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SUPPLEMENT 31 TO NUREG-0933
"A PRIORITIZATION OF GENERIC SAFETY ISSUES"

REVISION INSERTION INSTRUCTIONS

	Remove	Insert
Introduction:	pp. 27 to 68, Rev. 30	pp. 27 to 68, Rev. 31
Section 3:	pp. 3.196-1 to 6	pp. 3.196-1 to 6, Rev. 1
	-	pp. 3.198-1 to 12
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References:	pp. R-1 to R-126, Rev. 20	pp. R-1 to R-129, Rev. 21
Appendix B	pp. A.B-1 to 13, Rev. 21	pp. A.B-1 to 13, Rev. 22
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	p. A.F.14-1	p. A.F.14-1, Rev. 1
	p. A.F.16-1	p. A.F.16-1, 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
Continue	- As defined in NRC Management Directive 6.4[1858]

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	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	R. 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	R. 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	R. Colmar	NRR/DHFS/LQB	LI (NOTE 3)	6	12/31/97	NA
I.A.2.5	Plant Drills	R. 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	R. Colmar	NRR/DHFT/HFIB	NOTE 3(a)	6	12/31/97	NA
I.A.2.6(2)	Staff Review of NRR 80-117	R. Colmar	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.2.6(3)	Revise 10 CFR 55	R. Colmar	NRR/DHFS/LQB	I.A.2.2	6	12/31/97	NA
I.A.2.6(4)	Operator Workshops	R. Colmar	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.2.6(5)	Develop Inspection Procedures for Training Program	R. Colmar	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.2.6(6)	Nuclear Power Fundamentals	R. Colmar	NRR/DHFS/LQB	DROP	6	12/31/97	NA
I.A.2.7	Accreditation of Training Institutions	R. 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	R. Emrit	NRR/DHFS/LQB	I	6	12/31/97	
I.A.3.2	Operator Licensing Program Changes	R. Emrit	NRR/DHFS/OLB	NOTE 3(b)	6	12/31/97	NA
I.A.3.3	Requirements for Operator Fitness	R. Colmar	RES/DRAO/HFSB	NOTE 3(b)	6	12/31/97	NA
I.A.3.4	Licensing of Additional Operations Personnel	D. Thatcher	NRR/DHFS/LQB	NOTE 3(b)	6	12/31/97	NA
I.A.3.5	Establish Statement of Understanding with INPO and DOE	D. 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	D. Thatcher	NRR/DHFS/OLB	NOTE 3(b)	6	12/31/97	NA
I.A.4.1(2)	Interim Changes in Training Simulators	D. Thatcher	NRR/DHFS/OLB	NOTE 3(a)	6	12/31/97	

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	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	R. Colmar	NRR/DHFT/HFIB	NOTE 3(a)	6	12/31/97	
I.A.4.2(2)	Upgrade Training Simulator Standards	R. Colmar	RES/DFO/HFBR	NOTE 3(a)	6	12/31/97	
I.A.4.2(3)	Regulatory Guide on Training Simulators	R. Colmar	RES/DFO/HFBR	NOTE 3(a)	6	12/31/97	
I.A.4.2(4)	Review Simulators for Conformance to Criteria	R. Colmar	NRR/DLPQ/LOLB	NOTE 3(a)	6	12/31/97	
I.A.4.3	Feasibility Study of Procurement of NRC Training Simulator	R. Colmar	RES/DAE/RSRB	LI (NOTE 3)	6	12/31/97	NA
I.A.4.4	Feasibility Study of NRC Engineering Computer	R. 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	R. Colmar	NRR/DHFT/HFIB	NOTE 3(b)	4	12/31/97	NA
I.B.1.1(2)	Prepare Commission Paper	R. 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	R. Colmar	NRR/DHFT/HFIB	NOTE 3(b)	4	12/31/97	NA
I.B.1.1(4)	Review Responses to Determine Acceptability	R. Colmar	NRR/DHFT/HFIB	NOTE 3(b)	4	12/31/97	NA
I.B.1.1(5)	Review Implementation of the Upgrading Activities	R. 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	R. 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	R. 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	G. Sege	RES	LI (NOTE 3)	4	12/31/97	NA
I.B.1.3(2)	Use Existing Enforcement Options to Accomplish Safest Shutdown Cooling	G. Sege	RES	LI (NOTE 3)	4	12/31/97	NA
I.B.1.3(3)	Use Non-Fiscal Approaches to Accomplish Safest Shutdown Cooling	G. 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	G. Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	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	G. 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	G. 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	G. 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	G. Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.1(6)	Observe Routine Maintenance	G. 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	G. Sege	OIE/DQASIP/RCPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.2	Resident Inspector at Operating Reactors	G. Sege	OIE/DQASIP/ORPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.3	Regional Evaluations	G. Sege	OIE/DQASIP/ORPB	LI (NOTE 3)	1	12/31/97	NA
I.B.2.4	Overview of Licensee Performance	G. 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	R. 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	R. 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	D. Thatcher	RES/DE/MEB	NOTE 3(b)	8	12/31/97	NA
I.D.4	Control Room Design Standard	D. Thatcher	RES/DRPS/RHFB	NOTE 3(b)	8	12/31/97	NA
I.D.5	Improved Control Room Instrumentation Research	-	-	-			

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
I.D.5(1)	Operator-Process Communication	D. Thatcher	RES/DFO/HFBR	NOTE 3(b)	8	12/31/97	NA
I.D.5(2)	Plant Status and Post-Accident Monitoring	D. Thatcher	RES/DFO/HFBR	NOTE 3(a)	8	12/31/97	
I.D.5(3)	On-Line Reactor Surveillance System	D. Thatcher	RES/DE/MEB	NOTE 3(b)	8	12/31/97	NA
I.D.5(4)	Process Monitoring Instrumentation	D. Thatcher	RES/DFO/ICBR	NOTE 3(b)	8	12/31/97	NA
I.D.5(5)	Disturbance Analysis Systems	D. Thatcher	RES/DRPS/RHFB	LI (NOTE 3)	8	12/31/97	NA
I.D.6	Technology Transfer Conference	D. Thatcher	RES/DFO/HFBR	LI (NOTE 3)	8	12/31/97	NA
I.E	ANALYSIS AND DISSEMINATION OF OPERATING EXPERIENCE						
I.E.1	Office for Analysis and Evaluation of Operational Data	P. Matthews	AEOD/PTB	LI (NOTE 3)	3	12/31/97	NA
I.E.2	Program Office Operational Data Evaluation	P. Matthews	NRR/DL/ORAB	LI (NOTE 3)	3	12/31/97	NA
I.E.3	Operational Safety Data Analysis	P. Matthews	RES/DRA/RRBR	LI (NOTE 3)	3	12/31/97	NA
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs	P. Matthews	AEOD/PTB	LI (NOTE 3)	3	12/31/97	NA
I.E.5	Nuclear Plant Reliability Data System	P. Matthews	AEOD/PTB	LI (NOTE 3)	3	12/31/97	NA
I.E.6	Reporting Requirements	P. Matthews	AEOD/PTB	LI (NOTE 3)	3	12/31/97	NA
I.E.7	Foreign Sources	P. Matthews	IP	LI (NOTE 3)	3	12/31/97	NA
I.E.8	Human Error Rate Analysis	P. Matthews	RES/DFO/HFBR	LI (NOTE 3)	3	12/31/97	NA
I.F	QUALITY ASSURANCE						
I.F.1	Expand QA List	J. 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	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	J. 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	J. 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	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA
I.F.2(5)	Establish Qualification Requirements for QA and QC Personnel	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA
I.F.2(6)	Increase the Size of Licensees' QA Staff	J. 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	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA
I.F.2(8)	Compare NRC QA Requirements with Those of Other Agencies	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	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	J. Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	4	12/31/98	NA
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/30/98	NA
I.F.2(11)	Define Role of QA in Design and Analysis Activities	J. 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	H. Vandermolen	NRR/DHFS/PSRB	NOTE 3(a)	3	12/31/97	NA
II.A	SITING						
II.A.1	Siting Policy Reformulation	H. Vandermolen	NRR/DE/SAB	NOTE 3(b)	2	12/31/97	NA
II.A.2	Site Evaluation of Existing Facilities	H. 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	H. Vandermolen	RES/DSR/AEB	LI (NOTE 5)	4	12/31/97	NA
II.B.5(2)	Behavior of Core-Melt	H. 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	H. 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	J. Pittman	NRR/DST/RRAB	NOTE 3(a)	4	12/31/97	
II.B.7	Analysis of Hydrogen Control	P. Matthews	NRR/DSI/CSB	II.B.8	4	12/31/97	
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	H. 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	J. Pittman	RES/DRAO/RRB	NOTE 3(b)	3	12/31/97	NA
II.C.2	Continuation of Interim Reliability Evaluation Program	J. Pittman	NRR/DST/RRAB	NOTE 3(b)	3	12/31/97	NA
II.C.3	Systems Interaction	J. Pittman	NRR/DST/GIB	A-17	3	12/31/97	NA
II.C.4	Reliability Engineering	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	3	12/31/97	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	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	R. 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	R. 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	R. 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	R. Riggs	RES/DAE/RSRB	NOTE 3(b)	3	12/31/98	NA
II.E.2.3	Uncertainties in Performance Predictions	H. 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	H. Vandermolen	NRR/DST/GIB	A-45	2	12/31/97	NA
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements	H. Vandermolen	NRR/DST/GIB	A-45	2	12/31/97	NA
II.E.3.4	Alternate Concepts Research	R. Riggs	RES/DAE/FBRB	NOTE 3(b)	2	12/31/97	NA
II.E.3.5	Regulatory Guide	R. 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	W. 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	W. 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	W. Milstead	NRR/DSI/CSB	NOTE 3(a)	2	12/31/97	
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability	W. Milstead	NRR/DSI/CSB	NOTE 3(a)	2	12/31/97	
II.E.4.4(4)	Evaluate Purging and Venting During Normal Operation	W. Milstead	NRR/DSI/CSB	NOTE 3(b)	2	12/31/97	NA
II.E.4.4(5)	Issue Modified Purging and Venting Requirement	W. Milstead	NRR/DSI/CSB	NOTE 3(b)	2	12/31/97	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.E.5	Design Sensitivity of B&W Reactors						
II.E.5.1	Design Evaluation	D. Thatcher	NRR/DSI/RSB	NOTE 3(a)	2	12/31/98	
II.E.5.2	B&W Reactor Transient Response Task Force	D. Thatcher	NRR/DL/ORAB	NOTE 3(a)	2	12/31/98	
II.E.6	In Situ Testing of Valves						
II.E.6.1	Test Adequacy Study	D. Thatcher	RES/DE/EIB	NOTE 3(a)	2	12/31/98	
II.F	INSTRUMENTATION AND CONTROLS						
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	H. Vandermolen	RES/DFO/ICBR	NOTE 3(a)	3	12/31/98	
II.F.4	Study of Control and Protective Action Design Requirements	D. Thatcher	NRR/DSI/ICSB	DROP	3	12/31/98	NA
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	D. Thatcher	RES/DE	LI (NOTE 3)	3	12/31/98	NA
II.G	ELECTRICAL POWER						
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	-	NRR	I	1	12/31/98	NA
II.H	TMI-2 CLEANUP AND EXAMINATION						
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact	P. 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	W. Milstead	RES/DRAA/AEB	NOTE 3(b)	3	12/31/98	NA
II.H.3	Evaluate and Feed Back Information Obtained from TMI	W. Milstead	NRR/TMIPO	II.H.2	3	12/31/98	NA
II.H.4	Determine Impact of TMI on Socioeconomic and Real Property Values	W. Milstead	RES/DHSWM/SEBR	LI (NOTE 3)	3	12/31/98	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	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	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.1.2	Modify Existing Vendor Inspection Program	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers	L. 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	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.2.3	Assign Resident Inspectors to All Construction Sites	L. 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	J. Pittman	NRR/DHFS/LQB	I.B.1.1	1	12/31/98	NA
II.J.3.2	Issue Regulatory Guide	J. 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	L. 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	R. 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	R. 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	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(4)	Review Operating Procedures and Training Instructions	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(5)	Safety-Related Valve Position Description	R. Emrit	NRR	NOTE 3(a)		12/31/84	-

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	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	R. 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	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	R. 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	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(10	Review and Modify Procedures for Removing Safety-Related Systems from Service	R. 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	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(12)	One Hour Notification Requirement and Continuous Communications Channels	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	R. 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	R. 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	R. 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	R. 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	R. 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	R. 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	R. 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	R. Emrit	NRR	NOTE 3(a)		12/31/84	-

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	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	R. 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	R. 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	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(25)	Develop Operator Action Guidelines	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	R. 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	R. 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	R. 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	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	R. Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(4)	Small-Break LOCA Analysis, Procedures and Operator Training	R. Emrit	NRR/DHFS/OLB	NOTE 3(a)		12/31/84	-
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2(6)	Reevaluate Analysis for Dual-Level Setpoint Control	R. Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(7)	Reevaluate Transient of September 24, 1977	R. Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(8)	Continued Upgrading of AFW System	R. 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	R. Emrit	NRR	I		12/31/84	F-27
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	R. Emrit	NRR	I		12/31/84	F-28
II.K.2(11)	Operator Training and Drilling	R. Emrit	NRR	I		12/31/84	F-29
II.K.2(12)	Transient Analysis and Procedures for Management of Small Breaks	R. 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	R. Emrit	NRR	I		12/31/84	F-30
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	R. 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	R. Emrit	NRR	I		12/31/84	-
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA With Loss of Offsite Power	R. Emrit	NRR	I		12/31/84	F-32
II.K.2(17)	Analysis of Potential Voiding in RCS During Anticipated Transients	R. Emrit	NRR	I		12/31/84	F-33

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	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	R. 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	R. 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	R. Emrit	NRR	I		12/31/84	F-35
II.K.2(21)	LOFT L3-1 Predictions	R. 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	R. Emrit	NRR	I		12/31/84	F-36
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	R. Emrit	NRR	I		12/31/84	F-37
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	R. 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	R. 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	R. Emrit	NRR	I		12/31/84	F-39, G-01
II.K.3(6)	Instrumentation to Verify Natural Circulation	R. 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	R. Emrit	NRR	I		12/31/84	-
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs	R. Emrit	NRR/DST/GIB	II.C.1, II.E.3.3		12/31/84	NA
II.K.3(9)	Proportional Integral Derivative Controller Modification	R. 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	R. 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	R. Emrit	NRR	I		12/31/84	-
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	R. Emrit	NRR	I		12/31/84	F-42
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	R. Emrit	NRR	I		12/31/84	F-43
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	R. 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	R. Emrit	NRR	I		12/31/84	F-45

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	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	R. Emrit	NRR	I		12/31/84	F-46
II.K.3(17)	Report on Outage of ECC Systems - Licensee Report and Technical Specification Changes	R. 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	R. Emrit	NRR	I		12/31/84	F-48
II.K.3(19)	Interlock on Recirculation Pump Loops	R. Emrit	NRR	I		12/31/84	F-49
II.K.3(20)	Loss of Service Water for Big Rock Point	R. Emrit	NRR	I		12/31/84	-
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level - Design and Modification	R. Emrit	NRR	I		12/31/84	F-50
II.K.3(22)	Automatic Switchover of RCIC System Suction - Verify Procedures and Modify Design	R. Emrit	NRR	I		12/31/84	F-51
II.K.3(23)	Central Water Level Recording	R. 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	R. Emrit	NRR	I		12/31/84	F-52
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	R. Emrit	NRR	I		12/31/84	F-53
II.K.3(26)	Study Effect on RHR Reliability of Its Use for Fuel Pool Cooling	R. Emrit	NRR/DSI	II.E.2.1		12/31/84	NA
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	R. Emrit	NRR	I		12/31/84	F-54
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	R. Emrit	NRR	I		12/31/84	F-55
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	R. 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	R. Emrit	NRR	I		12/31/84	F-57
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	R. Emrit	NRR	I		12/31/84	F-58
II.K.3(32)	Provide Experimental Verification of Two-Phase Natural Circulation Models	R. Emrit	NRR/DSI	II.E.2.2		12/31/84	NA
II.K.3(33)	Evaluate Elimination of PORV Function	R. Emrit	NRR	II.C.1		12/31/84	NA
II.K.3(34)	Relap-4 Model Development	R. 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	R. 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	R. Emrit	NRR	I.C.1(3)		12/31/84	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	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	R. 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	R. 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	R. 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	R. Emrit	NRR	II.K.2(16)		12/31/84	NA
II.K.3(41)	Submit Predictions for LOFT Test L3-6 with RCPs Running	R. Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(42)	Submit Requested Information on the Effects of Non-Condensible Gases	R. 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	R. 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	R. Emrit	NRR	I		12/31/84	F-59
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	R. Emrit	NRR	I		12/31/84	F-60
II.K.3(46)	Response to List of Concerns from ACRS Consultant	R. 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	R. 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	R. Emrit	NRR	II.C.1, II.C.2		12/31/84	NA
II.K.3(49)	Review of Procedures (NRC)	R. Emrit	NRR/DHFS/PSRB	I.C.8, I.C.9		12/31/84	NA
II.K.3(50)	Review of Procedures (NSSS Vendors)	R. Emrit	NRR/DHFS/PSRB	I.C.7, I.C.9		12/31/84	NA
II.K.3(51)	Symptom-Based Emergency Procedures	R. Emrit	NRR/DHFS/PSRB	I.C.9		12/31/84	NA
II.K.3(52)	Operator Awareness of Revised Emergency Procedures	R. 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	R. Emrit	NRR	I.A.1.3		12/31/84	NA
II.K.3(54)	Simulator Upgrade for Small-Break LOCAs	R. Emrit	NRR	I.A.4.1(2)		12/31/84	NA
II.K.3(55)	Operator Monitoring of Control Board	R. Emrit	NRR	I.C.1(3), I.D.2, I.D.3		12/31/84	NA
II.K.3(56)	Simulator Training Requirements	R. 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	R. Emrit	NRR	I		12/31/84	F-62

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	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	R. Riggs	OIE/DEPER/EPB	NOTE 3(b)	2	06/30/91	NA
III.A.1.3(2)	Public	R. 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	R. 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	R. 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	R. Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(4)	Prepare Commission Paper	R. 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	R. Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.2	Improve Operations Centers	R. 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	J. Pittman	OIE/DEPER/IRDB	NOTE 3(a)	1	06/30/85	NA
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems	J. Pittman	OIE/DEPER/IRDB	NOTE 3(a)	1	06/30/85	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
III.A.3.4	Nuclear Data Link	D. Thatcher	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	
III.A.3.5	Training, Drills, and Tests	J. 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	J. Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	06/30/85	NA
III.A.3.6(2)	Federal	J. Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	06/30/85	NA
III.A.3.6(3)	State and Local	J. 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	W. 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	W. Milstead	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.B.2(2)	Federal Guidance	W. 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	J. Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.1(2)	Recommend Publication of Additional Information	J. Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.1(3)	Program of Seminars for News Media Personnel	J. 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	J. Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.2(2)	Provide Training for Members of the Technical Staff	J. 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	R. Emrit	RES/DRA/ARGIB	DROP	1	12/31/88	
III.D.1.1(3)	Develop Proposed System Acceptance Criteria	R. Emrit	RES/DRA/ARGIB	DROP	1	12/31/88	
III.D.1.2	Radioactive Gas Management	R. 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	R. Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
III.D.1.3(2)	Review and Revise SRP	R. Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems	R. Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Adsorber	R. 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	R. 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	R. 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	R. Emrit	NRR/DSI/METB	LOW	3	12/31/98	NA
III.D.2.1(3)	Revise Regulatory Guides	R. 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	R. Emrit	NRR/DSI/RAB	NOTE 3(b)	3	12/31/98	NA
III.D.2.2(2)	Evaluate Data Collected at Quad Cities	R. 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	R. Emrit	NRR/DSI/RAB	III.D.2.5	3	12/31/98	NA
III.D.2.2(4)	Revise SRP and Regulatory Guides	R. 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	R. 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	R. Emrit	NRR/DE/EHEB	NOTE 3(b)	3	12/31/98	NA
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction	R. Emrit	NRR/DE/EHEB	NOTE 3(b)	3	12/31/98	NA
III.D.2.3(4)	Prepare a Summary Assessment	R. 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	H. Vandermolen	NRR/DSI/RAB	NOTE 3(b)	3	12/31/98	NA
III.D.2.4(2)	Place 50 TLDs Around Each Site	H. Vandermolen	OIE/DRP/ORPB	LI (NOTE 3)	3	12/31/98	NA
III.D.2.5	Offsite Dose Calculation Manual	H. Vandermolen	NRR/DSI/RAB	NOTE 3(b)	3	12/31/98	NA
III.D.2.6	Independent Radiological Measurements	H. 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	H. 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	H. Vandermolen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(2)	Issue a Regulatory Guide	H. Vandermolen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
III.D.3.2(3)	Develop Standard Performance Criteria	H. 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	H. 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	H. 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	H. Vandermolen	DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
III.D.3.5(3)	Revise 10 CFR 20	H. Vandermolen	DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
IV.A	STRENGTHEN ENFORCEMENT PROCESS						
IV.A.1	Seek Legislative Authority	R. Emrit	GC	LI (NOTE 3)		11/30/83	NA
IV.A.2	Revise Enforcement Policy	R. Emrit	OIE/ES	LI (NOTE 3)		11/30/83	NA
IV.B	ISSUANCE OF INSTRUCTIONS AND INFORMATION TO LICENSEES						
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	R. Emrit	OIE/DEPER	LI (NOTE 3)		11/30/83	NA
IV.C	EXTEND LESSONS LEARNED TO LICENSED ACTIVITIES OTHER THAN POWER REACTORS						
IV.C.1	Extend Lessons Learned from TMI to Other NRC Programs	R. Emrit	NMSS/WM	NOTE 3(b)		11/30/83	NA
IV.D	NRC STAFF TRAINING						
IV.D.1	NRC Staff Training	R. Emrit	ADM/MDTS	LI (NOTE 3)		11/30/83	NA

Table II (Continued)

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

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	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	R. Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.C.2	Study Need for Additional Advisory Committees	R. Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.C.3	Study the Need to Establish an Independent Nuclear Safety Board	R. 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	R. Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.D.2	Study Construction-During-Adjudication Rules	R. Emrit	GC	LI (NOTE 5)		12/31/86	NA
V.D.3	Reexamine Commission Role in Adjudication	R. Emrit	GC	LI (NOTE 5)		12/31/86	NA
V.D.4	Study the Reform of the Licensing Process	R. Emrit	GC	LI (NOTE 5)		12/31/86	NA
V.E	LEGISLATIVE NEEDS						
V.E.1	Study the Need for TMI-Related Legislation	R. Emrit	GC	LI (NOTE 5)		12/31/86	NA
V.F	ORGANIZATION AND MANAGEMENT						
V.F.1	Study NRC Top Management Structure and Process	R. Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.F.2	Reexamine Organization and Functions of the NRC Offices	R. Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.F.3	Revise Delegations of Authority to Staff	R. 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	R. Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.F.5	Authority to Delegate Emergency Response Functions to a Single Commissioner	R. Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.G	CONSOLIDATION OF NRC LOCATIONS						
V.G.1	Achieve Single Location, Long-Term	R. Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.G.2	Achieve Single Location, Interim	R. Emrit	GC	LI (NOTE 3)		12/31/86	NA
	TASK ACTION PLAN ITEMS						
A-1	Water Hammer (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	D-10

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Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	R. Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	12/31/88	
A-4	CE Steam Generator Tube Integrity (former USI)	R. Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	12/31/88	
A-5	B&W Steam Generator Tube Integrity (former USI)	R. Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	12/31/88	
A-6	Mark I Short-Term Program (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-7	Mark I Long-Term Program (former USI)	R. 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)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-9	ATWS (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-10	BWR Feedwater Nozzle Cracking (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-25
A-11	Reactor Vessel Materials Toughness (former USI)	R. 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)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-13	Snubber Operability Assurance	R. Emrit	NRR/DE/MEB	NOTE 3(a)	1	06/30/91	B-17, B-22
A-14	Flaw Detection	P. Matthews	NRR/DE/MTEB	DROP		11/30/83	NA
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning	J. Pittman	NRR/DE/CHEB	NOTE 3(b)		11/30/83	NA
A-16	Steam Effects on BWR Core Spray Distribution	R. Emrit	NRR/DSI/CPB	NOTE 3(a)		11/30/83	D-12
A-17	Systems Interactions in Nuclear Power Plants (former (USI)	R. Emrit	RES/DSIR/EIB	NOTE 3(b)	1	12/31/89	NA
A-18	Pipe Rupture Design Criteria	R. Emrit	NRR/DE/MEB	DROP		11/30/83	NA
A-19	Digital Computer Protection System	W. 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	H. 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	P. Matthews	NRR/DSI/CSB	RI (NOTE 5)		11/30/83	
A-24	Qualification of Class 1E Safety-Related Equipment (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-60
A-25	Non-Safety Loads on Class 1E Power Sources	D. Thatcher	NRR/DSI/PSB	NOTE 3(a)		11/30/83	
A-26	Reactor Vessel Pressure Transient Protection (former (USI)	R. 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	R. Colmar	NRR/DE/SGEB	NOTE 3(a)		11/30/83	
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	R. Colmar	RES/DRPS/RPSI	NOTE 3(b)	1	12/31/89	NA
A-30	Adequacy of Safety-Related DC Power Supplies	G. Sege	NRR/DSI/PSB	128	1	12/31/86	NA
A-31	RHR Shutdown Requirements (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-32	Missile Effects	J. Pittman	NRR/DE/MTEB	A-37, A-38, B-68		11/30/83	NA

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Action Plan Item/ Issue No.	Title	Priority Analyst	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	R. Emrit	NRR/DSI/PSB	NOTE 3(a)	1	12/31/94	B-23
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	R. Emrit	NRR/DSI/GIB	NOTE 3(a)	2	06/30/04	C-10, C-15
A-37	Turbine Missiles	J. Pittman	NRR/DE/MTEB	DROP		11/30/83	NA
A-38	Tornado Missiles	G. 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)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-40	Seismic Design Criteria (former USI)	R. Emrit	RES/DSIR/EIB	NOTE 3(a)	1	12/31/89	NA
A-41	Long-Term Seismic Program	L. Riani	NRR/DE/MEB	NOTE 3(b)	1	12/31/84	NA
A-42	Pipe Cracks in Boiling Water Reactors (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-05
A-43	Containment Emergency Sump Performance (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	12/31/87	
A-44	Station Blackout (former USI)	R. Emrit	RES/DRPS/RPSI	NOTE 3(a)	1	06/30/88	
A-45	Shutdown Decay Heat Removal Requirements (former USI)	R. Emrit	RES/DRPS/RPSI	NOTE 3(b)	1	12/31/88	NA
A-46	Seismic Qualification of Equipment in Operating Plants (former USI)	R. Emrit	NRR/DSRO/EIB	NOTE 3(a)	2	06/30/00	
A-47	Safety Implications of Control Systems (former USI)	R. Emrit	RES/DSIR/EIB	NOTE 3(a)	1	12/31/89	
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	R. Emrit	NRR/DSIR/SAIB	NOTE 3(a)	1	06/30/89	
A-49	Pressurized Thermal Shock (former USI)	R. 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	R. 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	D. Thatcher	RES/DE/EIB	NOTE 3(b)	1	06/30/88	NA
B-6	Loads, Load Combinations, Stress Limits	J. 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	R. Riggs	NRR/DSI/RSB	DROP	1	12/31/94	NA
B-9	Electrical Cable Penetrations of Containment	R. Emrit	NRR/DSI/PSB	NOTE 3(b)		11/30/83	NA
B-10	Behavior of BWR Mark III Containments	H. 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)	R. 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	R. 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	R. Emrit	NRR/DE/MEB	A-18		11/30/83	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
B-17	Criteria for Safety-Related Operator Actions	W. Milstead	RES/DST/CIHFB	NOTE 3(b)	3	06/30/00	
B-18	Vortex Suppression Requirements for Containment Sumps	R. Emrit	NRR/DST/GIB	A-43		11/30/83	NA
B-19	Thermal-Hydraulic Stability	L. Riani	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	R. 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	R. 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	R. 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	J. 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	W. Milstead	NRR/DE/SGEB	LI (NOTE 3)	1	06/30/89	NA
B-32	Ice Effects on Safety-Related Water Supplies	J. 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	R. 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	R. 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	L. Riani	NRR/DE/MTEB	DROP		11/30/83	NA
B-48	BWR Control Rod Drive Mechanical Failures	R. 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 Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
B-50	Post-Operating Basis Earthquake Inspection	L. Riani	NRR/DE/SGEB	RI (NOTE 3)	1	06/30/85	NA
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components	R. Emrit	NRR/DE/MEB	A-40		11/30/83	NA
B-52	Fuel Assembly Seismic and LOCA Responses	R. Emrit	NRR/DST/GIB	A-2		11/30/83	NA
B-53	Load Break Switch	G. Sege	NRR/DSI/PSB	RI (NOTE 3)		11/30/83	
B-54	Ice Condenser Containments	W. Milstead	NRR/DSI/CSB	NOTE 3(b)	1	12/31/84	NA
B-55	Improved Reliability of Target Rock Safety Relief Valves	H. Vandermolen	NRR/DE/EMEB	NOTE 3(b)	1	06/30/00	
B-56	Diesel Reliability	W. Milstead	RES/DRPS/RPSI	NOTE 3(a)	2	06/30/95	D-19
B-57	Station Blackout	R. Emrit	NRR/DST/GIB	A-44		11/30/83	
B-58	Passive Mechanical Failures	L. Riani	NRR/DE/EQB	NOTE 3(b)	1	12/31/85	NA
B-59	(N-1) Loop Operation in BWRs and PWRs	L. Riani	NRR/DSI/RSB	RI (NOTE 3)	1	06/30/85	E-04,E-05
B-60	Loose Parts Monitoring Systems	R. Emrit	NRR/DSI/CPB	NOTE 3(b)	1	12/31/84	NA
B-61	Allowable ECCS Equipment Outage Periods	J. 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	R. Emrit	NRR/DE/MEB	NOTE 3(a)		11/30/83	B-45
B-64	Decommissioning of Reactors	L. Riani	RES/DE/MEB	NOTE 3(a)	2	06/30/95	NA
B-65	Iodine Spiking	W. Milstead	NRR/DSI/AEB	DROP	2	12/31/84	NA
B-66	Control Room Infiltration Measurements	P. Matthews	NRR/DSI/AEB	NOTE 3(a)		11/30/83	
B-67	Effluent and Process Monitoring Instrumentation	L. Riani	NRR/DSI/METB	III.D.2.1		11/30/83	NA
B-68	Pump Overspeed During LOCA	L. Riani	NRR/DSI/ASB	DROP		11/30/83	NA
B-69	ECCS Leakage Ex-Containment	L. 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	R. Emrit	NRR/DSI/PSB	NOTE 3(b)		11/30/83	
B-71	Incident Response	L. 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	D. 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	W. 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	R. Emrit	NRR/DSI/CSB	NOTE 3(b)		11/30/83	NA
C-3	Insulation Usage Within Containment	R. Emrit	NRR/DST/GIB	A-43	1	06/30/91	NA
C-4	Statistical Methods for ECCS Analysis	R. Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-5	Decay Heat Update	R. Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-6	LOCA Heat Sources	R. Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-7	PWR System Piping	R. Emrit	NRR/DE/MTEB	NOTE 3(b)		11/30/83	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
C-8	Main Steam Line Leakage Control Systems	W. 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	R. Emrit	NRR/DSI/AEB	NOTE 3(a)		11/30/83	NA
C-11	Assessment of Failure and Reliability of Pumps and Valves	R. Emrit	NRR/DE/MEB	NOTE 3(b)		12/31/85	NA
C-12	Primary System Vibration Assessment	D. Thatcher	NRR/DE/MEB	NOTE 3(b)		11/30/83	NA
C-13	Non-Random Failures	R. Emrit	NRR/DST/GIB	A-17	1	06/30/91	NA
C-14	Storm Surge Model for Coastal Sites	R. 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	R. Emrit	NRR/DSI/METB	NOTE 3(a)		11/30/83	NA
D-1	Advisability of a Seismic Scram	D. Thatcher	RES/DET/MSEB	DROP	1	12/31/98	NA
D-2	Emergency Core Cooling System Capability for Future Plants	R. Emrit	RES/DRA/ARGIB	DROP		12/31/88	NA
D-3	Control Rod Drop Accident	R. Emrit	NRR/DSI/CPB	NOTE 3(b)		11/30/83	NA
	NEW GENERIC ISSUES						
1.	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems	R. 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	R. Emrit	NRR/DSIR/RPSIB	NOTE 3(b)	1	06/30/86	NA
4.	End-of-Life and Maintenance Criteria	D. Thatcher	NRR/DE/EQB	NOTE 3(b)		11/30/83	NA
5.	Design Check and Audit of Balance-of-Plant Equipment	J. 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	H. Vandermolen	NRR/DSI/CPB	NOTE 3(b)	1	12/31/94	NA
7.	Failures Due to Flow-Induced Vibrations	H. Vandermolen	NRR/DSI/RSB	DROP	1	06/30/91	NA
8.	Inadvertent Actuation of Safety Injection in PWRs	L. Riani	NRR/DSI/RSB	I.C.1		11/30/83	NA
9.	Reevaluation of Reactor Coolant Pump Trip Criteria	R. Emrit	NRR/DSI/RSB	II.K.3(5)		11/30/83	NA
10.	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges	R. Riggs	NRR/DSI/ICSB	DROP		11/30/83	NA
11.	Turbine Disc Cracking	J. Pittman	NRR/DE/MTEB	A-37		11/30/83	NA
12.	BWR Jet Pump Integrity	G. Sege	NRR/DE/MTEB, MEB	NOTE 3(b)	1	12/31/84	NA
13.	Small Break LOCA from Extended Overheating of Pressurizer Heaters	L. Riani	NRR/DSI/RSB	DROP		11/30/83	NA
14.	PWR Pipe Cracks	R. Emrit	NRR/DE/MTEB	NOTE 3(b)	2	12/31/94	NA
15.	Radiation Effects on Reactor Vessel Supports	R. Emrit	RES/DET/EMMEB	NOTE 3(b)	3	06/30/96	NA

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

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

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Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
67.2.1	Integrity of Steam Generator Tube Sleeves	R. Riggs	NRR/DE/MEB	135	4	06/30/94	NA
67.3.1	Steam Generator Overfill	R. Riggs	NRR/DST/GIB NRR/DSI/RSB	A-47, I.C.1	4	06/30/94	NA
67.3.2	Pressurized Thermal Shock	R. Riggs	NRR/DST/GIB	A-49	4	06/30/94	NA
67.3.3	Improved Accident Monitoring	R. Riggs	NRR/DSI/ICSB	NOTE 3(a)	4	06/30/94	A-17
67.3.4	Reactor Vessel Inventory Measurement	R. Riggs	NRR/DSI/CPB	II.F.2	4	06/30/94	NA
67.4.1	RCP Trip	R. Riggs	NRR/DSI/RSB	II.K.3(5)	4	06/30/94	G-01
67.4.2	Control Room Design Review	R. Riggs	NRR/DHFS/HFEB	I.D.1	4	06/30/94	F-08
67.4.3	Emergency Operating Procedures	R. Riggs	NRC/DHFS/PSRB	I.C.1	4	06/30/94	F-05
67.5.1	Reassessment of Radiological Consequences	R. Riggs	RES/DRPS/RPSI	LI (NOTE 3)	4	06/30/94	NA
67.5.2	Reevaluation of SGTR Design Basis	R. Riggs	RES/DRPS/RPSI	LI (67.5.1)	4	06/30/94	NA
67.5.3	Secondary System Isolation	R. Riggs	NRR/DSI/RSB	DROP	4	06/30/94	NA
67.6.0	Organizational Responses	R. Riggs	OIE/DEPER/IRDB	III.A.3	4	06/30/94	NA
67.7.0	Improved Eddy Current Tests	R. Riggs	RES/DE/EIB	135	4	06/30/94	NA
67.8.0	Denting Criteria	R. Riggs	NRR/DE/MTEB	135	4	06/30/94	NA
67.9.0	Reactor Coolant System Pressure Control	R. 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	R. 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	J. Pittman	NRR/DSI/ASB	124	3	06/30/91	NA
69.	Make-up Nozzle Cracking in B&W Plants	R. Colmar	NRR/DE/MEB, MTEB	NOTE 3(b)	1	12/31/84	B43
70.	PORV and Block Valve Reliability	R. 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	J. Pittman	RES/DRA/ARGIB	DROP	3	06/30/01	NA
72.	Control Rod Drive Guide Tube Support Pin Failures	R. Riggs	RES	DROP	1	06/30/91	NA
73.	Detached Thermal Sleeves	R. Emrit	RES/DSIR/EIB	NOTE 3(a)	3	06/30/95	NA
74.	Reactor Coolant Activity Limits for Operating Reactors	W. Milstead	NRR/DSI/AEB	DROP	1	06/30/86	NA
75.	Generic Implications of ATWS Events at the Salem Nuclear Plant	R. 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 Analyst	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	L. Riani	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	L. Riani	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	H. Vandermolen	RES/DSARE/REAHFB	NOTE 3(b)	4	06/30/06	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	H. Vandermolen	RES/DRPS/RPSI	NOTE 3(b)	3	06/30/04	NA
83.	Control Room Habitability	R. Emrit	RES/DST/AEB	NOTE 3(b)	3	06/30/03	NA
84.	CE PORVs	R. 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	W. Milstead	NRR/DSI/CSB	DROP	2	06/30/91	NA
86.	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	R. Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	06/30/88	B-84
87.	Failure of HPCI Steam Line Without Isolation	J. Pittman	RES/DSIR/EIB	NOTE 3(a)	2	06/30/95	
88.	Earthquakes and Emergency Planning	R. Riggs	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
89.	Stiff Pipe Clamps	T.Y. Chang	RES/DSIR/EIB	LOW	2	06/30/95	NA
90.	Technical Specifications for Anticipatory Trips	H. Vandermolen	NRR/DSI/RSB, ICSB	DROP	2	12/31/98	NA
91.	Main Crankshaft Failures in Transamerica DeLaval Emergency Diesel Generators	R. Emrit	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
92.	Fuel Crumbling During LOCA	H. Vandermolen	NRR/DSI/RSB, CPB	DROP	1	12/31/98	NA
93.	Steam Binding of Auxiliary Feedwater Pumps	J. Pittman	RES/DRPS/RPSI	NOTE 3(a)		06/30/88	B-98
94.	Additional Low Temperature Overpressure Protection for Light Water Reactors	J. Pittman	RES/DSIR/RPSI	NOTE 3(a)		06/30/90	
95.	Loss of Effective Volume for Containment Recirculation Spray	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)		06/30/90	NA
96.	RHR Suction Valve Testing	W. Milstead	RES/DRA/ARGIB	105		06/30/90	NA
97.	PWR Reactor Cavity Uncontrolled Exposures	H. Vandermolen	NRR/DSI/RAB	III.D.3.1		06/30/85	NA
98.	CRD Accumulator Check Valve Leakage	J. Pittman	NRR/DSI/ASB	DROP		06/30/85	NA
99.	RCS/RHR Suction Line Valve Interlock on PWRs	J. Pittman	RES/DRPS/RPSI	NOTE 3(a)	3	06/30/91	L-817
100.	Once-Through Steam Generator Level	J. Jackson	RES/DSIR/EIB	DROP	1	06/30/95	NA

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

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Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
122.2	Initiating Feed-and-Bleed	H. Vandermolen	NRR/DEST/SRXB	NOTE 3(b)	4	12/31/98	NA
122.3	Physical Security System Constraints	H. 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	W. Milstead	RES/DSIR/SAIB	DROP	1	06/30/95	NA
124.	Auxiliary Feedwater System Reliability	R. 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	H. 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	H. Vandermolen	NRR/DSRO/SPEB	70	7	12/31/98	NA
125.I.2.b	Need for PORV Surveillance Tests to Confirm Operational Readiness	H. Vandermolen	NRR/DSRO/SPEB	70	7	12/31/98	NA
125.I.2.c	Need for Additional Protection Against PORV Failure	H. Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.I.2.d	Capability of the PORV to Support Feed-and-Bleed	H. Vandermolen	NRR/DSRO/SPEB	A-45	7	12/31/98	NA
125.I.3	SPDS Availability	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)	7	12/31/98	NA
125.I.4	Plant-Specific Simulator	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.5	Safety Systems Tested in All Conditions Required by DBA	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.6	Valve Torque Limit and Bypass Switch Settings	H. Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.7	Operator Training Adequacy	-	-	-			
125.I.7.a	Recover Failed Equipment	J. Pittman	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.7.b	Realistic Hands-On Training	H. Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.8	Procedures and Staffing for Reporting to NRC Emergency Response Center	H. 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	H. Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.1.b	Review Existing AFW Systems for Single Failure	H. Vandermolen	NRR/DSRO/SPEB	124	7	12/31/98	NA
125.II.1.c	NUREG-0737 Reliability Improvements	H. 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	H. Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.2	Adequacy of Existing Maintenance Requirements for Safety-Related Systems	R. 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	R. 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	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	
125.II.6	Reexamine PRA Estimates of Core Damage Risk from Loss	H. Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA

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Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	of All Feedwater						
125.II.7	Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break	H. Vandermolen	RES/DRPS/RPSI	NOTE 3(b)	7	12/31/98	NA
125.II.8	Reassess Criteria for Feed-and-Bleed Initiation	H. Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.9	Enhanced Feed-and-Bleed Capability	H. Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.10	Hierarchy of Impromptu Operator Actions	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.11	Recovery of Main Feedwater as Alternative to Auxiliary Feedwater	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.12	Adequacy of Training Regarding PORV Operation	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.13	Operator Job Aids	J. Pittman	NRR/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.14	Remote Operation of Equipment Which Must Now Be Operated Locally	H. Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
126.	Reliability of PWR Main Steam Safety Valves	R. Riggs	RES/DRA/ARGIB	LI (NOTE 3)		06/30/88	NA
127.	Maintenance and Testing of Manual Valves in Safety-Related Systems	J. Pittman	RES/DRA/ARGIB	LOW		12/31/87	NA
128.	Electrical Power Reliability	R. Emrit	RES/DSIR/EIB	NOTE 3(a)	2	06/30/95	
129.	Valve Interlocks to Prevent Vessel Drainage During Shutdown Cooling	W. Milstead	RES/DRA/ARGIB	DROP		06/30/90	NA
130.	Essential Service Water Pump Failures at Multiplant Sites	R. 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	R. 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	J. Pittman	NRR/DLPQ/LHFB	LI (NOTE 3)	1	12/31/91	NA
134.	Rule on Degree and Experience Requirement	J. Pittman	RES/DRA/RDB	NOTE 3(b)		12/31/89	NA
135.	Steam Generator and Steam Line Overfill	R. Emrit	RES/DSIR/EIB	NOTE 3(b)	3	06/30/95	NA
136.	Storage and Use of Large Quantities of Cryogenic Combustibles On Site	W. Milstead	RES/DRA/ARGIB	LI (NOTE 3)		06/30/88	NA
137.	Refueling Cavity Seal Failure	W. 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	W. Milstead	RES/DSIR/SAIB	DROP	2	12/31/98	NA
139.	Thinning of Carbon Steel Piping in LWRs	R. Riggs	RES/DRA/ARGIB	RI (NOTE 3)	1	06/30/95	NA
140.	Fission Product Removal Systems	R. Riggs	RES/DRA/ARGIB	DROP		06/30/90	NA
141.	Large-Break LOCA With Consequential SGTR	R. Riggs	RES/DRA/ARGIB	DROP		06/30/90	NA
142.	Leakage Through Electrical Isolators in Instrumentation Circuits	W. Milstead	RES/DSIR/EIB	NOTE 3(b)	4	12/31/97	NA
143.	Availability of Chilled Water Systems and Room Cooling	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)	2	06/30/95	NA
144.	Scram Without a Turbine/Generator Trip	Hrabal	RES/DSIR/EIB	DROP	2	12/31/98	NA

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Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
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	W. 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	R. Emrit	RES/DSIR/EIB	DROP	2	12/31/98	NA
150.	Overpressurization of Containment Penetrations	W. Milstead	RES/DSIR/SAIB	DROP	1	06/30/95	NA
151.	Reliability of Anticipated Transient Without SCRAM Recirculation Pump Trip in BWRs	W. 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	R. Emrit	RES/DSIR/EIB	DROP	3	06/30/01	NA
153.	Loss of Essential Service Water in LWRs	R. 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.	Generic Concerns Arising from TMI-2 Cleanup	-	-	-			
155.1	More Realistic Source Term Assumptions	R. Emrit	RES/DST/AEB	NOTE 3(a)	2	06/30/95	NA
155.2	Establish Licensing Requirements for Non-Operating Facilities	R. Emrit	RES/DSIR/EIB	RI (NOTE 5)	2	06/30/95	NA
155.3	Improve Design Requirements for Nuclear Facilities	R. Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.4	Improve Criticality Calculations	R. Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.5	More Realistic Severe Reactor Accident Scenario	R. Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.6	Improve Decontamination Regulations	R. Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.7	Improve Decommissioning Regulations	R. Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
156.	Systematic Evaluation Program	-	-	-			
156.1.1	Settlement of Foundations and Buried Equipment	T.Y. Chang	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.1.2	Dam Integrity and Site Flooding	J. Chen	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.1.3	Site Hydrology and Ability to Withstand Floods	J. Chen	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.1.4	Industrial Hazards	C. Ferrell	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.1.5	Tornado Missiles	J. Chen	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.1.6	Turbine Missiles	R. Emrit	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.2.1	Severe Weather Effects on Structures	J. Chen	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.2.2	Design Codes, Criteria, and Load Combinations	R. Kirkwood	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.2.3	Containment Design and Inspection	S. Shaukat	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.2.4	Seismic Design of Structures, Systems, and Components	J. Chen	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.1.1	Shutdown Systems	R. Woods	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.1.2	Electrical Instrumentation and Controls	R. Woods	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.2	Service and Cooling Water Systems	N. Su	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.3	Ventilation Systems	G. Burdick	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.4	Isolation of High and Low Pressure Systems	G. Burdick	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.5	Automatic ECCS Switchover	W. Milstead	RES/DSIR/SAIB	24	7	06/30/01	NA
156.3.6.1	Emergency AC Power	R. Emrit	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.3.6.2	Emergency DC Power	C. Rourk	RES/DSIR/EIB	DROP	7	06/30/01	NA

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Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
156.3.8	Shared Systems	R. Emrit	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.4.1	RPS and ESFS Isolation	R. Emrit	RES/DSIR/EIB	142	7	06/30/01	NA
156.4.2	Testing of the RPS and ESFS	T.Y. Chang	RES/DSIR/SAIB	120	7	06/30/01	NA
156.6.1	Pipe Break Effects on Systems and Components.	J. Page	RES/DET/GSIB	HIGH	7	06/30/01	
157.	Containment Performance	J. Shaperow	RES/DSIR/SAIB	NOTE 3(b)		06/30/95	NA
158.	Performance of Power-Operated Valves Under Design Basis Conditions	C. Hrabal	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
159.	Qualification of Safety-Related Pumps While Running on Minimum Flow	N. Su	RES/DSIR/SAIB	DROP	1	06/30/95	NA
160.	Spurious Actions of Instrumentation Upon Restoration of Power	C. Rourk	RES/DSIR/EIB	DROP	1	06/30/95	NA
161.	Use of Non-Safety-Related Power Supplies in Safety-Related Circuits	C. 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	U. 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	R. Emrit	RES/DSIR/EIB	DROP	1	06/30/95	NA
165.	Safety and Safety/Relief Valve Reliability	C. Hrabal	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
166.	Adequacy of Fatigue Life of Metal Components	R. Emrit	NRR/DE/EMEB	NOTE 3(b)	2	12/31/97	NA
167.	Hydrogen Storage Facility Separation	G. Burdick	RES/DSIR/SAIB	LOW	1	06/30/95	NA
168.	Environmental Qualification of Electrical Equipment	R. Emrit	NRR/DSSA/SPLB	NOTE 3(b)	3	06/30/04	NA
169.	BWR MSIV Common Mode Failure Due to Loss of Accumulator Pressure	R. Emrit	RES/DET/GSIB	DROP	1	06/30/00	NA
170.	Fuel Damage Criteria for High Burnup Fuel	R. Emrit	RES/DET/GSIB	NOTE 3(b)	2	06/30/01	NA
171.	ESF Failure from LOOP Subsequent to a LOCA	C. Rourk	RES/DET/GSIB	NOTE 3(b)	1	12/31/98	NA
172.	Multiple System Responses Program	R. Emrit	RES/DET/GSIB	NOTE 3(b)	2	06/30/02	NA
173.	<u>Spent Fuel Storage Pool</u>	-	-				
173.A	Operating Facilities	R. Emrit	RES/DET/GSIB	NOTE 3(b)	4	06/30/02	NA
173.B	Permanently Shutdown Facilities	R. Emrit	RES/DET/GSIB	NOTE 3(b)	4	06/30/02	NA
174.	<u>Fastener Gaging Practices</u>	-	-				
174.A	SONGS Employees' Concern	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
174.B	Johnson Gage Company Concern	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
175.	Nuclear Power Plant Shift Staffing	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
176.	Loss of Fill-Oil in Rosemount Transmitters	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
177.	Vehicle Intrusion at TMI	R. Emrit	RES/DET/GSIB	NOTE 3(a)	1	06/30/00	NA
178.	Effect of Hurricane Andrew on Turkey Point	R. Emrit	RES/DET/GSIB	LI (NOTE 3)	2	06/30/00	
179.	Core Performance	R. Emrit	RES/DET/GSIB	LI (NOTE 5)	1	06/30/00	
180.	Notice of Enforcement Discretion	R. Emrit	RES/DET/GSIB	LI (NOTE 3)	1	06/30/00	
181.	Fire Protection	R. Emrit	RES/DET/GSIB	LI (NOTE 5)	1	06/30/00	
182.	General Electric Extended Power Uprate	R. Emrit	RES/DET/GSIB	RI (NOTE 5)	1	06/30/00	

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
183.	Cycle-Specific Parameter Limits in Technical Specifications	R. Emrit	RES/DET/GSIB	RI (NOTE 3)	2	06/30/00	
184.	Endangered Species	R. Emrit	RES/DET/GSIB	EI (NOTE 5)	1	06/30/00	
185.	Control of Recriticality Following Small-Break LOCA In PWRs	H. Vandermolen	RES/DSARE/REAHFB	NOTE 3(b)	1	06/30/06	NA
186.	Potential Risk and Consequences of Heavy Load Drops	R. Lloyd	RES/DSARE/REAHFB	CONTINUE		06/30/04	
187.	The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump in Nuclear Power Plants	H. Vandermolen	RES/DSARE/REAHFB	DROP		06/30/01	NA
188.	Steam Generator Tube Leaks/Ruptures Concurrent with Containment Bypass	H. Vandermolen	RES/DSARE/REAHFB	NOTE 3(b)	1	06/30/06	NA
189.	Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Combustion During A Severe Accident	H. Vandermolen	RES/DSARE/REAHFB	CONTINUE		06/30/02	
190.	Fatigue Evaluation of Metal Components for 60-Year Plant Life	S. Shaukat	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
191.	Assessment of Debris Accumulation on PWR Sump Performance	M. Marshall	RES/DET/GSIB	HIGH	1	12/31/98	
192.	Secondary Containment Drawdown Time	H. Vandermolen	RES/DSARE/REAHFB	DROP		06/30/03	NA
193.	BWR ECCS Suction Concerns	H. Vandermolen	RES/DSARE/REAHFB	CONTINUE		06/30/04	
194.	Implications of Updated Probabilistic Seismic Hazard Estimates	D. Harrison	NRR/DSSA/SPSB	DROP		06/30/04	NA
195.	Hydrogen Combustion in Foreign BWR Piping	H. Vandermolen	RES/DSARE/REAHFB	DROP		06/30/04	NA
196.	Boral Degradation	H. Vandermolen	RES/DSARE/ARREB	NOTE 3(b)	1	06/30/07	NA
197.	Iodine Spiking Phenomena	H. Vandermolen	RES/DSARE/ARREB	DROP		06/30/06	NA
198.	Hydrogen Combustion in PWR Piping	H. Vandermolen	RES/DRASP/OERA	DROP		06/30/07	NA
199.	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States	T. Mitts	RES/DRASP/OERA	NOTE 4		(Later)	
200.	Tin Whiskers	C. Antonescu	RES/DRASP/OERA	DROP		06/30/07	NA
201.	Small-Break LOCA and Loss of Offsite Power Scenario	A. Salomon	RES/DRASP/OERA	DROP		06/30/07	NA
202.	Spent Fuel Pool Leakage Limits	T. Mitts	RES/DRASP/OERA	DROP		06/30/07	NA
203.	Potential Safety Issues with Cranes that Lift Spent Fuel Casks	T. Mitts	RES/DRASP/OERA	DROP		06/30/07	NA

HUMAN FACTORS ISSUES

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
HF1	STAFFING AND QUALIFICATIONS						
HF1.1	Shift Staffing	J. Pittman	RES/DRPS/RHFB	NOTE 3(a)	2	06/30/89	
HF1.2	Engineering Expertise on Shift	J. Pittman	NRR/DHFT/HFIB	NOTE 3(b)	2	06/30/89	NA
HF1.3	Guidance on Limits and Conditions of Shift Work	J. Pittman	NRR/DHFT/HFIB	NOTE 3(b)	2	06/30/89	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
HF2	TRAINING						
HF2.1	Evaluate Industry Training	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF2.2	Evaluate INPO Accreditation	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF2.3	Revise SRP Section 13.2	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF3	OPERATOR LICENSING EXAMINATIONS						
HF3.1	Develop Job Knowledge Catalog	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF3.2	Develop License Examination Handbook	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF3.3	Develop Criteria for Nuclear Power Plant Simulators	J. Pittman	NRR/DHFT/HFIB	I.A.4.2(4)	2	12/31/87	NA
HF3.4	Examination Requirements	J. Pittman	NRR/DHFT/HFIB	I.A.2.6(1)	2	12/31/87	NA
HF3.5	Develop Computerized Exam System	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF4	PROCEDURES						
HF4.1	Inspection Procedure for Upgraded Emergency Operating Procedures	J. Pittman	NRR/DLPQ/LHFB	NOTE 3(b)	6	06/30/95	NA
HF4.2	Procedures Generation Package Effectiveness Evaluation	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	6	06/30/95	NA
HF4.3	Criteria for Safety-Related Operator Actions	J. Pittman	NRR/DHFT/HFIB	B-17	6	06/30/95	NA
HF4.4	Guidelines for Upgrading Other Procedures	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	6	06/30/95	NA
HF4.5	Application of Automation and Artificial Intelligence	J. Pittman	NRR/DHFT/HFIB	HF5.2	6	06/30/95	NA
HF5	MAN-MACHINE INTERFACE						
HF5.1	Local Control Stations	J. 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	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	4	06/30/95	NA
HF5.3	Evaluation of Operational Aid Systems	J. Pittman	NRR/DHFT/HFIB	HF5.2	4	06/30/95	NA
HF5.4	Computers and Computer Displays	J. Pittman	NRR/DHFT/HFIB	HF5.2	4	06/30/95	NA
HF6	MANAGEMENT AND ORGANIZATION						
HF6.1	Develop Regulatory Position on Management and Organization	J. 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	J. Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF7	HUMAN RELIABILITY						

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
HF7.1	Human Error Data Acquisition	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.2	Human Error Data Storage and Retrieval	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.3	Reliability Evaluation Specialist Aids	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.4	Safety Event Analysis Results Applications	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF8	Maintenance and Surveillance Program	J. 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	R. Emrit	NRR/DLPQ/LHFB	LI (NOTE 5)		06/30/89	NA
CH1.1B	Procedure Violations	R. 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	R. Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.2B	NRC Testing Requirements	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.3	Bypassing Safety Systems	-	-				
CH1.3A	Revise Regulatory Guide 1.47	R. Emrit	RES/DE/EMEB	LI (NOTE 5)		06/30/89	NA
CH1.4	Availability of Engineered Safety Features	-	-				
CH1.4A	Engineered Safety Feature Availability	R. Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.4B	Technical Specifications Bases	R. Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.4C	Low Power and Shutdown	R. Emrit	RES/DSR/PRAB	LI (NOTE 5)		06/30/89	NA
CH1.5	Operating Staff Attitudes Toward Safety	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH1.6	Management Systems	-	-				
CH1.6A	Assessment of NRC Requirements on Management	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.7	Accident Management	-	-				
CH1.7A	Accident Management	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH2	DESIGN						
CH2.1	Reactivity Accidents	-	-				
CH2.1A	Reactivity Transients	R. Emrit	RES/DSR/RPSB	LI (NOTE 5)		06/30/89	NA
CH2.2	Accidents at Low Power and at Zero Power	R. Emrit	RES/DRA/ARGIB	CH1.4		06/30/89	NA
CH2.3	Miltiple-Unit Protection	-	-				
CH2.3A	Control Room Habitability	R. Emrit	RES/DRA/ARGIB	83		06/30/89	NA
CH2.3B	Contamination Outside Control Room	R. Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.3C	Smoke Control	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH2.3D	Shared Shutdown Systems	R. Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.4	Fire Protection	-	-				
CH2.4A	Firefighting With Radiation Present	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
CH3	CONTAINMENT						
CH3.1	Containment Performance During Severe Accidents	-	-				
CH3.1A	Containment Performance	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH3.2	Filtered Venting	-	-				
CH3.2A	Filtered Venting	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4	EMERGENCY PLANNING						
CH4.1	Size of the Emergency Planning Zones	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH4.2	Medical Services	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH4.3	Ingestion Pathway Measures	-	-				
CH4.3A	Ingestion Pathway Protective Measures	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4.4	Decontamination and Relocation	-	-				
CH4.4A	Decontamination	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4.4B	Relocation	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH5	SEVERE ACCIDENT PHENOMENA						
CH5.1	Source Term	-	-				
CH5.1A	Mechanical Dispersal in Fission Product Release	R. Emrit	RES/DSR/AEB	LI (NOTE 5)		06/30/89	NA
CH5.1B	Stripping in Fission Product Release	R. Emrit	RES/DSR/AEB	LI (NOTE 5)		06/30/89	NA
CH5.2	Steam Explosions	-	-				
CH5.2A	Steam Explosions	R. Emrit	RES/DSR/AEB	LI (NOTE 5)		06/30/89	NA
CH5.3	Combustible Gas	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH6	GRAPHITE-MODERATED REACTORS						
CH6.1	Graphite-Moderated Reactors	-	-				
CH6.1A	The Fort St. Vrain Reactor and the Modular HTGR	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH6.1B	Structural Graphite Experiments	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH6.2	Assessment	R. 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
Continue	- As defined in NRC Management Directive 6.4[1858]

TABLE III (Continued)

ACTION ITEM/ISSUE GROUP	I	S	RESOLVED STAGES			USI	HIGH	MEDIUM	LOW	DROP	CONT.	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	-	-	-	70
RI	-	-	-	-	6	-	-	-	-	-	-	-	1	7
LI	-	-	-	-	11	-	-	-	-	-	-	-	12	23
EI	-	-	-	-	13	-	-	-	-	-	-	-	2	15
NEW GENERIC ISSUES (283)														
GSI	-	54	0	0	87	0	4	0	4	105	2	1	-	257
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	421	0	4	0	16	129	2	1	64	853

ISSUE 196: BORAL DEGRADATION

DESCRIPTION

Historical Background

This issue was raised[1854, 1855] by an NMSS staff member whose concern arose from the observation that spent fuel pool racks using Boral for neutron absorption had experienced some problems with swelling and degradation of the Boral plates over long periods of time. Because the Boral material is commonly used for neutron absorption and shielding in a wide spectrum of applications, this degradation may have a number of implications for safety purposes. However, this issue specifically addresses the use of Boral in long-term dry storage casks for spent reactor fuel.

Safety Significance

Composition of Boral: "Boral" is the trade name for a product of AAR Advanced Structures of Livonia, Michigan and is a neutron absorber plate material which uses boron carbide for neutron absorption. One isotope of boron, specifically boron-10 (or ^{10}B), has a thermal neutron absorption cross section of over 3800 barns. Moreover, the nuclear reaction is unusual in that the resulting compound nucleus emits an alpha particle rather than de-exciting by gamma emission. Thus, unlike many other high cross section nuclides, boron-10 does not emit high energy secondary gammas, which (along with the high cross section) makes it an excellent shielding material. Consequently, Boral plate is widely used in many industrial, medical, and laboratory applications.

Boral is made by mixing boron carbide granules and aluminum powder inside an aluminum box, heating the box and its content to form an ingot, and then hot-rolling the ingot to form a plate consisting of a coarse core of B_4C-Al composite material bonded between two thin sheets of aluminum cladding.

Experience with Boral: One of the uses of Boral is in spent fuel pool racks. When the first generation of nuclear reactors was built, spent fuel racks ensured subcriticality by using rack designs which kept the spent fuel assemblies widely separated. (The thermal neutron absorption cross-section of the hydrogen in the water was sufficient to keep the array subcritical. In PWR pools, boric acid is dissolved in the water as well.) However, as more and more spent fuel was discharged into these pools, it was necessary to install new racks which held the spent fuel assemblies in a much more compact array. To ensure subcriticality, the new spent fuel racks incorporated Boral sheets between the fuel assemblies.

The Boral sheets were sandwiched (clad) within seal-welded stainless steel cover plates, apparently to keep water from contacting the Boral. Nevertheless, there were several instances (dating back to 1983) where the stainless steel cover plates experienced bulging, to the point where mechanical interference with the fuel assemblies became a problem. It was discovered upon investigation that there had been water ingress into the stainless steel sandwich, and the aluminum in the Boral had reacted chemically with the water to produce hydrogen gas and aluminum oxide. The hydrogen gas pressure had built up to the point where the stainless steel cladding bulged.

One fix was to clip the corners of the stainless steel cladding, allowing the hydrogen gas to escape. However, in follow-up investigations, it was found in some cases that the corrosion reactions resulted in a partial debonding of the Boral's aluminum cladding from the composite core absorber material, with some limited losses of B_4C granules and aluminum binder from the edges of the Boral plates.

A similar occurrence was discovered at an early generation BWR, where the spent fuel had been stored in the spent fuel pool since 1985. This plant had installed Boral cans around each fuel assembly, but without stainless steel cladding. It was discovered that there were blisters on about 5% of the Boral cans. The blisters were generally one inch in diameter or less, and tended to occur near the edge of the Boral sheet, where the internal composite core was in contact with the pool water.

Long-Term Dry Spent Fuel Storage Casks: Calculations showed that the observed B_4C losses did not result in a significant loss of shutdown margin in spent fuel pools, and that is not the focus of this generic issue. Instead, the question involves the situation with long term dry spent fuel storage casks. To understand the safety significance, it is necessary to first review the cask design and intended use. For the purposes of this screening analysis, the Holtec[1856] design was used. (Other designs exist, with somewhat different capacities, etc., but the designs are rather similar, and the differences should not affect any conclusions.)

These casks are intended for spent fuel which has been out of the reactor and in the spent fuel pool for a long time. As time goes on, the inventories of the various radioactive species in a spent fuel assembly will decrease in accordance with the half-life of each nuclide. Decay heat production will slowly diminish, and after several years, will be low enough that liquid coolant is no longer necessary to keep the spent fuel from overheating. (It is still necessary for shielding, however.) The Holtec HI-STORM design uses a multipurpose cannister (MPC) which can hold many fuel assemblies. (The MPC-24 will hold 24 PWR fuel assemblies; the MPC-68 will hold 68 BWR fuel assemblies.) Each MPC design consists of a sealed metallic canister, and the external dimensions are the same regardless of the intended contents. Once loaded, the MPC is backfilled with helium and sealed. The MPC can be placed into either a HI-TRAC transfer cask or a HI-STORM storage "overpack." The transfer cask uses lead and a water jacket for shielding, whereas the storage overpack is intended for long-term storage, and uses plain concrete for shielding. Cooling is passive; the large storage overpack incorporates air ducts for natural convection cooling.

The MPC designs include "baskets" to hold fuel assemblies, heat conduction elements that help transfer heat to the MPC shell, and Boral sheets between the baskets to provide reactivity control. Criticality is a concern; these MPCs hold about one-third of a full core for a small reactor such as Yankee Rowe.

The Boral sheets are necessary when the MPC is being loaded with spent fuel. Once the MPC is removed from the spent fuel pool, drained, and seal-welded, the lack of water as a moderator makes criticality unlikely. The MPC can reside within the HI-STORM storage overpack for many years.

Safety Concern: The HI-STORM dry storage system is designed for long term storage, but is not intended to be a permanent repository. Eventually, these MPC units will be transferred to transfer casks and shipped to a permanent repository. The criticality concern affects any MPC units that, years later, must be reopened for repairs of any kind. One scheme for doing so is to re-immerse the MPC in water for shielding, and perform the repair operations under water. Water immersion has several advantages, including shielding, a lower working temperature, a transparent medium, and some limiting of the spread of any contamination.

However, if the MPC is re-flooded with water, the Boral sheets again become necessary to ensure a subcritical configuration. When the MPC was first loaded with fuel, these Boral sheets will have been soaked in water, with some water ingress into the coarse B_4C-Al composite material within the aluminum cladding. (The edges of the sheet are not sealed; the composite material, which is porous, will be exposed to the water.) The vacuum drying is likely to leave some residual water within the composite core. During long-term storage, these sheets will then be subjected to temperatures on the order of 500°F for many years. This is a more severe environment than that experienced by Boral sheets immersed in the spent fuel pool, where blistering has been observed after several years in warm water. It is likely that steam blisters will form in the short term, and possibly hydrogen blisters in the long term.

Thus, if there is any problem with the integrity of the Boral sheet, it is possible that, under such conditions, the material may crumble or otherwise relocate in storage, or may be physically damaged when "quenched" by reflooding of the MPC. Moreover, the blistering will displace some of the water, which will affect reactivity somewhat even if the B_4C-Al composite material does not relocate.

It is possible to form a critical array with sixteen to twenty fresh BWR fuel assemblies in cold clean water (NUREG-75/110, pp. 4-14).[1857] PWR assemblies, which are generally equivalent to four BWR assemblies each, would be expected to approach criticality with a commensurately smaller number. Of course, the fuel stored within the MPC will be, except in a few cases, fuel with significant burnup. Nevertheless, this spent fuel was still capable of producing some power (with equilibrium xenon and at reactor temperatures) before it was discharged. (Equilibrium xenon is typically worth 2.5% to 3% $\Delta K/K$, and the moderator and Doppler defects are generally worth 3% to 4% in addition.)

Thus, although this spent fuel is not likely to achieve high power levels, it is quite credible that reflooding an MPC unit will result in an inadvertent criticality if the Boral neutron absorber is not present. Such an event might not damage the fuel cladding, but it would certainly produce high neutron and fission gamma radiation fields, which can be quite hazardous to personnel unless adequate shielding is in place.

There are two other aspects to such an inadvertent criticality event. First, there will not be any "scram" system or similar safety system available to rapidly insert negative reactivity, and it may not be immediately obvious to the personnel what should be done to terminate the event.

Second, the existing neutron flux from transuranics in the spent fuel may not be high enough to ensure a controlled startup. This can lead to a classic criticality accident, where a critical configuration is achieved, but nothing happens because there are not enough neutrons to start the chain reaction. Then, as the evolution continues, the configuration might be significantly supercritical before the reaction starts, and when it does start, neutron flux will escalate with a very fast period, leading to a very hazardous situation.

Possible Solution

The proposed solution for this generic issue is in two steps. The first step would be to test samples of Boral under conditions duplicating the environmental conditions that would be experienced in these MPC units. This experiment can be done quite readily, and at a modest cost. If there is no evidence for crumbling or relocation of the B_4C-Al composite material, the issue would be considered resolved.

However, if the experimental evidence indicates that relocation of the B_4C-Al composite material is credible, the second step would be to ensure that these MPC units either are repaired under dry conditions, or that the water used in submerged operations contain a soluble neutron absorber such as boric acid (or some other means be used for reactivity control).

Alternatively, it is the staff's understanding that the manufacturer has been conducting research to find ways to improve the performance of Boral. This also could resolve the issue.

SCREENING ANALYSIS

Criteria: The usual criteria for screening generic issues, specifically core damage frequency, large early release frequency, and person-rem per reactor-year, are not applicable to this issue. However, there are some statements in the regulations that address accidental criticalities.

10CFR72, Part 72.124, "Criteria for nuclear criticality safety," states, in part:

> (a) *Design for criticality safety.* Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety.

10CFR72, Part 72.236, "Specific requirements for spent fuel storage cask approval and fabrication," goes on to say:

> (c) The spent fuel storage cask must be designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions.

It also states:

> (h) The spent fuel storage cask must be compatible with wet or dry spent fuel loading and unloading facilities.

The essence of these statements is that an accidental criticality is not to occur, i.e., the probability of such an event should be low. For the purpose of this screening analysis, and in the absence of any formal guidance, a probability of 0.1 of an inadvertent criticality event will be used as a screening threshold.

Frequency Estimate

It is not possible to perform a probabilistic analysis in the usual sense, since there is no data upon which to base the analysis. Nevertheless, it is possible to have some qualitative probabilistic insights into the likelihood of such an event.

The central question of this generic issue is, will there be extensive blistering and degradation of the Boral sheets? If the blistering is extensive, it is quite likely that, in at least some of the MPCs, there will be an axial location where there is insufficient boron absorption to maintain a subcritical condition. It will be assumed that at least one MPC in 100 will be damaged to the extent that an inadvertent criticality is possible if the MPC is refilled with water. This is, of course, an educated guess. However, if blistering is this extensive, it should be detectable by a few experiments.

At the time of writing of this analysis, there were approximately 117 reactors with spent fuel pools, twelve of which were no longer operating. A simple tabulation was used to estimate the number of MPC units needed for this entire population, assuming a 40-year lifetime (or the actual lifetime for the shutdown units), a 1.5 year fuel cycle, a third core replacement for each fuel cycle, and the Holtec design of 68 BWR fuel assemblies per MPC, or 24 PWR assemblies per MPC. The result was an estimate of 8,783 MPC units. Assuming a 20-year license extension for the units that were not already shut down, this number increased to 12,850 MPC units.

Based on this, it was reasonable to assume that the total number of MPCs needed to accommodate the present generation of power reactors was on the order of 10^4 units.

For an inadvertent criticality event to occur, two events must happen. First, the Boral sheets in an MPC unit must be damaged to the point where criticality becomes possible. Second, this same MPC unit must be flooded with water for repair.

Based purely on engineering judgment, it was assumed that less than 1% of the MPCs will need to be opened and reflooded for any reason. However, as a practical matter, it was doubtful that this number of MPCs which must be opened would be less that one per thousand, based on general engineering experience. Therefore, it was assumed that the number opened would be at least one per thousand, the lower limit.

Putting these figures together, with a population of 10,000 MPC units, and assumptions that at least one unit per thousand would be opened underwater for repair (or any other reason), and at least one per hundred will be damaged to the point where criticality is possible, the total number of criticality incidents will be at least 0.1. Thus, the probability of at least one criticality was given by the Poisson formula:

$$P(n \geq 1) = 1 - e^{-x}$$

where x is the expected number of criticalities. If x, the expectation value, is 0.1 or greater, the probability of at least one event is 0.095 (essentially equal to the expectation value) or greater. Thus, under these assumptions, this generic issue meets the screening criterion described earlier.

Other Considerations

The semi-quantitative estimate developed above assumes that the likelihood of opening the MPC underwater, and the likelihood of damage such that criticality is possible, are independent. This may not be completely true. For example, if an MPC unit were involved in a transportation accident of any kind, the robustness of the MPC and its transfer cask would preclude any release of radioactive material to the surroundings. However, the physical assault might cause relocation of the B_4C-Al composite material, and also increase the likelihood of the MPC being opened for inspection. This would tend to increase the probability of an accidental criticality above that estimated by the assumption of randomness above.

CONCLUSION

Based on the likelihood of an accidental criticality described above, and on the relatively modest resources needed for resolution, technical assessment of the issue was pursued.[1875] The issue was later RESOLVED with no new requirements or guidance for licensees.[1876]

REFERENCES

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1856. "Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM Cask System)," Docket 72-1014, July 19, 2000. [ML003777866]

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ISSUE 198: HYDROGEN COMBUSTION IN PWR PIPING

DESCRIPTION

Historical Background

This issue was initiated[1833] in response to a recommendation in the screening analysis of Generic Issue (G)I-195, "Potential Hydrogen Combustion in BWR Piping." GI-195 investigated the potential safety significance of hydrogen combustion events that had actually occurred in BWR primary system piping. Although no similar events had been observed in PWRs, it was felt that the potential safety significance of such events in PWRs should also be investigated.

Safety Significance

In any water-cooled reactor, radiolysis of the water in the reactor core is always producing some elemental hydrogen and oxygen. In the BWR experience described in GI-195, these gases, because of their buoyancy in a steam atmosphere, tended to build up in high, stagnant points in the primary system over a period of time. This hydrogen-oxygen mixture, although somewhat diluted by steam, is also at reactor operating conditions, i.e., at a pressure of about 1000 psi and a temperature near 500°F. At these conditions, relatively little energy is required to ignite the mixture. There were three events where a combustible mixture built up in piping connected to the primary system and detonated. Although the gas buildup in all three cases was in a pipe volume isolated from the primary system by a check valve, the force of the detonations was sufficient to rupture the piping. In three other events, a combustible mixture accumulated in the top works of a safety/relief valve. When the mixture ignited, mechanical damage caused the valves to fail open and blow down the primary system.

The screening analysis of GI-195 concluded that the frequency of such events was sufficiently low that, given the many diverse systems available to mitigate such an event in BWRs, there was insufficient safety significance to justify regulatory action on BWRs. However, the review panel noted that, although a search of PWR operational history found no such events, PWRs also do not have as many ways of mitigating a loss of coolant accident. Because of this, the panel recommended that the hydrogen combustion phenomenon also be investigated for PWRs.

SCREENING ANALYSIS

Although there have been hydrogen fires at PWR plants, particularly during maintenance activities, no events which occurred in the primary system at power have been reported. Therefore, some deterministic methods must be used to estimate where such events are likely, and how frequently they might occur.

The PWR primary system is a closed system. In order to limit corrosion, it is common practice to operate with an excess of dissolved hydrogen in the primary coolant, which has the effect of scavenging the oxygen produced by radiolysis. This is accomplished by providing a hydrogen cover gas in the volume control tank in the Chemical and Volume Control System (CVCS), which supplies charging fluid to the primary system. Normally, dissolved oxygen is maintained below 0.1 ppm, and dissolved hydrogen will be in the range of 25 to 35 cc (at STP) per kilogram of water, which is equivalent to a weight percentage of 2.2 to 3.1 ppm.

Deflagration or detonation of a hydrogen-oxygen mixture could occur if a gas bubble formed in the primary system and was trapped in a stagnant volume in the upper portion of the system (e.g., the control rod travel housings, the reactor vessel head vent, or any other upward-leading pipe that normally carries no flow). Alternatively, the gases could accumulate in the pressurizer steam space. At pressures around 2000 psi and temperatures approaching 600°F, relatively little energy would be needed for ignition.

Phenomenology

A review of the physical phenomenology of gas solubility and bubble formation is helpful here. A more complete description can be found in NUREG/CR-2726.[1872] Consider a volume containing liquid water and hydrogen gas, in equilibrium with pressure P and temperature T. There will be some hydrogen gas dissolved in the liquid phase in the lower portion of the volume, and some water vapor in the gas phase above.

According to Henry's Law, the solubility of hydrogen is proportional to the partial pressure $P(H_2)$ in the cover gas.

$$P(H_2) = H(T)X(H_2)$$

Where

Where	$P(H_2)$	=	partial pressure of hydrogen
	$H(T)$	=	Henry's Law constant
	$X(H_2)$	=	mole fraction of hydrogen in the liquid phase.

The total pressure is the sum of the partial pressure of hydrogen and the partial pressure of the water vapor, which is just the saturation pressure for water at temperature T.

$$P = P(H_2) + P_{Sat}(T)$$

For example, consider the conditions at the top of a PWR reactor vessel, above the core. From the PWR training manual,

P	=	2235 psig = 2250 psia
T	=	610.7°F = 321.5°C = 594.66°K
H(T)	=	1.25E5 psia/mole fraction[1872]

At this temperature, the saturation pressure of water is

P(sat) = 1667.8 psia

The partial pressure of the hydrogen is the total pressure minus this saturation pressure:

$P(H_2)$ = 2250 - 1667.8 = 582.2 psia

Then, using Henry's Law,

$$X(H_2) = \frac{P(H_2)}{H(T)}$$

Working this out, $X(H_2)$ = 582.2 psia/1.25E5 psia/mole fraction = 4.66E-4 mole fraction, which works out to about 515 ppm by weight.

This is the equilibrium concentration. If more hydrogen is added to form a higher concentration, the solution would be supersaturated, and the excess hydrogen would bubble out in the gas phase. Conversely, if the hydrogen concentration in the liquid water were below 515 ppm, the gas bubble would dissolve into the liquid. Since the normal reactor chemistry maintains the hydrogen concentration below 3.1 ppm (about 0.6% of the saturation value), gas bubbles would not be expected to form in this area.

Similarly, the equilibrium concentration for oxygen is about 3560 ppm, and the normal concentration in the coolant is about 0.1 ppm *(D. M. Himmelblau, "Solubilities of Inert Gases in Water," Journal of Chemical & Engineering Data, 5:10-15, 1960)*. Clearly, oxygen is even less likely to come out of solution and form a bubble, since its equilibrium solubility is higher, and its normal concentration is lower. Similar calculations for various locations in the primary system result in the following equilibrium concentrations:

Location	Pressure (psia)	Temperature, (F)	Equilibrium hydrogen concentration (ppm)	Equilibrium oxygen concentration (ppm)
Reactor vessel head, nominal conditions	2250	610.7	515	3560
RCS cold legs & vessel downcomer	2250	545.7	323	2150
Control rod drive travel housings	2250	248	340	1980
Pressurizer	2250	653	essentially zero	essentially zero
Volume Control Tank (one atm. H_2 overpressure)	30	115	3.5	—

Again, given that the normal hydrogen concentration is 2.2 to 3.1 ppm, and normal oxygen concentration is 0.1 ppm, gas bubbles are not expected to form in the normally liquid-filled portions of the primary system. If such bubbles were somehow introduced, they would be expected to dissolve into the liquid coolant over a period of time. Therefore, collection and detonation of these gases in the liquid-filled portion of the primary system do not appear to be credible.

The exception is the pressurizer steam space. In theory, if the pressure were exactly equal to the saturation pressure for water at that temperature, the partial pressure of hydrogen (and oxygen) would be zero and any dissolved gas would be driven out of solution until some partial pressure of hydrogen and oxygen existed in the steam space. For 3.1 ppm, this translates to a partial pressure of 2.27 psia of hydrogen in the pressurizer steam space, if the gas were evenly dispersed. For a nominal steam volume of 720 cubic feet, this is equivalent to 62 moles of hydrogen gas, which, at 2250 psia and 653°F, would occupy a bubble approximately 14 inches in diameter if collected into one volume.

Similarly, if the oxygen concentration in the liquid coolant is 0.1 ppm, the partial pressure at equilibrium conditions would be 0.00647 psia, implying the presence of slightly less than six grams of (uniformly dispersed) oxygen in the steam space. If gathered together in one volume, this is equivalent to a bubble approximately two inches in diameter.

However, the hydrogen and oxygen are less dense than steam at the same temperature. The densities work out as follows:

Hydrogen gas	0.38	lb/ft^3.	(Based on ideal gas law)
Oxygen gas	6.02	lb/ft^3.	(Based on ideal gas law)
Saturated steam	6.373	lb/ft^3.	(Based on steam tables)

Depending on the degree of stagnation in the pressurizer steam space, these gases, because of their buoyancy, will tend to collect at the top of the pressurizer steam space, rather than being evenly distributed. If the mixture of gases is quiescent, with no turbulence or convection currents, the hydrogen will rise to the top. The oxygen is less buoyant, but will still tend to rise and form a layer at the interface between the hydrogen and the steam. This will tend to reduce the partial pressure of both gases at the liquid surface, causing more gases to come out of solution. Because more dissolved hydrogen (and oxygen) is continuously being introduced into the primary coolant, the collection of a significant amount of these gases at the top of the pressurizer steam space does appear to be credible. Moreover, because of the tendency to stratify, there will be a locus somewhere in the hydrogen/oxygen interface where the ratio of the two gases permits deflagration.

Conversely, if this volume has enough circulation to remain well mixed, the hydrogen and oxygen will be highly diluted in steam, and are unlikely to form a combustible mixture in this volume. However, even if this volume is well-mixed, any connected piping can still collect pockets of hydrogen and oxygen if this piping leads in an upward direction and is slightly cooler than the main pressurizer steam space. The two combustible gases will tend to rise and collect in the upper portions of the piping, and the steam will tend to condense on the pipe walls and run back into the main volume.

In reality, the pressurizer steam space is likely to have some convection currents whenever the heaters are in operation. Also, normally a small amount of flow is maintained in the spray line to keep the boron concentration in the liquid-filled pressurizer volume equal to that in the rest of the reactor coolant system. Thus, there is likely to be some mixing, such that the hydrogen and oxygen are unlikely to be present in well-defined, horizontal layers, but the presence of a combustible mixture, particularly in attached piping, cannot be ruled out.

The top of the pressurizer does have some locations where collection and ignition of a hydrogen-oxygen mixture could pose a problem. For a typical Westinghouse design, these include:

- The top of the head itself. However, detonation of a gas stratum here would not be likely to do much damage, since the gas volume would not be confined.

- A bolted personnel access hatch ("manway"). Again, detonation of a gas stratum here would not be likely to do much damage, since the gas volume would not be confined.

- The pressurizer spray nozzles (supplied by a four-inch line), which are located at the top of the upper head in the Westinghouse design. However, normal operational practice is to maintain a small flow through these lines all the time, in order to ensure a uniform boron concentration throughout the primary system. Thus, the line is normally liquid-filled, which greatly reduces the likelihood of noncondensible gases accumulating in this volume.

- Safety Valves: These are connected via 6-inch pipes, but the pipes are shaped in the form of a loop seal, keeping liquid in contact with the valve seat. (In contrast to the case of BWRs, catalysis by the plating in the valve internals is not possible.) The only volume which can accumulate gases is in the pipe upstream of the loop seal. A rupture of one of these lines would be an "intermediate" (i.e., up to six-inch equivalent diameter) LOCA.

- PORV Lines (Including Block Valves): The PWR training manual lists this as a six-inch line. However, the PORV capacity varies significantly from plant to plant, and presumably this pipe size can vary as well. However, a six-inch equivalent diameter LOCA should bound any PORV line break for any plant.

- Various instrument taps and small pipes, such as the level instrumentation, sampling lines, valve leakoffs, etc. A rupture of one of these lines would be a small break LOCA.

The design specifics can vary from product line to product line, and from reactor vendor to reactor vendor. For example, some B&W pressurizers have a vent valve at the very top, and the spray line penetration is off to the side, with a pipe running inside the steam space to the spray nozzle, which is located just below the center of the upper head. Nevertheless, all designs will have similar potential break locations.

<u>Initiating Event Frequency Estimate</u>

According to a private communication from the IAEA, PWR experience is approximately 6280 reactor-years at the time of this writing. An examination of the IAEA database has not found any instances of hydrogen deflagration in the primary system of a PWR.

In contrast to the PWR experience, there have been six such events in approximately 2325 BWR-years, as was discussed in GI-195. Thus, the credibility of such an event in a PWR is based on the experience of BWRs. There has been no PWR event which would indicate the possibility of such an event in a PWR. Clearly, the frequency of such an event in a PWR must be much less than the frequency of occurrence in BWRs, given that no PWR events have occurred, even though the accumulated experience of PWRs is almost triple that of BWRs. It will be necessary to make some assumptions regarding the combustion phenomena in order to extrapolate the BWR experience to PWRs.

One candidate explanation for the difference of event frequencies would be the PWR practice of intentionally operating with an excess of hydrogen in the coolant, with the express purpose of reducing the oxygen concentration. However, many BWRs use an analogous practice, adding hydrogen or noble metals to the feedwater and thereby reducing oxygen concentrations to as low as 0.005 ppm - actually less than oxygen concentrations in PWRs. This technique was being used in the Japanese plant that experienced a rupture of a steam condensation line. (See GI-195.) Thus, the PWR hydrogen strategy does not appear to provide a viable explanation for the difference in frequencies.

There are three other major differences between the steam spaces in a BWR and in a PWR. First, the PWR pressurizer steam space operates at double the pressure of a BWR, and consequently is about 100°F hotter. It is not obvious how this would affect the likelihood of a deflagration of hydrogen and oxygen. However, a higher temperature and pressure normally increase rather than decrease the likelihood of a spontaneous ignition. Thus, this difference also does not appear to provide a viable explanation for the difference in frequencies.

Second, the steam volume in a PWR pressurizer is fairly small - about 720 cubic feet. There is no readily-available data on the steam-filled volume in a boiling water reactor, but a rough calculation based on vessel dimensions, and including the steam lines, gave an estimate of approximately

7960 cubic feet - about eleven times that of a PWR. It does appear likely that more steam volume implies a higher likelihood of a combustion event. (Certainly, reducing the volume to zero would reduce the combustion probability to zero.)

Third, a PWR pressurizer does not have as many valve-isolated volumes connected to it. It should be noted that in all three BWR events that resulted in a pipe rupture, the rupture occurred in a pipe volume which was isolated by a check valve or isolation valve from the primary system. Although this (fortunately) meant that the events involved no significant coolant inventory loss, it also raises a question as to whether the pipes would have ruptured if the volume had not been confined. (Even a check valve, which would normally have relieved pressure by passing inventory back into the primary coolant system, might provide enough resistance to allow a deflagration event to transition into a detonation event.)

In the remaining three BWR events, combustible gases collected in the top works of a safety/relief valve. In these events, there was no piping rupture, but the safety/relief valve failed open. Such an event is less likely in a PWR, because of the use of loop seals, and because PWRs generally use spring safety valves which do not have the same top works as a BWR S/RV. It would be possible for a pressurizer PORV to be damaged and fail open by this mechanism, but the block valve would be available to stop the inventory loss.

Thus, the lower number of isolated volumes connected to a PWR steam space would also help explain the lower frequency of observed combustion events in PWRs.

In the absence of any better information, it will be assumed that the frequency of a combustion event in a PWR can be estimated from the BWR event frequency in linear proportion to the steam volumes of the two designs. For this purpose, the BWR safety/relief valve openings will not be included, since the analogous events in a PWR are rendered far less likely by the loop seals and the valve internals design. The BWR isolated-pipe-rupture event frequency used in the screening analysis of GI-195, based on three events in 2325 BWR-years, was 1.3E-3 event/ BWR-year, normally distributed with a standard deviation of 7.5E-5/BWR-year. Ratioing this by the steam space volumes gives an estimated PWR combustion event frequency of 1.17E-4 event/PWR-year, with a standard deviation of 6.8E-5/PWR-year.

A combustion event such as this does not necessarily lead to a loss of coolant, as the BWR experience shows. The screening analysis for GI-195 used an exponential distribution (with a mean of 0.21) to estimate the likelihood of a non-isolatable pipe rupture, given a combustion event. It should be noted that this is not a distribution with a peak at 0.21. Instead, this distribution has a maximum value at zero, a mean of 21%, and a tail that is "chopped off" with a value of 0.04 at unity.

In the absence of any better information, this same exponential distribution will be used in this analysis. However, it should be noted that, in the BWR screening analysis, this distribution was primarily a geometrical parameter used to describe the likelihood that a pipe rupture would be in a location not isolatable, i.e., the complementary likelihood would be the likelihood of a rupture in an isolatable location. In the present use for a PWR scenario, this is the likelihood that a combustion event will result in coolant inventory loss. The complementary likelihood is that of a combustion event that causes no significant damage.

Before going further, an examination of the reasonableness of these numbers is in order. Given that there are currently 6280 PWR-years of experience worldwide, and no observed events, it is reasonable to expect frequencies that are less than 1/6280, or about 1.6E-4. However, estimates that are several orders of magnitude less than this upper limit would need considerable justification.

The mean PWR combustion event frequency distribution above is 1.17E-4/PWR-year, estimated from the BWR experience. This is not much less than what would be estimated if an actual event had been observed in the existing 6280 PWR-years of experience, which would be about 1.59 event/PWR-year. Thus, the estimate extrapolated from BWR experience is not highly non-conservative.

However, it could be argued that PWR combustion events might already have occurred but not have been noticed. To address this, a numerical calculation of the frequency of pipe ruptures was performed, using the normal combustion event frequency distribution and the exponential pipe rupture probability distribution discussed above. The results were a mean estimated pipe rupture frequency of 2.47E-5 event/PWR-year. (The number of significant figures does not imply accuracy to this degree, but instead are provided as an aid in following the calculation.) This is about 15% of the value corresponding to one actual event in the existing 6280 PWR-years of experience. Again, this appears to be reasonable.

Discussion

Again, there are two modeling assumptions in the approach to initiating event frequency above: that the PWR combustion event frequency can be extrapolated from the observed BWR frequency in proportion to steam space volumes, and that the likelihood that such a combustion event will result in a breach of the reactor coolant pressure boundary can be estimated by exponential assumption used in GI-195. The numerical effect of both of these assumptions will be explored in the section on "core damage frequency" below.

Both of these assumptions are being made because, if there were no BWR events, there would be no experience indicating that such an event would be expected in a PWR. There have been no recorded events of this nature in 6280 PWR-years of experience, worldwide. One other approach would be to focus only on the PWR experience and see if a bounding value can be inferred. This can be done by assuming an exponential distribution for the initiating event frequency, and choosing the exponential parameter such that 95% of the distribution lies below a frequency of one event in 6280 PWR-years, consistent with the usual use of a 95% confidence interval. This approach, which also will be explored, makes no assumptions based on BWR experience.

Core Damage Frequency

The event of interest is a breach in the reactor coolant pressure boundary caused by a detonation in the pressurizer steam space or associated piping. In view of the various pipe sizes where a combustible mixture might accumulate, the possible break sizes include:

- A very small break (S3 - 1/2 inch or less equivalent diameter) caused by a ruptured instrumentation line or a PORV leak.

- A small break (S2 - 1/2 inch to 2 inches equivalent diameter) caused by damage to the PORV and/or block valve internals.

- An intermediate-size break (S1 - 2 inches to 6 inches equivalent diameter) caused by rupture of the line leading to the PORV or by rupture of the pressurizer spray line.

Based on their positions in the upper portions of the pressurizer, the PORV and spray lines are the most credible locations, although the spray line is normally liquid-filled. Thus, the probabilistic calculation will assume an intermediate-size ("S1") break. (This is somewhat conservative, in that instrument line breaks, sampling line breaks, etc. would normally be classified as S2 LOCAs, and evaluated using a separate event tree. This screening analysis therefore is bounding in the sense that all breaks are evaluated as S1 breaks.)

The NUREG-1150[1081] PRA for the Sequoyah plant was chosen for the analysis. This plant has a somewhat higher vulnerability to loss-of-coolant accidents because of its manual switchover to ECCS recirculation mode, and thus should bound most PWR plants.

As was done for the GI-195 screening analysis, this scenario was analyzed by constructing a new event tree. This new event tree was a simple copy of the existing event tree for the intermediate break "S1" LOCA, but the initiating event at the beginning of the tree was replaced by two top events - the detonation-induced pipe break frequency followed by the probability of not isolating the break, as described in the paragraphs above. The remainder of the event tree is exactly the same as that for the "S1" LOCA.

Five separate cases were run, to test first the event tree itself, and then to explore the effect of the various modeling assumptions described earlier. Case I was intended to see if the results of the calculation would match the published results for the SI LOCA event in the NUREG-1150[1081] analyses. To do this, the initiating event frequency was set to 10^{-3} S1 LOCA event/year, the probability of a non-isolatable break was set to unity, and the calculation was run using a sequence cutoff frequency of 10^{-7} and limited Latin hypercube sampling. The result was a mean core damage frequency (CDF) of 6.86E-06/year. The mean S1 LOCA sequences given on pp. 5-16 and 5-17 of NUREG/CR-4550,[1318] Vol. 5, Part 1, Rev. 1, sum to 6.8E-06, which compares quite well.

The sequence cutoff of 10^{-7} which was used in the NUREG-1150[1081] PRAs made the calculation much more practical by greatly reducing the number of cut sets to be calculated. Although such a cutoff tends to slightly reduce the CDF estimate, this is an appropriate numerical technique for the original PRA, since the S1 sequences contributed less than 10% of the total CDF. However, the sequences of interest for this GI are quite likely to be in the 10^{-7} range, and using this cutoff would significantly affect the results. Therefore, Case II is a repeat of Case I, i.e., a calculation of the S1 LOCA tree, but with the cutoff frequency lowered to 10^{-10}/year, and using regular Monte Carlo sampling instead of the limited Latin hypercube method.

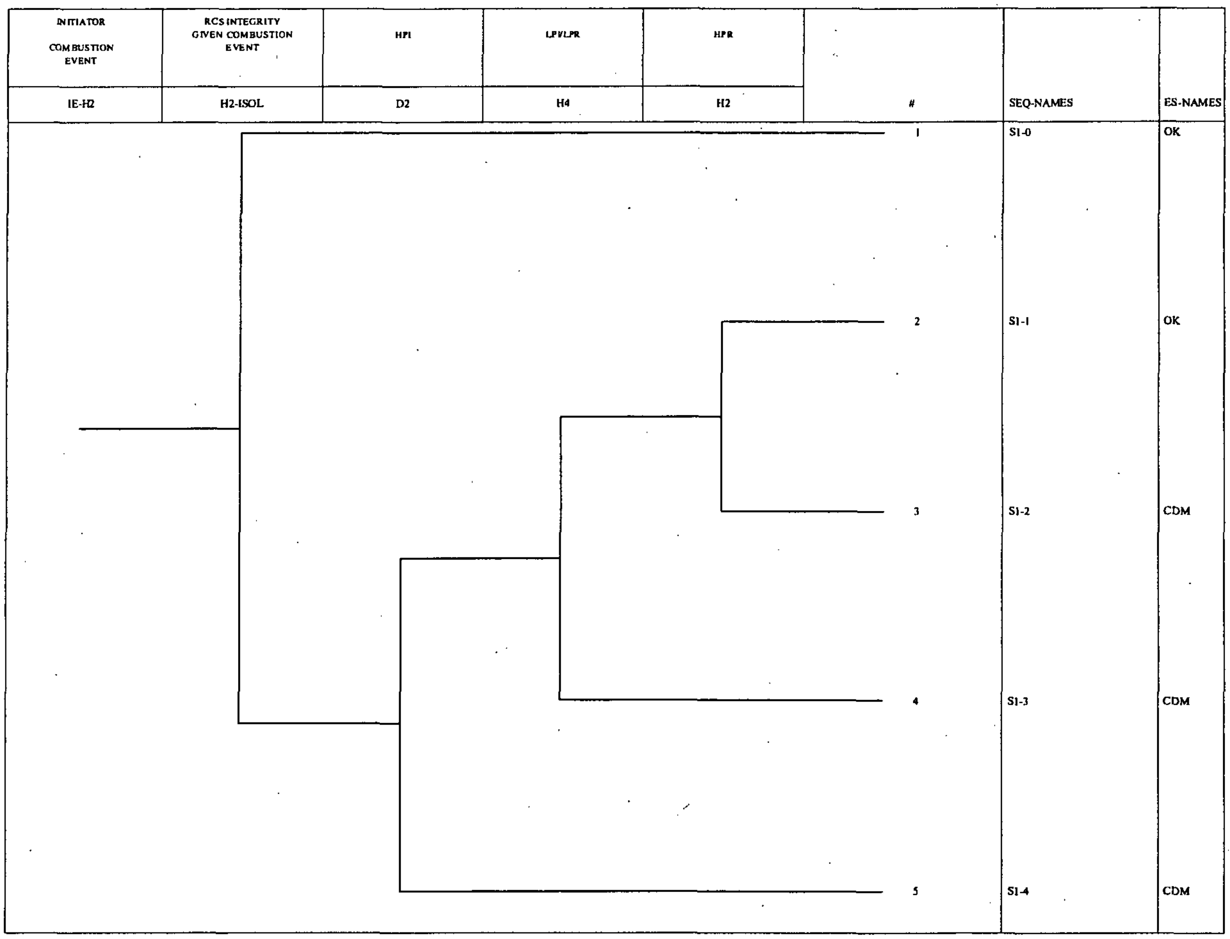
Case III is the same event tree, but with the initiating event frequency lowered to 1.17E-4/PWR-year, the estimate derived from BWR experience, and the probability of a non-isolatable break still set to unity. This calculation is not a "test case" like the first two cases, but instead corresponds to a CDF associated with this GI, although it assumes that every combustion event causes a LOCA.

Case IV is similar to Case III, but this time the probability of a non-isolatable break is calculated by means of the exponential distribution described earlier. This case is the "best" estimate of the safety significance of the GI, based on extrapolation from the experience in BWRs.

Finally, Case V uses the limiting exponential initiating event frequency based only on PWR experience, with the probability of a non-isolatable break set to unity. This case is intended as a check on the modeling assumptions used in Cases III and IV. The results of these cases are as shown in Table 3.198.1.

(Results in this and in subsequent tables are given to three significant figures for the convenience of the reader who wishes to follow the calculations, and are not intended to imply that these parameters are known to this accuracy, as the percentile range given in the table itself clearly shows.)

As can be seen by comparing the various cases tabulated above, the "limiting" case V CDF is only



about a factor of two greater than that estimated by Case IV. Also, the effect of the rather questionable extrapolation of pipe break probability given an ignition event can be seen by comparing the mean estimates based on Case IV vs. Case III: this assumption reduces the calculated CDF by about a factor of five. More importantly, all three cases associated with this GI (i.e., Cases III, IV, and V) have core damage frequencies below the GI screening criterion of 10^{-6} core damage event/RY.

In addition, an audit calculation was performed using a SPAR model for a PWR with the highest CDF for the medium LOCA. (A description of this calculation is attached to this report.) The audit calculation is similar to Case V, in that it is based on the PWR data only, but uses a more sophisticated Bayesian approach. This calculation gave a mean CDF of 8.3E-7, which agrees quite well with Case V above, and also is below the GI screening criterion of 10^{-6} core damage event/RY.

Table 3.198-1

Calculation	Mean	5th Percentile	Median	95th Percentile
Case I existing S1 initiating event frequency, cutoff = 10^{-7}, LHS sampling	6.860E-06	1.645E-07	1.931E-06	2.770E-05
Case II Existing S1 initiating event frequency, but cutoff = 10^{-10} and Monte Carlo sampling	1.003E-05	1.982E-07	2.415E-06	3.259E-05
Case III Initiating event frequency set to ratioed BWR ignition frequency, but assuming every ignition causes a break	1.232E-06	1.327E-07	7.018E-07	3.344E-06
Case IV Initiating event frequency set to ratioed BWR ignition frequency and including non-isolatable break probability	2.425E-07	4.413E-09	9.520E-08	8.662E-07
Case V Initiating event frequency set to 95% confidence distribution based on PWR experience.	5.113E-07	1.614E-08	2.354E-07	1.680E-06

Consequence Estimate

In the base PRA, the S1 LOCA sequences were placed in Plant Damage State Group 2 (see NUREG/CR-4551,[1795] page 2.9). A rough estimate of the consequences was made, using the CRIC-ET Code[1796] and the Sequoyah model, but using a consequence file loaded with the GIs program standard site parameters. The results, using the NUREG-1150[1081] technique of limited Latin hypercube sampling of 200 samples, were as follows:

	Person-rem per reactor-year
Mean	0.14
5th percentile	1.15E-03
95th percentile	0.813
Median	2.59E-02

The distribution is based on the uncertainties associated with the Level II and Level III analyses, and does not include the Level I distribution. (Again, the results in this table are given to three significant figures for the convenience of the reader who wishes to follow the calculations, and are not intended to imply that these parameters are known to this accuracy, as the percentile range given in the table itself clearly shows.) These estimates are well below the screening cutoff of 100 person-rem/RY given in the Handbook for Management Directive (MD) 6.4.

Large Early Release Frequency (LERF)

The NUREG-1150[1081] models loaded into the CRIC-ET code precede the development of large early release models, such as those described in NUREG/CR-6595.[1873] Nevertheless, an approximation can be made using the CRIC-ET code by selecting accident progression sequences which involve a large release. For this analysis, the following were selected:

Accident progression characteristic	Description	Mean frequency	Expected early fatalities, conditional on event
1C	Containment failure during core degradation	1.47E-08	10.9
1D	Containment failure at vessel breach	3.11E-08	17.9
6A	Induced Steam Generator Tube Rupture	1.20E-10	28.6
10C	Containment failed due to small hole, leak, failure to isolate	3.78E-08	0.77
Combination	all above sequences	6.74E-08	---

The frequencies and conditional early fatalities were calculated using the CRIC-ET code and the standard Generic Issues Program site parameters. Other accident progression sequences either were not applicable to a LOCA-initiated event (e.g., characteristics 1A and 1B are vessel rupture failures), or were not large and early, and (not surprisingly) had an estimated number of early fatalities below 0.5. The "combination" result in the bottom row is not a simple sum of the other rows, since there are some sequences which would be included in more than one row.

The combination row is a reasonable approximation for LERF. The statistical distribution is:

mean	6.74E-08	large early releases/RY
5^{th} percentile	1.34E-09	
95^{th} percentile	2.92E-07	
median	2.18E-08	

The distribution is based on the uncertainties associated with the Level II and Level III analyses, and does not include the Level I distribution. Again, the mean is well below the screening cutoff of 10^{-7} large early release/RY given in the Handbook for MD 6.4.

Cost Estimate

Because of the low CDF and risk, a cost estimate will not affect the conclusion of this analysis. Therefore, no cost analysis was performed.

Uncertainties

One of the more unusual aspects of the screening analysis calculations for this GI is that the initiating event is based on actual data, which results in an uncertainty distribution which, although quite wide by experimental standards, is much smaller than the uncertainty limits often seen in probabilistic analyses. Also, the distribution is normal rather than log-normal. The effect is to produce an estimate of CDF where the uncertainty bounds are not as wide as usual. However, some caution is called for in understanding these uncertainty limits, since they do not include the modeling uncertainty associated with extrapolating BWR experience to PWRs.

Discussion

The estimates above are all predicated on the fact that no hydrogen combustion events have thus far been observed thus far in PWRs. If such an event does occur in the future, this GI should be reevaluated. Similarly, if more BWR events occur, such that the estimated BWR initiating event frequency increases above the current estimate of 1.3E-3 combustion event/BWR-year, this GI should be reevaluated.

CONCLUSION

The CDF, LERF, and risk associated with this GI are below the screening thresholds given in Appendix C of the Handbook for MD 6.4, "Generic Issues Program." Therefore, it was concluded that there was insufficient justification for this GI to continue to the technical assessment stage, and the issue was DROPPED from further consideration.[1874]

REFERENCES

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1833. Memorandum to A. Thadani from J. Flack, "Results of Initial Screening of Generic Issue 195, 'Hydrogen Combustion in Foreign BWR Piping,'" February 23, 2004. [ML040850566]

1872. NUREG/CR-2726, "Light Water Reactor Hydrogen Manual," U.S. Nuclear Regulatory Commission, August 1983.

1873. NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," U.S. Nuclear Regulatory Commission, December 1999.

1874. Memorandum to B. Sheron from S. Bahadur, "Results of Initial Screening of Generic Issue 198, 'Hydrogen Combustion in PWR Piping,'" March 22, 2007. [ML070580447]

ISSUE 200: TIN WHISKERS

DESCRIPTION

Historical Background

This issue was identified by RES after concern was raised for the number of component failures in the solid state protection systems of nuclear power plants that were caused by the growth of tin whiskers.[1877]

In 1987, Dresden Nuclear Power Station Unit 2, while in Run Mode at 94% power, experienced a trip of the "B" channel of the RPS because of a high signal on an APRM channel (see LER #87-022-00). The APRM channels take LPRM output signals as their input. The root cause of the failure was determined to be the growth of metallic whiskers on the fission chamber outer electrode. An arc was generated between electrodes causing a transient short circuit which later burned away. After the whisker melted, the LPRM returned to normal operation.

On April 1, 1990, Duane Arnold Energy Center experienced an automatic reactor scram during reactor startup at 8% thermal power after an increase in flux greater than 15% was observed in APRM channels "C" and "D" (see LER #90-004-00). The intermediate cause of the event was a momentary LPRM spike cause by a transient short circuit between the anode and the cathode in the LPRM detector. This momentary spike in the LPRM output generated an APRM reading indicating a sudden power increase which in turn actuates the RPS logic that generates a scram. After the event, the affected LPRM was bypassed and the plant was started up. This event was not the only one of this nature at this plant. In February 1995, a similar event occurred during a controlled shutdown for a scheduled refueling outage. An upscale APRM reading occurred with the reactor shut down.

In December 1997, also at Dresden Unit 2, an unexpected full reactor scram occurred during an Instrumentation Surveillance due to an unexpected half-scram signal generated after RPS Channel "A" received a spiking LPRM reading in conjunction with an expected RPS Channel "B" half-scram signal (see LER #97-019-00). The unexpected LPRM reading was due to a short circuit caused by a whisker present within the detector. Actions taken after this event included complete I-V plots performance, and LPRM replacement with upgraded types.

On September 1, 1999, at South Texas Project Unit 2, a reactor pre-trip alarm occurred due to a low-low level indication in channel 2 of steam generator 2D while in Mode 1 at 100% power (see LER #1999-006-00). The failure of the input control relay that indicated the low-low level on the steam generator was attributed to tin whiskers. The corrective actions included the replacement of the relay.

On April 17, 2005, Millstone Unit 3 experienced a reactor trip from full power, and one of two trains of the Safety Injection and Main Steam Isolation actuated when low steam line pressure was sensed on one of four steam generators. The cause was the growth of a tin whisker between two components on a Westinghouse SSPS universal logic card, causing a short circuit to ground and triggering the single train of SI and the subsequent automatic reactor trip. A Westinghouse SSPS engineer revealed that the component where the whisker grew was a diode with a blue jacket.

On August 25, 2005, the NRC issued Information Notice 2005-25.[1878] which specifically discussed the inadvertent trip at Millstone Unit 3 caused by tin whiskers, notified licensees about recent operating experience related to the growth of tin whiskers in electronic circuits in nuclear power stations, and informed licensees to consider appropriate actions to avoid similar problems.

Safety Significance

Tin whiskers are electrically conductive, crystalline structures of tin that can grow from surfaces where tin (especially electroplated tin) is used as a final finish. Tin whiskers can range from several millimeters to as long as 10 mm in length. Their formation in electronic systems has been cited as the root cause for electronic system failure in military aviation and weapons systems, medical equipment, satellites, and in various instances within the electronics and power industries. Electronic component failures caused by tin whiskers have the potential to affect instrumentation and control (I&C) systems in nuclear power plants.

Until recently, this phenomenon has received relatively little attention in the nuclear industry. However, component failures attributed to tin whiskers have occurred at nuclear power plants, ranging in severity from spurious alarms and faulty signals in protection systems to reactor trips.

The use of electroplated tin has been driven by a movement to lead-free products. There is a possibility that the nuclear industry will be increasing the use of commercial off-the-shelf (COTS) -based I&C systems. Although formation of the tin whiskers have the potential to pose safety and reliability issues to all makers and users of high-reliability electronics and associated hardware, the electronics industry and COTS users have experience in documenting and tackling this problem.

Tin whisker incidents in nuclear power plants to date have not been safety significant. There have been no Accident Sequence Precursor (ASP) events, no impairment of redundant safety functions, no CCF events, no increasing trend of recent events, and no apparent decrease in reliability of systems/components due to tin whiskers. Lastly, this issue falls within the scope of the maintenance rule and no new rule/regulation is apparently needed.

Possible Solution

If the issue does not meet the criteria for pursuit, it could be dropped or be pursued as a compliance issue, as warranted. If further study is needed to better understand and establish the technical basis and safety impact of tin whiskers, NRR could make use of the User Need process to request RES assistance.

CONCLUSION

The low number of reported events associated with this issue, the lack of any increasing trend, the lack of any apparent decrease in reliability of systems or components due to tin whiskers, the existence of applicable regulatory requirements and programs (i.e., 10 CFR Part 21, the maintenance rule requirements, and the Reactor Oversight Program), and the issuance of Information Notice 2005-25[1878] to alert licensees collectively indicated that tin whiskers did not meet the requirements of NRC Management Directive 6.4. "Generic Issues Program," for further pursuit. Based on the considerations discussed above, RES recommended that the issue be returned to the originator to be evaluated for other possible options. As a result, the issue was DROPPED from further pursuit.[1879]

REFERENCES

1877. Memorandum to F. Eltawila from R. Barrett, "Proposed Generic Safety Issue: Tin Whiskers," August 18, 2005. [ML052080291]

1878. Information Notice 2005-25, "Inadvertent Reactor Trip and Partial Safety Injection Actuation Due to Tin Whisker," U.S. Nuclear Regulatory Commission, August 25, 2005. [ML052150404]

1879. Memorandum to B. Sheron from C. Lui, "Results of Initial Screening of Generic Isuue 200, 'Tin Whiskers,'" January 26, 2007. [ML070250399]

ISSUE 201: SMALL-BREAK LOCA AND LOSS OF OFFSITE POWER

DESCRIPTION

Historical Background

This issue was identified[1880] by NRR following an allegation that was submitted to the NRC in March 2006 that described a scenario in which a SBLOCA event in a PWR has progressed to the sump recirculation mode of core cooling with the ECCS aligned for high-pressure recirculation. The allegation also described the safety concern relative to "the plant response starting with this ECCS alignment, should a LOOP occur," and stated that for "some PWR designs and operating procedures, the plant response to a LOOP will cause the emergency diesel generators to start and loads to be automatically sequenced onto the emergency buses." This may cause the high-pressure ECCS pumps to be sequenced onto the emergency bus before the low-pressure pumps come onto the emergency bus – resulting in the high-pressure ECCS pumps starting with insufficient suction head, likely causing pump damage.

Safety Significance

In the event of a LOCA in a PWR, approximately 300,000 gallons of water are available in the RWST for post LOCA injection. Eventually a switchover to the recirculation mode is necessary after a LOCA as the RWST inventory is depleted. The timing of the switchover to recirculation mode depends on the size of the break, due to the flow from the centrifugal HPI pumps – increasing as the pressure decreases. For large breaks, the maximum injection pressure of the LPI pumps will be reached relatively quickly, and the HPI pumps are not needed for recirculation. The HPI pumps are needed during recirculation for small breaks. The scenario of interest is an SBLOCA event where the RWST is depleted and the ECCS is aligned in the HPI recirculation mode, approximately 4 hours into the event. At this lineup, the LPI pumps provide suction head to intermediate/or high head pumps, depending on the design. The potential safety concern is (the possibility) that if a LOOP occurs in this situation, the HPI pumps may be sequenced onto the EDG prior to the sequencing of the low head pumps. This would result in the HPI pumps being restarted with inadequate suction head, with likelihood of pump damage. It should be noted that there are potentially other scenarios, such as a transient relying upon primary system feed-and-bleed that result in conditions similar to the SBLOCA-LOOP scenario of interest.

SCREENING ANALYSIS

On detection of an SBLOCA, steam generators are used to cool the RCS. Centrifugal charging pumps are used for inventory control. Typically, HPI pumps (1500 psig discharge pressure) run with mini-flow valves to the RWST open – ensuring no damage. After a period of 2 to 4 hours, the RWST level will necessitate swap over to recirculation sump suction and allow recirculation (and closure of the HPI mini-flow valves).

If a LOOP occurs at this point, a valid recirculation signal will be present. Therefore, when the safety injection sequencer initiates after EDG breaker closing, the equipment's recirculation configuration will not change. Also, procedurally, operators are required to validate that the configuration is in its correct alignment for the plant condition (recirculation).

For most accident sequences, this issue is not expected to be a concern after about 4 hours following an SBLOCA; nevertheless, a 24-hour exposure period to account for uncertainties was used in this analysis. At this condition, the RCS temperature and pressure will be low enough to begin to transfer to long-term RHR. At this pressure and temperature (350°F, 350 psig), the HPI will inject enough water so that loss of mini-flow is not a problem. Similarly, since Westinghouse sometimes uses RHR to the SI pump piggy-back alignment to supply water from the sump to the RCS, and RHR spray is initiated to control the containment environment, RHR, mini-flow is not a concern in this situation.

Millstone 3 has a sub-atmospheric containment and relies on sump recirculation early in the event. As such, it is considered a bounding plant for this risk analysis. For most U.S. PWRs, following an SBLOCA, it is possible to use steam generator cooling to bring the plant pressure and temperature to a point at which RHR can be used for decay heat removal. In other words, most plants can use RHR without going to recirculation. Thus, the hypothetical frequency of going to recirculation at Millstone 3 is higher than for other plants. Also, based on our analysis of all SPAR PWR models, Millstone 3 has the same or a higher core damage frequency for the SBLOCA event than all others. Thus, it was used in the screening analysis.

At CE plants, LPI pumps are turned off with a recirculation signal, and the HPI pumps take suction directly from the sump. At Westinghouse plants, the low pressure pumps take suction from the sump and piggy-back the flow to the RCS through a combination of high pressure and low pressure pumps. For B&W plants following a LOCA, flow is initiated in the HPI and LPI systems from the borated water storage tank (BWST) to the reactor vessel. Flow is also initiated by the reactor building spray (RBS) system to the building spray headers. When the BWST inventory is depleted, recirculation from the reactor building sump is initiated by the operator for both the LPI flow and the reactor building sprays. If elevated RCS pressure requires piggyback operation, recirculation will also occur through the HPI System.

Long-term core cooling occurs by recirculation of injection water from the reactor building sump to the core through the LPI system for large breaks, or through the LPI system and the HPI system – in series – for small breaks where primary pressure remains above the shutoff head of the LPI pumps.

Although the HPI and the LPI systems operate to provide full protection across the entire range of break sizes, each system may operate individually and is initiated independently. The HPI system prevents uncovering of the core for small coolant piping leaks where high RCS pressure is maintained and delays uncovering of the core for intermediate sized breaks. The core flooding and LPI systems are designed to recover the core at intermediate to low RCS pressures and to insure adequate core cooling for break sizes ranging from intermediate breaks to the double-ended rupture of the largest pipe. The LPI system is also designed to permit long-term core cooling in the recirculation mode after a LOCA. The LPI system and the HPI system are designed to permit the recirculation mode at various system pressures following a LOCA. This is accomplished using LPI directly to the core for the low RCS pressures that exist following a large-break LOCA. The LPI discharge provides suction to the HPI in the "piggyback" mode of operation for higher RCS pressures which may occur following a small break. Pumped injection includes both HPI and LPI, each with separate and independent flow paths. One flow path from HPI system and one flow path from LPI system and the core flooding tanks are capable of providing 100% of necessary core injection.

During an SBLOCA, if system pressure remains above the LPI shutoff head upon depletion of the BWST, the LPI suction is manually realigned to the containment sump and the discharge is aligned to the suction of the HPI pumps (piggyback operation). Operation of the HPI system continues until the system operation is manually terminated.

To assure adequate makeup capability for the full range of SBLOCAs, each HPI pump is piped to all four injection lines. Redundant flow instrumentation and throttling globe valves were added to prevent pump run-out and to allow flow balancing among these four paths. Operation of this system does not normally depend on the operation of any other engineered safeguard; however, the system can be operated in series with the LPI system, in recirculation mode.

<u>Frequency Estimate</u>

A screening analysis was performed for sequences that require high pressure recirculation for continued core cooling during an SBLOCA event. The LOOP occurs during or after switchover to recirculation mode. The LPI system and the HPI system are designed to permit the recirculation mode at various system pressures following a LOCA. If the LPI pump loads before the HPI pump, the LPI discharge provides suction to the HPI in "piggyback" mode for higher RCS pressures which may exist following a small break. Therefore, there will be no pump runout and the HPI will have adequate suction flow. This concern is also applicable in sequences involving “feed and bleed” and stuck-open PORVs since plant behavior is equivalent to SBLOCA-initiated sequences.

<u>Probability of a LOOP</u>: We are concerned if a LOOP happens during the swapover to recirculation or during the recirculation. The LOOP could happen at random and could occur several times during recirculation. This random process can be represented by a Poisson process.

The probability of having failures (LOOPs) is:

$$P(n) = \frac{u^n e^{-u}}{n!}$$

$$P(n \geq 1) = 1 - P(0)$$

$$P(n \geq 1) = 1 - \frac{u^0 e^{-u}}{o!} = 1 - e^{-u}$$

where $u = (\lambda_{LOOP} + \lambda_{Weather} + \lambda_{Seismic})t = \lambda_{total}t = \lambda t$.

The LOOP frequency was 3.59E-02/year[1881] and exposure time was conservatively assumed to be 24 hours. Therefore:

$$u = \frac{3.59E-02}{year} * 24\ hours * \frac{1\ year}{8760\ hours} = 9.8E-05$$

Since u is much smaller than 1 and expanding the above equation using Taylor series, we get

$$P(n \geq 1) = 1 - (1 - u + \frac{u^2}{2!} - \frac{u^3}{3!} + \cdots).$$

If we ignore the small terms, we get:

$$P(n \geq 1) = u.$$

Therefore, the λt approximation for the occurrence probability of multiple LOOPs is equal to the exact expression.

Risk Calculation: A full train of ECCS is required for cooling during recirculation. It is conservatively assumed that at the occurrence of the LOOP the high pressure recirculation (HPR) is failed. The event trees for small LOCA and Transient are shown in figures below. We are only interested in sequences that involve recirculation, including those resulting from SBLOCA and consequential SBLOCA such as "feed and bleed" and stuck open PORVs. To determine the frequency of being in recirculation (i.e., being vulnerable to the GI-201 scenario), the probability of operator to initiate the HPR is set to one and the calculation was done using the SPAR (Standardized Plant Analysis Risk) model for Millstone 3 with a sequence cutoff frequency of 10^{-12}. The ΔCDF in Table 3.210-1 is the CDF increase for failed HPR function, which is equivalent to the frequency of entering HPR.

$$CDF = (Frequency\ being\ in\ \mathrm{Re}circulation) * P(LOOP)$$

Figures 3.201-1 and 3.201-2 show the transient and SBLOCA event trees. Other event trees referred to in Table 3.201-1 are similar to the transient (event) tree and are not included. The frequency of being in recirculation is the summation of being in recirculation during small LOCA, "feed and bleed" and stuck open PORVs modes (the scenarios are marked in the figures 1 and 2). Using the expression for the CDF – with frequency 4.1E-04 from Table 3.202-1, we get

CDF = (4.1E-04)/year * 9.8E-05 or, CDF = 4E-08/year.

This CDF is a bounding estimate of the risk associated with getting a LOOP during recirculation because:

(1) The analysis assumed all HPR pumps would start when the EDG started. This is not true if there is no SI signal present or if some HPR pumps are not designed to autostart.

(2) The EDG load sequencer may load differently for different operational modes. To ensure that we have bounded the issue, we assume that the most conservative loading pattern occurs, i.e., the loading sequence always causes all HPI pumps to fail.

(3) The analysis assumed that all HPR pumps will fail. They may survive until the LPI pumps are started and providing water.

(4) The analysis assumed that the plant was in HPR for 24 hours. The time in HPR varies for different sequences and, for the purpose of risk analysis, would always be less than 24 hours.

(5) The plant chosen for analysis has a simple AFW system and required HPR for successful mitigation of large and medium LOCAs.

(6) The plant chosen is one of the few plants that does not credit the ability to cool down and place the plant in RHR without going to recirculation.

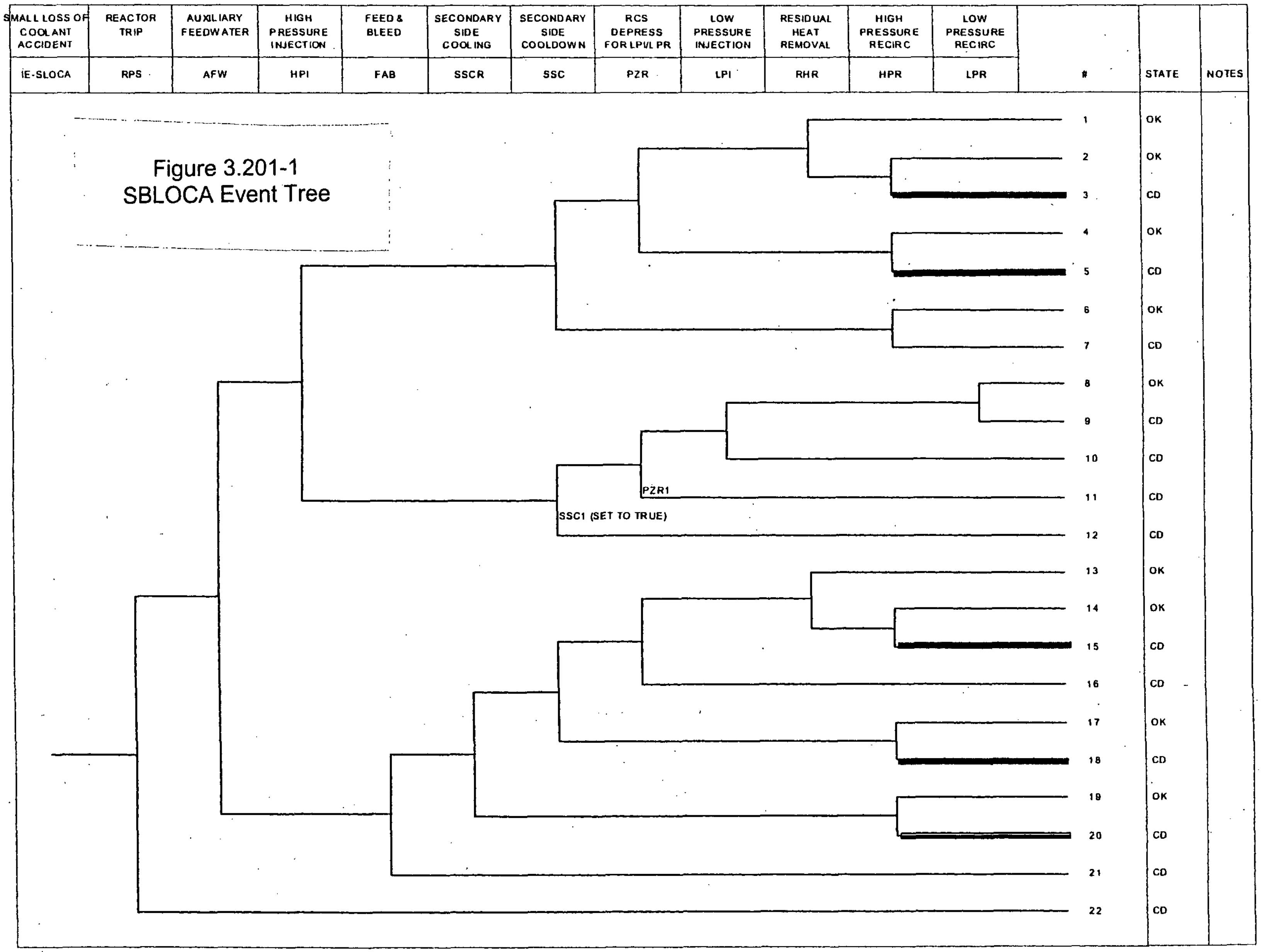

Figure 3.201-1
SBLOCA Event Tree

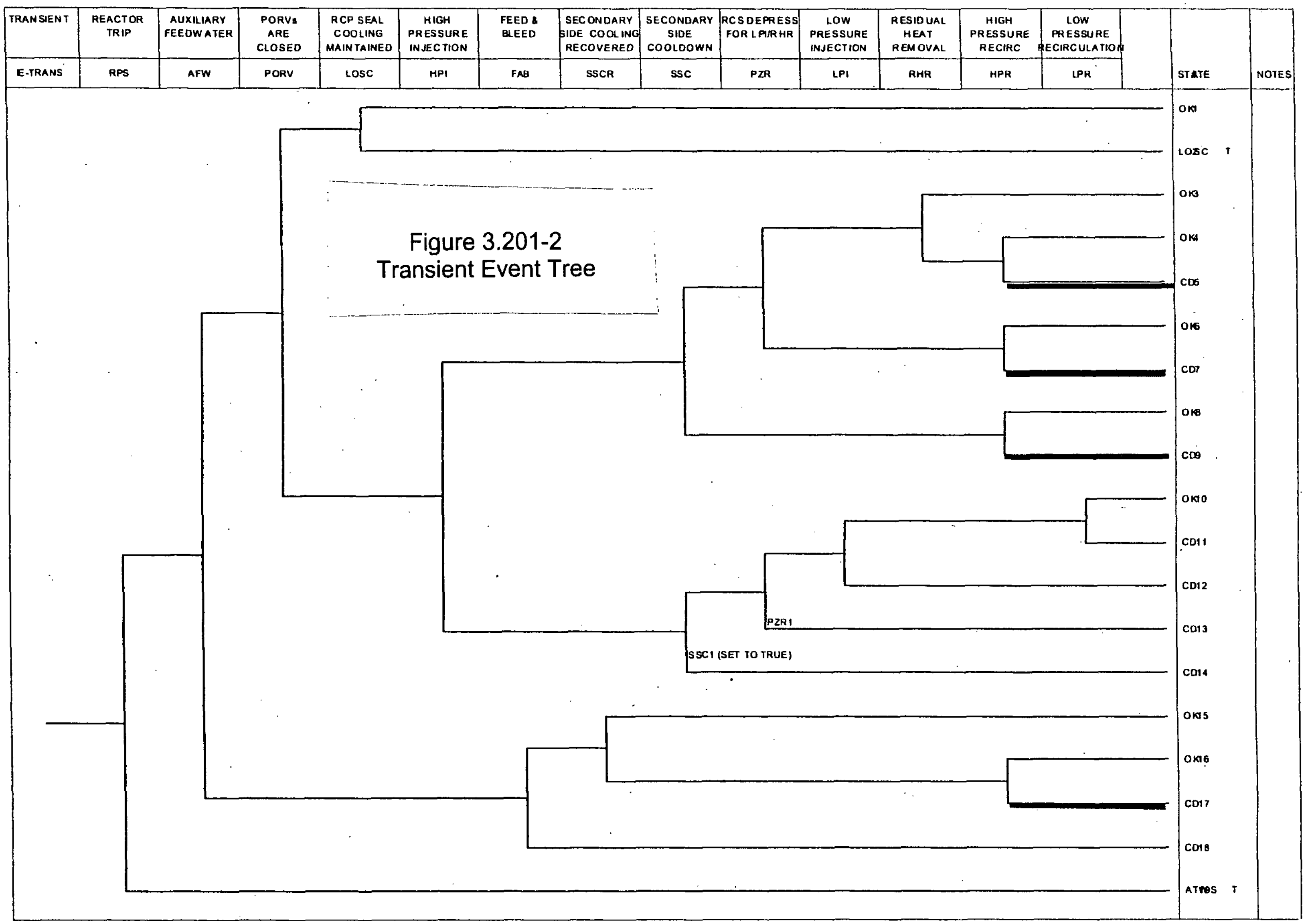

Figure 3.201-2
Transient Event Tree

With a CDF this low, calculations of LERF and person-rem/RY were not warranted. If it is assumed that all CDF ended in LERF (probability that CDF causes LERF is one), the LERF value is 4E-08, which is less than the LERF threshhold value of 1E-07.

Table 3.201-1

Event Tree	Sequence	CDF with HPR function Failed	Baseline CDF	ΔCDF
SLOCA	3	4.00E-04	2.40E-06	4.00E-04
TRANS	17	1.30E-06	7.50E-09	1.30E-06
SLOCA	5	1.10E-06	6.20E-09	1.10E-06
LOMFW	19	9.60E-07	2.00E-08	9.40E-07
LOCHS	19	8.60E-07	1.80E-08	8.40E-07
TRANS	5	5.10E-07	2.90E-09	5.00E-07
SLOCA	7	4.00E-07	2.20E-08	3.80E-07
LOMFW	18	1.90E-07	1.00E-09	1.90E-07
LOCHS	18	1.70E-07	9.10E-10	1.70E-07
LOCHS	5	6.50E-08	3.60E-10	6.50E-08
			Sum	4.05E-04

CONCLUSION

The negligible CDF increase associated with a SBLOCA and LOOP indicated that the issue did not require a technical assessment, in accordance with the guidelines of NRC Management Directive 6.4, "Generic Issues Program." Therefore, the issue was DROPPED from further pursuit.[1882]

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ISSUE 202: SPENT FUEL POOL LEAKAGE IMPACTS

DESCRIPTION

Historical Background

Spent fuel pools (SFPs) at PWRs are Seismic Category I structures that contain borated water, maintain spent fuel temperatures, and provide radiation shielding. The SFPs are typically lined with stainless steel plates, joined by full-penetration welds, on the inner surface of the reinforced concrete SFP structures. The vast majority of SFPs also have leakage collection systems (channels embedded in concrete at weld seams) that are designed to collect any borated SFP water that might leak through the liner for processing as radioactive waste water. These systems also provide a means of monitoring SFP leakage.

This issue involves the potential for long-term leakage of borated water through the SFP liners, reactor cavities, and fuel transfer canals at PWRs, to degrade the concrete support structures and associated reinforcement steel (i.e., rebar) if the leak-off channels are clogged. If not properly drained, long-term leakage of borated water through the liners will accumulate, wet the concrete behind the liner, seep through cracks in the concrete, and reach the rebar within the concrete. If not corrected, over a long period of time this condition could degrade the concrete and rebar and potentially compromise the structural integrity.

In September 2002, Salem Unit 1 identified evidence of radioactive water leakage through the interior walls or penetrations in both the auxiliary and fuel handling buildings (FHB). Investigations of the leakage by the licensee in February 2003 revealed the radionuclide tritium in the groundwater near the FHB. Further licensee inspections identified long-term leakage of borated water from the SFP through cracks into the structural walls of the SFP. The licensee determined that the SFP leak-off channels (i.e., tell-tale drains) that drain borated SFP water collected between the pool liner and concrete walls were clogged. Since the Salem Unit 1 finding, similar conditions involving SFP structures were discovered at Indian Point and Seabrook (transfer canal). Based on these discoveries, NRR proposed[1883] that the issue be addressed as a generic issue, in accordance with NRC Management Directive (MD) 6.4, "Generic Issues Program."

Safety Significance

SFP leakage has the potential to degrade the integrity of SFP structures due to the adverse impact of borated water on concrete and its rebar. That is, borated water can seep through cracks in the concrete, reduce the inherent high alkaline environment of concrete, and possibly expose rebar in the vicinity of these cracks to mildly acidic conditions. The degree of potential degradation would be expected to vary among facilities as the SFP leak rates, concrete conditions (alkalinity, porosity, and cracking), and degree of leak-off channel flow restrictions vary. In accordance with their corrective action programs, the licensees of plants where this condition was discovered have evaluated the short-term and long-term safety concerns (e.g., potential degradation of the SFP basemat and potential reductions in design license margins of the SFP structures). In each case, the licensees' evaluations determined there was no immediate safety concern and no long-term degradation in the design license margins of the SFP structures from borated water leakage.[1883,1884] The potential adverse impacts of SFP leakage on groundwater and possible consequent effect of the leakage on public health and safety was evaluated in 2006 as part of an NRC task force on ground water contamination from leakage of radioactive water from various sources at several licensee facilities.[1885]

Possible Solution

Information on the impact of borated water leakage through reinforced concrete structures at PWRs is limited. In many cases, the affected structures are subterranean and inaccessible, and leakage may not be readily detectable or repaired. The NRC issued generic communications to licensees to inform them of these conditions and the potential adverse consequences.[1886,1887] Licensees are required to maintain the design license basis of plant structures, systems, and components important to safety. The NRC monitors licensee performance and responds to conditions adverse to safety and quality at plants under the reactor oversight process (ROP). Accordingly, the existing regulatory framework provides sufficient means for licensees and the NRC to prevent, detect, and correct SFP leakage conditions that could adversely impact SFP structural integrity over the long term. If industry experience were to show an adverse trend developing in this area, then research to gain additional information on the impact of borated water leakage through reinforced concrete structures might be warranted.

SCREENING ANALYSIS

The PWR licensees that discovered SFP leakage conditions described above have taken corrective actions to preclude adverse impact to the long-term integrity of the Seismic Category I structures. For these few instances, the NRC used the ROP to review and assess the condition and the licensee actions to address the condition, and determined that the licensees' actions were adequate to maintain the plants' design license basis. The NRC also issued generic communications to inform plant licensees of these conditions and their potential adverse consequences. Accordingly, this issue was considered to be adequately monitored and addressed through existing regulatory programs. The following summarizes specific SFP leakage conditions discovered at Salem, Indian Point, and Seabrook plants.

Salem Unit 1: NRC and licensee reviews identified long-term leakage of the SFP through structural cracks to onsite groundwater. A visual inspection of the tell-tale leak-off drains revealed significant blockage with boric acid and calcium deposits. The licensee concluded that the blocked leak-off drains resulted in SFP water accumulation in the annulus area between the concrete pool wall and stainless steel liner, which eventually resulted in leakage of the water through construction joints and cracks in the SFP and FHB walls. The licensee initiated action to clear the drains to establish flow in the tell-tales and effectively drain the annulus area. Subsequently, the tell-tale drain rate held steady at about 100 gallons/day from liner leakage that was not specifically located or repaired. The leakage is being collected and processed as radioactive waste. Licensee monitoring of leakage did not proactively detect the leakage. Rather, a personnel contamination event prompted identification of the through-wall leakage. The NRC issued Information Notice No. 2004-05[1886] to inform the industry of this event.

The licensee initiated an assessment of potential long-term effects on the structure due to boric acid attack. The assessment covered an extended period of time to effectively simulate long-term boric acid effects on representative concrete specimens, including rebar. The licensee's assessment determined that the design license basis for the SFP structural integrity was maintained for the life of the plant, including during possible license extension. The licensee committed to routine monitoring of the leak-off flow of the tell-tale drain channels.

Indian Point 2: This plant does not have a SFP leak collection system, and has a history of leakage of borated water which may potentially affect the concrete and steel reinforcement. The licensee and NRC reviewed the SFP and concluded that the crack did not affect the SFP wall structural integrity. However, groundwater contamination was detected.

Seabrook: The licensee experienced sporadic leakage from its cask loading and transfer canal area during flood-up to support refueling. The area was drained after outages. The cause of this leakage was repaired, and a licensee analysis of the structural integrity of the canal did not identify any resulting adverse conditions. However, groundwater contamination was detected.

CONCLUSION

In addition to 10 CFR 50, Appendices A and B, industry standards existed to ensure a proper and timely evaluation of leaks that may develop in SFPs and the resultant impact on their concrete structures. The concerns of this issue involved long-term degradation of the structures and the adequacy of individual licensee corrective actions for this condition. Therefore, consistent with MD 6.4 and Generic Issues Program improvements described in SECY-07-0022,[1888] this issue was eliminated from further assessment as a generic issue. Additional research on the adverse impact of borated water on SFP concrete and its rebar might be warranted if the industry experienced an increasing adverse trend of this condition with the potential to degrade the design license basis of the structures.[1889]

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ISSUE 203: POTENTIAL SAFETY ISSUES WITH CRANES THAT LIFT SPENT FUEL CASKS

DESCRIPTION

This issue was identified by an NMSS engineer in December 2006 and addressed concerns about potential safety issues with cranes that are used to lift spent fuel casks at nuclear power plants. The areas of concern involved questions about the technical adequacy or programmatic effectiveness of existing regulatory programs and activities that are implemented through the NRC reactor oversight process (ROP). These were as follows:

(1) Cranes that do not conform to original design specifications: load drop analysis is not part of design basis, inadequate design basis documentation/information from parts vendors, and NRC inspection responsibilities for review of load drop analysis are unclear.

(2) Cranes that licensees modified without performing safety evaluation reviews required by 10 CFR 50.59, which may indicate that some licensees assume that these reviews are not required.

(3) Cranes that have had inadequate maintenance: overlooks important operating experience, invalidates single-failure proof capability, and tolerated by deficient NRC requirements for maintenance.

(4) *Cranes that are not single-failure proof or lack credible validation for single-failure proof* status have been used without adequate load path protection, or other mitigative measures, and this condition is not adequately considered in NRC's probabilistic risk assessment of crane events.

SCREENING ANALYSIS

RIS 2005-25[1890] describes the NRC's regulatory position with respect to the areas of concern identified in this issue, and identifies the design and inspection bases for cranes having the potential to impact SSCs important to safety. RIS 2005-25[1890] addresses: (1) single-failure-proof cranes, including guidelines for upgrading cranes to single-failure-proof status as well as for crane inspection, testing, and maintenance; and (2) conditions requiring load drop analyses. RIS 2005-25[1890] provides guidance on load drop analysis assumptions and methods, and incorporated operating experience from NUREG-1774,[1846] including results from staff work associated with Issue186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants," which formed the risk-informed basis for its use by the NRC to clarify NRC guidelines for licensees' programs for the control of heavy loads.

One possible net impact of the four areas of concern is that licensees have maintained inadequate control of changes to cranes over long periods of time. Sources of crane changes to consider include: design modifications, parts and information from vendors, aging, maintenance (or lack thereof), and incorporation of lessons learned from operating experience. Potential consequences from licensees' inadequate control of these changes may result in conditions where cranes no longer conform to original design specifications, conditions that invalidate load drop analysis or the single-failure-proof status of cranes. Licensees' control of the design basis of cranes that provide functions important to safety is clearly within the purview of the ROP inspection procedures. In addition, guidance from RIS 2005-25[1890] describes the NRC's regulatory position that licensees can maintain adequate defense-in-depth through the use of single-failure-proof cranes, by providing various appropriate forms of load path protection, or by performing adequate load drop analyses that demonstrate acceptable consequences. Thus, licensees may choose among these alternatives to maintain adequate defense-in-depth (i.e., they are not specifically obligated to maintain original design basis, to perform load drop analysis, or to provide load path protection for cranes providing important to safety functions). Again, these considerations are part of the existing regulatory framework.

CONCLUSION

The four areas of concern identified in this issue are covered under existing inspection procedures of the ROP. The guidance provided to licensees in RIS 2005-25[1890] clearly describes NRC's regulatory position for licensees' control of heavy loads programs covering the areas of concern identified. Therefore, the areas of concern represented licensee compliance issues and, as such, were not suitable for further assessment in the Generic Issues Program, as delineated in MD 6.4. The concerns were entered into the ROP Feedback Program, in accordance with Inspection Manual Chapter 0801, and assigned the ROP Feedback Form Item Number 60854-1-1113. Thus, this issue was DROPPED from further pursuit as a generic issue.[1891]

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APPENDIX B

APPLICABILITY OF NUREG-0933 ISSUES TO OPERATING AND FUTURE REACTOR PLANTS

This appendix contains a listing of those residual GSIs that are applicable to operating and future reactor plants and includes: issues that have been resolved with requirements [I, NOTE 3(a)]; USI, HIGH- and MEDIUM-priority issues scheduled for resolution; nearly-resolved issues scheduled for resolution (NOTES 1 and 2); and issues that are scheduled for prioritization (NOTE 4). The priority designations for all issues are consistent with those listed in Table II of the Introduction. In accordance with 10 CFR 52.47(a)(1)(iv), any future application for design certification must contain proposed technical resolutions for the issues in this listing that are designated USI, HIGH, MEDIUM, NOTE 1, and NOTE 2. (In July 1998, the priority categories NOTES 1 and 2 were eliminated and all GSIs in these categories were given a HIGH priority ranking.[1718]) Also included in this listing are those GSIs that were either prioritized or resolved with no impact on operating reactor plants but contain recommendations for future reactor plants (NOTE 6).

Legend

NOTES: 1	- Possible Resolution Identified for Evaluation (Discontinued 07-06-98)
2	- Resolution Available [Documented in NUREG, NRC Memorandum, SER or equivalent] (Discontinued 07-06-98)
3(a)	- Resolution Resulted in the Establishment of New Regulatory Requirements [Rule, Regulatory Guide, SRP Change, or equivalent]
4	- Issue to be Prioritized in the Future
6	- New Requirements for Future Plants Recommended
B&W	- Babcock & Wilcox Company
CE	- Combustion Engineering Company
GE	- General Electric Company
CONTINUE	- Work on the issue continues in accordance NRC Management Directive 6.4[1858]
HIGH	- High Safety Priority
I	- Resolved TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737
MEDIUM	- Medium Safety Priority
MPA	- Multiplant Action
NA	- Not Applicable
TBD	- To Be Determined
USI	- Unresolved Safety Issue
<u>W</u>	- Westinghouse Electric Corporation

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor BWR	PWR	Operating Plants- MPA No	Operating Plants - Effective Date	Future Plants- Effective Date
	TMI ACTION PLAN ITEMS						
I.A	OPERATING PERSONNEL						
I.A.1	Operating Personnel and Staffing						
I.A.1.2	Shift Technical Advisor	I	All	All	F-01	09/13/79	09/27/79
I.A.1.2	Shift Supervisor Administrative Duties	I	All	All		09/13/79	09/27/79
I.A.1.3	Shift Manning	I	All	All	F-02	07/31/80	06/26/80
I.A.1.4	Long-Term Upgrading	NOTE 3(a)	All	All		04/28/83	04/28/83
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	I	All	All	F-03	03/28/80	03/28/80
I.A.2.1(2)	Training	I	All	All	F-03	03/28/80	03/28/80
I.A.2.1(3)	Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	I	All	All	F-03	03/28/80	03/28/80
I.A.2.3	Administration of Training Programs	I	All	All		03/28/80	03/28/80
I.A.2.6	Long-Term Upgrading of Training and Qualifications	-	-	-	-	-	-
I.A.2.6(1)	Revise Regulatory Guide 1.8	NOTE 3(a)	All	All		TBD	05/--/87
I.A.3	Licensing and Requalification of Operating Personnel						
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	I	All	All		03/28/80	03/28/80
I.A.4	Simulator Use and Development						
I.A.4.1	Initial Simulator Improvement	-	-	-	-	-	-
I.A.4.1(2)	Interim Changes in Training Simulators	NOTE 3(a)	All	All		04/--/81	03/28/81
I.A.4.2	Long-Term Training Simulator Upgrade	-	-	-	-	-	-
I.A.4.2(1)	Research on Training Simulators	NOTE 3(a)	All	All		04/--/87	04/--/87
I.A.4.2(2)	Upgrade Training Simulator Standards	NOTE 3(a)	All	All		04/--/81	04/--/81
I.A.4.2(3)	Regulatory Guide on Training Simulators	NOTE 3(a)	All	All		04/--/81	04/--/81
I.A.4.2(4)	Review Simulators for Conformance to Criteria	NOTE 3(a)	All	All		03/25/87	03/25/87

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
I.C	OPERATING PROCEDURES						
I.C.1	Short-Term Accident Analysis and Procedures Revision	-	-	-	-	-	-
I.C.1(1)	Small Break LOCAs	I	All	All		09/13/79	09/13/79
I.C.1(2)	Inadequate Core Cooling	I	All	All	F-04	09/13/79	09/13/79
I.C.1(3)	Transients and Accidents	I	All	All	F-05	09/13/79	09/27/79
I.C.2	Shift and Relief Turnover Procedures	I	All	All		09/13/79	09/27/79
I.C.3	Shift Supervisor Responsibilities	I	All	All		09/13/79	09/27/79
I.C.4	Control Room Access	I	All	All		09/13/79	09/27/79
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	I	All	All	F-06	05/07/80	06/26/80
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	I	All	All	F-07	10/31/80	10/31/80
I.C.7	NSSS Vendor Review of Procedures	I	All	All		NA	06/26/80
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	I	All	All		NA	06/26/80
I.C.9	Long-Term Program Plan for Upgrading of Procedures	NOTE 3(a)	All	All		09/13/79	06/--/85
I.D	CONTROL ROOM DESIGN						
I.D.1	Control Room Design Reviews	I	All	All	F-08	06/26/80	06/26/80
I.D.2	Plant Safety Parameter Display Console	I	All	All	F-09	06/26/80	06/26/80
I.D.5	Improved Control Room Instrumentation Research	-	-	-	-	-	-
I.D.5(2)	Plant Status and Post-Accident Monitoring	NOTE 3(a)	All	All		NA	12/--/80
I.F	QUALITY ASSURANCE						
I.F.2	Develop More Detailed QA Criteria	-	-	-	-	-	-
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	NOTE 3(a)	All	All		NA	07/--/81
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	NOTE 3(a)	All	All		NA	07/--/81
I.F.2(6)	Increase the Size of Licensees' QA Staff	NOTE 3(a)	All	All		NA	07/--/81
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	NOTE 3(a)	All	All		NA	07/--/81
I.G	PREOPERATIONAL AND LOW-POWER TESTING						
I.G.1	Training Requirements	I	All	All		NA	06/26/80
I.G.2	Scope of Test Program	NOTE 3(a)	All	All		NA	07/--/81

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants- MPA No	Operating Plants - Effective Date	Future Plants- Effective Date
			BWR	PWR			
II.B	CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW						
II.B.1	Reactor Coolant System Vents	I	All	All	F-10	09/13/79	09/27/79
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	I	All	All	F-11	09/13/79	09/27/79
II.B.3	Post-Accident Sampling	I	All	All	F-12	09/13/79	09/27/79
II.B.4	Training for Mitigating Core Damage	I	All	All	F-13	03/28/80	03/28/80
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	NOTE 3(a)	All	All		TBD	NA
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	NOTE 3(a)	All	All		TBD	01/25/85
II.D	REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES						
II.D.1	Testing Requirements	I	All	All	F-14	09/13/79	09/27/79
II.D.3	Relief and Safety Valve Position Indication	I	All	All		07/21/79	09/27/79
II.E	SYSTEM DESIGN						
II.E.1	Auxiliary Feedwater System						
II.E.1.1	Auxiliary Feedwater System Evaluation	I	NA	All	F15	03/10/80	03/10/80
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	I	NA	All	F-16, F-17	09/13/79	09/27/79
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	NOTE 3(a)	All	All		NA	07/--/81
II.E.3	Decay Heat Removal						
II.E.3.1	Reliability of Power Supplies for Natural Circulation	I	NA	All		09/13/79	09/27/79
II.E.4	Containment Design						
II.E.4.1	Dedicated Penetrations	I	All	All	F-18	09/13/79	09/27/79
II.E.4.2	Isolation Dependability	I	All	All	F-19	09/13/79	09/27/79
II.E.4.4	Purging	-	-	-	-	-	-
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging	NOTE 3(a)	All	All		11/28/78	NA
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Letter	NOTE 3(a)	All	All		10/22/79	NA
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability	NOTE 3(a)	All	All		09/27/79	NA

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
II.E.5	Design Sensitivity of B&W Reactors						
II.E.5.1	Design Evaluation	NOTE 3(a)	NA	B&W			
II.E.5.2	B&W Reactor Transient Response Task Force	NOTE 3(a)	NA	B&W			
II.E.6	In Situ Testing of Valves						
II.E.6.1	Test Adequacy Study	NOTE 3(a)	All	All		06/--/89	06/--/89
II.F	INSTRUMENTATION AND CONTROLS						
II.F.1	Additional Accident Monitoring Instrumentation	I	All	All	F-20, F-21 F-22, F-23 F-24, F-25	09/13/79	09/27/79
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	I	All	All	F-26	070/2/79	09/27/79
II.F.3	Instruments for Monitoring Accident Conditions	NOTE 3(a)	All	All		NA	12/--/80
II.G	ELECTRICAL POWER						
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	I	NA	All		09/13/79	09/27/79
II.J	GENERAL IMPLICATIONS OF TMI FOR DESIGN AND CONSTRUCTION ACTIVITIES						
II.J.4	Revise Deficiency Reporting Requirements						
II.J.4.1	Revise Deficiency Reporting Requirements	NOTE 3(a)	All	All		07/31/91	07/31/91
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	NOTE 3(a)	All	All		03/31/80	NA
II.K.1(2)	Review Transients Similar to TMI-2 That Have Occurred at Other Facilities and NRC Evaluation of Davis-Besse Event	NOTE 3(a)	NA	B&W		03/31/80	NA
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	NOTE 3(a)	NA	All		03/31/80	NA

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
II.K.1(4)	Review Operating Procedures and Training Instructions	NOTE 3(a)	All	All		03/31/80	NA
II.K.1(5)	Safety-Related Valve Position Description	NOTE 3(a)	All	All		03/31/80	03/31/80
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	NOTE 3(a)	All	All		03/31/80	NA
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	NOTE 3(a)	NA	B&W		03/31/80	NA
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	NOTE 3(a)	NA	B&W		03/31/80	NA
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred out of Containment Inadvertently	NOTE 3(a)	All	All		03/31/80	NA
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	NOTE 3(a)	All	All		03/31/80	03/31/80
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	NOTE 3(a)	All	All		03/31/80	NA
II.K.1(12)	One Hour Notification Requirement and Continuous Communications Channels	NOTE 3(a)	All	All			NA
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	NOTE 3(a)	All	All		01/01/81	01/01/81
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	NOTE 3(a)	GE	CE, W		03/31/80	NA
II.K.1(15)	For Facilities with Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW	NOTE 3(a)	NA	CE, W		NA	
II.K.1(16)	Implement Procedures That Identify PRZ PORV "Open" Indications and That Direct Operator to Close Manually at "Reset" Setpoint	NOTE 3(a)	NA	CE, W		NA	
II.K.1(17)	Trip PZR Level Bistable so That PZR Low Pressure Will Initiate Safety Injection	NOTE 3(a)	NA	W			
II.K.1(18)	Develop Procedures and Train Operators on Methods of Establishing and Maintaining Natural Circulation	NOTE 3(a)	NA	B&W		NA	
II.K.1(19)	Describe Design and Procedure Modifications to Reduce Likelihood of Automatic PZR PORV Actuation in Transients	NOTE 3(a)	NA	B&W		03/31/80	NA
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	NOTE 3(a)	NA	B&W		03/31/80	03/31/80

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor Trip for LOFW, TT, or Significant Decrease in SG Level	NOTE 3(a)	NA	B&W		03/31/80	03/31/80
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	NOTE 3(a)	All	NA		03/31/80	03/31/80
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	NOTE 3(a)	All	NA		03/31/80	03/31/80
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	NOTE 3(a)	NA	All		NA	
II.K.1(25)	Develop Operator Action Guidelines	NOTE 3(a)	NA	All		NA	
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	NOTE 3(a)	NA	All		NA	
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	NOTE 3(a)	NA	All		NA	
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	NOTE 3(a)	NA	All		01/01/81	01/01/82
II.K.2	Commission Orders on B&W Plants	-	-	-	-	-	-
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System	NOTE 3(a)	NA	B&W		NA	
II.K.2(2)	Procedures and Training to Initiate and Control AFW Independent of Integrated Control System	NOTE 3(a)	NA	B&W		NA	
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	NOTE 3(a)	NA	B&W		NA	
II.K.2(4)	Small-Break LOCA Analysis, Procedures and Operator Training	NOTE 3(a)	NA	B&W		NA	
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	NOTE 3(a)	NA	B&W		NA	
II.K.2(6)	Reevaluate Analysis for Dual-Level Setpoint Control	NOTE 3(a)	NA	B&W		NA	
II.K.2(7)	Reevaluate Transient of September 24, 1977	NOTE 3(a)	NA	B&W		NA	
II.K.2(9)	Analysis and Upgrading of Integrated Control System	I	NA	B&W	F-27	01/01/81	01/01/81
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	I	NA	B&W	F-28	01/01/81	01/01/81
II.K.2(11)	Operator Training and Drilling	I	NA	B&W	F-29	01/01/81	01/01/81
II.K.2(13)	Thermal-Mechanical Report on Effect of HPI on Vessel Integrity for Small-Break LOCA With No AFW	I	NA	B&W	F-30	01/01/81	01/01/81
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	I	NA	B&W	F-31	01/01/81	01/01/81
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes After Primary System Voiding	I	NA	B&W		06/01/80	06/01/80
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA With Loss of Offsite Power	I	NA	B&W	F-32	06/01/80	06/01/80

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Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
II.K.2(17)	Analysis of Potential Voiding in RCS During Anticipated Transients	I	NA	B&W	F-33	NA	
II.K.2(19)	Benchmark Analysis of Sequential AFW Flow to Once-Through Steam Generator	I	NA	B&W	F-34	01/01/81	NA
II.K.2(20)	Analysis of Steam Response to Small-Break LOCA That Causes System Pressure to Exceed PORV Setpoint	I	NA	B&W	F-35	01/01/81	NA
II.K.2(21)	LOFT L3-1 Predictions	NOTE 3(a)	NA	B&W		NA	
II.K.3	Final Recommendations of Bulletins and Orders Task Force	-	-	-	-	-	-
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	I	NA	All	F-36	07/01/81	07/01/81
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	I	NA	All	F-37	01/01/81	01/01/81
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	I	All	All	F-38	04/01/80	04/01/80
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	I	NA	All	F-39, G-01	01/01/81	01/01/81
II.K.3(7)	Evaluation of PORV Opening Probability During Overpressure Transient	I	NA	B&W		01/01/81	01/01/81
II.K.3(9)	Proportional Integral Derivative Controller Modification	I	NA	W	F-40	07/01/80	07/01/80
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	I	NA	W	F-41		
II.K.3(11)	Control Use of PORV Supplied by Control Components, Inc. Until Further Review Complete	I	All	All			
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	I	NA	W	F-42	07/01/80	07/01/80
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	I	GE	NA	F-43	10/01/80	10/01/80
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	I	GE	NA	F-44	01/01/81	NA
II.K.3(15)	Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	I	GE	NA	F-45	01/01/81	01/01/81
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification	I	GE	NA	F-46	01/01/81	01/01/81
II.K.3(17)	Report on Outage of ECC Systems - Licensee Report and Technical Specification Changes	I	GE	NA	F-47	01/01/81	01/01/81
II.K.3(18)	Modification of ADS Logic - Feasibility Study and Modification for Increased Diversity for Some Event Sequences	I	GE	NA	F-48	01/01/81	01/01/81
II.K.3(19)	Interlock on Recirculation Pump Loops	I	GE	NA	F-49	01/01/81	NA
II.K.3(20)	Loss of Service Water for Big Rock Point	I	GE	NA		01/01/81	NA

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level - Design and Modification	I	GE	NA	F-50	01/01/81	01/01/81
II.K.3(22)	Automatic Switchover of RCIC System Suction - Verify Procedures and Modify Design	I	GE	NA	F-51	01/01/81	01/01/81
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	I	GE	NA	F-52	01/01/82	01/01/82
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	I	GE	NA	F-53	01/01/82	01/01/82
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	I	GE	NA	F-54	10/01/80	10/01/80
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	I	GE	NA	F-55	01/01/82	01/01/82
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	I	GE	NA	F-56	04/01/81	NA
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	I	All	All	F-57	01/01/83	01/01/83
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	I	All	All	F-58	01/01/83	01/01/83
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	I	GE	NA	F-59	01/01/81	01/01/81
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	I	GE	NA	F-60	01/01/81	01/01/81
II.K.3(46)	Response to List of Concerns from ACRS Consultant	I	GE	NA	F-61	07/01/80	07/01/80
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	I	GE	NA	F-62	10/01/80	NA
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	I	All	All		10/10/79	08/19/80
III.A.1.2	Upgrade Licensee Emergency Support Facilities	-	-	-	-	-	-
III.A.1.2(1)	Technical Support Center	I	All	All	F-63	09/13/79	09/27/79
III.A.1.2(2)	On-Site Operational Support Center	I	All	All	F-64	09/13/79	09/27/79
III.A.1.2(3)	Near-Site Emergency Operations Facility	I	All	All	F-65	09/13/79	09/27/79
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	NOTE 3(a)	All	All			
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	I	All	All	F-67		

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Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants- MPA No	Operating Plants - Effective Date	Future Plants- Effective Date
			BWR	PWR			
III.A.2.2	Development of Guidance and Criteria	I	All	All	F-68		
III.A.3	Improving NRC Emergency Preparedness						
III.A.3.3	Communications	-	-	-	-	-	-
III.A.3.3(1)	Install Direct Dedicated Telephone Lines	NOTE 3(a)	All	All			
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems	NOTE 3(a)	All	All			
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	I	All	All		07/02/79	09/27/79
III.D.3	Worker Radiation Protection Improvement						
III.D.3.3	Inplant Radiation Monitoring	-	-	-	-	-	-
III.D.3.3(1)	Issue Letter Requiring Improved Radiation Sampling Instrumentation	I	All	All	F-69	09/13/79	09/27/79
III.D.3.3(2)	Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	NOTE 3(a)	All	All		09/13/79	09/27/79
III.D.3.3(3)	Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation-Monitoring Instruments	NOTE 3(a)	All	All		09/13/79	09/27/79
III.D.3.3(4)	Issue a Regulatory Guide	NOTE 3(a)	All	All		09/13/79	09/27/79
III.D.3.4	Control Room Habitability	I	All	All	F-70	05/07/80	06/26/80
	TASK ACTION PLAN ITEMS						
A-1	Water Hammer (former USI)	NOTE 3(a)	All	All		NA	03/15/84
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems (former USI)	NOTE 3(a)	NA	All	D-10	01/--/81	01/--/81
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	NOTE 3(a)	NA	W		04/17/85	04/17/85
A-4	CE Steam Generator Tube Integrity (former USI)	NOTE 3(a)	NA	CE		04/17/85	04/17/85
A-5	B&W Steam Generator Tube Integrity (former USI)	NOTE 3(a)	NA	B&W		04/17/85	04/17/85
A-6	Mark I Short-Term Program (former USI)	NOTE 3(a)	GE	NA		12/--/77	NA
A-7	Mark I Long-Term Program (former USI)	NOTE 3(a)	GE	NA	D-01	08/--/82	08/--/82
A-8	Mark II Containment Pool Dyanmic Loads - Long Term Program (former USI)	NOTE 3(a)	GE	NA		08/--/81	08/--/81
A-9	ATWS (former USI)	NOTE 3(a)	All	All		06/26/84	06/26/84

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
A-10	BWR Feedwater Nozzle Cracking (former USI)	NOTE 3(a)	All	NA	B-25	11/--/80	11/--/80
A-11	Reactor Vessel Materials Toughness (former USI)	NOTE 3(a)	All	All		10/--/82	NA
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (former USI)	NOTE 3(a)	NA	All		NA	TBD
A-13	Snubber Operability Assurance	NOTE 3(a)	All	All	B-17, B-22	1980	1980
A-16	Steam Effects on BWR Core Spray Distribution	NOTE 3(a)	GE	NA	D-12	NA	
A-24	Qualification of Class 1E Safety Related Equipment (former USI)	NOTE 3(a)	All	All	B-60	08/--/81	08/--/81
A-25	Non-Safety Loads on Class 1E Power Sources	NOTE 3(a)	All	All		09/--/78	
A-26	Reactor Vessel Pressure Transient Protection (former USI)	NOTE 3(a)	NA	All	B-04	09/--/78	09/--/78
A-28	Increase in Spent Fuel Pool Storage Capacity	NOTE 3(a)	All	All		04/17/78	NA
A-31	RHR Shutdown Requirements (former USI)	NOTE 3(a)	All	All		05/--/78	10/01/78
A-35	Adequacy of Offsite Power Systems	NOTE 3(a)	All	All	B-23	06/02/77	1980
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	NOTE 3(a)	All	All	C-10, C-15	07/--/80	07/--/80
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	NOTE 3(a)	GE	NA		02/29/80	09/30/80
A-40	Seismic Design Criteria (former USI)	NOTE 3(a)	All	All		TBD	09/--/89
A-42	Pipe Cracks in Boiling Water Reactors (former USI)	NOTE 3(a)	All	NA	B-05	02/--/81	02/--/81
A-43	Containment Emergency Sump Performance (former USI)	NOTE 3(a)	NA	All		NA	11/--/85
A-44	Station Blackout (former USI)	NOTE 3(a)	All	All		TBD	06/--/88
A-46	Seismic Qualification of Equipment in Operating Plants (former USI)	NOTE 3(a)	All	All		02/--/87	NA
A-47	Safety Implications of Control Systems (former USI)	NOTE 3(a)	All	All		09/20/89	09/20/89
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	NOTE 3(a)	All	W		12/--/81	12/--/81
A-49	Pressurized Thermal Shock (former USI)	NOTE 3(a)	NA	All	A-21	TBD	07/--/85
B-10	Behavior of BWR Mark III Containments	NOTE 3(a)	GE	NA		NA	09/--/84
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	NOTE 3(a)	All	All		03/--/78	
B-56	Diesel Reliability	NOTE 3(a)	All	All	D-19	06/--/93	06/--/93
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	NOTE 3(a)	All	All	B-45	04/20/81	
B-64	Decommissioning of Reactors	NOTE 3(a)	All	All		06/27/88	NA
B-66	Control Room Infiltration Measurements	NOTE 3(a)	All	All		NA	07/--/81
C-1	Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	NOTE 3(a)	All	All		05/27/80	05/27/80

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
C-10	Effective Operation of Containment Sprays in a LOCA	NOTE 3(a)	All	All		NA	
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	NOTE 3(a)	All	All		12/27/82	12/27/82
	NEW GENERIC ISSUES						
25.	Automatic Air Header Dump on BWR Scram System	NOTE 3(a)	All	NA		01/09/81	01/09/81
40.	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	NOTE 3(a)	All	NA	B-65	08/31/81	08/31/81
41.	BWR Scram Discharge Volume Systems	NOTE 3(a)	All	NA	B-58	12/09/80	NA
43.	Reliability of Air Systems	NOTE 3(a)	All	All	B-107	08/08/88	08/08/88
45	Inoperability of Instrumentation Due to Extreme Cold Weather	NOTE 3(a)	All	All		NA	09/01/83
51.	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	NOTE 3(a)	All	All	L-913	07/18/89	07/18/89
67.	Steam Generator Staff Actions	-	-	-	-	-	-
67.3.3	Improved Accident Monitoring	NOTE 3(a)	All	All	A-17	12/17/82	12/17/82
70.	PORV and Block Valve Reliability	NOTE 3(a)	NA	All		06/25/90	06/25/90
73.	Detached Thermal Sleeves	NOTE 3(a)	NA	W		NA	
75.	Generic Implications of ATWS Events at the Salem Nuclear Plant	NOTE 3(a)	All	All	B-76, B-77, B-78, B-79, B-80, B-81, B-82, B-85, B-86, B-87, B-88, B-89, B-90, B-91, B-92, B-93	07/08/83	TBD
86.	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	NOTE 3(a)	All	NA	B-84	TBD	TBD
87.	Failure of HPCI Steam Line Without Isolation	NOTE 3(a)	All	All		06/28/89	06/28/89
89.	Stiff Pipe Clamps	NOTE 6	All	All	NA	NA	TBD
93.	Steam Binding of Auxiliary Feedwater Pumps	NOTE 3(a)	NA	All	B-98	10/--/85	10/--/85
94.	Additional Low Temperature Overpressure Protection for Light Water Reactors	NOTE 3(a)	NA	CE, W		06/25/90	06/25/90
99.	RCS/RHR Suction Line Valve Interlock on PWRs	NOTE 3(a)	NA	All	L-817	10/17/88	10/17/88
103.	Design for Probable Maximum Precipitation	NOTE 3(a)	All	All		10/19/89	10/19/89
118.	Tendon Anchorage Failure	NOTE 3(a)	All	All	NA	NA	07/--/90
124.	Auxiliary Feedwater System Reliability	NOTE 3(a)	All	All		TBD	TBD
128.	Electrical Power Reliability	NOTE 3(a)	All	All		04/29/91	04/29/91
130.	Essential Service Water Pump Failures at Multiplant	NOTE 3(a)	NA	All		09/19/91	09/19/91

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
	Sites						
155	Generic Concerns Arising from TMI-2 Cleanup	-	-	-	-	-	-
155.1	More Realistic Source Term Assumptions	NOTE 3(a)	All	All	NA	NA	02/--/95
156	Systematic Evaluation Program	-	-	-	-	-	-
156.6.1	Pipe Break Effects on Systems and Components	HIGH	All	All		TBD	TBD
163.	Multiple Steam Generator Tube Leakage	HIGH	NA	All		TBD	TBD
177.	Vehicle Intrusion at TMI	NOTE 3(a)	All	All		08/01/94	08/01/94
186.	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants	CONTINUE	All	All		TBD	TBD
189.	Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Combustion During A Severe Accident	CONTINUE	All	All		TBD	TBD
191.	Assessment of Debris Accumulation on PWR Sump Performance	HIGH	NA	All		TBD	TBD
193.	BWR ECCS Suction Concerns	CONTINUE	All	NA		TBD	TBD
199.	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States	NOTE 4	All	All		TBD	TBD
	HUMAN FACTORS ISSUES						
HF1	STAFFING AND QUALIFICATIONS						
HF.1.1	Shift Staffing	NOTE 3(a)	All	All		01/--/84	01/--/84

APPENDIX F

NUCLEAR MATERIAL SAFETY AND SAFEGUARDS GSIs

This appendix documents those non-reactor GSIs identified, prioritized, and resolved by NMSS. As stated in SECY-98-001,[1724] the prioritization procedure for these issues is contained in NMSS Policy and Procedures Letter 1-57,[1725] "NMSS Generic Issues Program."

TABLE F.1

LISTING OF NMSS GSIs

This table contains the priority designations for all NMSS GSIs listed in Appendix F.

Legend

NOTES: 3(a) - Resolution Resulted in the Establishment of New Requirements
3(b) - Resolution Resulted in the Establishment of No New Requirements
4 - Issue to be Prioritized in the Future
HIGH - High Safety Priority
MEDIUM - Medium Safety Priority
LOW - Low Safety Priority

Issue No.	Title	Priority Engineer	LeadOffice/ Division/Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date
NMSS-0001	Door Interlock Failure Resulting from Faulty MicroSelectron-High Dose Rate Remote Afterloader	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0002	Significant Quantities of Fixed Contamination Remain in Krypton-85 Leak-Detection Devices After Venting	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0003	Corrosion of Sealed Sources Caused by Sensitization of Stainless Steel Source Capsules During Shipment	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0004	Overexposures Caused by Sources Stolen from Facility of Bankrupt Licensee	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0005	Potential for Erroneous Calibration, Dose Rate, or Radiation Exposure Measurements With Victoreen Electrometers	Ramsey	NMSS/IMNS/IMOB	NOTE 3(a)		12/31/1998
NMSS-0006	Criticality in Low-Level Waste	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)	1	06/30/2000
NMSS-0007	Criticality Benchmarks Greater Than 5% Enrichment	Ramsey	NMSS/FCSS	HIGH	1	06/30/2001
NMSS-0008	Year 2000 Computer Problem - Non-Reactor Licensees	Ramsey	NMSS/IMNS	NOTE 3(b)	1	06/30/2000
NMSS-0009	Amersham Radiography Source Cable Failures	Ramsey	NMSS/IMNS	NOTE 3(b)		12/31/1998
NMSS-0010	Troxler Gauge Source Rod Weld Failures	Ramsey	NMSS/IMNS	NOTE 3(b	1	06/30/2002

Revision 5

Issue No.	Title	Priority Engineer	LeadOffice/ Division/Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date
NMSS-0011	Spent Fuel Dry Cask Weld Cracks	Ramsey	NMSS/SFPO	NOTE 3(b)		12/31/1998
NMSS-0012	Inadequate Transportation Packaging Puncture Tests	Ramsey	NMSS/SFPO	NOTE 3(b)	1	06/30/2000
NMSS-0013	Use of Different Dose Equivalent Models to Show Compliance	Ramsey	NMSS/IMNS	NOTE 3(b)	1	06/30/2000
NMSS-0014	Surety Estimates for Groundwater Restoration at In-Situ Leach Fields	Ramsey	NMSS/DWM	NOTE 3(b)	1	06/30/2007
NMSS-0015	Adequacy of 10 CFR 150 Criticality Requirements	Ramsey	NMSS/DWM	NOTE 3(b)	1	06/30/2000
NMSS-0016	Adequacy of 0.05 Weight Percent Limit in 10 CFR 40	Ramsey	NMSS/IMNS	NOTE 3(b)	1	06/30/2007
NMSS-0017	Misleading Marketing Information to General Licensees	C. Mattsen	NMSS/IMNS	NOTE 3(a)		06/30/2001
NMSS-0018	Problems Encountered When Manually Editing Treatment Planning Data on Nucletron MicroSelectron-HDR Model 105.999	B. Ayres	NMSS/IMNS	NOTE 3(b)		06/30/2001
NMSS-0019	Control Unit Failures of Classic Nucletron HDR Units	B. Ayres	NMSS/IMNS	NOTE 3(b)		06/30/2001
NMSS-0020	Leaking Pools	M. Sitek	NMSS/IMNS	DROP		06/30/2001
NMSS-0021	Unlikely Events	M. Sitek	NMSS/IMNS	DROP		06/30/2001
NMSS-0022	Gamma Stereotactic Radiosurgery	M. Sitek	NMSS/IMNS	DROP		06/30/2001

NMSS-0014: SURETY ESTIMATES FOR GROUNDWATER RESTORATION AT IN-SITU LEACH FACILITIES

DESCRIPTION

This issue was identified[1723] by NMSS to pursue research to provide a methodology to calculate surety for groundwater restoration activities at in situ leach uranium extraction facilities and a post-restoration groundwater quality stability monitoring methodology. The following tasks were envisioned: (1) review approaches used to estimate pore volumes and to calculate surety amounts and obtain data to evaluate these approaches; (2) develop a pore volume estimation methodology and document it in a NUREG report; (3) develop cost estimation methodology for use in evaluating the level of financial surety required; (4) brief regulators on the surety methodology; (5) review the existing approaches used to determine an appropriate time period for post-restoration monitoring period and obtain datasets to evaluate the methodologies; (6) use the datasets to develop and test the methodologies; (7) develop a robust methodology; and (8) transfer the methodology to regulators through briefings and a NUREG report.

CONCLUSION

This issue was given a medium priority ranking and resolution was pursued.[1723] The staff's work resulted in the publication of NUREG/CR-6870[1892] which provides a useful perspective on the costs associated with groundwater restoration at in-situ leach facilities, and describes several methods for estimating these costs at new facilities, including a rigorous approach for development of a site-specific model that considers groundwater flow, solute transport, and geochemical reactions in the uranium ore zone of interest. With this modeling approach, groundwater restoration volumes and, correspondingly, their associated costs, can be estimated. As such, NUREG/CR-6870[1892] provides the methodology originally requested by the Uranium Recovery Program. Thus, the issue was resolved with no new requirements or guidance for licensees.[1893]

REFERENCES

1723. Memorandum to J. Craig from F. Combs, "Submittal of New Generic Issues for Tracking in the Generic Issues Management and Control System (GIMCS)," July 23, 1998.

1892. NUREG/CR-6870, "Consideration of Geochemical Issues in Groundwater Restoration at Uranium In-Situ Leach Mining Facilities," U.S. Nuclear Regulatory Commission, January 2007.

1893. Memorandum to L. Reyes from C. Miller, "Closeout of Generic Safety Issue NMSS-0014, 'Surety Estimates for Groundwater Restoration at In-Situ Leach Facilities,'" May 25, 2007. [ML070790303]

NMSS-0016: ADEQUACY OF 0.05 WEIGHT PERCENT LIMIT IN 10 CFR 40

DESCRIPTION

Exposure to the "unimportant quantities" of source material, defined in 10 CFR 40.13(a) as less than 0.05 wt% uranium or thorium, could result in annual doses to the public of hundreds of millirem exceeding NRC's public dose limit of 100 mem/year from all sources. In July 1996, NMSS developed a Draft User Need memorandum requesting development of a regulation to limit the transfer of source material meeting the "unimportant quantity" limit or to revise the definition of source material. Discussions in 1996 and 1997 with RES and OGC indicated that there were several options available to the staff to revise the definition of source material. However, the User Need memorandum was never finalized.

Subsequently, the Division of Fuel Cycle Safety (FCSS)/NMSS received a licensee request to transfer baghouse dust containing less than 0.05 wt% uranium and thorium to an exempt person per 10 CFR 40.51(b)(3) and 40.13(a). Some conservative dose estimates indicated that the transfer could result in doses exceeding the public dose limit. This issue was identified[1723] by NMSS to pursue rulemeking to immediately cease transfers under 10 CFR 40.51(b)(3) and 40.51(b)(4) of source material exempted under 10 CFR 40.13(a). By eliminating these provisions, any future transfers would have to meet existing general license conditions or be specifically approved on a case-by-case basis.

CONCLUSION

This issue was given a medium priority ranking and resolution was pursued.[1723] The staff initially pursued a modification to 10 CFR 40.51, "Transfer of source or byproduct material," to codify the Commission's direction given in various SRMs. However, in November 2006, NMSS concluded that the existing NRC policy for handling the source material satisfied the intent of the Commission directives, and the issue was closed by the Office of Federal and State Materials and Environmental Management Programs (FSME)/NMSS with no new requirements or guidance for licensees. The staff plans to take further action related to the proposed rule (i.e., publish the final rule or rescind the proposed rule) when the EPA finalizes activities related to its ANPR, and/or the recommendation of the Interagency Jurisdictional Working Group(IJWG) is implemented.[1894]

REFERENCE

1723. Memorandum to J. Craig from F. Combs, "Submittal of New Generic Issues for Tracking in the Generic Issues Management and Control System (GIMCS)," July 23, 1998.

1894. Memorandum to L. Reyes from C. Miller, "Closeout of Generic Safety Issue NMSS-016, 'Adequacy of 0.05 Weight Percent Limit in 10 CFR 40,'" November 27, 2006. [ML062000507]

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report presents the safety priority rankings for generic safety issues related to nuclear power plants. The purpose of these rankings is to assist in the timely and efficient allocation of NRC resources for the resolution of those safety issues that have a significant potential for reducing risk. The safety priority rankings are HIGH, MEDIUM, LOW, DROP, and CONTINUE, and have been assigned on the basis of risk significance estimates, the ratio of risk to costs and other impacts estimated to result if resolution of the safety issues were implemented, and the consideration of uncertainties and other quantitative or qualitative factors. To the extent practical, estimates are quantitative.

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