 2 torus basement	1.10E-07
	Unit 2 battery room	5.89E-08
	Unit 2 reactor building ground floor	4.69E-08
	Unit 3 mezzanine floor	3.95E-08
	Unit 2 reactor building switchgear area	1.37E-08
	Unit 2/3 standby gas treatment system and TBCCW heat exchanger	5.29E-09
	Unit 3 battery charger room	4.48E-09
Dresden 3	Main control room/control room	7.11E-06
	Unit 3 west corridor and trackway	6.85E-06
	Unit 3 mezzanine floor	4.23E-06
	Unit 3 second floor reactor building	3.54E-06
	Electrical equipment room/auxiliary relay room	2.53E-06
	Cable vault or cable tunnel	2.12E-06
	Turbine building corridor	8.36E-07
	Unit 3 reactor building ground floor	7.16E-07
	Unit 2/3 standby gas treatment system and TBCCW heat exchanger	5.32E-07
	Unit 2 north trackway/switchgear area	4.94E-07
	Unit 3 cond. PP area	4.85E-07
	Control room backup ventilation	4.59E-07
	Vent room over NE switchgear	2.39E-07
	Cribhouse upper	2.38E-07
	Unit 3 DG	2.19E-07
	Unit 3 traveling in-core probe room	1.10E-07
	Unit 3 reactor building switchgear area	2.06E-08
	Unit 2 battery room	1.59E-08
	Unit 3 torus basement	7.57E-09
	Unit 3 battery charger room	1.07E-09
	Cable spreading room	
	Switchgear room	
	Switchgear room	
	Turbine room/hall/building	
Duane Arnold	Switchgear room	5.61E-06
	Switchgear room	4.92E-06
	Main control room/control room	5.02E-07
	Cable spreading room	2.33E-07
Hatch 1	Cable spreading room	1.93E-06
	Switchgear room	1.45E-06
	Switchgear room	1.38E-06
	Main control room/control room	7.10E-07
	4kV switchgear room 1F	7.04E-07
	Common area housing units 1 & 2 main control rooms	3.10E-07
	East cableway	2.05E-07
	Reactor building north working floor on elevation 130 ft. (Control rod drive area)	2.03E-07
	Station battery room 1A - Division I	1.49E-07
	Vertical cable chase	1.04E-07

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Annunciator room	9.94E-08
	West 600V switchgear room 1C - Division I	5.76E-08
	Control building north and south corridor on elevation 130 ft.	5.73E-08
	West dc switchgear room 1A - Division I	4.26E-08
	Computer room	4.23E-08
	Cable spreading room (fires initiated in cables of adjacent unit)	1.80E-08
	Control building, vertical cableway (fires initiated in cables of adjacent unit)	1.50E-08
	Turbine building, west cableway (fires initiated in cables of adjacent unit)	8.80E-10
Hatch 2	Switchgear room	1.10E-06
	Switchgear room	8.88E-07
	Cable spreading room	8.55E-07
	Main control room/control room	7.10E-07
	4kV switchgear room 2F	4.66E-07
	Common area housing units 1 & 2 main control rooms	3.10E-07
	West 600V switchgear room 2C - Division I	2.89E-07
	East cableway	1.89E-07
	Station battery room 2A - Division I	1.40E-07
	Reactor building south working floor on elevation 130 ft. (Control rod drive area)	9.51E-08
	Reactor building north working floor on elevation 130 ft. (Control rod drive area)	7.03E-08
	Computer room	5.70E-08
	Turbine room/hall/building	4.80E-08
	Reactor building north torus chamber	4.62E-08
	Control building working floor and corridor on elevation 112 ft.	4.49E-08
	Vertical cable chase	4.39E-08
	West dc switchgear room 2A - Division I	4.16E-08
	Reactor building south torus chamber	3.98E-08
	Control building, vertical cableway (fires initiated in cables of adjacent unit)	3.20E-08
	Cable spreading room (fires initiated in cables of adjacent unit)	1.60E-08
	Turbine building, west cableway (fires initiated in cables of adjacent unit)	6.40E-09
Fermi	Main control room/control room	7.36E-06
	Switchgear room	4.51E-06
	Electrical equipment room/auxiliary relay room	2.77E-06
	Turbine room/hall/building	2.72E-06
	Switchgear room	2.54E-06
	Third floor auxiliary room - major Div. I portion	1.90E-06
	NE quadrant reactor building rooms	1.45E-06
	Second floor reactor building	1.00E-06
	Cable vault or cable tunnel	4.08E-07
	Cable spreading room	1.05E-07
Fort Calhoun	Main control room/control room	7.90E-06
	AFW and air compressor area	6.01E-06
	Turbine room/hall/building	3.97E-06
	Upper electrical penetration area	3.62E-06
	Auxiliary building lower corridor and adjoining rooms	2.05E-06
	Switchgear room	7.84E-07
	Transformer yard area	6.18E-07
	Intake structure	5.96E-07

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Group I MCC area	5.66E-07
	Lower electrical penetration area	5.30E-07
	Ground level general area - auxiliary building	3.14E-07
	Charging pump area - auxiliary building	2.45E-07
	Switchgear room	2.27E-07
	Cable spreading room	7.42E-08
Grand Gulf	Main control room/control room	3.85E-06
	Switchgear room	9.30E-07
	Auxiliary building corridors, 139'-0" elevation	6.70E-07
	Auxiliary building corridors, 119' -0" elevation	6.19E-07
	Switchgear room	6.08E-07
	Auxiliary building corridors, 93' -0" elevation	5.74E-07
	Division 2 switchgear room - control building, 111' -0" elevation	4.06E-07
	Turbine room/hall/building	3.24E-07
	Lower cable room - control building, 148' -0" elevation	2.82E-07
	Hot machine shop - control building, 93' -0" elevation	2.42E-07
	HVAC equipment room - control building, 133' -0" elevation	2.10E-07
	Division 3 (HPCS) diesel generator building	1.72E-07
	Turbine building, 113' -0" elevation	7.10E-09
Haddam Neck	Switchgear room	2.59E-05
	Main control room/control room	1.40E-05
	Primary auxiliary building	1.17E-05
	Diesel generator room B	6.50E-06
	Cable spreading room	9.44E-07
	Intake structure	7.13E-07
	Diesel generator room A	4.50E-07
	Turbine room/hall/building	1.08E-07
	Switchgear room	1.03E-07
Hope Creek	Main control room/control room	2.51E-05
	Switchgear room	1.30E-05
	Diesel generator room (channel A)	5.30E-06
	Reactor building, CRD pump area	4.15E-06
	Diesel generator room (channel C)	4.10E-06
	Diesel generator room (channel B)	3.70E-06
	Auxiliary building 137' elevation, electrical access area	3.07E-06
	Switchgear room	3.00E-06
	Auxiliary building, upper control equipment room/computer room	2.68E-06
	Auxiliary building 102' elevation, electrical access room	2.67E-06
	Diesel generator room (channel D)	2.60E-06
	Auxiliary building124' elevation, electrical access area	2.07E-06
	Reactor building, 102' elevation-north side and Division I SACS area	1.77E-06
	Auxiliary building, lower (control) electric equipment room	1.73E-06
	Turbine building., access and unloading area	1.23E-06
	Reactor building, motor-control center (MCC) area	1.12E-06
	Turbine building., electrical equipment mezzanine	7.79E-07
	Radwaste building, middle section of the 3rd floor	7.20E-07
	Reactor building, RACS pump & heat exchanger area	6.34E-07

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Auxiliary building, DG area HVAC equipment room	5.30E-07
	Switchyard blockhouse	3.00E-07
	Control room/equipment room/mezzanine	2.90E-07
	Auxiliary building, Class 1E inverter room	2.16E-07
	Reactor building, RHR pump DP202 room	1.58E-07
	Auxiliary building, electrical access area (Div. I)	8.94E-08
	Cable spreading room	5.86E-08
	Auxiliary building, DG combustible air intake room	5.64E-08
	Reactor building, 102' elevation-inside cylinder - south side (Div. II)	4.06E-08
	Reactor building, torus water cleanup room/motor-control center (MCC)	3.01E-08
	Electrical equipment room/auxiliary relay room	1.81E-08
	Reactor building, RHR heat exchanger room (BP202 & heat exchanger BE205)	1.52E-08
	Reactor building, core spray DP206 room	1.21E-08
	Reactor building, RHR pump A202 & HX AE205 room (and vestibule)	1.07E-08
	Auxiliary building, electrical access area/corridor	9.17E-09
	Auxiliary building, HVAC equipment room	9.00E-09
	Reactor building, 102' elevation-inside cylinder - north side (Division I)	8.10E-09
	Aux building, auxiliary Electrical access area & common area in RW building	5.00E-09
	Auxiliary building, DG combustible air intake room	1.57E-09
Indian Point 2	Main control room/control room	7.07E-06
	Cable spreading room	4.28E-06
	Switchgear room	3.84E-06
	Electrical penetration area	1.11E-06
	Primary water makeup area	1.05E-06
	Electrical tunnel/pipe penetration area	9.19E-07
	Cable vault or cable tunnel	9.62E-08
	SW intake	7.46E-09
	Auxiliary feedwater pump room	6.15E-09
	CCW pump room	2.19E-09
	Drumming and storage station	1.53E-09
Indian Point 3	Switchgear room	3.51E-05
	Cable spreading room	6.83E-06
	Switchgear room	4.49E-06
	Main control room/control room	3.65E-06
	Diesel generator 31 room	2.13E-06
	Diesel generator 33 room	1.93E-06
	Upper electrical tunnel	7.14E-07
	Diesel generator 32 room	3.38E-07
	Lower electrical tunnel	2.78E-07
	Auxiliary feedwater pump room	2.28E-07
	Turbine room/hall/building	3.78E-08
	PAB corridor	3.17E-08

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
FitzPatrick	Cable spreading room	6.57E-06
	Relay room	5.40E-06
	Main control room or control room	3.00E-06
	Reactor building westside (elevation. 272) and southwest quadrant (elevation. 300)	1.35E-06
	Trains "A & C" EDG switchgear room south (elevation. 272)	1.32E-06
	Reactor building east crescent (elevation. 227 & 242)	1.02E-06
	West cable tunnel	7.21E-07
	Turbine building/relay room	6.26E-07
	Trains "B & D" EDG switchgear room north (elevation. 272)	6.05E-07
	South cable tunnel/relay room	5.52E-07
	"A" train battery room 2	4.62E-07
	North cable tunnel/relay room	4.62E-07
	Relay room/administration building, office area, records area, computer rooms & technical support center (elevation. 286)	3.80E-07
	Turbine room or hall or building	3.73E-07
	"B" train battery room 3	3.30E-07
	Turbine building/east cable tunnel	2.89E-07
	Administration building, Machine shop, locker rooms, stores & lunch room/reactor building eastside (elevation. 272'), southeast quadrant (elevation. 300'), entire floor at elevation 326, 344, & 369	2.54E-07
	"A" train battery charger room 1	2.40E-07
	"B" train battery charger room 4/"A" Train Battery Charger Room 1	2.24E-07
	East cable tunnel	2.24E-07
	Reactor building northeast & northwest quadrants (elevation. 300')	2.19E-07
	Motor generator set room & fan room/reactor building, northeast & northwest quadrants (elevation. 300')	2.06E-07
	Reactor building westside (elevation. 272) and southwest quadrant (elevation. 300')	1.52E-07
	Administration building, machine shop, locker rooms, stores & lunch room/cable spreading room	1.37E-07
	Reactor building, east crescent (elevation. 227 & 242)/reactor building. Eastside (elevation. 272), southeast quadrant (elevation. 300'), entire floor at elevations 326, 344, & 369	1.11E-07
	"B" train battery charger room 4	1.04E-07
	Reactor building west crescent (elevation 227 & 242)	9.52E-08
	Battery rooms corridor (elevation 272)	8.50E-08
	Standby gas filter room/reactor building, eastside (elevation. 272), southeast quadrant (elevation. 300'), entire floor at elevations 326, 344, & 369	3.72E-08
	Reactor building eastside (elevation. 272'), southeast quadrant (elevation. 300'), entire floor at elevations 326, 344 & 369	2.25E-08
	Relay room/reactor building, eastside (elevation 272), southeast quadrant (elevation 300'), entire floor at elevations 326, 344, & 369	1.33E-08
	Relay room/turbine building	1.33E-08
	Turbine building, west electric bay (elevation. 272')	6.61E-10
	Turbine building, east electric bay (elevation. 272')	2.37E-10

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
Farley 1	Auxiliary building 4160V switchgear room, train A	5.43E-05
	Train A electrical penetration room	2.90E-05
	Main control room/control room	2.68E-05
	Service water pump room	1.40E-05
	Component cooling water heat exchanger/pump room	1.27E-05
	Auxiliary building 4160V switchgear room, train B	8.38E-06
	Diesel building train A switchgear room	7.83E-06
	Low-voltage switchyard	4.12E-06
	Train B electrical penetration room	2.53E-06
	Cable spreading room	2.02E-06
	Turbine room/hall/building	1.47E-06
	Auxiliary building train B dc switchgear room	9.49E-07
	Auxiliary building train A dc switchgear room	5.50E-07
	Lower equipment room	3.66E-07
	Auxiliary building, 121' hallway	3.42E-07
	Auxiliary building, elevation 155'	2.51E-07
	Diesel building train B switchgear room	3.39E-08
Farley 2	Auxiliary building 4160V switchgear room, train A	5.26E-05
	Main control room/control room	2.68E-05
	Service water Pump Room	1.40E-05
	Auxiliary building 4160V switchgear room, train B	8.28E-06
	Diesel building train A switchgear room	7.83E-06
	Component cooling water heat exchanger/pump room	7.70E-06
	Cable spreading room	4.43E-06
	Low-voltage switchyard	4.12E-06
	Auxiliary building, elevation 155'	6.31E-07
	Turbine room/hall/building	6.01E-07
	Lower equipment room	4.65E-07
	Diesel building train B switchgear room	3.39E-08
	Boric acid area (auxiliary building, elevation 100')	3.26E-08
	Chemical drain tankroom	1.43E-08
Kewaunee	Diesel generator room B	6.15E-05
	Auxiliary feedwater pump A room	5.27E-05
	Main control room/control room	3.20E-05
	Auxiliary feedwater pump B room and 480V switchgear buses 61 and 62 room	2.97E-05
	Switchgear room	3.33E-06
	Electrical equipment room/auxiliary relay room	3.21E-07
	Auxiliary building, refueling water storage tank area, corridor	2.81E-09
	Switchgear room	1.41E-09
La Salle 1	Not analyzed, see Unit 2	
La Salle 2	Main control room/control room	1.39E-05
	Switchgear room	8.51E-06
	Switchgear room	5.15E-06
	Auxiliary equipment room	2.63E-06
	Turbine building corridor	6.20E-07
	Electrical equipment room/auxiliary relay room	5.73E-07

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Cable shaft area	5.42E-07
	Cable spreading room	1.63E-07
	Auxiliary building rad chem offices	3.58E-08
	BOP cable area north	7.31E-09
Limerick 1	No information provided.	
Limerick 2	No information provided.	
McGuire 1	Vital I & C room	6.47E-07
	Control room or cable room	8.13E-08
	Turbine room/hall/building	2.72E-08
	Main feed pump	2.02E-08
	Main control room/control room	1.00E-08
	Switchgear room	1.00E-08
	Switchgear room	1.00E-08
McGuire 2	Not analyzed, see Unit 1	
Nine Mile Point 1	Turbine building elevation 261', south	1.30E-05
	Cable spreading room	2.00E-06
	Main control room/control room	1.40E-06
	Auxiliary control room	1.10E-06
	Turbine building, elevation 250', south & west	1.00E-06
Nine Mile Point 2	Main control room/control room	1.40E-06
North Anna 1	Switchgear room	3.28E-06
	Cable vault or cable tunnel	4.51E-07
	Main control room/control room	1.69E-07
	Auxiliary building	1.24E-08
North Anna 2	Switchgear room	3.28E-06
	Cable vault or cable tunnel	4.51E-07
	Auxiliary building	1.81E-07
	Main control room/control room	1.69E-07
Oconee 1	Not analyzed, see Unit 3	
Oconee 2	Not analyzed, see Unit 3	
Oconee 3	Turbine room/hall/building	5.80E-06
	Cable shaft	1.56E-07
Oyster Creek	Cable spreading room	8.60E-06
	Switchgear room	5.10E-06
	Turbine building basement (south end)	1.90E-06
	Main control room/control room	3.30E-07
	Switchgear room	3.10E-07
Palisades	Cable spreading room	1.11E-05
	Main control room/control room	8.10E-06
	Switchgear room	4.89E-06
	Switchgear room	2.51E-06
	Turbine building (east side)	2.15E-06
	West engineered safeguards	1.11E-06
	Turbine building (south side)	8.64E-07
	Auxiliary building 590' corridor (south finger)	6.73E-07
	Intake structure - service water system	4.59E-07

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Battery room No.2	2.77E-07
	Auxiliary feedwater pump room	1.92E-07
	Battery room No. 1	1.62E-07
	590' auxiliary building (all not included other zones)	1.60E-07
	Diesel generator 1-2 room	1.32E-07
	Diesel generator 1-1 room	9.54E-08
	Turbine building (west side)	7.28E-08
	Electrical equipment room/auxiliary relay room	5.83E-08
	Engineering safeguards panel room	3.35E-08
	Auxiliary building 590' corridor (middle finger)	2.26E-08
	Spent fuel pool equipment room	2.19E-08
	East engineered safeguards	2.04E-08
	Manholes Nos. 1, 2, 3	1.01E-08
	Component cooling pump room	9.20E-09
	Charging pump room	5.36E-09
Palo Verde 1	Switchgear room	2.07E-05
	Turbine room/hall/building	1.47E-05
	Main control room/control room	1.07E-05
	Switchgear room	9.73E-06
	Corridor building., 120 ft. elevation	9.36E-06
	Train B containment electrical penetration room	7.14E-06
	Channel A dc equipment room	6.74E-06
	Channel B dc equipment room	3.97E-06
	Train A containment electrical penetration room	3.62E-06
	Electrical equipment room/auxiliary relay room	9.15E-07
Palo Verde 2	Not analyzed, see Unit 1	
Palo Verde 3	Not analyzed, see Unit 1	
Peach Bottom 2	Turbine building U3 wing area (see comment)	3.70E-06
	Reactor building U2 north (see comment)	3.20E-06
	Reactor building U3 north (see comment)	2.70E-06
	4kV switchgear room (34) (see comment)	2.50E-06
	4kV switchgear room (32) (see comment)	2.20E-06
	Turbine building 13.2kV switchgear area (see comment)	1.40E-06
Peach Bottom 3	Not analyzed, see Unit 2	
Perry	Main control room/control room	1.06E-05
	Switchgear room	1.05E-05
	Switchgear room	3.38E-06
	Unit 1 turbine power complex switchgear room	3.28E-06
	Control complex, elevation 574' 0"	2.03E-06
	Fuel handling building, elevation 620'	1.63E-06
	Turbine room/hall/building	1.30E-06
Pilgrim	Switchgear room	6.10E-06
	Switchgear room	3.10E-06
	Vital motor generator set room	2.40E-06
	Turbine building heater bay	2.10E-06
	Train "B" RBCCW/TBCCW pump and heat exchanger room	2.00E-06

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Main Control room/control room	1.60E-06
	Main transformer	1.50E-06
	Train "A" RBCCW/TBCCW pump and heat exchanger room	9.80E-07
	Reactor building west, elevation 21	9.70E-07
	Cable spreading room	9.50E-07
Point Beach 1	Gas-fired turbine generator area	2.04E-05
	Diesel generator room G02	5.84E-06
	Diesel generator room G01	5.52E-06
	Monitor tank room auxiliary operator's station	4.95E-06
	Main control room/control room	4.58E-06
	Switchgear room	3.70E-06
	Cable spreading room	2.68E-06
	Switchgear room	2.51E-06
	MCC 2B-32 room outside charging pump rooms	1.07E-06
	Electrical equipment room/auxiliary relay room	3E-07
Point Beach 2	Not analyzed, see Unit 1	
Prairie Island 1	Main control room/control room	3.22E-05
	"B" train hot shutdown panel & air compressor/AFW room	8.23E-06
	Switchgear room	2.24E-06
	Switchgear room	1.74E-06
	Auxiliary building mezzanine floor Unit 1	1.45E-06
	Auxiliary building ground floor Unit 2	1.28E-06
	Cable spreading room	1.08E-06
	Turbine room/hall/building	1.08E-06
Prairie Island 2	Not analyzed, see Unit 1	
Quad Cities 1	Turbine room/hall/building	2.03E-05
	Main control room/control room	1.00E-05
	Auxiliary transformer 11 and reserve auxiliary transformer 12	6.64E-06
	Unit 1 mezzanine floor	4.56E-06
	Cable vault or cable tunnel	3.25E-06
	Switchgear room	3.18E-06
	Unit 2 cable tunnel	3.16E-06
	Unit 2 turbine building ground floor	2.64E-06
	Auxiliary electric room	2.38E-06
	Unit 1/2 mezzanine floor	2.04E-06
	Cable spreading room (only or upper)	1.84E-06
	Unit 1 dc panel room	1.75E-06
	Unit 1 reactor building mezzanine level	1.51E-06
	Unit 1/2 turbine building ground floor	1.21E-06
	Unit 1 torus	1.09E-06
	Unit 2 upper basement	2.80E-07
	Unit 2 mezzanine floor	6.54E-09
	Unit 1 traveling in-core probe room	5.47E-09
	Unit 1 condensate pump room	2.06E-09
	Unit 1 MSIV room	1.06E-09
	Old computer room	4.91E-10

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Unit 1 reactor building main floor	4.28E-10
	Unit 1 upper basement	2.43E-11
Quad Cities 2	Turbine room/hall/building	2.28E-05
	Cable vault or cable tunnel	1.12E-05
	Main control room/control room	1.00E-05
	Unit 2 mezzanine floor	3.43E-06
	Unit 1/2 turbine building ground floor	3.25E-06
	Switchgear room	3.20E-06
	Unit 2 dc panel room	2.23E-06
	Cable spreading room	1.05E-06
	Unit 2 torus area	9.17E-07
	Unit 2 reactor building mezzanine floor	8.49E-07
	Auxiliary electric room	4.83E-07
	Old computer room	4.16E-07
	Unit 2 southeast corner room	3.04E-07
	Unit 1 reactor building mezzanine level	1.32E-07
	Unit 1 turbine building ground floor	1.07E-07
	Unit 2 reactor building ground floor	1.54E-08
	Unit 1/2 mezzanine floor	7.84E-09
	Unit 1 mezzanine floor	1.53E-09
	Unit 2 condensate pump room	6.71E-10
Ginna	Main control room	7.95E-06
	Turbine building mezzanine level	4.61E-06
	Turbine building basement level	3.90E-06
	Battery room 1A, control building	2.27E-06
	Control building relay room	2.14E-06
	Transformer yard	1.94E-06
	Auxiliary building, basement level	1.90E-06
	Battery room 1B, control building	1.67E-06
	Diesel generator room 1B	1.50E-06
	Containment	1.44E-06
	Auxiliary building, mezzanine level	1.20E-06
	Screen house basement level	1.00E-06
	Auxiliary building, operating level	9.69E-07
	Intermediate building basement level north	4.00E-07
	Cable vault/cable tunnel	2.70E-07
	Air handling room	1.34E-07
	Technical support center	3.34E-08
River Bend	Main control room/control room	4.87E-06
	Switchgear room	4.75E-06
	Control room ventilation room (elevation 116')	4.56E-06
	ACU west room	3.31E-06
	HPCS & HPCS hatch area	2.23E-06
	Cable vault or cable tunnel	1.48E-06
	Auxiliary building, west side crescent area	1.26E-06

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
Robinson	Electrical equipment room/auxiliary relay room	2.38E-05
	Yard transformers	1.99E-05
	Cable spreading room	1.50E-05
	Auxiliary building hallway (ground floor)	1.11E-05
	Main control room/control room	6.86E-06
	Service water pump area	4.37E-06
	Emergency diesel generator "B" room	3.92E-06
	Turbine room/hall/building	3.85E-06
	Battery room	2.61E-06
	Diesel generator "A" room	8.53E-07
Salem 1	Electrical equipment room/auxiliary relay room	7.20E-06
	Main control room/control room	7.00E-06
	Switchgear room	1.70E-06
	Switchgear room	1.70E-06
	Lower electrical penetration area	1.40E-06
	Upper electrical and piping penetration area	1.30E-06
	Reactor plant auxiliary equipment area, auxiliary building elevation 84'	1.10E-06
	Turbine room/hall/building	6.40E-07
	Service water intake	4.20E-07
	Reactor plant auxiliary equipment area, auxiliary building elevation 100'	2.90E-07
	Service water duct, manhole	2.10E-07
	CO_2 equipment room	6.00E-08
	Reactor plant auxiliary equipment area, auxiliary building elevation 45'	9.40E-09
	Reactor plant auxiliary equipment area, auxiliary building elevation 45'	9.30E-09
	Mechanical penetration area	7.30E-09
	Reactor plant auxiliary equipment area, auxiliary building elevation 64'	5.10E-09
	Hallway of the auxiliary building elevation 100'	1.50E-09
Salem 2	Electrical equipment room/auxiliary relay room	7.20E-06
	Main control room/control room	7.00E-06
	Switchgear room	1.70E-06
	Switchgear room	1.70E-06
	Lower electrical penetration area	1.40E-06
	Upper electrical and piping penetration area	1.30E-06
	Reactor plant auxiliary equipment area, auxiliary building elevation 84'	1.10E-06
	Turbine room/hall/building	6.40E-07
	Service water intake	4.20E-07
	Reactor plant auxiliary equipment area, auxiliary building elevation 100'	2.90E-07
	Service water duct, manhole	2.10E-07
	CO_2 equipment room	6.00E-08
	Reactor plant auxiliary equipment area, auxiliary building elevation 45'	9.40E-09
	Reactor plant auxiliary equipment area, auxiliary building elevation 45'	9.30E-09
	Mechanical penetration area	7.30E-09
	Reactor plant auxiliary equipment area, auxiliary building elevation 64'	5.10E-09
	Hallway of the auxiliary building elevation 100'	1.50E-09
San Onofre 2	Switchgear room	3.30E-06
	Switchgear room	2.90E-06

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Turbine room/hall/building	2.20E-06
	Electrical penetration, 2-PE-63-3B	1.70E-06
	Electrical penetration, 3-PE-63-3B	1.60E-06
	Electrical penetration, 2-PE-45-3A	1.00E-06
	Switchgear room, 2-AC-85-71	9.40E-07
	Diesel generator room, 2-DG-30-155	9.30E-07
	Diesel generator room, 2-DG-30-158	9.30E-07
	Electrical equipment room/auxiliary relay room	9.20E-07
San Onofre 3	Not analyzed, see Unit 2	
Seabrook	Main control room/control room	4.30E-06
	Primary auxiliary building	3.20E-06
	Turbine room/hall/building	1.60E-06
	Switchgear room	1.40E-06
	Service water pumphouse	1.10E-06
	Switchgear room	8.20E-07
	Electrical tunnel room B	2.20E-08
	Electrical tunnel room A	5.40E-09
Sequoyah 1	Corridor	9.78E-07
	Main control room/control room	9.33E-07
	Turbine room/hall/building	6.78E-07
	Corridor	5.53E-07
	Unit 2 auxiliary instrument room	3.83E-07
	Unit 1 auxiliary instrument room	3.76E-07
	Cable spreading room (only or upper)	3.67E-07
	Electrical equipment room/auxiliary relay room	3.66E-07
	480-V board room 1B	3.58E-07
	250-V battery board room 1 & 2 and corridor	2.54E-07
	480-V board room 2B	2.50E-07
	480-V shutdown board room 1B2	1.90E-07
	480-V shutdown board room 2A2	1.77E-07
	Computer room	1.58E-07
	6.9KV shutdown board room B	1.54E-07
	Mechanical equipment room	8.21E-08
	Auxiliary control room	8.01E-08
	250-V battery room No. 1	5.69E-08
	480-V shutdown board room 1A2	4.45E-08
	Personnel and equipment access room	4.38E-08
	6.9KV shutdown board room A	1.95E-08
	480-V shutdown board room 1A1	1.07E-08
Sequoyah 2	Corridor	9.78E-07
	Main control room/control room	9.33E-07
	Turbine room/hall/building	6.78E-07
	Corridor	5.53E-07
	Unit 2 auxiliary instrument room	3.83E-07
	Unit 1 auxiliary instrument room	3.76E-07
	Cable spreading room (only or upper)	3.67E-07

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Electrical equipment room/auxiliary relay room	3.66E-07
	480-V board room 1B	3.58E-07
	250-V battery board room 1 & 2 and corridor	2.54E-07
	480-V board room 2B	2.50E-07
	480-V shutdown board room 1B2	1.90E-07
	480-V shutdown board room 2A2	1.77E-07
	Computer room	1.58E-07
	6.9KV shutdown board room B	1.54E-07
	Mechanical equipment room	8.21E-08
	Auxiliary control room	8.01E-08
	250-V battery room No. 1	5.69E-08
	480-V shutdown board room 1A2	4.45E-08
	Personnel and equipment access room	4.38E-08
	6.9KV shutdown board room A	1.95E-08
	480-V shutdown board room 1A1	1.07E-08
Shearon Harris	Main control room/control room	4.30E-06
	Switchgear room	4.00E-06
	Switchgear room	3.10E-06
	1-A-4-COMB (No other compartment identification could be found.)	1.30E-06
South Texas 1	Individual contributors are not identified in submittal	
South Texas 2	Not analyzed, see Unit 1	
Saint Lucie 1	Main control room/control room	7.49E-05
	Cable spreading room (only or upper)	6.96E-05
	Switchgear room	4.30E-05
Saint Lucie 2	Main control room/control room	5.90E-05
	Cable spreading room	5.64E-05
	FA-I21/32/33/51W	2.67E-05
	FA-O	1.34E-05
	Switchgear room	4.48E-06
Surry 1	Switchgear room	4.18E-06
	Switchgear room	1.93E-06
	Turbine room/hall/building	3.30E-07
	Cable spreading room	2.75E-07
	Electrical equipment room/auxiliary relay room	2.42E-07
	Main control room/control room	8.23E-08
	Cable vault or cable tunnel	8.18E-08
Surry 2	Switchgear room	4.18E-06
	Switchgear room	1.93E-06
	Turbine room/hall/building	3.30E-07
	Cable spreading room	2.75E-07
	Electrical equipment room/auxiliary relay room	2.42E-07
	Main control room/control room	8.23E-08
	Cable vault or cable tunnel	8.18E-08
Susquehanna 1	Sum of 15 areas (cable chases in control structure, plus two compartments in reactor building)	2.6E-08
	Main control room/control room	4.1E-09

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Cable spreading room (lower)	2.6E-09
	1-2B Unit 1 reactor building	1.7E-09
	Battery charger rooms	1.0E-09
	Cable spreading room (upper)	2.8E-10
Susquehanna 2	Not analyzed, see Unit 1	
Three Mile Island	West inverter room	5.81E-06
	Switchgear room	4.96E-06
	East inverter room	4.94E-06
	Switchgear room	3.94E-06
	Main control room/control room	3.12E-06
	Electrical equipment room/auxiliary relay room	2.59E-07
Turkey Point 3	Main control room/control room	1.90E-04
	Cable spreading room	2.80E-06
	Reactor control rod equipment room, Unit 3	2.70E-06
	Unit 3 intake cooling water structure	7.80E-07
Turkey Point 4	For most areas, refer to corresponding Unit 3 fire area	
	Reactor control rod equipment room, Unit 4	2.70E-06
	Unit 4 intake cooling water structure	7.80E-07
Vermont Yankee	Cable vault or cable tunnel	1.50E-05
	Switchgear room	9.00E-06
	Switchgear room	7.00E-06
	Main control room/control room	5.70E-06
	Reactor building., elevation 252', zone RB3 (north)	5.10E-06
	Reactor building., elevation 252', zone RB4 (south)	3.30E-06
	Cable vault battery room, elevation 262'	3.20E-06
	Reactor building, elevation 252', separation zone Div. S1 & S2 trays	1.30E-06
	Turbine room/hall/building	1.10E-06
	Reactor building., torus room, elevation 213', zone RB2 (south)	7.40E-07
	Reactor building, elevation 280', zone RB5 (north)	7.30E-07
	Reactor building, elevation 303'	4.90E-07
	Emergency diesel generator room B	4.60E-07
	Emergency diesel generator room A	4.50E-07
	Relay and metering house, 345kv switchyard	4.00e-07
	Reactor building., elevation 280', zone RB6 (south)	3.50E-07
	Reactor building., elevation 280', recirculation MG set fire	3.40E-07
	Intake structure, service water pump room fire	3.10E-07
	Startup transformer fire/propagation to turbine building	2.80E-07
	Advanced off gas building fire	1.40E-07
	Reactor building., torus room, elevation 213', zone RB1 (north)	1.30E-07
	Main/auxiliary transformer fire w/propagation to turbine building	6.80E-08
	Reactor building, lower RCIC corner room, elevation 213' at NW corner	6.70E-08
	Radwaste corridor fire	5.20E-08
	Reactor building., upper RCIC corner room, elevation 232' at NW corner	4.50E-08
	Reactor building, elevation 318'	1.90E-08
	EDG fuel oil storage tank and transfer pump house fire	1.20E-08
	Reactor building, southeast ECCS corner room, elevations 213' & 232'	1.00E-08

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Reactor building, HPCI room, elevation 213'	9.00E-09
	Reactor building, northeast ECCS corner room, elevations 213' & 232'	3.80E-09
	Intake structure, circulating water pump room fire	1.60E-09
	Reactor building., elevation 345'	1.50E-09
	Discharge structure fire	9.40E-10
Summer	Main control room/control room	3.44E-05
	Switchgear room	2.44E-05
	Electrical equipment room/auxiliary relay room	1.28E-05
	Intermediate building, IB-21.1	9.10E-06
	Service water pumphouse, SWPH-1	8.15E-06
	Turbine room/hall/building	7.09E-06
	Service water pumphouse, SWPH-3	5.96E-06
	Service water pumphouse, SWPH-5.1/5.2	5.14E-06
	Switchgear room	2.75E-06
	Intermediate building, IB-22.1	1.91E-06
	Cable spreading room	1.08E-06
	Intermediate building, IB-25.1.2 [412' elevation general area]	8.49E-07
	A diesel generator room, DG-1.1/1.2	6.98E-07
	B diesel generator room, DG-2.1/2.2	6.19E-07
	Intermediate building, IB-25.1.3 [412' elevation general area]	5.82E-07
	Control building, CB-1.1, 412' elevation	3.87E-07
	Intermediate building, IB-25.1.5 [412' elevation general area]	3.43E-07
	Auxiliary building, AB-1.29 [auxiliary building switchgear room]	2.15E-07
	HVAC chilled water pump rooms, IB-7.2	4.34E-08
	Intermediate building, IB-3 [battery charger room A]	1.79E-08
	Control building, CB-1.2 , 425' elevation	9.33E-09
	Intermediate building, IB-25.2 [turbine drive efw pump room]	3.47E-09
	Intermediate building, IB-23 HVAC water chiller equipment room A]	2.64E-09
	Intermediate building, IB-9 [HVAC water chiller equipment room B]	2.62E-09
	Auxiliary building, AB-1.7 [charging pump room A]	2.62E-09
	Auxiliary building, AB-1.5 [charging pump room B]	2.35E-09
Vogtle 1	Additional detailed analysis fire risk contribution (additional information not provided)	2.19E-06
	Fire risk contribution of scenarios screened from spatial interactions quantitative screening (additional information not provided)	1.78E-06
	Main control room/control room	1.61E-06
	Switchgear room	1.21E-06
	Switchgear room	7.14E-07
	Lower cable spreading room - train A	5.59E-07
	Train B electrical penetration area	5.51E-07
	Level A east-west corridor and cable chase	5.29E-07
	Train A electrical mezzanine	3.73E-07
	Upper cable spreading room - train B	3.46E-07
	Train b electrical raceway room	2.29E-07
	Electrical equipment room/auxiliary relay room	4.59E-09
Vogtle 2	Not analyzed, see Unit 1	

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
Waterford 3	Main control room/control room	2.00E-06
	Essential chillers room (H&V mechanical room)	1.90E-06
	Switchgear room (switchgear "A" room, switchgear "B" room, & switchgear "AB" room)	1.50E-06
	Emergency diesel generator B	5.90E-07
	Electrical penetration area A	4.30E-07
	Turbine room/hall/building	3.30E-07
	Electrical equipment room/auxiliary relay room	1.60E-07
	Cable spreading room	9.90E-08
	Reactor auxiliary building, -4' corridor and passageways	5.50E-08
	Reactor auxiliary building, 35' and 4' general areas	2.00E-08
Watts Bar 1	Auxiliary control instrument room 2A	8.82E-07
	125V vital battery room IV	7.94E-07
	Main control room/control room	7.01E-07
	Electric driven fire pump room B	6.88E-07
	Decontamination room	5.52E-07
	Corridor	5.31E-07
	ERCW pump room B	5.28E-07
	6.9kV and 480V shutdown board room A	5.22E-07
	480V board room 1B	5.13E-07
	ERCW pump room A	5.05E-07
	Screen wash pump and electric driven fire pump room A	4.72E-07
	Auxiliary instrument room 1	4.39E-07
	Corridor	4.24E-07
	Auxiliary instrument room 2	4.17E-07
	480V shutdown board room 1B	4.16E-07
	480V board room 2B	4.11E-07
	125V vital battery room III	4.04E-07
	480V electric board room	4.03E-07
	480V board room 2A	3.94E-07
	6.9kV and 480V shutdown board room B	3.78E-07
	125V vital battery I	3.68E-07
	480V board room 1A	3.15E-07
	480V transformer room 2B	2.93E-07
	ERCW strainer room A	2.60E-07
	ERCW strainer room B	2.49E-07
	Corridor	2.44E-07
	125V vital battery board I room	2.38E-07
	480V transformer room 1A	1.88E-07
	Mechanical equipment room	1.86E-07
	Mechanical equipment room	1.83E-07
	480V transformer room 2A	1.81E-07
	125V vital battery board room III	1.74E-07
	480V shutdown board room 2A	1.66E-07
	Diesel generator 1A-A room	1.64E-07
	Diesel generator 2A-A room	1.64E-07

Table 3.3: Significant fire area CDFs (Continued)

Plant	Significant fire areas	CDF
	Plant computer room	1.64E-07
	Diesel generator 1B-B room	1.52E-07
	Diesel generator 2B-B room	1.52E-07
	125V vital battery board room IV	1.31E-07
	125V vital battery II	1.25E-07
	Cable spreading room	1.25E-07
	24V and 48V battery board and charger room	1.20E-07
	250V battery board room 2	1.15E-07
	250V battery board room 1	1.11E-07
	Mechanical equipment room	1.06E-07
	Turbine room/hall/building	8.60E-08
	Electrical equipment room/auxiliary relay room	2.62E-08
Wolf Creek	Switchgear room (train A ESF switchgear room, north (3301))	2.58E-06
	Switchgear room (train B ESF switchgear room, south (3302))	2.12E-06
	Main control room/control room	1.43E-06
	Electrical penetration room, north (room 1410)	5.36E-07
	Auxiliary building, elevation 2000', general area (rooms 1301, 1313, 1314, 1315, 1318, 1320 and 1321)	3.43E-07
	Electrical equipment room/auxiliary relay room	2.40E-07
	Auxiliary building, elevation 2026', general area (rooms 1401, 1402, 1406 and 1408)	2.08E-07
	Reactor trip switchgear room (1403)	1.95E-07
	Electrical penetration room, south (room 1409)	1.80E-07
	Cable spreading room (lower cable spreading room CDF = 3.9E-08; upper cable spreading room CDF = 3.7E-08)	7.60E-08

* Formerly known as Washington Nuclear Project Number 2.

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities

Plant	Statement on vulnerabilities	Plant definition of vulnerability
Arkansas Nuclear Unit 1	No vulnerabilities were identified. [IPEEE, page 4-2]	The core damage frequency results and the failures associated with the unscreened zones do not represent any vulnerabilities for ANO based on the following: 1) All zones have a CDF that is well below the 1.0E-04 criteria set by the NRC's safety goal policy (NEI 91-04, Severe Accident Issue Closure Guidelines), 2) No new or unusual failures were identified, 3) Per the guidance of NEI's Severe Accident Closure Guidelines (see page 17), no further modifications nor procedure enhancements are required. [IPEEE, pages 4-3]
Arkansas Nuclear Unit 2	No vulnerabilities were identified. [IPEEE, page 4-2]	see Unit 1
Beaver Valley Unit 1	No specific vulnerabilities were identified with respect to external events. [IPEEE, page 1-3]	Duquesne Light Company's (DLC's) approach to identifying vulnerabilities was as follows. If the overall CDF and early release frequencies were consistent with study results from other similar plants that have been accepted by the U.S. NRC, then there are no vulnerabilities requiring enhancement. However, the most important contributors to risk at the plant should still be evaluated to see if cost-effective improvements can be made. [IPEEE, page 7-1]
Beaver Valley Unit 2	See Unit 1	See Unit 1
Big Rock Point	n/a	n/a
Braidwood Unit 1	No vulnerabilities were identified. [IPEEE, pages 8-1 through 8-3]	A definition is implied by the following statement from the IPEEE: "Each individual fire compartment has a Core Damage Frequency (CDF) of less than 1.0E-06 per reactor year. FIVE states that fire compartments with a CDF of less than 1.0E-06 are not risk significant. Therefore, no fire vulnerabilities exist for Braidwood Station." [IPEEE, page 8-1]
Braidwood Unit 2	See Unit 1	See Unit 1
Browns Ferry Unit 1	See Unit 2	See Unit 2
Browns Ferry Unit 2	No vulnerabilities were identified. [IPEEE, page 1-2]	A definition is implied by the following statement from the IPEEE: "There are no potential vulnerabilities to internal fires, high winds, external floods, or facilities/transportation accidents identified which reduce the plant's safety margins." [IPEEE, page 7-1]

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities (Continued)

Plant	Statement on vulnerabilities	Plant definition of vulnerability
Browns Ferry Unit 3	No fire vulnerabilities were identified. [IPEEE, page 7-6]	No definition of vulnerability was provided. However, one may be implied by the following statement: "In keeping with the requirements of Supplement 4 to Generic Letter 88-20 (NUREG-1407) and the guidance provided by the EPRI FIVE documentation, this evaluation has confirmed that there are no fire-induced vulnerabilities associated with the continued operation of Browns Ferry Unit 3." [IPEEE, page 7-1]
Brunswick Unit 1	See Brunswick Unit 2	
Brunswick Unit 2	The licensee found no significant vulnerabilities.	The licensee states: "Rather than attempt to define vulnerabilities, CP&L uses the criteria in NEI 91-04 'Severe Accident Issue Closure Guidelines' for deciding on the appropriate resolution for each significant accident sequence." [IPEEE, page 1-8] "Core damage sequences were grouped on the basis of location (e.g., fire area/compartment) and the group frequency compared to the closure guidelines which are provided in Tables 1 and 2 of the NEI document. These tables provide guidance as to the nature of appropriate action, ranging from effective hardware fixes (if the CDF of the group is greater than 1.0E-04 or >50% of the total CDF), to no action required (if the group CDF is less than 1.0E-06)." [IPEEE, page 2-4]
Byron Unit 1	No vulnerabilities that warrant modifications were identified. [IPEEE, pages 1-3 to 6]	A definition is implied by the following statement from the IPEEE: "Each individual fire compartment has a Core Damage Frequency (CDF) of less than 1.0E-06 per reactor year. FIVE states that fire compartments with a CDF of less than 1.0E-06 are not risk significant. Therefore, no fire vulnerabilities exist for Byron Station." [IPEEE, page 1-3]
Byron Unit 2	See Byron Unit 1	
Callaway	No vulnerabilities were identified. [IPEEE, page 1-10]	Callaway used the NEI 91-04 severe accident closure guidelines to evaluate the need for plant improvements. Each fire area or compartment above the FIVE screening threshold (1.0E-06/year) was compared to the NEI closure guidelines listed in Table 1 of NEI 91-04. [IPEEE, page 1-8]
Calvert Cliffs Unit 1	No vulnerabilities were identified. [IPEEE, page 8-1]	A specific definition of vulnerability was not provided in the IPEEE.

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities (Continued)

Plant	Statement on vulnerabilities	Plant definition of vulnerability
Calvert Cliffs Unit 2	See Calvert Cliffs Unit 1	
Catawba Unit 1	The basic finding of the evaluations in the IPEEE was that there are no fundamental weaknesses or vulnerabilities with regard to severe accident risk at Catawba Nuclear Station. [IPEEE, page 1-4]	The IPEEE states: "The major findings from this examination are that there are no unduly significant sequences (vulnerabilities) from external events." [IPEEE, page 1-3]
Catawba Unit 2	See Catawba Unit 1	
Clinton	No vulnerabilities were identified.	An explicit definition of vulnerability was not given.
Columbia Generating*	Vulnerabilities are not discussed in the IPEEE.	An explicit definition of vulnerability is not provided.
Comanche Peak Unit 1	No vulnerabilities were identified. [IPEEE, page 1-3]	Licensee states: "the relatively low core damage frequency and its uniform distribution among various contributors demonstrate that no plant-specific vulnerability to severe accidents exists at CPSES from fires." [IPEEE, page 1-4, 8-2]
Comanche Peak Unit 2	See Comanche Peak Unit 1	
Cook Unit 1	No vulnerabilities were identified.	An explicit definition of vulnerability was not given.
Cook Unit 2	See Unit 1	
Cooper	No significant vulnerabilities were discovered during the CNS IPEEE evaluation. [IPEEE Introductory Letter]	An explicit definition of vulnerability was not given.
Crystal River Unit 3	No plant vulnerabilities were identified.	An explicit definition of vulnerability was not given.
Davis-Besse	No vulnerabilities were identified. [IPEEE, page 1-2]	No explicit definition of vulnerability was provided.
Diablo Canyon Unit 1	No vulnerabilities were identified and there were no cost-effective design changes identified that could significantly reduce overall plant risk. [IPEEE, page 1-4]	"A vulnerability refers to any component, system, operator action, or accident sequence that contributes more than 50 percent to the CDF or has a frequency that exceeds 1E-04 per year .... Any containment bypass or large early release that exceeds 1E-05 per year is considered a containment performance vulnerability." [IPEEE, page 1-6]

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities (Continued)

Plant	Statement on vulnerabilities	Plant definition of vulnerability
Diablo Canyon Unit 2	See Diablo Canyon Unit 1	
Dresden Unit 2	No vulnerabilities were identified.	An explicit definition of vulnerability in the context of fire was not given. The revised IPEEE highlighted nine insights related to fire risk (see Insights). [IPEEE, page 1-6] A definition may be implied in the statement that there were no other external events identified that have any significant impact on the core damage frequency at Dresden. [IPEEE, page 1-7]
Dresden Unit 3	See Dresden Unit 2	
Duane Arnold	The plant did not identify any vulnerabilities. However, potential plant improvements or procedural strategies were identified as part of the IPEEE. [IPEEE, page 7-1]	An explicit definition of vulnerability was not given. However, the following is presented in the IPEEE, "In Section 7.2 a discussion of significant hazards is given. Hazards that were identified by comparison with the NUREG-1407 core damage frequency screening criterion of 1E-06/yr or by comparison to the NUREG/CR-5088 Fire Risk Scoping Study issues are included in this discussion." [IPEEE, page 7-1]
Farley Unit 1	No fire vulnerabilities were identified.	Potential vulnerabilities were dispositioned consistent with NEI 91-04, Revision 1 for the fire analyses, see ImpMatrix. [IPEEE, Introductory Letter; IPEEE, page 4-57]
Farley Unit 2	See Unit 1	
Fermi Unit 2	No vulnerabilities were identified. [IPEEE, pages 8-2, 4, 5]	A definition of vulnerability may be implied by the following statement: "The EPRI Fire-Induced Vulnerability Evaluation (FIVE) technique is used for the fire portion of the Fermi 2 IPEEE. This technique identifies fire initiators by compartment and then uses a multi-step screening process to ascertain if the probability of going to core damage is less than 1.0E-06/yr for each identified fire compartment. This screening effort includes a walkdown to verify assumptions credited in the screening process. Those compartments that do not screen out are then evaluated as potential vulnerabilities." [IPEEE, page 1-5]
FitzPatrick	No vulnerabilities were identified. [IPEEE, pages 1-7 to 1-11]	An explicit definition of vulnerability was not given.

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities (Continued)

Plant	Statement on vulnerabilities	Plant definition of vulnerability
Fort Calhoun	The submittal states that based on the analyses of external events, no event sequence has been identified which is considered to be a severe accident vulnerability. [IPEEE, page 1-5]	Fire areas that did not screen out were assessed based on the guidance provided in NEI 91-04 (*Severe Accident Issue Closure Guidelines (Rev. 1)*, NEI 91-04, Dec. 1994). NEI 91-04 provides a fire IPEEE closure approach for plants that implemented a fire PRA based on a mean CDF per fire compartment or mean containment bypass frequency per fire compartment considering both the actual CDF magnitude and relative contribution of any one compartment to total fire CDF. [IPEEE, pages 4-79 to 81]
Ginna	No major fire vulnerabilities were identified. [IPEEE, Introductory Letter]	An explicit definition of vulnerability was not given.
Grand Gulf Unit 1	No vulnerabilities with regard to Seismic, Fire, or HFO events were found. [IPEEE, page 8]	Vulnerability screening was based on application of the NEI 91-04 *Severe Accident Closure Guidelines*. None of the compartments were found to meet this test. [IPEEE, page 116]
Haddam Neck	The IPEEE cover letter states that "the major vulnerabilities associated with internal fires had already been identified and resolved as a result of the CY Fire PRA performed in 1986 and Appendix R related modifications." It goes on to state that "additional insights were gained from performing the IPEEE analysis." [IPEEE, cover letter and page 1-2].	An explicit definition of vulnerability was not given. However, the identification of vulnerabilities was apparently based on the consideration of "risk outliers" [IPEEE cover letter and pages 1-2 and 7-1]
Hatch Unit 1	The major finding from the IPEEE is that the plant has no fundamental weaknesses or vulnerabilities to severe accident risk in regard to the external events related to seismic, fire, high winds, floods, transportation and nearby facility accidents, and other external hazards. [IPEEE, page 8-1]	The term vulnerabilities, as used in the IPEEE, refers to "those components, systems, operator actions, and/or plant design configuration that contribute significantly to an unacceptably high severe accident risk." [IPEEE, page 7-2]
Hatch Unit 2	See Unit 1	

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities (Continued)

Plant	Statement on vulnerabilities	Plant definition of vulnerability
Hope Creek Unit 1	No external event accident vulnerabilities were identified. [IPEEE, page 1-13]	A vulnerability is defined as "a scenario which contributes inordinately to the HCGS core damage frequency (CDF), as compared to other plants of similar type and vintage (as available from published risk assessment results), thus representing a substantial design weakness of the plant." [IPEEE, page 2-1] The evaluation of severe accident vulnerabilities was accomplished by reference to the Severe Accident Issue Closure Guidelines NEI 91-04. Core Damage sequences were grouped by categories (e.g., fire-induced loss of core cooling) and the group frequency compared to the closure guidelines which are provided in NEI, 1994, Tables 1 and 2. [IPEEE, page 2-7]
Indian Point Unit 2	No vulnerabilities were discovered during the IPEEE but several opportunities for improvement were identified which are being incorporated or evaluated. [IPEEE, pg. 1-6]	The IPEEE states: "In this study the external event induced sequences have been categorized and evaluated in accordance with the guidelines provided in the Nuclear Energy Institute (NEI) Severe Accident Closure Guidelines (NEI 91-04)." [IPEEE, page 9-3]
Indian Point Unit 3	No vulnerabilities were identified as a result of the IPEEE. [IPEEE, pages 1-7, 8]	An explicit definition of vulnerability was not given.
Kewaunee	No major plant changes were deemed necessary based on the results of the Kewaunee IPEEE. However, some equipment outliers were identified. [IPEEE, page 7-2]	An explicit definition of vulnerability was not given.
LaSalle Unit 1	See Unit 2	

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities (Continued)

Plant	Statement on vulnerabilities	Plant definition of vulnerability
LaSalle Unit 2	The detailed review (of the NRC's Risk Methods Integration Evaluation Program (RMIEP) analysis) process conducted by Commonwealth Edison determined that no vulnerabilities regarding severe accident issues were indicated in the results of the RMIEP analysis of LaSalle County Station Unit 2. [IPEEE, Executive Summary]	A definition of vulnerability is implied in the following statement: "The RMIEP results are well within the safety goals established by the Nuclear Regulatory Commission." [IPEEE, Executive Summary]
Limerick Unit 1	No vulnerabilities were determined to exist at LGS. [IPEEE, page 2-3]	A fire vulnerability was defined as any fire compartment that is well above the compartment screening criterion of 1E-06. [IPEEE, page 7-1]
Limerick Unit 2	See Limerick, Unit 1	
Maine Yankee	The IPEEE concludes that vulnerabilities with regard to severe accident risk at Maine Yankee have been satisfactorily addressed either through installed or planned plant modifications and that the remaining risk from severe accidents is acceptably low. [IPEEE, page 1-10]	An explicit definition of vulnerability was not given.
McGuire Unit 1	The IPEEE concluded that there are no vulnerabilities to severe accident risk from external events. [IPEEE, pages 1-3 and 8-1]	An explicit definition of vulnerability was not given.
McGuire Unit 2	see Unit 1	
Millstone Unit 1	N/A	

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities (Continued)

Plant	Statement on vulnerabilities	Plant definition of vulnerability
Millstone Unit 2	The IPEEE states: "Through the evaluations performed, several plant vulnerabilities (outliers) to severe external events were identified." [IPEEE, page 1-2]. The referenced table [Table 7.1-1] identified one fire and three seismic/fire interaction items.	An explicit definition of vulnerability was not given beyond the equating of "vulnerabilities" with "outliers" in the cited quotation from Section 1 of the submittal. However, the threshold for identification of vulnerabilities appears to have been low in comparison to other licensees. For example, the cited table of plant vulnerabilities [Table 7.1-1] is entitled "Opportunities for Improvement."
Millstone Unit 3	"No major severe accident vulnerabilities requiring immediate corrective action have been identified or are outstanding." [IPE page 7]	No definition of vulnerability is given.
Monticello	No external event accident vulnerabilities were identified. [Revised IPEEE, page iv, v]	A vulnerability in the context of fire was defined as any direct-to-core-damage fire sequences as implied by the following statement: "The principal finding of the fire portion of the IPEEE is that there is no area in the plant in which a fire would lead directly to the inability to cool the core. Without additional random equipment failures unrelated to damage caused by the fire, core damage will not occur. As a result, this study concludes that there are no vulnerabilities due to fire events at the Monticello Nuclear Generating Station." [IPEEE, page v]
Nine Mile Point Unit 1	No vulnerabilities were identified.	An explicit definition of vulnerability was not provided. However, a definition may be inferred from the following statement: "The results of the IPEEE analysis suggest that operation of NMP1 poses no undue risk to the public and the containment evaluation indicated that the NMP1 containment does not have any unusual characteristics that result in poor containment performance." [IPEEE page 1-1]
Nine Mile Point Unit 2	NMP2 did not discuss vulnerabilities.	An explicit definition of vulnerability was not provided. However, a definition may be inferred from the following statement: "For the same reasons described above (i.e., relatively new plant designed to the latest conservative requirements), the detailed analysis of seismic and fire hazards found the risks to be relatively low. Core damage frequency for each hazard was assessed to be on the order of 1E-06/yr or less." [IPEEE, page 1-6]

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities (Continued)

Plant	Statement on vulnerabilities	Plant definition of vulnerability
North Anna Unit 1	The analysis results showed that North Anna is not vulnerable to non-seismic external events or fires. [IPEEE, Introductory letter]	An explicit definition of vulnerability was not provided. However, a definition was clearly implied in the following statement: "The NUMARC severe accident closure guidelines promulgated in NUMARC 91-04 have been used to evaluate plant improvements. Since a fire PRA was performed the appropriate guidelines are those presented in Table 1 of the document. (NUMARC, 1991)." [IPEEE, page 4-94]
North Anna Unit 2	See Unit 1	
Oconee Unit 1	See Unit 3	
Oconee Unit 2	See Unit 3	
Oconee Unit 3	The IPEEE concludes that there are no fundamental weaknesses or vulnerabilities with regard to severe accident risk at Oconee. [IPEEE, page 1-6; 8-1]	An explicit definition of vulnerability was not given. One may be implied in the statement that there were no plant changes identified that would significantly reduce the risk from external events. [IPEEE, page 1-3]
Oyster Creek	The IPEEE concludes that there are no vulnerabilities with regard to severe accident risk from external. [IPEEE, pages 1-6, 1-9]	The term vulnerability is defined as any core damage sequence that exceeds 1E-04 per reactor-year, or any containment bypass sequence or large early containment failure sequence that exceeds 1E-06 per reactor-year. [IPEEE, pages 1-9, 3-142]
Palisades	There were no other external events identified that have an impact on the core damage frequency at Palisades. [IPEEE, Revision 1, page 8-2]	An explicit definition of vulnerabilities was not presented. However, one was implied by the following statement: "The functional reporting requirements presented in GL 88-20 and NUREG-1407 are: 1) Functional sequences with a CDF greater than 1E-06/yr. 2) Functional sequences that contribute 5% or more to total CDF. 3) Sequences determined by Palisades to be important contributors to CDF or containment performance." [IPEEE, page 3-50, 4-72]
Palo Verde Unit 1	The IPEEE concludes that no vulnerabilities to fire initiating events exist at Palo Verde. [IPEEE, page 1-3]	A fire vulnerability was said to exist if core damage sequences were identified which were in excess of the screening criterion of 1E-06/reactor-year and which resulted in containment failure sequences that were either unique or unbounded by similar sequences contained in the internal events IPE. [IPEEE, page 7-3]
Palo Verde Unit 2	See Unit 1	

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities (Continued)

Plant	Statement on vulnerabilities	Plant definition of vulnerability
Palo Verde Unit 3	See Unit 1	
Peach Bottom Unit 2	The IPEEE concludes that no vulnerabilities to seismic, fires, high winds or floods or "others" were found to exist. [IPEEE, page 7-1]	A fire vulnerability is a fire compartment that significantly exceeds the FIVE screening criteria and provides a major accident risk. [IPEEE, page 7-1]
Peach Bottom Unit 3	See Unit 2	
Perry Unit 1	There were no vulnerabilities identified during the performance of the IPEEE. [IPEEE, page 7-2]	An explicit definition of vulnerability was not provided.
Pilgrim Unit 1	The IPEEE concludes that Pilgrim Station does not contain any significant vulnerabilities or "outliers" in the fire risk. [IPEEE, pages 7-3 and 7-4]	The IPEEE used the same definition of vulnerability as that cited in the IPE. Section 5 of the Internal Events IPE uses the following criteria to determine if any plant vulnerabilities exist: 1) Are there any new or unusual means by which core damage or containment failure occur as compared to those identified in other PRAs? 2) Do the results suggest that the Pilgrim core damage frequency would not be able to meet the NRC's safety goal for core damage? [IPEEE, page 7-1]
Point Beach Unit 1	The IPEEE concludes that no significant fire concerns were discovered in the Point Beach Nuclear Plant Fire Analysis. [IPEEE, Section 8.2, page 4]	An explicit definition of vulnerability in the fire context was not provided. One may be implied by the following statement: "Since the resulting fission product release frequency due to seismic events is judged to be <1E-06/year, it is concluded that PBNP has no severe accident vulnerabilities due to a seismic event." [IPEEE, Section 8.1, page 2] However, an equivalent statement was not found for fire or HFO events.
Point Beach Unit 2	See Unit 1	

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities (Continued)

Plant	Statement on vulnerabilities	Plant definition of vulnerability
Prairie Island Unit 1	The IPEEE concluded that there are no significant vulnerabilities to severe accidents that exist at Prairie Island that would be attributable to seismic, fire, or other external events. [IPEEE, Revision 1, page v]	An explicit definition of vulnerability was not provided. A definition may be implied in that the IPEEE fire analysis concluded that the overall core damage frequency is low. [IPEEE, Revision 1, page 3] A definition may also be implied in the screening criteria "used to identify sequences to be discussed" as follows: "The criteria are identical to the functional reporting requirements presented in GL 88-20 as required by NUREG-1407: 1) functional sequences with a CDF greater than 1E-06/year, 2) functional sequences that contribute 5% or more to total CDF, and 3) sequences determined by the utility to be important contributors to CDF or containment performance." [IPEEE, Revision 1, page B-91]
Prairie Island Unit 2	See Unit 1	
Quad Cities Unit 1	The original IPEEE submittal stated that, with regard to fire, "Five potential vulnerabilities have been identified by the IPEEE" [original IPEEE, page 7-3]. The revised analysis concluded that no fire vulnerabilities remained.	No explicit definition of vulnerability is stated. The discussion of each of the five items implies that a vulnerability was a condition that contributed to the high CDF value estimated in the original analysis. [IPEEE Section 7.2.2, page 7-3]
Quad Cities Unit 2	See Unit 1.	

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities (Continued)

Plant	Statement on vulnerabilities	Plant definition of vulnerability
River Bend Unit 1	The IPEEE concludes that there were no vulnerabilities due to internal fires. [IPEEE, page 151]	The exact definition of a vulnerability in the context of fire was not entirely clear. It appears that the overall fire results were initially compared against the vulnerability criteria established by the NEI 91-04 *Severe Accident Issue Closure Guidelines*. [see the fire IPEEE, page 1-5]. However, the IPEEE also cites that the licensee's IPE analysis had considered and rejected the NEI guidelines. It would appear that in the end, a set of criteria consistent with the IPE criteria was applied. In particular: "Instead (of the NEI criteria) the vulnerability screening criteria for River Bend is based on the NRC's Safety Goal Policy Statement. The criteria for River Bend is that if the total core damage frequency or the core damage frequency of any functional accident sequence exceeds 1.0E-04 per year, a vulnerability associated with the overall plant or sequence is assumed to exist. In addition, the contribution that exceeds the criteria must be 'real' and not an artifact of conservative modeling or analysis assumptions."
Robinson Unit 2	No vulnerabilities were identified.	The IPEEE states: "The evaluation of severe accident vulnerabilities was accomplished by reference to the Severe Accident Issue Closure Guidelines, NUMARC 91-04, Severe Accident Issue Closure Guidelines (NUMARC, 1991). Core damage sequences were grouped primarily on the basis of location (e.g., fire area/compartment) and also on the nature of the sequence (non-LOCA, LOCA, fire-induced containment bypass), and the group frequency compared to the closure guidelines which are provided in Tables 1 and 2 of the NUMARC Document. These tables provide guidance as to the nature of appropriate action, ranging from effective hardware fixes (if the CDF of the group is greater than 1.0E-04 or greater than 50% of the total CDF), to no action required (if the group CDF is less than 1.0E-06)." [IPEEE, page 2-3]

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities (Continued)

Plant	Statement on vulnerabilities	Plant definition of vulnerability
Saint Lucie Unit 1	The IPEEE concludes that "there are no vulnerabilities to severe accident risk from external events." [IPEEE Section 1.4, page 7] and that "no scenario or event sequence has been identified which is considered to be a severe accident vulnerability." [IPEEE Section 8, page 123]	No definition of vulnerability is given.
Saint Lucie Unit 2	See Unit 1.	
Salem Unit 1	The IPEEE concludes that "SGS has no significant vulnerabilities to external events." [IPEEE, page 8-6]	A vulnerability is defined as "a contribution to unusually high risk, as compared to other plants of similar type and vintage (as available from published risk assessment results), which represents a substantial design weakness of the plant" [IPEEE, page 2-1]. Furthermore, the evaluation of severe accident vulnerabilities was accomplished by reference to the NEI Severe Accident Issue Closure Guidelines NEI 91-04 (NEI, 1994). [IPEEE, page 2-4]
Salem Unit 2	See Unit 1	
San Onofre Unit 2	No fire vulnerabilities are identified.	The IPEEE states: "A vulnerability in a PWR is a plant feature which contributes a disproportionately large percentage to either core damage or significant release probabilities which are in turn significantly higher than those of an average PWR. This definition is applicable for the seismic, internal fire, and other hazards analysis" [IPEEE, page 2-3]. The submittal also intimates that a vulnerability is a condition that might expose the plant to non-conservatisms outlined in SECY-93-143. [IPEEE, page 4-121]
San Onofre Unit 3	See Unit 2.	See Unit 2.
Seabrook Unit 1	The IPEEE concludes that there are no vulnerabilities to severe accident risk from external events. [IPEEE, page 1-5]	An explicit definition of vulnerability is not provided.

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities (Continued)

Plant	Statement on vulnerabilities	Plant definition of vulnerability
Sequoyah Unit 1	No vulnerabilities were identified.	The term vulnerabilities refers to "those components, systems, operator actions, and/or plant design configurations that contribute significantly to an unacceptably high severe accident risk." [IPEEE page 1-9]
Sequoyah Unit 2	See Unit 1	
Shearon Harris Unit 1	"The IPEEE has demonstrated that the SHNPP has no significant vulnerabilities to external events." [page 8-5]	A vulnerability was defined as any core damage sequence that exceeds 1E-4/ry or any containment bypass sequence or large early containment failure sequence that exceeds 1E-6/ry.
South Texas Project Unit 1	The IPEEE does not identify and vulnerabilities.	An explicit definition of vulnerability is not provided. However, for fire, the IPEEE states that the various fire sequences considered are "not important to plant risk." [IPEEE page 9.5-1]
South Texas Project Unit 2	See Unit 1	
Summer	No vulnerabilities are identified.	An explicit definition of vulnerability is not provided. The IPEEE states: "SCE&G's IPEEE Fire Evaluation demonstrates that, in concert, the existing VCSNS Appendix R and FPER Evaluation, selected shutdown systems, Fire Emergency Procedures, and VCSNS's overall Fire Protection and Equipment Maintenance Programs are sufficient to maintain the Virgil C. Summer Nuclear Station at a negligible vulnerability to a fire initiated core damage event." [IPEEE, Fire Portion, page 3]
Surry Unit 1	No fire vulnerabilities are identified.	An explicit definition of vulnerability was not provided in the IPEEE submittal. However, use of the NEI 91-04 *Severe Accident Issue Closure Guidelines*, is implied.
Surry Unit 2	See Surry Unit 1	
Susquehanna Unit 1	No fire vulnerabilities were identified.	No explicit definition provided although a definition is implied in that the IPEEE states that "...the PRA demonstrates that defense in depth against core damage exists for any fire. That is, no fire with a single independent equipment failure results in core damage." [IPEEE page 4-74]
Susquehanna Unit 2	See Unit 1.	

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities (Continued)

Plant	Statement on vulnerabilities	Plant definition of vulnerability
Three Mile Island Unit 1	The IPEEE states that there are no vulnerabilities to severe accident risk from external events. [IPEEE, page 1-4]	The term vulnerability is defined as any core damage sequence that exceeds 1.0E-04 per reactor-year, or any containment bypass sequence or large early containment failure sequence that exceeds 1.0E-06 per reactor-year. [IPEEE, page 1-9]
Turkey Point Unit 3	The IPEEE states that there are no vulnerabilities to severe accident risk from external events. [IPEEE, pages 8, 62]	No definition of vulnerability is given.
Turkey Point Unit 4	See Unit 3.	
Vermont Yankee	No vulnerabilities were identified.	No definition is provided.
Vogtle Unit 1	The IPEEE concludes that "VEGP has no fundamental weaknesses or vulnerabilities to severe accident risk in regard to the external events related to seismic, fire, high winds, floods, transportation and nearby facility accidents, and other external hazards." [IPEEE, page 1-2]	An explicit definition of vulnerability was not provided.
Vogtle Unit 2	See Unit 1	
Waterford Unit 3	The IPEEE concludes that there are no fire vulnerabilities at Waterford. [IPEEE, page 1-4]	The lack of fire vulnerabilities was based on three points: 1) no individual fire scenario has a CDF greater than 2E-6 (i.e., less than 1E-4); (2) no individual fire scenario contributes more than 31% of the total core damage frequency due to fires; and (3) no unusual and significant failures were found. [IPEEE page 1-4].
Watts Bar Unit 1	The submittal states that "The IPEEE program did not uncover any serious fire vulnerabilities" [IPEEE, page 4].	An explicit definition of vulnerability was not provided.
Wolf Creek Unit 1	The IPEEE concludes that no event sequence has been identified which is considered to be a severe accident vulnerability. [IPEEE, pg. 8-3]	An explicit definition of vulnerability was not provided.

Table 3.4: Licensees' statements on, and definitions of, fire vulnerabilities (Continued)

Plant	Statement on vulnerabilities	Plant definition of vulnerability
Zion Unit 1	No fire vulnerabilities are identified.	An explicit definition of vulnerability was not provided. The submittal states that risk vulnerabilities are identified in accordance with the guidance provided in GL 88-20, Supplement 4, and NUREG-1407. Vulnerabilities are in the form core damage and containment failure frequency for the fire PRA. [IPEEE, page 2-3]
Zion Unit 2	See Unit 1	

* Formerly known as Washington Nuclear Project Number 2.

Table 3.5: Fire-related plant improvements

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
ANO 1&2	None cited.								
Beaver Valley 1	Cable Tunnel (CV-3) Fire CDF Key Contributor: reroute river water pump power cable.	X						X	
Beaver Valley 1	Cable spreading room (CS-1) Fire (SW corner) Key CDF Contributor: refine emergency switchgear room heat up analysis to provide additional time margin.	X						X	
Beaver Valley 1	Primary auxiliary building General Area E (PA-1E) Fire Key CDF Contributor: reroute CCR pump or high head safety injection (HHSI) suction MOV cables.	X						X	
Beaver Valley 1	Cable Spreading Room (CS-1) Fire (NE corner) Key CDF Contributor: reroute river water or auxiliary river water pump power and control cables.	X						X	
Beaver Valley 1	Normal Switchgear Room (NS-1) Fire (South Wall) Key CDF Contributor: reroute river water pump control cables or Auxiliary river water pump power cables.	X						X	
Beaver Valley 2	Control Room (CB-3) Fire Key CDF Contributor: provide operator credit for recovery of auxiliary feedwater from outside the control room.	X						X	
Beaver Valley 2	Cable Tunnel (CT-1) Fire Key CDF Contributor: install qualified fire barriers between fire areas Communication, Instrumentation, and Relay Room (CB-1), Cable Spreading Room (CB-2), and Cable Tunnel (CT-1).	X						X	
Beaver Valley 2	Normal Switchgear Room (SB-4) Fire Key CDF Contributor: install an automatic CO_2 fire suppression system.	X						X	
Beaver Valley 2	West Cable Vault Area Elevation 735' (CV-1) Fire Key CDF Contributor: reroute purple train service water pump/MOV power and control cables.	X						X	
Beaver Valley 2	West Cable Vault Area Elevation 755' (CV-3) Fire Key CDF Contributor: reroute orange train CCP/thermal barrier cooling MOV and service water power and control cables.	X						X	
Big Rock Point	Modifications performed in response to Fire Protection and Appendix R Modifications: fire doors added or replaced in the machine shop, access control to turbine building stairway and electrical equipment room. In addition, a fusible link closure device was added to the third floor hallway to the turbine building.			SRP (Section 9.5.1) & Appendix R to 10CFR50	X				
Big Rock Point	Modifications performed in response to Fire Protection and Appendix R Modifications: smoke and or fire detectors were added to the emergency diesel generator room, screenhouse, control room, electrical equipment room, condensate pump room, control rod drive accumulator area, core spray pump room, shutdown heat exchanger room, and the reactor recirculating water pump room.			SRP (Section 9.5.1) & Appendix R to 10CFR50	X				
Big Rock Point	Modifications performed in response to Fire Protection and Appendix R Modifications: fire hose reels were installed in the interior cable penetration area and the reactor cooling water pump and heat exchanger room. Sprinklers were added in the reactor recirculating water pump room. A valve hose connection manifold was added to the discharge of the diesel fire pump.			SRP (Section 9.5.1) & Appendix R to 10CFR50	X				

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Big Rock Point	Modifications performed in response to Fire Protection and Appendix R Modifications: fire barriers or fire stops were added to the following locations, as applicable. Typical openings sealed include: (1) Locations where cable trays penetrate walls, ceilings, and floors. (2) Locations where conduit or pipe provide an opening between a wall, ceiling, or floor. (3) Large openings in walls or between walls where concrete fill is not applicable or desirable. (4) Openings between vent ducts and walls, floors, or ceilings.			SRP (Section 9.5.1) & Appendix R to 10CFR50	X				
Big Rock Point	Modifications performed in response to Fire Protection and Appendix R Modifications: Electrical switchgear was modified to protect the energized equipment from the effects of fire protection sprinkler spray. Modifications included: (1) Sealing non-ventilation openings and hardware mountings, (2) Installing shields over ventilation louvers, and (3) Installing shields to totally cover transformers.			SRP (Section 9.5.1) & Appendix R to 10CFR50	X				
Big Rock Point	Modifications performed in response to Fire Protection and Appendix R Modifications: installation of self-contained battery lighting units for vital safe-shutdown areas and access ways.			SRP (Section 9.5.1) & Appendix R to 10CFR50	X				
Big Rock Point	Modifications performed in response to Fire Protection and Appendix R Modifications: smoke and heat removal unit installed above vent duct chase on roof of the electrical equipment room.			SRP (Section 9.5.1) & Appendix R to 10CFR50	X				
Big Rock Point	Modifications performed in response to Fire Protection and Appendix R Modifications: fire dampers were installed in large vent ducts where they penetrate fire barriers. A manual smoke damper was installed in control room ducts.			SRP (Section 9.5.1) & Appendix R to 10CFR50	X				
Big Rock Point	Modifications performed in response to Fire Protection and Appendix R Modifications: design and installation of the alternate shutdown system. Included is the design of the electrical system and the alternate shutdown building.			SRP (Section 9.5.1) & Appendix R to 10CFR50	X				
Big Rock Point	The fire penetration, barriers and doors were inspected during the performance of the T545-01 Procedure, Fire Door and Fire Damper Inspection: fire barriers were identified and labeled in response to this notice and also identified on applicable plant drawings. Initial inspection identified some deficiencies which have been corrected.	X			X				
Big Rock Point	Requirement that the safe shutdown circuits are physically independent of, or can be isolated from, the control room in the event of a fire affecting equipment control from the control room: circuit separation was accomplished with the installation of the alternate shutdown system (ASD). There are two methods identified for safe shutdown cooling after a fire. One method involves the reactor depressurization system (RDS) in conjunction with the core spray system (CSS). The second method involves using the Emergency Condenser Systems (ECS).	X			X				

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Braidwood 1 & 2	None cited.								
Browns Ferry 1, 2, 3	None cited.								
Brunswick 1&2	None cited.								
Byron 1&2	None cited								
Callaway	The main control room has an overall core damage frequency of 2.65E-06/ry: NEI closure guidelines recommend that severe accident management guidelines (SAMG) be put in place with emphasis on prevention/mitigation of core damage. (Note: The NEI recommendation is not needed. The NRC found the design and procedures associated with the control room to be acceptable during an extensive NRC review prior to Callaway's receipt of its Operating License. Since that time, no substantive changes have been made to the control room design or procedures.)	X						X	
Callaway	The main control room has an overall core damage frequency of 2.65E-06/ry: the SAMG effort at Callaway will develop responses to spurious actuations which can result from a control room fire. Emphasis will be on those actions that need to be accomplished outside the control room to mitigate spurious actuations caused by a fire confined to a cabinet (e.g., loss of seal injection due to a hot short in the control circuitry associated with the seal injection line containment isolation valves). These actions outside the control room will permit the control room operators to safe shut down the plant from the control room.	X					X		
Callaway	For two safety-related ac switchgear rooms with CDFs of 2.26E-06/yr and 1.29E-06/yr, NEI closure guidelines recommend that SAMG be put in place. (Note: There are no obvious recovery actions that will reduce core damage frequency. The fire areas are equipped with area-wide suppression which is effective. These is also a great deal of uncertainty in the fire ignition frequencies and the extent of damage to key components. In addition, the overall SAMG effort will focus on recovery of failed equipment. Emphasis will be on those actions that need to be accomplished outside the control room. The overall Callaway SAMG effort is sufficient to reduce the impact of a fire in either of these fire areas.)	X					X		
Calvert Cliffs	Smoke infiltration into the main control room via ventilation intake: applicable procedures will be revised to direct the MCR operator to place the MCR and cable spreading rooms (CSRs) into recirculation if it appears likely that smoke could be drawn into the MCR ventilation intake. Operator training was also initiated.	X							X

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Calvert Cliffs	Inadvertent isolation of the switchgear room and CSR ventilation: improvement in procedural direction and training for the operators was initiated. The procedures will direct that the cable spreading rooms and switchgear ventilation systems be restored following an inadvertent actuation. Measures were also being taken to place these ventilation systems in recirculation for fires outside the cable spreading room and switchgear rooms if smoke from a fire external to them has potential for getting into the ventilation intakes. In addition, the switchgear room ventilation systems were to be evaluated for a method to ensure effective recovery from inadvertent actuation of Halon. A realistic switchgear room heat-up model was to be developed and was expected to show that there is considerable time available to recover the loss of ventilation. An evaluation based on the new heat-up model will determine if modifications or procedure changes are required to: (1) either prevent the loss of ventilation or (2) restore ventilation.	X							X
Calvert Cliffs	Fire barriers and components, such as fire dampers, fire penetration seals and fire doors, were not included in the plant surveillance and maintenance program: incorporate them into an appropriate control and/or inspection program.	X							X
Calvert Cliffs	Hot work in cable chases at power: procedure change has been initiated adding restrictions to hot work in cable chases.	X							X
Catawba 1 & 2	A fire in a Diesel generator load sequencer could cause load shed of a 4160 volt bus: a procedure enhancement has been made by placing additional instructions in the pre-fire plant for the ETB switchgear area.	X			X				
Catawba 1 & 2	Sufficient redundancy for fires: replace reciprocal air compressors with centrifugal compressors and the cables for the newly installed instrument air compressors are to be routed.	X							X
Catawba 1 & 2	Auxiliary shutdown panel NEMA 4 cabinets are missing door bolts: reinstall bolts.	X			X				
Clinton	Response to generic letter 92-08, "Thermo-Lag 330-1 Fire Barriers": cables routed from Division 2' inverter through the Division 1 cable spreading room and then through the Division 3 switchgear room have been rerouted.			Generic Letter 92-08 (Fire)					X
Columbia Generating*	The recovery of the critical ac buses SM-7 and SM-8 was shown to be significant in reducing fire induced CDF: this recovery action is proceduralized and it has been recommended that specific training scenarios be included in the operator training.	X				X			
Comanche Peak 1&2	None cited.								
Cook 1&2	None cited.								
Cooper	Fires in board C and vertical board F in the MCR were identified as completely disabling the control of switchyard breakers: licensee indicated that they were examining additional (unspecified) features that would allow for control of the switchyard breakers either from the switchyard itself or from an alternate area, or have a preplanned recovery/repair action in place.	X						X	

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Cooper	A fire in the SW pump house could disable all of the SW pumps and one of the motor-driven fire water pumps, especially if the fire suppression system fails: licensee is examining the feasibility of providing the SW system with water supplies that are diverse from the pumps in the SW pump room.	X						X	
Crystal River 3	Two transient fire storage areas, sources were in a specific location and were significant contributors to total core damage risk due to fire: place administrative limits on.	X				X			
Davis-Besse	Inadequate mounting of two cylinders containing compressed flammable gas were identified: licensee will take actions to address this.	X				X			
Davis-Besse	The licensee noted that four plant compartments had calculated bounding core damage frequency values above the screening criterion of 1E-6/ry after completion of the fire analysis: The Severe Accident Closure Guidelines were reviewed to ascertain the relative importance of these estimations. The Closure Guidelines indicate that for fire compartments that fall in this CDF range (1E-06 to 1E-05), the licensee should ensure that severe accident management guidelines will be in place with emphasis on prevention/mitigation of core damage or vessel failure, and containment failures. Licensee will review the fire response procedures associated with these areas to ensure that specified actions are optimized with respect to maintaining overall plant risk as low as reasonably achievable.	X					X		
Diablo Canyon	Control room fire located in cabinets that could result in loss of CCW or auxiliary saltwater (ASW) systems: modify control room evacuation procedure to require the reactor coolant pumps to be tripped.	X					X		
Diablo Canyon	Diesel generator 1-3 could only support the vital ac F bus of one unit if needed during a plant transient: add a sixth emergency diesel generator which allows each vital ac bus to be supported and increases the availability of backup power for vital ac bus F. Installation of the sixth diesel is calculated to have reduced the contribution of loss of offsite power events to the overall core damage frequency and to have reduced the likelihood of ASW or CCW system failures leading to a loss of RCP seal cooling.			X	X				
Diablo Canyon	A single failure of the motor-operated discharge damper could have failed the 480V switchgear ventilation system: make a design change.			X	X				
Diablo Canyon	Ensure the RCP seal cooling is maintained to prevent RCP seal LOCAs: revise operating procedure.			X	X				
Diablo Canyon	Improve the reliability and availability of the plant process protection system: upgrade Eagle 21 Process Protection System and eliminate resistance temperature detector (RTD) bypass to reduce plant downtime and radiation exposures to plant personnel.			X	X				
Diablo Canyon	Instrument inverter fails: replace instrument invertors with invertors of increased capacity, increase reliability by including automatic backup switching (static switch).			X	X				
Dresden 2&3	None identified.								

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Duane Arnold	Two air handlers in the HPCI room were identified as flood/spray outliers because nearby piping could potentially impact fire protection sprinkler piping and break off the sprinkler heads, spray from the sprinkler heads could damage the air handlers' motors: evaluate further. (Calculation determined that it is adequate as-is.)		X					X	
Duane Arnold	Unrestrained nitrogen gas bottles near MCC could fall over and cause missiles or fires: secure bottles.		X		X				
Duane Arnold	Gas bottles stored adjacent to chiller could fall over during a seismic event and initiate a fire or missile: remove bottles from area.		X		X				
Duane Arnold	Unavailability of one train of the river water system due to maintenance in the essential switchgear rooms: optimization of the river water system maintenance outage time and staging or readying of fire hoses.	X							X
Duane Arnold	A rupture of fire protection pipe (2" or 4" in diameter) located in the HVAC room above the control room could cause flooding of the HVAC shaft which could cause subsequent collapse of the ductwork in the shaft and establish a substantial flow rate into the essential switchgear rooms via the HVAC ductwork leading to direct failure of the key electrical equipment controlling safe shutdown or flooding of the control building basement rooms, and subsequent failure of the key electrical equipment: modify piping design to eliminate the flooding sequences by converting the two fire protection pipes in the HVAC room into "dry pipe systems each of which have a manually controlled valve located outside the HVAC room for control of water to each pipe system."	X			X				
Duane Arnold	Cables for Division II equipment (required for the remote shutdown of the plant) pass through cable spreading room: reroute cables.	X			X				
Farley	The RCP seals require cooling from either component cooling water (CCW) to the RCP thermal barrier heat exchanger, or seal injection flow from the charging pumps. Loss of both of these sources for a prolonged period is expected to cause RCP seal failure resulting in a LOCA. Loss of the operating (on-service) train of CCW, which can be caused by failures within CCW or support systems, such as the SW system, or the electrical distribution system, results in loss of RCP seal cooling almost immediately. It also results in loss of cooling to the running charging pump, which will cause charging pump failure and, thus, loss of seal injection flow in a relatively short time: implement procedure enhancements to improve operator response capability for fire events which can lead to a loss of RCP seal cooling. Install high temperature O-rings for Units 1 and 2 RCPs during maintenance overhauls. Expected to provide a substantial reduction in CDF to fire scenarios involving electrical switchgear rooms or electrical penetration rooms.	X				X			

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Farley	If cooling from the on-service CCW train fails, but the standby train is available, the operators can establish seal cooling by manually starting the standby CCW train and manually aligning the miscellaneous CCW header (which provides RCP thermal barrier heat exchanger cooling) to that train. This action was estimated to require at least 20 minutes, which was judged to be too long to ensure RCP seal integrity without injection. However, RCP seal injection can be maintained for a prolonged period by starting the standby charging train and aligning charging pump suction to the RWST to maintain a cool water source. This requires operation of the normally operating charging pump, without cooling for a short time, while the alternate alignment is made: revise appropriate abnormal operating procedure (AOP) to include instructions directing the FNP operators to perform one of several sequences of steps, depending on available equipment, to maintain RCP seal cooling in the event of fires in the electrical switchgear. The pertinent steps include aligning charging pump suction to the RWST, and isolating RCP seal return flow to minimize heat-up of the injection flow due to the addition of pump heat in the charging pump mini-flow line.	X				X			
Farley	CCW to the RCP thermal barrier heat exchangers is provided from the miscellaneous header, which is aligned to the on-service CCW train A (supported by train-dedicated CCW pump C and swing CCW pump B). If the on-service CCW pump A fails due to loss of train A SW support and the standby train CCW pump A fails, it is possible to restore CCW cooling to the standby train B charging pump by realigning swing CCW pump B to discharge through the train B CCW heat exchanger. Realignment of the swing CCW pump B from train A to train B normally requires that both the electric+A352cal power alignment and the mechanical alignment be changed. The total time to complete this action was estimated to be at least 20 minutes, which was judged to be too long to ensure RCP seal integrity without injection or thermal barrier cooling. However, with both trains of electrical power available, it is possible to restore CCW flow to the train B charging pump by allowing the swing pump to be powered by train A while it is mechanically align to the train B CCW heat exchanger: revise appropriately AOP, as necessary, including the addition of a caution statement for response to fire in the SWIS to inform the operators that electrical realignment of the swing CCW pump may be delayed if required by plant conditions. The pertinent steps include aligning the discharge flow from swing CCW pump B to the train B CCW heat exchanger.	X				X			
Farley	Fires in the electrical penetration rooms can result in inadvertent movement of the normally open, motor-operated SW supply valve to the CCW heat exchanger. This can result in loss of RCP thermal barrier cooling if the SW supply to the on-service CCW heat exchanger inadvertently closes: add steps to the appropriate AOP to instruct operators to locally verify that the SW supply valve to the on-service CCW heat exchanger is open in the event of a fire in the electrical penetration room associated with the on-service train.	X				X			

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Farley	Room cooling is required for continued operation of the motor driven AFW (MDAFW) pumps following an initiating event. Room cooling for the MDAFW pumps is supplied by train-oriented room coolers, which require the support of the SW system and the 600V emergency ac distribution system: add steps to appropriate AOP to direct operators to initiate action to monitor pump room temperatures and establish temporary ventilation if MDAFW pump operation is required following a fire in the SWIS or auxiliary building fire area.	X				X			
Fermi	An unscreened compartment leads to the fire insight that dominating contributors are cabinets used for dedicated shutdown whose loss would isolate the affected equipment from the main control room thereby causing loss of equipment function: even though adequately covered by current operator training, additional fire brigade drills in the vicinity of these cabinets are planned to increase the brigade's awareness of the need to quickly isolate and extinguish such fires.	X				X			
FitzPatrick	Reduce the contribution of reactor building fires to the CDF: addition of keylock bypass switches to allow opening of valves 10MOV-25A/B and 14MOV-12A/B. Also, plant fire procedure AOP-28 directs the operators to use the switches if necessary and includes a tabulation of potentially unavailable equipment in each fire zone.	X			X				
FitzPatrick	Fire-induced core damage resulting from fires in the cable spreading room: evaluate whether the heat detectors which automatically initiate CO_2 suppression in the cable spreading room need to be relocated from the bottom of the ceiling beams (approximately 2 ft below the ceiling) to the ceiling, placing detectors on both sides of ceiling beams. This action could potentially achieve a 66% reduction in the dominant contributor of fire-induced core damage resulting from fires in the cable spreading room.	X					X		
FitzPatrick	Contribution of fires attributable to transient combustibles: revised AP14.02, "Combustible and Flammable Material Control," to impose strict limitations on the use of unattended combustible materials in the Cable Spreading Room. (Details requirements for the use and storage of combustible and flammable materials within the power block and applicable adjacent areas. Under the provisions of this procedure, transient combustibles in the cable spreading room require the approval of a qualified individuals on the fire protection staff.)	X			X				
Fort Calhoun	Hydrogen piping, fuel oil, seal oil tank, & flammable storage area in turbine building, CDF of a 0.1g event causing turbine building failure due to hydrogen fire/explosion was determined to be 6.87E-08/ry. (Low CDF, no further evaluation done.)		X					X	
Fort Calhoun	Flammable liquid cabinets are located throughout the plant: move cabinets out of critical area or determine if cabinets can be anchored.		X				X		
Fort Calhoun	A control room fire occurs that does not require control room evacuation: revise AOP-6 to instruct operators to de-energize the PORVs.	X				X			
Fort Calhoun	Control room fire initiates an interfacing LOCA system: if feasible, remove power from HCV-347 and HCV-348.	X					X		

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Ginna	Fuses will be installed on control circuits routed in the screen house associated with the functioning of 4160V ac circuit breakers: the fuses will be designed to open if grounding occurs, as is postulated to occur for screen house fires, permitting the over current protective function of the circuit breakers to remain intact.	X			X				
Ginna	Perform local recovery of the pressurizer heaters if control of the heaters is lost from the control room (the pressurizer heaters are one means of providing long-term RCS circulation): enhancement to an operating procedure.	X						X	
Ginna	A spurious opening of MOV 857B fails RHR closed cycle cooling: insertion of a warning in the alternate shutdown procedure ER-FIRE.1 was being considered to indicate that this valve may need to be closed locally.	X						X	
Ginna	Transient combustibles storage in the auxiliary building basement: installation of additional sealed containers for combustibles storage was being considered.	X						X	
Ginna	Spurious opening of MOVs 850A and 850B due to hot shorts can lead to draining of the RWST volume into the containment sump: methods to reduce this potential were being considered in combination with a similar modification of MOV 857B.	X						X	
Ginna	Assist operators in switching to sump recirculation, particularly for fire scenarios in which the Control Room is evacuated: installation of a local pressure gauge to permit RWST level measurements to be obtained in the event that electrical RWST level sensors are damaged by fire.	X						X	
Ginna	House heating boiler was found to be inadequately anchored and thus could shift during an earthquake causing damage to the attached natural gas line: implement design change to anchor the boiler.		X			X			
Ginna	Several locations were identified where block wall failures could result in the release of combustibles - an oxygen line in the auxiliary building, a hydrogen line and valve station in the intermediate building, and hydrogen cylinders in the turbine building: the oxygen line does not pose a significant risk because it is only connected after an accident, the hydrogen line is not a risk since it is not valved on during power operation. The hydrogen cylinders do not pose a risk since the hydrogen is diluted with nitrogen and any release would thus result in a low hydrogen concentration.		X					X	
Ginna	Failure of block walls potentially causing the actuation of two fire suppression systems: the inadvertent actuation of a deluge system in the relay room would not have a significant impact since the relay cabinets are closed on top and the cable penetrations are sealed. Actuation of a deluge system in the intermediate building would only spray the turbine-driven AFW pump. Seismic actuation of a pre-action system in the same area was dismissed because of the existence of fusible link sprinkler heads in the system.		X					X	
Ginna	Failure of block walls between the service and intermediate buildings and between the turbine and intermediate buildings during an earthquake could impact the fire protection of safety-related equipment: licensee says the potential for fires initiated in these areas is small.		X					X	

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Ginna	PSA identified a fire scenario in the DG B vault, located beneath the DG B room, in which both trains of ac electric power could be affected. Basically, a worst-case fire could fail the B electrical train and fail offsite power and all control power to Bus 18 of the A electrical train (DG A would still remain available). This, in turn, would result in the loss of all SW: ACTION report 99-948 was generated to evaluate the scenario. The result of the ACTION report was to recommend consideration of procedural changes to instruct plant personnel to manually close the required Bus 18 breakers to prevent leaving the plant in a station blackout condition.			Fire, PSA			X		
Grand Gulf	None identified.								
Hatch	Regulatory issues associated with the use of Thermo-Lag fire retardant: cable rerouting modification.			X					X
Haddam Neck	Relatively high probability of diesel B unavailability: develop procedure for connecting air cooled DG.	X				X			
Haddam Neck	Potential coincidental loss of dc bus A and BX due to their spatial proximity: develop procedure to deal with the loss of dc buses A and BX.	X				X			
Haddam Neck	Cable vault and cable spreading area - large consequences of a fire in an area where there is a high concentration of cables of both trains A & B: change training philosophy to increase sensitivity to and awareness of transient combustibles and maintenance activities in the cable spreading area and cable vault to same level as for control room and switchgear rooms.	X				X			
Haddam Neck	Cable spreading area cable separation concerns: installation of additional sprinkler heads on trays where numerous cable sprays are stacked above each other.	X							X
Haddam Neck	Trains A & B control cable separation in the control room and cable spreading area: improve AOP 3.2.57 for recovery from control room, switchgear room A and cable spreading area fires. Also, for the cable spreading area revise procedures for 'A' charging pump so that it can be credited for fire scenario in cable spreading area.	X				X			
Haddam Neck	Trains A & B cable separation in PAB corridor (trays C1 and C7) for charging pump supports: re-route either charging pump B main or auxiliary lube oil pump cables.	X				X			
Haddam Neck	Vertical unanchored waste oil tank in the waste oil area can topple over flammable liquid containers in that area: installation of additional anchorage to prevent toppling.		X						X
Haddam Neck	End of bottle of Carbon Dioxide fire suppression system for containment cable vault requires restraint modification (N 94-12): modify end bottle restraint to adequately restrain bottle.		X			X			
Haddam Neck	Batteries to the diesel fire pump require restraining: installation of anchorage to prevent battery movement.		X						X
Haddam Neck	Lack of electrical separation in switchgear room A: built a new switchgear building with cable separation between trains A and B for most of the safe shutdown systems.			Fire PSS, Feb. 1986	X				
Haddam Neck	Dependence of both ECCS trains on a single motor control center: built new Switchgear building that includes the capability to shutdown from the alternate shutdown panel.			Fire PSS, Feb. 1986	X				

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Haddam Neck	Four 4160/480V oil filled transformers: replace with dry transformers.			Fire PSS, Feb. 1986	X				
Hatch 1 & 2	None identified.								
Hope Creek	None identified.								
Indian Point 2	None identified.								
Indian Point 3	Eliminate susceptibility of multiple EDG exhaust fans (and thus multiple EDGs) to fire within a single fire zone: realign the power feeds to the EDG exhaust fans and auxiliaries.	X				X			
Indian Point 3	Reduce the susceptibility of the plant to switchgear room fires: recommend that the area-wide, total flooding CO_2 fire suppression system within the switchgear room be restored to automatic actuation.	X							X
Kewaunee	None Identified								
LaSalle 1&2	A RCIC "sneak circuit" could cause the isolation of RCIC each time a loss of offsite power occurred. Under these conditions a false, loss-of-power induced high RCIC room temperature signal was generated and the in-board ac-powered isolation valve received a signal to close. However, the valve could not close because it had no ac power. When ac was restored to the valve, a relay race ensued, and the relay associated with room high temperature was energized before the loss of power contact opened. The valve would shut, isolating RCIC because it "sensed" RCIC room high temperature before it "sensed" a loss of power. This event could occur during station blackout, loss of offsite power or due to a loss of a train of ac power: changes were made to LaSalle Procedure LOA-AP-07, "Loss of Auxiliary Electrical Power," to identify the sequence of events which will result in the non-recoverable isolation of the RCIC inboard steam isolation valve. In addition, the operators receive training during every training cycle targeted specifically at the "sneak circuit" concern.	X			X				
LaSalle 1&2	RCIC room temperature isolation logic, in cases where train A ac power has failed but train B ac power is available, isolates if no other emergency core cooling system is working: change the RCIC room temperature isolation logic so that, in cases where train A ac power has failed but train B ac power is available, RCIC does not isolate if no other emergency core cooling system is working.	X		NUREG/CR-4832, Vol. 3, Section 7.4		X			

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
LaSalle 1&2	For long-term containment heat removal accidents and ATWS sequences, decay heat can be removed by venting the containment. The use of a rubber boot connecting the vent pipe to the standby gas treatment system results in steam being released into the reactor building which creates a severe environment for components. This severe environment can affect the ability of the systems to perform their functions. The degree of this impact will depend on the environment produced and the qualification of the equipment subject to the severe environment: the IPE submittal does not contain specific criteria for resolution of the DHR vulnerability issue. (Procedures exist to use the fire protection system's water supply as an alternate water supply for vessel injection. The EOPs provide several different means of vessel depressurization. Revision 4 of the BWR Owners Group Emergency Procedures Guidelines has been implemented. A hardened vent path does exist.)	X			X				
Limerick 1&2	Plant improvements that allow fire areas to be screened out: designate fire compartments 1 (corridor areas, recombiner rooms, backwash tank and pump rooms, recombiner access area, water analyzer rooms), 7 (4 kV switchgear corridor), 22 (Unit 1 cable spreading room) and 23 (Unit 2 cable spreading room) as transient combustible-free zones.	X			X				
Limerick 1&2	Plant improvements that allow fire areas to be screened out: Replace wood scaffolding with metal scaffolding and revise procedures to prevent further use of wood scaffolding.	X							X
Limerick 1&2	Plant improvements that allow fire areas to be screened out: the combustion control procedure will be revised to provide more conservative combustible control guidelines in safety-related areas within the reactor enclosures.	X				X			
Limerick 1&2	Plant improvements that allow fire areas to be screened out: additional doors will be administratively controlled by the Hazard Barrier Procedure as "fire" doors to limit the amount of air available for combustion. The existing doors are fire rated.	X				X			
Limerick 1&2	Thermo-Lag issue: plant changes not specified.	X							X
Limerick 1&2	Fixed combustibles present a fire risk impact in fire compartments 2 (13.2 kV switchgear room), 20 (Unit 1 static inverter room), and 26 (remote shutdown panel room): increase fire brigade drill activities and brigade awareness in these areas. These fire compartments were reviewed against the NEI 91-04 closure guidance and procedural changes were deemed appropriate.	X				X			
Maine Yankee	Recommended enhancements as a result of the fire analysis: Abnormal Operating Procedures, "Plant Shutdown Plan for Fire in Containment, Spray Pump Area, Steam/Feed Valve House, Containment Electrical Penetration In South Elevation 46'," will be updated to indicate: (1) that the potential exists for steam generator low pressure (SGLP) isolation in the event of a fire in the main steam and feedwater valve house, and (2) to indicate that there is the potential to hot short open containment penetrations 64, 65, and 66 to manually isolate the main steam drains, if open. [IPEEE, pg. 4-239]	X				X			

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Maine Yankee	Recommended enhancements as a result of the fire analysis: AOP, "Plant Shutdown Plan for Fire in Control Room, Control Room Cable Chase, Protected Cable Vault, Protected Cable Tray Room, and Protected Switchgear Room," will also be updated to: (1) indicate that the potential exists for an SGLP trip isolation (dual loss of vital buses), in the event of a fire in the protected cable tray room and the protected switchgear room, and (2) to indicate that there is the potential to hot short open containment penetrations 23, 24, 39, 60, 64, 65, 66, and 92 and that valves are available to manually isolate these leak paths.	X				X			
Maine Yankee	Recommended enhancements as a result of the fire analysis: AOP, "Plant Shutdown Plan for Fire in Reactor - MCC - Elevation 21' and 33' Containment Electrical Penetration Room - North Elevation 46'," will be updated to warn the operator about the possibility that disconnecting power to MCC-7B and MCC-8B would result in removing power to HPSI suction valves from the RWST. If a such open PORV results in a safety injection signal, and the HPSI suction valves are closed and de-powered, the HPSI pumps could fail due to the loss of suction. The problem can be avoided if the HPSI suction valves are opened prior to removing power from MCC-7B and MCC-8B.	X				X			
Maine Yankee	Recommended enhancements as a result of the fire analysis: AOP, "Plant Shutdown for Fire in Turbine Hall or Circulating Water Pump House," will specifically indicate that the protected switchgear room could be affected because of a loss of HVAC due to a fire in the turbine building and the need for temporary measures to maintain cooling.	X				X			
Maine Yankee	Station operating practice is to isolate the hydrogen supply at the tube trailer after filling the generator and volume control tank (VCT): in order to provide more positive control, the station administrative procedure was updated to reflect this requirement.	X			X				
Maine Yankee	Fires (and potentially floods) in the control room, cable vault, protected cable tray room, protected SWGR and EFW pump room have the potential to cause multiple hot shorts or power failures to both in-board and out-board containment isolation valves: incorporate guidance into Severe Accident Management Guidelines being developed.	X				X			
Maine Yankee	AFW availability is functionally important to mitigation of various external events: take actions to improve AFW availability.	X							X
McGuire 1&2	None identified								
Millstone 1	N/A								
Millstone 2	Oil fires involving main condenser vacuum pump, transformer, main feedwater pump or turbine generator located in the turbine building: reroute turbine driven auxiliary feedwater (TDAFW) control cable to remove it from turbine building fire area.	X				X			

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Millstone 2	Fire disabling the "B" turbine building component cooling water pump, "A" and "C" turbine building component cooling water pumps, or the turbine generator: eliminate dependence of service water (SW) and the "C" SW pump power cable for emergency diesel generator cooling by installation of a permanent cross tie pipe between fire protection and SW to the EDG cooling connection. This modification ensures that ac power will remain available in the event that a fire disables all three pumps by eliminating EDG dependence upon service water.	X				X			
Millstone 2	Large quantities of transient combustibles (protective clothing) in open storage racks placed near concentration of cable trays: reduce quantity; store in enclosed fire-related lockers, and/or remove from area.	X			X				
Millstone 2	MP2 relies on the MP1 fire suppression system for fire protection. The seismic capacity of the MP1 diesel fire pump fuel tank may not be adequate. Fires generated as a result of earthquakes are common. Fire pumps driven using offsite power cannot be depended upon since most earthquakes result in a loss of offsite power: perform additional evaluation to ensure seismic adequacy. If determined to be inadequate, perform modification to improve seismic ruggedness.		X				X		
Millstone 2	A long run of fire water header system piping along the turbine building's north wall appears to have a very low seismic capacity because the pipe run is inadequately attached to its supports: attach pipe to its supports adequately.		X						X
Millstone 2	The block wall construction of the fire pump house (shared by MP1and MP2) may not provide adequate seismic ruggedness: evaluate ruggedness and, if ruggedness is low, enhance the structure.		X				X		
Millstone 3	None identified								
Monticello	None identified								
Nine Mile Point 1&2	Fire and high winds can lead to SBO scenarios where recovery is not likely for much longer than the 8 hours currently considered for SBO mitigation: enhance operator training on procedure N1-SOP-14, "Alternate Instrumentation," to include station blackout mitigation without dc power.	X			X				
Nine Mile Point 1&2	Cables associated with both divisions of emergency ac, dc, and various front-line systems (i.e., feedwater) are located in the southeast corner of the turbine building (elevation 261'), a number of combustibles are in this immediate vicinity. These combustibles included: five drums filled with oily rags, paint cans, bags of trash, electronic equipment, and aerosol spray cans: storage of combustibles in this area should be curtailed or more tightly controlled.	X			X				
Nine Mile Point 1&2	Instrument air is required to align containment vent and containment spray in the torus cooling mode. Containment vent valves could be opened with handwheels. Containment spray valves currently fail as is (normally open) on loss of instrument air and have no handwheels for manual operation: add manual handwheels to valves so operators could align torus cooling without instrument air.	X				X			

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Nine Mile Point 1&2	The criteria suggest that a relatively insignificant event could lead to evacuation. This creates uncertainty with regard to what conditions really lead to evacuation, yet the control room is the preferred location for plant recovery. Once the remote shutdown procedure is entered, one interpretation would be that the control room is evacuated, yet a more likely situation may be to use the remote shutdown panel to enhance plant recovery. Giving up control of reactor inventory in the control room when only long term heat removal needs recovery from outside the control room is not considered an appropriate strategy. Given that the remote shutdown panel is being used due to control room conditions, it is likely to be outside the event-driven remote shutdown procedures and utilization of the EOPs is not explicitly addressed: Operations decided to make the following revisions: (1) to ensure that the control room would be evacuated only under extreme conditions, the symptoms indicative of conditions requiring entry to N2-SOP-78 "Remote Shutdown" have been revised and the procedure explicitly states the senior shift supervisor determines whether the control room is uninhabitable. (2) Although the SOP mentions the use of the RHR in a "pseudo" LPCI mode, there is no explicit guidance. Revision of the procedure is being considered. (3) A satellite master copy of N2-EOP-RPV has been placed in the remote shutdown rooms to ensure the operators have adequate guidance when the control room is evacuated.	X			X				
North Anna 1&2	ESGR room fire analysis results require action according to the NUMARC severe accident closure guidelines: Based on the frequency criterion, only Severe Accident Management Guidelines (SAMG) procedures are required to be developed to mitigate severe accident scenarios resulting from fires in this area. These procedures are being developed and implemented on a schedule independent of the IPEEE. The fire results have been transmitted to the group developing these procedures so appropriate enhancements can be incorporated.	X				X			
North Anna 1&2	Action plan for structural steel fireproof coating in the cable spreading room: follow up on Appendix R commitments.			X					X
North Anna 1&2	Action plan for foam installation concerns pointed out in Information Notice 88-56: revision of station procedure.			X					X
North Anna 1&2	Action plan for Appendix R fire dampers: develop a new periodic test.			X		X			
North Anna 1&2	Action plan for periodic tests: incorporate barrier mark numbers and show all fire barrier penetrations on controlled drawings for use in periodic tests.			X		X			
North Anna 1&2	Thermo-Lag is used in a limited fashion in the North Anna containments to provide radiant energy shielding between redundant equipment and cables separated by less than 20 ft. The redundant components protected in this way are the RHR pump motors, fuel building cable penetration, two transmitters, a conduit (from transmitters to penetration) and a conduit (neutron flux indication): Thermo-Lag in Unit 2 has either been sheathed with stainless steel or replaced with a new radiant energy shield of Marinite board sheathed with stainless steel. Unit 1 is to complete the same modifications.	X							X

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement: IPEEE Fire	IPEEE Seismic/Fire	Other	Implementation Status of Improvement: Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
North Anna 1&2	Non-seismic IPEEE and fire procedures enhancements: revise procedures to provide for the comparison of the instrumentation in the remote monitoring panel and the auxiliary shutdown panel with the main control room instrumentation to identify possible instrumentation malfunctions during MCR, emergency switchgear room, or cable vault &tunnel fires.	X				X			
North Anna 1&2	Non-seismic IPEEE and fire procedures enhancements: revise procedures to direct operators to use the auxiliary shutdown panel's alternate control circuits to operate specific equipment if automatic actuation and the main control room control switch fails.	X				X			
North Anna 1&2	Non-seismic IPEEE and fire procedures enhancements: revise procedures to preferentially operate non-failed equipment from the main control room unless the MCR is evacuated.	X				X			
North Anna 1&2	Non-seismic IPEEE and fire procedures enhancements: revise procedures to identify the significance of auxiliary feedwater and charging pump controls for recovery actions from the auxiliary shutdown panel.	X				X			
North Anna 1&2	Non-seismic IPEEE and fire procedures enhancements: revise procedures to remove the control circuit fuses and locally close the 4160V breakers necessary to re-establish feedwater flow in the event of no running condensate, main feedwater or auxiliary feedwater pumps due to possible control circuit failures.	X				X			
North Anna 1&2	Non-seismic IPEEE and fire procedures enhancements: revise procedures to remove the control circuit fuses and locally close the 4160V breakers necessary to re-establish charging/HHSI flow in the event of no running charging pumps due to possible control circuit failures.	X				X			
North Anna 1&2	Non-seismic IPEEE and fire procedures enhancements: revise procedures to de-energize the pressurizer PORV control circuits added to other procedures as action to be taken if the pressurizer PORV control switch does not close the valve and the block valve can not be closed.	X				X			
North Anna 1&2	Modification which significantly enhances the ability of the plant to respond to fires, especially those originating in the MCR: alternative monitoring capability has been provided for the primary system process parameters that need to be monitored for safe shutdown. The auxiliary monitoring panels located in the fuel building provide the indications (but not control) for each unit. The instrument loops that are used for the auxiliary monitoring panels are routed via the fuel building penetrations, which are independent of the normal instrument circuit routing to the control room. The power source for the instrumentation can be supplied from either unit and is therefore independent of the normal instrument power supplies. The auxiliary monitoring panels together with auxiliary shutdown panels have been designed to provide alternative shutdown capability for the primary and secondary process monitoring variables independent of the cable vault and tunnel, emergency switchgear/instrument rack room, and control room.			Appendix R Program (Fire)	X				

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
North Anna 1&2	Modification which significantly enhances the ability of the plant to respond to fires, especially those originating in the MCR: The auxiliary shutdown panel, located in the emergency switchgear room of each unit, is an alternate means of control to bring the plant to a hot standby condition. In the case of the MCR becoming inaccessible due to a fire, the plant may be safety controlled and monitored from the auxiliary shutdown panel and auxiliary monitoring panels for an extended period of time. Emergency diesel generators 1H and 2H are provided with local control isolation independent of the MCR. The control panels are located in the diesel generator room and in the emergency switchgear room. The control panels provide isolation from the control room along with local control, indication, and metering capabilities for the emergency diesel generator and the 4160V emergency bus breakers.			Appendix R Program (Fire)	X				
North Anna 1&2	Modification which significantly enhances the ability of the plant to respond to fires, especially those originating in the MCR: spurious operations, the high/low pressure boundary interface at the pressurizer PORV and block valves were reviewed. The pressurizer PORVs are normally shut; the block valves are normally open and can be shut to isolate flow through the PORVs. These valves are located inside the containment. It may be postulated that a single fire could cause both the PORVs to operate spuriously and disable the block valves in an open position, resulting in a fire-initiated loss of coolant. The 125V dc high/low pressure circuits for the pressurizer PORVs have been modified to include the following: (1) motive power to the 125V dc solenoids is routed in dedicated rigid steel conduits from the solenoid (inside containment) to the MCR, (2) flex steel conduit is used at the solenoid valve, penetration areas, and through the floor sleeve at the emergency switchgear room to the MCR, (3) Procedures are in place requiring the circuits listed on the Summary Evaluation Table to be de-energized in the event of a fire. (4) A new isolation switch is provided in the emergency switchgear room to ensure the circuit can be de-energized from either the existing switch in the MCR or the emergency switchgear rooms.			Appendix R Program (Fire)	X				
North Anna 1&2	Generic issue for reactor coolant pump seal failure during normal operation: procedural changes to ensure that adequate instructions are available to provide backup seal injection and thermal barrier cooling by adding steps to ECA-0.0 which refer the operators to the seal cooling abnormal procedure.			IPE Fire Analysis		X			
North Anna 1&2	Generic issue for reactor coolant pump seal failure during off-normal conditions: licensee has committed to supply backup diesel generator capability. This added equipment will ensure the provision of seal injection during off-normal conditions.			IPE Fire Analysis		X			
Oconee 1-3	Recommendation for improvement: water spray and smoke control observations and precautions should be placed in the next revision of the Oconee pre-fire plan.	X				X			
Oconee 1-3	Recommendation for improvement: the combustible storage locker near the SSF diesel should be mounted so that the combustible materials cannot spill around the diesel during a seismic event.	X							X

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Oconee 1-3	Recommendation for improvement: the wall between the HPI pump room, the LPI and reactor building spray rooms and the spent fuel pool cooling pump room should be sealed to limit smoke migration.	X							X
Oconee 1-3	Recommendation for improvement: open-head water sprinklers in the cable rooms, equipment rooms, and cable shafts should be replaced with a closed-head design.	X							X
Oconee 1-3	Recommendation for improvement: a water-based suppression system should be evaluated for the turbine bearings.	X							X
Oconee 1-3	Recommendation for improvement: the pre-fire plan update should include an expansion of the pre-fire plan to include all fire zones.	X							X
Oconee 1-3	Recommendation for improvement: fire detectors should be installed on the turbine building side of the maintenance support building elevator to activate the elevator fire lockout feature.	X							X
Oconee 1-3	Recommendation for improvement: flammable storage cabinets should be rigidly mounted.	X							X
Oconee 1-3	Recommendation for improvement: members of the Keowee workforce should be advisors to the fire brigade.	X							X
Oconee 1-3	Recommendation for improvement: the Unit 2 equipment room smoke purge fan should be removed (it has been determined that this fan serves no fire protection function).	X							X
Oconee 1-3	Recommendation for improvement: fire protection drawings should be updated.	X							X
Oconee 1-3	Recommendation for improvement: the door to the CT-4 blockhouse should have a fire link on both sides (currently there is one on one side).	X							X
Oconee 1-3	Recommendation for improvement: evaluate appropriate procedures coupled with potential physical changes to improve the seismic adequacy of the turbine building oil sump barrels and lube oil drums located in the Powdex Area.	X					X		
Oyster Creek	Continued transient combustible control and good housekeeping are essential elements of a successful fire protection program: continue good housekeeping practices and continue attention to the control of transient combustibles.		X		X				
Oyster Creek	The high pressure CO_2 system cylinders in the turbine building could potentially become missiles following a seismic event resulting in the potential loss of turbine bearing No. 10 and turbine generator exciter fire suppression: consider upgrading the anchorage of the CO_2 system.		X						X
Oyster Creek	The small oil filter of the turbine generator hydrogen seal oil unit is supported only by a vertical stanchion, and no lateral support is provided. The small diameter piping to and from the filter forms an approximately 8' cantilever which was found be inflexible: consider additional support of the oil filter.		X						X

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Oyster Creek	During the seismic-fire interaction walkdowns an anchorage chain was not attached to an embedded eye hook. During a seismic event, interaction of the Arrowhead Demineralizer trailer with the station blackout transformer fire suppression system could result in inadvertent suppression system actuation: anchorage was re-installed, which prevents the trailer from becoming a missile in high wind scenarios.		X		X				
Oyster Creek	Following a seismic event the high pressure generator purge CO_2 rack outside the turbine building could overturn and result in missiles. The low pressure CO_2 tank could be disabled by these missiles due to its close proximity to the high pressure CO_2 storage racks: high pressure CO_2 rack and anchorage have been replaced.		X		X				
Oyster Creek	Following a seismic event, actuation of the fire protection drop-weight-actuated deluge valves could result in diversion of fire suppressant from actual fire events or fire suppressant spray effects on safety-related equipment: although walkdowns verified that electrical panels and safety equipment are generally well sealed or spray protected, new deluge valves with less potential for seismic actuation would provide additional margin from fire suppressant effects and flow diversion.		X						X
Oyster Creek	Consider operator and fire brigade training on fire analysis scenarios from the most significant Oyster Creek fire areas (fire areas for which a detailed evaluation is performed): particular emphasis should be placed on the unscreened fire areas (the cable spreading room and the "A" 480V ac switchgear room).		X						X
Palisades	Fire analysis identified several significant operator actions that impact fire core damage frequency: operator training will be conducted on all operator actions credited in the IPEEE that were not credited in the IPE, including the fire-risk-significant operator actions.	X				X			
Palo Verde 1-3	The essential air cooling unit (ACU) for the train B dc equipment rooms does not have a remote disconnect switch to disconnect it from the control room in the event of a control room fire: until this switch is installed, instructions exist to lift leads, if necessary, to operate the ACU.			Appendix R (Fire)		X			
Palo Verde 1-3	Fire panel control circuits that actuate dampers for cooling air flow in the train A and train B essential switchgear rooms were not designed for train separation; i.e., a fire in train B switchgear room could cause a loss of cooling to train A switchgear and dc equipment rooms: perform modification that will separate the damper actuation circuitry and reconfigure the fire damper control panels.			Appendix R (Fire)		X			
Palo Verde 1-3	Certain safe shutdown and non-safe shutdown control circuits in each train have common fuse protection, with non-safe shutdown circuits routed through a common area. This could lead to a single fire causing loss of control power to safe shutdown circuits of both trains: contingency plans are in place to address this concern, in the event that a fire occurs prior to the time that a design change is implemented to install additional fuses.			Appendix R (Fire)		X			

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Palo Verde 1-3	A fire in the Unit 1 control room, upper cable spreading room or corridor building cable shaft has the potential to cause a major disruption to the Palo Verde switchyard and, in turn, lead to a loss of offsite power to all three Palo Verde Units (Units 2 and 3 would remain on line connected to the two remaining offsite transmission lines): perform a grid stability analysis under the worst case scenario of tripping Unit 1 and loss of three of the five offsite transmission lines.	X						X	
Palo Verde 1-3	A fire in the Unit 1 control room, upper cable spreading room or corridor building cable shaft has the potential to cause a major disruption to the Palo Verde switchyard and, in turn, lead to a loss of offsite power to all three Palo Verde Units (Units 2 and 3 would remain on line connected to the two remaining offsite transmission lines): developed a procedure to disconnect the Unit 1 switchyard control circuits, re-establish alignment of the transmission lines and re-energize the startup transformers to restore offsite power availability to the Palo Verde units; this was integrated into the interface procedures used between the SRP power dispatch office, the APS energy control center, and the Palo Verde Unit 1 control room, which has primary responsibility for site interface with the switchyard.	X			X				
Palo Verde 1-3	FIVE pilot project completion in 1991 resulted in: development of a computer-based training course to inform plant staff of the project findings. In addition, enhancements were made to the annual site access training received by all personnel having unescorted access to the protected area regarding fire prevention and identification of those fire-sensitive areas, including those non-Appendix R compartments where offsite power conductors and control circuits are located.	X			X				
Peach Bottom 2&3	Reactor Unit 2 RHR pump fire area: revise procedure to allow venting of Unit 2 containment.	X				X			
Peach Bottom 2&3	Control room/CSR, 4 kV switchgear rooms (34 (shared area), 35(shared area), 32(shared area)), reactor Unit 2 reactor building north, reactor Unit 3 cooling water, reactor Unit 2 recirculation motor generator set: enhance control of transient combustibles in area and increase fire brigade awareness of the area and its hazards.	X				X			
Peach Bottom 2&3	Reactor Unit 2 cooling water fire area: enhance control of transient combustibles in area and increase fire brigade awareness of area. Create operator action to regain dc batter charger.	X				X			
Peach Bottom 2&3	Reactor Unit 3 recirculation motor generator set fire compartment: upgrade fire compartment to provide separation from other fire areas. Increase fire brigade awareness of area and its hazards. Enhance control of transient combustibles in fire area. Create operator action to locally operate valve.	X				X			
Peach Bottom 2&3	RW miscellaneous areas, reactor Unit 3 reactor building north, turbine building Unit 2 wing area and turbine building Unit 3 lube oil tank: upgrade fire compartment barriers to provide separation from other fire areas. Increase fire brigade awareness of area and its hazards. Enhance control of transient combustibles in fire area.	X				X			

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Peach Bottom 2&3	4kV switchgear corridor: upgrade fire compartment barriers to provide separation from other fire areas. Increase fire brigade awareness of area and its hazards. Create transient combustible free zone.	X				X			
Peach Bottom 2&3	Reactor Unit 2 torus and reactor Unit 3 torus fire compartment: create new combustible free zone (Thermo-Lag change). Increase fire brigade awareness of area and its hazards.	X				X			
Peach Bottom 2&3	Reactor Unit 2 core spray: create operator actions to allow operation of valve at MCC.	X				X			
Peach Bottom 2&3	Reactor Unit 2 reactor building south: enhance control of transient combustibles in fire area. Increase fire brigade awareness of area and its hazards. Revise procedure to allow venting of Unit 3 containment.	X				X			
Peach Bottom 2&3	Reactor Unit 3 core spray and battery room fire compartment: revise procedure to allow venting of Unit 3 containment.	X				X			
Peach Bottom 2&3	Reactor Unit 3 reactor building south: revise procedure to allow venting of Unit 3 containment. Create operator action to allow valve operation at MCC. Enhance control of combustibles in fire area.	X				X			
Peach Bottom 2&3	Battery room fire compartment (shared area): enhance control of transient combustibles in the fire area. Increase fire brigade awareness of area and its hazards. Create operator action to allow valve operation at MCC.	X				X			
Peach Bottom 2&3	4kV switchgear room 33 (shared area): enhance control of transient combustibles in fire area. Increase fire brigade awareness of area and its hazards. Create operator action to swap to alternate power supply and operate valve at MCC.	X				X			
Peach Bottom 2&3	4kV switchgear room 38 (shared area): enhance control of transient combustibles in fire area. Increase fire brigade awareness of area and its hazards. Create operator action to manually align switchgear to diesel.	X				X			
Peach Bottom 2&3	4 kV switchgear room 38 (shared area): enhance control of transient combustibles in fire area. Increase fire brigade awareness of area and its hazards. Create operator action to swap to alternate power supply.	X				X			
Peach Bottom 2&3	4 kV switchgear room 39 (shared area): enhance control of transient combustibles in fire area. Increase fire brigade awareness of area and its hazards. Revise procedure to allow venting of Unit 2 containment.	X				X			
Peach Bottom 2&3	Turbine building: Unit 2 lube oil 50L (shared area), Unit 3 lube oil 50P (shared area), Unit 2/Unit 3 pipe tunnels 50R-3 (shared area), Unit 2 lube oil tank 50R-5 (shared area), Unit 2 lube oil equipment 50R-7 (shared area), Unit 3 lube oil equipment 50R-8 (shared area), Unit 2 RFPT C lube oil 50R-10 (shared area), Unit 3 RFPT C lube oil 50R-11 (shared area): upgrade fire barriers to provide separation from other fire areas.	X				X			

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Peach Bottom 2&3	Turbine building: Unit 3 wing area 50R-2&4 (shared area) and 13.2kV switchgear area 50R-9a (shared area): upgrade fire compartment barriers to provide fire separation from other fire areas. Increase fire brigade awareness of area and its hazards. Enhance control of transient combustibles in fire area. Modify offsite power control to retain offsite power in fire area.	X				X			
Peach Bottom 2&3	Reactor building A (shared area) and B (shared area): upgrade fire compartment barriers to provide separation from other fire areas. Create transient combustible free zone.	X				X			
Peach Bottom 2&3	Turbine building Unit 2/Unit 3 remainder: upgrade fire compartment barriers to provide separation from other fire areas. Increase fire brigade awareness of area and its hazards. Enhance control of transient combustibles in fire area. Modify offsite power control to retain offsite power in fire area. Automate existing fire suppression system on elevation 116'.	X				X			
Peach Bottom 2&3	Mercury switches in water suppression system manual pull stations (4): replace with non-mercury switches.		X			X			
Peach Bottom 2&3	CO_2 system panel Cardox relays (4): establish procedural controls to mitigate results of spurious relay operation.		X			X			
Peach Bottom 2&3	DG Cardox tank 00S101: Add restraints to restrict motion during seismic event.		X			X			
Peach Bottom 2&3	CO_2 tanks 20S101, 30S101, and 20S112: perform an engineering evaluation to determine the impact of a seismically induced CO_2 release on plant equipment in the turbine building.		X			X			
Perry	None identified								
Pilgrim	None identified								
Point Beach 1&2	Control room/cable spreading room fire procedure: Abnormal Operating Procedure for fire in the control room is being revised to add verification that additional valves which were identified as part of the Point Beach IPEEE process are closed for containment isolation.	X				X			
Point Beach 1&2	Evaluation of smoke detectors in the control room: the control room smoke detectors are located below a grated ceiling. Smoke would not accumulate to the level of the detectors until the section of the control room above the ceiling is filled with smoke. An engineering evaluation on this condition is being conducted.	X					X		

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement: IPEEE Fire	Basis for Improvement: IPEEE Seismic/Fire	Basis for Improvement: Other	Implementation Status of Improvement: Have Impl.	Implementation Status of Improvement: Plan to Impl.	Implementation Status of Improvement: Being Evalu.	Implementation Status of Improvement: Reject	Implementation Status of Improvement: Not Stated
Prairie Island 1&2	Add fire wrap or other barrier material to the exposed length of cable 1DCB-1 above cable tray 1SG-LB22 in FA 32: an issue has been identified involving a fire in FA 32 (Unit 1 side AFW pump/instrument air compressor room) could affect the control power for both safeguards 4160V buses. A large lube oil spill fire on 121 IA compressor (fire suppression system fails) could result in the failure of cables 1DCA-1 and 1DCB-1, which run in the overhead near the compressor. Cable 1DCA-1 (control power to Bus 15) is not fire wrapped in this area. Cable 1DCB-1(control power to Bus 16) has a radiant energy shield (thermal board) for the portion of the cable that runs through tray 1SG-LB22. However, above the air compressor, 1DCB-1 then exits the tray and runs up through a penetration in the ceiling to enter the Bus 16 room. No wrap or other barrier protects the cable in this region. Fire modeling predicts that a fire "jet layer" will exist above a height of 16 feet above the floor in this room. All unprotected equipment for a radial distance of 14 feet from the fire must be expected to fail due to the fire in this region.	X			X				
Prairie Island 1&2	In many fire core damage sequences (fire may be initiated in a number of fire areas), the 121 cooling water pump and a roof exhaust fan are available, but since (in these sequences) the fan and pump are powered from the opposite train, the fan is not running. This leads to failure of the 121 CL pump due to lack of sufficient ventilation: add instructions to Fire Safety Procedure F5 Appendix D for the operator to locally start an available roof exhaust fan to reestablish safeguards screenhouse ventilation. (Subsequent review revealed that procedures already exist to accomplish this task for fires that cause loss of power from MCC 1AB1 or 1AB2.)	X						x	
Prairie Island 1&2	On an air compressor large oil spill fire the assumption is that the fire causes spurious closure of MV-32027 prior to loss of power from MCC 1A2. The cooling water supply valve MV-32027 could also be opened. This operator action is credited in the SSA: add instructions to Fire Safety Procedure F5 Appendix D for the operator to manually open a suction supply valve to the 12 AF pump on a fire in FA 32 (Unit 1 side AFW pump/IA compressor room). Alternative - fire wrap the length of conduit for cable 1A2-6A that runs in FA 32. (Upon review of the procedure, it was noted that direction is included in F5 Appendix D to de-energize MCC 1A2 and manually operate as necessary the suction motor valves for 12 MDAFWP for a fire in FA-32. No credit for this recommendation (or for the operator action at all) was given in the final quantification. It was decided to conservatively remove credit for this action.)	X						X	
Prairie Island 1&2	Need for shutdown outside the control room: ensure that existing training for manual fire suppression in the mitigation of fires in the control room and relay room (fire brigade to relay room) includes a discussion of the risk significance of this action in the prevention of a core damage accident.	X				X			
Prairie Island 1&2	Action to prevent a core damage accident: ensure that existing training for the operator task to shut down the plant from outside the control room per F5, Appendix B includes a discussion of the risk significance of this action in the prevention of a core damage accident.	X				X			

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Prairie Island 1&2	Action to prevent a core damage accident due to internal fires: ensure that existing training for the operator task to perform bleed and feed cooling of the RCS includes a discussion of the risk significance of this action in the prevention of a core damage accident due to internal fires.	X				X			
Prairie Island 1&2	Operator task to perform dc panel switching in the battery room and relay room for a fire in FA 59: ensure that training (lesson plans, out plant checkoffs, etc., as appropriate) exists for this operator task. Training should include information relative to the importance of this action to stopping loss of inventory through the RCS vent solenoid valves. (A job performance measure was also created to address this issue.)	X				X			
Prairie Island 1&2	Verify cable separation in the G panel due to the potential for a large fire internal to the panel to cause the loss of offsite and onsite power: power would then have to be restored from the diesel generators from outside the control room. The current panel design configuration meets criteria that support prevention of this scenario for most fires. This recommendation is made only to provide added assurance of this critical assumption with respect to its impact on plant risk due to fires. (A visual inspection was performed on the G panel and confirmation was made on the proper design separation between trains. In addition, through the plant design change process, proper separation of cables throughout the plant was verified.)	X			X				
Prairie Island 1&2	During walkdown activities conducted as part of the IPEEE improvement process (Fire Risk Scoping Study issues task) ERIN Engineering determined that a potentially weak anchorage exists for the main CO_2 storage tank in the Unit 1 turbine building. This is a concern for the seismic-fire interactions review, in that this tank is required for fire suppression in the relay room. Suppression in the relay room is important due to the critical equipment for plant safe shutdown located in this room: upgrade the anchorage for the main Cardox tank for relay room automatic fire suppression.		X		X				
Prairie Island 1&2	During the walkdown activities conducted as part of the IPEEE improvement process (Fire Risk Scoping Study issues task) ERIN Engineering determined that a potentially weak anchorage exists for the diesel driven fire water pump batteries and fuel oil day tank in the plant screenhouse. This is a concern for the seismic-fire interactions review, in that a seismic event of sufficient magnitude to cause the loss of offsite power (cause loss of the motor-driven fire water pumps) could also render the diesel fire pump unavailable: upgrade the anchorage for the diesel driven fire water pump batteries and fuel oil day tank.		X		X				

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Quad Cities 1 & 2	The increased risk from the (non-qualified) cables at Quad Cities results from the fact that they can be damaged more quickly by an exposure fire than cables qualified to the provision of IEEE 383. And once ignited, they burn more intensely with both higher heat release rate and mass loss rates. The use of possible additional fire protection features to overcome the inherent liabilities of non-qualified cable will be fully evaluated. It is possible that simple steps, such as additional suppression or detection, new fire barriers, or radiant heat shields, sealing the top of electrical and switchgear cabinets, or other yet-to-be-identified solutions, may provide sufficient additional margin to reduce the contribution of non-qualified cables to an acceptable level.	X					X		
Quad Cities 1 & 2	Human factors (number of steps in QARPs). When a severe, uncontrolled fire requires the plant operators to enter the QARPs to achieve safe shutdown, the process is more complex and does not have the options available in normal shutdown procedures. Five operators can be required to perform more than 20 separate steps, all coordinated by the Licensed Reactor Operator using portable radios. Two potential solutions to the human factors contribution to the CDF are immediately apparent. The first is to evaluate possible upgrades to the QARPs. As a minimum, they should be more clearly written in line with Human Factors. The second potential solution is to evaluate the use of additional transfer/isolation switches. The prudent installation and use of such transfer/isolation switches may substantially reduce the actual number of manual actions currently required by the QARPs.	X					X		
Quad Cities 1 & 2	Use of opposite unit equipment. Because so many fire scenarios assume damage to and unavailability of redundant trains of equipment, most fire scenarios direct the operators to the QARPs and use of opposite equipment to achieve safe shutdown. This contributes to the CDF because of three interrelated issues: equipment availability, multi-unit LCOs, and contribution to QARP complexity. Quad Cities will evaluate the availability of alternative (not shared) equipment in the unit of concern that has not previously been identified and that may be used and credited without reliance on equipment in the opposite unit. Quad Cities will also evaluate the need for additional protection of specific cables and components of the shutdown trains.	X					X		
Quad Cities 1 & 2	Both divisions of safe shutdown cables are located in the same fire compartment; e.g., cable tunnels, cable spreading room, etc. The evaluation of additional protection for specific cables and components of the shutdown trains, and the evaluation of availability of alternative equipment discussed above will also address the concerns of lack of separation between redundant divisions where such lack of separation exists.	X					X		

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Quad Cities 1 & 2	Because of lack of specific information, the QARPs assume that all equipment in the fire area of concern is damaged and consequently all potential spurious actuations occur. The response from the QARPs is to strip all control circuits from potentially involved buses which further increases the number of required manual actions. As part of QARP revisions, Quad Cities will consider the selective stripping of control circuits from identified buses rather than the indiscriminate stripping of all circuits simply on the basis that they are all assumed to be damaged. The evaluation of transfer/isolation switches and the evaluation of protection of specific equipment/cables will also address specific concerns with spurious actuations during a fire.	X					X		
River Bend	None identified.								
Robinson	Fire originating in Battery Room "A-16" in motor control center (MCC) cabinets MCC "A" or MCC "B," combined with failure to perform manual fire suppression, which leads to a loss of Train "A" and "B" direct current (dc) power: seal open conduits emerging from the top of MCC "A" and MCC "B" to avert the formation of a hot gas layer from a fire in the battery room sufficient to prevent effective fire suppression.	X				X			
Robinson	Fire in a ruptured transformer in the switchyard that results in a loss of offsite power and subsequent loss of the dedicated shutdown (DS) diesel generator: revise procedure used by the fire brigade to include instructions to the fire brigade to emphasize protection of the DS conduit by aiming water streams on the DS conduit to counter the damaging effects of radiant heat.	X				X			
Robinson	Eight fire scenarios consist of a fire in a reactor turbine generator board (RTGB) panel, located in the control room, that is suppressed within the RTGB cabinet or that propagates to other RTGB cabinets. Another fire scenario is a fire that originates in a control room location other than an RTGB panel that results in an evacuation of the control room: evaluate and select early warning fire detection systems that utilize air sample technology to detect fires in the pre-incipient stage of combustion for the RTGB cabinets in the control room.	X				X			
Salem 1 & 2	There are two sets of cables supplying offsite power to the 4kV vital buses and these are routed through one elevation of the turbine and service buildings before entering the auxiliary building. The two sets provide a redundant source of power to the vital 4kV buses. Thus, if one set is damaged by fire, the second set could provide power to all three buses. In the turbine building, there is an area in which the two redundant sets of cables are separated by less than 10 feet: Transient combustible controls similar to those in place for the auxiliary building, penetration areas and service water intake structure will be put into effect for this area of the turbine building. Daily walkdowns will be performed for the elevation on which the cables are routed to ensure that combustibles do not accumulate beneath the cables, and fire watches will be posted if any normally active suppression systems are disabled. Procedures are being revised to ensure these activities are accomplished through periodic monitoring.	X			X				

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Salem 1 & 2	The IPE analysis credits operator intervention for the provision of alternate means for providing cooling to the switchgear and control areas of the Salem Generating Station in the event that the normal HVAC systems cannot meet the areas' requirements. Review of operator action, with respect to external events concerns indicates that a more current and substantial basis is required for crediting these actions: revised heat-up calculations for these areas to address a wider spectrum of accident scenarios and provide a more detailed response methodology were being done. Additionally, based on those calculations, a new, stand-alone procedure is to be developed to address loss of HVAC to these areas.		X			X			
San Onofre 2	For fire compartment 2-AC-30-20A (control room and cabinet area), implementation of an administrative change to Procedure SO23-13-2, "Shutdown from Outside the Control Room," would allow operators to use offsite power in the event that the reserve auxiliary transformers are not inadvertently tripped by fire-induced damage to panel 2/3CR-63.	X				X			
San Onofre 2	For fire compartments 2-DG-30-155 and 2-DG-30-158 (diesel generator rooms), implementation of an administrative change to SO23-13-21 (fire) would allow operators to recover power to the 4 kV switchgear by disconnecting power to the diesel generator feeder breaker and reclosing the offsite power breaker on the switchgear.	X				X			
San Onofre 2	For fire compartments 2-AC-50-44, 2-AC-50-45, 2-AC-50-46, 2-AC-50-47 (distribution rooms), 2-AC-50-35 and 2-AC-50-40 (switchgear rooms), implementation of an administrative change to alarm response procedure SO23-15-60.A1 (annunciator panel 60A, emergency HVAC) would allow operators to use air duct and gas-driven fans to prevent room heat-up.	X			X				
San Onofre 3	The applicability of fire improvements to Unit 3 is not clear. The procedures modified apparently apply to both units (as noted by the SO23 designation). However, the fire compartments appear to be part of Unit 2. It is noted, however, that the units share a common control room and the other compartments might also be shared.	X							X
Seabrook	Turbine building relay room: install fire detection.	X				X			
Seabrook	Combustible materials stored near the west wall of the turbine building: expand water suppression along the west wall.	X				X			
Seabrook	In Appendix A of Fire Response Procedure OS1200.01, indicate that PCC can be impacted by a PAB fire: modify the fire response procedure. Also, in Fire Procedure OS1200.02(b), response should incorporate an instruction for verifying PORV status.	X				X			
Seabrook	Highlight important fire areas: place tighter restrictions on the amount of combustible material allowed in important fire risk areas, e.g., the control room, CSR, PCC Pump Area, and SW Pump Area.	X				X			
Seabrook	Reduce the incidence of RCP seal LOCAs: install high temperature O-rings.			IPE		X			
Seabrook	Reduce offsite release risk: make improvements to the RCS depressurization procedures.			IPE		X			
Seabrook	Reduce offsite release risk: use administrative controls to reduce the time that containment purge valves are allowed to be open.			IPE		X			

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Sequoyah 1&2	None identified.								
Shearon Harris 1	A potential LOCA can be mitigated by the closing of the appropriate PORV block valve from the alternate control panel. The procedure will be revised to specifically check the status of the pressurizer PORVs after transfer to the alternate control panel to require closure of a block valve if necessary to isolate a failed open relief valve.	X				X			
South Texas 1&2	None identified.								
St. Lucie 1 & 2	For fires and possibly other events, it is unlikely, but possible, for one unit to be "black" while the other unit still has offsite power and continues to operate. In screening 6 areas, it was assumed power could be fed from the other unit. FPL will perform an engineering evaluation to determine cost effective methods for reduction of CDFs for the 6 areas. Although the focus will be on use of the power crosstie, other alternatives will be considered. FPL will also provide a schedule for implementation of any procedure revisions or other alternatives resulting from the evaluation.	X					X		
St. Lucie 1 & 2	For each turbine generator building switchgear room, fire modeling showed fires (fixed or transient combustibles) would not propagate throughout the room. However, a transient spill (and fire) could occur between the two trains affecting both A2 and B2 4.16kV buses, resulting in loss of offsite power. There are (usually open) roll-up doors in the concrete block walls between the "A" and "B" trains which would prevent spread of such a spill. The FIVE analysis assumed such a spill (affecting both buses) would not occur in screening this area. FPL will revise Fire Protection procedures to maintain the west roll-up door in each unit turbine generator building switchgear room closed.	X				X			
St. Lucie 1 & 2	The Unit 2 "B" switchgear room (compart C) also contains the 2A5 480 V load center. This meets Appendix R criteria since 2A5 is separated from 2B5 by a wall and powers no Appendix R safe shutdown equipment. However, it does power equipment used in the PSA and this area will not screen assuming loss of "B" train and 2A5. Fire modeling showed fires (fixed and transient combustibles) would not propagate throughout the room. The FIVE analysis assumes that the "B" train is lost but does not assume the 2A5 Load Center is simultaneously lost. (An analysis was done which verified loss of adjacent 2A5 and 2B5 Load Centers would screen.) Also, 2A5 and most control power enters the load center from underneath (a separate fire compartment). However, one control cable is routed through the room such that it could be affected by a fire in/around the "B" switchgear. Even with action on this cable, compartment C does not screen but the conditional damage frequency is reduced by a factor of five from 2.28E-06/ry to 4.51E-07/ry. FPL will perform an engineering evaluation to determine cost effective methods for reduction of core damage frequencies for this area. Although focus will be on the cable mentioned above, other alternatives will be considered. FPL will provide a schedule for any resulting modifications or procedural enhancements.	X					X		

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Summer	Several potential procedural enhancements were identified during the detailed HRA evaluation: a note or caution to not isolate offsite power to the safe shutdown bus before the diesel generator is ready to load would reinforce fire emergency procedure (FEP) training on this point.	X							X
Summer	Several potential procedural enhancements were identified during the detailed HRA evaluation: specific steps to operate the breakers required to restore offsite power, if the diesel generator fails after being loaded, would reduce the HEP for this evolution that is currently performed using a generic procedure attachment for breaker operation.	X							X
Summer	Several potential procedural enhancements were identified during the detailed HRA evaluation: a step to ensure that the turbine driven EFW pump is shut down would reduce the chance of a potential pump start going unnoticed. This start is not normally expected to occur but is possible if the sequence of actions being taken by two independent operators is reversed.	X							X
Summer	The details of the 1998 VCSNS Fire Induced Vulnerability Addendum confirmed operator knowledge that use of the FEPs in an inappropriate scenario will increase the CCDP: these details will be used to enhance operator training by providing scenario analyses that quantitatively illustrate this point. They will also provide a better basis for the SRO's judgement of when FEP use is appropriate.	X							X
Summer	The details of the 1998 VCSNS Fire Induced Vulnerability Evaluation Addendum can be used to enhance fire brigade member training: they are currently trained on which components are important for functions required to achieve and maintain safe shutdown but this will be enhanced with information on the areas in which severe uncontrolled fires could require use of the FEPs.	X							X
Surry 1&2	Prevent breakers from automatically reopening due to control circuit failures caused by fires: revise fire contingency action procedures to allow removal of control circuit fuses of auxiliary feedwater, main feedwater, and condensate pumps for 4160V breakers between offsite power and the emergency buses.	X				X			
Surry 1&2	Consider using main feedwater or condensate if auxiliary feedwater has failed: revise fire contingency action procedures	X				X			
Surry 1&2	In the turbine building hydrogen flat area of Unit 1, a stainless steel tube is connected to the hydrogen piping near an elbow. The piping appeared to be flexible which may cause relatively large displacements of occur: flexibility of tubing will be further reviewed and if needed, a modification will be performed.		X				X		

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Surry 1&2	Housekeeping/conduct of maintenance issues related to hydrogen cylinders in the turbine building: address via procedure. A carbon dioxide bottle in the emergency service water pump house was not clamped properly: address via housekeeping procedure. In the turbine building, adjacent to the skid containing 1(2)-LO-FL-1, was an unanchored oil drum resting on the floor and a small container containing oil which is supported approximately 5' off the floor by four angles: unanchored drum is also a housekeeping issue and will be addressed programmatically. Further review will be performed on the smaller container.		X				X		
Susquehanna 1&2	The improvements are mainly procedural. They include restrictions on combustible materials storage, ban on smoking inside buildings, special covers for barrels, and opening of the floor drain in the Lower Cable Spreading Room to allow sprinkler system water to drain.	X			X				
Three Mile Island 1	None identified.								
Turkey Point 3&4	None identified.								
Vermont Yankee	Ensure that limit switches and torque switches would not be bypassed assuming a fire-induced hot short for control room and cable vault fire events: reconfigure control circuits of the Appendix R motor-operated valves (MOVs). With these modifications, a hot short may cause an MOV to inadvertently transfer position; however, the motor operator will remain protected via the limit and torque switches. Thus, the MOV itself is not damaged and remains available for later manipulation at the alternate shutdown panel.			Appendix R	X				
Vermont Yankee	A hot short of the power cables for the SRVs causing spurious opening of an SRV: the power cables for two of the four SRVs were re-routed to provide spatial separation. The control cables for each SRV are enclosed in dedicated, grounded steel conduit in the reactor building and cable vault. Based on this modification, a hot short is judged to be very remote. Also, the ADS bypass switch in the control room is protected with a 1 hour rated fire barrier to prevent completion of a spurious ADS signal.			Appendix R	X				
Vermont Yankee	RCIC failure (the preferred injection source), the licensee would de-pressurize the RPV from the ASD panel and use an LPCI pump for low pressure injection: enhance alternate shutdown controls to include remote control of two SRVs for pressure control and/or initiation of RPV de-pressurization.			Appendix R	X				
Vermont Yankee	Simplify initiation of alternate shutdown: use the vernon tie as the preferred alternate shutdown ac power source instead of EDG-1A.			Appendix R	X				
Vermont Yankee	Improvements made to plant procedures: including the alternate shutdown and fire response procedures, fire barrier surveillance requirements procedures and natural phenomena procedures.			Appendix R	X				
Vermont Yankee	Expeditious control of alternate shutdown equipment without having to physically replace individual control circuit fuses which may have blown due to fire-induced circuit damage: install back-up fuses in the circuits of the Appendix R safe shutdown equipment. These backup fuses are automatically switched into the circuits whenever the alternate shutdown selector switches are operated.			Appendix R	X				

Table 3.5: Fire-related plant improvements (Continued)

Plant	Description of fire-related plant improvements cited in the IPEEE submittal (regardless of status or source)	Basis for Improvement			Implementation Status of Improvement				
		IPEEE Fire	IPEEE Seismic/Fire	Other	Have Impl.	Plan to Impl.	Being Evalu.	Reject	Not Stated
Vermont Yankee	Enhance fire independence between the turbine building and the radwaste corridor: install a 3 hour fire rated door between these fire areas.			Appendix R	X				
Vermont Yankee	Potential fire damage interaction for Switchgear room fires: improve and separate the RHR system minimum flow valve pump interlocks and RHR Room cooler start circuits.			Appendix R	X				
Vermont Yankee	Improvement opportunity, north wall lower NE corner room: include the top 6" of the north wall in the lower NE ECCS corner room (just under floor elevation 232' 6") in the plant fire barrier inspection program.	X			X				
Vermont Yankee	Improvement opportunity, vertical cable tray fire stops: enhance inspection and maintenance of vertical cable tray stops at each floor in the reactor building, to limit fire spread from one elevation to another.	X			X				
Vermont Yankee	Improvement opportunity, periodic fire prevention inspections: perform periodic fire prevention inspections of the reactor building and control building on a more frequent (monthly) basis.	X			X				
Vermont Yankee	Improvement opportunity, Vernon Tie breaker cables: relocate or otherwise protect the control cables for the Vernon Tie breakers 3V, 4V, and 3V4 in the east and west switchgear rooms from fires that are likely to damage offsite power control cables.	X					X		
Vogtle 1&2	None identified.								
Waterford 3	In the essential chiller room, a fire on chiller A or chilled water pump A could damage cables associated with chiller train B: although the design meets the requirements of Appendix R, due to the availability of the AB train during this scenario, the robustness of the plant to fire hazards in this fire area could be improved by adding fire wrap to the B chilled water cables in the vicinity of the A chiller.	X							X
Watts Bar 1	None identified								
Wolf Creek 1	None identified								

* Formerly known as Washington Nuclear Project Number 2.

Table 3.6: Comparison of a set of past fire PRAs with respective IPEEE submittals

Plant	Completion date[1]	PRA CDF (/ry)[1]	PRA important contributors[1]	IPEEE CDF (/ry)	IPEEE important contributors	Comments
Indian Point 2	1982	2.0E-04	Electrical tunnels, switchgear room	1.8E-05	Main control room, cable spreading room, switchgear room, electrical penetration area, primary makeup area	IPEEE did not use 1982 study. The 1982 study CDF does not include plant modifications since completion of the study.
Indian Point 3	1982	6.3E-05	Switchgear room, electrical tunnel, cable spreading room	5.6E-05	Switchgear rooms, cable spreading room, main control room, diesel generator rooms	IPEEE did not use 1982 study. The 1982 study CDF does not include plant modifications since completion of the study.
Limerick	1983	2.3E-05	Equipment rooms, switchgear room, assess area, main control room, cable spreading room [2]	Not Reported	Main control room, remote shutdown panel room, auxiliary equipment room, switchgear room, static inverter room	IPEEE submittal does not use CDF explicitly. Final screening of main control room & auxiliary equipment room is based on 1983 fire PRA.
Millstone 3	1983	4.8E-06	Main control room, instrument rack room, cable spreading room	4.8E-06	Charging pumps, CCW pumps, cable spreading room, MCC & rod control areas, main control room, instrument rack room	
Seabrook	1983	1.7E-05	Main control room, instrument rack room, cable spreading room	1.2E-05	Main control room, primary auxiliary building, turbine building, switchgear room, service water pump house	IPEEE submittal is a new analysis but acknowledges 1983 study and compares the differences in the method and data employed.
Oconee	1984	1.0E-05		6.0E-06	turbine building, cable shaft	Original study was updated in 1990 and used for preparing IPEEE.
TMI-1	1987	8.6E-05	MCC area, switchgear room, electrical cabinet area	2.2E-05	Inverter rooms, switchgear rooms, main control rooms, auxiliary relay room	Used 1987 study as a starting point and modified it for IPEEE.
South Texas	1989	<1.2E-06	Main control room	5.1E-07		IPEEE submittal partly based on the 1989 study.
Diablo Canyon	1990	2.9E-05	Cable spreading room, main control room	2.7E-05	Cable spreading room, main control room	Original study was updated for IPEEE.
Peach Bottom 2	1990	2.0E-05	Main control room, switchgear rooms, cable spreading room	Not Reported	turbine building, reactor building, switchgear rooms	1990 study done by SNL for U.S.NRC. Licensee conducted its own study for IPEEE. Licensee did not report a fire CDF.

Table 3.6: Comparison of a set of past fire PRAs with respective IPEEE submittals (Continued)

Plant	Completion date[1]	PRA CDF (/ry)[1]	PRA important contributors[1]	IPEEE CDF (/ry)	IPEEE important contributors	Comments
Surry 1	1990	1.1E-05	Switchgear room, main control room, auxiliary building, cable vault or tunnel	6.3E-06	Switchgear rooms, turbine building, cable spreading room, electrical equipment room, main control room, cable vault	1990 study done by SNL for U.S.NRC. Licensee conducted a new fire analysis for IPEEE.
LaSalle 2	1993	3.2E-05	Main control room, switchgear rooms, equipment rooms, turbine building, cable shaft	3.2E-05	Main control room, switchgear rooms, equipment rooms, turbine building, cable shaft	1990 SNL for U.S.NRC study referenced.

1. The list of fire PRAs and related information is taken from NUREG/CP0162, "Research Needs in Fire Risk Assessment," Proceedings of 25th U.S. Nuclear Regulatory Commission Water Reactor Safety Information Meeting, Bethesda, Maryland, October 20-22, 1997, Volume 2, pages 93-116.
2. IPEEE did not use the 1983 study.

4. HIGH WINDS, FLOODS, AND OTHER EXTERNAL EVENTS TABLES

This section contains a summary table of the methodologies and results for the IPEEE HFO events. Included in the table are (1) an identification of the methodologies used for the different severe accident event categories, (2) the core damage frequency contributions for those cases where PRA information was reported, (3) HFO-related plant improvements, and (4) comments regarding the plant HFO results and review.

Table 4.1: Methodologies and results for the HFO external events

Plant	Methodology		CDF (/ry)	HFO-related improvements	Comments
	High winds & floods	Others			
ANO 1&2	Compliance with SRP (qualitative progressive screening per NUREG-1407)	Compliance with SRP	Not estimated	Added scuppers to parapet walls of auxiliary building to relieve roof load during heavy rainfall	
Beaver Valley 1	Compliance with SRP	Compliance with SRP (est. chemical release (hazard) frequency <7E-7/year)	Not estimated	None identified; backup cooling water intake structure was added prior to IPEEE as further protection against barge accidents	The licensee stated that, based on their estimates of the various HFO contributions to CDF, the plant's HFO-related events were dominated by chemical spills and releases from onsite and offsite (chemical factory within 5 miles & nearby railroad) though < 1E-6/ry
Beaver Valley 2	Compliance with SRP	Compliance with SRP (est. chemical release (hazard) frequency <7E-7/year)	Not estimated	None reported	Dominated by chemical spill (like Unit 1); CDF contribution from lightning strikes is bounded by LOOP and other trip events in IPE
Braidwood	Compliance with SRP	Compliance with SRP	Not estimated	None reported	
Browns Ferry 2&3	Compliance with SRP	Compliance with SRP	Not estimated	None reported	

Table 4.1: Methodologies and results for the HFO external events (Continued)

Plant	Methodology		CDF (/ry)	HFO-related improvements	Comments
	High winds & floods	Others			
Brunswick 1&2	PRA	Compliance with SRP	High Winds: 4E-6 Floods: 2E-7	Development of severe accident management guideline for high winds under consideration	
Byron 1&2	Compliance with SRP	Compliance with SRP	Not estimated	None reported	
Callaway	Compliance with SRP	Compliance with SRP	Not estimated	None reported	
Calvert Cliffs 1&2	Compliance with SRP	Compliance with SRP	Not estimated	(1) Emergency procedures (EP) revised to prepare portable ventilation fans and generator for adequate ventilation in switchgear rooms during hurricanes; (2) restrictions added to prevent air flights over protected plant areas	
Catawba	Tornadoes: PRA Others: compliance with SRP	Compliance with SRP	Tornadoes: 3E-6 (~11% total external CDF)	None reported	

Table 4.1: Methodologies and results for the HFO external events (Continued)

Plant	Methodology: High winds & floods	Methodology: Others	CDF (/ry)	HFO-related improvements	Comments
Clinton	Compliance with SRP	Compliance with SRP	Not estimated	None reported	During flooding walkdown, identified & repaired a potential leak path on a hatch over shutdown service water pipe tunnel and noted that sump pumps in service water pump room could fail due to MCC flooding in non-safety screen house; however, such failures are preventable using existing procedures
Columbia	Compliance with SRP	Compliance with SRP	Not estimated	Procedures were modified to ensure that C-Van containers are not stacked in close proximity to safety-related buildings	Formerly known as Washington Nuclear Project Number 2
Commanche Peak 1&2	Tornadoes: PRA Others: compliance with SRP	Compliance with SRP	Tornadoes: 4E-6 (~15% of total external CDF)	None reported	
Cooper	Compliance with SRP	Compliance with SRP	Not estimated	Protection of DG exhaust from tornadoes	

Table 4.1: Methodologies and results for the HFO external events (Continued)

Plant	Methodology		CDF (/ry)	HFO-related improvements	Comments
	High winds & floods	Others			
Crystal River 3	Compliance with SRP	Compliance with SRP	Not estimated	None reported	
Davis-Besse	Compliance with SRP	Compliance with SRP	Not estimated	(1) A potential condition adverse to quality report (PCAQR) was initiated to address onsite hazards from hazardous materials; (2) revise description of hazards from chemical stored or transported onsite in USAR; (3) controlled materials program revised so that new onsite materials will be evaluated for control room habitability; (4) PCAQR initated for monitoring roof drains & standing water on aux building roof (643 ft elevation)	
Cook 1&2	Compliance with SRP	Compliance with SRP	Not estimated	None reported	
Diablo Canyon	Compliance with SRP	Compliance with SRP	Not estimated	None reported	
Dresden 2&3	Compliance with SRP	Compliance with SRP	Not estimated	Installed scuppers in roof parapets of turbine, reactor, & crib house buildings	

Table 4.1: Methodologies and results for the HFO external events (Continued)

Plant	Methodology		CDF (/ry)	HFO-related improvements	Comments
	High winds & floods	Others			
Duane Arnold	High winds: PRA Floods: Compliance with SRP	Compliance with SRP	High winds: on the order of 1E-6	(1) Concrete barriers installed around propane tank near DG rooms to protect against vehicle impact & explosion-fire; (2) increase distance between enlarged H_2 storage system and safety equipment	
Farley 1	Compliance with SRP	Compliance with SRP	Not estimated	None reported	
Fermi 2	Compliance with SRP	Compliance with SRP	Not estimated	To prevent ice formation in service water pumps causing common mode failure of DG, the licensee (1) implemented procedures to check on this condition, (2) installed permanent temperature monitoring equipment, (3) installed fiberglass curtain to reduce wind chill effects, & (4) modified terminations of cold weather lines to reduce chilling effect	
FitzPatrick	Compliance with SRP	Bounding analysis	< 1E-6	AOP enhanced to (1) warn of potential DG loss of air supply and (2) instructions how to prevent the loss of air supply	

Table 4.1: Methodologies and results for the HFO external events (Continued)

Plant	Methodology		CDF (/ry)	HFO-related improvements	Comments
	High winds & floods	Others			
Ft Calhoun	High winds: Compliance with SRP Floods: Compliance with SRP, with bounding deterministic and simplified PRA	Compliance with SRP	Dam break-induced flooding:6E-7 Other periodic flooding:3E-6	Provisions provided for flooding: portable pumps; new, detailed procedures; upgraded doors; sandbags; and sealing conduits	Bounding events were (1) dam break induced flooding and (2) periodic flooding from intense rainstorms
Ginna	Compliance with SRP	Compliance with SRP	Not estimated	Added roof scupper to reduce roof ponding	
Grand Gulf	Compliance with SRP	Compliance with SRP	Not estimated	(1) increased maintenance on drains, (2) revised plant flood mitigation procedures, and (3) add inspection of roof drains, overflows, and drainage system	
Haddam Neck	High winds: PRA Floods: bounding PRA	Bounding PRA	High winds: 6E-5 Floods: 5E-6 Lightning: 8E-6 Snow & ice: 7E-6 Others: <1E-6	(1) Added air cooled DG (before IPEEE); (2) arrangement to have fuel delivered <24 hours of high winds; (3) Procedures to install flood door <8 hours during flooding conditions; and (4) procedure to remove snow & ice during winter storms	

Table 4.1: Methodologies and results for the HFO external events (Continued)

Plant	Methodology		CDF (/ry)	HFO-related improvements	Comments
	High winds & floods	Others			
Hatch 1&2	Compliance with SRP	Compliance with SRP	Not estimated	Added guideline for company pilots to not fly over nuclear facility structures	
Hope Creek	Compliance with SRP	Compliance with SRP	< 1E-6	(1) installed tornado missile shield at technical support center door, (2) had Coast Guard stop shipments of explosives on river near plant	
Indian Point 2	PRA for certain events, compliance with SRP for others	Compliance with SRP	Tornadoes: 2E-5 Extra tropical Cyclones: 1E-5 Flooding: 7E-6	(1) Added surveillance of control building drain flapper valve flow, (2) add weather stripping to doors transformer area to switchgear room, (3) add screens on 480V switchgear room equipment hub drains	
Indian Point 3	Compliance with SRP	Hydrogen: PRA Others: compliance with SRP	Hydrogen explosions: Slightly greater than 1E-6	Considering installation of H_2 supply line excess flow valve from line ruptures in turbine or auxiliary buildings	Licensee estimated HFO CDF dominated by H_2 explosions in turbine building causing severe damage to DG fire panel, 6.9KV switchgear, and other cables, leading to SBO

Table 4.1: Methodologies and results for the HFO external events (Continued)

Plant	Methodology		CDF (/ry)	HFO-related improvements	Comments
	High winds & floods	Others			
Kewaunee	Compliance with SRP	Compliance with SRP	Not estimated	None reported	Underground diesel oil storage tank vents are susceptible to tornado-generated missiles. Was to be resolved in 1996.
LaSalle	Compliance with SRP or probabilistic bounding analysis	Compliance with SRP or probabilistic bounding analysis	Aircraft impact: 5E-7 Turbine-generated missiles: 1E-7 Tornadoes: 3E-7	None reported	Licensee did not provide sufficient information to resolve several IPEEE issues, including an HFO-related issue (i.e., external flooding and site drainage)
Limerick	Compliance with SRP	Compliance with SRP	Not estimated	None reported	
McGuire	PRA	PRA	Tornadoes: 2E-5	None reported	Some minor plant fixes (e.g., replacement of corroded nut and missing bolt) were made during earlier PRA development walkdown

Table 4.1: Methodologies and results for the HFO external events (Continued)

Plant	Methodology		CDF (/ry)	HFO-related improvements	Comments
	High winds & floods	Others			
Millstone 2	Compliance with SRP	Compliance with SRP	Not estimated	Added roof scuppers; considering (1) protection of cooling ducts and dampers in control and DG rooms from high wind, (2) closure time for flood gates, (3) flood protection for service water pump motor, and (4) revising plant grading to reduce potential site flooding	
Millstone 3	PRA	PRA	"insignificant contributors"	None reported	
Monticello	Tornadoes: bounding PRA Others: compliance with SRP	Compliance with SRP	Tornadoes: <1E-6	None reported	
Nine Mile Point 1	PRA and compliance with SRP	Compliance with SRP	Tornadoes & high winds: 2E-6 Tornado missiles impacting DG doors: 4E-7 Flooding from heavy rainfall: 6E-7	Provision for specific operator training in the use of two different instrument rooms together in the event of tornadoes	

Table 4.1: Methodologies and results for the HFO external events (Continued)

Plant	Methodology		CDF (/ry)	HFO-related improvements	Comments
	High winds & floods	Others			
Nine Mile Point 2	Compliance with SRP	Compliance with SRP	< 1E-6	None reported	
North Anna 1&2	Compliance with SRP	Compliance with SRP	Not estimated	None reported	
Oconee	PRA	PRA	Tornadoes: 1E-5 Flooding: 7E-6	(1) Additional training (study) by plant personnel regarding tornado events, (2) additional evaluation of sheltering plans for plant personnel during tornadoes, (3) gas explosion protection modification to each letdown storage tank room's ventilation exhausts, (4) additional operator guidance to prevent H_2 buildup in letdown storage tank rooms if ventilation becomes unavailable	
Oyster Creek	High winds: PRA Floods: compliance with SRP	Compliance with SRP	High winds & tornadoes: < 1E-6 (staff estimate)	None reported	
Palisades	Compliance with SRP	Compliance with SRP	Not estimated	Added seiche protection barrier to protect DG fuel oil transfer pumps during heavy rainfall	

Table 4.1: Methodologies and results for the HFO external events (Continued)

Plant	Methodology		CDF (/ry)	HFO-related improvements	Comments
	High winds & floods	Others			
Palo Verde	High winds: PRA Floods: compliance with SRP	Compliance with SRP	Tornadoes: Unit 1: 2E-7 Unit 2: 4E-7 Unit 3: 3E-9	None reported	
Peach Bottom	Compliance with SRP	Compliance with SRP	< 1E-6	None reported	
Perry	NUREG/CR-4839 (RMIEP)	NUREG/CR-4839 (RMIEP)	Not estimated	None reported	
Pilgrim	High winds & local intense precipitation: calculated low hazard frequency Floods: compliance with SRP	Aircraft crashes: calculated low hazard frequency Others: compliance with SRP	Not estimated	None reported	
Point Beach 1&2	PRA and compliance with SRP	Compliance with SRP (low frequency of occurrence)	High winds: 3E-7 Floods: 3E-6	DG exhaust stacks were modified to accommodate higher wind loads	
Prairie Island 1&2	Compliance with SRP	Compliance with SRP	Not estimated	None reported	
Quad Cities 1&2	Compliance with SRP	Compliance with SRP	Not estimated	None reported	
River Bend	Compliance with SRP	Compliance with SRP	Not estimated	None reported	

Table 4.1: Methodologies and results for the HFO external events (Continued)

	Methodology				
Plant	High winds & floods	Others	CDF (/ry)	HFO-related improvements	Comments
Robinson	High winds: PRA Floods: compliance with SRP	Compliance with SRP	Tornado/wind-induced missiles: 2E-6 Other wind effects: 8E-6	Emergency procedures enhanced to (1) ensure walkdown of DG fuel oil transfer pumps following severe wind conditions and (2) isolate oil leakage & provide make up, if needed	Unique site features which enhance recovery of fuel oil to the DGs include a number of cross-connections and ability to pump oil directly from a fuel truck
Salem 1&2	Compliance with SRP, supplemented with bounding PRA	Compliance with SRP, supplemented with bounding PRA	Not estimated	(1) Improved service & auxiliary building penetration seals, (2) improved H_2 tank hold downs, (3) CWS intake structure modification to protect against detritus (blockage)	Licensee reported improvement reduced estimated contribution to CDF from flooding by approximately three orders of magnitude (from about 1E-4 to about 1E-7)
San Onofre	Compliance with SRP, supplemented by quantitative hazard screening	Compliance with SRP	< 1E-6	None reported	
Seabrook	PRA	PRA	Floods: 1E-6 Transportation: 1E-6 Others:<1E-6	(1) Modified SWS pump house roof to allow scuppers to function properly, (2) modified several exterior doors to withstand pressure differential from high wind	
Sequoyah	Compliance with SRP	Compliance with SRP	Not estimated	None reported	

Table 4.1: Methodologies and results for the HFO external events (Continued)

Plant	Methodology: High winds & floods	Methodology: Others	CDF (/ry)	HFO-related improvements	Comments
Shearon Harris	Compliance with SRP	Compliance with SRP	Not estimated	None reported	
South Texas 1&2	High winds: compliance with SRP (low hazard frequency) Floods: PRA	Chemical releases: PRA Others: compliance with SRP	Chemical release from nearby chemical facilities: 8E-6	None reported	
St. Lucie 1&2	Compliance with SRP	Compliance with SRP	Not estimated	Revised administrative procedure regarding severe weather preparations based on lessons learned from Hurricane Andrew	
Summer	Compliance with SRP	Compliance with SRP	Not estimated	None reported	
Surry 1&2	PRA and compliance with SRP (low hazard frequency)	PRA and compliance with SRP (low hazard frequency)	Not provided	(1) Modified parapet to reduce ponding, (2) heavy rainfall procedures modified to allow water to flow out of turbine building and restrict water flow into the main control room	Licensee used NUREG/CR-4550 results for some cases and bounding quantitative analyses for others
Susquehanna 1&2	Compliance with SRP	Compliance with SRP	Not estimated	None reported	

Table 4.1: Methodologies and results for the HFO external events (Continued)

Plant	Methodology		CDF (/ry)	HFO-related improvements	Comments
	High winds & floods	Others			
Three Mile Island 1	PRA	PRA	High winds: 7E-7 Floods: 8E-5 Chemical: 2E-7 Aircraft: 4E-7	(1) Unspecified modification and (2) added special procedures to mitigate river flooding consequences	
Turkey Point 1&2	Compliance with SRP	Compliance with SRP	< 1E-6	(1) Refurbish existing flood wall and stop logs, (2) EDG fuel oil transfer pump elevated to protect against hurricane surge, (3) EP revised to improve protection against severe storms, (4) two nearby fossil plant stacks strengthened and procedures and other plant modifications as a result of lessons learned from Hurricane Andrew	

Table 4.1: Methodologies and results for the HFO external events (Continued)

Plant	Methodology		CDF (/ry)	HFO-related improvements	Comments
	High winds & floods	Others			
Vermont Yankee	Compliance with SRP	Compliance with SRP	Not estimated	(1) Underground conduit to switchgear rooms sealed from external flooding, (2) enhanced procedures to provide guidance following tornado or hurricane and to address site flooding, (3) evaluating possible OP revision for dam-failure-induced site flooding to protect switchgear room, (4) diesel fuel oil transfer pump house wall penetration sealed against external flooding	
Vogtle	Compliance with SRP	Compliance with SRP	Not estimated	None reported	
Waterford	Compliance with SRP	Compliance with SRP	Not estimated	(1) Pump added to cooling tower basin to mitigate excess ponding, (2) Pump was added to the surveillance testing program	

Table 4.1: Methodologies and results for the HFO external events (Continued)

Plant	Methodology		CDF (/ry)	HFO-related improvements	Comments
	High winds & floods	Others			
Watts Bar	Compliance with SRP	Compliance with SRP	Not estimated	A shield plate installed to protected auxiliary building opening against tornado-generated missiles	
Wolf Creek	Compliance with SRP	Compliance with SRP	Not estimated	None reported	

5. UNRESOLVED SAFETY ISSUES AND GENERIC SAFETY ISSUES TABLES

This chapter contains tables with plant-specific information related to the unresolved safety issues (USIs) and generic safety issues (GSIs) discussed in Chapter 5 of Volume 1. Table 5.2 in Volume 1 lists all the plants and gives the staff's overall assessment of the verification of each generic issue. The tables in this chapter supplements Chapter 5 of Volume 1 by identifying significant plant-specific information related to the generic issues and sub-issues. For example, Table 5.4 on GSI-147 identifies plant-specific features related to this issue, and, in the few cases where there was inadequate information to verify all aspects of this generic issue, the table identifies the missing information.

The information in these tables was derived from the NRC's Staff Evaluation Reports and the supporting Technical Evaluation Reports. These tables address GSI-57, "Fire Protection System Impact on Safety-Related Equipment" (Table 5.1); the effects of a revised probable maximum precipitation (PMP) in rainfall and flood elevation (Table 5.2) and related plant improvements (Table 5.3) as part of GSI-103, "Design for Probable Maximum Precipitation"; GSI-147, "Fire-Induced Alternate Shutdown/Control Room Panel Interactions" (Table 5.4); GSI-148, "Smoke Control and Manual Fire-Fighting Effectiveness" (Table 5.5); GSI-156, "Systematic Evaluation Program" (Table 5.6); GSI-172, "Multiple System Responses Program" (Table 5.7); and the Sandia "Fire Risk Scoping Study" (Table 5.8).

Table 5.1: GSI-57, Fire protection systems impact on safety-related equipment

Plant name	Licensee considered	
	Suppression diversion	Suppression activation
Arkansas 1 & 2	Yes	Yes
Beaver Valley 1 & 2	Yes	Yes
Braidwood 1 & 2	Yes	Yes
Browns Ferry 2 & 3	Yes	Yes
Brunswick 1 & 2	Yes	Yes
Byron 1 & 2	Yes	Yes
Callaway	Yes	Yes
Calvert Cliffs 1 & 2	Yes	Yes
Catawba 1 & 2	Yes	Yes
Clinton	Yes	Yes
Columbia*	Yes	Yes
Comanche Peak	Yes	Yes
Cooper	Yes	Yes
Crystal River 3	Yes	Yes
D. C. Cook 1 & 2	Yes	Yes
Davis-Besse	Yes	Yes
Diablo Canyon 1 & 2	Yes	Yes
Dresden 2 & 3	Yes	Yes
Duane Arnold	Yes	Yes
Farley 1	Yes	Yes
Fermi 2	Yes	Yes
FitzPatrick	Yes	Yes
Fort Calhoun	Yes	Yes
Ginna	Yes	Yes
Grand Gulf	Yes	Yes

Table 5.1: GSI-57, Fire protection systems impact on safety-related equipment (Continued)

Plant name	Licensee considered	
	Suppression diversion	Suppression activation
Hatch 1 & 2	Yes	Yes
Hope Creek	Yes	Yes
Indian Point 2	Yes	Yes
Indian Point 3	Yes	Yes
Kewaunee	Yes	Yes
LaSalle	No	No
Limerick 1 & 2	Yes	Yes
Millstone 2	Yes	Yes
Millstone 3	Yes	Yes
Monticello	Yes	Yes
McGuire 1 & 2	Yes	Yes
Nine Mile Point 1	Yes	Yes
Nine Mile Point 2	Yes	Yes
North Anna 1 & 2	Yes	Yes
Oconee 1, 2, & 3	Yes	Yes
Oyster Creek	Yes	Yes
Palisades	Yes	Yes
Palo Verde 1, 2, & 3	Yes	Yes
Peach Bottom 2 & 3	Yes	Yes
Perry	Yes	Yes
Pilgrim	Yes	Yes
Point Beach 1 & 2	Yes	Yes
Prairie Island 1 & 2	Yes	Yes
Quad Cities 1 & 2	Yes	Yes

Table 5.1: GSI-57, Fire protection systems impact on safety-related equipment (Continued)

Plant name	Licensee considered	
	Suppression diversion	Suppression activation
River Bend	Yes	Yes
Robinson 2	Yes	Yes
Salem 1 & 2	Yes	Yes
San Onofre 2 & 3	Yes	Yes
Seabrook	Yes	Yes
Sequoyah 1 & 2	Yes	Yes
Shearon Harris	Yes	Yes
South Texas 1 & 2	Yes	Yes
St. Lucie 1 & 2	Yes	No
Summer	Yes	Yes
Surry 1 & 2	Yes	Yes
Susquehanna 1 & 2	Yes	Yes
Three Mile Island 1	Yes	Yes
Turkey Point 1 & 2	Yes	No
Vermont Yankee	Yes	Yes
Vogtle 1 & 2	Yes	Yes
Waterford 3	Yes	Yes
Watts Bar 1	Yes	Yes
Wolf Creek	Yes	Yes

* Formerly known as Washington Nuclear Project No. 2 (WNP-2).

Table 5.2: Probable maximum precipitation data provided in IPEEE submittals*

Plant name	Rain fall (inches / hours)		Flood protection elevation (feet above mean sea level)	
	Design basis	Revised	Protection to	Maximum flood
Arkansas 1 & 2		19 / 1	369	358
Beaver Valley 1 & 2			732	730
Braidwood 1 & 2				571
Browns Ferry 2 & 3			578	572.5
Brunswick			<22.0[1]	25.6
Byron			871[2]	870.5
Callaway			840	839.87
Calvert Cliffs 1 & 2[3]				
Catawba	5 / 1	30 / 6		
Clinton			~714[4]	714
Columbia**	11.7 / 6	9.2 / 6	441	433.3
Comanche Peak	6 / 1	2 / 5 min	810	794.7
Cooper	9.7 / 1	18.2 / 1	906.5	< 906.5
Crystal River 3		19.4 / 1	129[5]	127[5]
D.C. Cook 1 & 2				
Davis-Besse		17.5 / 24	585	579
Diablo Canyon	4 / 1		48 ft mean lower low water level[6]	
Dresden 2 & 3				
Duane Arnold				
Farley			158	144
Fermi 2	10.2 / 1	17.3 / 1	586.9	586.9

Table 5.2: Probable maximum precipitation data provided in IPEEE submittals (Continued)

Plant name	Rain fall (inches / hours)		Flood protection elevation (feet above mean sea level)	
	Design basis	Revised	Protection to	Maximum flood
FitzPatrick	4 / 1	16 / 1	10 ft above Lake Ontario's maximum probable flood level	
Fort Calhoun		18 / 1	15 ft above grade[7]	25 ft above grade
Ginna		16.5 / 1		
Grand Gulf	16.4 / 1	28.2 / 1	133	116
Hatch 1 & 2	> 24.8 / N.S.	24.8 / N.S.	110[8]	113
Hope Creek			35.4[9]	35.4
Indian Point 2 & 3			15.5	14
Kewaunee[10]			605	596
LaSalle 1 & 2	32 / 24	No change		
Limerick 1 & 2			217	216.4[11]
Millstone 2		17.3 / 1	28	25.1
Millstone 3				
Monticello	7.7 (static depth)	9 / 0.25[12]	939.2	939.2
McGuire 1 & 2				
Nine Mile Point 1		29.8 / 1	<261.75[13]	261.75
Nine Mile Point 2				262.5
North Anna 1 & 2	14.5 / 1	18.6 / 1[14]	271	267
Oconee 1, 2, & 3		26.6 / 48		
Oyster Creek				

Table 5.2: Probable maximum precipitation data provided in IPEEE submittals (Continued)

Plant name	Rain fall (inches / hours)		Flood protection elevation (feet above mean sea level)	
	Design basis	Revised	Protection to	Maximum flood
Palisades	7.7 (static depth)	7.7 (static depth with drainage)	594.7	593.5
Palo Verde 1, 2, & 3			957.5, 960.5, 963.5	
Peach Bottom 2 & 3			135	133.8
Perry		17.1 / 1	620	608
Pilgrim			23	14.7+[15]
Point Beach 1 & 2		6.17 / 24	588.2[16]	596[16]
Prairie Island 1 & 2		17 / 1	706.7[17]	706.7
Quad Cities 1 & 2	4 / 1	3 / 1		8 ft above plant grade
River Bend				
Robinson 2	30 / 6			3 ft below plant grade
Salem 1 & 2				
San Onofre 2 & 3	>>7 (static depth)[18]	7 (static depth)[18]		
Seabrook		16.2 / 1		
Sequoyah 1 & 2			725.4[19]	725.4
Shearon Harris[20]				
South Texas 1 & 2[21]				
St Lucie 1 & 2[22]	6 / 1	24.1 / 24	19.5	17.2
Summer				
Surry 1 & 2		18.6 / 1		
Susquehanna 1 & 2			670	548

Table 5.2: Probable maximum precipitation data provided in IPEEE submittals (Continued)

Plant name	Rain fall (inches / hours)		Flood protection elevation (feet above mean sea level)	
	Design basis	Revised	Protection to	Maximum flood
Three Mile Island 1			310	310
Turkey Point 3 & 4	6 (static depth)[23]	6 (static depth)[23]	18[23]	18
Vermont Yankee		16.4 / 1	254	254
Vogtle 1 & 2	18 (static depth)	15 / 1	220	165
Waterford 3			29.25	
Watts Bar 1			728	740[24]
Wolf Creek[25]				

* A blank entry means that the information was not specified in the IPEEE submittal.
N.S. - Not Specified.

** Formerly known as Washington Nuclear Project No. 2 (WNP-2)

1 Brunswick: the recurrence frequency for this flood level is 1667 years.

2 Byron: plant grade is 870 ft mean sea level (MSL) but there is a 1-ft-high curb at building entrances.

3 Calvert Cliffs: roof drains are inspected monthly by two groups of engineers.

4 Clinton: "approximately the same level up to which the plant's safety-related equipment is protected by waterproofing."

5 Crystal River 3: mean low water level.

6 Diablo Canyon: this is the ventilation intake for the intake pump house, dam failures could cause higher water levels for short durations.

7 Fort Calhoun: it take 3.9 days after the failure of the Oahe earthen dam for the water to crest at the plant. Thereafter core damage is assumed by the licensee. At issuance of plant operating license, design basis flood was 2 ft MSL with the plant grade at 9.5 ft MSL. Army Corps of Engineers has increased the anticipated flood level to 15 ft MSL with a 1000 recurrence frequency. Licensee has sealed potential flood water entry locations against the 15 ft MSL flood.

8 Hatch: the flood level is 110 ft MSL with wave runup to 113 ft MSL. The licensee stated that sump pumps inside the intake structure are expected to prevent leakage around the doors from flooding safety-related components.

9 Hope Creek: there are doors and hatches below this elevation that require administrative action to be closed before the river flooding reaches an elevation "at which important systems could be compromised."

10 Kewaunee: the roof is designed for 7.7 inches of standing water while the maximum water depth (because of a ledge) is 3 inches.

11 Limerick: this is from the revised PMP; the Schuykill River water level is far below plant grade.

Table 5.2: Probable maximum precipitation data provided in IPEEE submittals (Continued)

12 Monticello: licensee reviewed the structures and determined they could withstand the ponding from the PMP; the 1000 year flood level is 921 ft MSL; above 930 ft MSL the licensee must employ other measures, e.g., sand bags and portable pumps.

13 Nine Mile Point 1: after IPEEE review, the licensee made a critical electrical termination below 261.75 ft MSL, which would result in loss of offsite power and diesel generator failure. The licensee estimated the core damage frequency (CDF) associated with a PMP event to be 6E-7 per reactor-year. In the SER, the staff considered this estimate to be overly optimistic and estimated the CDF to be 3E-6 per reactor-year and concluded that this was not a severe accident vulnerability.

14 North Anna: roof ponding is a potential problem for the turbine building. Further review will determine if scuppers need to be cut into the parapets to relieve roof loading.

15 Pilgrim: this does not include wave runup. The probability of this flood was determined to be less than 10^{-6}/ry.

16 Point Beach: sand bags are used to provide protection above grade level (588.2); elevations are related to International Great Lakes Datum (IGLD) and not MSL. Probabilistically, the floods are as follows: 588.2 IGLD is 3.69×10^{-2}/ry, 593.1 IGLD is 2.53×10^{-4}/ry, and 596 IGLD is 2.8×10^{-6}/ry.

17 Prairie Island: below this elevation, openings are protected by use of stop logs.

18 San Onofre: the site flooding was determined to be worse from thunderstorms than from frontal storm flood levels; thus, the PMP was determined from thunderstorms.

19 Sequoyah: penetrations below this elevation have water-tight seals or the equipment not protected to this elevation is designed for submerged operation.

20 Shearon Harris: declared the plant in compliance by virtue of meeting the 1975 SRP guidance, as identified in NUREG-1407.

21 South Texas: "A new PMP evaluation ... was not required because the impact of the new PMP criteria has been evaluated previously as part of the operating license (OL) process in 1989."

22 St Lucie: no roof ponding in excess of 2 inches is possible, except for the shield building, which has an 18-inch parapet.

23 Turkey Point: maximum water accumulation on the roof is 6 inches; above 6 inches the water spills off the roof. Greater precipitation rates only decrease the time until water spills off the roof. A dike protects the plant to 18 feet. The PMF with storm surge will top the wall and inundate the critical safety-related equipment (emergency switchgear); the PMF frequency is estimated to have an upper bound of 10^{-4}/ry and a lower bound of 10^{-6}/ry.

24 Watts Bar: the plant is required to shut down if flood levels exceed 728 feet. No other information was provided in the SER or TER.

25 Wolf Creek: is protected from flooding by dikes, drainage, and site grading.

Table 5.3: Plant improvements

Plant name	Improvements completed or to be completed	Improvements under review
Arkansas 1	None identified	Scuppers may be needed to prevent overloading roofs of buildings with safety-related equipment.
Dresden 2 & 3	Installed roof scuppers in reactor and turbine buildings and the crib house	None identified
Ginna	Modify control building roof or add additional scupper to limit roof ponding	None identified
Millstone 2	Installed roof scuppers, confirmed service water pumps protected by walkdown, time needed to close flood gates determined to be adequate	Functionality of the backwater valves
North Anna	None identified	turbine building roof loading; if necessary, scuppers will be cut into the parapets by 12/31/99
Salem 1 & 2	Installed penetration seal between buildings	None identified
Surry 1 & 2	(1) turbine building parapet being modified to prevent roof failure. (2) An active design change program is addressing the causes of water intrusion into the buildings.	None identified
Three Mile Island	Develop flood mitigating guidelines in event of severe Susquehanna River flooding; "flood-related plant improvements to mitigate the consequences of" floods >310 ft MSL	None identified

Table 5.4: GSI-147, Fire-induced alternate shutdown/control room panel interactions

Plant name	Comments
Arkansas 1 & 2	Does not have remote shutdown panels; shut down is accomplished by local operator actions using equipment isolated from the control room; seeks to maintain availability of both shutdown trains
Beaver Valley 1 & 2	
Braidwood 1 & 2	
Browns Ferry 2 & 3	
Brunswick 1 & 2	
Byron 1 & 2	
Callaway	
Calvert Cliffs 1 & 2	Procedure requires shedding and manual restart of most electrical loads -- if manual restart is unsuccessful, a SISBO will result; alternate shutdown is credited only for MCR abandonment, which also includes large turbine building fires and various fires in the yard
Catawba 1 & 2	Independent standby shutdown system in plant yard
Clinton	Capability for safe shutdown using either Division 1 or Division 2 equipment; annual training to shut down from the remote shutdown panel
Columbia Generating*	Remote and alternate remote shutdown panels
Comanche Peak	FIVE guidance used
Cooper	
Crystal River 3	
D. C. Cook 1 & 2	FIVE guidance used
Davis-Besse	
Diablo Canyon 1 & 2	FIVE guidance used
Dresden 2 & 3	
Duane Arnold	The evaluation of hot shorts, as documented in the submittal, is not as robust as at other facilities and may represent a weakness in the plant's protection
Farley 1	

Table 5.4: GSI-147, Fire-induced alternate shutdown/control room panel interactions (Continued)

Plant name	Comments
Fermi 2	Black-start combustion turbine-generator set provides power to the remote shutdown panel and equipment in the event of a loss of offsite power.
FitzPatrick	Independent safe shutdown facility with required diesels locally isolated and controlled
Fort Calhoun	FIVE guidance used
Ginna	Eight staff members are required for alternate shutdown; two primary shutdown stations manned continuously, six support stations manned as needed, and several valve locations manned for short period for manual operation
Grand Gulf	
Hatch 1 & 2	The submittal did not contain adequate information related to spurious signal/hot short issue, which could compromise recovery.
Hope Creek	Independent safe shutdown facility and a remote shutdown system
Indian Point 2	FIVE guidance used
Indian Point 3	Appendix R diesel generator and separate 480 volt switchgear reduces risk
Kewaunee	FIVE guidance used
LaSalle	
Limerick 1 & 2	FIVE guidance used
Millstone 2	Self-induced station blackout (SISBO); core uncovery ~1.5 hours; to get power from Unit 1 requires ~ 2 hours; operator actions to regain control of plant take 2 to 2.5 hours, which is less than the required 3 hours; probability that operators will *not* follow procedures and initiate SISBO if ECCS starts is 0.5; only the hot short causing a PORV to open was considered
Millstone 3	
Monticello	
McGuire 1 & 2	FIVE guidance used — these issues are assumed to have been considered by licensee
Nine Mile Point 1	Emergency condensers automatically initiate themselves (even in control room fires) and need no attention for 1-2 hours
Nine Mile Point 2	FIVE guidance used

Table 5.4: GSI-147, Fire-induced alternate shutdown/control room panel interactions (Continued)

Plant name	Comments
North Anna 1 & 2	The submittal did not address potential for spurious actuation causing component damage which could compromise recovery. FIVE guidance used; addressed spurious actuation that might lead to a LOCA or ISLOCA.
Oconee 1, 2, & 3	Safe shutdown facility is physically and electrically independent
Oyster Creek	No credit taken for alternate shutdown panel as the control room fire CDF was conservatively estimated to be 3.3E-7 per ry
Palisades	Spurious actuation (hot shorts) was not addressed
Palo Verde 1, 2, & 3	Alternate shutdown capability is in the Train B cable spreading room
Peach Bottom 2 & 3	
Perry	PRA also includes potential for total loss of shutdown system functions even with successful transfer of control to remote shutdown panel
Pilgrim	Necessary to control the plant from the 14 alternate shutdown panels
Point Beach 1 & 2	FIVE guidance used
Prairie Island 1 & 2	FIVE guidance used; CDF from failure to shutdown outside the control room was 5.3E-6/ry (~11% of total CDF); if shutdown from outside control room had a failure probability of 1.0, fire CDF would increase by a factor of 3
Quad Cities 1 & 2	FIVE guidance used
River Bend	
Robinson 2	FIVE guidance used
Salem 1 & 2	
San Onofre 2 & 3	FIVE guidance used
Seabrook	"None of the four specific areas [electrical independence, prevention of loss of control and/or power, prevention of spurious signals/hot short, or prevention of the total loss of system function] ... are addressed in the submittal."
Sequoyah 1 & 2	
Shearon Harris	"Generally immune to the effects of control system interactions"
South Texas 1 & 2	
St. Lucie 1 & 2	FIVE guidance used

Table 5.4: GSI-147, Fire-induced alternate shutdown/control room panel interactions (Continued)

Plant name	Comments
Summer	
Surry 1 & 2	In addition to the alternate shutdown panel (ASP), remote monitoring panels in the Unit 1 cable spreading room monitors shutdown of both units independent of normal instrument circuits; circuits were modified to (1) ensure DG & ASP isolation from control room and (2) greatly reduce likelihood of spurious PORV actuations
Susquehanna 1-2	FIVE guidance used
Three Mile Island 1	
Turkey Point 3 & 4	FIVE guidance used
Vermont Yankee	
Vogtle 1 & 2	
Waterford 3	
Watts Bar 1	Auxiliary control room and shutdown boards are located in the auxiliary building
Wolf Creek	

* Formerly known as Washington Nuclear Project No. 2 (WNP-2)

Table 5.5: GSI-148, Smoke control and manual fire-fighting effectiveness

Plant name	Manual fire-fighting	Effectiveness	Smoke control	Comments
Arkansas	Not credited	FIVE guidance used	FIVE guidance used	FIVE guidance used
Beaver Valley	Credited	Addressed	Discussion limited to training/procedures for fires in each fire area	Results of actual training exercised provided, including timing
Braidwood	Credited	Addressed	Portable ventilation used	FIVE guidance used
Browns Ferry 2 & 3	Credited only for the control bay, control room, & cable spreading room	Addressed	Not addressed except for identification of a smoke ejector for the control room	Effects of misdirected manual fire suppression activities were not addressed
Brunswick 1&2	Credited only for control room	Addressed	Not addressed	
Byron 1 & 2	Credited	Addressed	Portable ventilation used	FIVE guidance used
Callaway	Credited	FIVE guidance used	Not addressed	
Calvert Cliffs	Credited only for control room	FPRAIG guidance used	Addressed	Ability of operators to perform actions in control room were probabilistically evaluated to determine degradation due to fire impacts
Catawba	Not specified	Addressed	Addressed	
Clinton	Credited only for control room	Addressed	Addressed	

Table 5.5: GSI-148, Smoke control and manual fire-fighting effectiveness (Continued)

Plant name	Manual fire-fighting	Effectiveness	Smoke control	Comments
Columbia Generating Station*	Not credited	FIVE guidance used	Procedures in place to use ventilation (including portable) for smoke removal	Fire brigade trained in smoke removal and potential for misdirected suppression
Comanche Peak	Not credited	Addressed	Addressed	
Cooper	Credited only in the control room and the non-essential switchgear room	Addressed	Addressed	Training under live smoke conditions, including use of self-contained breathing apparatus (SCBA) equipment and ventilation techniques
Crystal River 3	Credited	Addressed	Addressed	
D.C. Cook	Credited	Addressed	Addressed	
Davis-Besse	Credited only for the control room and for transient fires	Addressed	Addressed	FIVE guidance used; negative effects of misdirected manual fire suppression efforts not discussed
Diablo Canyon 1 & 2	Credited only for control room fires	FIVE guidance used	FIVE guidance used	Only uses wet-pipe fire protection system
Dresden 2 & 3	Credited only for control room fires	Addressed	Addressed	
Duane Arnold	Credited only for control room fires	FIVE guidance used	Not addressed	Effects of smoke-induced misdirected manual fire suppression activities were not addressed

Table 5.5: GSI-148, Smoke control and manual fire-fighting effectiveness (Continued)

Plant name	Manual fire-fighting	Effectiveness	Smoke control	Comments
Farley	Credited only for control room fires	FIVE guidance used	Discussion limited to use of SCBA equipment	Credit for manual fire fighting not identified in other plant areas; no discussion on impacts of fire or suppression activities on other equipment
Fermi 2	"Generally" not credited	Addressed	Addressed	Not addressed is the potential for fire suppression activities to adversely impact other equipment
FitzPatrick	Credited only for the control bay, control room, and cable spreading room	FIVE guidance used	Addressed	Manual suppression times estimates usually longer than damage times in discussed scenarios
Fort Calhoun	Credited only for control room fires	FIVE guidance used	FIVE guidance used	
Ginna	Credited	Addressed	Addressed	Average time from fire alarm until drill was over was 13 minutes
Grand Gulf	Credited	Addressed	Addressed	Training with actual fire events with smoke control in configurations representative of plant conditions, training in potential toxic and corrosive characteristics of combustion products
Hatch 1 & 2	Credited	Addressed	Addressed	Not addressed is the potential for fire suppression activities to adversely impact other equipment; fire non-suppression factor used in PRA included detection, brigade notification and response, and fire control times

Table 5.5: GSI-148, Smoke control and manual fire-fighting effectiveness (Continued)

Plant name	Manual fire-fighting	Effectiveness	Smoke control	Comments
Hope Creek	Not credited	Addressed	Addressed	Drills use live smoke (SCBA)
Indian Point 2	Credited	Addressed	Smoke impact considerations and recovery probability modified	Effects of smoke-induced misdirected manual fire suppression activities were not addressed
Indian Point 3	"Minimally" credited	FIVE guidance used	FIVE guidance used	FIVE guidance used; effects of smoke-induced misdirected manual fire suppression activities were not addressed
Kewaunee	Not credited	Not addressed	Not addressed	Effects of smoke-induced misdirected manual fire suppression activities were not addressed
LaSalle 1 & 2	Credited	Not addressed	Potential negative effects of smoke not addressed	No SCBA equipment, no discussion on response time, training, potential for misdirected suppression causing damage, or of any fire pre-plans
Limerick 1&2	Credited in some fire scenarios	Partially addressed	Not addressed	Effects of smoke-induced misdirected manual fire suppression activities and manual fire-fighting effectiveness were not addressed
Millstone 2	Credited	Addressed	Closed	Weakness: treatment in the IPEEE submittal of the effects of smoke-induced misdirected manual fire suppression activities; strength: extensive fire brigade training, including contractor-operated simulator and unannounced fire drills

Table 5.5: GSI-148, Smoke control and manual fire-fighting effectiveness (Continued)

Plant name	Manual fire-fighting	Effectiveness	Smoke control	Comments
Millstone 3	Closed	Closed	Closed	
Monticello	Credited only for control room fires	Addressed	Addressed	FIVE guidance used; except in the control room, fire was assumed to damage all cables and equipment before suppression
McGuire 1&2	Not specified	Addressed by walkdown	Addressed by walkdown	
Nine Mile Point 1	Not specified	Addressed	Addressed	Effects of smoke-induced misdirected manual fire suppression activities were not addressed; discussion of fire identification (including potentially affected equipment), available equipment, procedures, training (including SCBA gear with live smoke), smoke removal, drills, and records
Nine Mile Point 2	Credited only for control room fires	Addressed	Addressed	
North Anna	Credited	Addressed (FIVE)	Addressed	FIVE guidance used; effects of smoke-induced misdirected manual fire suppression activities were not addressed; *operator* effectiveness in a smoke-filled environment was not adequately addressed
Oconee 1, 2, & 3	Credited	Addressed	Addressed	Effects of smoke-induced misdirected manual fire suppression activities were not addressed
Oyster Creek	Not credited	Addressed	Partial	Effects of smoke-induced misdirected manual fire suppression activities were not addressed

Table 5.5: GSI-148, Smoke control and manual fire-fighting effectiveness (Continued)

Plant name	Manual fire-fighting	Effectiveness	Smoke control	Comments
Palisades	Credited only for fires in control room, cable spreading room, and two vital switchgear rooms	Addressed	Addressed	
Palo Verde	Not credited	Addressed	Addressed	
Peach Bottom 2 & 3	Credited for several fire areas	FIVE guidance used; 30 minutes to control a fire	FIVE guidance used	
Perry	Credited if time to damage greater than detection/suppression time	Addressed	Addressed	
Pilgrim	Credited	Addressed	Addressed	
Point Beach	Credited	Addressed	Addressed	
Prairie Island	Credited only for control room, relay room, and auxiliary feedwater pump room fires	Addressed	Addressed	
Quad Cities	Not credited	FIVE guidance used	FIVE guidance used	Negative effects of misdirected manual fire suppression efforts not discussed
River Bend				

Table 5.5: GSI-148, Smoke control and manual fire-fighting effectiveness (Continued)

Plant name	Manual fire-fighting	Effectiveness	Smoke control	Comments
Robinson 2	Credited	FIVE guidance used	FIVE guidance used	Thorough discussion of available equipment, procedures, communication, training (including SCBA gear), drills, and records
Salem 1 & 2	Not specified	FIVE guidance used	FIVE guidance used	Dedicated fire department for Salem
San Onofre 2 & 3	Considered in the submittal but not credited in any of the fire compartments assessed	FIVE guidance used	FIVE guidance used	
Seabrook 1 & 2	Credited on a case-by-case basis	FIVE guidance used	Open	Negative effects of misdirected manual fire suppression efforts and potential of breach of fire barriers were not discussed
Sequoyah	Credited	Addressed	Addressed	
Shearon Harris	Credited for some fire scenarios	FIVE guidance used	FIVE guidance used	Negative effects of misdirected manual fire suppression efforts not discussed
South Texas	Closed	Closed	Closed	
St. Lucie 1 & 2	Not credited	Not addressed	Not addressed	
Summer	Not credited	FIVE guidance used	FIVE guidance used	
Surry 1 & 2	Not specified	FIVE guidance used	FIVE guidance used	
Susquehanna	Not credited	Closed	Closed	

Table 5.5: GSI-148, Smoke control and manual fire-fighting effectiveness (Continued)

Plant name	Manual fire-fighting	Effectiveness	Smoke control	Comments
Three Mile Island 1	Not credited	Addressed	Addressed	
Turkey Point	Not credited	Not addressed	Not addressed	
Vermont Yankee	Not specified	Addressed	Addressed	FIVE guidance used
Vogtle 1 & 2	Credited	Addressed	Addressed	Negative effects of misdirected manual fire suppression efforts and effects of compromised barriers during fire-fighting activities were not discussed
Waterford 3	Credited only for welding/cutting-initiated fire scenarios and one control room fire scenario	Plant fire brigade trained in accordance with plant training procedures, which meets Appendix R requirements. Fire brigade members are educated on toxic and corrosive characteristics of combustion products	Self-contained breathing apparatus available; impact of smoke on operator's ability to safely shut down the plant considered	Negative effects of misdirected manual fire suppression efforts not discussed
Watts Bar	Not specified	FIVE guidance used	Self-contained breathing apparatus and portable ventilation equipment provided at key plant locations	
Wolf Creek	Not specified	FIVE guidance used	FIVE guidance used	

* Formerly known as Washington Nuclear Project No. 2 (WNP-2).

Table 5.6: GSI-156, Systematic evaluation program

Plant name	Hydrology & withstand floods	Industrial hazards	Tornado missiles	Severe weather	Design codes, criteria, and loadings	Dam integrity, site flooding	Settlement & liquefaction	Seismic design	Shutdown I&C
Arkansas 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Beaver Valley 1 & 2	1975 SRP plant								
Braidwood 1 & 2	1975 SRP plant								
Browns Ferry 3	Yes	Yes	Yes	Yes	Yes	Yes	N.E.	Yes	Yes
Brunswick 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Byron 1 & 2	1975 SRP plant								
Callaway	1975 SRP plant								
Calvert Cliffs 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Catawba 1 & 2	1975 SRP plant								
Clinton	1975 SRP plant								
Columbia Generating*	1975 SRP plant								
Comanche Peak	1975 SRP plant								
Cooper	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Crystal River 3	1975 SRP plant								
D.C. Cook 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Davis-Besse	1975 SRP plant								
Diablo Canyon 1 & 2	1975 SRP plant								

Table 5.6: GSI-156, Systematic evaluation program (Continued)

Plant name	Hydrology & withstand floods	Industrial hazards	Tornado missiles	Severe weather	Design codes, criteria, and loadings	Dam integrity, site flooding	Settlement & liquefaction	Seismic design	Shutdown I&C
Dresden 2 & 3	Yes	Yes	Yes	Yes	Yes	Yes	N.A.	Yes	Yes
Duane Arnold	Yes	Yes	Yes	Yes	Yes	Yes	N.E.	Yes	Yes
Farley 1	1975 SRP plant								
Fermi 2	1975 SRP plant								
FitzPatrick	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Fort Calhoun	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Ginna	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Grand Gulf	1975 SRP plant								
Hatch 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Hope Creek	1975 SRP plant								
Indian Point 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Indian Point 3	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Kewaunee	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
LaSalle	1975 SRP plant								
Limerick 1 & 2	1975 SRP plant								
Millstone 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Millstone 3	1975 SRP plant								

Table 5.6: GSI-156, Systematic evaluation program (Continued)

Plant name	Hydrology & withstand floods	Industrial hazards	Tornado missiles	Severe weather	Design codes, criteria, and loadings	Dam integrity, site flooding	Settlement & liquefaction	Seismic design	Shutdown I&C
Monticello	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
McGuire 1 & 2	1975 SRP plant								
Nine Mile Point 1	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Nine Mile Point 2	1975 SRP plant								
North Anna 1 & 2	1975 SRP plant								
Oconee 1, 2, & 3	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Oyster Creek	Resolved by NUREG-0822 (January 1983) and NUREG-0822, Supplement 1 (July 1988)								
Palisades	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Palo Verde 1, 2, & 3	1975 SRP plant								
Peach Bottom 2 & 3	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Perry	1975 SRP plant								
Pilgrim	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Point Beach 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Prairie Island 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Quad Cities 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
River Bend	1975 SRP plant								
Robinson 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes

Table 5.6: GSI-156, Systematic evaluation program (Continued)

Plant name	Hydrology & withstand floods	Industrial hazards	Tornado missiles	Severe weather	Design codes, criteria, and loadings	Dam integrity, site flooding	Settlement & liquefaction	Seismic design	Shutdown I&C
Salem 1 & 2	1975 SRP plant								
San Onofre 2 & 3	1975 SRP plant								
Seabrook	1975 SRP plant								
Sequoyah 1 & 2	1975 SRP plant								
Shearon Harris	1975 SRP plant								
South Texas 1 & 2	1975 SRP plant								
St. Lucie 1 & 2	1975 SRP plant								
Summer	1975 SRP plant								
Surry 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Susquehanna 1 & 2	1975 SRP plant								
Three Mile Island 1	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Turkey Point 3 & 4	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Vermont Yankee	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Vogtle 1 & 2	1975 SRP plant								
Waterford 3	1975 SRP plant								
Watts Bar 1	1975 SRP plant								
Wolf Creek	1975 SRP plant								

Table 5.6: GSI-156, Systematic evaluation program (Continued)

N.A. Not Applicable; soil related issues do not apply to rock sites.
N.E. Not Evaluated; Generic Letter 88-20, Supplement 5, states that soil failures need not be evaluated by focused-scope plants.
* Formerly known as Washington Nuclear Project No. 2 (WNP-2).

Table 5.7: GSI-172, Multiple system responses program

Plant name	Effects of activation[1]	Seismic activation[2]	Seismic fires[3]	Hydrogen ruptures[4]	Systems' depend[5]	Flood and moisture[6]	Spatial interactions[7]	Seismic flooding[8]	Relay chatter[9]	Common cause[10]	Beyond SSE[11]
Arkansas 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Beaver Valley 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Braidwood 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Browns Ferry 3	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	No	Yes
Brunswick 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Byron 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Callaway	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Calvert Cliffs 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Catawba 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Clinton	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Columbia*	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Comanche Peak	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Cooper	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Crystal River 3	Yes	Yes	Yes	Yes	Yes	Yes	Yes	No	Yes	No	Yes
D.C. Cook 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Davis-Besse	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Diablo Canyon 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Dresden 2 & 3	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes

Table 5.7: GSI-172, Multiple system responses program (Continued)

Plant name	Effects of activation[1]	Seismic activation[2]	Seismic fires[3]	Hydrogen ruptures[4]	Systems' depend[5]	Flood and moisture[6]	Spatial interactions[7]	Seismic flooding[8]	Relay chatter[9]	Common cause[10]	Beyond SSE[11]
Duane Arnold	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Farley 1	Yes	Yes	Yes	Yes	Yes	No	Yes	Yes	Yes	No	N.E.
Fermi 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
FitzPatrick	Yes	Yes (not credible < 0.3 g)	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Fort Calhoun	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Ginna	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Grand Gulf	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Hatch 1 & 2	Yes	Yes	Partial	Partial	Partial	Yes	Yes	Yes	Yes	Yes	Yes
Hope Creek	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Indian Point 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Indian Point 3	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Kewaunee	Yes	Yes	Yes	No	Yes	Yes	Yes	No	Yes	Yes	Yes
LaSalle	No	Yes	No	No	No	Partial	No	No	Yes	Yes	Yes
Limerick 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Partial	Yes	Partial	Yes
Millstone 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Millstone 3	Yes	Yes	Yes[12]	Yes[12]	Yes	Yes	Yes[12]	Yes[12]	Yes	Yes	Yes
Monticello	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes

Table 5.7: GSI-172, Multiple system responses program (Continued)

Plant name	Effects of activation[1]	Seismic activation[2]	Seismic fires[3]	Hydrogen ruptures[4]	Systems' depend[5]	Flood and moisture[6]	Spatial interactions[7]	Seismic flooding[8]	Relay chatter[9]	Common cause[10]	Beyond SSE[11]
McGuire 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Nine Mile Point 1	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Nine Mile Point 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
North Anna 1 & 2	Yes	No	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Partial	Yes
Oconee 1, 2, & 3	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Oyster Creek	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Palisades	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Palo Verde 1, 2, & 3	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Peach Bottom 2 & 3	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Perry	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Pilgrim	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Point Beach 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Prairie Island 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Quad Cities 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
River Bend	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Robinson 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Salem 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
San Onofre 2 & 3	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes

Table 5.7: GSI-172, Multiple system responses program (Continued)

Plant name	Effects of activation[1]	Seismic activation[2]	Seismic fires[3]	Hydrogen ruptures[4]	Systems' depend[5]	Flood and moisture[6]	Spatial interactions[7]	Seismic flooding[8]	Relay chatter[9]	Common cause[10]	Beyond SSE[11]
Seabrook	Yes	Yes	Partial	Partial	No	Yes	Yes	Yes	Yes	Yes	Yes
Sequoyah 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Shearon Harris	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
South Texas 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
St. Lucie 1 & 2	No	No	Yes	Yes	Yes	Yes	No	No	Yes	No	N.E.
Summer	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Surry 1 & 2	Yes	Yes	Yes	Yes**	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Susquehanna 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Three Mile Island 1	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Turkey Point 3 & 4	Yes	No	Yes	Yes	Yes	Yes	Yes	No	Yes	No	Yes
Vermont Yankee	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Vogtle 1 & 2	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	No	Yes
Waterford 3	Yes	Yes	Yes	No	Yes	Yes	Yes	Yes	N.A.	Yes	N.A.
Watts Bar 1	Yes	Yes	Yes	Yes	No	No	Yes	Yes	Yes	Yes	Yes
Wolf Creek	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes

* Formerly known as Washington Nuclear Project No. 2 (WNP-2).
** Surry: two issues (excessive flexibility of stainless steel tubes carrying hydrogen and poor restraint of hydrogen cylinders) are identified for future resolution.
N.E. No evaluation needed for reduced-scope plants (NUREG-1407, Section 3.2.4.5)

Table 5.7: GSI-172, Multiple system responses program (Continued)

[1] Effects of fire protection system actuation on non-safety related and safety-related equipment.
[2] Seismically induced for suppression system actuations.
[3] Seismically induced fires.
[4] Effects of hydrogen line ruptures.
[5] Non-safety related control system/safety-related system dependencies.
[6] Effects of flooding and/or moisture intrusion on non-safety related and safety-related equipment.
[7] Seismically induced spatial and functional interactions.
[8] Seismically induced flooding.
[9] Seismically induced relay chatter.
[10] IPEEE-related aspects of common cause failures related to human actions.
[11] Evaluation of earthquake magnitudes greater than the safe shutdown earthquake.
[12] Millstone 3: resolved by walkdown based on composition of walkdown team and expertise of licensee's independent reviewer.

Table 5.8: Sandia fire risk scoping study issues

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Arkansas 1 & 2	Yes	Yes	Yes	Yes	Yes	FIVE guidance used
Beaver Valley 1 & 2	Yes	Yes	Yes	Yes	Yes	
Braidwood 1 & 2	Yes (FIVE)	Yes (FIVE)	Yes (FIVE)	Yes	Yes	FIVE guidance used, seismic/fire walkdowns, fire sources (flammable storage cabinets and oil drums) were identified but acceptable because no ignition or target identified
Browns Ferry 2 & 3	Yes	Yes (FIVE)	Yes	No (FIVE)	Yes	Assumed effects of smoke bounded by fire effects; effects of fire suppressants on equipment not addressed; operator effectiveness addressed by procedures, training, & equipment (SCBA)
Brunswick 1 & 2	Yes	Yes (FIVE)	Yes	Yes	Yes	Seismic walkdowns, addressed effects of combustion products

Table 5.8: Sandia fire risk scoping study issues (Continued)

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Byron 1 & 2	Yes	Yes	Yes	Yes	Yes	FIVE guidance used, seismic/fire walkdowns, fire sources (flammable storage cabinets, gas bottles, oil drums, switchgear) were identified but acceptable because no target identified, non-rated barrier failures considered in multi-compartment analysis
Callaway	Yes	Yes	Yes	Yes	Yes	Seismic walkdown, many new procedures implemented
Calvert Cliffs 1 & 2	Yes	Yes	Yes, FPRAIG guidance used	Yes	Yes	Seismic walkdowns performed; past inadvertent activations did not affect equipment operability; modifications made (install conduit and cabinet seals, holes in junction box bottoms, and water shields) as part of GSI-57 before IPEEE and credited in IPEEE

Table 5.8: Sandia fire risk scoping study issues (Continued)

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Catawba 1 & 2	Yes	Yes	Yes	Yes	Yes	Independent standby shutdown system in plant yard
Clinton	Yes	Yes	Yes	Yes	Yes	Seismic walkdown; Thermo-Lag (to be modified or replaced during next two refueling outages); seismically designed to same quality as other systems in the same area.
Columbia Generating Station*	Yes	Yes (NFPA)	Yes	Yes	Yes	Seismic & fire walkdowns, meets inspection standards of NFPA 80 & 90A
Comanche Peak	Yes	Yes	Yes	Yes	Yes	Seismic walkdowns
Cooper	Yes	Yes	Yes	Yes	Yes	Fire protection systems installed in accordance with NFPA standards
Crystal River 3	Yes	Yes	Yes	Yes	Yes	
D.C. Cook 1 & 2	Yes	Yes	Yes	Yes	Yes	Remote shutdown through local shutdown indication panels located in several spots in the auxiliary building

Table 5.8: Sandia fire risk scoping study issues (Continued)

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Davis-Besse	Yes	Yes	Yes	Yes	Yes	Effects of smoke-induced misdirected manual fire suppression activities were not addressed; proposed action to correct seismic mounting of two small flammable compressed gas bottles
Diablo Canyon 1 & 2	Yes	Yes	Yes	Yes	Yes	Not addressed: combustion products effects on equipment, relays freezing from CO_2, and inadvertent discharge
Dresden 2 & 3	Yes	Yes	Yes	Yes	Yes	Seismic walkdowns; multi-compartment fires evaluated; fire barriers and 10% of the penetration seals are inspected every 18 months
Duane Arnold	Yes	Yes	No	Yes	Yes	No credit is taken for manual fire fighting (except in the control room)
Farley 1	Yes	Yes (FIVE)	Yes (FIVE)	Yes	Yes	

Table 5.8: Sandia fire risk scoping study issues (Continued)

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Fermi 2	Yes	Yes	No	Yes	Yes	Effects of smoke-induced misdirected manual fire suppression activities were not addressed
FitzPatrick	Yes	Yes	Yes (FIVE)	Yes	Yes	Remote shutdown is from four panels in different parts of the plant; instrumentation is on a shutdown panel and several instrumentation racks; training is with live-fire and smoke environment gear; FIVE guidance used
Fort Calhoun	Yes	Yes	Yes	Yes	Yes	Explicitly considered survival from combustion products, fire suppression actuation, operator actions, flames, and hot gas layer; probability of barrier failure included in fire PRA

Table 5.8: Sandia fire risk scoping study issues (Continued)

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Ginna	Yes	Yes	Yes	Yes	Yes	Fire wrap on charging pump A provides "significant" (unspecified) risk reduction - "apparently" not Thermo-Lag; no failures of fire doors, fire dampers, or penetration seals that were not promptly detected during daily plant tours
Grand Gulf	Yes	Yes	Yes	Yes	Yes	Thermo-Lag being upgraded to ensure hourly fire endurance rating
Hatch 1 & 2	Partial	Partial	Partial	Yes	Partial	Inadequate information to resolve issues (left to right): anchorage of electrical cabinets not on safe shutdown equipment list, failure of active barriers, effects of misdirected spray, and hot shorts
Hope Creek	Yes	Yes	Yes	Yes	Yes	Inspection procedures include fire barrier penetration seals
Indian Point 2	Yes	Yes	No	Yes	Yes	FIVE guidance used

Table 5.8: Sandia fire risk scoping study issues (Continued)

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Indian Point 3	Yes	Yes	Partial	Yes	Yes	FIVE guidance used; effects of smoke-induced misdirected manual fire suppression activities were not addressed
Kewaunee	Yes	Yes	No	Yes	Yes	Effects of smoke-induced misdirected manual fire suppression activities were not addressed
LaSalle 1 & 2	No	No	No	No	Yes	Thermo-Lag is cited by name
Limerick 1 & 2	Yes	Yes	Yes	Yes	Yes	FIVE guidance used
Millstone 2	Yes	Yes	Yes	Yes	Yes	FIVE guidance used; Thermo-Lag is cited by name; separate walkdowns to address FRSS issues; 27% of events result in damaged equipment from water spray; weaknesses: diesel fire pump fuel tank could tip in a seismic event, block walls in fire water pump houses, support for a long run of fire water pipe

Table 5.8: Sandia fire risk scoping study issues (Continued)

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Millstone 3	Yes	Yes	Yes	Yes	Yes	Walkdown performed in 1985 by licensee & independent reviewer
Monticello	Yes	Yes	Yes	Yes	Yes	FIVE guidance and seismic walkdowns used; fire barrier inspection procedures; brigade training includes drills and communications; survivability includes effects of combustion products
McGuire 1 & 2	Yes	Yes	Yes	Yes	Yes	Standby shutdown system mitigates adverse effects of failure of redundant trains caused by breach of fire barriers

Table 5.8: Sandia fire risk scoping study issues (Continued)

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Nine Mile Point 1	Yes	Yes	No	Yes	Yes	Effects of smoke-induced misdirected manual fire suppression activities were not addressed; walkdowns reviewed fire protection systems for compliance with NFPA codes and standards; all Appendix R doors are inspected daily and all seals were inspected in 1989-1990; all barriers in inspection and maintenance program
Nine Mile Point 2	Yes	Yes	Yes	Yes	Yes	FIVE guidance used
North Anna 1 & 2	Yes	Yes	Yes	Yes	Yes	FIVE guidance used
Oconee 1, 2, & 3	Yes	Yes	No	Yes	Yes	Effects of smoke-induced misdirected manual fire suppression activities were not addressed
Oyster Creek	Yes	Yes	No	Yes	Yes	Effects of smoke-induced misdirected manual fire suppression activities were not addressed

Table 5.8: Sandia fire risk scoping study issues (Continued)

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Palisades	Yes	Yes	Yes	Yes	Yes	
Palo Verde 1, 2, 3	Yes	Yes	Yes	Yes	Yes	FIVE and fire compartment interaction analysis (FCIA) guidance used
Peach Bottom 2 & 3	Yes	Yes	Yes	Yes	Yes	Seismic walkdown performed; no adverse impact of mercury relay switches on safety systems; weakness: CO_2 system's tank anchorage & batteries; improvements to be completed by 12/31/00
Perry	Yes	Yes	Yes	Yes	Yes	Seismic walkdown performed; fire barrier inspection frequencies identified
Pilgrim	Yes	Yes	Yes	Yes	Yes	FIVE guidance used; fire protection systems installed in accordance with NFPA codes and standards
Point Beach 1 & 2	Yes	Yes	Yes	Yes	Yes	FIVE guidance used; seismic and fire walkdowns performed

Table 5.8: Sandia fire risk scoping study issues (Continued)

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Prairie Island 1 & 2	Yes	Yes	Yes	Yes	Yes	FIVE guidance used
Quad Cities 1 & 2	Yes	Yes	Yes	Yes	Yes	FIVE guidance used
River Bend						
Robinson 2	Yes	Yes	Yes	Yes	Yes	FIVE guidance used; seismic and fire walkdowns performed
Salem 1 & 2	Yes	Yes	Yes	Yes	Yes	FIVE guidance used; seismic and fire walkdowns performed
San Onofre 2 & 3	Yes	Yes	Yes	Yes	Yes	Seismic walkdown, seismic Category II/I analysis performed; several analyses conducted on effects of fire suppression system actuation on equipment

Table 5.8: Sandia fire risk scoping study issues (Continued)

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Seabrook	Yes	Partial	Yes	Yes	No	FIVE guidance used for *some* issues; the submittal did not address manual fire-fighting breaching fire barriers, failure of fire barriers, alternate shutdown electrical independence, loss of control or power before transfer, spurious actuations, and total loss of system function.
Sequoyah 1 & 2	Yes	Yes	Yes	Yes	Yes	FIVE guidance used.
Shearon Harris	Yes	Yes	No	Yes	Yes	FIVE guidance used; effects of smoke-induced misdirected manual fire suppression activities were not addressed
South Texas 1 & 2	Yes	Yes	Yes	Yes	Yes	
St. Lucie 1 & 2	No	Yes	No	No	Yes	Nothing in the TER indicates these issues should be open
Summer	Yes	Yes	Yes	Yes	Yes	FIVE guidance used

Table 5.8: Sandia fire risk scoping study issues (Continued)

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Surry 1 & 2	Yes	Yes	Yes	Yes	Yes	FIVE guidance used; identified items to be resolved are (1) hydrogen line modifications, (2) anchorage & storage issues for CO_2 and hydrogen cylinders, and (3) anchorage of oil drums and other storage containers
Susquehanna 1 & 2	Yes	Yes	Yes	Yes	Yes	Seismic and fire walkdowns; ~200 member fire brigade; strength: independent, black-start, portable diesel generator to support shutdown; weaknesses: (1) pumps in non-seismically designed building, (2) CO_2 tank not seismically supported, (3) batteries (to start diesel pump) had no spacers between cells or end stops on battery racks, and (4) small, unsupported metal cabinets are free to tip over

Table 5.8: Sandia fire risk scoping study issues (Continued)

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Three Mile Island 1	Yes	Yes	Yes	Yes	Yes	Seismic walkdown; a 0.1 factor was applied to account for damage to all equipment in a fire zone, including redundant equipment; Thermo-Lag installation is used
Turkey Point 3 & 4	Yes	Yes	No	No	Yes	All unescorted personnel must undergo fire watch training
Vermont Yankee	Yes	Yes	Yes	Yes	Yes	Seismic-fire interactions identified three weaknesses: two to be evaluated (H_2 piping & fuel oil tank supports) and one to be upgraded (anchorage of buses 1 and 2 to prevent sliding)

Table 5.8: Sandia fire risk scoping study issues (Continued)

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Vogtle 1 & 2	Yes	Yes	Partial	Yes	Yes	Effects of smoke-induced misdirected manual fire suppression activities and potential barrier breaches were not addressed; comments: seismic walkdown; curbing around equipment containing significant quantities of flammable liquids; fire protection systems installed to category II/I criteria; failure rates are 0.13/door-yr, 0.0011/damper-yr, 0.003/seal-yr, unavailability of door is estimated to be 5.94E-5
Waterford 3	Yes	Yes	Yes	Yes	Yes	Seismic and fire walkdowns; all penetration seals inspected in 1988
Watts Bar 1	Yes	Yes	Yes	Yes	Yes	FIVE guidance used; seismic walkdowns; fire protection systems installed to category II/I criteria

Table 5.8: Sandia fire risk scoping study issues (Continued)

Plant name	Seismic-fire interactions	Adequacy of fire barriers	Smoke control and manual fire-fighting effectiveness	Equipment survival in a fire-induced environment	Alternate shutdown-control room panel interactions	Comments
Wolf Creek	Yes	Yes	Yes	Yes	Yes	Fire walkdown; fire protection systems piping adequately supported against SSE in safety-related areas; emergency lighting installed; SCBA; training includes encountering toxic and corrosive combustion products; control room fire PRA included effects of smoke for abandonment; remote shutdown uses Train B

* Formerly known as Washington Nuclear Project No. 2 (WNP-2).

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10. SUPPLEMENTARY NOTES

Alan M. Rubin, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The NRC requested by Generic Letter 88-20, Supplement 4, and NUREG-1407, that each licensee perform an IPEEE to identify and report all plant-specific vulnerabilities to severe accidents caused by external events. The external events considered included seismic events; internal fires; and high winds, floods, and other external initiating events including transportation or nearby facility accidents and plant-unique hazards. All currently operating U.S. nuclear power plants have completed their assessments.

The objective of the NRC's IPEEE submittal reviews was to ascertain whether the licensees' IPEEE processes were capable of identifying severe accident vulnerabilities to such external events, and implementing cost-effective safety improvements to either eliminate or reduce the impact of those vulnerabilities. The reviews did not attempt to validate or verify the licensees' results.

The purpose of this report is to document the perspectives gleaned from the technical reviews of the IPEEE submittals. These include a description of the overall IPEEE process and findings; conclusions regarding the dominant risk contributors for the major areas of evaluation; an overview of plant improvements; a description of the overall strengths and weaknesses in the licensees' implementation of the IPEEE evaluation methodologies; and an assessment of the overall effectiveness in meeting the IPEEE objectives.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
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