UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

JUNE 2001

SUPPLEMENT 25 TO NUREG-0933
"A PRIORITIZATION OF GENERIC SAFETY ISSUES"

REVISION INSERTION INSTRUCTIONS

	Remove	Insert
Introduction:	pp. 29 to 68, Rev. 24	pp. 29 to 68, Rev. 25
	pp. 63 to 66, Rev. 10	pp. 69 to 72, Rev. 11
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Section 3:	pp. 3.71-1 to 8, Rev. 2	pp. 3.71-1 to 8, Rev. 3
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References:	pp. R-1 to R-120, Rev. 14	pp. R-1 to R-121, Rev. 15
Appendix B:	pp. A.B-1 to 13, Rev. 15	pp. A.B-1 to 13, Rev. 16
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	pp. A.F.7-1	pp. A.F.7-1, Rev. 1
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TABLE II

LISTING OF ALL TMI ACTION PLAN ITEMS, TASK ACTION PLAN ITEMS, NEW GENERIC ISSUES, HUMAN FACTORS ISSUES, AND CHERNOBYL ISSUES

This table contains the priority designations for all issues listed in this report. For those issues found to be covered in other issues described in this document, the appropriate notations have been made in the Safety Priority Ranking column, e.g., I.A.2.2 in the Safety Priority Ranking column means that Item I.A.2.6(3) is covered in Item I.A.2.2. For those issues found to be covered in programs not described in this document, the notation (S) was made in the Safety Priority Ranking column. For resolved issues that have resulted in new requirements for operating plants, the appropriate multiplant licensing action number is listed. The licensing action numbering system bears no relationship to the numbering systems used for identifying the prioritized issues. An explanation of the classification and status of the issues is provided in the legend below.

Legend

NOTES:	1 - Possible Resolution Identified for Evaluation
	2 - Resolution Available (Documented in NUREG, NRC Memorandum, SER, or equivalent)
	3 - Resolution Resulted in either: (a) The Establishment of New Regulatory Requirements (By Rule, SRP Change, or equivalent) or (b) No New Requirements
	4 - Issue to be Prioritized in the Future
	5 - Issue that is not a Generic Safety Issue but should be Assigned Resources for Completion
HIGH	- High Safety Priority
MEDIUM	- Medium Safety Priority
LOW	- Low Safety Priority
DROP	- Issue Dropped as a Generic Issue
EI	- Environmental Issue
I	- Resolved TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737
LI	- Licensing Issue
MPA	- Multiplant Action
NA	- Not Applicable
RI	- Regulatory Impact Issue
S	- Issue Covered in an NRC Program Outside the Scope of This Document
USI	- Unresolved Safety Issue

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	TMI ACTION PLAN ITEMS						
I.A	OPERATING PERSONNEL						
I.A.1	Operating Personnel and Staffing						
I.A.1.1	Shift Technical Advisor	-	NRR/DHFS/LQB	I	3	12/31/97	F-01
I.A.1.2	Shift Supervisor Administrative Duties	-	NRR/DHFS/LQB	I	3	12/31/97	
I.A.1.3	Shift Manning	-	NRR/DHFS/LQB	I	3	12/31/97	F-02
I.A.1.4	Long-Term Upgrading	Colmar	RES/DFO/HFBR	NOTE 3(a)	3	12/31/97	
I.A.2	Training and Qualifications of Operating Personnel						
I.A.2.1	Immediate Upgrading of Operator and Senior Operator Training and Qualifications	-	-	-			
I.A.2.1(1)	Qualifications - Experience	-	NRR/DHFS/LQB	I	6	12/31/97	F-03
I.A.2.1(2)	Training	-	NRR/DHFS/LQB	I	6	12/31/97	F-03
I.A.2.1(3)	Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	-	NRR/DHFS/LQB	I	6	12/31/97	F-03
I.A.2.2	Training and Qualifications of Operations Personnel	Colmar	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.2.3	Administration of Training Programs	-	NRR/DHFS/LQB	I	6	12/31/97	
I.A.2.4	NRR Participation in Inspector Training	Colmar	NRR/DHFS/LQB	LI (NOTE 3)	6	12/31/97	NA
I.A.2.5	Plant Drills	Colmar	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.2.6	Long-Term Upgrading of Training and Qualifications	-	-	-			
I.A.2.6(1)	Revise Regulatory Guide 1.8	Colmar	NRR/DHFT/HFIB	NOTE 3(a)	6	12/31/97	NA
I.A.2.6(2)	Staff Review of NRR 80-117	Colmar	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.2.6(3)	Revise 10 CFR 55	Colmar	NRR/DHFS/LQB	I.A.2.2	6	12/31/97	NA
I.A.2.6(4)	Operator Workshops	Colmar	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.2.6(5)	Develop Inspection Procedures for Training Program	Colmar	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.2.6(6)	Nuclear Power Fundamentals	Colmar	NRR/DHFS/LQB	DROP	6	12/31/97	NA
I.A.2.7	Accreditation of Training Institutions	Colmar	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.3	Licensing and Requalification of Operating Personnel						
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	Emrit	NRR/DHFS/LQB	I	6	12/31/97	
I.A.3.2	Operator Licensing Program Changes	Emrit	NRR/DHFS/OLB	NOTE 3(b)	6	12/31/97	NA
I.A.3.3	Requirements for Operator Fitness	Colmar	RES/DRAO/HFSB	NOTE 3(b)	6	12/31/97	NA
I.A.3.4	Licensing of Additional Operations Personnel	Thatcher	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.3.5	Establish Statement of Understanding with INPO and DOE	Thatcher	NRR/DHFS/HFEB	LI (NOTE 3)	6	12/31/97	NA
I.A.4	Simulator Use and Development						
I.A.4.1	Initial Simulator Improvement	-	-	-			
I.A.4.1(1)	Short-Term Study of Training Simulators	Thatcher	NRR/DHFS/OLB	NOTE 3(b)	6	12/31/97	NA
I.A.4.1(2)	Interim Changes in Training Simulators	Thatcher	NRR/DHFS/OLB	NOTE 3(a)	6	12/31/97	

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
I.A.4.2	Long-Term Training Simulator Upgrade	-	-	-			
I.A.4.2(1)	Research on Training Simulators	Colmar	NRR/DHFT/HFIB	NOTE 3(a)	6	12/31/97	
I.A.4.2(2)	Upgrade Training Simulator Standards	Colmar	RES/DFO/HFBR	NOTE 3(a)	6	12/31/97	
I.A.4.2(3)	Regulatory Guide on Training Simulators	Colmar	RES/DFO/HFBR	NOTE 3(a)	6	12/31/97	
I.A.4.2(4)	Review Simulators for Conformance to Criteria	Colmar	NRR/DLPQ/LOLB	NOTE 3(a)	6	12/31/97	
I.A.4.3	Feasibility Study of Procurement of NRC Training Simulator	Colmar	RES/DAE/RSRB	LI (NOTE 3)	6	12/31/97	NA
I.A.4.4	Feasibility Study of NRC Engineering Computer	Colmar	RES/DAE/RSRB	LI (NOTE 3)	6	12/31/97	NA
I.B.	SUPPORT PERSONNEL						
I.B.1	Management for Operations						
I.B.1.1	Organization and Management Long-Term Improvements	-	-	-			
I.B.1.1(1)	Prepare Draft Criteria	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	4	12/31/97	NA
I.B.1.1(2)	Prepare Commission Paper	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	4	12/31/97	NA
I.B.1.1(3)	Issue Requirements for the Upgrading of Management and Technical Resources	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	4	12/31/97	NA
I.B.1.1(4)	Review Responses to Determine Acceptability	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	4	12/31/97	NA
I.B.1.1(5)	Review Implementation of the Upgrading Activities	Colmar	OIE/DQASIP/ORPB	NOTE 3(b)	4	12/31/97	NA
I.B.1.1(6)	Prepare Revisions to Regulatory Guides 1.33 and 1.8	Colmar	NRR/DHFS/LQB	I.A.2.6(1), 75	4	12/31/97	NA
I.B.1.1(7)	Issue Regulatory Guides 1.33 and 1.8	Colmar	NRR/DHFS/LQB	I.A.2.6(1), 75	4	12/31/97	NA
I.B.1.2	Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants		- -	-			
I.B.1.2(1)	Prepare Draft Criteria	-	NRR/DHFS/LQB	NOTE 3(b)	4	12/31/97	NA
I.B.1.2(2)	Review Near-Term Operating License Facilities	-	NRR/DHFS/LQB	NOTE 3(b)	4	12/31/97	NA
I.B.1.2(3)	Include Findings in the SER for Each Near-Term Operating License Facility	-	NRR/DL/ORAB	NOTE 3(b)	4	12/13/97	NA
I.B.1.3	Loss of Safety Function	-	-	-			
I.B.1.3(1)	Require Licensees to Place Plant in Safest Shutdown Cooling Following a Loss of Safety Function Due to Personnel Error	Sege	RES	LI (NOTE 3)	4	12/31/97	NA
I.B.1.3(2)	Use Existing Enforcement Options to Accomplish Safest Shutdown Cooling	Sege	RES	LI (NOTE 3)	4	12/31/97	NA
I.B.1.3(3)	Use Non-Fiscal Approaches to Accomplish Safest Shutdown Cooling	Sege	RES	LI (NOTE 3)	4	12/31/97	NA
I.B.2	Inspection of Operating Reactors						
I.B.2.1	Revise OIE Inspection Program	-	-	-			
I.B.2.1(1)	Verify the Adequacy of Management and Procedural Controls and Staff Discipline	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
I.B.2.1(2)	Verify that Systems Required to Be Operable Are Properly Aligned	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.1(3)	Follow-up on Completed Maintenance Work Orders to Assure Proper Testing and Return to Service	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.1(4)	Observe Surveillance Tests to Determine Whether Test Instruments Are Properly Calibrated	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.1(5)	Verify that Licensees Are Complying with Technical Specifications	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.1(6)	Observe Routine Maintenance	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.1(7)	Inspect Terminal Boards, Panels, and Instrument Racks for Unauthorized Jumpers and Bypasses	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.2	Resident Inspector at Operating Reactors	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.3	Regional Evaluations	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.4	Overview of Licensee Performance	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)	1	12/31/97	NA
I.C	OPERATING PROCEDURES						
I.C.1	Short-Term Accident Analysis and Procedures Revision	-	-	-			
I.C.1(1)	Small Break LOCAs	-	NRR	I	4	12/31/97	
I.C.1(2)	Inadequate Core Cooling	-	NRR	I	4	12/31/97	F-04
I.C.1(3)	Transients and Accidents	-	NRR	I	4	12/31/97	F-05
I.C.1(4)	Confirmatory Analyses of Selected Transients	Riggs	NRR/DSI/RSB	NOTE 3(b)	4	12/31/97	NA
I.C.2	Shift and Relief Turnover Procedures	-	NRR	I	4	12/31/97	
I.C.3	Shift Supervisor Responsibilities	-	NRR	I	4	12/31/97	
I.C.4	Control Room Access	-	NRR	I	4	12/31/97	
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	-	NRR/DL	I	4	12/31/97	F-06
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	-	NRR/DL	I	4	12/31/97	F-07
I.C.7	NSSS Vendor Review of Procedures	-	NRR/DHFS/PSRB	I	4	12/31/97	
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	-	NRR/DHFS/PSRB	I	4	12/31/97	
I.C.9	Long-Term Program Plan for Upgrading of Procedures	Riggs	NRR/DHFS/PSRB	NOTE 3(b)	4	12/31/97	NA
I.D	CONTROL ROOM DESIGN						
I.D.1	Control Room Design Reviews	-	NRR/DL	I	8	12/31/97	F-08
I.D.2	Plant Safety Parameter Display Console	-	NRR/DL	I	8	12/31/97	F-09
I.D.3	Safety System Status Monitoring	Thatcher	RES/DE/MEB	NOTE 3(b)	8	12/31/97	NA
I.D.4	Control Room Design Standard	Thatcher	RES/DRPS/RHFB	NOTE 3(b)	8	12/31/97	NA
I.D.5	Improved Control Room Instrumentation Research	-	-	-			

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
I.D.5(1)	Operator-Process Communication	Thatcher	RES/DFO/HFBR	NOTE 3(b)	8	12/31/97	NA
I.D.5(2)	Plant Status and Post-Accident Monitoring	Thatcher	RES/DFO/HFBR	NOTE 3(a)	8	12/31/97	
I.D.5(3)	On-Line Reactor Surveillance System	Thatcher	RES/DE/MEB	NOTE 3(b)	8	12/31/97	NA
I.D.5(4)	Process Monitoring Instrumentation	Thatcher	RES/DFO/ICBR	NOTE 3(b)	8	12/31/97	NA
I.D.5(5)	Disturbance Analysis Systems	Thatcher	RES/DRPS/RHFB	LI (NOTE 3)	8	12/31/97	NA
I.D.6	Technology Transfer Conference	Thatcher	RES/DFO/HFBR	LI (NOTE 3)	8	12/31/97	NA
<u>I.E</u>	<u>ANALYSIS AND DISSEMINATION OF OPERATING EXPERIENCE</u>						
I.E.1	Office for Analysis and Evaluation of Operational Data	Matthews	AEOD/PTB	LI (NOTE 3)	3	12/31/97	NA
I.E.2	Program Office Operational Data Evaluation	Matthews	NRR/DL/ORAB	LI (NOTE 3)	3	12/31/97	NA
I.E.3	Operational Safety Data Analysis	Matthews	RES/DRA/RRBR	LI (NOTE 3)	3	12/31/97	NA
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs	Matthews	AEOD/PTB	LI (NOTE 3)	3	12/31/97	NA
I.E.5	Nuclear Plant Reliability Data System	Matthews	AEOD/PTB	LI (NOTE 3)	3	12/31/97	NA
I.E.6	Reporting Requirements	Matthews	AEOD/PTB	LI (NOTE 3)	3	12/31/97	NA
I.E.7	Foreign Sources	Matthews	IP	LI (NOTE 3)	3	12/31/97	NA
I.E.8	Human Error Rate Analysis	Matthews	RES/DFO/HFBR	LI (NOTE 3)	3	12/31/97	NA
<u>I.F</u>	<u>QUALITY ASSURANCE</u>						
I.F.1	Expand QA List	Pittman	RES/DRA/ARGIB	NOTE 3(b)	4	12/31/98	NA
I.F.2	Develop More Detailed QA Criteria	-	-	-			
I.F.2(1)	Assure the Independence of the Organization Performing the Checking Function	Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	4	12/31/98	NA
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	4	12/31/98	NA
I.F.2(4)	Establish Criteria for Determining QA Requirements for Specific Classes of Equipment	Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA
I.F.2(5)	Establish Qualification Requirements for QA and QC Personnel	Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA
I.F.2(6)	Increase the Size of Licensees' QA Staff	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	4	12/31/98	NA
I.F.2(7)	Clarify that the QA Program Is a Condition of the Construction Permit and Operating License	Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA
I.F.2(8)	Compare NRC QA Requirements with Those of Other Agencies	Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	4	12/31/98	NA
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation	Pittman	OIE/DQASIP/QUAB	LOW	4	12/30/98	NA
I.F.2(11)	Define Role of QA in Design and Analysis Activities	Pittman	OIE/DQASIP/QUAB	LOW	4	12/30/98	NA
I.G	PREOPERATIONAL AND LOW-POWER TESTING						
I.G.1	Training Requirements	-	NRR/DHFS/PSRB	I	3	12/31/97	
I.G.2	Scope of Test Program	Vandermolen	NRR/DHFS/PSRB	NOTE 3(a)	3	12/31/97	NA
II.A	SITING						
II.A.1	Siting Policy Reformulation	Vandermolen	NRR/DE/SAB	NOTE 3(b)	2	12/31/97	NA
II.A.2	Site Evaluation of Existing Facilities	Vandermolen	NRR/DE/SAB	V.A.1	2	12/31/97	NA
II.B	CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW						
II.B.1	Reactor Coolant System Vents	-	NRR/DL	I	4	12/31/97	F-10
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	-	NRR/DL	I	4	12/31/97	F-11
II.B.3	Post-Accident Sampling	-	NRR/DL	I	4	12/31/97	F-12
II.B.4	Training for Mitigating Core Damage	-	NRR/DL	I	4	12/31/97	F-13
II.B.5	Research on Phenomena Associated with Core Degradation and Fuel Melting	-	-	-			
II.B.5(1)	Behavior of Severely Damaged Fuel	Vandermolen	RES/DSR/AEB	LI (NOTE 5)	4	12/31/97	NA
II.B.5(2)	Behavior of Core-Melt	Vandermolen	RES/DSR/AEB	LI (NOTE 5)	4	12/31/97	NA
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structure	Vandermolen	RES/DSR/AEB	LI (NOTE 5)	4	12/31/97	NA
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	Pittman	NRR/DST/RRAB	NOTE 3(a)	4	12/31/97	
II.B.7	Analysis of Hydrogen Control	Matthews	NRR/DSI/CSB	II.B.8	4	12/31/97	
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	Vandermolen	RES/DRAO/RAMR	NOTE 3(a)	4	12/31/97	
II.C	RELIABILITY ENGINEERING AND RISK ASSESSMENT						
II.C.1	Interim Reliability Evaluation Program	Pittman	RES/DRAO/RRB	NOTE 3(b)	3	12/31/97	NA
II.C.2	Continuation of Interim Reliability Evaluation Program	Pittman	NRR/DST/RRAB	NOTE 3(b)	3	12/31/97	NA
II.C.3	Systems Interaction	Pittman	NRR/DST/GIB	A-17	3	12/31/97	NA
II.C.4	Reliability Engineering	Pittman	RES/DRPS/RHFB	NOTE 3(b)	3	12/31/97	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.D	REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES						
II.D.1	Testing Requirements	-	NRR/DL	I	3	12/31/98	F-14
II.D.2	Research on Relief and Safety Valve Test Requirements	Riggs	RES	LOW	3	12/31/98	NA
II.D.3	Relief and Safety Valve Position Indication	-	NRR	I	3	12/31/98	
II.E	SYSTEM DESIGN						
II.E.1	Auxiliary Feedwater System						
II.E.1.1	Auxiliary Feedwater System Evaluation	-	NRR/DL	I	2	12/31/97	F-15
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	-	NRR/DL	I	2	12/31/97	F-16, F-17
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	Riggs	RES/DRA/RRBR	NOTE 3(a)	2	12/31/97	
II.E.2	Emergency Core Cooling System						
II.E.2.1	Reliance on ECCS	Riggs	NRR/DSI/RSB	II.K.3(17)	3	12/31/98	NA
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	Riggs	RES/DAE/RSRB	NOTE 3(b)	3	12/31/98	NA
II.E.2.3	Uncertainties in Performance Predictions	Vandermolen	NRR/DSI/RSB	LOW	3	12/31/98	NA
II.E.3	Decay Heat Removal						
II.E.3.1	Reliability of Power Supplies for Natural Circulation	-	NRR/DL	I	2	12/31/97	
II.E.3.2	Systems Reliability	Vandermolen	NRR/DST/GIB	A-45	2	12/31/97	NA
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements	Vandermolen	NRR/DST/GIB	A-45	2	12/31/97	NA
II.E.3.4	Alternate Concepts Research	Riggs	RES/DAE/FBRB	NOTE 3(b)	2	12/31/97	NA
II.E.3.5	Regulatory Guide	Riggs	NRR/DST/GIB	A-45	2	12/31/97	NA
II.E.4	Containment Design						
II.E.4.1	Dedicated Penetrations	-	NRR/DL	I	2	12/31/97	F-18
II.E.4.2	Isolation Dependability	-	NRR/DL	I	2	12/31/97	F-19
II.E.4.3	Integrity Check	Milstead	RES/DRPS/RPSI	NOTE 3(b)	2	12/31/97	NA
II.E.4.4	Purging	-	-	-			
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging	Milstead	NRR/DSI/CSB	NOTE 3(a)	2	12/31/97	
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Letter	Milstead	NRR/DSI/CSB	NOTE 3(a)	2	12/31/97	
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability	Milstead	NRR/DSI/CSB	NOTE 3(a)	2	12/31/97	
II.E.4.4(4)	Evaluate Purging and Venting During Normal Operation	Milstead	NRR/DSI/CSB	NOTE 3(b)	2	12/31/97	NA
II.E.4.4(5)	Issue Modified Purging and Venting Requirement	Milstead	NRR/DSI/CSB	NOTE 3(b)	2	12/31/97	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
<u>II.E.5</u>	<u>Design Sensitivity of B&W Reactors</u>						
II.E.5.1	Design Evaluation	Thatcher	NRR/DSI/RSB	NOTE 3(a)	2	12/31/98	
II.E.5.2	B&W Reactor Transient Response Task Force	Thatcher	NRR/DL/ORAB	NOTE 3(a)	2	12/31/98	
<u>II.E.6</u>	<u>In Situ Testing of Valves</u>						
II.E.6.1	Test Adequacy Study	Thatcher	RES/DE/EIB	NOTE 3(a)	2	12/31/98	
<u>II.F</u>	<u>INSTRUMENTATION AND CONTROLS</u>						
II.F.1	Additional Accident Monitoring Instrumentation	-	NRR/DL	I	3	12/31/98	F-20, F-21, F-22, F-23, F-24, F-25
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	-	NRR/DL	I	3	12/31/98	F-26
II.F.3	Instruments for Monitoring Accident Conditions	Vandermolen	RES/DFO/ICBR	NOTE 3(a)	3	12/31/98	
II.F.4	Study of Control and Protective Action Design Requirements	Thatcher	NRR/DSI/ICSB	DROP	3	12/31/98	NA
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	Thatcher	RES/DE	LI (NOTE 3)	3	12/31/98	NA
<u>II.G</u>	<u>ELECTRICAL POWER</u>						
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	-	NRR	I	1	12/31/98	NA
<u>II.H</u>	<u>TMI-2 CLEANUP AND EXAMINATION</u>						
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact	Matthews	NRR/TMIPO	NOTE 3(b)	3	12/31/98	NA
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	Milstead	RES/DRAA/AEB	NOTE 3(b)	3	12/31/98	NA
II.H.3	Evaluate and Feed Back Information Obtained from TMI	Milstead	NRR/TMIPO	II.H.2	3	12/31/98	NA
II.H.4	Determine Impact of TMI on Socioeconomic and Real Property Values	Milstead	RES/DHSWM/SEBR	LI (NOTE 3)	3	12/31/98	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.J	GENERAL IMPLICATIONS OF TMI FOR DESIGN AND CONSTRUCTION ACTIVITIES						
II.J.1	Vendor Inspection Program						
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections	Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.1.2	Modify Existing Vendor Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees	Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers	Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.2	Construction Inspection Program						
II.J.2.1	Reorient Construction Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.2.3	Assign Resident Inspectors to All Construction Sites	Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.3	Management for Design and Construction						
II.J.3.1	Organization and Staffing to Oversee Design and Construction	Pittman	NRR/DHFS/LQB	I.B.1.1	1	12/31/98	NA
II.J.3.2	Issue Regulatory Guide	Pittman	NRR/DHFS/LQB	I.B.1.1	1	12/31/98	NA
II.J.4	Revise Deficiency Reporting Requirements						
II.J.4.1	Revise Deficiency Reporting Requirements	Riani	AEOD/DSP/ROAB	NOTE 3(a)	3	12/31/98	NA
II.K	MEASURES TO MITIGATE SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS AND LOSS-OF-FEEDWATER ACCIDENTS						
II.K.1	IE Bulletins	-	-	-			
II.K.1(1)	Review TMI-2 PNs and Detailed Chronology of the TMI-2 Accident	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(2)	Review Transients Similar to TMI-2 That Have Occurred at Other Facilities and NRC Evaluation of Davis-Besse Event	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(4)	Review Operating Procedures and Training Instructions	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(5)	Safety-Related Valve Position Description	Emrit	NRR	NOTE 3(a)		12/31/84	-

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred out of Containment Inadvertently	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(10	Review and Modify Procedures for Removing Safety-Related Systems from Service	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(11)	Make All Operating and Maintenance Personnel Aware of the Seriousness and Consequences of the Erroneous Actions Leading up to, and in Early Phases of, the TMI-2 Accident	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(12)	One Hour Notification Requirement and Continuous Communications Channels	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(15)	For Facilities with Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(16)	Implement Procedures That Identify PRZ PORV "Open" Indications and That Direct Operator to Close Manually at "Reset" Setpoint	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(17)	Trip PZR Level Bistable so That PZR Low Pressure Will Initiate Safety Injection	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(18)	Develop Procedures and Train Operators on Methods of Establishing and Maintaining Natural Circulation	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(19)	Describe Design and Procedure Modifications to Reduce Likelihood of Automatic PZR PORV Actuation in Transients	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(20)	Provide Procedures and Training to Operators for Prompt Manual Reactor Trip for LOFW, TT, MSIV Closure, LOOP, LOSG Level, and LO PZR Level	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor Trip for LOFW, TT, or Significant Decrease in SG Level	Emrit	NRR	NOTE 3(a)		12/31/84	-

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(24)	Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and RCP Trip	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(25)	Develop Operator Action Guidelines	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2	Commission Orders on B&W Plants	-	-	-			
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(2)	Procedures and Training to Initiate and Control AFW Independent of Integrated Control System	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(4)	Small-Break LOCA Analysis, Procedures and Operator Training	Emrit	NRR/DHFS/OLB	NOTE 3(a)		12/31/84	-
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2(6)	Reevaluate Analysis for Dual-Level Setpoint Control	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(7)	Reevaluate Transient of September 24, 1977	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(8)	Continued Upgrading of AFW System	Emrit	NRR	II.E.1.1, II.E.1.2		12/31/84	NA
II.K.2(9)	Analysis and Upgrading of Integrated Control System	Emrit	NRR	I		12/31/84	F-27
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	Emrit	NRR	I		12/31/84	F-28
II.K.2(11)	Operator Training and Drilling	Emrit	NRR	I		12/31/84	F-29
II.K.2(12)	Transient Analysis and Procedures for Management of Small Breaks	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.2(13)	Thermal-Mechanical Report on Effect of HPI on Vessel Integrity for Small-Break LOCA With No AFW	Emrit	NRR	I		12/31/84	F-30
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	Emrit	NRR	I		12/31/84	F-31
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes After Primary System Voiding	Emrit	NRR	I		12/31/84	-
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA With Loss of Offsite Power	Emrit	NRR	I		12/31/84	F-32
II.K.2(17)	Analysis of Potential Voiding in RCS During Anticipated Transients	Emrit	NRR	I		12/31/84	F-33

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.K.2(18)	Analysis of Loss of Feedwater and Other Anticipated Transients	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.2(19)	Benchmark Analysis of Sequential AFW Flow to Once-Through Steam Generator	Emrit	NRR	I		12/31/84	F-34
II.K.2(20)	Analysis of Steam Response to Small-Break LOCA That Causes System Pressure to Exceed PORV Setpoint	Emrit	NRR	I		12/31/84	F-35
II.K.2(21)	LOFT L3-1 Predictions	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.3	Final Recommendations of Bulletins and Orders Task Force	-	-	-			
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	Emrit	NRR	I		12/31/84	F-36
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	Emrit	NRR	I		12/31/84	F-37
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	Emrit	NRR	I		12/31/84	F-38
II.K.3(4)	Review and Upgrade Reliability and Redundancy of Non-Safety Equipment for Small-Break LOCA Mitigation	Emrit	NRR	II.C.1, II.C.2, II.C.3		12/31/84	NA
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	Emrit	NRR	I		12/31/84	F-39, G-01
II.K.3(6)	Instrumentation to Verify Natural Circulation	Emrit	NRR/DSI	I.C.1(3), II.F.2, II.F.3		12/31/84	NA
II.K.3(7)	Evaluation of PORV Opening Probability During Overpressure Transient	Emrit	NRR	I		12/31/84	-
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs	Emrit	NRR/DST/GIB	II.C.1, II.E.3.3		12/31/84	NA
II.K.3(9)	Proportional Integral Derivative Controller Modification	Emrit	NRR	I		12/31/84	F-40
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	Emrit	NRR	I		12/31/84	F-41
II.K.3(11)	Control Use of PORV Supplied by Control Components, Inc. Until Further Review Complete	Emrit	NRR	I		12/31/84	-
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	Emrit	NRR	I		12/31/84	F-42
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	Emrit	NRR	I		12/31/84	F-43
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	Emrit	NRR	I		12/31/84	F-44
II.K.3(15)	Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	Emrit	NRR	I		12/31/84	F-45

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification	Emrit	NRR	I		12/31/84	F-46
II.K.3(17)	Report on Outage of ECC Systems - Licensee Report and Technical Specification Changes	Emrit	NRR	I		12/31/84	F-47
II.K.3(18)	Modification of ADS Logic - Feasibility Study and Modification for Increased Diversity for Some Event Sequences	Emrit	NRR	I		12/31/84	F-48
II.K.3(19)	Interlock on Recirculation Pump Loops	Emrit	NRR	I		12/31/84	F-49
II.K.3(20)	Loss of Service Water for Big Rock Point	Emrit	NRR	I		12/31/84	-
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level - Design and Modification	Emrit	NRR	I		12/31/84	F-50
II.K.3(22)	Automatic Switchover of RCIC System Suction - Verify Procedures and Modify Design	Emrit	NRR	I		12/31/84	F-51
II.K.3(23)	Central Water Level Recording	Emrit	NRR	I.D.2, III.A.1.2(1), III.A.3.4		12/31/84	NA
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	Emrit	NRR	I		12/31/84	F-52
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	Emrit	NRR	I		12/31/84	F-53
II.K.3(26)	Study Effect on RHR Reliability of Its Use for Fuel Pool Cooling	Emrit	NRR/DSI	II.E.2.1		12/31/84	NA
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	Emrit	NRR	I		12/31/84	F-54
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	Emrit	NRR	I		12/31/84	F-55
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	Emrit	NRR	I		12/31/84	F-56
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	Emrit	NRR	I		12/31/84	F-57
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	Emrit	NRR	I		12/31/84	F-58
II.K.3(32)	Provide Experimental Verification of Two-Phase Natural Circulation Models	Emrit	NRR/DSI	II.E.2.2		12/31/84	NA
II.K.3(33)	Evaluate Elimination of PORV Function	Emrit	NRR	II.C.1		12/31/84	NA
II.K.3(34)	Relap-4 Model Development	Emrit	NRR/DSI	II.E.2.2		12/31/84	NA
II.K.3(35)	Evaluation of Effects of Core Flood Tank Injection on Small-Break LOCAs	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(36)	Additional Staff Audit Calculations of B&W Small-Break LOCA Analyses	Emrit	NRR	I.C.1(3)		12/31/84	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.K.3(37)	Analysis of B&W Response to Isolated Small-Break LOCA	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(38)	Analysis of Plant Response to a Small-Break LOCA in the Pressurizer Spray Line	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(39)	Evaluation of Effects of Water Slugs in Piping Caused by HPI and CFT Flows	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(40)	Evaluation of RCP Seal Damage and Leakage During a Small-Break LOCA	Emrit	NRR	II.K.2(16)		12/31/84	NA
II.K.3(41)	Submit Predictions for LOFT Test L3-6 with RCPs Running	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(42)	Submit Requested Information on the Effects of Non-Condensible Gases	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(43)	Evaluation of Mechanical Effects of Slug Flow on Steam Generator Tubes	Emrit	NRR	II.K.2(15)		12/31/84	NA
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	Emrit	NRR	I		12/31/84	F-59
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	Emrit	NRR	I		12/31/84	F-60
II.K.3(46)	Response to List of Concerns from ACRS Consultant	Emrit	NRR	I		12/31/84	F-61
II.K.3(47)	Test Program for Small-Break LOCA Model Verification Pretest Prediction, Test Program, and Model Verification	Emrit	NRR	I.C.1(3), II.E.2.2		12/31/84	NA
II.K.3(48)	Assess Change in Safety Reliability as a Result of Implementing B&OTF Recommendations	Emrit	NRR	II.C.1, II.C.2		12/31/84	NA
II.K.3(49)	Review of Procedures (NRC)	Emrit	NRR/DHFS/PSRB	I.C.8, I.C.9		12/31/84	NA
II.K.3(50)	Review of Procedures (NSSS Vendors)	Emrit	NRR/DHFS/PSRB	I.C.7, I.C.9		12/31/84	NA
II.K.3(51)	Symptom-Based Emergency Procedures	Emrit	NRR/DHFS/PSRB	I.C.9		12/31/84	NA
II.K.3(52)	Operator Awareness of Revised Emergency Procedures	Emrit	NRR	I.B.1.1, I.C.2, I.C.5		12/31/84	NA
II.K.3(53)	Two Operators in Control Room	Emrit	NRR	I.A.1.3		12/31/84	NA
II.K.3(54)	Simulator Upgrade for Small-Break LOCAs	Emrit	NRR	I.A.4.1(2)		12/31/84	NA
II.K.3(55)	Operator Monitoring of Control Board	Emrit	NRR	I.C.1(3), I.D.2, I.D.3		12/31/84	NA
II.K.3(56)	Simulator Training Requirements	Emrit	NRR/DHFS/OLB	I.A.2.6(3), I.A.3.1		12/31/84	NA
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	Emrit	NRR	I		12/31/84	F-62

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
III.A	EMERGENCY PREPAREDNESS AND RADIATION EFFECTS						
III.A.1	Improve Licensee Emergency Preparedness - Short-Term						
III.A.1.1	Upgrade Emergency Preparedness	-		-			
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	-	OIE/DEPER/EPB I		2	06/30/91	
III.A.1.1(2)	Perform an Integrated Assessment of the Implementation	-	OIE/DEPER/EPB	NOTE 3(b)	2	06/30/91	NA
III.A.1.2	Upgrade Licensee Emergency Support Facilities	-	-	-	2	06/30/91	
III.A.1.2(1)	Technical Support Center	-	OIE/DEPER/EPB	I	2	06/30/91	F-63
III.A.1.2(2)	On-Site Operational Support Center	-	OIE/DEPER/EPB I		2	06/30/91	F-64
III.A.1.2(3)	Near-Site Emergency Operations Facility	-	OIE/DEPER/EPB I		2	06/30/91	F-65
III.A.1.3	Maintain Supplies of Thyroid-Blocking Agent	-	-	-	2	06/30/91	
III.A.1.3(1)	Workers	Riggs	OIE/DEPER/EPB	NOTE 3(b)	2	06/30/91	NA
III.A.1.3(2)	Public	Riggs	OIE/DEPER/EPB	NOTE 3(b)	2	06/30/91	NA
III.A.2	Improving Licensee Emergency Preparedness - Long-Term						
III.A.2.1	Amend 10 CFR 50 and 10 CFR 50, Appendix E	-	-	-			
III.A.2.1(1)	Publish Proposed Amendments to the Rules	-	RES	NOTE 3(a)		12/31/94	NA
III.A.2.1(2)	Conduct Public Regional Meetings	-	RES	NOTE 3(b)		12/31/94	NA
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules	-	RES	NOTE 3(b)		12/31/94	NA
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	-	OIE	I			F-67
III.A.2.2	Development of Guidance and Criteria	-	NRR/DL	I			F-68
III.A.3	Improving NRC Emergency Preparedness						
III.A.3.1	NRC Role in Responding to Nuclear Emergencies	-	-	-			
III.A.3.1(1)	Define NRC Role in Emergency Situations	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(2)	Revise and Upgrade Plans and Procedures for the NRC Emergency Operations Center	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(3)	Revise Manual Chapter 0502, Other Agency Procedures, and NUREG-0610	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(4)	Prepare Commission Paper	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(5)	Revise Implementing Procedures and Instructions for Regional Offices	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.2	Improve Operations Centers	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.3	Communications	-	-	-			
III.A.3.3(1)	Install Direct Dedicated Telephone Lines	Pittman	OIE/DEPER/IRDB	NOTE 3(a)	1	06/30/85	NA
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems	Pittman	OIE/DEPER/IRDB	NOTE 3(a)	1	06/30/85	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
III.A.3.4	Nuclear Data Link	Thatcher	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	
III.A.3.5	Training, Drills, and Tests	Pittman	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.6	Interaction of NRC and Other Agencies	-	-	-			
III.A.3.6(1)	International	Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	06/30/85	NA
III.A.3.6(2)	Federal	Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	06/30/85	NA
III.A.3.6(3)	State and Local	Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	06/30/85	NA
III.B	EMERGENCY PREPAREDNESS OF STATE AND LOCAL GOVERNMENTS						
III.B.1	Transfer of Responsibilities to FEMA	Milstead	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.B.2	Implementation of NRC and FEMA Responsibilities	-	-	-			
III.B.2(1)	The Licensing Process	Milstead	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.B.2(2)	Federal Guidance	Milstead	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.C	PUBLIC INFORMATION						
III.C.1	Have Information Available for the News Media and the Public	-	-	-			
III.C.1(1)	Review Publicly Available Documents	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.1(2)	Recommend Publication of Additional Information	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.1(3)	Program of Seminars for News Media Personnel	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.2	Develop Policy and Provide Training for Interfacing With the News Media	-	-	-			
III.C.2(1)	Develop Policy and Procedures for Dealing With Briefing Requests	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.2(2)	Provide Training for Members of the Technical Staff	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.D	RADIATION PROTECTION						
III.D.1	Radiation Source Control						
III.D.1.1	Primary Coolant Sources Outside the Containment Structure	-	-	-			
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining to Reducing Leakage from Operating Systems	-	NRR	I	1	12/31/88	
III.D.1.1(2)	Review Information on Provisions for Leak Detection	Emrit	RES/DRA/ARGIB	DROP	1	12/31/88	
III.D.1.1(3)	Develop Proposed System Acceptance Criteria	Emrit	RES/DRA/ARGIB	DROP	1	12/31/88	
III.D.1.2	Radioactive Gas Management	Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3	Ventilation System and Radioiodine Adsorber Criteria	-	-	-			
III.D.1.3(1)	Decide Whether Licensees Should Perform Studies and Make Modifications	Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
III.D.1.3(2)	Review and Revise SRP	Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems	Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Adsorber	Emrit	NRR/DSI/METB	NOTE 3(b)	1	12/31/88	NA
III.D.1.4	Radwaste System Design Features to Aid in Accident Recovery and Decontamination	Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.2	Public Radiation Protection Improvement						
III.D.2.1	Radiological Monitoring of Effluents	-	-	-			
III.D.2.1(1)	Evaluate the Feasibility and Perform a Value-Impact Analysis of Modifying Effluent-Monitoring Design Criteria	Emrit	NRR/DSI/METB	LOW	3	12/31/98	NA
III.D.2.1(2)	Study the Feasibility of Requiring the Development of Effective Means for Monitoring and Sampling Noble Gases and Radioiodine Released to the Atmosphere	Emrit	NRR/DSI/METB	LOW	3	12/31/98	NA
III.D.2.1(3)	Revise Regulatory Guides	Emrit	NRR/DSI/METB	LOW	3	12/31/98	NA
III.D.2.2	Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis	-	-	-			
III.D.2.2(1)	Perform Study of Radioiodine, Carbon-14, and Tritium Behavior	Emrit	NRR/DSI/RAB	NOTE 3(b)	3	12/31/98	NA
III.D.2.2(2)	Evaluate Data Collected at Quad Cities	Emrit	NRR/DSI/RAB	III.D.2.5	3	12/31/98	NA
III.D.2.2(3)	Determine the Distribution of the Chemical Species of Radioiodine in Air-Water-Steam Mixtures	Emrit	NRR/DSI/RAB	III.D.2.5	3	12/31/98	NA
III.D.2.2(4)	Revise SRP and Regulatory Guides	Emrit	NRR/DSI/RAB	III.D.2.5	3	12/31/98	NA
III.D.2.3	Liquid Pathway Radiological Control	-	-	-			
III.D.2.3(1)	Develop Procedures to Discriminate Between Sites/Plants	Emrit	NRR/DE/EHEB	NOTE 3(b)	3	12/31/98	NA
III.D.2.3(2)	Discriminate Between Sites and Plants That Require Consideration of Liquid Pathway Interdiction Techniques	Emrit	NRR/DE/EHEB	NOTE 3(b)	3	12/31/98	NA
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction	Emrit	NRR/DE/EHEB	NOTE 3(b)	3	12/31/98	NA
III.D.2.3(4)	Prepare a Summary Assessment	Emrit	NRR/DE/EHEB	NOTE 3(b)	3	12/31/98	NA
III.D.2.4	Offsite Dose Measurements	-	-	-			
III.D.2.4(1)	Study Feasibility of Environmental Monitors	Vandermolen	NRR/DSI/RAB	NOTE 3(b)	3	12/31/98	NA
III.D.2.4(2)	Place 50 TLDs Around Each Site	Vandermolen	OIE/DRP/ORPB	LI (NOTE 3)	3	12/31/98	NA
III.D.2.5	Offsite Dose Calculation Manual	Vandermolen	NRR/DSI/RAB	NOTE 3(b)	3	12/31/98	NA
III.D.2.6	Independent Radiological Measurements	Vandermolen	OIE/DRP/ORPB	LI (NOTE 3)	3	12/31/98	NA
III.D.3	Worker Radiation Protection Improvement						
III.D.3.1	Radiation Protection Plans	Vandermolen	NRR/DSI/RAB	NOTE 3(b)	3	12/31/87	NA
III.D.3.2	Health Physics Improvements	-	-	-			
III.D.3.2(1)	Amend 10 CFR 20	Vandermolen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(2)	Issue a Regulatory Guide	Vandermolen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
III.D.3.2(3)	Develop Standard Performance Criteria	Vandermolen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(4)	Develop Method for Testing and Certifying Air-Purifying Respirators	Vandermolen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.3	In-plant Radiation Monitoring	-	-	-			
III.D.3.3(1)	Issue Letter Requiring Improved Radiation Sampling Instrumentation	-	NRR/DL	I	2	12/31/86	F-69
III.D.3.3(2)	Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	-	NRR	NOTE 3(a)	2	12/31/86	NA
III.D.3.3(3)	Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation-Monitoring Instruments	-	RES	NOTE 3(a)	2	12/31/86	NA
III.D.3.3(4)	Issue a Regulatory Guide	-	RES	NOTE 3(a)	2	12/31/86	NA
III.D.3.4	Control Room Habitability	-	NRR/DL	I	2	12/31/86	F-70
III.D.3.5	Radiation Worker Exposure	-	-	-			
III.D.3.5(1)	Develop Format for Data To Be Collected by Utilities Regarding Total Radiation Exposure to Workers	Vandermolen	DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
III.D.3.5(2)	Investigative Methods of Obtaining Employee Health Data by Nonlegislative Means	Vandermolen	DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
III.D.3.5(3)	Revise 10 CFR 20	Vandermolen	DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
<u>IV.A</u>	<u>STRENGTHEN ENFORCEMENT PROCESS</u>						
IV.A.1	Seek Legislative Authority	Emrit	GC	LI (NOTE 3)		11/30/83	NA
IV.A.2	Revise Enforcement Policy	Emrit	OIE/ES	LI (NOTE 3)		11/30/83	NA
<u>IV.B</u>	<u>ISSUANCE OF INSTRUCTIONS AND INFORMATION TO LICENSEES</u>						
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	Emrit	OIE/DEPER	LI (NOTE 3)		11/30/83	NA
<u>IV.C</u>	<u>EXTEND LESSONS LEARNED TO LICENSED ACTIVITIES OTHER THAN POWER REACTORS</u>						
IV.C.1	Extend Lessons Learned from TMI to Other NRC Programs	Emrit	NMSS/WM	NOTE 3(b)		11/30/83	NA
<u>IV.D</u>	<u>NRC STAFF TRAINING</u>						
IV.D.1	NRC Staff Training	Emrit	ADM/MDTS	LI (NOTE 3)		11/30/83	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
<u>IV.E</u>	<u>SAFETY DECISION-MAKING</u>						
IV.E.1	Expand Research on Quantification of Safety Decision-Making	Colmar	RES/DRA/RABR	LI (NOTE 3)	2	12/31/86	NA
IV.E.2	Plan for Early Resolution of Safety Issues	Emrit	NRR/DST/SPEB	LI (NOTE 3)	2	12/31/86	NA
IV.E.3	Plan for Resolving Issues at the CP Stage	Colmar	RES/DRA/RABR	LI (NOTE 5)	2	12/31/86	NA
IV.E.4	Resolve Generic Issues by Rulemaking	Colmar	RES/DRA/RABR	LI (NOTE 3)	2	12/31/86	NA
IV.E.5	Assess Currently Operating Reactors	Matthews	NRR/DL/SEPB	NOTE 3(b)	2	12/31/86	NA
<u>IV.F</u>	<u>FINANCIAL DISINCENTIVES TO SAFETY</u>						
IV.F.1	Increased OIE Scrutiny of the Power-Ascension Test Program	Thatcher	OIE/DQASIP	NOTE 3(b)	1	12/31/86	NA
IV.F.2	Evaluate the Impacts of Financial Disincentives to the Safety of Nuclear Power Plants	Matthews	SP	NOTE 3(b)	1	12/31/86	NA
<u>IV.G</u>	<u>IMPROVE SAFETY RULEMAKING PROCEDURES</u>						
IV.G.1	Develop a Public Agenda for Rulemaking	Emrit	ADM/RPB	LI (NOTE 3)	1	12/31/86	NA
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules	Milstead	RES/DRA/RABR	LI (NOTE 3)	1	12/31/86	NA
IV.G.3	Improve Rulemaking Procedures	Milstead	RES/DRA/RABR	LI (NOTE 3)	1	12/31/86	NA
IV.G.4	Study Alternatives for Improved Rulemaking Process	Milstead	RES/DRA/RABR	LI (NOTE 3)	1	12/31/86	NA
<u>IV.H</u>	<u>NRC PARTICIPATION IN THE RADIATION POLICY COUNCIL</u>						
IV.H.1	NRC Participation in the Radiation Policy Council	Sege	RES/DHSWM/HEBR	LI (NOTE 3)		11/30/83	NA
<u>V.A</u>	<u>DEVELOPMENT OF SAFETY POLICY</u>						
V.A.1	Develop NRC Policy Statement on Safety	Emrit	GC	LI (NOTE 3)		12/31/86	NA
<u>V.B</u>	<u>POSSIBLE ELIMINATION OF NONSAFETY RESPONSIBILITIES</u>						
V.B.1	Study and Recommend, as Appropriate, Elimination of Nonsafety Responsibilities	Emrit	GC	LI (NOTE 3)		12/31/86	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
V.C	ADVISORY COMMITTEES						
V.C.1	Strengthen the Role of Advisory Committee on Reactor Safeguards	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.C.2	Study Need for Additional Advisory Committees	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.C.3	Study the Need to Establish an Independent Nuclear Safety Board	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.D	LICENSING PROCESS						
V.D.1	Improve Public and Intervenor Participation in the Hearing Process	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.D.2	Study Construction-During-Adjudication Rules	Emrit	GC	LI (NOTE 5)		12/31/86	NA
V.D.3	Reexamine Commission Role in Adjudication	Emrit	GC	LI (NOTE 5)		12/31/86	NA
V.D.4	Study the Reform of the Licensing Process	Emrit	GC	LI (NOTE 5)		12/31/86	NA
V.E	LEGISLATIVE NEEDS						
V.E.1	Study the Need for TMI-Related Legislation	Emrit	GC	LI (NOTE 5)		12/31/86	NA
V.F	ORGANIZATION AND MANAGEMENT						
V.F.1	Study NRC Top Management Structure and Process	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.F.2	Reexamine Organization and Functions of the NRC Offices	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.F.3	Revise Delegations of Authority to Staff	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.F.4	Clarify and Strengthen the Respective Roles of Chairman, Commission, and Executive Director for Operations	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.F.5	Authority to Delegate Emergency Response Functions to a Single Commissioner	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.G	CONSOLIDATION OF NRC LOCATIONS						
V.G.1	Achieve Single Location, Long-Term	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.G.2	Achieve Single Location, Interim	Emrit	GC	LI (NOTE 3)		12/31/86	NA
	TASK ACTION PLAN ITEMS						
A-1	Water Hammer (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	D-10

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	12/31/88	
A-4	CE Steam Generator Tube Integrity (former USI)	Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	12/31/88	
A-5	B&W Steam Generator Tube Integrity (former USI)	Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	12/31/88	
A-6	Mark I Short-Term Program (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-7	Mark I Long-Term Program (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	D-01
A-8	Mark II Containment Pool Dyanmic Loads Long-Term Program (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-9	ATWS (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-10	BWR Feedwater Nozzle Cracking (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-25
A-11	Reactor Vessel Materials Toughness (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-13	Snubber Operability Assurance	Emrit	NRR/DE/MEB	NOTE 3(a)	1	06/30/91	B-17, B-22
A-14	Flaw Detection	Matthews	NRR/DE/MTEB	DROP		11/30/83	NA
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning	Pittman	NRR/DE/CHEB	NOTE 3(b)		11/30/83	NA
A-16	Steam Effects on BWR Core Spray Distribution	Emrit	NRR/DSI/CPB	NOTE 3(a)		11/30/83	D-12
A-17	Systems Interactions in Nuclear Power Plants (former (USI)	Emrit	RES/DSIR/EIB	NOTE 3(b)	1	12/31/89	NA
A-18	Pipe Rupture Design Criteria	Emrit	NRR/DE/MEB	DROP		11/30/83	NA
A-19	Digital Computer Protection System	Milstead	RES/DSR/HFB	LI (NOTE 5)	1	06/30/91	NA
A-20	Impacts of the Coal Fuel Cycle	-	NRR/DE/EHEB	LI (NOTE 5)		11/30/83	NA
A-21	Main Steamline Break Inside Containment - Evaluation of Environmental Conditions for Equipment Qualification	Vandermolen	NRR/DSI/CSB	DROP	1	12/31/98	NA
A-22	PWR Main Steamline Break - Core, Reactor Vessel and Containment Building Response	V'Molen	NRR/DSI/CSB	DROP		11/30/83	NA
A-23	Containment Leak Testing	Matthews	NRR/DSI/CSB	RI (NOTE 5)		11/30/83	
A-24	Qualification of Class 1E Safety-Related Equipment (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-60
A-25	Non-Safety Loads on Class 1E Power Sources	Thatcher	NRR/DSI/PSB	NOTE 3(a)		11/30/83	
A-26	Reactor Vessel Pressure Transient Protection (former (USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-04
A-27	Reload Applications	-	NRR/DSI/CPB	LI (NOTE 5)		11/30/83	NA
A-28	Increase in Spent Fuel Pool Storage Capacity	Colmar	NRR/DE/SGEB	NOTE 3(a)		11/30/83	
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	Colmar	RES/DRPS/RPSI	NOTE 3(b)	1	12/31/89	NA
A-30	Adequacy of Safety-Related DC Power Supplies	Sege	NRR/DSI/PSB	128	1	12/31/86	NA
A-31	RHR Shutdown Requirements (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-32	Missile Effects	Pittman	NRR/DE/MTEB	A-37, A-38, B-68		11/30/83	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
A-33	NEPA Review of Accident Risks	-	NRR/DSI/AEB	EI(NOTE 3)		11/30/83	NA
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents	V'Molen	NRR/DSI/ICSB	II.F.3		11/30/83	NA
A-35	Adequacy of Offsite Power Systems	Emrit	NRR/DSI/PSB	NOTE 3(a)	1	12/31/94	B-23
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	Emrit	NRR/DSI/GIB	NOTE 3(a)	1	06/30/85	C-10, C-15
A-37	Turbine Missiles	Pittman	NRR/DE/MTEB	DROP		11/30/83	NA
A-38	Tornado Missiles	Sege	NRR/DSI/ASB	DROP	3	06/30/00	NA
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
~~A-40~~	~~Seismic Design Criteria (former USI)~~	~~Emrit~~	~~RES/DSIR/EIB~~	~~NOTE 3(a)~~	~~1~~	~~12/31/89~~	~~NA~~
A-41	Long-Term Seismic Program	Colmar	NRR/DE/MEB	NOTE 3(b)	1	12/31/84	NA
A-42	Pipe Cracks in Boiling Water Reactors (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-05
A-43	Containment Emergency Sump Performance (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	12/31/87	
A-44	Station Blackout (former USI)	Emrit	RES/DRPS/RPSI	NOTE 3(a)	1	06/30/88	
A-45	Shutdown Decay Heat Removal Requirements (former USI)	Emrit	RES/DRPS/RPSI	NOTE 3(b)	1	12/31/88	NA
A-46	Seismic Qualification of Equipment in Operating Plants (former USI)	Emrit	NRR/DSRO/EIB	NOTE 3(a)	2	06/30/00	
A-47	Safety Implications of Control Systems (former USI)	Emrit	RES/DSIR/EIB	NOTE 3(a)	1	12/31/89	
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	Emrit	NRR/DSIR/SAIB	NOTE 3(a)	1	06/30/89	
A-49	Pressurized Thermal Shock (former USI)	Emrit	NRR/DSRO/RSIB	NOTE 3(a)	1	12/31/87	A-21
B-1	Environmental Technical Specifications	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-2	Forecasting Electricity Demand	-	NRR	EI (NOTE 3)		11/30/83	NA
B-3	Event Categorization	-	NRR/DSI/RSB	LI (NOTE 3)		11/30/83	NA
B-4	ECCS Reliability	Emrit	NRR/DSI/RSB	II.E.3.2		11/30/83	NA
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	Thatcher	RES/DE/EIB	NOTE 3(b)	1	06/30/88	NA
B-6	Loads, Load Combinations, Stress Limits	Pittman	NRR/DSRO/EIB	119.1		12/31/87	NA
B-7	Secondary Accident Consequence Modeling	-	NRR/DSI/AEB	LI (NOTE 3)		11/30/83	NA
B-8	Locking Out of ECCS Power Operated Valves	Riggs	NRR/DSI/RSB	DROP	1	12/31/94	NA
B-9	Electrical Cable Penetrations of Containment	Emrit	NRR/DSI/PSB	NOTE 3(b)		11/30/83	NA
B-10	Behavior of BWR Mark III Containments	Vandermolen	NRR/DSI/CSB	NOTE 3(a)	1	12/31/84	NA
B-11	Subcompartment Standard Problems	-	NRR/DSI/CSB	LI (NOTE 5)		11/30/83	NA
B-12	Containment Cooling Requirements (Non-LOCA)	Emrit	NRR/DSI/CSB	NOTE 3(b)	1	12/31/86	NA
B-13	Marviken Test Data Evaluation	-	NRR/DSI/CSB	LI (NOTE 5)		11/30/83	NA
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA	Emrit	NRR/DST/GIB	A-48		11/30/83	NA
B-15	CONTEMPT Computer Code Maintenance	-	NRR/DSI/CSB	LI (NOTE 3)		11/30/83	NA
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	Emrit	NRR/DE/MEB	A-18		11/30/83	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
B-17	Criteria for Safety-Related Operator Actions	Milstead	RES/DST/CIHFB	NOTE 3(b)	3	06/30/00	
B-18	Vortex Suppression Requirements for Containment Sumps	Emrit	NRR/DST/GIB	A-43		11/30/83	NA
B-19	Thermal-Hydraulic Stability	Colmar	NRR/DSI/CPB	NOTE 3(b)		06/30/85	NA
B-20	Standard Problem Analysis	-	RES/DAE/AMBR	LI (NOTE 5)		11/30/83	
B-21	Core Physics	-	NRR/DSI/CPB	LI (NOTE 3)		11/30/83	NA
B-22	LWR Fuel	Emrit	RES/DSIR/RPSIB	DROP	2	06/30/95	NA
B-23	LMFBR Fuel	-	NRR/DSI/CPB	LI (NOTE 3)		11/30/83	NA
B-24	Seismic Qualification of Electrical and Mechanical Equipment	Emrit	NRR	A-46		11/30/83	NA
B-25	Piping Benchmark Problems	-	NRR/DE/MEB	LI (NOTE 5)		11/30/83	
B-26	Structural Integrity of Containment Penetrations	Riggs	NRR/DE/MTEB	NOTE 3(b)	1	12/31/84	NA
B-27	Implementation and Use of Subsection NF	-	NRR/DE/MEB	LI (NOTE 5)		11/30/83	
B-28	Radionuclide/Sediment Transport Program	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-29	Effectiveness of Ultimate Heat Sinks	Pittman	NRR/DE/EHEB	LI (NOTE 3)	1	06/30/91	NA
B-30	Design Basis Floods and Probability	-	NRR/DE/EHEB	LI (NOTE 5)		11/30/83	
B-31	Dam Failure Model	Milstead	NRR/DE/SGEB	LI (NOTE 3)	1	06/30/89	NA
B-32	Ice Effects on Safety-Related Water Supplies	Pittman	NRR/DE/EHEB	153	1	06/30/91	NA
B-33	Dose Assessment Methodology	-	NRR/DSI/RAB	LI (NOTE 3)		11/30/83	NA
B-34	Occupational Radiation Exposure Reduction	Emrit	NRR/DSI/RAB	III.D.3.1		11/30/83	NA
B-35	Confirmation of Appendix I Models for Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light Water Cooled Power Reactors	-	NRR/DSI/METB	LI (NOTE 5)		11/30/83	
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems	Emrit	NRR/DSI/METB	NOTE 3(a)		11/30/83	
B-37	Chemical Discharges to Receiving Waters	-	NRR/DE/EHEB	EI (NOTE 5)		11/30/83	
B-38	Reconnaissance Level Investigations	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-39	Transmission Lines	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-40	Effects of Power Plant Entrainment on Plankton	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-41	Impacts on Fisheries	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-42	Socioeconomic Environmental Impacts	-	NRR/DE/SAB	EI (NOTE 3)		11/30/83	NA
B-43	Value of Aerial Photographs for Site Evaluation	-	NRR/DE/EHEB	EI (NOTE 5)		11/30/83	
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants	-	NRR/DE/SAB	EI (NOTE 3)		11/30/83	NA
B-45	Need for Power - Energy Conservation	-	NRR/DE/SAB	EI (NOTE 3)		11/30/83	NA
B-46	Cost of Alternatives in Environmental Design	-	NRR/DE/SAB	EI (NOTE 3)		11/30/83	NA
B-47	Inservice Inspection of Supports-Classes 1, 2, 3, and MC Components	Colmar	NRR/DE/MTEB	DROP		11/30/83	NA
B-48	BWR Control Rod Drive Mechanical Failures	Emrit	NRR/DE/MTEB	NOTE 3(b)		11/30/83	
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments	-	NRR	LI (NOTE 5)		11/30/83	

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
B-50	Post-Operating Basis Earthquake Inspection	Colmar	NRR/DE/SGEB	RI (NOTE 3)	1	06/30/85	NA
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components	Emrit	NRR/DE/MEB	A-40		11/30/83	NA
B-52	Fuel Assembly Seismic and LOCA Responses	Emrit	NRR/DST/GIB	A-2		11/30/83	NA
B-53	Load Break Switch	Sege	NRR/DSI/PSB	RI (NOTE 3)		11/30/83	
B-54	Ice Condenser Containments	Milstead	NRR/DSI/CSB	NOTE 3(b)	1	12/31/84	NA
B-55	Improved Reliability of Target Rock Safety Relief Valves	Vandermolen	NRR/DE/EMEB	NOTE 3(b)	1	06/30/00	
B-56	Diesel Reliability	Milstead	RES/DRPS/RPSI	NOTE 3(a)	2	06/30/95	D-19
B-57	Station Blackout	Emrit	NRR/DST/GIB	A-44		11/30/83	
B-58	Passive Mechanical Failures	Colmar	NRR/DE/EQB	NOTE 3(b)	1	12/31/85	NA
B-59	(N-1) Loop Operation in BWRs and PWRs	Colmar	NRR/DSI/RSB	RI (NOTE 3)	1	06/30/85	E-04,E-05
B-60	Loose Parts Monitoring Systems	Emrit	NRR/DSI/CPB	NOTE 3(b)	1	12/31/84	NA
B-61	Allowable ECCS Equipment Outage Periods	Pittman	RES/DST/PRAB	NOTE 3(b)	1	06/30/00	
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions	-	NRR/DSI/CPB	LI (NOTE 3)		11/30/83	NA
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	Emrit	NRR/DE/MEB	NOTE 3(a)		11/30/83	B-45
B-64	Decommissioning of Reactors	Colmar	RES/DE/MEB	NOTE 3(a)	2	06/30/95	NA
B-65	Iodine Spiking	Milstead	NRR/DSI/AEB	DROP	2	12/31/84	NA
B-66	Control Room Infiltration Measurements	Matthews	NRR/DSI/AEB	NOTE 3(a)		11/30/83	
B-67	Effluent and Process Monitoring Instrumentation	Colmar	NRR/DSI/METB	III.D.2.1		11/30/83	NA
B-68	Pump Overspeed During LOCA	Riani	NRR/DSI/ASB	DROP		11/30/83	NA
B-69	ECCS Leakage Ex-Containment	Riani	NRR/DSI/METB	III.D.1.1(1)		11/30/83	NA
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps	Emrit	NRR/DSI/PSB	NOTE 3(b)		11/30/83	
B-71	Incident Response	Riani	NRR	III.A.3.1		11/30/83	NA
B-72	Health Effects and Life Shortening from Uranium and Coal Fuel Cycles	-	NRR/DSI/RAB	LI (NOTE 5)		11/30/83	NA
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel	Thatcher	NRR/DE/MEB	C-12		11/30/83	NA
C-1	Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	Milstead	NRR/DE/EQB	NOTE 3(a)		11/30/83	
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure	Emrit	NRR/DSI/CSB	NOTE 3(b)		11/30/83	NA
C-3	Insulation Usage Within Containment	Emrit	NRR/DST/GIB	A-43	1	06/30/91	NA
C-4	Statistical Methods for ECCS Analysis	Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-5	Decay Heat Update	Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-6	LOCA Heat Sources	Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-7	PWR System Piping	Emrit	NRR/DE/MTEB	NOTE 3(b)		11/30/83	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
C-8	Main Steam Line Leakage Control Systems	Milstead	RES/DRPS/RPSI	NOTE 3(b)	1	06/30/90	NA
C-9	RHR Heat Exchanger Tube Failures	V'Molen	NRR/DSI/RSB	DROP		11/30/83	NA
C-10	Effective Operation of Containment Sprays in a LOCA	Emrit	NRR/DSI/AEB	NOTE 3(a)		11/30/83	NA
C-11	Assessment of Failure and Reliability of Pumps and Valves	Emrit	NRR/DE/MEB	NOTE 3(b)		12/31/85	NA
C-12	Primary System Vibration Assessment	Thatcher	NRR/DE/MEB	NOTE 3(b)		11/30/83	NA
C-13	Non-Random Failures	Emrit	NRR/DST/GIB	A-17	1	06/30/91	NA
C-14	Storm Surge Model for Coastal Sites	Emrit	NRR/DE/EHEB	LI (NOTE 3)		06/30/88	NA
C-15	NUREG Report for Liquid Tank Failure Analysis	-	NRR/DE/EHEB	LI (NOTE 3)		11/30/83	NA
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	Emrit	NRR/DSI/METB	NOTE 3(a)		11/30/83	NA
D-1	Advisability of a Seismic Scram	Thatcher	RES/DET/MSEB	DROP	1	12/31/98	NA
D-2	Emergency Core Cooling System Capability for Future Plants	Emrit	RES/DRA/ARGIB	DROP		12/31/88	NA
D-3	Control Rod Drop Accident	Emrit	NRR/DSI/CPB	NOTE 3(b)		11/30/83	NA
		NEW GENERIC ISSUES					
1.	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
2.	Failure of Protective Devices on Essential Equipment	Diab	RES/DSIR/EIB	DROP	2	06/30/95	NA
3.	Set Point Drift in Instrumentation	Emrit	NRR/DSIR/RPSIB	NOTE 3(b)	1	06/30/86	NA
4.	End-of-Life and Maintenance Criteria	Thatcher	NRR/DE/EQB	NOTE 3(b)		11/30/83	NA
5.	Design Check and Audit of Balance-of-Plant Equipment	Pittman	NRR/DSI/ASB	I.F.1		11/30/83	NA
6	Separation of Control Rod from Its Drive and BWR High Rod Worth Events	Vandermolen	NRR/DSI/CPB	NOTE 3(b)	1	12/31/94	NA
7.	Failures Due to Flow-Induced Vibrations	Vandermolen	NRR/DSI/RSB	DROP	1	06/30/91	NA
8.	Inadvertent Actuation of Safety Injection in PWRs	Colmar	NRR/DSI/RSB	I.C.1		11/30/83	NA
9.	Reevaluation of Reactor Coolant Pump Trip Criteria	Emrit	NRR/DSI/RSB	II.K.3(5)		11/30/83	NA
10.	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges	Riggs	NRR/DSI/ICSB	DROP		11/30/83	NA
11.	Turbine Disc Cracking	Pittman	NRR/DE/MTEB	A-37		11/30/83	NA
12.	BWR Jet Pump Integrity	Sege	NRR/DE/MTEB, MEB	NOTE 3(b)	1	12/31/84	NA
13.	Small Break LOCA from Extended Overheating of Pressurizer Heaters	Riani	NRR/DSI/RSB	DROP		11/30/83	NA
14.	PWR Pipe Cracks	Emrit	NRR/DE/MTEB	NOTE 3(b)	2	12/31/94	NA
15.	Radiation Effects on Reactor Vessel Supports	Emrit	RES/DET/EMMEB	NOTE 3(b)	3	06/30/96	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
16.	BWR Main Steam Isolation Valve Leakage Control Systems	Milstead	NRR/DSI/ASB	C-8		11/30/83	NA
17.	Loss of Offsite Power Subsequent to a LOCA	Colmar	NRR/DSI/PSB, ICSB	DROP		11/30/83	NA
18.	Steam Line Break with Consequential Small LOCA	Riggs	NRR/DSI/RSB	I.C.1		11/30/83	NA
19.	Safety Implications of Nonsafety Instrument and Control Power Supply Bus	Sege	NRR/DST/GIB	A-47		11/30/83	NA
20.	Effects of Electromagnetic Pulse on Nuclear Power Plants	Thatcher	NRR/DSI/ICSB	NOTE 3(b)	1	06/30/84	NA
21.	Vibration Qualification of Equipment	Riggs	NRR/DE/EIB	DROP	2	06/30/91	NA
22.	Inadvertent Boron Dilution Events	Vandermolen	NRR/DSI/RSB	NOTE 3(b)	2	12/31/94	NA
23.	Reactor Coolant Pump Seal Failures	Riggs	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
24.	Automatic ECCS Switchover to Recirculation	Milstead	RES/DET/GSIB	NOTE 3(b)	3	12/31/95	NA
25.	Automatic Air Header Dump on BWR Scram System	Milstead	NRR/DSI/RSB	NOTE 3(a)		11/30/83	
26.	Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power	Emrit	NRR/DSI/ASB	17		11/30/83	NA
27.	Manual vs. Automated Actions	Pittman	NRR/DSI/RSB	B-17		11/30/83	NA
28.	Pressurized Thermal Shock	Emrit	NRR/DST/GIB	A-49		11/30/83	NA
29.	Bolting Degradation or Failure in Nuclear Power Plants	Vandermolen	RES/DSIR/EIB	NOTE 3(b)	2	06/30/95	NA
30.	Potential Generator Missiles - Generator Rotor Retaining Rings	Pittman	NRR/DE/MEB	DROP	1	12/31/85	NA
31.	Natural Circulation Cooldown	Riggs	NRR/DSI/RSB	I.C.1		11/30/83	NA
32.	Flow Blockage in Essential Equipment Caused by Corbicula	Emrit	NRR/DSI/ASB	51		11/30/83	NA
33.	Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power	Pittman	NRR/DSI/ICSB	A-47		11/30/83	NA
34.	RCS Leak	Riggs	NRR/DHFS/PSRB	DROP	1	06/30/84	NA
35.	Degradation of Internal Appurtenances in LWRs	Vandermolen	NRR/DSI/CPB, RSB	DROP	2	12/31/98	NA
36.	Loss of Service Water	Colmar	NRR/DSI/ASB, AEB, RSB	NOTE 3(b)	3	06/30/91	NA
37.	Steam Generator Overfill and Combined Primary and Secondary Blowdown	Colmar	NRR/DST/GIB, NRR/DSI/RSB	A-47, I.C.1(2)	1	06/30/85	NA
38.	Potential Recirculation System Failure as a Consequence of Ingestion of Containment Paint Flakes or Other Fine Debris	Emrit	RES/DSIR/RPSIB	DROP	2	06/30/95	NA
39.	Potential for Unacceptable Interaction Between the CRD System and Non-Essential Control Air System	Pittman	NRR/DSI/ASB	25	1	06/30/95	NA
40.	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	Colmar	NRR/DSI/ASB	NOTE 3(a)	1	06/30/84	B-65
41.	BWR Scram Discharge Volume Systems	Vandermolen	NRR/DSI/RSB	NOTE 3(a)		11/30/83	B-58
42.	Combination Primary/Secondary System LOCA	Riggs	NRR/DSI/RSB	I.C.1	1	06/30/85	NA
43.	Reliability of Air Systems	Milstead	RES/DSIR/RPSI	NOTE 3(a)	2	12/31/88	B-107

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
44.	Failure of Saltwater Cooling System	Milstead	NRR/DSI/ASB	43	1	12/31/88	NA
45.	Inoperability of Instrumentation Due to Extreme Cold Weather	Milstead	NRR/DSI/ICSB	NOTE 3(a)	2	06/30/91	
46.	Loss of 125 Volt DC Bus	Sege	NRR/DSI/PSB	76		11/30/83	NA
47.	Loss of Offsite Power	Thatcher	NRR/DSI/RSB, ASB	NOTE 3(b)		11/30/83	
48.	LCO for Class 1E Vital Instrument Buses in Operating Reactors	Sege	NRR/DSI/PSB	128	1	12/31/86	NA
49.	Interlocks and LCOs for Redundant Class 1E Tie-Breakers	Sege	NRR/DSI/PSB	128	3	06/30/91	NA
50.	Reactor Vessel Level Instrumentation in BWRs	Thatcher	NRR/DSI/RSB, ICSB	NOTE 3(b)	1	12/31/84	NA
51.	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	Emrit	RES/DE/EIB	NOTE 3(a)	1	12/31/89	L-913
52.	SSW Flow Blockage by Blue Mussels	Emrit	NRR/DSI/ASB	51		11/30/83	NA
53.	Consequences of a Postulated Flow Blockage Incident in a BWR	Vandermolen	NRR/DSI/CPB, RSB	DROP	1	12/31/84	NA
54.	Valve Operator-Related Events Occurring During 1978, 1979, and 1980	Colmar	NRR/DE/MEB	II.E.6.1	1	06/30/85	NA
55.	Failure of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	Emrit	NRR/DSI/PSB	DROP	2	06/30/91	NA
56.	Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event	Colmar	NRR/DHFS/HFEB	A-47, I.D.1		11/30/83	NA
57.	Effects of Fire Protection System Actuation on Safety-Related Equipment	Milstead	RES/DRA/ARGIB	NOTE 3(b)	3	06/30/95	NA
58.	Inadvertent Containment Flooding	Sege	NRR/DSI/ASB, CSB	DROP		11/30/83	
59.	Technical Specification Requirements for Plant Shutdown when Equipment for Safe Shutdown is Degraded or Inoperable	Emrit	NRR/DST/TSIP	RI (NOTE 5)	1	06/30/85	NA
60.	Lamellar Tearing of Reactor Systems Structural Supports	Colmar	NRR/DST/GIB	A-12		11/30/83	NA
61.	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments	Milstead	NRR/DSI/CSB	NOTE 3(b)	2	12/31/86	NA
62.	Reactor Systems Bolting Applications	Riggs	RES/DSIR/EIB	29	1	12/31/88	NA
63.	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis	Pittman	RES/DRA/ARGIB	DROP	1	06/30/90	NA
64.	Identification of Protection System Instrument Sensing Lines	Thatcher	NRR/DSI/ICSB	NOTE 3(b)		11/30/83	
65.	Probability of Core-Melt Due to Component Cooling Water System Failures	Vandermolen	NRR/DSI/ASB	23	1	12/31/86	NA
66.	Steam Generator Requirements	Riggs	NRR/DEST/EMTB	NOTE 3(b)	2	12/31/88	NA
67.	<u>Steam Generator Staff Actions</u>	-	-	-			

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
67.2.1	Integrity of Steam Generator Tube Sleeves	Riggs	NRR/DE/MEB	135	4	06/30/94	NA
67.3.1	Steam Generator Overfill	Riggs	NRR/DST/GIB NRR/DSI/RSB	A-47, I.C.1	4	06/30/94	NA
67.3.2	Pressurized Thermal Shock	Riggs	NRR/DST/GIB	A-49	4	06/30/94	NA
67.3.3	Improved Accident Monitoring	Riggs	NRR/DSI/ICSB	NOTE 3(a)	4	06/30/94	A-17
67.3.4	Reactor Vessel Inventory Measurement	Riggs	NRR/DSI/CPB	II.F.2	4	06/30/94	NA
67.4.1	RCP Trip	Riggs	NRR/DSI/RSB	II.K.3(5)	4	06/30/94	G-01
67.4.2	Control Room Design Review	Riggs	NRR/DHFS/HFEB	I.D.1	4	06/30/94	F-08
67.4.3	Emergency Operating Procedures	Riggs	NRC/DHFS/PSRB	I.C.1	4	06/30/94	F-05
67.5.1	~~Reassessment of Radiological Consequences~~	~~Riggs~~	~~RES/DRPS/RPSI~~	~~LI (NOTE 3)~~	~~4~~	~~06/30/94~~	~~NA~~
67.5.2	Reevaluation of SGTR Design Basis	Riggs	RES/DRPS/RPSI	LI (67.5.1)	4	06/30/94	NA
67.5.3	Secondary System Isolation	Riggs	NRR/DSI/RSB	DROP	4	06/30/94	NA
67.6.0	Organizational Responses	Riggs	OIE/DEPER/IRDB	III.A.3	4	06/30/94	NA
67.7.0	Improved Eddy Current Tests	Riggs	RES/DE/EIB	135	4	06/30/94	NA
67.8.0	Denting Criteria	Riggs	NRR/DE/MTEB	135	4	06/30/94	NA
67.9.0	Reactor Coolant System Pressure Control	Riggs	NRR/DSI/GIB NRR/DSI/RSB	A-45, I.C.1 (2,3)	4	06/30/94	NA
67.10.0	Supplemental Tube Inspections	Riggs	NRR/DL/ORAB	LI (NOTE 5)	4	06/30/94	NA
68.	Postulated Loss of Auxiliary Feedwater System Resulting from Turbine-Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	Pittman	NRR/DSI/ASB	124	3	06/30/91	NA
69.	Make-up Nozzle Cracking in B&W Plants	Colmar	NRR/DE/MEB, MTEB	NOTE 3(b)	1	12/31/84	B43
70.	PORV and Block Valve Reliability	Riggs	RES/DE/EIB	NOTE 3(a)	3	06/30/91	
71.	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety	Pittman	RES/DRA/ARGIB	DROP	3	06/30/01	NA
72.	Control Rod Drive Guide Tube Support Pin Failures	Riggs	RES	DROP	1	06/30/91	NA
73.	Detached Thermal Sleeves	Emrit	RES/DSIR/EIB	NOTE 3(a)	3	06/30/95	NA
74.	Reactor Coolant Activity Limits for Operating Reactors	Milstead	NRR/DSI/AEB	DROP	1	06/30/86	NA
75.	Generic Implications of ATWS Events at the Salem Nuclear Plant	Emrit	RES/DRA/ARGIB	NOTE 3(a)	1	06/30/90	B-76, B-77, B-78, B-79, B-80, B-81, B-82, B-85 B-86, B-87, B-88, B-89,

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
75.	(Cont.)						B-90, B-91, B-92, B-93
76.	Instrumentation and Control Power Interactions	Zimmerman	RES/DSIR/EIB	DROP	3	06/30/95	NA
77.	Flooding of Safety Equipment Compartments by Back-flow Through Floor Drains	Colmar	RES/DE/EIB	A-17		12/31/87	NA
78.	Monitoring of Fatigue Transient Limits for Reactor Coolant System	Rourk	RES/DET/GSIB	NOTE 3(b)	3	12/31/97	
79.	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	Colmar	RES/DSIR/EIB	NOTE 3(b)	3	06/30/95	NA
80.	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments	Vandermolen	NRR/DSI/RSB, ASB, CPB	DROP	2	12/31/98	NA
81.	Impact of Locked Doors and Barriers on Plant and Personnel Safety	Rourk	RES/DSIR/EIB	LOW	4	06/30/95	NA
82.	Beyond Design Basis Accidents in Spent Fuel Pools	Vandermolen	RES/DRPS/RPSI	NOTE 3(b)	1	06/30/89	NA
83.	Control Room Habitability	Emrit	RES/DST/AEB	NOTE 3(b)	2	06/30/96	NA
84.	CE PORVs	Riggs	RES/DSIR/RPSI	NOTE 3(b)	2	06/30/90	NA
85.	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	Milstead	NRR/DSI/CSB	DROP	2	06/30/91	NA
86.	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	06/30/88	B-84
87.	Failure of HPCI Steam Line Without Isolation	Pittman	RES/DSIR/EIB	NOTE 3(a)	2	06/30/95	
88.	Earthquakes and Emergency Planning	Riggs	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
89.	Stiff Pipe Clamps	Chang	RES/DSIR/EIB	LOW	2	06/30/95	NA
90.	Technical Specifications for Anticipatory Trips	Vandermolen	NRR/DSI/RSB, ICSB	DROP	2	12/31/98	NA
91.	Main Crankshaft Failures in Transamerica DeLaval Emergency Diesel Generators	Emrit	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
92.	Fuel Crumbling During LOCA	Vandermolen	NRR/DSI/RSB, CPB	DROP	1	12/31/98	NA
93.	Steam Binding of Auxiliary Feedwater Pumps	Pittman	RES/DRPS/RPSI	NOTE 3(a)		06/30/88	B-98
94.	Additional Low Temperature Overpressure Protection for Light Water Reactors	Pittman	RES/DSIR/RPSI	NOTE 3(a)		06/30/90	
95.	Loss of Effective Volume for Containment Recirculation Spray	Milstead	RES/DRA/ARGIB	NOTE 3(b)		06/30/90	NA
96.	RHR Suction Valve Testing	Milstead	RES/DRA/ARGIB	105		06/30/90	NA
97.	PWR Reactor Cavity Uncontrolled Exposures	Vandermolen	NRR/DSI/RAB	III.D.3.1		06/30/85	NA
98.	CRD Accumulator Check Valve Leakage	Pittman	NRR/DSI/ASB	DROP		06/30/85	NA
99.	RCS/RHR Suction Line Valve Interlock on PWRs	Pittman	RES/DRPS/RPSI	NOTE 3(a)	3	06/30/91	L-817

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
100.	Once-Through Steam Generator Level	Jackson	RES/DSIR/EIB	DROP	1	06/30/95	NA
101.	BWR Water Level Redundancy	Vandermolen	RES/DE/EIB	NOTE 3(b)	1	06/30/89	NA
102.	Human Error in Events Involving Wrong Unit or Wrong Train	Emrit	NRR/DLPQ/LPEB	NOTE 3(b)	2	12/31/88	NA
103.	Design for Probable Maximum Precipitation	Emrit	RES/DE/EIB	NOTE 3(a)	1	12/31/89	NA
104.	Reduction of Boron Dilution Requirements	Pittman	RES/DRA/ARGIB	DROP		12/31/88	NA
105.	Interfacing Systems LOCA at LWRs	Milstead	RES/DE/EIB	NOTE 3(b)	4	06/30/95	NA
106.	Piping and Use of Highly Combustible Gases in Vital Areas	Milstead	RES/DRPS	NOTE 3(b)	2	06/30/95	NA
107.	Main Transformer Failures	Milstead	RES/DRA/ARGIB	DROP	3	06/30/00	NA
108.	BWR Suppression Pool Temperature Limits	Colmar	NRR/DSI/CSB	RI (NOTE 3)		06/30/85	NA
109.	Reactor Vessel Closure Failure	Riggs	RES/DRA/ARGIB	DROP		06/30/90	NA
110.	Equipment Protective Devices on Engineered Safety Features	Diab	RES/DSIR/EIB	DROP	1	06/30/95	NA
111.	Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments	Riggs	NRR/DE/MTEB	LI (NOTE 5)	1	06/30/91	NA
112.	Westinghouse RPS Surveillance Frequencies and Out-of-Service Times	Pittman	NRR/DSI/ICSB	RI (NOTE 3)		12/31/85	NA
113.	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	Riggs	RES/DSIR/EIB	NOTE 3(b)	2	06/30/95	NA
114.	Seismic-Induced Relay Chatter	Riggs	NRR/DSRO/SPEB	A-46	1	06/30/91	NA
115.	Enhancement of the Reliability of Westinghouse Solid State Protection System	Milstead	RES/DRPS/RPSI	NOTE 3(b)	2	06/30/00	NA
116.	Accident Management	Pittman	RES/DRA/ARGIB	S		06/30/91	NA
117.	Allowable Time for Diverse Simultaneous Equipment Outages	Pittman	RES/DRA/ARGIB	DROP		06/30/90	NA
118.	Tendon Anchorage Failure	Shaukat	RES/DSIR/EIB	NOTE 3(a)	1	06/30/95	NA
119.	Piping Review Committee Recommendations	-	-	-			
119.1	Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads	Riggs	NRR/DE	RI (NOTE 3)	3	12/31/97	NA
119.2	Piping Damping Values	Riggs	NRR/DE	RI (DROP)	3	12/31/97	NA
119.3	Decoupling the OBE from the SSE	Riggs	NRR/DE	RI (S)	3	12/31/97	NA
119.4	BWR Piping Materials	Riggs	NRR/DE	RI (NOTE 5)	3	12/31/97	NA
119.5	Leak Detection Requirements	Riggs	NRR/DE	RI (NOTE 5)	3	12/31/97	NA
120.	On-Line Testability of Protection Systems	Milstead	RES/DRA/ARGIB	NOTE 3(b)	2	06/30/95	NA
121.	Hydrogen Control for Large, Dry PWR Containments	Emrit	RES/DSIR/SAIB	NOTE 3(b)	2	06/30/95	NA
122.	Davis-Besse Loss of All Feedwater Event of June 9, 1985: Short-Term Actions						
122.1	Potential Inability to Remove Reactor Decay Heat	-	-	-			
122.1.a	Failure of Isolation Valves in Closed Position	Vandermolen	NRR/DSRO/RSIB	124	4	12/31/98	NA
122.1.b	Recovery of Auxiliary Feedwater	Vandermolen	NRR/DSRO/RSIB	124	4	12/31/98	NA

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
122.1.c.	Interruption of Auxiliary Feedwater Flow	Vandermolen	NRR/DSRO/RSIB	124	4	12/31/98	NA
122.2	Initiating Feed-and-Bleed	Vandermolen	NRR/DEST/SRXB	NOTE 3(b)	4	12/31/98	NA
122.3	Physical Security System Constraints	Vandermolen	NRR/DSRO/SPEB	DROP	4	12/31/98	NA
123.	Deficiencies in the Regulations Governing DBA and Single-Failure Criteria Suggested by the Davis-Besse Event of June 9, 1985	Milstead	RES/DSIR/SAIB	DROP	1	06/30/95	NA
124.	Auxiliary Feedwater System Reliability	Emrit	NRR/DEST/SRXB	NOTE 3(a)	3	06/30/91	
125.	Davis-Besse Loss of All Feedwater Event of June 9, 1985: Long-Term Actions	-	-	-			
125.I.1	Availability of the Shift Technical Advisor	Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.2	PORV Reliability	-	-	-	7	12/31/98	
125.I.2.a	Need for a Test Program to Establish Reliability of the PORV	Vandermolen	NRR/DSRO/SPEB	70	7	12/31/98	NA
125.I.2.b	Need for PORV Surveillance Tests to Confirm Operational Readiness	Vandermolen	NRR/DSRO/SPEB	70	7	12/31/98	NA
125.I.2.c	Need for Additional Protection Against PORV Failure	Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.I.2.d	Capability of the PORV to Support Feed-and-Bleed	Vandermolen	NRR/DSRO/SPEB	A-45	7	12/31/98	NA
125.I.3	SPDS Availability	Milstead	RES/DRA/ARGIB	NOTE 3(b)	7	12/31/98	NA
125.I.4	Plant-Specific Simulator	Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.5	Safety Systems Tested in All Conditions Required by DBA	Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.6	Valve Torque Limit and Bypass Switch Settings	Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.7	Operator Training Adequacy	-	-	-			
125.I.7.a	Recover Failed Equipment	Pittman	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.7.b	Realistic Hands-On Training	Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.8	Procedures and Staffing for Reporting to NRC Emergency Response Center	Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.1	Need for Additional Actions on AFW Systems	-	-	-			
125.II.1.a	Two-Train AFW Unavailability	Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.1.b	Review Existing AFW Systems for Single Failure	Vandermolen	NRR/DSRO/SPEB	124	7	12/31/98	NA
125.II.1.c	NUREG-0737 Reliability Improvements	Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.1.d	AFW/Steam and Feedwater Rupture Control System/ICS Interactions in B&W Plants	Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.2	Adequacy of Existing Maintenance Requirements for Safety-Related Systems	Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.3	Review Steam/Feedline Break Mitigation Systems for Single Failure	V'Molen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.4	Thermal Stress of OTSG Components	Riggs	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.5	Thermal-Hydraulic Effects of Loss and Restoration of Feedwater on Primary System Components	Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
125.II.6	Reexamine PRA Estimates of Core Damage Risk from Loss of All Feedwater	Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.7	Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break	Vandermolen	RES/DRPS/RPSI	NOTE 3(b)	7	12/31/98	NA
125.II.8	Reassess Criteria for Feed-and-Bleed Initiation	Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.9	Enhanced Feed-and-Bleed Capability	Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.10	Hierarchy of Impromptu Operator Actions	Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.11	Recovery of Main Feedwater as Alternative to Auxiliary Feedwater	Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.12	Adequacy of Training Regarding PORV Operation	Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.13	Operator Job Aids	Pittman	NRR/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.14	Remote Operation of Equipment Which Must Now Be Operated Locally	Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
126.	Reliability of PWR Main Steam Safety Valves	Riggs	RES/DRA/ARGIB	LI (NOTE 3)		06/30/88	NA
127.	Maintenance and Testing of Manual Valves in Safety-Related Systems	Pittman	RES/DRA/ARGIB	LOW		12/31/87	NA
128.	Electrical Power Reliability	Emrit	RES/DSIR/EIB	NOTE 3(a)	2	06/30/95	
129.	Valve Interlocks to Prevent Vessel Drainage During Shutdown Cooling	Milstead	RES/DRA/ARGIB	DROP		06/30/90	NA
130.	Essential Service Water Pump Failures at Multiplant Sites	Riggs	RES/DSIR/RPSIB	NOTE 3(a)	2	12/31/95	
131.	Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse-Designed Plants	Riggs	RES/DRA/ARGIB	S	1	06/30/91	NA
132.	RHR System Inside Containment	Su	RES/DSIR/SAIB	DROP	1	12/31/95	NA
133.	Update Policy Statement on Nuclear Plant Staff Working Hours	Pittman	NRR/DLPQ/LHFB	LI (NOTE 3)	1	12/31/91	NA
134.	Rule on Degree and Experience Requirement	Pittman	RES/DRA/RDB	NOTE 3(b)		12/31/89	NA
135.	Steam Generator and Steam Line Overfill	Emrit	RES/DSIR/EIB	NOTE 3(b)	3	06/30/95	NA
136.	Storage and Use of Large Quantities of Cryogenic Combustibles On Site	Milstead	RES/DRA/ARGIB	LI (NOTE 3)		06/30/88	NA
137.	Refueling Cavity Seal Failure	Milstead	RES/DRA/ARGIB	DROP		06/30/90	NA
138.	Deinerting of BWR Mark I and II Containments During Power Operations Upon Discovery of RCS Leakage or a Train of a Safety System Inoperable	Milstead	RES/DSIR/SAIB	DROP	2	12/31/98	NA
139.	Thinning of Carbon Steel Piping in LWRs	Riggs	RES/DRA/ARGIB	RI (NOTE 3)	1	06/30/95	NA
140.	Fission Product Removal Systems	Riggs	RES/DRA/ARGIB	DROP		06/30/90	NA
141.	Large-Break LOCA With Consequential SGTR	Riggs	RES/DRA/ARGIB	DROP		06/30/90	NA
142.	Leakage Through Electrical Isolators in Instrumentation Circuits	Milstead	RES/DSIR/EIB	NOTE 3(b)	4	12/31/97	NA
143.	Availability of Chilled Water Systems and Room Cooling	Milstead	RES/DRA/ARGIB	NOTE 3(b)	2	06/30/95	NA

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
144.	Scram Without a Turbine/Generator Trip	Hrabal	RES/DSIR/EIB	DROP	2	12/31/98	NA
145.	Actions to Reduce Common Cause Failures	Rasmuson	RES/DST/PRAB	NOTE 3(b)	3	06/30/00	NA
146.	Support Flexibility of Equipment and Components	Chang	RES/DSIR/EIB	NOTE 3(b)	2	06/30/95	NA
147.	Fire-Induced Alternate Shutdown/Control Room Panel Interactions	Milstead	RES/DSIR/SAIB	LI (NOTE 3)	1	06/30/94	NA
148.	Smoke Control and Manual Fire-Fighting Effectiveness	Basdekas	RES/DSIR/RPSIB	LI (NOTE 3)	1	06/30/00	NA
149.	Adequacy of Fire Barriers	Emrit	RES/DSIR/EIB	DROP	2	12/31/98	NA
150.	Overpressurization of Containment Penetrations	Milstead	RES/DSIR/SAIB	DROP	1	06/30/95	NA
151.	Reliability of Anticipated Transient Without SCRAM Recirculation Pump Trip in BWRs	Milstead	RES/DSIR/SAIB	NOTE 3(b)	2	06/30/95	NA
152.	Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads	Emrit	RES/DSIR/EIB	DROP	3	06/30/01	NA
153.	Loss of Essential Service Water in LWRs	Riggs	RES/DRA/ARGIB	NOTE 3(b)	2	12/31/95	NA
154.	Adequacy of Emergency and Essential Lighting	Woods	RES/DSIR/SAIB	DROP	2	12/31/98	NA
155.	<u>Generic Concerns Arising from TMI-2 Cleanup</u>	-	-	-			
155.1	More Realistic Source Term Assumptions	Emrit	RES/DST/AEB	NOTE 3(a)	2	06/30/95	NA
155.2	Establish Licensing Requirements for Non-Operating Facilities	Emrit	RES/DSIR/EIB	RI (NOTE 5)	2	06/30/95	NA
155.3	Improve Design Requirements for Nuclear Facilities	Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.4	Improve Criticality Calculations	Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.5	More Realistic Severe Reactor Accident Scenario	Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.6	Improve Decontamination Regulations	Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.7	Improve Decommissioning Regulations	Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
156.	<u>Systematic Evaluation Program</u>	-	-	-			
156.1.1	Settlement of Foundations and Buried Equipment	Chang	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.1.2	Dam Integrity and Site Flooding	Chen	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.1.3	Site Hydrology and Ability to Withstand Floods	Chen	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.1.4	Industrial Hazards	Ferrell	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.1.5	Tornado Missiles	Chen	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.1.6	Turbine Missiles	Emrit	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.2.1	Severe Weather Effects on Structures	Chen	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.2.2	Design Codes, Criteria, and Load Combinations	Kirkwood	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.2.3	Containment Design and Inspection	Shaukat	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.2.4	Seismic Design of Structures, Systems, and Components	Chen	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.1.1	Shutdown Systems	Woods	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.1.2	Electrical Instrumentation and Controls	Woods	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.2	Service and Cooling Water Systems	Su	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.3	Ventilation Systems	Burdick	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.4	Isolation of High and Low Pressure Systems	Burdick	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.5	Automatic ECCS Switchover	Milstead	RES/DSIR/SAIB	24	7	06/30/01	NA
156.3.6.1	Emergency AC Power	Emrit	RES/DSIR/EIB	DROP	7	06/30/01	NA

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
156.3.6.2	Emergency DC Power	Rourk	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.3.8	Shared Systems	Emrit	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.4.1	RPS and ESFS Isolation	Emrit	RES/DSIR/EIB	142	7	06/30/01	NA
156.4.2	Testing of the RPS and ESFS	Chang	RES/DSIR/SAIB	120	7	06/30/01	NA
156.6.1	Pipe Break Effects on Systems and Components	Page	RES/DET/GSIB	HIGH	7	06/30/01	
157.	Containment Performance	Shaperow	RES/DSIR/SAIB	NOTE 3(b)		06/30/95	NA
158.	Performance of Power-Operated Valves Under Design Basis Conditions	Hrabal	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
159.	Qualification of Safety-Related Pumps While Running on Minimum Flow	Su	RES/DSIR/SAIB	DROP	1	06/30/95	NA
160.	Spurious Actions of Instrumentation Upon Restoration of Power	Rourk	RES/DSIR/EIB	DROP	1	06/30/95	NA
161.	Use of Non-Safety-Related Power Supplies in Safety-Related Circuits	Rourk	RES/DSIR/EIB	DROP	1	06/30/95	NA
162.	Inadequate Technical Specifications for Shared Systems at Multiplant Sites When One Unit Is Shut Down	Cheh	RES/DSIR/SAIB	DROP	1	06/30/95	NA
163.	Multiple Steam Generator Tube Leakage	Coffman	RES/DET/GSIB	HIGH		12/31/97	
164.	Neutron Fluence in Reactor Vessel	Emrit	RES/DSIR/EIB	DROP	1	06/30/95	NA
165.	Safety and Safety/Relief Valve Reliability	Hrabal	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
166.	Adequacy of Fatigue Life of Metal Components	Emrit	NRR/DE/EMEB	NOTE 3(b)	2	12/31/97	NA
167.	Hydrogen Storage Facility Separation	Burdick	RES/DSIR/SAIB	LOW	1	06/30/95	NA
168.	Environmental Qualification of Electrical Equipment	Emrit	NRR/DSSA/SPLB	HIGH	2	12/31/98	
169.	BWR MSIV Common Mode Failure Due to Loss of Accumulator Pressure	Emrit	RES/DET/GSIB	DROP	1	06/30/00	NA
170.	Fuel Damage Criteria for High Burnup Fuel	Emrit	RES/DET/GSIB	NOTE 3(b)	2	06/30/01	NA
171.	ESF Failure from LOOP Subsequent to a LOCA	Rourk	RES/DET/GSIB	NOTE 3(b)	1	12/31/98	NA
172.	Multiple System Responses Program	Emrit	RES/DET/GSIB	HIGH	1	12/31/98	
173.	Spent Fuel Storage Pool	-	-				
173.A	Operating Facilities	Emrit	RES/DET/GSIB	HIGH	3	06/30/00	
173.B	Permanently Shutdown Facilities	Emrit	RES/DET/GSIB	NOTE 3(b)	3	06/30/00	NA
174.	Fastener Gaging Practices	-	-				
174.A	SONGS Employees' Concern	Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
174.B	Johnson Gage Company Concern	Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
175.	Nuclear Power Plant Shift Staffing	Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
176.	Loss of Fill-Oil in Rosemount Transmitters	Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
177.	Vehicle Intrusion at TMI	Emrit	RES/DET/GSIB	NOTE 3(a)	1	06/30/00	NA
178.	Effect of Hurricane Andrew on Turkey Point	Emrit	RES/DET/GSIB	LI (NOTE 3)	2	06/30/00	
179.	Core Performance	Emrit	RES/DET/GSIB	LI (NOTE 5)	1	06/30/00	
180.	Notice of Enforcement Discretion	Emrit	RES/DET/GSIB	LI (NOTE 3)	1	06/30/00	
181.	Fire Protection	Emrit	RES/DET/GSIB	LI (NOTE 5)	1	06/30/00	

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
182.	General Electric Extended Power Uprate	Emrit	RES/DET/GSIB	RI (NOTE 5)	1	06/30/00	
183.	Cycle-Specific Parameter Limits in Technical Specifications	Emrit	RES/DET/GSIB	RI (NOTE 3)	2	06/30/00	
184.	Endangered Species	Emrit	RES/DET/GSIB	EI (NOTE 5)	1	06/30/00	
185.	Control of Recriticality Following Small-Break LOCA In PWRs	Vandermolen	RES/DSARE/REAHFB	HIGH		06/30/01	
186.	Potential Risk and Consequences of Heavy Load Drops	Lloyd	RES/DSARE/REAHFB	NOTE 4		(Later)	
187.	The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump in Nuclear Power Plants	Vandermolen	RES/DSARE/REAHFB	DROP		06/30/01	NA
188.	Steam Generator Tube Leaks/Ruptures Concurrent with Containment Bypass	TBD	RES/DSARE/REAHFB	NOTE 4		(Later)	
189.	Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Combustion During A Severe Accident	TBD	RES/DSARE/REAHFB	NOTE 4		(Later)	
190.	Fatigue Evaluation of Metal Components for 60-Year Plant Life	Shaukat	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
191.	Assessment of Debris Accumulation on PWR Sump Performance	Marshall	RES/DET/GSIB	HIGH	1	12/31/98	
	HUMAN FACTORS ISSUES						
HF1	STAFFING AND QUALIFICATIONS						
HF1.1	Shift Staffing	Pittman	RES/DRPS/RHFB	NOTE 3(a)	2	06/30/89	
HF1.2	Engineering Expertise on Shift	Pittman	NRR/DHFT/HFIB	NOTE 3(b)	2	06/30/89	
HF1.3	Guidance on Limits and Conditions of Shift Work	Pittman	NRR/DHFT/HFIB	NOTE 3(b)	2	06/30/89	
HF2	TRAINING						
HF2.1	Evaluate Industry Training	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF2.2	Evaluate INPO Accreditation	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF2.3	Revise SRP Section 13.2	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF3	OPERATOR LICENSING EXAMINATIONS						
HF3.1	Develop Job Knowledge Catalog	Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF3.2	Develop License Examination Handbook	Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF3.3	Develop Criteria for Nuclear Power Plant Simulators	Pittman	NRR/DHFT/HFIB	I.A.4.2(4)	2	12/31/87	NA
HF3.4	Examination Requirements	Pittman	NRR/DHFT/HFIB	I.A.2.6(1)	2	12/31/87	NA
HF3.5	Develop Computerized Exam System	Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
HF4	PROCEDURES						
HF4.1	Inspection Procedure for Upgraded Emergency Operating Procedures	Pittman	NRR/DLPQ/LHFB	NOTE 3(b)	6	06/30/95	NA
HF4.2	Procedures Generation Package Effectiveness Evaluation	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	6	06/30/95	NA
HF4.3	Criteria for Safety-Related Operator Actions	Pittman	NRR/DHFT/HFIB	B-17	6	06/30/95	NA
HF4.4	Guidelines for Upgrading Other Procedures	Pittman	RES/DRPS/RHFB	NOTE 3(b)	6	06/30/95	NA
HF4.5	Application of Automation and Artificial Intelligence	Pittman	NRR/DHFT/HFIB	HF5.2	6	06/30/95	NA
HF5	MAN-MACHINE INTERFACE						
HF5.1	Local Control Stations	Pittman	RES/DRPS/RHFB	NOTE 3(b)	4	06/30/95	NA
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	Pittman	RES/DRPS/RHFB	NOTE 3(b)	4	06/30/95	NA
HF5.3	Evaluation of Operational Aid Systems	Pittman	NRR/DHFT/HFIB	HF5.2	4	06/30/95	NA
HF5.4	Computers and Computer Displays	Pittman	NRR/DHFT/HFIB	HF5.2	4	06/30/95	NA
HF6	MANAGEMENT AND ORGANIZATION						
HF6.1	Develop Regulatory Position on Management and Organization	Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF6.2	Regulatory Position on Management and Organization at Operating Reactors	Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF7	HUMAN RELIABILITY						
HF7.1	Human Error Data Acquisition	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.2	Human Error Data Storage and Retrieval	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.3	Reliability Evaluation Specialist Aids	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.4	Safety Event Analysis Results Applications	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF8	Maintenance and Surveillance Program	Pittman	NRR/DLPQ/LPEB	NOTE 3(b)	2	06/30/88	NA
		CHERNOBYL ISSUES					
CH1	ADMINISTRATIVE CONTROLS AND OPERATIONAL PRACTICES						
CH1.1	Administrative Controls to Ensure That Procedures Are Followed and That Procedures Are Adequate	-	-				
CH1.1A	Symptom-Based EOPs	Emrit	NRR/DLPQ/LHFB	LI (NOTE 5)		06/30/89	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
CH1.1B	Procedure Violations	Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.2	Approval of Tests and Other Unusual Operations	-	-				
CH1.2A	Test, Change, and Experiment Review Guidelines	Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.2B	NRC Testing Requirements	Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.3	Bypassing Safety Systems	-	-				
CH1.3A	Revise Regulatory Guide 1.47	Emrit	RES/DE/EMEB	LI (NOTE 5)		06/30/89	NA
CH1.4	Availability of Engineered Safety Features	-	-				
CH1.4A	Engineered Safety Feature Availability	Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.4B	Technical Specifications Bases	Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.4C	Low Power and Shutdown	Emrit	RES/DSR/PRAB	LI (NOTE 5)		06/30/89	NA
CH1.5	Operating Staff Attitudes Toward Safety	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH1.6	Management Systems	-	-				
CH1.6A	Assessment of NRC Requirements on Management	Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.7	Accident Management	-	-				
CH1.7A	Accident Management	Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH2	DESIGN						
CH2.1	Reactivity Accidents	-	-				
CH2.1A	Reactivity Transients	Emrit	RES/DSR/RPSB	LI (NOTE 5)		06/30/89	NA
CH2.2	Accidents at Low Power and at Zero Power	Emrit	RES/DRA/ARGIB	CH1.4		06/30/89	NA
CH2.3	Miltiple-Unit Protection	-	-				
CH2.3A	Control Room Habitability	Emrit	RES/DRA/ARGIB	83		06/30/89	NA
CH2.3B	Contamination Outside Control Room	Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.3C	Smoke Control	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH2.3D	Shared Shutdown Systems	Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.4	Fire Protection	-	-				
CH2.4A	Firefighting With Radiation Present	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH3	CONTAINMENT						
CH3.1	Containment Performance During Severe Accidents	-	-				
CH3.1A	Containment Performance	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH3.2	Filtered Venting	-	-				
CH3.2A	Filtered Venting	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4	EMERGENCY PLANNING						
CH4.1	Size of the Emergency Planning Zones	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH4.2	Medical Services	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH4.3	Ingestion Pathway Measures	-	-				

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
CH4.3A	Ingestion Pathway Protective Measures	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4.4	Decontamination and Relocation	-	-				
CH4.4A	Decontamination	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4.4B	Relocation	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
<u>CH5</u>	<u>SEVERE ACCIDENT PHENOMENA</u>						
CH5.1	Source Term	-	-				
CH5.1A	Mechanical Dispersal in Fission Product Release	Emrit	RES/DSR/AEB	LI (NOTE 5)		06/30/89	NA
CH5.1B	Stripping in Fission Product Release	Emrit	RES/DSR/AEB	LI (NOTE 5)		06/30/89	NA
CH5.2	Steam Explosions	-	-				
CH5.2A	Steam Explosions	Emrit	RES/DSR/AEB	LI (NOTE 5)		06/30/89	NA
CH5.3	Combustible Gas	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
<u>CH6</u>	<u>GRAPHITE-MODERATED REACTORS</u>						
CH6.1	Graphite-Moderated Reactors	-	-				
CH6.1A	The Fort St. Vrain Reactor and the Modular HTGR	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH6.1B	Structural Graphite Experiments	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH6.2	Assessment	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA

TABLE III

SUMMARY OF THE PRIORITIZATION OF ALL TMI ACTION PLAN ITEMS, TASK ACTION PLAN ITEMS, NEW GENERIC ISSUES, HUMAN FACTORS ISSUES, AND CHERNOBYL ISSUES

Legend

NOTES:	1 - Possible Resolution Identified for Evaluation
	2 - Resolution Available
	3 - Resolution Resulted in either the Establishment of New Requirements or No New Requirements
	4 - Issues to be Prioritized in the Future
	5 - Issues that are not GSIs but Should be Assigned Resources for Completion
DROP	- GSI Dropped from Further Pursuit
EI	- Environmental Issue
GSI	- Generic Safety Issue
HIGH	- High Safety Priority
I	- TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737
LI	- Licensing Issue
LOW	- Low Safety Priority
MEDIUM	- Medium Safety Priority
RI	- Regulatory Impact Issue
USI	- Unresolved Safety Issue

TABLE III

TABLE III (Continued)

ACTION ITEM/ISSUE GROUP	I	S	RESOLVED STAGES			USI	HIGH	MEDIUM	LOW	DROP	NOTE 4	NOTE 5	TOTAL
			NOTE 1	NOTE 2	NOTE 3								
TMI ACTION PLAN ITEM (369)													
GSI	84	46	0	0	135	0	0	0	12	9	-	-	286
LI	-	0	-	-	75	-	-	-	-	-	-	8	83
TASK ACTION PLAN ITEMS (142)													
USI	-	-	-	-	27	0	-	-	-	-	-	-	27
GSI	-	20	0	0	36	-	0	0	0	14	0	-	70
RI	-	-	-	-	6	-	-	-	-	-	-	1	7
LI	-	-	-	-	11	-	-	-	-	-	-	12	23
EI	-	-	-	-	13	-	-	-	-	-	-	2	15
NEW GENERIC ISSUES (271)													
GSI	-	54	0	0	80	0	8	0	4	97	2	-	245
RI	-	1	-	-	5	-	-	-	-	1	-	5	12
LI	-	1	-	-	8	-	-	-	-	-	-	4	13
EI	-	-	-	-	-	-	-	-	-	-	-	1	1
HUMAN FACTORS ISSUES (27)													
GSI	-	8	0	0	8	0	0	0	0	0	-	-	16
LI	-	-	-	-	3	-	-	-	-	-	-	8	11
CHERNOBYL ISSUES (32)													
LI	-	2	-	-	7	-	-	-	-	-	-	23	32
TOTAL:	84	132	0	0	414	0	8	0	16	121	2	64	841

TABLE IV

LISTING OF AEOD REPORTS AND RELATED GENERIC ISSUES

This listing shows all AEOD reports that have been addressed either as completely new safety issues or as part of existing safety issues. It should be noted that, in some cases, more than one AEOD report has been generated on a single topic. However, all AEOD reports related to the identified safety issues are listed alphanumerically including those that have been superseded by other AEOD reports. The following is a description of the types of AEOD reports:

C - Reactor Case Study
E - Reactor Engineering Evaluation
S - Special Study Report
T - Technical Review Report

AEOD REPORT NO.	AEOD REPORT TITLE	RELATED ISSUE NO.	RELATED AEOD REPORT
C001	Report on the Browns Ferry 3 Partial Failure to Scram Event on June 28, 1980	41	-
C003	Report on Loss of Offsite Power Event at Arkansas Nuclear One, Units 1 and 2	47	-
C004	AEOD Actions Concerning the Crystal River 3 Loss of Non-Nuclear Instrumentation and Integrated Control System Power on February 26, 1980	33	E122
C005	AEOD Observations and Recommendations Concerning the Problem of Steam Generator Overfill and Combined Primary and Secondary Side Blowdown	37, 42	-
C101	Report on the Saint Lucie 1 Natural Circulation Cooldown on June 11, 1980	31	-
C102	H. B. Robinson Reactor Coolant System Leak on January 29, 1981	34	-
C103	AEOD Safety Concerns Associated with Pipe Breaks in the BWR Scram System	40	-
C104	Millstone Unit 2 Loss of 125 V DC Bus Event on January 2, 1981	46	-
C105	Report on the Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980	36	-

TABLE IV

AEOD REPORT NO.	AEOD REPORT TITLE	RELATED ISSUE NO.	RELATED AEOD REPORT
C201	Safety Concern Associated with Reactor Vessel Level Instrumentation in Boiling Water Reactors	50, 101	-
C202	Report on Service Water System Flow Blockages by Bivalve Mollusks at Arkansas Nuclear One and Brunswick	32	E016
C203	Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980	54	E305
C204	San Onofre Unit 1 Loss of Salt Water Cooling Event of March 10, 1980	44	-
C205	Abnormal Transient Operating Guidelines (ATOG) as Applied to the April 1981 Overfill Event at Arkansas Nuclear One, Unit 1	56	-
C301	Failures of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	55	-
C401	Low Temperature Overpressure Events at Turkey Point Unit 4	94	E426
C403	Edwin I. Hatch Unit No. 2 Plant Systems Interaction Event on August 25, 1982	85	E322
C404	Steam Binding of Auxiliary Feedwater Pumps	93	E325
C501	Safety Implications Associated with In-Plant Pressurized Gas Storage and Distribution Systems in Nuclear Power Plants	106	E902
C503	Decay Heat Removal Problems at U.S. Pressurized Water Reactors	99	-
C701	Air Systems Reliability	43	E123
E002	BWR Jet Pump Integrity	12	-
E005	Operational Restrictions for Class 1E 120 VAC Vital Instrument Buses	48	-
E007	Potential for Unacceptable Interaction Between the Control Rod Drive System and Non-Essential Control Air System at the Browns Ferry Plant	39	-
E010	Tie Breaker Between Redundant Class 1E Buses - Point Beach Nuclear Plant, Units 1 and 2	49	-

TABLE IV

AEOD REPORT NO.	AEOD REPORT TITLE	RELATED ISSUE NO.	RELATED AEOD REPORT
E011	Concerns Relating to the Integrity of a Polymer Coating for Surfaces Inside Containment	38	-
E016	Flow Blockage in Essential Equipment at ANO Caused by Corbicula sp. (Asiatic Clams)	32	C202
E101	Degradation of Internal Appurtenances in LWR Piping	35	-
E112	Inoperability of Instrumentation Due to Extreme Cold Weather	45	E226
E122	AEOD Concern Regarding Inadvertent Opening of Atmospheric Dump Valves on B&W Plants During Loss of ICS/NNI Power	33	C004
E123	Common Cause Failure Potential at Rancho Seco - Desiccant Contamination of Air Lines	43	C701
E204	Effects of Fire Protection System Actuation on Safety-Related Equipment	57	-
E209	Generator Rotor Retaining Ring as a Potential Missile (Incident at Barseback 1 on 4/13/79)	30	-
E215	Engineering Evaluation of the Salt Service Water System Flow Blockage at the Pilgrim Nuclear Power Station by Blue Mussels	52	-
E226	Inoperability of Instrumentation Due to Extreme Cold Weather	45	E112
E304	Investigation of Backflow Protection in Common Equipment and Floor Drain Systems to Prevent Flooding of Vital Equipment in Safety-Related Compartments	77	-
E305	Inoperable Motor-Operational Valve Assemblies Due to Premature Degradation of Motors and/or Improper Limit Switch/Torque Switch Adjustment	54	C203
E322	Damage to Vacuum Breaker Valves as a Result of Relief Valve Lifting	85	C403
E325	Vapor Binding of Auxiliary Feedwater Pumps at Robinson 2	93	C404
E414	Stuck Open Isolation Check Valve on the Residual Heat Removal System at Hatch Unit 2	105	-

TABLE IV

AEOD REPORT NO.	AEOD REPORT TITLE	RELATED ISSUE NO.	RELATED AEOD REPORT
E417	Loosening of Flange Bolts on RHR Heat Exchanger Leading to Primary to Secondary Side Leakage	C-9	-
E426	Single Failure Vulnerability of Power Operated Relief Valve (PORV) Actuation Circuitry for Low Temperature Overpressure Protection (LTOP)	94	C401
E609	Inadvertent Draining of Reactor Vessel During Shutdown Cooling Operation	129	-
E804	Reliability of Non-Safety-Related Field Breakers During ATWS Events	151	-
E807	Pump Damage Due to Low Flow Cavitation	159	-
S401	Human Error in Events Involving Wrong Unit or Wrong Train	102	-
S92-02	Safety and Safety/Relief Valve Reliability	165	-
T302	Postulated Loss of Auxiliary Feedwater System Resulting from a Turbine Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	68	-
T305	Flow Blockage in Essential Raw Cooling Water System Due to Asiatic Clam Intrusion at Sequoyah 1	51	-
T420	Failure of an Isolation Valve of the Reactor Core Isolation Cooling System to Open Against Operating Reactor Pressure	87	-

TABLE V
SUMMARY OF CONSOLIDATED GENERIC ISSUES

This table shows the consolidation of those issues whose technical concerns were found to be addressed either partially or completely in other (major issues). The table reflects the findings of the prioritization process that are summarized in Table II.

MAJOR ITEM/ISSUE NO.	PRIORITY	ITEM(S)/ISSUE(S) COVERED IN MAJOR ITEMS
TMI ACTION PLAN ITEMS		
I.A.1.3	I	II.K.3(53)
I.A.2.2	NOTE 3(b)	I.A.2.6(3) [II.K.3(56)]
I.A.2.6(1)	NOTE 3(a)	I.B.1.1.(6); I.B.1.1(7); HF3.4
I.A.3.1	I	II.K.3(56)
I.A.4.1(2)	NOTE 3(a)	II.K.3(54)
I.A.4.2(4)	NOTE 3(a)	HF3.3
I.B.1.1 (1,2,3,4)	NOTE 3(b)	II.J.3.1; II.J.3.2; II.K.3(52); HF6.1; HF6.2
I.C.1	-	8; 18; 31; 42; 67.3.1; 67.4.3
I.C.1(2)	I	37; 67.9.0
I.C.1(3)	I	II.K.2(12); II.K.2(18); II.K.3(6); II.K.3(35); II.K.3(36); II.K.3(37); II.K.3(38); II.K.3(39); II.K.3(41); II.K.3(42); II.K.3(47); II.K.3(55); 67.9.0
I.C.2	I	II.K.3(52)
I.C.5	I	II.K.3(52)
I.C.7	I	II.K.3(50)
I.C.8	I	II.K.3(49)
I.C.9	NOTE 3(b)	II.K.3(49); II.K.3(50); II.K.3(51)
I.D.1	I	56; 67.4.2

TABLE V

MAJOR ITEM/ISSUE NO.	PRIORITY	ITEM(S)/ISSUE(S) COVERED IN MAJOR ITEMS
I.D.2	I	II.K.3(23); II.K.3(55)
I.D.3	NOTE 3(b)	II.K.3(55)
I.F.1	NOTE 3(b)	5
II.B.8	NOTE 3(a)	II.B.7
II.C.1	NOTE 3(b)	II.K.3(4); II.K.3(8); II.K.3(33); II.K.3(48)
II.C.2	NOTE 3(b)	II.K.3(4); II.K.3(48)
II.E.1.1	I	II.K.2(8)
II.E.1.2	I	II.K.2(8)
II.E.2.2	NOTE 3(b)	II.K.3(32); II.K.3(34); II.K.3(47)
II.E.6.1	NOTE 3(a)	54
II.F.2	I	II.K.3(6); 67.3.4
II.F.3	NOTE 3(a)	II.K.3(6); A-34
II.H.2	NOTE 3(b)	II.H.3
II.K.2(15)	I	II.K.3(43)
II.K.2(16)	I	II.K.3(40)
II.K.3(5)	I	9; 67.4.1
II.K.3(17)	I	II.E.2.1[II.K.3(26)]
III.A.1.2(1)	I	II.K.3(23)
III.A.3.1	NOTE 3(b)	B-71
III.A.3	-	67.6.0
III.A.3.4	NOTE 3(b)	II.K.3(23)

TABLE V

MAJOR ITEM/ISSUE NO.	PRIORITY	ITEM(S)/ISSUE(S) COVERED IN MAJOR ITEMS
III.D.1.1(1)	I	B-69
III.D.2.1	LOW	B-67
III.D.2.5	NOTE 3(b)	III.D.2.2(2); III.D.2.2(3); III.D.2.2(4)
III.D.3.1	NOTE 3(b)	B-34; 97
V.A.1	LI (NOTE 3)	II.A.2
TASK ACTION PLAN ITEMS		
A-2	NOTE 3(a)	B-52
A-12	NOTE 3(a)	60
A-17	NOTE 3(b)	II.C.3[II.K.3(4)]; C-13; 77
A-18	DROP	B-16
A-37	DROP	A-32; 11
A-38	DROP	A-32
A-40	NOTE 3(a)	B-51
A-43	NOTE 3(a)	B-18; C-3
A-44	NOTE 3(a)	B-57
A-45	NOTE 3(b)	II.E.3.2[B-4]; II.E.3.3[II.K.3(8)]; II.E.3.5; 67.9.0; 125.I.2.d
A-46	NOTE 3(a)	B-24; 114
A-47	NOTE 3(a)	19; 33; 37; 56; 67.3.1
A-48	NOTE 3(a)	B-14
A-49	NOTE 3(a)	28; 67.3.2
B-2	EI (NOTE 3)	B-45

TABLE V

MAJOR ITEM/ISSUE NO.	PRIORITY	ITEM(S)/ISSUE(S) COVERED IN MAJOR ITEMS
B-17	NOTE 3(b)	27; HF4.3
B-68	DROP	A-32
C-8	NOTE 3(b)	16
C-12	NOTE 3(b)	B-73
NEW GENERIC ISSUES		
17	DROP	26
23	NOTE 3(b)	65
24	NOTE 3(b)	156.3.5
25	NOTE 3(a)	39
29	NOTE 3(b)	62
43	NOTE 3(a)	44
51	NOTE 3(a)	32; 52
67.5.2	LI(NOTE 3)	36; 67.5.1
70	NOTE 3(a)	125.I.2.a; 125.I.2.b
75	NOTE 3(a)	I.B.1.1(6); I.B.1.1(7)
76	DROP	46
83	NOTE 3(b)	CH 2.3A
105	NOTE 3(b)	96
119.1	RI(NOTE 3)	B-6
120	NOTE 3(b)	156.4.2
124	NOTE 3(a)	68; 122.1.a; 122.1.b; 122.1.c; 122.II.1.b

TABLE V

MAJOR ITEM/ISSUE NO.	PRIORITY	ITEM(S)/ISSUE(S) COVERED IN MAJOR ITEMS
128	NOTE 3(a)	48; 49; A-30
135	NOTE 3(b)	67.2.1; 67.7.0; 67.8.0
142	NOTE 3(b)	156.4.1
153	NOTE 3(b)	B-32
HUMAN FACTORS ISSUES		
HF5.2	NOTE 3(b)	HF4.5; HF5.3; HF5.4
CHERNOBYL ISSUES		
CH1.4	LI(NOTE 5)	CH2.2

ISSUE 71: FAILURE OF RESIN DEMINERALIZER SYSTEMS AND THEIR EFFECTS ON NUCLEAR POWER PLANT SAFETY

DESCRIPTION

Historical Background

This issue was raised[516] by DSI/NRR in August 1982[516] following a search of LERs which suggested that additional licensing attention was needed for certain ancillary power plant equipment. The available information showed that failures of resin bed demineralizer sub-systems occurred within the process systems (both nuclear and non-nuclear) of nuclear power plants. These process systems, by definition, do not directly perform any reactor protection or engineered safeguard functions, yet their failure could seriously impair the capability of safety grade systems to perform by rendering their redundant trains inoperable (i.e., causing a common mode failure). The chief concern was the possibility that these types of events may not be bounded by the current licensing basis for nuclear power plants and could cause plants to be inadequately protected. The types of failures considered were: (1) introduction of resin into other areas of the system (either by breakthrough of the resin during normal operation or by improper recharging); (2) introduction of gas into other areas of the system by improper recharging; and (3) loss of water chemistry.

Safety Significance

Failures of resin demineralizers can be caused by operator error or by equipment failure and have produced the following: (1) clogging of pump strainers (due to resin introduction into the system) and the subsequent tripping of the pumps; and (2) introduction of gas into systems (subsequently causing pump trips) due to improper demineralizer back-flushing. Systems containing demineralizers are:

PWR
(a) Chemical and Volume Control System
(b) Condensate and Feedwater System
(c) Component Cooling Water System
(d) Service Water System
(e) Spent Fuel Pool Cooling and Purification System

Two failure modes were considered: (1) introduction of resin or gas into a system which subsequently causes one or more additional failures; and (2) loss of water chemistry control which affects corrosion rates. The first failure mode can be caused by operator error or by equipment failure and has the potential of affecting the following systems:

PWR
(a) High Head Safety Injection System
(b) Condensate and Feedwater System
(c) RHR System
(d) Containment Spray System
(e) Chemical and Volume Control System
(f) Component Cooling Water System

(g) Spent Fuel Pool Cooling and Purification System

BWR
(a) Sensor Output from Reactor Protection System
(b) Condensate and Feedwater System
(c) RHR System
(d) Containment Cooling System
(e) Reactor Water Cleanup System
(f) Emergency Equipment Cooling Water System
(g) Fuel Pool Cooling and Cleanup System

Since some of these systems perform a safety function or support systems which perform a safety function, their failure could reduce the ability of a plant to maintain safe shutdown conditions. The following are a few examples of where demineralizer failures caused a loss of safety grade equipment.

(1) Following a review of a TMI-2 event that occurred in September 1977 during hot functional testing prior to fuel loading, it was concluded that, had the reactor been fueled and at power when the event occurred, there might have been core uncovery followed by fuel damage.[516] TMI-2 has a full-flow, condensate polishing system in the condensate and feedwater system and, as a result of its malfunction, resin from the system was carried over into the plant's demineralized water system from which it migrated to all other parts of the plant, including the nuclear steam supply system and the turbine. The most significant result was that the resin clogged the strainers to all of the circulating pumps in the nuclear service closed cooling water system causing them to trip. This removed essential cooling water from all related reactor pressure and ESF systems and components and also all non-essential nuclear systems and components, i.e., RCPs, spent fuel coolers, instrument air compressors, and after-coolers. The loss of coolant to the RCPs caused the pumps to trip and the pressurizer heaters to shut off resulting in depressurization of the reactor coolant system. It was concluded that the net result of the polishing system malfunction was the potential loss of primary system heat removal capability, i.e., forced convection using RCPs, natural circulation cooling, and feed-and-bleed using HPSI pumps.

(2) During RHR operation at cold shutdown at San Onofre-2, there was a system malfunction or operator error while reprocessing of a demineralizer subsystem.[1172] During this operating mode, the demineralizers of the related CVCS were lined up with the RHR to accomplish RCS cleanup and pressure control. Backflushing of one of the related filters was initiated and, during this process, by either system malfunction or operator error, nitrogen gas used during this procedure passed through the subsystem into the suction lines of all the RHR pumps with resultant loss of operability. The RHR pumps are also the LPSI pumps. In this case, redundant systems important to protection of the facility during an accident, as well as orderly cold shutdown of the plant from 350°F, were rendered inoperable.

(3) At Pilgrim-1, there was a system malfunction which caused an improper recharging of a demineralizer in the RWCS.[1173] This resulted in resin entering the RCS and caused the indicated flow rate input to the APRM flow bias trip settings to read high, thus providing a non-conservative input to two trip functions. In this case, a demineralizer problem affected the ability of a safety system to perform its function.

The loss of the ability to shut down or to maintain a safe shutdown condition for the reactor is considered of highest safety significance and the effect demineralizer failures could have on public risk associated with core-melt were evaluated below. The loss of spent fuel cooling and water cleanup capability was assumed to be of much less safety significance due to the long lead time available to restore cooling. Therefore, it was not considered a large contributor to risk and was not evaluated below.

The second failure mode (loss of water chemistry control) has the potential of changing the corrosion rate for the affected system. However, since a loss of water chemistry and the subsequent change in corrosion rate do not lead to immediate failures, do not affect all parts of the system at the same rate, and can be detected and corrected prior to having any significant impact, this failure mode was not considered a significant contributor to public risk and was not considered further below.

Therefore, based on the above, the rest of this evaluation addressed the failure mode of resin or gas introduction into a system which then leads to immediate failures of other safety systems.

Possible Solutions

Possible solutions included hardware and administrative changes. Specifically, a combination of the following could be done: (1) install filters on the outlet of all demineralizer units which would stop resin from entering the system through the demineralizer outlet nozzle; and (2) evaluate existing procedures, job aids, and training to discern where improvements can be made to enhance operator capability and further reduce the chances for human error which result in resin or gas intrusion into a system during demineralizer recharging.

PRIORITY DETERMINATION

Assumptions

No provision was made in the safety analysis of the operating LWRs to account for the effects or consequences of demineralizer problems or failures. Therefore, by considering the possibility of demineralizer failures, the additional risk these present to the public must be determined. The system failure probabilities used were those summarized in NUREG/CR-2800[64] and were based on the Oconee-3 PRA for PWRs and the Grand Gulf-1 PRA for BWRs. The number of plants affected by this issue was conservatively assumed to be all operating and planned plants (78 PWRs and 39 BWRs) and their average remaining life was assumed to be 30 years.

Frequency Estimate

The frequency of demineralizer failures was estimated using data from an LER search for the period June 1982 through June 1984. LERs prior to 1982 were not searched since old data did not reflect existing operating practice and improvements in procedures, training, etc., subsequent to TMI-2 and, therefore, may not have been an accurate estimate of failure rate.

From the LER search, it was determined that there were 15 events involving abnormalities caused by demineralizer-related problems. Of these 15 events, 2 led to degradation of a safety system. (See References 516, 1172, 1173.) An additional LER search covering the years from 1984 through 1987 was performed to identify LERs that involved demineralizer systems; no additional LERs were identified involving demineralizers that caused a degradation of a safety system. The span from 1982 through 1987 comprised 277 PWR-years of operating experience. Hence, for PWRs, the frequency of safety system failure due to demineralizer problems was 2 failures in 277 PWR-years or 7.2×10^{-3} failure/ RY. For BWRs, there were no recorded LERs involving the loss of safety systems resulting from demineralizer problems. However, the event described[1172] at San Onofre-2 could have occurred in a BWR. Hence, for BWRs, it was assumed that one failure occurred over the span of 166 BWR-years or 6.2×10^{-3} failure/RY.

In the 1984 through 1987 LER assessment, 3 events involving BWRs were found to have occurred which resulted in either an automatic or manual scram. These scrams were the result of high main steam line radiation readings which were believed to be due to either resin or corrosion particles. It was conceivable that all 3 could have resulted from resin particles. Assuming 3 transient events in the 116 BWR-years resulted in 0.026 transient per BWR-year due to demineralizer failures. PWRs were not susceptible to these same occurrences. However, a PWR scram was found which resulted from a demineralizer fault. In the TMI-2 accident,[516] the loss of feedwater resulted in a scram. With one transient trip in 227 PWR-years, a transient frequency of 3.6×10^{-3} event/RY resulted from demineralizer-related events.

Consequence Estimate

Demineralizer system failures and their resulting impact on other plant systems cannot, by themselves, lead to a core-melt or containment failure. They can, however, remove some of the systems which provide lines of defense against such core-melt and containment failure events, or result in transient-induced scrams. In the case of PWRs, the systems which provide a line of defense and which could be rendered inoperable due to a demineralizer failure are: High Head Safety Injection System (for reactor shutdown); Condensate and Feedwater System (for normal decay heat removal); RHR System (for shutdown decay heat removal); and Containment Spray System (for containment pressure and temperature control). For BWRs, the systems are: Reactor Protection System (for reactor shutdown); Condensate and Feedwater System (for normal decay heat removal); and RHR System (for shutdown decay heat removal and containment cooling).

The consequences associated with these events were estimated by considering the following scenario. While at full power, a malfunction in the plant required the plant protection system to automatically shut down the plant. However, a demineralizer problem caused the loss of function of one of the safety systems which could be affected by demineralizers. Other safety systems were assumed to fail with probabilities as defined in the Oconee-3 and Grand Gulf-1 PRAs leading to a core-melt with containment failure. Since this event could result in a loss of core cooling, containment cooling, or containment spray, it was considered to be bounded by the PWR-2 and BWR-2 release categories which have estimated dose consequences of 4.8×10^6 and 7.1×10^6 man-rem/event, respectively. The transient-related accidents T_{23} for BWRs and T_3 for PWRs were expected to result in BWR release categories 1, 2, 3, and 4, and in PWR release categories 3, 5 and 7, respectively.[64]

To estimate the reduction in risk associated with the elimination of demineralizer failures, two calculations were involved: (1) the additional probability of reaching a core-melt due to demineralizer failure which rendered a safety injection system inoperable; and (2) the reduction in core-melt frequency resulting from a reduction in transient-induced scrams. The first was done by assuming that the effect of demineralizer failure contributed directly to the probability of core-melt by adding directly to the failure probability of those systems that can be affected by demineralizer failures. This contribution was calculated by examining the dominant accident sequences for PWRs and BWRs (using the Oconee-3 and Grand Gulf-1 PRAs as representative of these plants) and, for those sequences that involve systems whose performance could by affected by demineralizer problems, adding to that system an annual unavailability of (2×10^{-5}) for PWRs and (1.4×10^{-6}) for BWRs. This would then represent the incremental increase in the frequency of a core-melt accident for a plant. The values calculated for these increases in frequency were 6.4×10^{-8}/RY and 8.8×10^{-8}/RY for PWRs and BWRs, respectively. The transient reductions were based upon the frequency reduction values given previously. The transient reductions resulted in a reduction in core-melt accident frequency of 1×10^{-9}/RY for PWRs and 8×10^{-8}/RY for BWRs. The risk reduction associated with resolution of this issue was calculated below.

<u>PWRs</u>: <u>System Failure</u>

Risk Reduction = (6.4×10^{-8}/RY)(4.8×10^{6} man-rem/event)(30 years)
= 9.2 man-rem/reactor

<u>Transient Risk Reduction</u>

PWR-3 = (0.5)(9.9×10^{-10}/RY)(5.4×10^{6} man-rem/event)(30 years)
= 8×10^{-3} man-rem/reactor

PWR-5 = (0.0073)(9.9×10^{-10}/RY)(1×10^{6} man-rem/event)(30 years)
= 2.2×10^{-4} man-rem/reactor

PWR-7 = (0.5)(9.9×10^{-10}/RY)(2.3×10^{3} man-rem/event)(30 years)
= 3.4×10^{-5} man-rem/reactor

Total PWR dose reduction = 9.3 man-rem/reactor

<u>BWRs</u>: <u>System Failure</u>

Risk Reduction = (8.8×10^{-8}/RY)(7.1×10^{6} man-rem/event)(30 years)
= 15.1 man-rem/reactor

<u>Transient Risk Reduction</u>

BWR-1 = (0.01)(1.4×10^{-8}/RY)(5.4×10^{6} man-rem/event)(30 years)
= 0.022 man-rem/reactor

BWR-2 = (1.0)(7.8×10^{-8}/RY)(7.1×10^{6} man-rem/event)(30 years)
= 16.6 man-rem/reactor

BWR-3 = (0.5)(2×10^{-9}/RY)(5.1×10^{6} man-rem/event)(30 years)
= 0.15 man-rem/reactor

BWR-4 = (0.5)(2×10^{-9}/RY)(6.1×10^{5} man-rem/event)(30 years)
= 0.018 man-rem/reactor

Total BWR dose reduction = 32 man-rem/reactor

In addition, since hardware fixes were assumed to be part of the solution of this issue, the occupational dose associated with the installation of these fixes were considered. The addition of 6 strainers per plant on the outlet of demineralizers was assumed as the hardware fix.

The occupational dose received from the installation of demineralizer strainers was estimated as follows: (1) it was assumed that the installation of each strainer involved 40 man-hours of labor in a radiation zone; and (2) from Chapter 12 of the Oconee-3 and Grand Gulf-1 FSARs, the dose rate in the areas where demineralizers are present was approximately 100 millirem/hr when the plant is shutdown. Therefore, the occupational dose received from the installation of 6 outlet strainers was (40 man-hrs)(6)(0.1 rem/hr)=24 man-rem/reactor.

Since this occupational dose was less than the risk reduction dose consequences, it appeared that there was some benefit to implementing such fixes. The impact of additional strainers on increased occupational dose due to maintenance was assumed to be negligible.

Cost Estimate

Industry Cost: The cost associated with resolution of this issue involve hardware additions (demineralizer outlet strainers) to mitigate the consequences of demineralizer failures, procedure changes, and additional operator training. Hardware fixes were estimated to cost $600,000 based on the addition of 6 outlet strainers/plant. Procedural changes were estimated to cost $12,000 assuming 1 man-month/plant. Based on 1 man-week/RY, additional operator training was estimated to cost ($3,000/RY)(30 years)or $90,000. Thus, the total industry cost was estimated to be $700,000. It was assumed that all of the fixes could be done during normally scheduled downtime; therefore, the cost of replacement power was not a factor.

Additional maintenance costs to monitor implementation were assumed to be negligible. However, it was also possible that a reduction in demineralizer problems would also reduce undesired plant shutdowns and thus save licensees the cost of replacement power. From the LER search, it was determined that, of the 15 events reported involving demineralizers, 2 caused plant shutdowns to correct the problem. It was assumed that half of these could be avoided by the better training procedures and mitigation effects of demineralizer outlet filters. Therefore, based on the LER data, a plant will avoid [(1)(30 years)/(75)(2.5 years)]=0.16 shutdown/plant due to demineralizer problems over its life. This resulted in a cost savings to each plant of (0.16 shutdown)($500,000/ shutdown) =$80,000/plant over its life. (Each shutdown was assumed to last 1 day at a cost of $500,000/day.) Therefore, the total cost/plant to resolve this issue was estimated to be $(700,000 - 80,000) or $620,000.

NRC Cost: NRC costs were negligible.

Total Cost: The total industry and NRC cost associated with a possible solution was estimated to be $0.62/reactor.

Value/Impact Assessment

Based on estimated public risk reductions of 9.3 man-rem/reactor and 32 man-rem/reactor for PWR and BWRs, respectively, and a cost of $0.62M/reactor for a possible solution, the value/impact scores were given by:

(1) PWRs:

$$S = \frac{9.3\ \text{man-rem/reactor}}{\$0.62\text{M/reactor}}$$

$$< 15\ \text{man-rem/\$M}$$

(2) BWRs:

$$S = \frac{32\ \text{man-rem/reactor}}{\$0.62\text{M/reactor}}$$

$$< 52\ \text{man-rem/\$M}$$

Other Considerations

(1) The assumptions used in this evaluation regarding frequency and consequence estimates were conservative because the estimates of frequencies of transients and failures in BWRs were high and the bounding of non-transient accidents by BWR-2 and PWR-2 releases resulted in high public dose estimates. Therefore, the value/impact scores were considered to be high estimates.

(2) Many demineralizer failures can and are detected (via water chemistry, etc.) prior to their affecting other equipment.

(3) Generally, a demineralizer failure affects only one system and this is not enough to prevent a plant from performing its safety functions. In the one case at TMI-2 where more than one system was affected,[516] the plant was in the pre-operational testing phase, prior to certification that the plant condition (equipment and procedures) was suitable for power operation.

(4) At the time of this evaluation, fixes following the TMI-2 failure appeared to have reduced the frequency of occurrences.

CONCLUSION

Based on the above value/impact scores, the issue was given a low priority ranking (see Appendix C) in February 1990. Further prioritization, using the conversion factor of $2,000/man-rem approved[1689] by the Commission in September 1995, resulted in impact/value ratios (R) of $66,666/man-rem and $19,375/man-rem for PWRs and BWRs, respectively, which placed the issue in the DROP category.

Following a periodic review of LOW-priority issues, new information was provided[1773] by Region IV that required a reevaluation of the issue. However, consideration of this new information did not result in any change in the priority of the issue.[1774]

REFERENCES

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516. Memorandum for W. Johnston and L. Rubenstein from T. Speis, "Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety," August 6, 1982.

1172. Letter to R. Engelken (NRC) from H. Ray (Southern California Edison Company), "Docket No. 50-361, Licensee Event Report, Numbers 82-002 and 82-003, San Onofre Nuclear Generating Station, Unit 2," March 30, 1982.

1173. Letter to R. Haynes (NRC) from C. Mathis (Boston Edison Company), "Docket No. 50-293, License DPR-35," September 15, 1982.

1689. Memorandum to J. Taylor from J. Hoyle, "COMSECY-95-033 - Proposed Dollar per Person-Rem Conversion Factor; Response to SRM Concerning Issuance of Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission and SRM Concerning the Need for a Backfit Rule for Materials Licensees (RES-950225) (WITS-9100294)," September 18, 1995.

1773. Memorandum for E. Beckjord from J. Milhoan, "Periodic Review of Low-Priority Generic Safety Issues," June 4, 1993.

1774. Memorandum to A. Thadani from S. Bahadur, "Reprioritization of GSI-71, 'Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety,'" December 20, 2000.

ISSUE 152: DESIGN BASIS FOR VALVES THAT MIGHT BE SUBJECTED TO SIGNIFICANT BLOWDOWN LOADS

DESCRIPTION

Historical Background

This issue was identified[1416] by DSIR/RES following ACRS concerns raised during the review of the resolution of Issue 87, "Failure of HPCI Steam Line Without Isolation," which addressed the design bases for those MOVs that isolate the HPCI, RCIC, and RWCU systems in BWRs. These design bases required that the MOVs close against loads imposed by a double-ended pipe break at design basis flow conditions.

In resolving Issue 87, the staff issued Generic Letter No. 89-10[1217] which required licensees to identify safety-related valves that might not perform adequately under design basis conditions. However, the ACRS believed that the design basis for the HPCI steam line valves and other valves in some plants might not specify the type of heavy duty. Thus, it was possible that heavy duty loads might not be considered for these valves by licensees in response to Generic Letter No. 89-10.[1217] The ACRS recommended that the staff amend the generic letter to require licensees to examine their design bases to determine if safety-related valves, including but not limited to MOVs, were capable of operating against blowdown loads that might not have been considered (by licensees) in their original designs.

Safety Significance

The inability of valves that might be subjected to significant blowdown loads to meet their design bases is a compliance concern. Therefore, the safety significance of this issue lies in the environmental conditions that could result from the inability of containment isolation valves to close under accident conditions. The resulting environmental conditions could cause the malfunction of equipment required to cool the reactor. This issue affects all operating and future plants.

Possible Solution

A possible solution to this issue would include the following: (1) amendment of Generic Letter No. 89-10[1217] to ensure complete compliance with the original design bases; (2) licensee review of design bases for compliance; (3) licensee analyses to assess operability of valves; and (4) hardware modification of isolation valves and additional licensee analyses to bring the valves into compliance with the original design bases.

PRIORITY DETERMINATION

Assumptions

It was assumed that 50% of all 112 operating plants will find that they are in compliance with the amended generic letter. Of the remaining 50% that will have to perform analyses, 80% will demonstrate compliance. Thus, only 10% of all operating plants will make hardware modifications and perform additional analyses to comply with the amended generic letter. Therefore, the potential exists for a reduction in public risk and occupational dose at approximately 11 plants: 7 PWRs and 4 BWRs. Future plants would not require any modifications since their design would be based on the requirements of the amended generic letter. Oconee-3 and Grand Gulf-1 were selected as the representative PWR and BWR, respectively.

Frequency Estimate

For PWRs, a steam line break was assumed to correspond to an S_3 LOCA. If this LOCA is not isolated, the potential exists for introducing a harsh environment into the containment which may affect the operation of certain components needed to mitigate the LOCA. These components were assumed to be MOVs and pumps, specifically for failure modes designated as hardware or control circuitry, found in accident sequences initiated by an S_3 LOCA.

It was assumed[64] that the potential for increased failure under harsh environmental conditions was not factored into the failure probabilities of the affected parameters in the original plant evaluations. Therefore, the base case failure probabilities were assumed to be 10% higher than their original values. For Oconee 3, this resulted in a base case core-melt frequency of 1.18×10^{-6} /RY.[64]

For BWRs, a steam line break was assumed to correspond to an S LOCA. Assuming the same accident scenario and resultant effects described above for PWRs, the base case core-melt frequency for Grand Gulf 1 was estimated[64] to be 2.48×10^{-7}/RY.

It was assumed that resolution of the issue would return the failure probabilities to their original values in both PWRs and BWRs; this represented a 10% reduction in the base case values. Thus, the adjusted case core-melt frequencies were estimated to be 1.01×10^{-6}/RY and 2.09×10^{-7} for Oconee-3 and Grand Gulf-1, respectively. The potential core-melt frequency reduction associated with the possible solution was calculated to be 1.7×10^{-7}/RY and 3.9×10^{-8}/RY for the affected PWRs and BWRs, respectively.

Consequence Estimate

The affected release categories for Oconee-3 were PWR-2,-3,-4,-5,-6, and -7, and the base case and adjusted case public risk were estimated to be 3.14 man-rem/RY and 2.68 man-rem/RY, respectively, with a potential reduction of 0.46 man-rem/RY. For the 7 affected PWRS with an average remaining life of 25.8 years, the public risk reduction was estimated to be (0.46)(7)(25.8) man-rem or 83 man-rem.

Affected release categories for Grand Gulf 1 were BWR-1 and -2 and the base case and adjusted case public risk were estimated to be 1.76 man-rem/RY and 1.48 man-rem/RY, respectively, with a potential reduction of 0.28 man-rem/RY. For the 4 affected BWRs with an average remaining life of 24.1 years, the estimated public risk reduction was (0.28)(4)(24.1) man-rem or 27 man-rem. Therefore, the total public risk reduction associated with the possible solution was estimated to be 110 man-rem.[64]

Cost Estimate

Industry Cost: The review of design bases was estimated to require 6 man-weeks/plant at all 112 operating plants affected by the amended generic letter. At $2,270/man-week, this cost was estimated to be $1.525M.

Additional analyses at 56 plants (50% of all affected plants) were estimated to require 12 man-weeks/plant for a total cost of $1.525M. Equipment costs were estimated to be $20,000/plant (10% of all affected plants) that will have to make valve modifications. These modifications were estimated to require 8 man-weeks of skilled labor and 16 man-weeks for additional engineering analyses. Thus, the total estimated cost for 11 plants that require modifications was $0.82M and the total industry cost associated with the possible solution was $3.87M.

NRC Cost: It was estimated that 8 man-weeks would be required to amend Generic Letter No. 89-10[1217] at a cost of $18,000. Review of licensee responses from all 112 plants was estimated to require 2 man-weeks/plant. Responding to the half of these plants that would have to submit analyses was estimated to require 6 man-weeks/plant. For the 11 plants that would have to be modified, NRC review of the additional analyses was estimated to require 12 man-weeks/plant. Thus, the total NRC review time was estimated to be 692 man-weeks. At $2,270 man-week, this translated to a cost of $1.57M.

Total Cost: The total industry and NRC cost associated with the possible solution was $(3.87 + 1.57)M or $5.44M.

Value/Impact Assessment

Based on a potential public risk reduction of 110 man-rem and an estimated cost of $5.44M for a possible solution, the value/impact score was given by:

$$S = \frac{110 \text{ man-rem}}{\$5.44\text{M}}$$

$$\sim 20 \text{ man-rem/\$M}$$

CONCLUSION

Based on the potential public risk reduction, the issue was given a low priority ranking (see Appendix C) in January 1993. Additional concerns raised[1509] by the ACRS on the ability of safety-related MOVs to close under pipe break conditions were addressed[1510] by the staff but did not affect the priority ranking of the issue. Consideration of a 20-year license renewal period did not change the priority of the issue.[1564] Further prioritization, using the conversion factor of $2,000/man-rem approved[1689] by the Commission in September 1995, resulted in an impact/value ratio (R) of $50,000/man-rem which placed the issue in the DROP category.

Following a periodic review of LOW-priority issues, new information was provided[1699] by NRR that required a reevaluation of the issue. However, consideration of this new information did not result in any change in the priority of the issue.[1775]

REFERENCES

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1217. NRC Letter to All Licensees of Operating Power Plants and Holders of Construction Permits for Nuclear Power Plants, "Safety-Related Motor-Operated Valve Testing and Surveillance (Generic Letter No. 89-10) - 10 CFR 50.54(f)," June 28, 1989, (Supplement 1) June 13, 1990, (Supplement 2) August 3, 1990, (Supplement 3) October 25, 1990, (Supplement 4) February 12, 1992.

1416. Memorandum for B. Morris from W. Minners, "Prioritization of Proposed New Generic Issue," December 4, 1989.

1509. Letter to J. Taylor from P. Shewmon, "Prioritization of Generic Issue 152, 'Design Basis for Valves that Might Be Subjected to Significant Blowdown Loads,'" April 23, 1993.

1510. Letter to J. Wilkins (ACRS) from J. Taylor(EDO), June 8, 1993.

1564. Memorandum for W. Russell from E. Beckjord, "License Renewal Implications of Generic Safety Issues (GSIs) Prioritized and/or Resolved Between October 1990 and March 1994," May 5, 1994.

1689. Memorandum to J. Taylor from J. Hoyle, "COMSECY-95-033 - Proposed Dollar per Person-Rem Conversion Factor; Response to SRM Concerning Issuance of Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission and SRM Concerning the Need for a Backfit Rule for Materials Licensees (RES-950225) (WITS-9100294)," September 18, 1995.

1699. Memorandum to D. Morrison from W. Russell, "Periodic Review of Low-Priority Generic Issues," April 11, 1996.

1775. Memorandum to A. Thadani from M. Mayfield, "Closeout of GSI-152, 'Design Basis for Valves That Might be Subjected to Significant Blowdown Loads,'" April 4, 2001.

ISSUE 156: SYSTEMATIC EVALUATION PROGRAM

In 1977, the NRC initiated the Systematic Evaluation Program (SEP) to review the designs of 51 older, operating nuclear power plants. The SEP was divided into 2 phases. In Phase I, the staff defined 137 issues for which regulatory requirements had changed enough over time to warrant an evaluation of those plants licensed before the issuance of the SRP.[11] In Phase II, the staff compared the design of 10 of the 51 older plants to the SRP[11] issued in 1975. Based on these reviews, the staff identified 27 of the original 137 issues that required some corrective action at one or more of the 10 plants that were reviewed. The staff referred to the issues on this smaller list as the SEP "lessons learned" issues and concluded that they would generally apply to operating plants that received operating licenses before the SRP[11] was issued in 1975.

In SECY-84-133,[814] the staff presented the 27 SEP issues to the Commission as part of a proposal for an ISAP, the intent of which was to review safety issues for a specific plant in an integrated manner. Two SEP plants participated in the ISAP pilot efforts. Following the review of these two pilot plants, ISAP was discontinued.

In SECY-90-160,[1443] the staff forwarded for Commission approval a proposed license renewal rule and supporting regulatory documents. In this paper, the staff stated that certain unresolved safety issues could weaken the generic justification of the adequacy of the current licensing bases argument. These issues included SEP topics for 41 older plants that had not been explicitly reviewed under Phase II of the SEP. The Commission requested that the staff keep it informed of the status of the program to determine how the SEP "lessons learned" issues had been factored into the licensing bases of operating plants.

Resolution of the 27 SEP issues was deemed by the staff to be important to the development of the license renewal rulemaking. The key regulatory principle underlying the license renewal rule is that the current licensing bases (CLBs) at all operating nuclear power plants, with the exception of age-related degradation, provide adequate protection to the public health and safety. This principle is reflected in the provisions of the license renewal rule which limit the renewal decision to whether age-related degradation has been adequately addressed to assure continued compliance with a plant's CLB. In order to adopt this approach, the NRC must be able to provide a technical basis for the key principle of license renewal. Accordingly, the rulemaking included a technical discussion documenting the adequacy of the CLB for all nuclear power plants, in both the statement of considerations and in NUREG-1412.[1444] However, as discussed in SECY-90-160,[1443] the staff identified a potential weakness in the discussion of the adequacy of the CLB with regard to the 41 older, non-SEP plants. To address this potential weakness, the staff undertook an effort to determine whether or not each SEP issue either had been or was being addressed by other regulatory programs and activities.

The staff completed this effort and placed each SEP issue into one of the following categories: (1) issues that had been completely resolved (i.e., necessary corrective actions had been identified by the staff, transmitted to licensees, and implemented by licensees); (2) issues that were of such low safety significance so as to require no further regulatory action; (3) issues that were unresolved, but for which the staff had identified existing regulatory programs that cover the scope of the technical concerns and whose implementation would resolve the specific SEP issue, such as the Individual Plant Examination (IPE) and the Individual Plant Examination of External Events

(IPEEE); and (4) issues that were unresolved and regulatory actions to resolve the issues had not been identified. The 27 SEP issues and applicable regulatory programs were summarized and presented in SECY-90-343.[1351] The staff concluded that the 22 SEP issues in Categories 3 and 4 remained unresolved for purposes of justifying the adequacy of the CLB for some portion of the 41 older, non-SEP plants. The following is an evaluation of these 22 issues: nineteen from Category 3 and three from Category 4.

ISSUE 156.1.1: SETTLEMENT OF FOUNDATIONS AND BURIED EQUIPMENT

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The objective of this issue was to ensure that safety-related structures, systems, and components were adequately protected against excessive settlement. The scope included the review of subsurface materials (soils or geologic) and foundations to assess the potential static and seismically-induced settlement of all safety-related structures and buried equipment.

Excessive settlement or collapse of foundations and buried equipment for structures, systems, and components under either static or seismic loading could result in failure of structures, interconnecting piping, control systems or cables, or other equipment (tanks, etc.) such that the capability to safely shut down a plant, or mitigate the consequences of an accident, could be compromised.

There were two specific concerns in this issue: (1) the potential impact of static soil settlements on foundations and buried equipment where the soil may not have been properly prepared; and (2) seismically-induced differential settlement and potential soil liquefaction following a postulated seismic event. These two concerns were limited only to plants that have soil-supported, safety-related structures (including vertical, field-erected tanks) and soil-buried piping and components (including tanks) that have the potential for excessive settlement but were not reviewed to the pertinent SRP[11] Sections 2.5.4 and 2.5.5.

For the 41 older, non-SEP plants with OLs issued before 1975, any impact of static settlement on structural foundations (including the foundations of buried components) should become noticeable in the first 5 to 10 years. Thus, any significant settlement would have been revealed already and warranted corrective action. In addition, the ongoing IPEEE program[1354] has elements in its seismic task which requires that, for plants on soil sites, potential seismically-induced settlement and soil liquefaction should be assessed during its implementation.

CONCLUSION

This issue is being addressed by the SRP[11] for future plants as well as for operating plants with OLs issued after 1975. For the 51 older, operating plants, this issue was considered resolved for the 10 SEP plants. For the remaining 41 non-SEP, operating plants, any significant static settlement would have been revealed already and warranted corrective action. The concern on the seismically-induced settlement and soil liquefaction for these 41 older, non-SEP operating plants will be addressed during the implementation of the IPEEE Program. Therefore, Issue 156.1.1 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.1.2: DAM INTEGRITY AND SITE FLOODING

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90 -343.[1351] The safety concern was the ability of a dam to prevent site flooding and ensure a cooling water supply. The safety features of a dam would normally include remaining stable under all conditions of reservoir operation, controlling seepage to prevent excessive uplifting water pressure or erosion of soil materials, and providing sufficient freeboard and outlet capacity to prevent overtopping. The objective of this issue was to ensure that adequate margins of safety are available under all loading conditions and uncontrolled releases of retained water are prevented. Plants must provide the basis for ensuring that all safety-related structures, systems, and components are adequately protected against flooding that might result from dam failures. Further, review of licensee procedures would determine whether an adequate supply of cooling water exists in the ultimate heat sink during normal and emergency operations. The 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

If a dam exists in the vicinity of a nuclear power plant, it will have to meet one of the following criteria:

(1) If the dam provides impoundment for an UHS at a plant or provides flood protection, the dam is an essential part of the plant and the safety of the dam needs to be ensured throughout the life of the plant. The dam has to be designed and remain stable under both static and seismic conditions.[688,916]

(2) If the dam provides impoundment only for plant operation, but not as a part of the UHS, there are no regulatory requirements for dam design. However, the flood conditions that could be caused by dam failures should be considered in establishing the design basis flood.[687] When upstream dams or other features that provide flood protection are present, in addition to the analyses of the most severe floods that may be induced by either hydrometeorological or seismic mechanisms, reasonable combinations of less severe flood conditions and seismic events should be considered in establishing the design basis flood.

The IPEEE Program will address the safety and the flooding effects of dams. Under this program, the safety of dams will be assessed by all licensees in the process of searching for severe accident vulnerabilities due to external events.[1222,1354] If the failure of these dams would have significant consequences, i.e., a breach of an UHS which might lead to a severe accident, they would have to be evaluated and inspected to assess their existing condition and vulnerability to earthquakes. If the failure of an upstream dam could lead to significant flooding at a site, i.e., the postulated flood exceeded the design basis flood and might lead to a severe accident, the effect of flooding will have to be addressed in the IPEEE.

CONCLUSION

The safety concerns of dam integrity and site flooding will be addressed in the implementation of the IPEEE Program at the 41 plants affected by this issue.[1575] Therefore, Issue 156.1.2 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.1.3: SITE HYDROLOGY AND ABILITY TO WITHSTAND FLOODS

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The concerns of this issue included identifying the site hydrologic characteristics, the capability of structures important to safety to withstand flooding, the determination of the adequacy of the cooling water supply, and the ISI of water control structures. Hydrologic considerations are the interface of the plant with the hydrosphere, the identification of hydrologic causal mechanisms that may require special plant design, or operating limitations with regard to floods, and water supply requirements. The specific items to be reviewed in this issue were:

(1) Hydrologic Description - To ensure that plant design reflects appropriate hydrologic conditions.

(2) Flooding Potential and Protection - To ensure that the plant is adequately protected against floods.

(3) Ultimate Heat Sink - To ensure an appropriate supply of cooling water is available during normal and emergency shutdowns.

(4) ISI of Water Control Structures - To ensure an adequate inspection program is in place to prevent water control structure deterioration or failure which could result in flooding or loss of the UHS.

The 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

At a nuclear plant, the safety-related structures, systems, and components, identified in accordance with Regulatory Guide 1.29,[916] must be designed to withstand the conditions resulting from the worst probable site-related flood and retain the capability for shutdown and maintenance.[687] Alternatively, NRC permits licensees not to design against the worst flood conditions for safety-related structures, systems, and components if sufficient warning time is shown to be available to shut down the plant and implement adequate emergency procedures. However, the safety-related structures, systems, and components must be designed to withstand the conditions resulting from a Standard Project Flood (with a flow-rate about 40% to 60% of the PMF).[687]

On June 28, 1991, the NRC requested all licensees to conduct an IPEEE to search for severe accident vulnerabilities due to external events[1222]; external flooding is one of the events that will be addressed in the IPEEE.[1354] All licensees will have to examine the flood designs and associated flood protection measures at their sites to determine if severe accident vulnerabilities due to external floods exist. Therefore, the above Items 1 and 2 have been addressed in the external flood portion of the IPEEE program.

Item 3 is related to maintaining the functioning of the SWS and the DHR system of a plant. The severe accident vulnerability resulting either from failure or unavailability of the UHS is one of the important items to be examined in the IPE and IPEEE programs.

The NRC will require the affected licensees to upgrade their ISI programs for water control structures where inspection findings and any subsequent analyses reveal inadequacies in meeting the intent of Item 4.

CONCLUSION

The safety concerns of site hydrologic characteristics and the capability of plants to withstand flooding will be addressed in the implementation of the IPE and IPEEE Programs at the 41 plants affected by this issue.[1575] Therefore, Issue 156.1.3 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.1.4: INDUSTRIAL HAZARDS

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The objective of this issue was to ensure that the integrity of safety-related structures, components, and systems will not be damaged by potential hazards from nearby transportation, storage, or industrial facilities. Such hazards include: (1) shock waves and thermal flux from nearby explosions of munitions or explosive gases or chemicals; (2) drifting toxic/explosive vapor clouds; (3) aircraft; and (4) missiles that can result from nearby explosions, such as a rocketing chemical tank car. In a few past licensing cases, reactor containment and intake structure hardening and pipeline relocation have been required to ensure safety of the plants. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

Regulatory Guide 4.7[1372] and SRP[11] Sections 2.2.1, 2.2.2, and 2.2.3 have been used since 1975 in the design of nuclear power plants for protection against industrial hazards. In addition, Regulatory Guides 1.78,[1373] 1.91,[1374] and 1.95[1375] were issued to provide further regulatory guidance in this area. Prior to the issuance of these criteria, offsite hazards had been an area of long-standing concern and were reviewed on a case-by-case basis.

Supplement 4 to Generic Letter No. 88-20[1222] required all licensees to conduct an IPEEE to search for severe accident vulnerabilities due to external events. Industrial hazards comprise one of the external events that will be addressed in the IPEEE.[1354]

CONCLUSION

Based on past staff reviews, existing review criteria and guidance, and the implementation of the IPEEE program for all plants, the concern for industrial hazards was adequately addressed. Therefore, Issue 156.1.4 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.1.5: TORNADO MISSILES

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] All plants licensed after 1972 were designed for protection against tornadoes. The concern existed, however, that plants constructed prior to 1972 may not be adequately protected, in particular, those reviewed before 1968 when criteria on tornado protection were first developed. The objective of this issue was to ensure that safety structures, systems, and components can withstand the impact of an appropriate postulated spectrum of tornado-generated missiles. The failure of safety-related structures, systems, or components due to a tornado-induced missile could compromise the ability of a plant to safely shut down. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

A plant must be designed to remain in a safe condition in the event that the most severe tornado that can be reasonably predicted occurs at the plant site as a result of severe meteorological conditions. All safety-related structures, systems, and components must be designed to withstand the effects of the design basis tornado, tornado-generated missiles, and other tornado-induced effects.[42,916]

Under the IPEEE program, all licensees are required to examine their plants to determine if severe accident vulnerabilities due to high winds/tornadoes exist.[1222,1354] The criteria used for plant design (such as the design basis wind speed, parameters of the design basis tornado along with missile spectrum, and the allowable stresses and load combinations) will be examined. The reporting criterion, 10^{-6}/year CDF, specified for the IPEEE, however, is considered to be less stringent compared to the CDF associated with tornado missiles design criteria (a product of combining the probability of exceedance associated with the design basis tornado and the conditional failure probability associated with engineering design and construction against tornado missiles). Therefore, meeting the objectives of the IPEEE does not mean, in this situation, that current NRC guidelines for tornado design have been met. Thus, the staff believes that any vulnerability associated with tornado missiles will be evaluated and reported in the IPEEE submittals.

CONCLUSION

The safety concern for tornado missiles will be addressed in the implementation of the IPEEE Program at the 41 plants affected by this issue. Therefore, Issue 156.1.5 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.1.6: TURBINE MISSILES

DESCRIPTION

This issue is one of the three Category 4 issues identified by NRR in SECY-90-343.[1351] The safety concern was the potential damage from turbine missiles in nuclear plants licensed before 1973.

As a result of turbine disc failures at two nuclear plants and a number of non-nuclear plants prior to 1973, the staff believed that high energy missiles could be generated from steam turbines with the potential for causing failures in safety-related systems. The two areas of concern were: (1)

failures at design overspeed because of degraded disc material, poor ISI of flaws, or chemistry conditions leading to SCC; and (2) destructive overspeed failures that would bring into question the reliability of electrical overspeed protection systems, the reliability and testing programs for stop and control valves, and the ISI of valves. For plants licensed after 1973, the safety concerns of this issue were reviewed by the staff as part of its OL activities; turbine overspeed protection designs were found acceptable and the magnitude of the potential damage from turbine missiles was determined to be plant-specific.

CONCLUSION

The safety concerns of this issue were addressed in the evaluation of Issue A-37, which focused primarily on plants licensed prior to November 1976; SRP[11] requirements for turbine design were issued for use by CP applicants after this date. Based on the historical failure rate of turbines used in the evaluation, Issue A-37 was determined to have little safety significance. No new data were provided in SECY-90-343[1351] that changed this conclusion. Therefore, this issue was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.2.1: SEVERE WEATHER EFFECTS ON STRUCTURES

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] Safety-related structures, systems, and components should be designed to function under all severe weather conditions to which they may be exposed. Meteorological phenomena to be considered include straight winds, tornadoes, snow and ice loads, and other phenomena judged to be significant for a particular site. The objective of this issue was to identify those meteorological conditions which should be considered in the structural reviews to determine the ability of structures to withstand conditions such as flooding, wind, tornadoes, hurricanes, tsunamis, and seiches. The dynamic effects of waves, tornado pressure drop loading, and possible in-leakage due to floods were to be considered. The 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

A nuclear power plant must be designed to remain in a safe condition in the event that the most severe weather conditions that can reasonably be predicted at the site occurs. All the safety-related structures must be designed to withstand the effects of the design basis flood, wind, hurricane, tornado, wind/tornado-generated missiles, and other wind/tornado-induced effects.[916]

Under the IPEEE Program, all licensees were requested to examine their plants to determine if severe accident vulnerabilities due to floods or high winds/tornadoes exist.[1222,1354] Licensees were expected to examine their design criteria (such as the design flood level, the hydrostatic pressures against the structures, the design basis wind speed, parameters of the design basis tornado along with missile spectrum, and the allowable stresses and load combinations) used for plant structures to determine if the 1975 SRP[11] criteria are satisfied. If a plant conforms to these criteria, it will be judged that the contribution to CDF from the effects of severe weather is less than 10^{-6}/year and the IPEEE screening criterion would be met. Otherwise, additional evaluation will have to be made to establish severe accident vulnerabilities due to the effects of severe weather. The reporting criterion of 10^{-6}/year CDF specified for the IPEEE will provide a means by which the ability of a nuclear power plant to withstand severe weather conditions can be reviewed and examined for severe weather-induced vulnerabilities.

Snow and ice loads, when accompanied by strong winds, have caused several complete and partial losses of offsite power and the potential of causing severe accidents at a particular site will be evaluated in the IPE program. Snow and ice loads alone, are judged, based on limited PRA experience, to be unlikely to cause significant structural failure that might lead to severe accidents at nuclear power plants.

CONCLUSION

The safety concern of severe weather effects on structures will be addressed in the implementation of the IPEEE program. Therefore, Issue 155.2.1 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.2.2: DESIGN CODES, CRITERIA, AND LOAD COMBINATIONS

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] With the development of nuclear power, provisions addressing nuclear power plants were progressively introduced into codes and standards to which plant buildings and structures are constructed. Because of this evolutionary development, older nuclear power plants conform to a number of different versions of codes and standards, some of which have since undergone considerable revision. There has likewise been a corresponding development of other licensing criteria, resulting in similar non-uniformity in many of the requirements to which plants have been licensed.

Individual SEP plant reviews identified specific areas of structural design code changes for which the previous codes used in the SEP review required greater safety margins than earlier versions of the codes, or for which no original code provision existed. Most plants demonstrated that safety margins in building structures were not significantly lower than those required by the codes and standards used in the SEP review. A few SEP plants required certain modifications to plant structures.

The concern of this issue was to provide assurance that building structures that house systems and components important to safety are capable of withstanding the effects of natural phenomena such as earthquakes,[916] tornadoes (See Issue 156.1.5), hurricanes, and floods without loss of capability to perform their safety function. These events could cause walls or roofs to collapse damaging equipment that perform a safety function, thereby increasing the likelihood of a transient or LOCA.

CONCLUSION

On June 28, 1991, Supplement 4 to Generic Letter 88-20[1222] was issued requesting all licensees to perform an IPEEE to determine if vulnerabilities to severe accidents initiated by natural phenomena existed.[1354] The as-built structures, systems, and components in conjunction with operating plant conditions will be used to assess the adequacy of plant safety. Although this program does not directly address the effects of specific structural design code changes, it does in part focus on evaluating the capability of building structures to withstand natural phenomena and to search for cost-effective improvements that can be made to either prevent or reduce the impact of severe accidents. Thus, the staff believed that any severe accident vulnerabilities associated with the effects of natura phenomena on building structures will be evaluated and reported in the IPEEE submittals.

The safety concern with respect to the capability of building structures to withstand the effects of natural phenomena will be sufficiently addressed in the implementation of the IPEEE Program at the 53 operating plants (34 PWRs and 19 BWRs) affected by this issue. Therefore, Issue 156.2.2 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.2.3: CONTAINMENT DESIGN AND INSPECTION

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The objective of this issue was to review the inspection program for tendons in prestressed concrete containment structures to determine whether the inspection programs included testing of prestressed tendons, checking for corrosion or relaxation and possible deterioration of prestressed containments, and whether the concrete in the containment dome or walls degraded due to shrinkage or creep. The 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

The concerns about the tendons were addressed in Issue 118 which was identified when a dented and leaking tendon grease cap was found during inspection at Farley Unit 2. The generic implications of tendon anchor head failures were studied under Issue 118 and tendon inspection and surveillance programs were developed that could be followed by licensees to mitigate or reduce such problems. The guidance for inspection and surveillance are contained in Regulatory Guides 1.35[481] and 1.35.1.[1360]

The containment dome or wall degradation due to shrinkage or creep is an age-related factor and is also addressed in Regulatory Guide 1.35.1.[1360] For license renewal applications, this concern was addressed in Draft Regulatory Guide DE-1009, "Standard Format and Content of Technical Information for Applications to Renew Nuclear Power Plant Operating Licenses," which will resolve the concern when issued in final form.

10 CFR 50 Appendix A (GDC 53), as implemented by Regulatory Guide 1.35,[481] requires that measured tendon forces (guidance provided in Regulatory Guide 1.35.1[1360]) be compared with acceptance criteria. This issue was reviewed by the staff for all SEP plants and accepted on a case-by-case basis, as documented in SERs; some of these plants also developed ISI programs.

CONCLUSION

The safety concerns of containment design and inspection at the 41 plants affected by this issue were addressed in the resolution of Issue 118. Beyond the normal life of the plants, the age-related concrete degradation concern will be addressed in the License Renewal Program. Therefore, 156.2.3 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.2.4: SEISMIC DESIGN OF STRUCTURES, SYSTEMS, AND COMPONENTS

DESCRIPTION

This issue is of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The objective of this issue was to review and evaluate the original seismic design (seismic input, analysis methods, design criteria, seismic instrumentation, seismic classification) of safety-related plant structures, systems, and components to ensure the capability of plants to withstand the effects of an earthquake. Further, this issue would verify whether the free field ground motion specified for plant design adequately represents the vibratory ground motion associated with a postulated SSE at each plant. The free field ground motion will be utilized as the input to analyses to verify the design adequacy of structures, piping, and equipment. This review and evaluation will address the SSE only, since it represents the most severe event that must be considered in plant design. The scope of the review includes three major areas: (1) the integrity of the reactor coolant pressure boundary; (2) the integrity of fluid and electrical distribution systems related to safe shutdown; and (3) the integrity of mechanical and electrical equipment and engineered safety features systems (including containment). This issue did not call for a detailed review of all safety-related structures, systems, and components; rather, a sampling approach supported by a set of confirmatory analyses were to be performed. The sample size and confirmatory analyses were to be increased, if necessary. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

GDC 2 of Appendix A to 10 CFR 50 requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. An earthquake is one of the natural phenomena whose effects nuclear power plants must be designed to withstand and remain in a safe condition.

In Supplement 4 to Generic Letter No. 88-20,[1222] licensees were required to conduct an IPEEE to search for severe accident vulnerabilities due to external events. A seismic event is one of the external events that should be addressed in the IPEEE.[1371] All licensees will have to review and evaluate the seismic capabilities of their plants (the as-built, as-operated plants) to withstand the earthquake effects well beyond the design basis and to determine if severe accident vulnerabilities due to seismic events exist at their plants. The seismic input has been evaluated by the staff in the Eastern United States Probabilistic Seismic Hazard Program and the results have been factored into the process of determining the seismic review scope in the IPEEE.

The seismic qualification of mechanical and electrical equipment is being resolved by the implementation of the resolution of Issue A-46. A seismic IPEEE can be accomplished by performing either a seismic PRA with enhancements or a seismic evaluation using a seismic margins method with enhancements. The review scope may vary from plant to plant depending on the selected method and the prescribed seismic hazard condition at the site. Even with the minimum effort under the IPEEE seismic program, at least two success paths (a preferred and an alternative) to shut down and maintain a plant in a safe shutdown condition will be evaluated.[1371] This process, when using the seismic margins approach, might not provide a detailed review of all safety-related structures, systems, and components, but it will represent a sampling approach, thus fulfilling the objective of Issue 156.2.4. Furthermore, if warranted as a result of staff review, additional analyses on selected safety-related structures, systems, and components can be performed.

CONCLUSION

The safety concerns for the seismic design of structures, systems, and components will be addressed in the implementation of the IPEEE. Therefore, Issue 156.2.4 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.3.1.1: SHUTDOWN SYSTEMS

DESCRIPTION

Issues 156.3.1.1 and 156.3.1.2 were combined and evaluated together. These issues are two of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by these issues.

Issue 156.3.1.1 addressed the capability of plants to ensure reliable shutdown using safety-grade equipment. Systems and components important to safety should be designed, fabricated, installed, and tested to quality standards commensurate with the safety function to be performed. Also, systems and components that are required to withstand the effects of an SSE and remain functional should be classified as Seismic Category I. Due to the evolutionary nature of design codes and standards, the staff believed that operating plants may have been designed to requirements that are not as conservative as those currently required. Systems needed to remove decay heat and reach safe shutdown should have sufficient redundancy to ensure that their function can be accomplished with a loss of offsite power and a single failure. Systems needed to shut down must also remain functional following external events. In addition, the plant operating procedures which direct the use of these systems during normal and abnormal events were to be evaluated.

Issue 156.3.1.2 addressed the review of electrical instrumentation and control features of systems required for safe shutdown, including support systems, to determine whether they met existing licensing requirements. This review was to include the capability and methods of bringing the plant from a high pressure to a low pressure cooling condition, assuming the use of only safety equipment.

The intent of these issues have been met by a number of NRC requirements and initiatives that are already in place to secure reliable plant shutdown capability. These are as follows:

(1) The fire protection rule (10 CFR 50, Appendix R) requires that the capability for shutdown be maintained, in the event of a fire in any location;

(2) The station blackout rule (10 CFR 50.63) requires the capability to cope with a complete loss of AC power and maintain safe shutdown at the same time;

(3) A number of initiatives under the TMI Action Plan[48] enhance auxiliary feedwater capability, including emergency power provisions;

(4) Improved capability for natural circulation cooldown was required by Generic Letter No. 81-21[1355] and improved TS that enhance RHR operability in all modes were required by Generic Letter Nos. 80-42 and 80-53[1356];

(5) TMI Action Plan[48] Item I.C.I requires upgraded procedures for emergency conditions, including alternate means of providing a heat sink;

(6) The TMI Action Plan,[48] as clarified by NUREG-0737,[98] resulted in the issuance of requirements to licensees to implement Regulatory Guide 1.97[55] which specifies instrumentation for monitoring important parameters such as pressure, flow, and temperature (Continuing improvements in emergency procedures and training also address these issues);

(7) The resolution of Issue A-46 and the imposition of Generic Letter Nos. 87-02[1069] and 87-03[1387] required licensees to address the seismic adequacy of equipment needed to bring a plant to hot shutdown and maintain that condition for a minimum of 72 hours;

(8) The resolution of Issue 99 addressed corrective actions to reduce risk during shutdown with requirements issued in Generic Letter No. 88-17.[1145] The program described in this letter was included in a broader program described in SECY-91-283[1370] to evaluate the risk associated with shutdown and low power.

The resolution of Issue A-45 spanned the period from March 1981 to September 1988 during which time, extensive, PRA-based determinations of the risk resulting from shutdown cooling system failures at 6 representative operating plants were made. These studies included (but were not limited to) the concerns of Issues 156.3.1.1 and 156.3.1.2. The technical resolution of Issue A-45 was described in SECY-88-260[1143] in which the following conclusions were presented:

(1) The risk due to loss of DHR systems could be unduly high for some plants;

(2) DHR failure vulnerabilities and the optimum corrective actions for those vulnerabilities are strongly plant-specific;

(3) Detailed plant-specific analyses under the IPE program, including extension of the IPE program to require consideration of externally-initiated events (anticipated at the time of the resolution of Issue A-45 but since accomplished), will be needed to impose and implement the resolution of this issue.

The staff concluded from the PRA studies that the risk from DHR-related failures might be too high at some plants, but a generic corrective action or a set of actions could not be identified that would both reduce that risk to an acceptable level and be cost-effective at all plants. It was believed, however, that cost-effective plant-specific actions might be possible that would reduce DHR-failure-related risk and it was concluded that the most efficient method to identify any such actions would be through the IPE program.

Appendix 5 of Generic Letter No. 88-20[1222] provided a specific description of those topics addressed in Issue A-45 and related to internally-initiated events (including those raised in Issues 156.3.1.1 and 156.3.1.2) that are to be considered in the IPE program. The IPE process was extended to include externally-initiated events (IPEEE) upon issuance of Supplement 4 to Generic Letter No. 88-20.[1222] Section 5 of this supplement specifically described how the IPEEE program was to be used to implement the technical resolution of those topics in Issue A-45 that are related to externally-initiated events.

The studies performed in the resolution of Issue A-45 included the analysis of events that initiate at full power conditions. Although the final results (total risk resulting from DHR-related failures) were increased by 20% for PWRs and 30% for BWRs to account for risk from DHR-related failures, during events that initiate when a plant is not at full power (such as hot standby and cold shutdown), such events were not investigated in detail. The IPE process was consistent with the analyses completed for Issue A-45 in that it only required consideration of events that initiate at full power conditions.

However, detailed attention is currently being paid to DHR failure-related events that initiate at conditions other than full power by an extensive NRC program initiated with the issuance of Generic Letter No. 88-17[1145] which resulted from an Augmented Inspection Team (AIT) investigation of a 1987 loss-of-DHR event at Diablo Canyon.[1369] This letter required licensees to investigate and, if necessary, improve procedures involving containment isolation and cooling and DHR-related equipment operation methods and training during non-power operations, when the reactor primary coolant inventory is reduced. This work received additional impetus since the issuance of Generic Letter No. 88-17[1145] by a loss-of-DHR event at the Vogtle nuclear plant. The Vogtle event resulted in the issuance of SECY-91-283[1370] which described all aspects of the extensive program including, but not limited to, the program outlined in Generic Letter No. 88-17.[1145] Some aspects of the program described in SECY-91-283[1370] will contribute to the imposition and implementation of the resolution of Issue A-45. This program now includes the NRC-sponsored Low Power and Shutdown (LP&S) Program which was originally formulated as part of the NRC response to the Chernobyl event.[1195] The LP&S work is being performed by BNL and SNL with additional work regarding seismically-initiated events being performed by Future Resources Associates (FRA), Inc. The objectives of the LP&S program were to: (1) assess the frequency and risk of accidents initiated during LP&S modes of operation for two nuclear power plants; (2) compare the assessed frequency and risk with those of accidents initiated during full power operations; and (3) develop new methods for assessing LP&S accident frequency and risk, as necessary.

CONCLUSION

The safety concerns of Issues 156.3.1.1 and 156.3.1.2 were addressed in the resolution of Issue A-45 and in the IPE and IPEEE programs which were supplemented by the Evaluation of Shutdown and Low Power Risk Issues Program described in SECY-91-283.[1370] Therefore, Issues 156.3.1.1 and 156.3.1.2 were DROPPED from further consideration as new and separate issues. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issues.

ISSUE 156.3.1.2: ELECTRICAL INSTRUMENTATION AND CONTROLS

This issue was evaluated with Issue 156.3.1.1 above and DROPPED from further consideration as a new and separate issue.

ISSUE 156.3.2: SERVICE AND COOLING WATER SYSTEMS

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The safety concern was the capability of service and cooling water systems to meet their design objective with adequate margin. This issue was raised to provide assurance that service and cooling water systems are: (1) capable of transferring heat from structures, systems, and components important to safety to the ultimate heat sink; (2) provided with adequate physical separation such that there are no adverse interactions among the systems under any mode of operation; and (3) provided with sufficient cooling water inventory or that adequate provisions for makeup are available. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

Concerns for the potential unavailability of SWS were addressed in Issues 51, 130, and 153. Issue 51 was resolved and implemented at operating plants in accordance with Generic Letter No. 89-13.[1259] The resolution identified a recommended improvement in the reliability of open cycle SWS that could result from reducing the potential for flow blockage in safety-related components caused by bivalves, sediment, and corrosion products. This improvement was in the form of an integrated, baseline fouling surveillance and control program for all nuclear power plant open cycle SWS.

Issue 130 was resolved and is being implemented at certain specific plants in accordance with Generic Letter 91-13.[1368] This issue addressed the concerns regarding the SWS reliability of 14 PWRs at multi-unit sites with two SWS trains per unit and a crosstie capability. The resolution identified several cost-effective options that were considered for reducing the risk from loss of SWS (due to causes other than fouling), including a backup means of RCP seal cooling plus additional SWS TS and emergency procedures.

Issue 153 affected all LWRs except those that were addressed in Issue 130. All potential causes of SWS unavailability were to be considered, except those that were resolved and implemented in accordance with Generic Letter No. 89-13.[1259] The resolution plan for Issue 153 was divided into two phases: Phase I, a pilot study; and Phase II, a generic evaluation. The results of Phase I were to be used to determine if an interim resolution was viable and how to proceed with Phase II; Issue B-32 was also addressed in the resolution of Issue 153.

Concerns for the availability of cooling water systems were addressed in the resolution of Issue 143. This issue addressed the potential unavailability of chilled water systems which provide room cooling to maintain adequate environmental temperature for non-safety-related and safety-related equipment. The potential loss of room cooling could affect the operability of the safety-related systems including the SWS system.

CONCLUSION

All of the concerns regarding the performance capability and reliability of service and cooling water systems at the 41 affected plants either have been addressed or are being addressed in the issues discussed above. Additionally, a staff action plan was developed that established NRR as the focal point to ensure that all existing and future SWS issues are adequately addressed.[1367] Therefore, Issue 156.3.2 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.3.3: VENTILATION SYSTEMS

DESCRIPTION

This issue is one of nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] At issue was the adequacy of ventilation systems to provide a safe environment for plant personnel and ESF systems under normal, anticipated transient, and design basis operational conditions. A safe environment is one that is effectively controlled with respect to radiation, heat, humidity, smoke, and toxic gases. Five ventilation systems were identified in SRP[11] Section 9.4 to effect ESF equipment and plant personnel: the control room area, spent fuel area, auxiliary and radwaste area, turbine area, and ESF area.

With respect to plant personnel, the concerns about ventilation are grouped under radiation exposure as the first, and exposure to excessive levels of environmental pollutants such as smoke, toxic gases, heat, and humidity as the second. These concerns may be considered for both normal operating and abnormal conditions. For normal conditions, the first concern is addressed by existing regulations in 10 CFR 20 which is quite clear and comprehensive concerning monitoring of restricted and unrestricted areas and radiation limits in each. In particular, 10 CFR 20.106 applies to radioactivity in effluent between restricted and unrestricted areas. Coverage includes limits of concentrations of radioactive material in air as well as water. For applications filed after January 2, 1971, 10 CFR 50.34a requires ALARA programs which are elaborated upon in 10 CFR 50, Appendix I. In addition, 10 CFR 50.34a requires design and installation of equipment "to maintain control over radioactive materials in gaseous and liquid effluent" not only during normal operations but also during expected operational occurrences. 10 CFR 50.36a requires TS on effluent from nuclear power reactors.

For normal operating conditions, the second concern is the responsibility of OSHA whenever the safety of licensed radioactive materials is not involved. This responsibility was outlined in an MOU between OSHA and the NRC issued on October 25, 1988. For abnormal conditions, the second concern comprises potentially unpleasant plant nuisance factors with the exception of the control room and turbine area. One potentially serious atmospheric contaminant in the turbine building and the auxiliary building of PWRs is H_2 with its potential for deflagration or detonation. Issue 106 addressed the role of ventilation systems in the prevention of H_2 deflagration from leaks in the H_2 distribution piping.

Issue 136 addressed the issue of vapor clouds from liquified combustible gases drifting into safety-related air intakes.

Abnormal control room environmental conditions could exist that adversely affect operator performance to a degree sufficient to cause operator-initiated transients. These conditions are within the NRC scope as defined in the above MOU. Conditions affecting mitigation of accidents are also clearly NRC responsibility. The resolution of Issue 83 will address the limits of plant personnel functioning from radiation and toxic gas exposure. The scope of Issue 83 includes "provisions for personnel to remain in the control room as needed to manage accidents which have the potential for offsite and onsite radiological consequences, and protection of control room occupants to the degree necessary to prevent an accident occurring as a result of operator incapacitation." SRP[11] Section 6.4, Rev. 2, describes review of the control room ventilation system with the objective of assuring protection for plant operators from the effects of accidental releases of toxic and radioactive gases. A third revision draft is under consideration as part of the resolution of Issue 83. Thus, accident initiation and mitigation capabilities of control room personnel are being addressed with respect to radiation and toxic gas exposure. Control room concerns remaining are high temperature and humidity and smoke.

With respect to high temperature and humidity, the ACRS recommended that "[t]emperature limits should be revised taking into account low air exchange rate, operation of ESF filter system heaters and perspiration." The ACRS considers a temperature limit of 120°F for the control room as unacceptable; this is a TS limit derived for control room equipment.[678] Under accident conditions, no NRC requirement exists for temperature limits for reliable performance of control room personnel. However, documentation exists that supports a maximum effective temperature of 85°F for reliable human performance. (A defined effective temperature includes some combination of dry bulb temperature, relative humidity, and air velocity). Although no accident condition temperature limit has been formalized, SRP[11] Section 9.4.1, "Control Room Area Ventilation System," concerns itself in part with "...the comfort of control room personnel during normal operating, anticipated operational transient, and design basis accident conditions." The control room area ventilation system (CRAVS) is reviewed, among other things, with respect to ability to maintain a suitable ambient temperature for control room personnel. The single failure criterion is applied in the CRAVS review. In addition, the CRAVS must function unaffected by loss of equipment that is not seismic Category 1 and the integrated system design must satisfy GDC 2 with respect to earthquakes. The designs are reviewed for protection from floods, hurricanes, tornadoes, internally- or externally-generated missiles, fires, and loss of offsite power. At some plants, the CRAVS is capable of functioning in an internal-filtered recirculation mode of operation.

A survey of 12 plants reported some problems with adequacy and demonstration of adequacy of control room cooling for a postulated 30-day accident period.[1371] The plants surveyed were a mix of ages, ranging from some of the oldest to some of the newest. While the problems identified produced no added industry requirements, a recommendation was made for more [staff] attention to detail in evaluations of control room cooling systems design and operations that rely on two separate cooling systems, i.e., a non-safety-related system for normal operations and a safety-related system for emergency operations only. In sum, no additional regulatory requirements or guidance are warranted for investigation with respect to high temperature and humidity vis-a-vis control room personnel under accident conditions.

Issue 143 is to be resolved and will address the importance of ventilation systems on cooling for the operation of ESF equipment. Activities in support of the resolution of Issue 143 will identify the vulnerabilities of safety-related systems and their support systems to the effects of HVAC and chilled water system failures and adverse temperature fluctuations. An evaluation will be made of equipment environmental qualification, equipment room heat load and heat-up rate to identify areas in which a reduction in the dependence of equipment operability on HVAC and room cooling may be required. The control of smoke in plants is being addressed in Issue 148.

CONCLUSION

The safety concerns of Issue 156.3.3 were either being addressed in ongoing staff actions on Issues 83, 106, 136, 143, and 148, or were covered by existing regulations. Therefore, Issue 156.3.3 was DROPPED from further pursuit as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.3.4: ISOLATION OF HIGH AND LOW PRESSURE SYSTEMS

DESCRIPTION

This issue is one of nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] At issue were low pressure systems (such as the RHR systems) that interface with the reactor coolant system through isolation valves. The concern was that systems with low design pressure, in comparison with reactor coolant pressure, will incur damage due to valve failure or inadvertent valve opening.

Issue 105 addressed the possible breach of those interfacing boundaries that are created by a series of PIVs and the consequences of failure of a boundary by mechanical failure, human error, or external event. Thus, Issue 105 covered all interfacing systems, including those identified in Issue 156.3.4. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

CONCLUSION

The safety concern of Issue 156.3.4 was addressed in the resolution of Issue 105. Therefore, Issue 156.3.4 was DROPPED from further pursuit as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.3.5: AUTOMATIC ECCS SWITCHOVER

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] Most PWRs require operator action to realign the ECCS for the recirculation mode following a LOCA. Existing guidelines state that automatic transfer to the recirculation mode is preferable to manual transfer. However, a design that provides manual switchover is sufficient provided that adequate instrumentation and information displays are available for the operator to manually transfer from the injection mode to the recirculation mode at the correct time. Automatic in lieu of manual switchover could possibly provide an improvement of ECCS reliability at a cost that could result in a worthwhile safety enhancement. This issue addressed the procedures for manual switchover, the adequacy of available instrumentation, and the possible operator errors associated with the switchover process. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

CONCLUSION

All 41 plants affected by this issue were to be considered in the resolution of Issue 24 which was directed at studying the merits of manual, automatic, and semi-automatic ECCS switchover to recirculation. Thus, Issue 156.3.5 was covered in the resolution of Issue 24. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change this conclusion.

ISSUE 156.3.6.1: EMERGENCY AC POWER

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The electrical independence and redundancy of safety-related onsite power sources must meet the single failure criterion. Diesel generators, which provide emergency standby power for safe reactor shutdown in the event of total loss of offsite power, have experienced a significant number of failures over the years that have been attributed to a variety of causes, including failure of the air startup, fuel oil, and combustion air system. The objective of this issue was to review the reliability of protection interlocks and testing of diesel generators to assure that diesel generator systems meet the availability requirements for providing emergency standby power to the engineered safety features, as well as the independence of onsite power distribution systems and features, such as automatic bus transfers and breaker connections, that could affect the independence of redundant trains. The 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

CONCLUSION

The safety concern of this issue was addressed in the resolution of Issues A-44, 128, and B-56. The requirements that resulted from the resolution of these three issues will affect the 41 non-SEP plants. In addition, MPAs B-23, "Degraded Grid Voltage," and B-48, "Adequacy of Station Electric Distribution Voltage," have been implemented at several of the 41 plants affected by this issue and will not have to be repeated in the implementation of the resolution of Issue A-44.[1108] Based on the above considerations, Issue 156.3.6.1 was DROPPED from further pursuit as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.3.6.2: EMERGENCY DC POWER

DESCRIPTION

Historical Background

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343[1351] following its study of how the lessons learned from the SEP have been factored into the licensing bases of operating plants. The issue addresses the concern that safety-related DC power system bus voltage monitoring and annunciation may not adequately notify operators of DC bus status. Responses to Generic Letter 91-06[1399] indicated that a significant number of licensees could be affected by the concerns of this issue. Based upon a PRA analysis of the DC power system at six plants, it was concluded that additional DC power system bus voltage monitoring and annunciation for licensed facilities would not have a significant impact on safety and would not be a cost-effective means of increasing plant safety.

This issue addressed the criteria in 10 CFR 50.55a(h) and 10 CFR 50 (GDC 2, 4, 5, 17, 18, and 19) which require that the control room operator be given timely indication of the status of the safety-related DC power system batteries and their availability. The current staff position is that the following separate and independent control room indications and alarms for the Class 1E DC power system status are recommended in order to meet these criteria:

(1) battery disconnect or circuit breaker open alarm
(2) battery charger disconnect or circuit breaker open alarm (both input AC and output DC)
(3) DC system ground alarm
(4) DC bus undervoltage alarm
(5) DC bus overvoltage alarm
(6) battery charger failure alarm
(7) battery discharge alarm
(8) battery float charge current ammeter
(9) battery circuit output current ammeter
(10) battery discharge indicator
(11) bus voltage voltmeter

These annunciators and alarms are needed in order to ensure that the control room operators are alerted in the event of DC power system or battery failure. If a less extensive configuration of equipment is used, it is possible that a DC power system or battery failure mode could exist which would not result in the actuation of any alarms or annunciators. In this event, the DC power supply would remain in the degraded condition until a periodic surveillance test or maintenance was performed to identify the condition of the batteries.

Safety Significance

Based upon the SEP reviews, it was apparent that some licensees had received operating licenses without providing the above recommended alarms and annunciators. However, in most cases the licensees in the SEP reviews were able to demonstrate to the staff that modifications were unnecessary. The concern in this issue is that some licensees that were not reviewed in the SEP program might have insufficient annunciators and alarms in the control room to alert the operators to some safety-related DC power supply or battery failure modes, which would increase the likelihood that a DC power supply is unavailable when needed.

PRIORITY DETERMINATION

The issue of control room annunciation and alarms for the safety-related DC power supplies was also addressed in Issue A-30 which was combined with other generic issues involving safety-related power supplies to form Issue 128. Generic Letters 91-06[1399] and 91-11[1400] were issued in the resolution of Issue 128; Generic Letter 91-06 addressed the concerns of Issue A-30. Industry organizations such as NUMARC and INPO asserted that most licensees already had alarm and annunciator configurations that were equivalent to the existing staff recommendations which were based in part on industry standards. Therefore, the questions in Generic Letter 91-06[1399] which addressed available alarms and annunciators did not represent a minimum acceptable configuration, but were formulated to provide sufficient information to the staff to determine if licensees had met or adequately addressed the current recommendations.

An INEL review[1457] of the responses to Generic Letter 91-06[1399] showed that 42 licensees do not have any separate and independent alarms in the control room for their DC power system. However, these licensees typically had local alarms which were separate and independent, and a single battery condition monitor which alarms in the control room in the event that one or more of the local battery alarms actuate. In addition, the INEL review indicated that 15 licensees have not performed a human factors review of their testing and maintenance procedures, and 5 licensees do not have procedures that specifically prevent simultaneous testing or maintenance of redundant safety-related DC power sources. In most cases, the licensees supplied justification for the discrepancies between their licensed configuration and the current staff position. INEL did not evaluate licensee responses to determine what modifications would be required to adequately resolve the concerns of Issue A-30, and recommended that the staff perform a PRA study to determine the impact on plant safety of existing configurations of safety-related DC power supply annunciation and alarms.

Frequency Estimate

The concern in this issue was that the safety-related DC power supplies might be unavailable because of inadequate control room annunciators and alarms. This concern correlates with the results of NUREG-0666,[164] which included a FMEA and a PRA of a model DC power system. This model system consisted of two independent DC buses each of which were supplied by a single battery charger and had a single battery back-up. In addition, this system had the following alarms and annunciators in the control room: (1) battery charger ground alarm; (2) battery charger AC power supply failure alarm; (3) DC bus undervoltage alarm; (4) battery charger DC ammeter; and (5) battery charger DC voltmeter.

NUREG-0666[164] concluded that battery unavailability is dominated by inadequate maintenance practices and failure to detect battery unavailability due to bus connection faults. By improving battery surveillance, DC power system unreliability could be decreased by a factor of two, and improving maintenance and testing practices could decrease DC power system unavailability by a factor of 10. The report does not quantify a safety benefit which would result from additional alarms or annunciators in the control room, but additional alarms and annunciators would result in the enhancement of surveillance, maintenance and testing capabilities. Additional recommendations were made in NUREG-0666,[164] but these relate to aspects of the DC system which would not be enhanced by the addition of alarms or annunciators, such as the addition of a third DC power train.

In addition to the concerns relating to alarms and annunciators, the responses to Generic Letter 91-06[1399] also identified concerns with the probability of CCF of the DC power supplies. In order to evaluate these two concerns, the PRAs for 6 licensees were reviewed and found to include basic events which modeled the probability of battery unavailability and common cause battery failure. A study was performed to determine the effect on the CDF of decreasing battery unavailability and common cause battery failure probability. This study was performed by the staff using the SARA[1456] software. The results are described below.

The assumption was made that improved alarms and annunciators would result in continuous battery condition indication and would essentially result in an undetected battery failure probability of zero, since the operators would be notified of a DC power system failure immediately. However, this approximation would give a greater estimate of the effectiveness of modifications of alarms and annunciators than could actually be obtained. A better estimate of the effect on DC power system reliability resulting from an increase in the number of alarms and annunciators in the control room was obtained by decreasing the battery unavailability from the base case value to a test case value of 10^{-6}. For the plants considered in this analysis, the base case values ranged from 6.12×10^{-3} to 7.2×10^{-4}, which reflects an hourly failure rate of approximately 10^{-6}/hour, and an interval between tests which are capable of detecting a failed battery ranging from 6,120 to 720 hours.

This modification in battery unavailability will also account for any decrease in the battery charger unavailability resulting from the additional hardware. Because the battery must be instantaneously available to supply power if the battery charger fails, the battery unavailability terms in a PRA model are always multiplied by the battery charger unavailability terms. This analysis is conservative because it overestimates the effectiveness of additional alarms and annunciators, which will improve DC power system reliability by a much smaller factor. In addition, this approximation is made under the assumption that the DC power systems have been accurately modeled by PRA analysts for the existing PRAs and is only valid if the configuration of alarms and annunciators modelled by the existing PRAs is less effective than the currently recommended configuration.

CCF of the DC power system can be caused by maintenance activity, the most significant of which is inadvertent connection of redundant trains. Generic Letter 91-11[1400] addressed the use of interconnections between Class 1E vital instrument buses and LCOs for Class 1E vital instrument buses. The purpose of this generic letter was to decrease the probability and sources of CCF of redundant Class 1E AC and DC buses and inverters. It was assumed that CCF of the Class 1E buses and inverters has been adequately addressed and the scope of this issue was limited to the batteries and battery chargers.

The SARA[1456] software was used to model the effect of decreasing battery unavailability. There are currently nine operating plants which have PRA models which can be used with SARA. These are listed below, in addition to the configuration of the DC power system at the plant.

Plant	Number of 125V DC Batteries	Number of Battery Chargers
Grand Gulf 1[1318]	3	6
Brunswick 1 & 2*	4 (each)	4 (each)
Peach Bottom 2*	4	4
Surry 1[1318]	2 + diesel	2
Sequoyah 1[1318]	2 + diesel + 1 common	2 + 1 common
Oconee-3[889]	2	3
Zion[1318]	2 + 1 common	2 + 1 common
Indian Point-2	4	4

* Based on IPE Submittal

Peach Bottom-2: This unit has two independent divisions of safety-related 125V DC power, one of which is required to safely shut down the plant. Each division is comprised of two batteries, each with it's own charger. The control room has 3 of 7 recommended alarms and 1 of 4 recommended annunciators. The Peach Bottom PRA included probability terms for battery unavailability due to common mode failure and unavailability of the individual Unit 2B and 3C battery banks. The terms for the remaining battery banks (2A, 2C, 2D, and 3D) were not included in any significant minimal cutsets, and decreasing these basic event probabilities would have a negligible effect on the CDF. The probability of battery unavailability was estimated in the original PRA to be 0.001.

Peach Bottom-2: Common Mode Battery Failure

Probability	CDF/RY	Change/RY
0.001	3.6×10^{-6}	base case
0.000001	3.4×10^{-6}	-2.0×10^{-7}

Peach Bottom-2: Battery 2B and 3C Failure

Probability	CDF/RY	Change/RY
0.001	3.6×10^{-6}	base case
0.000001	3.6×10^{-6}	-

Decreasing the probability of common mode battery unavailability by three orders of magnitude would result in a decrease in CDF of 2.0×10^{-7}/year, whereas decreasing the probability of the unavailability of batteries 2B and 3C would result in less than a 10^{-7} decrease in CDF.

Grand Gulf-1: This unit has three independent divisions of safety-related 125V DC power, two of which are required to safely shut down the plant. The control room has 1 of 7 recommended alarms and 1 of 4 recommended annunciators. The Grand Gulf PRA included terms for the probability of battery common mode failure and failure of the individual Unit 1A3, 1B3, and 1C3 battery banks. All battery banks were included in significant minimal cutsets.

Grand Gulf-1: Common Mode Battery Failure

Probability	CDF/RY	Change/RY
0.001	2.1×10^{-6}	base case
0.000001	1.6×10^{-6}	-5.0×10^{-7}

Grand Gulf 1 - Loss of Power from Batteries 1A3, 1B3, 1C3

Probability	CDF/RY	Change/RY
0.001	2.1×10^{-6}	base case
0.000001	1.9×10^{-6}	-2.0×10^{-7}

Decreasing common mode battery unavailability by three orders of magnitude would result in a decrease in CDF of 5×10^{-7}/RY, whereas decreasing the unavailability of battery 1A3, 1B3 and 1C3 would result in a decrease of 2×10^{-7} in CDF.

Brunswick-1 and 2: These units each have two independent divisions of safety-related 125V DC power, one of which is required to safely shut down the plant. Each division is comprised of two independent batteries, each with its own charger. The control room has 5 of 7 recommended alarms and 2 of 4 recommended annunciators. The Brunswick Units 1 and 2 PRAs included terms for the probability of individual battery bank unavailability but not for common cause unavailability. The terms for failure of three of the four batteries were included in some minimal cutsets.

Brunswick-1: Battery Bank 1A1, 1A2, and 1B1 Fault

Probability	CDF/RY	Change/RY
0.00033	2.47×10^{-5}	base case
0.000001	2.46×10^{-5}	-1.0×10^{-7}

Brunswick-2: Battery Bank 2A1, 2A2, and 2B1 Fault

Probability	CDF/RY	Change/RY
0.00033	2.08×10^{-5}	base case
0.000001	2.06×10^{-5}	-2.0×10^{-7}

Units 1 and 2 differed slightly in their response to battery failure rate changes. However, decreasing the unavailability of battery 2A1, 2A2, and 2B1 would result in a decrease of 10^{-7}/RY and 2×10^{-7}/RY in CDF for Unit 1 and 2, respectively.

Surry-1: This unit has two independent divisions of safety-related 125V DC power, one of which is required to safely shut down the plant. The unit also has dedicated batteries for starting the diesel generators. The control room has 4 of 7 recommended alarms and 1 of 4 recommended annunciators. The Surry PRA included terms for the probability of battery common mode failure and failure of the individual I and II battery banks. Neither the common mode battery failure term or individual battery failure terms were included in any significant minimal cutsets. The assumed battery unavailability was 7.2×10^{-4}, which suggests a 2-month interval between tests that would detect battery problems for the typical failure rate. Because the CDF magnitude cutoff for exclusion of core damage sequences from the group of minimal cutsets is usually less than 10^{-8}, decreasing battery unavailability or common mode failure probability would result in a negligible decrease in CDF.

Sequoyah-1: This unit has two independent divisions of safety-related 125V DC power, one of which is required to safely shut down the plant. The unit also has dedicated batteries for starting the diesel generators. The control room has zero of 7 recommended alarms and 3 of 4 recommended annunciators. The Sequoyah PRA included probabilities for battery common mode unavailability and unavailability of the individual I and II battery banks. Battery unavailability was initially estimated to be 7.2×10^{-4}, which suggests a two-month surveillance test or maintenance interval for a failure rate of 10^{-6}/hour. The common mode unavailability was estimated to be 5.8×10^{-6}. Neither the common mode unavailability or individual battery unavailability were included in any significant minimal cutsets. The unavailabilities used in this analysis were slightly lower than those used in other analyses. However, the CDF magnitude cutoff for exclusion of core damage sequences from the group of minimal cutsets is usually less than 10^{-8} or less. Therefore, decreasing battery unavailability or common mode failure probability would result in a negligible decrease in CDF.

Oconee-3: This unit has two independent divisions of safety-related DC power, one of which is required to safely shut down the plant. The control room has 1 of 7 recommended alarms and none of 4 recommended annunciators. The Oconee PRA[889] included terms for unavailability of the individual 1CA, 1CB, 3CA, and 3CB battery banks. The probability of battery unavailability was estimated to be 6.12×10^{-3}, which is based on a one-year surveillance test or maintenance interval and a failure rate of 1.4×10^{-6}/hour. Common mode unavailability was not included in the PRA model. The individual battery unavailability terms were not included in any significant minimal cutsets. The probabilities used in this analysis were significantly greater than those used in other analyses. However, the CDF magnitude cutoff for exclusion of core damage sequences from the group of minimal cutsets is usually less than 10^{-8} or less. Therefore, decreasing battery unavailability or common mode failure probability would result in a negligible decrease in CDF.

The average decrease in CDF from the proposed modifications was estimated to be approximately 10^{-7}/RY.

Consequence Estimate

It was assumed that all affected operating plants had an average remaining life of 20 years, based on their original licenses. It was also assumed that each of these plants would be granted a life extension of 20 years. Thus, the average remaining life for all affected plants was 40 years.

The public risk associated with the event considered in this issue was estimated[64] to be 6.76×10^{6} man-rem and 2.52×10^{6} man-rem for BWRs and PWRs, respectively. For BWRs, the total potential risk reduction was estimated to be $(6.76 \times 10^{6})(10^{-7})(40)$ man-rem/reactor or 27 man-rem/reactor. For PWRs, the total potential risk reduction was estimated to be $(2.52 \times 10^{6})(10^{-7})(40)$ man-rem/reactor or 10 man-rem/reactor.

Cost Estimate

Improving the control room annunciators and alarms for all safety-related DC power systems at each plant would involve a different amount of effort for each licensee, depending upon the amount of instrumentation currently installed, available space for additional annunciators and alarms, and whether existing raceway could hold additional cables. In addition, new procedures and operator training would be required. This additional hardware would include the following:

(1)	Data transmitters at each battery room. Design, installation and testing assumed to be \$100,000/battery room, with 3 battery rooms per facility	\$300,000
(2)	Raceway and cable from each battery room to the control room. Design, installation and testing costs assumed to be \$100 per linear foot, with 1000 linear feet of raceway per battery room and 3 battery rooms per facility	\$300,000
(3)	Control room modifications to add annunciators and alarms. Design, installation and testing assumed to be \$100,000/battery, 3 batteries per facility	\$300,000
(4)	Procedure changes, drawing changes, training, and administrative costs	\$100,000
	TOTAL:	\$1,000,000

Value/Impact Assessment

Separate value/impact scores were calculated for PWRs and BWRs.

BWRs: Based on a potential public risk reduction of 27 man-rem/reactor and an estimated cost of \$1M/reactor for a possible solution, the value/impact score was given by:

$$S = \frac{27 \text{ man-rem/reactor}}{\$1\text{M/reactor}}$$

$$= 27 \text{ man-rem/\$M}$$

PWRs: Based on a potential public risk reduction of 10 man-rem/reactor and an estimated cost of \$1M/reactor for a possible solution, the value/impact score was given by;

$$S = \frac{10 \text{ man-rem/reactor}}{\$1\text{M/reactor}}$$

$$= 10 \text{ man-rem/\$M}$$

Other Considerations

(1) It is important to monitor the condition of the safety-related DC power system, including the condition of batteries which may be needed in the event of a station blackout. In addition, it is also necessary to have procedures which minimize the probability of a common cause fault of the safety-related DC power systems. Operating experience so far does not indicate that significant problems exist in this area.

(2) Based upon the results of this study, it could be asserted that the control room alarms and annunciators recommended by the staff in current licensing guidelines do not result in a significant increase in plant safety beyond that realized by existing alarm and annunciator configurations and weekly or quarterly maintenance programs. It should be noted that the empirical battery failure rate of approximately 10^{-6}/hour, which is used to determine battery unavailability, is dependent upon the frequency of battery failures for systems with existing configurations of control room annunciators and alarms. Therefore, it might not be accurate to conclude that the existing recommendations for annunciators and alarms should be relaxed.

(3) Battery unavailability and CCF are recognized by some licensees to be sufficiently probable so as to require modeling in PRAs. Based upon these PRA models, decreasing the unavailability of the batteries and safety-related DC power supplies by several orders of magnitude over that used in the base case does not result in a significant decrease in CDF for these licensees. This observation must be tempered with the knowledge that licensees currently monitor important DC bus parameters, and that other DC power system design features, such as the number of batteries, have a greater impact on DC power system reliability than the number of alarms and annunciators.

CONCLUSION

Based on the potential public risk reduction, this issue had a low priority ranking for BWRs and was in the drop category for PWRs (see Appendix C). Overall, the issue was given a low priority ranking in March 1993. Consideration of a 20-year license renewal period did not change the priority of the issue.[1564] Further prioritization, using the conversion factor of \$2,000/man-rem approved by the

Commission in September 1995, resulted in an impact/value ratio (R) of $37,037/man-rem which placed the issue in the DROP category.

ISSUE 156.3.8: SHARED SYSTEMS

DESCRIPTION

This issue is one of the nineteen category 3 issues identified by NRR in SECY-90-343.[1351] The sharing of the ESFS for a multi-unit plant, including onsite emergency power systems and service systems, can result in a reduction of the number and capacity of onsite systems to below that which is needed to bring either unit to a safe shutdown condition, or to mitigate the consequences of an accident. Shared systems for multiple unit stations should include equipment powered from each of the units involved. There were 13 multi-unit sites that could be affected by this issue among the 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976.

CONCLUSION

The safety concerns associated with systems that are shared by two or more units at multi-unit sites have been previously identified by the staff. The most important contributors to core damage probability at these sites have been determined to be air, cooling water, and electric power systems. These systems have been adequately addressed in Issues 43, 130, 153, and A-44. Based on these considerations, this issue was DROPPED from further pursuit as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.4.1: RPS AND ESFS ISOLATION

DESCRIPTION

This issue is one of the three Category 4 issues identified by NRR in SECY-90-343.[1351] The safety concern was that, in the event of non-safety system failures, the lack of isolation devices could result in the propagation of faults to safety systems and common cause failures may result. In its study, the staff found that approximately 39 plants at 28 sites were not required to meet IEEE 279-1971[397] and have not been reviewed for this safety concern since the time of their licensing. Non-safety systems generally receive control signals from the RPS and ESF sensor current loops. The non-safety circuits are required to be isolated to ensure the independence of the RPS and ESF channels. Requirements for the design and qualification of isolation devices are quite specific. Evaluation of the quality of isolation devices is not the safety issue of concern; rather, the issue is the existence of isolation devices which will preclude the propagation of non-safety system faults to safety systems.

CONCLUSION

The safety concerns of leakage through electrical isolators in instrumentation circuits and electrical isolation in plants not required to meet IEEE 279-1971[397] were addressed in the resolution of Issue 142. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change this conclusion.

ISSUE 156.4.2: TESTING OF THE RPS AND ESFS

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The objective of this issue was to review plant designs to ensure that: (1) all ECCS components, including the pumps and valves, are included in the component and system test; (2) the frequency and scope of periodic testing are identified; and (3) the test programs will provide adequate assurance that the systems will function when needed. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

CONCLUSION

A portion of this issue was covered by existing requirements; specifically, ECCS pumps and valves are required to be tested quarterly by the ASME Code in accordance with 10 CFR 50.55(a), unless the NRC grants relief to defer testing until refueling outages. The remainder of this issue was covered in the resolution of Issue 120 which addressed the concern regarding on-line (at-power) testability af protection systems (both the RPS and the ESFS) and the possibility that some plants may not provide complete testing capability at power. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change this conclusion.

ISSUE 156.6.1: PIPE BREAK EFFECTS ON SYSTEMS AND COMPONENTS

DESCRIPTION

Historical Background

In 1967, the AEC published draft GDCs for comment and interim use and, until 1972, the staff's implementation of the GDCs required consideration of pipe break effects inside containment. However, due to the lack of documented review criteria, AEC staff positions continued to evolve. Review uniformity was finally developed in the early 1970s, initiated by a November 9, 1972, note from L. Rogers to R. Fraley, in which a Draft Safety Guide entitled "Protection Against Pipe Whip Inside Containment" was proposed. This Draft Guide contained some of the first documented deterministic criteria that the staff had used for several years (to varying degrees) as guidelines for selecting the locations and orientations of postulated pipe breaks inside containment, and for identifying the measures that should be taken to protect safety-related systems and equipment from the dynamic effects of such breaks. Prior to use of these deterministic criteria, the staff used non-deterministic guidelines on a plant-specific basis. The Draft Safety Guide was subsequently revised and issued in May 1973 as Regulatory Guide 1.46[18] for implementation on a forward-fit basis only.

The AEC issued two generic letters to all licensees and CP or OL applicants regarding pipe break effects outside containment in December 1972[139] and July 1973. These letters, known as the "Giambusso" and "O'Leary" letters, respectively, extended pipe break concerns to locations outside containment, and provided deterministic criteria for break postulation and evaluation of the dynamic effects of postulated breaks. The letters requested all recipients to submit a report to the staff summarizing each plant-specific analysis of the issue. All operating reactor licensees and license applicants submitted the requested analyses in separate correspondence or updated the SARs for their proposed plants to include the analysis. The staff reviewed the submitted analyses and prepared safety evaluations for all plants. In November 1975, the staff published SRP[11] Sections 3.6.1 and 3.6.2 that slightly revised the two generic letters discussed above. Thus, after 1975, the specific structural and environmental effects of pipe whip, jet impingement, flooding, etc., on systems and components relied on for safe reactor shutdown were considered.

As stated above, the AEC/NRC has provided requirements to the industry regarding pipe breaks outside of containment through the issuance of the Giambusso and O'Leary generic letters. Since these requirements are applicable to all the affected plants, pipe breaks outside of containment were judged to be a compliance issue and were not considered in this analysis. Compliance matters are dealt with promptly and do not await the generic issue resolution process. Therefore, the issue of pipe breaks outside of containment for the 41 affected plants was brought to the attention of NRR by separate correspondence.[1761] The remainder of this evaluation only addressed pipe breaks inside containment.

As a part of its plant-specific reviews between 1975 and 1981, the staff used the guidelines in Regulatory Guide 1.46[18] for postulated pipe breaks inside containment, and SRP[11] Sections 3.6.1 and 3.6.2 for outside containment. In July 1981, SRP[11] Sections 3.6.1 and 3.6.2 were revised to be applicable to both outside and inside containment, thus eliminating the need for further use of Regulatory Guide 1.46,[18] which was subsequently withdrawn.

Between the period 1983-1987, the general issue of pipe breaks inside and outside containment was revisited in the SEP. The objective of the SEP was to determine to what extent the earliest 10 plants (i.e., SEP-II) met the licensing criteria in existence at that time. This objective was later interpreted to ensure that the SEP also provided safety assessments adequate for conversion of provisional operating licenses (POLs) to full-term operating licenses (FTOLs). As a result of these reviews, plants were required to perform engineering evaluations, TS or procedural changes, and physical modifications both inside and outside containment. Regarding inside containment modifications: of the two SEP-II plants evaluated in this analysis (one BWR and one PWR), the BWR was required to modify four piping containment penetrations and the PWR was required to modify steam generator blowdown piping supports. This indicates there was a wide spectrum of implementation associated with the original reviews of these early plants for pipe breaks inside and outside containment.

As with the above-described evolution of uniform pipe break criteria, electrical systems design criteria were also in a state of development. Prior to 1974, electrical system designs were generally reviewed in accordance with the guidelines provided in IEEE-279; however, significant variations in interpretations of that document resulted in substantial design differences in plants. Specifically, true physical separation of wiring to redundant components was not necessarily accomplished. In 1974, Regulatory Guide 1.75 was published, clarifying the requirements.

An earlier evaluation of this issue resulted in a medium-priority ranking (see Appendix C) with the finding that the scope could be limited to pipe breaks inside containment, since the NRC had already provided requirements regarding outside containment pipe breaks to the industry through the issuance of the Giambusso and O'Leary generic letters. However, the uncertainty in the analysis was much wider than desired for a definitive priority ranking. Thus, the issue appeared to warrant additional analysis to enhance the prioritization. In July 1994, a contract was awarded to INEEL to:

(1) Review pipe failure rate data, pipe break methodologies, and related publications to determine recommended pipe failure rates (initiating events) applicable to the affected SEP-III plants.

(2) Review updated FSARs and related SERs for SEP-II, SEP-III, and for representative non-SEP plants to identify and prioritize potential safety concerns (i.e., accident sequences). Several plant visits and walkdowns were included as part of this review.

(3) Estimate changes to core damage frequencies for accident sequences that are determined to be of high or medium priority.

(4) Identify potential corrective actions and their estimated costs.

The evaluation that follows was based on the results of the INEEL research documented in Draft NUREG/CR-6395..

Safety Significance

GDC 4 is the primary regulatory requirement of concern. It requires, in part, that structures, systems and components important to safety be appropriately protected against the environmental and dynamic effects that may result from equipment failures, including the effects of pipe whipping and discharging fluids. Several possible scenarios for plants that do not have adequate protection against pipe whip were identified as a result of the research performed in support of the enhanced prioritization. Related regulatory criteria include common cause failures, protection system independence, and the single failure criterion.

Possible Solution

Issue generic letters to the affected plants requesting that they perform plant-specific reviews and walkdowns, identify vulnerable pipe break locations, and inform the NRC of proposed corrective actions.

PRIORITY DETERMINATION

Numerous scenarios of potential concern were evaluated. The following were considered important enough to be specifically identified for future consideration. All estimated frequencies and probabilities are mean values.

Frequency Estimate

BWRs

Case 1: Failure of Main Steam or Feedwater Piping Resulting in Pipe Whip and Containment Impact/Failure, with Resultant Failure of All Safety Injection Systems

This event (INEEL BWR Event 1) involved a BWR with a Mark I steel containment; 15 of the 16 affected BWRs were of this design. A DEGB of an unprotected (i.e., no pipe whip restraint or containment liner impact absorber) large reactor coolant recirculation pipe inside containment and near the containment liner might result in puncturing of the liner. The resulting unisolable LOCA steam environment would be introduced into the secondary containment building, possibly disabling the ECCS equipment located there. This scenario would greatly increase the probability of core damage and potential offsite doses.

All of the affected BWRs were more than 10 years old and most used Type 304SS in the primary system piping, a material that was susceptible to IGSCC degradation. It should be noted that piping of this material did not qualify for the extremely low rupture probability (leak-before-break) provision of GDC 4. From NUREG-1150,[1081] the recirculation loop DEGB frequency for this material was estimated to be 10^{-4}/RY. The fraction of BWR primary piping inside containment that was either main steam or feedwater was estimated to be 0.4. The fraction of main steam or feedwater piping that can impact the containment metal shell was estimated to be 0.25.

The research performed indicated that there was considerable variation among the affected plants regarding the amount of pipe whip protection provided and the proximity of high energy lines to potential targets of concern, including redundant trains (see Other Considerations). It was assumed that the probability of a main steam or feedwater broken pipe rupturing the containment metal shell was 0.25.

The postulated event may also cause a common mode failure of the ECCS system since much of this equipment was located within the secondary containment and will be exposed to a harsh environment beyond its design basis, or that the ECCS piping will fail due to overpressurization of the containment annulus. In most of the affected plants, the ECCS is located in four different quadrants outside the suppression pool (torus). On the other hand, as stated above, redundant electrical power systems and initiating circuitry may not be physically separated in the older plants. Also, if the ECCS operates initially, the ECCS equipment rooms may not be fully protected from internal flooding as the water from the suppression pool flows out the broken pipe into the secondary containment. Based on these considerations, the mean probability of loss of ECCS function was assumed to be 0.8. Based on the above assumptions, the mean value of change in CDF was 2 x 10^{-6}/RY.

From WASH-1400,[16] the nearest scenario to that described above was the large LOCA BWR-3 release category involving a large LOCA and subsequent containment failure. However, in the WASH-1400[16] case, the containment failure results from overpressurization, not from pipe whip. Three of the four specific BWR-3 large LOCA accident sequences have an incidence frequency of 10^{-7}/RY, and the remaining one is 10^{-6}/RY; 10^{-7}/RY was chosen as the base case for this analysis.

<u>Case 2:</u> <u>Failure of Recirculation Piping Resulting in Pipe Whip and Containment Impact/Failure, With Resultant Failure of All Emergency Core Cooling Systems</u>

This event (INEEL BWR Event 9) was similar to Case 1 but involved the recirculation system piping. From NUREG-1150,[1081] the recirculation loop DEGB mean frequency for this material was estimated to be 10^{-4}/RY. The fraction of BWR primary piping inside containment that is recirculation piping was estimated to be 0.2. The fraction of recirculation piping that can impact the containment metal shell was estimated to be 0.5. It was estimated that the mean probability of a recirculation system broken pipe rupturing the containment metal shell was 0.5. The mean probability of eventual failure of all ECCS by the same modes described for Case 1 was estimated to be 0.8. Based on the above assumptions, the mean value of change in CDF was 4 x 10^{-6}/RY.

Case 3: Failure of RHR Piping Resulting in Pipe Whip and Containment Impact/Failure, With Resultant Failure of All Emergency Core Cooling Systems

This event (INEEL BWR Event 12) was similar to Cases 1 and 2 but involved the RHR System piping. From NUREG-1150,[1081] the RHR DEGB frequency for this material was estimated to be 10^{-4}/RY. The fraction of BWR primary piping inside containment that is RHR piping was estimated to be 0.1. The fraction of RHR piping that can impact the containment metal shell was estimated to be 0.5. The mean probability of a recirculation system broken pipe rupturing the containment metal shell was 0.1. The mean probability of eventual failure of all ECCS by the same modes described for Cases 1 and 2 was estimated to be 0.8. Based on the above assumptions, the mean value of change in CDF/RY was 4×10^{-7}/RY.

Case 4: Failure of Recirculation Piping Resulting in Pipe Whip or Jet Impingement on Control Rod Drive Bundles, Causing Failure by Crimping of Enough Insert/Withdraw Lines to Result in Failure to Scram the Reactor

This case corresponded to INEEL BWR Event 5. From NUREG-1150,[1081] the recirculation loop DEGB frequency for this material was estimated to be 10^{-4}/RY. The fraction of BWR primary piping inside containment that is recirculation piping was estimated to be 0.2. The fraction of recirculation piping that can impact or impinge on the CRD lines was estimated to be 0.25. It was estimated that the mean probability of a broken RHR pipe crimping enough CRD lines to prevent a scram (about 5 to 10 adjacent lines) was 1. Based on the above assumptions, the mean value of change in CDF was estimated to be 5×10^{-6}/RY.

Case 5: Failure of RHR Piping Resulting in Pipe Whip or Jet Impingement on Control Rod Drive Bundles, Causing Failure by Crimping of Enough Insert/Withdraw Lines to Result in Failure to Scram the Reactor

This event (INEEL BWR Event 10) was similar to Case 3 but involved the RHR system piping. The research performed indicated that there was considerable variation among the affected plants regarding the amount of pipe whip protection provided and the proximity of high energy lines to potential targets of concern. Walkdowns showed that, in at least one case, a large "unisolable from the RCS" RHR line was routed directly between the two banks of CRD bundles. An RHR pipe break in this vicinity would impinge and/or impact on both banks simultaneously.

From NUREG-1150,[1081] the RHR DEGB frequency for this material was estimated to be 10^{-4}/RY. The fraction of BWR primary piping inside containment that consitutes RHR piping was estimated to be 0.1. The fraction of RHR piping that can impact or impinge on the CRD lines was estimated to be 0.25. It was estimated that the mean probability of a broken RHR pipe crimping enough CRD lines to prevent a scram (about 5 to 10 adjacent lines) was 1. Based on the above assumptions, the mean value of change in CDF was 2.5×10^{-6}/RY.

Case 6: Failure of High Energy Piping Resulting in Pipe Whip or Jet Impingement on Reactor Protection or Instrumentation & Control Electrical, Hydraulic or Pneumatic Lines, or Components and Eventually Resulting in Failure of Mitigation Systems and Core Damage

This case corresponded to INEEL BWR Event 14. From NUREG-1150,[1081] the large LOCA frequency is 10^{-4}/RY. All high energy piping inside containment was considered. The fraction of high energy piping that can impact or impinge on these lines or components was estimated to be

0.5. The mean probability of a broken high energy line failing some of these lines or components to the extent that core damage results was estimated to be 0.75. Based on the above assumptions, the mean value of change in CDF was 3.8×10^{-5}/RY.

Case 7: Failure of High Energy Piping Resulting in Pipe Whip Impact on Reactor Building Component Cooling Water (RBCCW) System to the Extent That the RBCCW Pressure Boundary is Broken, Potentially Opening a Path to Outside Containment if Containment Isolation Fails to Occur; Also Possible Loss of RBCCW Outside Containment for Mitigation

This case corresponded to INEEL BWR Event 16. From NUREG-1150,[1081] the large LOCA frequency was 10^{-4}/RY. All high energy piping inside containment was considered. The fraction of high energy piping that can impact the RBCCW system was estimated to be 0.1. The probability of an HELB broken pipe rupturing the RBCCW system was 0.5. The probability of failure to close of containment isolation check valve was 10^{-3}; the probability of failure to close of a containment isolation MOV was 3×10^{-3}. These scenarios had a combined total probability of 4×10^{-3}. Since the RBCCW surge tank in the secondary containment is vented to atmosphere and has a relatively small volume, it was assumed that its water inventory will drain quickly; for this reason, the mean probability of opening a path to atmosphere outside containment was 1. Once this scenario proceeds to this point, the RBCCW system in the secondary containment will become unavailable, including the RHR heat exchanger; therefore, the probability of losing the RBCCW function outside containment to the extent that core damage occurs was 1. Based on the above assumptions, the mean value of change in CDF was estimated to be 2×10^{-8}/RY.

The total change in CDF for the above 7 BWR cases was estimated to be 5.2×10^{-5}/RY. For all 16 affected BWRs, ΔCDF was 8.3×10^{-4}/RY.

PWRs

Case 1: Failure of Non-Leak-Before-Break Reactor Coolant System, Feedwater, or Main Steam Piping Resulting in Pipe Whip or Jet Impingement on Reactor Protection or Instrumentation & Control Electrical, Hydraulic or Pneumatic Lines or Components and Eventually Resulting in Failure of Mitigation Systems and Core Damage

This case corresponded to INEEL PWR Event 9. From NUREG-1150,[1081] the HELB frequency in the above-listed systems was 1.5×10^{-3}/RY. All of the listed high energy piping inside containment was considered. The fraction of high energy piping that can impact or impinge on these lines or components was estimated to be 0.1. The mean probability of a broken high energy line failing some of these lines or components to the extent that core damage results was estimated to be 0.5. Based on the above assumptions, the mean value of change in CDF was 7.5×10^{-5}/RY.

Case 2: Failure of Main Steam or Feedwater Piping Resulting in Pipe Whip and Containment Impact/Failure, with Resultant Failure of All Emergency Core Cooling Systems

This case corresponded to INEEL PWR Event 16. From NUREG-1150,[1081] the DEGB frequency in feedwater piping was estimated to be 4×10^{-4}/RY; for main steam piping, it was estimated to be 10^{-4} /RY. The fraction of feedwater piping that can impact the containment shell was estimated to be 0.1. The fraction of main steam piping was also estimated to be 0.1; this fraction remained 0.1. The mean probability of a feedwater or main steam system broken pipe rupturing the containment metal shell was 0.5. The mean probability of additional I&C or ECCS systems failures to the extent that core damage results was estimated to be 4.8 x 10^{-5} for the case involving feedwater piping breaks, and 9.8 x 10^{-5} for the case involving main steam piping breaks. Based on the above assumptions, the mean value of change in CDF was 1.4 x 10^{-9}/RY.

Case 3: Failure of Main Steam or Feedwater Piping Resulting in Pipe Whip Impact on CCW System to the Extent That the CCW Pressure Boundary is Broken, Potentially Opening a Path to Outside Containment if Containment Isolation Fails to Occur; Also Possible Loss of CCW Outside Containment for Mitigation

This case corresponded to INEEL PWR Event 17. From NUREG-1150,[1081] the DEGB frequency in feedwater piping was estimated to be 4 x 10^{-4}/RY; for main steam piping, it was estimated to be 10^{-4}/RY; this combined for a total frequency of 5 x 10^{-4}/RY. The fraction of feedwater piping that can impact the CCW system was estimated to be 0.1; the fraction of main steam piping was also estimated to be 0.1; this fraction remained 0.1. The probability of a feedwater or main steam system broken pipe rupturing the CCW system was 0.5. The probability of failure to close of containment isolation check valve was 10^{-3}; the probability of failure to close of a containment isolation MOV was 3 x 10^{-3}; this combined for a total probability of 4 x 10^{-3}. Since the CCW surge tank is in the auxiliary building near mitigation equipment, is vented to atmosphere, and has a relatively small volume, it was assumed that its water inventory will drain quickly. For this reason, the mean probability of opening a path to atmosphere outside containment was 1. Once this scenario proceeds to this point, the CCW system outside containment will become unavailable, including the RHR heat exchanger. Therefore, the probability of losing the CCW function outside containment, to the extent that core damage occurs, is 1. Based on the above assumptions, the mean value of change in CDF was 10^{-7} /RY.

The total change in CDF for the above three PWR cases was 7.5 x 10^{-5}/RY. For all 25 affected PWRs, the ΔCDF was estimated to be 1.9 x 10^{-3}/RY.

Consequence Estimate

TABLE 3.156-1
BWR Offsite Dose Table

NUREG/CR-6395 Event Number	ΔCDF (Event/RY)	WASH-1400[16] Release Category	WASH-1400[16] Offsite Dose (Man-rem/Event)	Offsite Dose (Man-rem/RY)
Event 1	2.0 x 10^{-6}	BWR-3	5.1 x 10^{6}	10.2
Event 5	5.0 x 10^{-6}	BWR-4	6.1 x 10^{5}	3.1
Event 9	4.0 x 10^{-6}	BWR-3	5.1 x 10^{6}	20.4
Event 10	2.5 x 10^{-6}	BWR-4	6.1 x 10^{5}	1.5
Event 12	4.0 x 10^{-7}	BWR-3	5.1 x 10^{6}	2.0
Event 14	3.8 x 10^{-5}	BWR-4	6.1 x 10^{5}	23.2
Event 16	2.0 x 10^{-8}	BWR-3	5.1 x 10^{6}	0.1
TOTAL:				**60.5**

For the 16 affected BWRs with an average remaining life of 17 years, the estimated change in offsite dose was (60.5 man-rem/RY)(16 reactors)(17years) or 16,464 man-rem.

TABLE 3.156-2
PWR Offsite Dose Table

NUREG/CR-6395 Event Number	ΔCDF (Event/RY)	WASH-1400[16] Release Category	WASH-1400[16] Offsite Dose (man-rem/event)	Offsite Dose (man-rem/RY)
Event 9	7.5×10^{-5}	PWR-6	1.5×10^{5}	11.3
Event 16	1.4×10^{-9}	PWR-4	2.7×10^{6}	0.004
Event 17	1.0×10^{-7}	PWR-4	2.7×10^{6}	0.3
TOTAL:				**11.6**

For the 25 affected PWRs with an average remaining life of 17 years, the estimated change in offsite dose was (11.6 man-rem/RY)(25 reactors)(17 years) or 4,925 man-rem. Thus, the estimated total offsite dose for the 41 affected plants was (16,464 + 4,925) man-rem or 21,389 man-rem.

Cost Estimate

Industry Cost: Implementation of the possible solution was assumed to require the performance of engineering analyses inside containment, perform system walkdowns, and provide a report to the NRC. Ultimately, it was expected that operating procedures and/or TS will be modified, inservice inspections will be enhanced, or physical modifications will be done either to piping (probably addition of pipe whip restraints or jet shields) or to the inside containment leakage detection system. It is expected that the cost to each plant will be $1M. Therefore, for the 41 affected plants (16 BWRs and 25 PWRs), the total implementation cost was estimated to be $41M. This estimate was based on the presumption that the level of effort at the affected plants would be similar to that which resulted for this issue during the SEP program review of the 10 earliest SEP plants.

NRC Cost: Development and implementation of a resolution was estimated to cost $1M, primarily involving review of industry submittals and possible proposed changes to hardware.

Total Cost: The total industry and NRC cost associated with the possible solution was estimated to be $42M.

Impact/Value Assessment

Based on a potential public risk reduction of 21,389 man-rem and an estimated cost of $42M for a possible solution, the impact/value ratio was given by:

$$R = \frac{\$42M}{21{,}389 \text{ man-rem}}$$

$$= \$1{,}960/\text{man-rem}$$

Other Considerations

(1) The updated SAR for an SEP-III BWR (i.e., one of the 41 plants potentially affected by this issue) stated that, in the event of a DEGB, the broken pipe would strike the Mark I Containment and deform it significantly. However, another BWR of about the same vintage is known to have been required to add energy absorbing structures to protect the Mark I Containment from pipe whip, prior to receipt of an operating license. Therefore, it appeared that there was considerable variation among the affected plants regarding the amount of pipe whip protection provided.

(2) Pipe breaks have actually occurred in the industry. Examples include a Surry feedwater line break, a WNP-2 Fire System valve structural pressure boundary failure, and a Ft. Calhoun 12" steam line break.

(3) Some suspect configurations were observed in the SEP-III walkdown plants, e.g., at one BWR a very close proximity exists between a large RHR (unisolable from RCS) pipe and both banks of the CRD piping, and at one PWR it appeared that a large volume of piping penetrated the containment near where a large amount of electrical wiring also penetrated the containment. This demonstrated that, even through modest efforts (i.e., sampling walkdowns of a sampling of plants), configurations of potential concern have been identified.

(4) Readily available plant documentation provides very little insights regarding actual proximity of high energy piping and potential targets or concern. The potential lack of adequate separation of redundant system targets (e.g., I&C electrical wiring) is also a concern.

(5) Uncertainty remains a significant factor because of the large scope of this issue. This is because of the large number and types of plants, and significant differences in the specific as-built details applicable to this issue.

(6) Many of the affected plants are either currently applying for life extension or are expected to in the near future. Most of the lead life extension applications will be from the affected plants for many years to come.

(7) Although there is a large apparent disparity between the BWR and PWR cases evaluated, it must be remembered that much of the background of this issue was based on sampling walkdowns, i.e., only selected portions of selected plants were available for these walkdowns. Therefore, it is important to treat the BWR and PWR evaluations equally during the next phase of the evaluation. Also, some of the listed scenarios seem to have low probabilities but potentially high consequences. They should be further evaluated.

(8) Assuming a life extension of 20 years for the 31 affected plants, the public risk reduction would be 35,824 man-rem and 10,725 man-rem for BWRs and PWRs, respectively. This would produce an impact/value ratio of $900/man-rem.

CONCLUSION

Several potential accident scenarios were identified; 7 for BWRs and 3 for PWRs. Mean values for core damage were estimated for each and the cumulative effect of each group was also estimated. The total change in CDF was 8.3×10^{-4}/year for the 16 affected BWRs and 7.5×10^{-5}/RY for the

3 PWR cases. This would give the issue a medium/high priority ranking (see Figure 2 of NUREG-0933). For all 25 affected PWRs, ΔCDF/Year was 1.9×10^{-3}, which would also give the issue a high/medium priority ranking. Further evaluations which included estimates of offsite doses and costs for potential solutions showed that the issue has a HIGH priority ranking.[399]

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ISSUE 170: FUEL DAMAGE CRITERIA FOR HIGH BURNUP FUEL

DESCRIPTION

Historical Background

Following the TMI-2 accident, the NRC converted its fuel behavior research program into a severe accident research program and, consequently, no further confirmatory work on fuel damage criteria was pursued. However, some work of this type was conducted by foreign agencies and, in March 1994, the NRC received the results from a reactivity test in the French Cabri test reactor which showed multiple brittle failures at a relatively low energy of 30 calories/gram (cal/gm) in the cladding of a commercial fuel rod with a burnup of 63 gigawatt-days/ton (GWd/t); dispersal of finely divided fuel particles was also observed. Test results from the Japanese Nuclear Safety Research Reactor (NSRR) and the Russian Impulse Graphite Reactor (IGR) also appeared to indicate reduced failure thresholds and fuel dispersal. The corresponding failure threshold used by the NRC for some similar situations is 170 cal/gm and no loss of fuel is assumed below 280 cal/gm.

Licensee requests for higher burnup fuel limits in operating reactors resulted in the issuance of an NRR request[1595] for an RES assessment of fuel damage thresholds for reactivity transients. Confirmatory analysis of fuel conditions required RES to request experimental data from the French test.[1596] Information Notice 94-64[1597] was issued and the Commission was informed[1598] of the staff's actions. Following staff review of the Cabri test data, an assessment[1599] of its safety significance was forwarded to the Commission. High burnup fuel behavior was the subject of Session 11 of the Twenty-Second Water Reactor Safety Information Meeting, the transactions of which were documented in NUREG/CP-0139.[1600]

Safety Significance

Changes in fuel pellets and cladding occur at high burnups that appear to reduce fuel resistance to damage. Two of these changes are: (1) a reduction in cladding ductility that results from neutron damage and internal hydriding associated with oxidation; and (2) the formation of a very fine grain structure in the fuel pellets and the accumulation of microscopic fission gas bubbles on the grain boundaries. While the underlying processes that cause these changes have been known for many years, the extent and effects of these changes were not realized until recently.

Fuel damage criteria for LOCAs are also brought into question at very high burnups. To avoid fragmentation during quenching, 10 CFR 50.46 requires that the peak cladding temperature not exceed 2200°F and that the total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. Reduced ductility at high burnup due to internal hydriding may affect the validity of these limits. Further, oxidation during normal operation to burnups above 50 GWd/t has been observed to be on the order of 15% of the cladding thickness, leaving little margin for additional oxidation during an accident.

GDC 10 requires that specified acceptable fuel design limits are not to be exceeded during normal operation, including anticipated operational occurrences. Such limits are specified by applicants and approved by NRC, rather than being prescribed. The limits are intended to keep the fuel rod cladding from leaking, thus protecting its function as a fission product barrier; a commonly used value is a 1% limit on cladding strain. Ductility of cladding from fuel with burnups around 60 GWd/t is found to be reduced by a factor on the order of 5, compared with unirradiated cladding, with observed uniform strains of 1% or less at the time of rupture. It is thus unlikely that a 1% strain limit is providing the protection desired for high burnup fuel. As discussed below, the safety significance of these fuel damage criteria varies depending on the type of event

Reactivity Transients: These transients can lead to pressure pulses in the coolant, loss of coolable fuel geometry, and releases of radionuclides. The absence of pressure pulses, implied by the present criteria, is probably not affected by high burnups; while fine particulates may be formed, high temperatures (above 300 cal/gm) required to generate pulses should not occur in high burnup fuel by an even wider margin. Fuel dispersal constitutes loss of coolable geometry and may occur with low energy depositions at high burnup; however, the worst reactivity transients are very localized, and localized loss of geometry should not lead to a core-melt. The lower threshold for release of gap activity plus the dispersal of particulate fuel at high burnup would increase plant activity and public dose, and the dispersal of fuel would alter the character of these Chapter 15 transients.

LOCAs: Excessive oxidation of cladding (i.e., more than 17%) on high burnup fuel could, in principle, lead to loss of coolable geometry from hydraulic loads (from the ECCS) without ever experiencing high temperatures during a blowdown. On the other hand, additional oxidation may not occur during the transient regardless of the amount of initial oxidation present on high burnup fuel. Transient temperatures for these accidents are very sensitive to power level, and even marginally lower power levels in high burnup fuel might keep transient temperatures sufficiently low to avoid further oxidation. If either the criteria or the safety analyses fail to provide a margin to loss of coolable geometry, then a core-melt could result from these design basis accidents.

Specified Acceptable Fuel Design Limits: Failure of the 1% strain limit or any other "specified acceptable fuel design limit" to provide the assurance that is assumed would result in plant releases or public doses that are not permitted during normal operation. None of these situations, however, should lead to core damage.

Possible Solution

Resolution of this issue could be accomplished by updating the existing burnup-independent criteria to include the effects of burnup, or to develop substitute criteria, as appropriate. Updated criteria could be incorporated in revisions to 10 CFR 50.46, Regulatory Guide 1.77,[1594] and SRP[11] Section 4.2, as necessary. Implementation of the resolution would require a screening of certain approved licensing topical reports and reloads that were reviewed previously to permit reactor operation to high burnups.

PRIORITY DETERMINATION

Existing and emerging data, largely from foreign sources, are expected to be adequate for the criteria revisions envisioned. At the time the issue was evaluated in May 1995, NRC programs to obtain and analyze these data were being planned or were in place; no major new testing programs were anticipated. The provisions in 10 CFR 50.46 that are in question were controversial when originally established and changes to this regulation will be avoided unless absolutely necessary; it is possible that the existing criteria can continue to be used as long as careful attention is given to initial oxidation and method of analysis for high burnup fuel.

Significant changes in exposure of plant operating staff are not expected during normal operation. Although the specified acceptable fuel design limits are probably not providing the protection intended, it is believed that licensees are employing more stringent measures that are not derived from the licensing safety analysis, e.g., power maneuvering restrictions and barrier fuel designs are being used to reduce fuel failures, which the 1% strain limit would not prevent. On the other hand, reductions would be expected in exposure of plant operating staff following a major transient or accident. These reductions could be accomplished by changes in operating conditions or fuel designs such that fewer fuel failures would occur during accidents and attendant fuel dispersal would be avoided.

CONCLUSION

At the time this issue was identified, an action plan for resolving it had been developed by the staff and presented to the Commission. Thus, resolution was planned and in progress and the issue was considered nearly-resolved in January 1995. It was later given a HIGH priority ranking in SECY-98-166.[1718] The impact of a license renewal period of 20 years was to be considered in the resolution of the issue.

The staff performed an evaluation of data collected and confirmed that the use of fuel up to the existing limits did not pose safety problems. Confirmatory research with industry cooperation was expected to refine the staff's understanding of issues that may arise from additional increases in burnups. Thus, the issue was RESOLVED with no new requirements.[1778]

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ISSUE 185: CONTROL OF RECRITICALITY FOLLOWING SMALL-BREAK LOCAs IN PWRs

DESCRIPTION

Historical Background

This issue was identified[1730] following an NRR request for reconsideration of the safety priority ranking (DROP) of GSI-22, "Inadvertent Boron Dilution Events," based on new information on high burn-up fuel and new calculations provided by the B&W Owners' Group (B&WOG). Reactivity insertion event tests indicated that high burn-up fuel may be more susceptible to reactivity events than previously expected, and fuel failure may occur at fuel enthalpy values that were previously judged acceptable. In addition, B&WOG calculations predicted prompt criticality with significant heat generation under conditions that may result from small-break (SB) LOCAs. NRR believed that there is no regulatory guidance applicable to this issue.

NRR had previously reviewed studies of deborated water formation during SBLOCAs in PWRs and concluded that: (1) recovery of natural circulation was unlikely to lead to core damage from reactivity transients; and (2) starting or "bumping" of RCPs could lead to a large reactivity transient. However, recent B&WOG calculations predict prompt criticality from natural circulation restart with an accompanying significant heat generation, which raised serious questions about potential reactivity events.

NRR was informed in June 1995 that, if a B&W-designed NSSS spends some time in a boiling/condensing mode following an SBLOCA, a substantial amount of deborated water may accumulate in the RCP suction piping.[1728] Analysis showed that RCP restart would pump the deborated water into the core and might cause a criticality. In July 1995, the scope of the issue was expanded to include: (1) deborated water in the steam generators, cold legs, reactor vessel downcomer, and reactor vessel lower plenum; (2) restart of natural circulation as a mechanism for causing deborated water to flow into the core, and possibly result in criticality; and (3) the potential for prompt criticality.[1728] In late 1996, Framatome Technologies, Inc. (FTI) developed guidance to restrict RCP restart to prevent potential fuel damage.[1728]

In June 1998, the B&WOG prepared a progress report which reiterated that, with conservative assumptions, displacement of deborated water had the potential to cause a prompt-critical condition due to insertion of several dollars of excess reactivity.[1729] In this report the B&WOG concluded that this was an operational issue, not a safety concern, and that potential plant consequences under 10 CFR 50.46 assumptions need not be determined. The June 1998 report was not sufficient to assess the work that had been completed and NRR did not concur with the B&WOG conclusions.

On September 11, 1998, the B&WOG reported new calculation results, provided PRA values to clarify the significance of the safety concern, committed to provide an in-depth investigation to substantiate the September 11, 1998, results, and stated that three utilities had responded to the FTI recommendations regarding RCP restart and two others were in the process of responding.[1728]

Safety Significance

Although the original request from NRR was for reopening Issue 22, "Inadvertent Boron Dilution Events," the scope of Issue 22 covered inadvertent boron dilution events when the reactor was in shutdown or refueling modes, a completely different scenario with different conditions, causes, and potential fixes. Thus, Issue 185 was initiated to address this new scenario.

Some SBLOCAs in PWRs involve steam generation in the core and condensation in the steam generators, causing deborated water to accumulate in part of the RCS. Restart of RCS circulation may cause a deboration event by moving this deborated water into the core. The problem is perceived to be greater in most NSSS designed by B&W than in the W and CE designs because the B&W lowered-loop geometry may favor the accumulation of more deborated water.

Although the B&WOG calculated that the restart of natural circulation following some SBLOCAs may result in prompt criticality with deposition of significant energy in the fuel, similar information has not been provided for operating W- and CE-designed NSSS, although W representatives have written that RCP restart with a large quantity of deborated water must be prevented.

Potential core damage associated with RCP restart was not addressed in the B&WOG PRA and ideally would be included, since operator error may lead to inappropriate RCP restart and there are uncertainties associated with the analysis underlying restart guidance. Consequently, NRR did not concur with the B&WOG conclusion that there is no regulatory concern associated with potential recriticality due to restart of natural circulation. Although this analysis focused on B&W reactors, the generic issue was applicable to all PWRs.

Possible Solution

Because of the potential consequences of an inappropriate RCP start, the B&WOG advised licensees with B&W-designed NSSS to restrict RCP restart following SBLOCAs until the deborated water has been adequately mixed with borated water. This industry voluntary action could be included in regulatory guidance to be issued to all plants.

At the time of the evaluation of this issue, RES was supporting a test program at the University of Maryland thermal-hydraulic test facility that represented the B&W NSSS configuration. Test data had been obtained for restart of RCPs and of natural circulation, but applicability to the issue of deborated water had not been established. (When confronted with a similar problem with the CE System 80+, the planned boron concentration in the refueling water storage tank was increased to ensure non-criticality.)

PRIORITY DETERMINATION

In the request for prioritization of this issue,[1730] NRR stated that "The fuel damage probability indicates that a significant safety problem is unlikely. Further, we judge that a backfit would not be cost-beneficial and would not be justified under 10 CFR 50.109. Nonetheless, modeling uncertainties are high and the potential consequences associated with prompt criticality are of sufficient concern that further assessment may be necessary."

The essence of the issue, as defined by NRR, was the thermal-hydraulic modeling uncertainty and the uncertainty in the potential consequences associated with prompt criticality. This analysis will therefore assess the importance of the thermal-hydraulic phenomena and the consequences of prompt criticality, i.e., the "worst" will be assumed for these two effects, namely that the boron dilution phenomenon will occur and that a prompt criticality will result in significant fuel damage, and the risk importance of the two effects, assuming the worst, will be estimated. These assumptions were appropriate for this analysis. The actual evaluation of the thermal-hydraulic phenomena and the consequences of prompt criticality was reserved for the resolution of the issue.

Frequency Estimate

Description of Sequence (B&W NSSS Design): The event sequence for a B&W design was explored first, since the thermal-hydraulic phenomena were somewhat simpler. (Other PWR designs were examined in a later section.) The plant chosen for analysis was Crystal River Unit 3, a fairly typical 177-fuel assembly lowered-loop design. This plant was chosen primarily because of the ready availability of a RELAP model and considerable design information.

The event of interest begins with an "S2" small LOCA. As reactor coolant escapes, ECCS and AFW start on low pressurizer pressure. (The emergency procedures instruct the operator to trip the RCPs once successful operation of high pressure injection is verified.) The high pressure injection pumps attempt to replace the lost coolant. However, the break size is too large and the primary system pressure too high for the HPI pumps to maintain inventory, and the coolant level in the pressurizer drops. Eventually, the pressurizer empties and steam spaces form at the tops of the hot leg pipes, just above the steam generators, because these locations are the highest points in the system (see Figure 1, taken from NUREG/CR-5640[1759]). When the level drops to the point where there is no longer a liquid pathway to the top of the steam generators, natural circulation ceases and the coolant in the reactor core region heats up and begins to boil, keeping system pressure high. The coolant level continues to drop and the upper portion of the steam generator tubes fill with steam.

The AFW systems in B&W plants spray feedwater into the upper portion of the steam generators. As the primary level drops further, more and more cool steam generator tube surface is exposed to the steam in the primary system, condensing it back into liquid. Eventually, as more and more steam generator tube surface is exposed to the vapor phase, the heat removal from condensation matches the heat generation in the core.

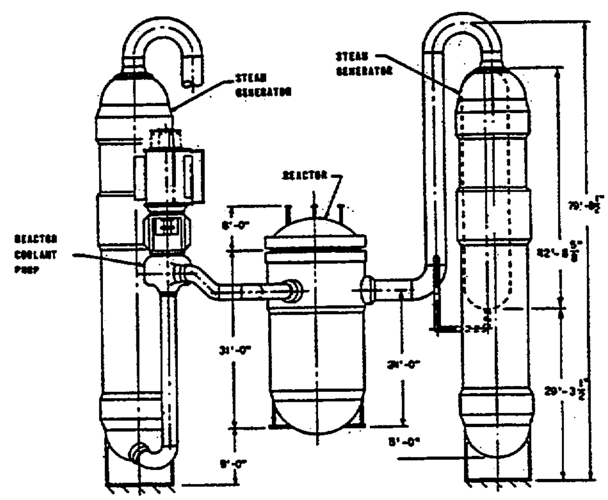

Figure 1: B&W NSSS

An equilibrium condition would be achieved, with the coolant boiling in the core and condensing in the steam generators, if it were not for the continued loss of coolant through the "S2" break. As level drops further, and still more cool steam generator tube surface is exposed to the vapor phase, primary pressure drops. (The heat generation rate in the core is also slowly decreasing due to radioactive decay, which contributes to the pressure drop.) As the pressure decreases, the flow rate from the high pressure coolant injection trains increases, and eventually the injection rate will equal the loss through the break.

This scenario is actually a successful operation of the ECCS which would avoid severe core damage. However, this method of core cooling, which boils coolant in the core, condenses coolant in the steam generator, and returns coolant to the core through the cold leg, also removes the soluble boron from the coolant via distillation. The condensed coolant in the steam generator lower plena and cold leg piping will have a nearly zero boron concentration, while the boron concentration in the reactor vessel core volume will increase. (There will be some injection of borated coolant at the RCP seals, but the coolant return flow will carry this boron into the reactor vessel.)

The deborated coolant region will not be troublesome as long as the system remains in the "reflux boiling" state, since deborated coolant entering the reactor will mix with the more concentrated boron solution in the core region. However, if the system is refilled to the point where liquid natural circulation restarts, or if the RCPs are started, the deborated, relatively cool coolant which has accumulated in the cold legs and steam generators will be swept into the reactor core. In a typical 177-fuel assembly B&W NSSS (including Crystal River), the tube side free water volume of each steam generator is 2030 cubic feet,[1759] while the water volume of the reactor vessel is 3910 cubic feet (from the Crystal River RELAP model). Thus, the two steam generators would contain a water volume slightly larger than that of the reactor vessel. It appeared plausible that, should natural circulation be reestablished, the deborated coolant could momentarily flush the borated coolant out of the core with relatively little mixing. As was stated above, it was assumed that this happens, consistent with the "worst-case" assumption. It should be noted that there was considerable uncertainty as to the reality of this phenomenon.

After shutdown, decay heat will drop rapidly to about 2% of rated thermal power and continue to decrease. At this power level, a simple hand calculation shows that, if natural circulation is lost, the core will boil enough coolant to fill the steam generators with condensed coolant in about 25 minutes. Thus, the scenario is credible. Since there is return flow of condensed coolant from the steam generators to the reactor through the cold legs, it is unlikely that any dissolved boric acid will diffuse back into the steam generator volumes. However, it is possible that deborated coolant will gradually fill the reactor vessel downcomer and lower plenum with soluble boron concentrating (and possibly precipitating) in the core region. How much mixing will occur in the lower plenum and downcomer is a source of uncertainty that will ultimately need to be resolved but, for this analysis, it was assumed that the deborated volume in the steam generators will be sufficient to (at least momentarily) flood the core region.

If the accident should occur early in the fuel cycle, there may be sufficient excess reactivity in the core for the deborated coolant to bring the core to criticality even though all the control rods have been inserted. The possible power excursion may be sufficient to cause severe damage to the core, even though the ECCS has successfully kept the core covered with coolant. It is this power excursion that formed the basis for this issue.

Event Tree: An event tree was constructed to quantify this scenario (see Figure 2).

Small Break LOCA: The initiating event for this scenario is a LOCA of the proper size - large enough for the high pressure injection to not keep up with coolant loss at full primary system pressure, but small enough to not depressurize the system. This is an "S2" break as defined in NUREG-1150,[1081] a break of ½ to 2 inches equivalent diameter, corresponding to a fluid loss rate of approximately 100 to 1500 gpm. The frequency of such breaks in NUREG-1150[1081] was 10^{-3}/RY.

Number of HPI trains: Once the break occurs, high pressure injection will initiate. This particular plant has three HPI trains, two of which will start automatically, and one of which is kept "in reserve," and may be manually initiated by the operator. For this analysis, which was intended to

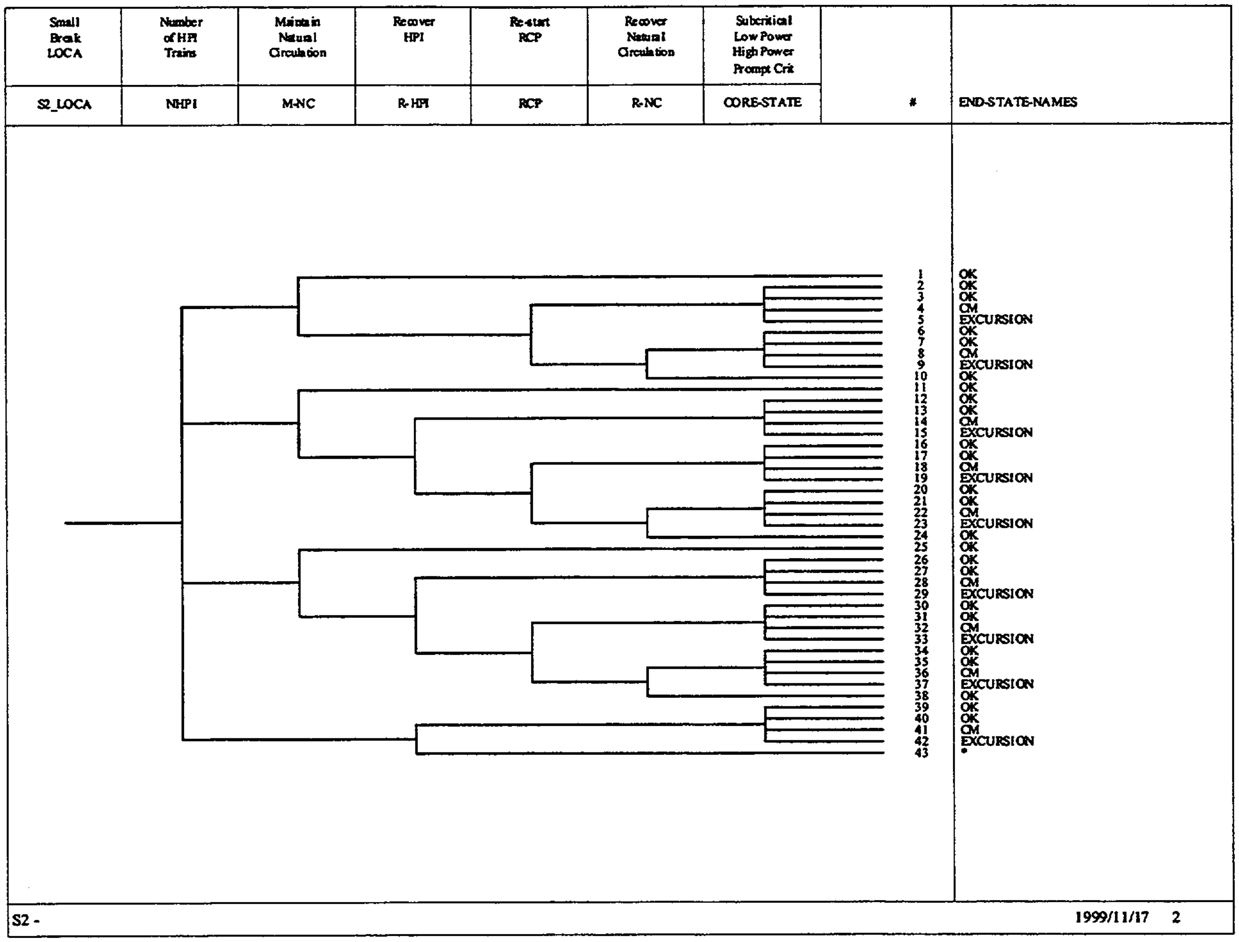

Figure 2: Event Tree

be more generic, it was assumed that all three trains will be started shortly after the onset of coolant loss. Thus, four outcomes were possible corresponding to zero, one, two, or three trains operating. A full calculation of the probabilities of these four system states was beyond the scope of this analysis. Instead, it was assumed that the likelihood of a single train failure would be dominated by the unavailability of the pump (3.8×10^{-3} in the Crystal River SPAR-2QA model). The SPAR-2QA model was presented at the 1998 Probabilistic Safety Assessment and Management (PSAM IV) Conference in New York by S. M. Long, P. D. O'Reilly, E. G. Rodrick, and M. B. Sattison in their paper on the "Current Status of the SAPHIRE Models for ASP Evaluations." For the failure probability of the entire system, the SPAR-2QA figure for the entire system was used (1.019×10^{-4}). If the unavailability of one pump is "p," the four probabilities, using the rare event approximation, are as follows:

$P(0) = 1.019 \times 10^{-4}$ (the SPAR-2QA number for the entire system[1761])

$P(1) = 3(1-p)p^2 = 4.32 \times 10^{-5}$

$P(2) = 3(1-p)^2p = 1.113 \times 10^{-2}$

$P(3) = 1 - [P(0) + P(1) + P(2)] = 0.9887$

Two caveats should be noted. First, the number of significant figures was used for the convenience of forming differences between numbers and for the reader who wishes to reproduce the calculation, and not because the unavailabilities were known to such high accuracy; appropriate rounding will be performed at the end of the calculation. Second, the approximation used assumed that all common cause failures will fail all three trains, and also that failure other than pump failures will fail all three trains. For this reason, P(0), the probability of no trains operating, was higher than P(1).

It was assumed that the operator will shut down the RCPs with a probability of unity. This is a standard "no miracles" assumption in all PRA calculations - a failure to follow procedures is never credited as a positive outcome.

Maintain Natural Circulation: If the flow out the break is less than or equal to the injection flow from the HPI trains, the coolant level will not drop out of the pressurizer, and natural circulation will be maintained. If the HPI trains cannot keep up with the break flow, the level will drop and natural circulation will be lost. (Eventually, pressure will drop to the saturation pressure for the existing coolant temperature, and HPI flow will increase as pressure drops.)

The likelihood of a particular break size would decrease as the equivalent diameter increases, which is why large break "A" LOCAs are less likely than small break "S1" LOCAs, which in turn are less likely than very small break "S2" LOCAs. However, for this analysis, it was assumed that the likelihood of a particular break size will be constant over the S2 size interval, which was assumed to be equivalent to the "G3" coolant loss rate assessed in NUREG/CR-5750.[1760] Comparing these coolant loss rates with the capability of the HPI pumps:

Number of Pumps	Flow at 1600 psi[1759] (gpm)	Flow at 2255 psi[1759] (gpm)	Fraction of 100-1500 gpm "G3" Spectrum Covered	Probability of Loss of Natural Circulation
1	400	270	21.4%	79%
2	800	540	50%	50%
3	1200	810	78.6%	21%

Thus, the likelihood of loss of natural circulation would depend on the number of HPI trains running. If all three trains of HPI fail, the probability of loss of natural circulation is unity.

Recover HPI: There is some likelihood that the operator will be able to recover a train of HPI. To estimate this probability, the operator's probability of recovery for the "SLOCA" sequences in the Crystal River SPAR-2QA model were used. This parameter, designated "SLOCA-XHE-NOREC" was 43% of non-recovery, implying a recovery probability of 57%.

Restart RCPs: For the usual small-break LOCA sequences, procedures call for the operator to trip the RCPs once it is verified that a train of HPI is operating. (The RCPs add a significant amount of energy to the primary system.) However, if the operator discovers that natural circulation has been lost and coolant is boiling in the core, the operator may elect to restart an RCP to ensure that the upper portion of the core does not rise above the liquid/vapor interface but instead is cooled by two-phase flow. There was essentially no precedent for this situation and, based purely on judgment, a probability of 10% was used for this parameter.

Recover Natural Circulation: The operator may be able to recover natural circulation, possibly by using the charging pumps (for which no credit has been given up to this point - the Crystal River plant does not have separate charging pumps, but other plants may be so equipped), by isolating the break (which might be a stuck-open valve for a LOCA in this size range), by manually starting a reserve train of HPI (in plants so equipped, such as Crystal River), or by blowing down the secondary side of a steam generator, thereby reducing the temperature and pressure in the primary, reducing flow out the break in the system, and permitting more injection flow from the HPI trains. Eventually, as decay heat slowly drops, the coolant level will rise. Again, there was no available estimate for this situation. Based on judgment, 50% was used for this parameter.

Core State: PWR cores must be designed with sufficient excess reactivity to be able to remain at power throughout the fuel cycle. At the end of the cycle, there is no soluble boron in the coolant. Conversely, a high boron concentration is present at the beginning of the cycle to compensate for the excess reactivity designed into the core. The longer the cycle, the more excess reactivity must be designed into the core, and the higher the beginning-of-cycle boron concentration. However, there is a limit to how high a boron concentration can be used, since the presence of soluble boron causes the moderator temperature coefficient (MTC) to be less negative. At the beginning of the cycle, the MTC is usually close to zero. The core designer may (and usually does) use burnable poison to further extend the cycle. The burnable poison holds reactivity "down" at the beginning of the cycle without causing the MTC to become excessively positive.

Boron concentration thus drops during the course of the cycle, very rapidly at first as xenon and samarium build up to equilibrium levels. Boron concentration as a function of burnup (commonly called "boron letdown curves") for the reactor under study is shown in Figure 3 (from the Crystal River updated FSAR). (It should be noted that the full equilibrium cycle for this plant is 310 effective full power days, even though the curve reaches zero boron concentration slightly before 300 days. It is at this point that the transient rod bank is moved out of the core, which extends core life by approximately 30 days.)

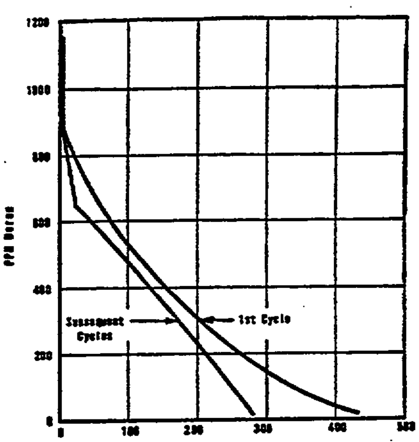

Figure 3: Boron Letdown

The significance for this analysis is that, at the beginning of the cycle, the reactivity worth of the soluble boron is greater than the worth of the control rods, Thus, if the soluble boron is swept out of the core and replaced with deborated coolant, the control rods do not have sufficient worth to keep the core in a subcritical state.

The boron letdown and reactivity characteristics can vary considerably from plant to plant or even from cycle to cycle, since the core designer may be aiming for a longer cycle, a flatter power distribution, maximum burnup on older fuel assemblies, or any number of other factors. Thus, although this calculation must of necessity be based on one set of core parameters, these numbers must not be taken as being universally applicable to all plants and all cycles.

This particular cycle (the equilibrium cycle described in the Crystal River updated FSAR) has a soluble boron worth of 0.01 %Δk/k per ppm of boron, a total rod worth of 7% (not including a stuck rod allowance of 1.6 %), and moderator and Doppler deficits of 0.2% and 1.7%, respectively. The excess reactivity was estimated and is shown in Figure 4.

As can be seen from Figure 4, there is an interval of approximately 24 days at the beginning of the cycle during which the control rod worth is insufficient to render the core subcritical. The probability of occurrence of such a criticality is just the number of days where this is possible (24) divided by the total number of days in the cycle (310), giving a probability of approximately 7.7%.

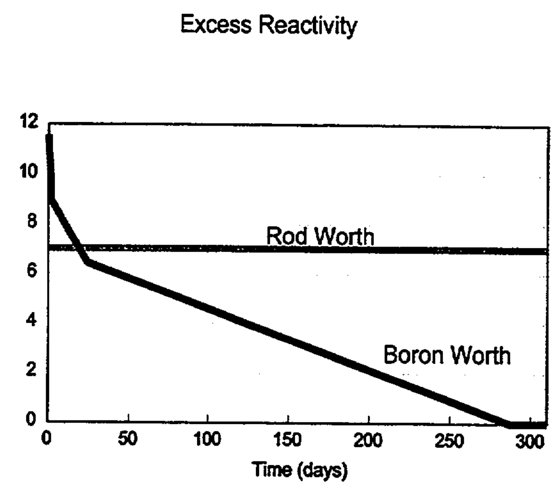

Figure 4: Excess Reactivity vs. Time

However, criticality does not automatically equate to severe core damage. In this scenario, AFW is operating, and both steam generators are capable of removing heat from the primary system. This plant is equipped with two AFW pumps, each capable of supplying 740 gpm of feedwater,[1761] which would accommodate approximately 7% of the reactor's rated thermal power. With both AFW pumps operating, and subtracting 2% for the decay heat being produced in the reactor core, the steam generators should be able to accommodate fission heat up to approximately 12% of rated power. However, the fission heat will not be continuous, but will "chug" as the deborated coolant sweeps in and out of the core. Therefore, it was assumed that the steam generators can accommodate power pulses of up to double the continuous power, or approximately 25% of rated thermal power. Any power pulse above 25% was assumed to result in core damage.

If the net reactivity is greater than approximately 0.5% Δk/k, the core will be in a state of prompt criticality and will experience a power excursion. This was also assumed to result in severe core damage consistent with the "worst-case" assumption discussed previously.

If the deborated coolant fills the core area relatively slowly, as would be expected in the case of a refill of the system and a restart of natural circulation, there will be time for the moderator temperature coefficient to limit core power. The situation is different if the RCPs are restarted. The design forced coolant flow rate (131.3 x 10^6 lb/hr) corresponds to a core transit time of approximately 0.6 seconds. All four coolant pumps will not be switched on simultaneously, so the deborated coolant may take two or three seconds to flood the core. This is still significantly less than the thermal time constant of the fuel rods (roughly 6 seconds for most designs), and there will be little negative feedback provided by the moderator temperature coefficient. Moreover, there is a fairly strong tendency for the incremental axial reactivity worth to concentrate near the top in any core with significant burnup, which will accelerate the incremental reactivity insertion rate. Therefore, only Doppler feedback was assumed for event sequences involving restart of the RCPs. (The moderator temperature coefficient is only slightly negative at the beginning of the cycle, and thus the two situations are not vastly different.)

There is also a timing window effect due to the xenon transient, as is shown in Figure 5 (from the NRC training manual for PWR plants). If the core is operating at full power and has achieved an equilibrium xenon concentration, the xenon concentration will increase and insert still more negative reactivity after the reactor shuts down. For a shutdown from full power, the negative reactivity peaks about eight hours after shutdown, returns to the equilibrium value after approximately one day, and then continues to decrease, which implies that still more shutdown reactivity is needed to keep the core in a subcritical condition. It was assumed that the operators will have the plant stabilized by the time a full day has gone by, and thus the effects of the xenon "tail" were not considered here.

It should be noted that, for the first few hours after reactor trip, if natural circulation or pump restart occurs later in time, the likelihood of a recriticality is less, because of the xenon transient. The excess reactivity at the very beginning of the cycle is sufficient to overcome the xenon overshoot even at its peak, but the xenon effect might prevent a criticality if the boron dilution event occurred after an hour or so and if the event occurred a little later in the fuel cycle.

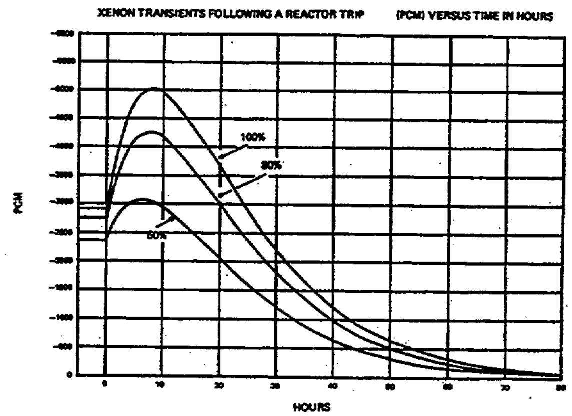

Figure 5: Typical Xenon Transients

The boron curve was digitized and the excess reactivity compared with the various deficits. Of the 310 days in the fuel cycle, criticality is possible with all rods in for approximately the first 20 days. The probabilities of the various branches were as follows:

	Probability of Prompt Criticality	Probability of Overpower	Probability of Criticality, Low Power	Probability of No Criticality
Slow reactivity insertion	2/310 (0.6%)	13/310 (4.2%)	5/310 (1.6%)	290/310 (93.6%)
Fast reactivity insertion	4/310 (1.3%)	11/310 (3.5%)	5/310 (1.6%)	290/310 (93.6%)

In summary, after the first four days of the fuel cycle, a reactivity excursion is no longer possible and, after 15 days, significant core damage is no longer possible. These figures can vary somewhat from plant to plant and cycle to cycle, however.

Results: The results of the event tree calculation for this B&W design were a CDF of 5.7×10^{-6} event/RY, of which 9×10^{-7} event/RY involved a reactivity excursion.

The highest frequency scenario corresponded to Sequences 8 and 9 on the event tree. The scenario is initiated by a small-break LOCA, all three HPI trains operate, but flow is not sufficient to maintain natural circulation. The RCPs are not restarted, but natural circulation re-starts after the steam generators fill with deborated coolant. The frequency of a reactivity excursion is 2×10^{-7} /RY and the frequency of severe core damage is an additional 4×10^{-6}/RY.

The second highest frequency scenario, corresponding to Sequences 4 and 5, is similar, but instead of recovering natural circulation, the RCPs are restarted. The total frequency is 10^{-6}/RY, which includes a frequency of excursion of 3×10^{-7}/RY.

The third highest frequency scenario, Sequences 14 and 15, starts with a small-break LOCA, but one train of HPI fails. Natural circulation is lost, the steam generators fill with deborated coolant, and then the inoperable HPI train is recovered. The frequency of this scenario is 10^{-7}/RY which includes a frequency of excursion of 2×10^{-8}/RY.

Description of Sequence (W design): The W design differs significantly from the B&W design and the thermal-hydraulic effects can be affected. The design is shown in Figures 6 and 7 of NUREG/CR-5640.[1759]

First, the steam generators are of the U-tube design and these tubes are completely submerged in liquid water on the secondary side. After a small LOCA, as coolant is lost out of the break, the pressurizer will empty, pressure will drop, and voids will form in the core area.

Unlike the situation in the B&W design where the voids will naturally collect and form a vapor space at the top of the hot leg, voids will be carried into the ascending half of the U-tubes and condense back into the liquid phase. As pressure and coolant inventory continue to drop, a greater fraction of the volume above the core and in the hot legs will be in the vapor phase. It is likely that re-condensed (and deborated) coolant will first flow back down the ascending half of the U-tubes and run down on the lower surfaces of the pipes back down to the upper plenum of the reactor, where it will mix rapidly with the more concentrated, turbulently boiling coolant just above the core. As more inventory is lost, eventually a state will be reached where the primary system is at saturation pressure, coolant in the vapor phase condenses in the steam generators, and at least some of the condensed, deborated coolant collects in the descending half of the U-tubes, and the outlet plena, cold legs, pump volume, and, eventually, the lower plenum of the reactor vessel.

Second, unlike the B&W "lowered loop" design, the steam generators are located at a higher elevation than the top of the reactor core. In this design, as the coolant level in the primary system drops, it will be more difficult for deborated coolant to remain in the steam generators. In contrast to this, in the B&W lowered loop design, the coolant level can drop to the top of the active core, and there will still be some deborated coolant in the steam generators.

Third, the available volume in the steam generators is somewhat less. The total volume of coolant in the reactor vessel is 4333 cubic feet (from the RELAP model for this plant), while the primary side of a "Model F" steam generator is 962 cubic feet.[1759] The total primary volume of the four steam generators is thus about 90% of the reactor volume. However, because of the U-tube design of the steam generators, it was not clear that the entire primary volume of the steam generators will fill with deborated coolant. If only the descending portion of the tubes are filled, the total liquid inventory in the steam generators will be only 45% of the reactor volume. It was not clear that, should natural circulation be restored, the core area will be flooded temporarily with deborated coolant. Conversely, the reactor downcomer and lower plenum volumes may slowly fill with unmixed, deborated coolant, as was discussed earlier, and this would be a sufficient volume to sweep the dissolved boron out of the core region. Thus, for this design, there was even more uncertainty regarding the credibility of this scenario than in the B&W example discussed previously. However, some experimental work at a test facility at the University of Maryland strongly suggested that the deborated coolant will sweep through the primary system as a "slug" with relatively little mixing. Again, assuming the "worst case" scenario, it was assumed that the accumulation of deborated coolant will occur.

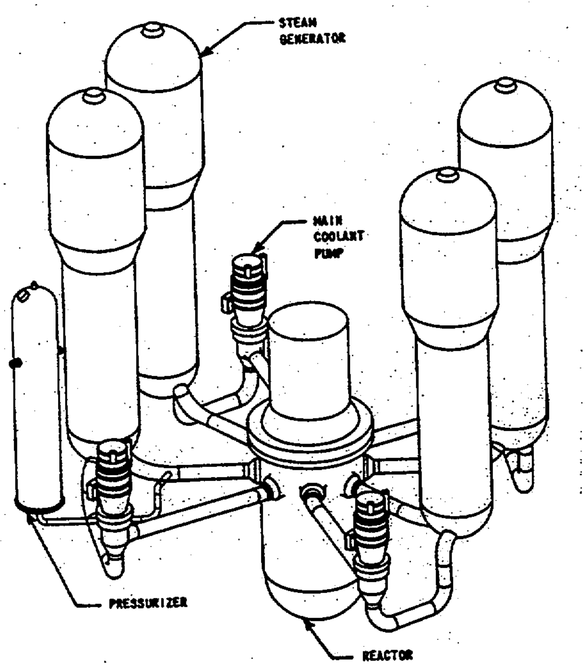

Figure 6: Westinghouse NSSS

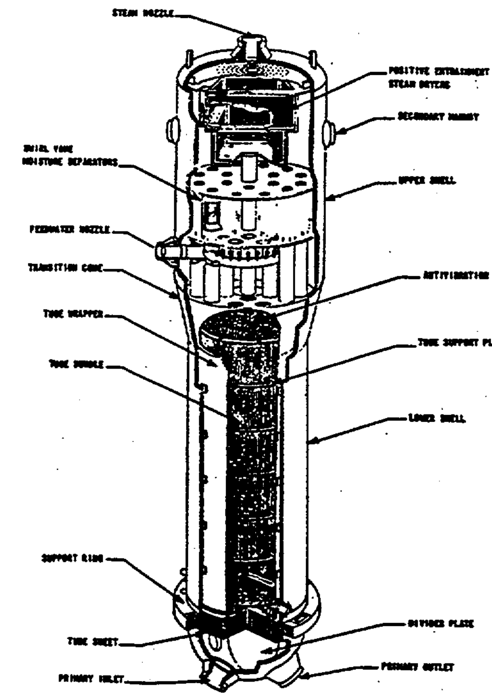

Figure 7: U-Tube Steam Generator

Event Tree: The event tree structure is essentially unchanged, but the values of certain split fractions must be changed because of the differences in the various systems. The Seabrook plant was chosen for analysis, again because of the ready availability of design information and the existence of a RELAP model.

Small Break LOCA: As before, the NUREG-1150[1081] S2 frequency of 10^{-3}/RY was used.

Probability of Maintaining Natural Circulation: Seabrook is equipped with three charging pumps, two of which are centrifugal, and one of which is a positive displacement pump.[1759] In addition, the plant is equipped with a two-train high-pressure safety injection (HPSI) system. The two HPSI pumps are centrifugal pumps, but have a shutoff head close to the saturation pressure of the primary system; they cannot inject at operating pressure. Pump capacities are given in the following table:

Pump Type	Flow at 1750 psi[1759]	Flow at PORV Setpoint[1759]
Charging, Centrifugal (2)	(unknown)	150 gpm (each)
Charging, Positive Displacement	98 gpm	98 gpm
HPSI, Centrifugal (2)	425 gpm (each)	zero

The positive displacement pump was neglected because of its low capacity. The flow near saturation pressure for the two centrifugal charging pumps was not given in NUREG/CR-5640.[1759] However, the SPAR-2QA model event tree for small-break LOCA has, as success criteria, either of the two HPSI pumps, or both of the two centrifugal charging pumps. Thus, the two charging pumps were treated together as if they were a third HPSI train with a combined flow of 425 gpm. Split fractions were calculated using the same assumptions as before and the results were as follows:

Number of Pumps	Flow at 1750 psi	Fraction of 100 to 1500 gpm "G3" Spectrum Covered	Probability of Loss of Natural Circulation
1	425 gpm	23.2%	76%
2	850 gpm	53.6%	46%
3	1275 gpm	83.9%	16%

Number of HPSI "Trains:" The SPAR-2QA model's HPSI fault tree for this plant was much more tractable than that of the B&W plant. From the SPAR-2QA model for this plant, calculations of the three total system and the individual trains gave the following results:

Probability of Failure of:	Parameters in SPAR-2QA Model[1761]M	Value
Entire HPSI System, including Charging Pumps	HPI	1.096E-5
Two Centrifugal Charging Pump Trains	CHV-SYS-F	8.77E-3
Both HPSI Trains (including Common Cause Failures)	HPI-TRAINS-F	1.624E-5
One HPSI Train	HPI-TRAINA-F or HPI-TRAINB-F	4.030E-3

Again, the numbers above did not have four significant figure accuracy. The extra digits were given for the convenience of the reader who wishes to repeat the calculation. The probability of a certain number of trains operating, P(n), was then calculated as follows:

Probability of n Trains Operating	Parameters in SPAR-2QA Model[1761]		Value
P(0)	HPI		1.096E-5
P(1)	(HPI-TRAINS-F)(1-CHV-SYS-F) + [(HPI-TRAINA-F)(CHV-SYS-F)](1-HPI-TRAINB-F) + [(HPI-TRAINB-F)(CHV-SYS-F)](1-HPI-TRAINA-F)	1.61E-5 + 3.52E-5 + 3.52E-5	8.65E-5
P(2)	HPI-TRAINA-F + HPI-TRAINB-F + CHV-SYS-F	4.03E-3 + 4.03E-3 + 8.77E-3	1.683E-2
P(3)	1 - P(0) - P(1) - P(2)		0.983

Recover HPSI: Using the Seabrook SPAR-2QA model, the parameter designated "SLOCA-XHE-NOREC" indicated a 43% probability of non-recovery which implied a recovery probability of 57%.

Restart RCPs: As in the B&W case, a probability of 10% was used, based purely on judgment.

Recover Natural Circulation: As in the B&W case, the operator may be able to recover natural circulation by isolating the break, using the positive displacement charging pump, or blowing down a steam generator. Based on judgment, 50% was again used for this parameter.

Core State: The boron letdown curve for the Seabrook core (fairly typical of a W "low leakage" design, and plotted versus burnup in megawatt-days per metric ton of uranium instead of days in the cycle) is shown in Figure 8 (from the Seabrook updated FSAR). As can be seen by comparing this curve with the B&W curve shown earlier, there are some marked differences. First, it should be noted that the licensee did not include the xenon and samarium build-in at the very beginning of the cycle, and thus the curve does not begin at zero burnup. Second, the full power boron concentration actually increases slightly at the beginning of the cycle, then decreases slowly, eventually becoming linear for the latter portion of the cycle until it becomes zero at the end of the cycle (17 GWD/MTU). This is due to the burnable poison loading, which is typically higher in W cores.

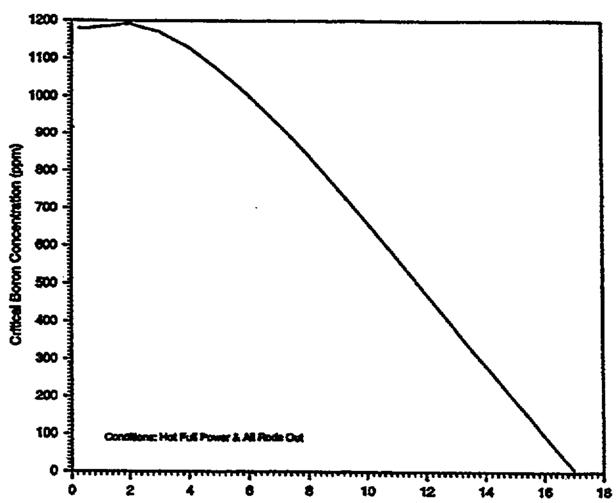

Figure 8: Westinghouse Boron Letdown

This curve was digitized and combined with other information in the Seabrook FSAR to produce a plot of boron worth and control rod worth over the cycle (with the xenon buildup added at the beginning of the cycle. For this core design, it is possible to achieve criticality for about 36% of the cycle, almost five times the 7.7% figure for the B&W core.

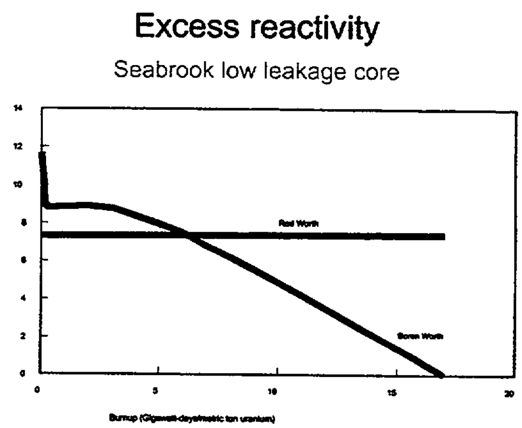

Figure 9: Excess Reactivity vs. Burnup

As before, criticality does not automatically equate to severe core damage. The Seabrook plant is equipped with two AFW trains, one motor-driven and one turbine-driven, each capable of supplying 710 gpm at a secondary side pressure of 1322 psi.[1759] This is somewhat less than the capacity of the Crystal River plant's AFW, and the rated thermal power of the Seabrook reactor core is actually greater than that of Crystal River. A rough calculation similar to the one done for the B&W design indicates that the AFW supply is capable of removing about 4.8% of rated thermal power per AFW train. If both trains are operating, allowing 2% of rated power for decay heat removal, and assuming the fission heat pulses with a 50% duty cycle, the AFW system can accommodate fission power of about 15% of rated - significantly less than that of the B&W design. However, unlike the B&W design, the W steam generators are likely to contain a significant inventory of secondary coolant, completely submerging the tubes on the secondary side, and are far less likely to dry out before the power pulses in the primary side die out due to boron mixing in the primary. There was no easy way to estimate this effect quantitatively. However, the probability of damage was not a very strong function of the power level assumed to be the threshold of severe fuel damage. Using the digitized curves, the following estimates were made:

Fuel Damage Assumption	Percentage of Fuel Cycle
Fuel melts at criticality	36%
Fuel melts at AFW limit (15% power)	33%
Fuel melts at 50% power	25%
Fuel melts at 100% power	15%

It was difficult to believe that a 100% power pulse would not result in damage. It was even more difficult to believe that a subcritical core would sustain any damage. The extreme range in damage threshold only leads to a range of 15% to 36% in the probability of severe core damage, given a boron dilution event. It was assumed, based purely on judgment, that severe core damage will result at 50% of rated power.

Regarding prompt criticality, a calculation indicated this to be possible only during the time of xenon buildup - about 1% of the fuel cycle. Once equilibrium is achieved, the burnable poison loading is such that the excess reactivity curve is relatively flat and does not rise sufficiently above the shutdown rod worth to permit a prompt criticality event. The digitized boron curve was used to calculate the probabilities of the various branches:

Sequence	Probability of Prompt Criticality	Probability of Overpower	Probability of Criticality, Low Power	Probability of No Criticality
Slow reactivity insertion	1%	24%	11%	64%
Fast reactivity insertion	1%	24%	11%	64%

Results: The results of the event tree calculation for this W design were a CDF of 2.2×10^{-5} event/RY, of which 10^{-6} event/RY involved a reactivity excursion.

As in the B&W case, the highest frequency scenario corresponded to Sequences 8 and 9 on the event tree. This scenario is initiated by a small break LOCA, all HPSI trains operate, but flow is not sufficient to maintain natural circulation. The RCPs are not restarted, but natural circulation restarts after the steam generators fill with deborated coolant. The frequency of a reactivity excursion was 7×10^{-7}/RY and the frequency of severe core damage was an additional 2×10^{-5}/RY.

The second highest frequency scenario, which corresponds to Sequences 4 and 5, is similar but instead of recovering natural circulation, the RCPs are restarted. The total frequency was 4×10^{-6} /RY which includes a frequency of excursion of 2×10^{-7}/RY.

The third highest frequency scenario, corresponding to Sequences 14 and 15, starts with a small-break LOCA but one train of HPSI fails. Natural circulation is lost, the steam generators fill with deborated coolant and then the inoperable HPSI train is recovered. The frequency of this scenario was 10^{-6}/RY, which included a frequency of excursion of 4×10^{-8}/RY.

Discussion: The CDF results were quite similar for both designs. This was not too surprising as the same event tree was used for both, and many of the split fractions were the same. Results for 2-loop or 3-loop W designs, or a CE design, were not likely to be greatly different. The W CDFs were about a factor of four higher than that estimated for the B&W design. This appeared to be primarily due to the higher burnable poison loading in the W core which causes the core to have a potential for criticality for almost five times as long a fraction of the fuel cycle. There was, however, somewhat less uncertainty in the thermal-hydraulic effects in the B&W design.

The nature of the highest frequency scenarios suggest that a procedural fix may be appropriate for this issue. All three scenarios involve natural circulation restarting due to actions taken by the operators, restarting the RCPs, or recovering a train of high pressure injection.

Consequence Estimate

To estimate consequences and risk, the standard analysis described in the Introduction to NUREG-0933 was used, i.e, the WASH-1400[16] Release Categories and a generic site. For the portion of the CDF associated with overpower damage to the fuel, the spectrum of consequences across the seven PWR Release Categories for the S2 LOCA in WASH-1400[16] was re-normalized to this issue's CDF. For the reactivity excursions, the entire event frequency was put into the PWR-1 release category, consistent with the worst case assumption discussed earlier. The results are shown in Table 3.185-1 below.

Table 3.185-1

Release Category	1	2	3	4	5	6	7	Total
WASH-1400 Spectrum of Release Categories[16]								
WASH-1400 S2 Frequencies	1.0e-07	3.0e-07	3.0e-06	3.0e-07	3.0e-07	2.0e-06	2.0e-05	2.6e-05
WASH-1400 Normalized Frequencies	0.38%	1.15%	11.54%	1.15%	1.15%	7.69%	76.92%	100.00%
Westinghouse Design								
Frequencies, Overpower Sequences	8.1e-08	2.4e-07	2.4e-06	2.4e-07	2.4e-07	1.6e-06	1.6e-05	2.1e-05
Excursion Event Frequency	1.0e-06							1.0e-06
Sum	1.1e-06	2.4e-07	2.4e-06	2.4e-07	2.4e-07	1.6e-06	1.6e-05	2.2e-05
Release Category Consequences (man-rem)	5.4e+06	4.8e+06	5.4e+06	2.7e+06	1.0e+06	1.5e+05	2.3e+03	
Risk (man-rem/RY)	5.8e+00	1.2e+00	1.3e+01	6.5e-01	2.4e-01	2.4e-01	3.7e-02	2.1e+01
B&W Design								
Frequencies, Overpower Sequences	1.8e-08	5.5e-08	5.5e-07	5.5e-08	5.5e-08	3.7e-07	3.7e-06	4.8e-06
Excursion Event Frequency	9.0e-07							9.0e-07
Sum	9.2e-07	5.5e-08	5.5e-07	5.5e-08	5.5e-08	3.7e-07	3.7e-06	5.7e-06
Release Category Consequences (man-rem)	5.4e+06	4.8e+06	5.4e+06	2.7e+06	1.0e+06	1.5e+05	2.3e+03	
Risk (man-rem/RY)	5.0e+00	2.7e-01	3.0e+00	1.5e-01	5.5e-02	5.5e-02	8.5e-03	8.5e+00

The net risk associated with this issue was thus estimated to be 8.5 man-rem/RY for the B&W design, and 21 man-rem/RY for the W and CE designs. In January 2000, the net benefit of this issue was estimated as follows:

Reactor Design	Number of Plants	Remaining Aggregate Life (RY)	Man-rem/RY	Risk benefit (man-rem)
B&W	10	190	8.5	1,615
Westinghouse	54	1100	21	23,100
CE	15	300	21	6,300
			Total:	31,015

The total risk benefit was estimated to be 31,000 man-rem, excluding the effect of license renewal which would increase the number significantly.

Cost Estimate

Industry Cost: The cost to a licensee would be the cost of writing and putting in place a complex change in emergency procedures. According to Table 4.1 of NUREG/CR-4627,[961] such a change would cost \$3,420 to \$4,350, with a point estimate of \$3,900. This complex procedure may well be an above-average cost and, therefore, the upper limit of \$4,350 was used. For approximately 80 PWRs, the total licensee cost was \$348,000.

NRC Cost: The cost to the NRC would be significant, since considerable work would need to be done to resolve the thermal-hydraulic uncertainties, plus all of the administrative effort involved in any type of regulatory action. Based purely on judgment, a cost of \$2M was assumed.

Total Cost: The total industry and NRC cost for the possible solution was estimated to be approximately \$2.4M and was dominated by the cost of confirmatory thermal-hydraulic research.

Impact/Value Assessment

Based on a potential public risk reduction of 31,000 man-rem and cost of \$2.4M for a possible solution, the impact/value score was estimated to be \$80/man-rem.

Other Considerations

(1) Because the contemplated fix would be procedural in nature, there were no implications for increased ORE to plant workers.

(2) Because the issue was well into the cost-beneficial range, avoided offsite costs of a potential accident were not estimated; inclusion of these costs would not change the conclusion.

(3) License Renewal: Assuming a license renewal period for 79 plants, the public risk reduction would be approximately doubled, to 60,000 man-rem.

Uncertainties

The calculations presented above were point estimates only. The Rev. 2 QA SPAR models from which many of the parameters were taken did not include uncertainty distributions. Moreover, some of the parameters were based only on judgment. Thus, a standard PRA uncertainty analysis was not feasible. Nevertheless, there were several limitations in the analysis:

- The estimates of the fraction of the fuel cycle during which the core can be brought to a critical state with all control rods inserted were based on calculations performed on FSAR data. These calculations were very primitive, core nuclear design parameters may differ for each fuel cycle, and the two estimates of this fraction, 7.7% for the B&W core and 36% for the W core, can vary. However, it is doubtful that these fractions will vary by orders of magnitude, which would be necessary to change the conclusion.

- The xenon reactivity transient was included only as a window effect. In reality, the xenon transient will become steadily more important as core burnup increases, and the "window" of time after shutdown during which it is possible to achieve criticality will steadily decrease.

- Conversely, the fact that the xenon will eventually decay away has not been included. The assumption was made that, by the time the xenon transient turned around, the operators would have taken appropriate corrective action. This "delayed criticality" effect is, in reality, still another accident scenario which should be incorporated into the resolution of this issue.

- The options available to the operator to refill the primary system (and thereby recover natural circulation) are plant-specific. In the particular case of Crystal River, it was assumed that all three HPI trains will be started to mitigate the loss of coolant. However, only two trains start automatically on an SI signal. If the operator manually starts the third train at the beginning of the accident sequence, this will be a good approximation. However, if the operator delays starting the manual train, and then starts the third train after observing that the automatically-initiated trains have either failed or are not sufficient to maintain primary coolant inventory, this late start will actually increase the likelihood of a return to criticality.

- The core power level associated with the onset of severe fuel damage was, at best, an educated guess. If there is any high burnup fuel in the core, severe damage might occur as a result of even a relatively mild reactivity excursion. Conversely, the steam generators are sized to accommodate full power operation and should be able to remove the integrated energy of a significant power pulse, limited primarily by the capacity of the AFW system and the capacity of the secondary side safety valves and ADVs.

- The actions of the operators were worthy of much more study, given the time windows involved in these scenarios and the lack of information on core reactivity. The plant operators would be faced with some confusing decisions about whether to restore failed trains, initiate forced circulation, etc.

- The thermal-hydraulic phenomena needed further investigation. Although the estimate for this study was $2M (roughly 10 staff-years), the investigation would be cost-effective even if this expense were much higher.

It should also be noted that, in its evaluation of the B&WOG PRA, NRR believed that the deborated water accumulation modeling, transport modeling, and reactivity analyses were highly approximate, incompletely understood, and subject to large uncertainties. Although the staff recognized these shortcomings, it expanded the B&WOG PRA to include approximations of additional variables and concluded that the fuel damage probability for natural circulation restart was probably between approximately 10^{-7}/RY and 10^{-5}/RY.[1730] This was completely independent of the analysis presented here, but nevertheless yielded similar results.

CONCLUSION

The CDF change associated with the issue was estimated to be 2.2×10^{-5} event/RY and the cost/benefit ratio was approximately $80/man-rem for W and CE plants. This class of PWRs dominated primarily because of a higher burnable poison loading and, consequently, a longer fraction of the fuel cycle in which recriticality is possible. The cost/benefit ratio was particularly favorable because the cost was low and was likely to be dominated by NRC research costs. Based on the cost/benefit criteria (shown in Figure 1 of the Introduction to NUREG-0933), the issue was assigned a HIGH priority ranking.

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ISSUE 187: THE POTENTIAL IMPACT OF POSTULATED CESIUM CONCENTRATION ON EQUIPMENT QUALIFICATION

DESCRIPTION

NRC regulatory requirements include the estimation of individual offsite dose from the design basis LOCA. The fission product source term in TID-14844[73] which is referenced by the regulations, has historically been used for this application. As an update to the source term in TID-14844,[73] the NRC developed NUREG-1465,[1465] which provides a more realistic source term based on two decades of severe accident research; its use in offsite dose analysis provides safety and cost benefits. Accordingly, the NRC issued a new regulation, 10 CFR 50.67, allowing licensees to implement an alternative source term. Together with the issuance of 10 CFR 50.67, the NRC issued a regulatory guide which states that one acceptable alternative source term (AST) is the gap and in-vessel releases described in NUREG-1465.[1465] In the implementation of NUREG-1465[1465] for estimating offsite dose, the issue arose[1777] as to whether any additional requirements were needed with respect to estimating doses for equipment exposed to sump water.

NRC regulatory requirements also include the environmental qualification of equipment for the duration that it is needed to perform its safety function. This includes qualification for radiation, temperature, pressure, and humidity. Regulatory Guide 1.89[91] states that it is acceptable to use the TID-14844[73] source term for this application. Regulatory Guide 1.89 also states that, for equipment that must be qualified for more than thirty days, a source term that incorporates considerable quantities of cesium, as suggested by the accident at TMI-2, may produce doses greater than those estimated by TID-14844,[73] which includes a 1% release of cesium. The gap and in-vessel releases described in NUREG-1465[1465] include a 30% release of cesium.

The SNL report *Evaluation of Radiological Consequences of Design Basis Accidents at Operating Reactors Using the Revised Source Term*, dated September 28, 1998, showed that, for equipment exposed to the containment atmosphere, the TID-14844[73] source term and the gap and in-vessel releases in the AST produced similar integrated doses. This report also showed that, for equipment exposed to sump water, the integrated doses calculated with the AST exceeded those calculated with TID-14844,[73] after 42 days for a PWR and 145 days for a BWR, because of the 30% vs. 1% release of cesium.

CONCLUSION

The staff concluded that there was no clear basis for backfitting the requirement to modify the design basis for equipment qualification to adopt the AST. There would be no discernible risk reduction associated with such a requirement. Licensees should be aware, however, that a more realistic source term would potentially involve a larger dose for equipment exposed to sump water for long periods of time. Longer term equipment operability issues associated with severe fuel damage accidents, (with which the AST is associated) could also be addressed under accident management or plant recovery actions as necessary. Thus, the issue was DROPPED from further pursuit.[1776]

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