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SUPPLEMENT 28 TO NUREG-0933
"A PRIORITIZATION OF GENERIC SAFETY ISSUES"

REVISION INSERTION INSTRUCTIONS

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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

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
<u>I.B.</u>	<u>SUPPORT PERSONNEL</u>						
<u>I.B.1</u>	<u>Management for Operations</u>						
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
<u>I.B.2</u>	<u>Inspection of Operating Reactors</u>						
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. Sego	OIE/DQASIP/ORPB	LI (NOTE 3)	1	12/31/97	NA
<u>I.C</u>	<u>OPERATING PROCEDURES</u>						
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
<u>I.D</u>	<u>CONTROL ROOM DESIGN</u>						
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
<u>I.G</u>	<u>PREOPERATIONAL AND LOW-POWER TESTING</u>						
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
<u>II.A</u>	<u>SITING</u>						
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
<u>II.B</u>	<u>CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW</u>						
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	
<u>II.C</u>	<u>RELIABILITY ENGINEERING AND RISK ASSESSMENT</u>						
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.
<u>II.E.5</u>	<u>Design Sensitivity of B&W Reactors</u>						
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	
<u>II.E.6</u>	<u>In Situ Testing of Valves</u>						
II.E.6.1	Test Adequacy Study	D. Thatcher	RES/DE/EIB	NOTE 3(a)	2	12/31/98	
<u>II.F</u>	<u>INSTRUMENTATION AND CONTROLS</u>						
II.F.1	Additional Accident Monitoring Instrumentation	-	NRR/DL	I	3	12/31/98	F-20, F-21, F-22, F-23, F-24, F-25
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	-	NRR/DL	I	3	12/31/98	F-26
II.F.3	Instruments for Monitoring Accident Conditions	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
<u>II.G</u>	<u>ELECTRICAL POWER</u>						
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	-	NRR	I	1	12/31/98	NA
<u>II.H</u>	<u>TMI-2 CLEANUP AND EXAMINATION</u>						
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact	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
<u>III.D.2</u>	<u>Public Radiation Protection Improvement</u>						
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
<u>III.D.3</u>	<u>Worker Radiation Protection Improvement</u>						
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

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.
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

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.
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

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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.
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	CONTINUE	3	06/30/03	
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.	<u>Davis-Besse Loss of All Feedwater Event of June 9, 1985: Long-Term Actions</u>	-	-	-			
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	HIGH	2	12/31/98	
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	HIGH		06/30/01	
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	Continue		06/30/02	
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 4		(Later)	
	HUMAN FACTORS ISSUES						
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	
HF1.3	Guidance on Limits and Conditions of Shift Work	J. Pittman	NRR/DHFT/HFIB	NOTE 3(b)	2	06/30/89	
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						

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.
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						
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

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.
	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
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	-	-				

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.
CH3.2A	Filtered Venting	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
<u>CH4</u>	<u>EMERGENCY PLANNING</u>						
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
<u>CH5</u>	<u>SEVERE ACCIDENT PHENOMENA</u>						
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
<u>CH6</u>	<u>GRAPHITE-MODERATED REACTORS</u>						
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

TABLE III

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 (276)														
GSI	-	54	0	0	83	0	5	0	4	99	4	1	-	250
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	417	0	5	0	16	123	4	1	64	846

ITEM A-36: CONTROL OF HEAVY LOADS NEAR SPENT FUEL

DESCRIPTION

At all nuclear plants, overhead cranes are used to lift heavy objects in the vicinity of spent fuel. If a heavy object such as a spent fuel shipping cask or shielding block were to fall onto spent fuel in the storage pool or reactor core during refueling and damage the fuel, there could be a release of radioactivity to the environment. Such an occurrence also has the potential for overexposing plant personnel to radiation. If the dropped object were large and the damaged fuel contained a considerable amount of undecayed fission products, radiation releases to the environment could exceed 10 CFR Part 100 guidelines. With the advent of increased and longer-term storage of spent fuel, the NRC determined that there was a need for a systematic review of requirements, facility designs, and TS regarding the movement of heavy loads to assess safety margins and improve them where necessary. This item was originally identified in NUREG-0371[2] and was later determined to be a USI in NUREG-0510.[186]

CONCLUSION

This USI was RESOLVED in December 1980 with the issuance of GL 80-113,[1842] following publication of the report on the NRC review of nuclear power plant load-handling operations in NUREG-0612.[747] GL 80-113[1842] requested licensees to review their controls for the handling of heavy loads to determine the extent to which guidelines in NUREG-0612[747] were being satisfied, and to identify the changes and modifications that would be required in order to fully satisfy the guidelines. GL 81-07[1843] was subsequently issued to correct errors in GL 80-113.[1842]

Licensee responses to NUREG-0612[747] were requested in two parts: Phase I (6-month response); and Phase II (9-month response). For operating plants, MPAs C-10 and C-15 were established by NRR/DL to track the implementation of Phases I and II, respectively.[60] For future plants, SRP[11] Section 9.1.5. was revised. At the completion of Phase II, the results of the NRC findings were published in GL 85-11.[1844] However, in April 1999, Issue 186 was identified to address the NRR concern that licensees operating within the GL 85-11[1844] regulatory guidelines were not taking adequate measures to assess and mitigate the consequences of dropped heavy loads.

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.

11. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (1st Edition) November 1975, (2nd Edition) March 1980, (3rd Edition) July 1981.

60. NUREG-0606, "Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission, Vol. 7 No. 3, August 1985.

186. NUREG-0510, "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants," U.S. Nuclear Regulatory Commission, January 1979.

747. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants Resolution of Generic Technical Activity A-36," U.S. Nuclear Regulatory Commission, July 1980.

1842. NRC Letter to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, "Control of Heavy Loads," (Generic Letter 80-113) December 22, 1980.

1843. NRC Letter to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, "Control of Heavy Loads (Generic Letter 81-07)," February 3, 1981.

1844. NRC Letter to All Licensees for Operating Reactors, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants,' NUREG-0612 (Generic Letter 85-11)" June 28, 1985.

ISSUE 82: BEYOND DESIGN BASIS ACCIDENTS IN SPENT FUEL POOLS

DESCRIPTION

Historical Background

The risks of beyond design basis accidents in the spent fuel storage pool were examined in WASH-1400[16] (App. I, pp. I-96ff). It was concluded that these risks were orders of magnitude below those involving the reactor core. The basic reason for this was the simplicity of the spent fuel storage pool -- the coolant is at atmospheric pressure, the spent fuel is always subcritical and the heat source is low, there is no piping which can drain the pool, and there are no anticipated operational transients that could interrupt cooling or cause criticality.

The reasons for reexamination of spent fuel storage pool accidents were two-fold. First, spent fuel is being stored instead of reprocessed. This has led to the expansion of onsite fuel storage by means of high density storage racks, which results in a larger inventory of fission products in the pool, a greater heat load on the pool cooling system, and less distance between adjacent fuel assemblies. Second, some laboratory studies have provided evidence of the possibility of fire propagation between assemblies in an air-cooled environment.[543,544] These two reasons, put together, provided the basis for an accident scenario which was not previously considered.

Safety Significance

A typical spent fuel storage pool with high density storage racks can hold roughly five times the fuel in the core. However, since reloads typically discharge one third of a core, much of the spent fuel stored in the pool will have had considerable decay time. This reduces the radioactive inventory somewhat. More importantly, after roughly three years of storage, spent fuel can be air-cooled, i.e., such fuel need not be submerged to prevent melting. (Submersion is still desirable for shielding and to reduce airborne activity, however.)

If the pool were to be drained of water, the discharged fuel from the previous two refuelings would still be "fresh" enough to melt under decay heat. However, the zircaloy cladding of this fuel could be ignited during the heatup.[543] The resulting fire, in a pool equipped with high density storage racks, would probably spread to most or all of the fuel in the pool. The heat of combustion, in combination with decay heat, would certainly release considerable gap activity from the fuel and would probably drive "borderline aged" fuel into a molten condition. Moreover, if the fire becomes oxygen-starved (quite probable for a fire located in the bottom of a pit such as this), the hot zirconium would rob oxygen from the uranium dioxide fuel, forming a liquid mixture of metallic uranium, zirconium, oxidized zirconium, and dissolved uranium dioxide. This would cause a release of fission products from the fuel matrix quite comparable to that of molten fuel.[545] In addition, although confined, spent fuel pools are almost always located outside of the primary containment. Thus, release to the atmosphere is more likely than for comparable accidents involving the reactor core.

Possible Solutions

At the time of the evaluation of this issue in December 1983, no generic solution to the potential problem had yet been identified. Several possibilities existed, however. The first possibility was to reprocess the spent fuel and thus reduce the inventory in the pool. Second, the pool could be compartmentalized by installing partitions (and individual coolant supply diffusers for each compartment) thus limiting the extent of an accident. Third, spray headers could be installed to provide cooling even when the pool is drained and not refloodable.

PRIORITY DETERMINATION

LWR spent fuel storage pools do not differ greatly. None are equipped with drains; a portable pump must be brought in when it is desired to empty the pool. The cooling systems are provided with anti-siphoning devices (check valves and/ or anti-siphoning holes) so that pipe breaks in the cooling system will not drain the pool. All are seismic Category I. One difference does exist: PWR pools are generally below grade (often on bedrock) while BWR pools are considerably above grade. Thus, even a hole in the bottom of the pool will not rapidly drain a PWR pool. This priority determination, therefore, is concentrated on a BWR pool because of its (somewhat) greater vulnerability.

Frequency Estimate

BWR spent fuel can be uncovered either by extended loss of pool cooling, which results in boiloff, or by an accident which drains the pool. Both mechanisms were considered.

Typically, a BWR spent fuel storage pool has no drains. Instead, coolant is withdrawn at the surface by skimmers which conduct the water into two surge tanks. The cooling system consists of two pumps and two heat exchangers which reject heat to the RBCCW system. These are not independent trains. The suction on the surge tanks is common and flow from the heat exchangers is combined to go through one filter/demineralizer before it is returned to the spent fuel pool. Return is by means of a set of diffusers located near the bottom of the pool. The piping connected to the diffusers contains check valves or some other antisiphoning device.

Immediately after a refueling, both pumps and heat exchangers are usually needed. After a few months of decay, the heat load will diminish to the point where only one pump and heat exchanger are needed. Water makeup is normally via the condensate transfer system which is connected to one of the surge tanks.

The spent fuel pool cooling system is cross-connected to one train of the RHR system at both inlet and outlet. The primary reasons for this is to allow use of RHR for supplementary fuel pool cooling during periods when an entire reactor core is off-loaded. However, this also provides a backup means of pool cooling. In addition, since the RHR suction can be lined up to the condensate storage tank or even to river water, RHR also provides a backup means of maintaining pool water inventory.

Control and operation of the spent fuel pool cooling system and RHR cross-ties are not performed from the control room; most of the valves involved are manually operated. However, if pool cooling is lost, it will take over two days for the pool temperature to rise to boiling and at least two days more for the level to drop to the top of the fuel assemblies, even under design heat load conditions. Moreover, there are level alarms on the surge tanks and the pool itself in the control room. Thus, even though the systems are not automatic, the long time intervals involved should be sufficient to prevent problems with human confusion, etc.

WASH-1400[16] estimated the frequency of loss of one spent fuel pool cooling "train" to be 0.1/RY. We will assume, based on experience with other systems, that the conditional probabilities of the second "train" also failing due to a common-mode problem is 5%, and due to a random failure, 1.5%. In addition to this, the second pump and heat exchanger are in use (i.e., are not a redundant backup) about 30% of the time. Thus, the combined frequency of a pool heatup event is 3.7×10^{-2}/RY.

To go from a pool heatup event to an event that threatens the fuel, several other failures must occur. First, the RHR system must fail, both as a cooling system and as a supply of makeup water. For this, we assume a conditional probability of 1.5%, based on RHR reliability in the LPCI mode.[16] Second, the condensate transfer system could be used as a makeup system, either by supply to the fuel pool cooling system suction or (if the pool cooling system is isolated) by overfilling the surge tanks and causing backflow into the fuel pool. Since the condensate system is not powered by emergency power buses, it may well be put out of service by any common mode failure of the spent fuel pool cooling system. Thus, we will assume a conditional failure probability of 5% for the condensate transfer system.

Ultimately, makeup to the pool could be supplied by bringing in a fire hose (60 gpm would suffice). Although one would expect that the failure probability associated with bringing in a hose (over a period of four or more days) would be very low, it must also be remembered that working next to 385,000 gallons of potentially contaminated boiling water on top of a 10-story building is not a trivial problem. We will assume, based purely on judgment, that the conditional failure probability for this method of makeup is on the order of 5%. When these probabilities are combined, the result is a frequency of 1.4×10^{-6}/RY for an accident initiated by loss of spent fuel pool cooling.

Several events could cause an accident by draining the pool. We will first examine those events which are not likely to cause gross failure of the confinement system. First, there is the possibility of a break in the cooling system (beyond the condensate transfer makeup capacity) which we estimate to happen no more often than once per thousand reactor-years (the "S2" frequency). To drain the pool, the anti-siphoning check valves must fail (conditional probability of 8%, based on a German component failure study) and there must be a failure of the pool cooling system to isolate (conditional failure probability of 1%, based purely on judgment). RHR should provide sifficient makeup, since each RHR pump can supply 10,000 gpm and normal maximum fuel pool flow is 1200 gpm. However, RHR may be inoperable, for which we assume a conditional probability of 1.5% (based on WASH-1400).[16] When these figures are combined, the siphoning scenario is estimated to occur with a frequency of 1.2×10^{-8}/RY.

In addition, the pool could be drained by a cask drop accident (2.5×10^{-7}/RY, from WASH-1400)[16] or a turbine missile (4.1×10^{-7}/RY, also from WASH-1400).[16] Here, the RHR might not have sufficient capacity and the time frame is not as long as the previous scenarios. Based again on judgment, it was assumed that the combined RHR conditional failure probability is 10%. This gives an accident frequency of 6.6×10^{-8}/RY. If the 1.2×10^{-8}/RY from the siphoning scenario is added, the total frequency for this class of accidents is 7.8×10^{-8}/RY.

Finally, we come to two scenarios which could open up the pool to the atmosphere as well as drain it. First, there is the tornado missile ($<5 \times 10^{-6}$/RY, from WASH-1400).[16] This should not simultaneously cause failure of RHR. However, RHR may be otherwise inoperable (in this shorter time frame) or have insufficient capacity. It was assumed that the combined RHR conditional failure probability is 5%. This gives an accident frequency of 2.5×10^{-7}/RY. Second, a seismic event could breach the pool. The WASH-1400[16] estimate for this is 10^{-5} to 10^{-7}/RY, depending on the site. We will use the higher figure, recognizing that this will limit the number of sites to which the analysis will apply.

After a seismic event severe enough to breach a seismic Category I spent fuel pool, the probability of RHR failure is higher than that of our previous scenarios. Moreover, the RHR might not be able to supply enough makeup. Finally, the time frame is very short, considering that manual valves must be opened and other earthquake-induced problems may be distracting plant personnel. We will assume that 90% of the time the draining rate will be slow enough to both be within the capacity of RHR makeup and also allow operator diagnosis and the necessary manual lineup of RHR to the pool. We will further assume a 90% probability of RHR remaining operable after the earthquake. This gives a total failure conditional probability of 19%.

Thus, for a site with a high seismic probability, the frequency of earthquake-induced accidents is estimated to be 1.9×10^{-6}/RY. Adding the tornado-induced accident frequency to this, we get a frequency for this class of accidents of 2.2×10^{-6}/RY.

<u>Consequence Estimate</u>

A BWR spent fuel storage pool with high density racks may contain almost 3500 fuel bundles, which is about 4½ times the inventory of the reactor core. Thus, an accident in the spent fuel pool can threaten much more fuel than a reactor accident. Compensating for this is the fact that much of the stored spent fuel has had considerable time for decay of hazardous radioactive fission products. To estimate the hazard to the public from melting of the spent fuel pool inventory, special CRAC2 runs were performed for the NRC by PNL, using a uniform population density of 340 persons per square mile, a central midwest plain meteorology, and no ingestion pathways. The calculations were performed for a spent fuel pool with a series of 1/3-core reload modules. The first module had one week decay time, the second, 18 months, the third, 3 years, and so on for a total of 13 modules. Cases were run using release fractions from the BWR-2, BWR-3 and BWR-4 release categories. This corresponds to release direct to atmosphere, release through a hole in the secondary containment, and release with the containment at design leakage and SGTS operable.

The results of the calculations and their corresponding frequencies from the previous section are given in the Table below:

Analagous Release Category	Frequency (/RY)	Consequences (man-rem)	Product (man-rem/RY)
BWR-2	2.2×10^{-6}	7.4×10^{6}	16.3
BWR-3	7.8×10^{-6}	6.5×10^{6}	0.5
BWR-4	1.4×10^{-6}	1.1×10^{6}	1.5
TOTAL:			18.3

It should be noted that this analysis is predicated on the assumption that the exposed elements will burn and that the fire will propagate throughout the pool. Additional research is necessary to substantiate this hypothesis. Assuming a 40-year plant life, the total risk reduction is approximately 700 man-rem/reactor.

Cost Estimate

As was discussed previously, no specific solution to the potential problem had yet been settled upon as of December 1983. However, any hardware addition probably would have had to be Seismic Category I and, thus, costs were unlikely to be less than $1M/reactor. NRC costs were estimated to be negligible compared to licensee costs.

Value/Impact Assessment

Based on an estimated risk reduction of 700 man-rem/reactor and a cost of $1M/reactor for a possible solution, the value/impact score was given by:

$$S = \frac{700 \text{ man-rem/reactor}}{\$1M/\text{reactor}}$$

$$= 700 \text{ man-rem/\$M}$$

Other Considerations

It should be noted that a low seismic probability would drop the above estimates to about 200 man-rem/reactor and 200 man-rem/$M. This would not change the final conclusion. In any case, this analysis was based on a specific pool design which was selected in an attempt to represent both generic and worst-case situations. The number of plants actually at risk may be limited.

CONCLUSION

Based on the available information and the above calculations, this item was given a medium priority ranking. Studies performed by the staff in resolving the issue showed that, although most of the spent fuel pool risk comes from beyond design basis earthquakes, this risk is no greater than the risk from core damage accidents due to seismic events beyond the safe shutdown earthquake. The staff's technical findings were published in NUREG/CR-4982,[1157] NUREG/CR-5176,[1196] and NUREG/CR-5281.[1197] The regulatory analysis published in NUREG-1353[1198] showed that there was no cost-effective alternative which, if implemented, would result in a substantial safety improvement.

The staff concluded that reducing the risk from spent fuel pools due to events beyond the SSE would still leave a comparable risk due to core damage accidents. Because of the large inherent safety margins in the design and construction of spent fuel pools, this issue was RESOLVED and no new requirements were established.[1199]

REFERENCES

16. WASH-1400 (NUREG-75/014), "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, October 1975.

543. Memorandum for T. Speis from R. Mattson, "Proposed Generic Issue on Beyond Design Basis Accidents in Spent Fuel Pools," August 10, 1983.

544. NUREG/CR-0649, "Spent Fuel Heatup Following Loss of Water During Storage," U.S. Nuclear Regulatory Commission, May 1979.

545. Memorandum for Z. Rosztoczy from P. Williams, "Trip Report: International Meeting on Severe Fuel Damage and Visit to Power Burst Facility," April 25, 1983.

1157. NUREG/CR-4982, "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82," U.S. Nuclear Regulatory Commission, July 1987.

1196. NUREG/CR-5176, "Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants," U.S. Nuclear Regulatory Commission, January 1989.

1197. NUREG/CR-5281, "Value/Impact Analyses of Accident Preventive and Mitigative Options for Spent Fuel Pools," U.S. Nuclear Regulatory Commission, March 1989.

1198. NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82 'Beyond Design Basis Accidents in Spent Fuel Pools,'" U.S. Nuclear Regulatory Commission, April 1989.

1199. Memorandum for V. Stello from E. Beckjord, "Resolution of Generic Issue, 'Beyond Design Basis Accidents in Spent Fuel Pools,'" April 24, 1989.

ISSUE 168: ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

DESCRIPTION

As discussed in SECY-93-049,[380] the staff reviewed significant license renewal issues and found that several related to environmental qualification (EQ). A key aspect of these issues was whether the licensing bases, particularly for older plants whose licensing bases differ from newer plants, should be reassessed or enhanced in connection with license renewal or whether they should be reassessed for the current license term. The staff concluded that differences in EQ requirements constituted a potential generic issue which should be evaluated for backfit independent of license renewal.[1518]

In the staff's development of an interoffice action plan to address upgrading EQ requirements for older plants during the current licensing term, the staff evaluated the technical adequacy of EQ requirements. As part of this evaluation, the staff reviewed tests of qualified cables performed by SNL, under contract with the NRC. The purpose of these tests was to determine the effects of aging on cable products used in nuclear power plants. After accelerated aging, some of the environmentally-qualified cables either failed or exhibited marginal insulation resistance during accident testing, indicating that qualification of some electric cables may have been non-conservative. Although the SNL tests may have been more severe than required by NRC regulations, the test results raised questions with respect to the EQ and accident performance capability of certain artificially-aged cables. Depending on the application, failure of these cables during or following design basis events could affect the performance of safety functions in nuclear power plants.

CONCLUSION

Based on the existence of an ongoing action plan to address the safety concern and the NRR decision[1517] to pursue its resolution, the issue was considered nearly-resolved in April 1993. It was later given a HIGH priority ranking in SECY-98-166.[1718] In accordance with an RES evaluation,[1564] the impact of a license renewal period of 20 years was to be considered in the resolution of the issue.

Accelerated aging tests on electrical equipment showed that some environmentally qualified cables either failed or exhibited marginal insulation resistance. Failure of these cables during or following a design-basis event could affect the performance of safety functions. After review and analysis of six LOCA tests, condition-monitoring tests on I&C cables, and information provided by the nuclear industry, the staff concluded that the existing equipment qualification process was adequate to ensure that I&C cables would perform their intended function. Regulatory Issue Summary 2003-09[1852] was issued in May 2003, and the issue was RESOLVED with no new requirements for licensees.[1853]

REFERENCES

380. SECY-93-049, "Implementation of 10 CFR Part 54, 'Requirements for Renewal of Operating Licenses for Nuclear Power Plants,'" March 1, 1993.

1517. Memorandum for J. Sniezek from T. Murley and E. Beckjord, "Resolution of Fatigue and Environmental Qualification Issues Related to License Renewal," April 1, 1993.

1518. Memorandum for The Chairman et al., from J. Taylor, "Environmental Qualification of Electric Equipment," May 27, 1993.

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

1718. SECY-98-166, "Summary of Activities Related to Generic Safety Issues," July 6, 1998.

1852. NRC Regulatory Issue Summary 2003-09, "Environmental Qualification of Low-Voltage Instrumentation and Control Cables," May2, 2003.

1853. Memorandum to W. Travers from R. Borchardt, "Closeout of Generic Safety Issue (GSI) 168, 'Environmental Qualification of Low-Voltage Instrumentation and Control Cables,'" August 14, 2003.

ISSUE 186: POTENTIAL RISK AND CONSEQUENCES OF HEAVY LOAD DROPS IN NUCLEAR POWER PLANTS

DESCRIPTION

Historical Background

This issue was identified[1845] by NRR in April 1999 when the concern was raised that licensees operating within the regulatory guidelines of GL 85-11[1844] may not have taken adequate measures to assess and mitigate the consequences of dropped heavy loads. Prior to the issuance of GL 85-11,[1844] GLs 80-113,[1842] 81-07,[1843] and 83-42[1847] were issued with requirements for operating licensees following the resolution of Issue A-36. In April 1996, NRC Bulletin 96-02[1848] was issued to alert licensees of potential high consequences that could result from a cask drop and to remind them of complying with existing regulatory guidelines on the control and handling of heavy loads.

Safety Significance

In nuclear plant operation, maintenance, and refueling activities, heavy loads may be handled in several plant areas. If these loads were to drop because of human error or crane failure, they could impact on stored spent fuel, fuel in the core, or on equipment that may be required to achieve safe shutdown or permit continued decay heat removal. In some instances, load drops at specific times, locations, and weights could potentially lead to offsite doses that exceed 10 CFR Part 100 limits.

Moreover, in 2003, many spent fuel pools were approaching their capacity. If a licensee elected to use long-term dry storage casks to store excess spent fuel, the large, heavy casks would have to be hoisted and transported to and from the spent fuel pool while the plant is at full power operation.

In general, very heavy load drops in BWR plants are more risk significant than very heavy load drops in PWRs because of plant systems layout. For PWRs, spent fuel cask transfers occur near ground level in an area separate from the reactor building and many safety-related systems. However, for BWRs, many very heavy loads are commonly lifted and moved on the upper floor of the reactor building or the auxiliary building. Should a floor breach occur during a load drop, there are many safety-related components located on lower floors which could be disabled. A load drop in certain areas could simultaneously initiate an accident and disable accident mitigation equipment. These types of events have the potential to defeat defense-in-depth.

ANALYSIS

Frequency Estimate

A comprehensive analysis of U.S. nuclear industry crane operating experience from 1968 through 2002 was conducted by the NRC and documented in NUREG-1774.[1846] Some of the NRC's findings were: (1) the human error rate for crane operating events increased significantly; (2) load drop events between the period 1993-2002 increased over the period 1981-1992; (3) the number of below-the-hook crane events (mainly rigging deficiencies or failures) increased greatly; (4) calculational methodologies, assumptions, and predicted consequences varied greatly from licensee to licensee for very similar accident scenarios; (5) the number of mobile crane events declined slightly; and (6) there were few load slips or drops involving very heavy loads.

Based on actual crane operating experience data from commercial U.S. nuclear power plants, it was estimated that the average rate of drops for very heavy loads was 5.6×10^{-5}/demand. This estimate could be higher or lower at a specific plant because of varying human error rates which appeared to dominate load drop events. Based on data estimates collected from the U.S. Navy, the frequency of a handling system failure for nuclear plant cranes was estimated in NUREG-0612[747] to be between 10^{-5} and 1.5×10^{-4} per lift. However, the Navy crane data did not indicate how many lifts were actually performed, i.e., only the number of problems was quantified.

Consequence Estimate

Of the 74 plants that responded to Bulletin 96-02,[1848] only eight indicated that a consequence analysis for heavy load drops had been done at their plants. While the number of operating power plants during the 1993-2002 period only increased 9% over the previous period from 1981 to1992, the number of crane-related injuries during the 1993-2002 period increased 100% over those in the 1981-1992 period. Between 1969 and 2002, there were 10 reported crane events that led to deaths in the nuclear industry and these deaths occurred primarily during the construction phase of the plants.

Other Considerations

The following observations were documented in NUREG-1774:[1846]

(1) Although single-failure-proof cranes share many common design features (e.g., dual reeving, redundant limit switches, and redundant brakes), the remaining criteria for declaring a crane as single-failure-proof (e.g., for new cranes or upgraded cranes) were applied inconsistently. Crane manufacturers were of the opinion that NUREG-0554[1849] was ambiguous in some areas and that clarifications or changes to both NUREG-0612[747] and NUREG-0554[1849] were needed. The industry suggested that a preferred approach would be to consider adopting ASME NOG-1 (Rules for Construction of Overhead and Gantry Cranes) Type I, with minor changes, as an acceptable approach to meeting NUREG-0554[1849] and for upgrading cranes to single-failure-proof status. NOG-1 contains much more specific design criteria for single-failure-proof cranes than does NUREG-0554.[1849] In addition, while some licensees listed their cranes as single-failure-proof or indicated that they met the NUREG-0612[747] upgrade requirements, all the single-failure-proof design criteria listed in NUREG-0554[1849] may not have been fully met. Among events occurring during the period 1968 through 2002 involving cranes suitable for an upgrade to a single-failure-proof design, most load drop events were the result of poor program implementation or human performance errors that led to hoist wire rope or below-the-hook failures. All three very heavy load drops were the result of rigging failures, not crane failures. Consequently, there were no very heavy load drop events that could have been prevented had only a single-failure-proof crane been employed in the lift. However, there were load or hook and block assembly drops that could have been prevented with the use of single-failure-proof cranes and lifting devices.

(2) Between 1976 and 2003, there were 29 NRC generic communications that involved load movements at U.S. nuclear power plants, nine of which addressed: (1) heavy loads moved on the refueling floor; (2) load drop analysis for heavy loads; (3) the identification of heavy

loads that are lifted over safe shutdown equipment; and (4) the consequence of a load drop on selected equipment. Among these communications were generic letters and a bulletin which requested licensees to provide information on their crane programs for NRC evaluation. The accuracy and consistency of information received in response to some of these communications were questionable. Many of the licensees that responded to the latest request (Bulletin 96-02[1848]) provided incomplete information. Also, in many instances, information previously provided to the NRC was not verified to be accurate.

CONCLUSION

The screening and technical assessment of the issue were documented in NUREG-1774.[1846] At the completion of the technical assessment, four recommendations were made for followup guidance development by the NRC staff:

(1) Evaluate the capability of various rigging components and materials to withstand rigging errors and issue necessary guidelines for rigging applications.

(2) Endorse ASME NOG-1 for Type I cranes as an acceptable method of qualifying new or upgraded cranes as single-failure-proof and issue guidance endorsing the standard, as appropriate.

(3) Reemphasize the need to follow Phase I guidelines involving good practices for crane operations and load movements and continue to assess licensee implementation of heavy load controls in safety-significant applications.

(4) Request the appropriate industry Code Committees to evaluate the need to standardize load drop calculational methodologies for nuclear power plants.

REFERENCES

747. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants Resolution of Generic Technical Activity A-36," U.S. Nuclear Regulatory Commission, July 1980.

1845. Memorandum to A. Thadani from B. Sheron, "Proposed Generic Safety Issue - Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants," April 19, 1999.

1846. NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," U.S. Nuclear Regulatory Commission, July 2003.

1847. NRC Letter to All Holders of Operating Licenses, Applicants for Operating Licenses and Holders of Construction Permits for Power Reactors (Generic Letter 83-42)," December 19, 1983.

1848. NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," April 11, 1996.

1849. NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, May 1979.

1850. Memorandum to J. Dyer from A. Thadani, "Proposed Recommendations for Generic Issue (GI)-186, 'Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants,'" November 12, 2003.

1851. Memorandum to M. Mayfield from F. Eltawila, "Proposed Recommendations for Generic Issue (GI)-186, 'Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants,'" November 21, 2003.

ISSUE 193 : BWR ECCS SUCTION CONCERNS

DESCRIPTION

Historical Background

This issue was identified[1814, 1815] by a Region III inspector and addressed the possible failure of low pressure emergency core cooling systems due to unanticipated, large quantities of entrained gas in the suction piping from suppression pools in BWR Mark I containments.

Safety Significance

Three specific concerns were listed in the identifying document:[1814]

(1) One of the bounding design basis accidents is a LOOP combined with a LOCA. While this may be bounding from an ECCS performance perspective, it may not be bounding from a gas entrainment perspective. Because the pumps may start sooner during a LOCA without a LOOP, bubbles generated during the initial blowdown may not have risen to the surface and more may become entrained in the ECCS suction piping. Since a LOCA without a LOOP was not considered, this aspect should be considered for further evaluation.

(2) An AEOD evaluation[1817] of potential air binding or performance degradation of RHR pumps only used the volume of water in the RHR suction piping to determine the amount of dissolved gas. However, the amount of gas that is potentially available to affect pump performance is the total volume of water in the suction piping and the suppression pool. The potential for pump air binding or performance degradation may need to consider the total volume of available water in determining the volume of gas.

(3) The swell/exclusion zone in the torus after a LOCA is considered to be limited to less than one diameter of the downcomer pipe. There does not appear to be a technical basis for this limitation, and it may not be conservative. The intrusion of non-condensable gas into the torus may be greater and the effect will potentially be worse due to the larger suction strainers installed in response to NRC Bulletin 96-03.[1816] Adequate bases to limit the exclusion zone to less than one diameter of the downcomer pipe should be established, especially with respect to the recently installed larger suction strainers.

Possible Solutions

There are several possible solutions to this potential problem. One alternative would be to install a sensor at the ECCS pump suctions, and inhibit either pump startup or discharge valve opening until a stable liquid-phase flow supply is verified. Another approach would be to change the sequencing of the pumps onto their individual buses during ECCS startup. Still another would be to line up one of the ECCS train suctions to the condensate storage tank or other alternative water source. In addition to the above, the installation of anti-vortexing devices to the ECCS suction strainers might be necessary.

It is not clear at this point which solution would be practical or cost-effective. However, because of the fast timing of the event in question, it is likely that the "fix" will involve some hardware modifications to the plant, and not be just procedural.

SCREENING ANALYSIS

Background

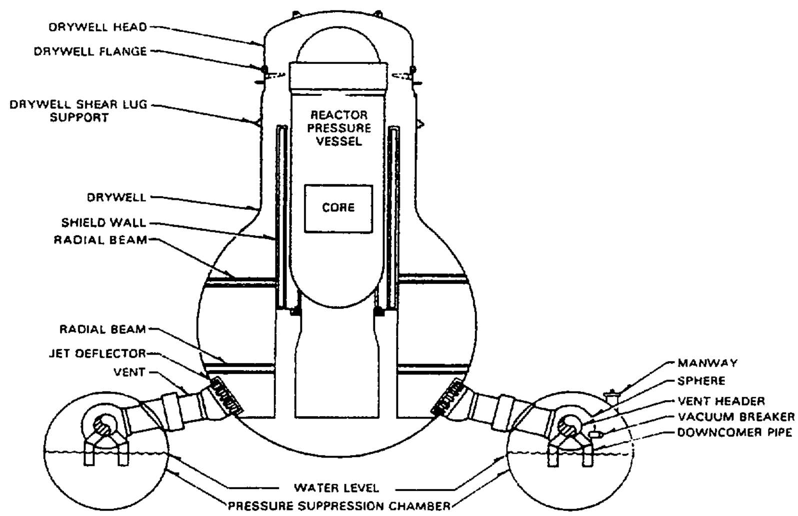

Figure 3.193-1

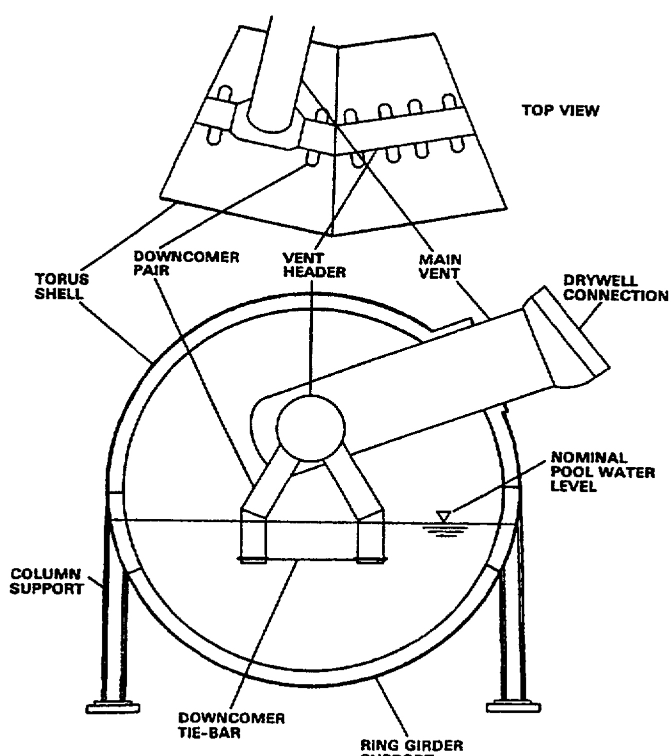

Figure 3.193-2

Pressure Suppression Design: The pressure suppression chamber, or torus, in a BWR Mark I containment, is a steel pressure vessel in the shape of a torus below and encircling the primary containment drywell, which contains the reactor vessel and recirculation system pumps and piping (Figure 3.193-1). In the event of a LOCA, steam released into the drywell airspace is forced through (typically) eight large vent pipes to the suppression chamber. The vent pipes exhaust into a large ring-shaped continuous vent header within the torus. The header is connected to a set of downcomer pipes, which extend into the suppression pool water, and end about four feet below the water surface (Figure 3.193-2). The steam is condensed in the suppression pool water, greatly limiting the peak containment pressure.

BWR Mark I containments operate with the containment atmosphere "inerted," i.e., with less that 4% oxygen by volume. Thus, in the text than follows, the term "air" generally refers to this containment atmosphere composition.

Dynamic Effects of Pressure Suppression: The dynamic effects of a primary system blowdown on the suppression chamber and pool have been studied rather extensively in the Mark I containment short-term and long-term programs (see Issue A-7, "Mark I Long-Term Program"). The primary thrust of this program was the evaluation of the loads (forces exerted) on the containment structure and components, not the effects of entrained non-condensible gases on the ECCS suction. Nevertheless, the phenomena are the same. The effects of the blowdown are well-described in NUREG-0661,[702] portions of which are quoted here:

> "With the instantaneous rupture of a steam or recirculation line, a shock wave exits the broken primary system pipe and expands into the drywell atmosphere. At the break exit point, the wave amplitude theoretically is equal to reactor operating pressure (1000 psia); however, there would be rapid attenuation as the wave front expands spherically outward into the drywell. Further attenuation would occur as the wave enters the drywell vent system and progresses into the suppression pool.
>
> "Because there would be a very rapid drywell pressure increase associated with the postulated LOCA, a compression wave would propagate into the water initially standing in the downcomers. Before this water is cleared from the downcomers, this compression wave would propagate through the suppression pool and result in a dynamic loading on the suppression chamber (torus). The compression wave could also result in a dynamic loading condition on any structures within the suppression pool.
>
> "With the drywell pressure increase, the water initially standing in the downcomers accelerates into the pool, and the downcomers clear of water. During this water-clearing process, a water jet forms in the suppression pool, and causes a potential water-jet-impingement load on the structures within the suppression pool and on the torus section beneath the downcomers.
>
> "Immediately following downcomer clearing, a bubble of [inerted] air starts to form at the exit of the downcomers. As the bubble forms, its pressure is nearly equal to the drywell pressure at the time of downcomer clearing. The bubble pressure is transmitted through the suppression pool water and results in a downward load on the torus.
>
> "When the air/steam flow from the drywell becomes established in the vent system, the initial bubble expands and subsequently decompresses as a result of over-expansion. During the early stages of this process, the pool will swell in bulk mode (i.e., a ligament of

solid water is being accelerated upward by the air bubble). During this phase of pool swell, structures close to the pool surface experience impact loads as the rising pool surface strikes the lower surfaces of the structures. This is followed by drag loads as the pool surface continues to rise past the structures. In addition to this impact and drag loads above the pool, there will also be drag loads as the bubble formation causes water flow past submerged structures and equipment.

"As the water slug continues to rise (pool swell), the bubble pressure falls below the torus airspace pressure. However, the momentum of the water slug causes it to continue to rise, this compresses the air volume above the pool and results in a net upward pressure loading on the torus. The thickness of the water slug will decrease as it rises. Aided by impact of the vent header, it will begin to break up and evolve into a two-phase "froth" of air and water. The froth will continue to rise of its own momentum, and it will impinge on structures above the pool breakthrough elevation.

"When the drywell air flow rate through the vent system decreases and the air/water mixture in the suppression pool experiences gravity-induced phase separation, the pool liquid upward movement stops, and the "fallback" process starts. During this process, structures in the torus may experience a downward loading, and the submerged portion of the torus could be subjected to a pressure increase. Following "fallback," waves may develop on the suppression pool surface, thereby presenting a potential source of dynamic loads on the downcomers, torus, and any other structures close to the water surface.

"The pool swell transient typically lasts on the order of 3 to 5 seconds. Because of the configuration of the pool, this period is dominated by the flow of the drywell atmosphere through the vent system. Steam flow will follow, beginning near the end of the pool swell transient, with a relatively high concentration of noncondensible gases. Throughout these periods, there is a significant pressure differential between the drywell and the torus. This, together with flow-induced reaction forces, leads to structural loads on the vent system."

It is common for BWRs with a MARK I containment to maintain a slight differential pressure between the drywell and the suppression pool airspace, to depress the water level in the downcomers and reduce the hydrodynamic forces caused by expelling a vertical column of water downward from the downcomer exits - the water level is maintained just above the end of the downward-leading pipes. This will reduce the hydrodynamic drag loads, but not the quantity of entrained air.

Thus, it seems reasonable to conclude that the blowdown of the primary system through the drywell and through the suppression pool is a rather violent process. Even though the suction header is somewhat protected from what is occurring within the torus itself, the originator of this issue has posed a reasonable question: will a significant amount of entrained containment atmosphere be sucked into the various ECCS pumps? Clearly, this is a question of timing, since the blowdown phenomena are transient, and the pool will eventually settle down.

Dissolved Gas: The originator of the issue also mentioned potential air binding or performance degradation of RHR pumps due to dissolved gas. This phenomenon was investigated by AEOD in 1982.[18][17] Because the suppression pool water is in equilibrium with the airspace above it, there is always some gas (primarily nitrogen) dissolved in the water. When this water quenches the steam from a primary system blowdown, the water heats up. As the water temperature rises, the solubility of gases decreases, and the dissolved gas comes out of solution and is liberated into its gaseous state. The experiments indicated that the gas was released in the form of a vast number of very small bubbles, less than one millimeter in diameter. Such small bubbles do not rise rapidly to the surface, and could be drawn into the ECCS suction piping.

ECCS System Timing: To see the effects of entrained gas, as postulated by this issue, it is necessary to review ECCS system timing. The details of ECCS initiation can vary from plant to plant. The description used here is based on Browns Ferry 1. Although this plant has been shut down for some time, it was used for many years as the basis for NRC training classes and, for this reason, its design details are readily available. The ECCS pump configuration and the details of the onsite and offsite power systems can vary significantly from plant to plant.

The originator of the issue stated[1814] the difference between ECCS initiation with offsite power available and ECCS initiation when the emergency diesel generators must be used. A diesel generator will always be wired to auto-start on loss of voltage on its associated 4160-Volt shutdown board. Thus, if a LOCA were caused by a seismic event, it is likely that the diesels will already be running when the LOCA occurs, since the same seismic event is likely to damage the transmission lines and cause a loss of offsite power.

In addition, there is an anticipatory diesel generator start signal which is generated be either a combination of high drywell pressure and "low" reactor vessel pressure, or by "low-low-low" reactor water level (by itself). The diesel generators are capable of accepting load within 10 seconds of receiving the automatic start signal. Once each diesel generator is ready, if voltage on its associated shutdown board is low or lost, the diesel generator will be connected to the board. If voltage is normal on the shutdown board, the diesel generator will continue to run at rated speed and voltage, immediately available to be connected.

There are two low pressure ECCS systems in BWRs from the BWR/3 design on. Each of these systems meets the single failure criterion. One is the LPCI mode of the RHR system. LPCI is a high volume reflooding system which injects emergency coolant into the recirculation pump discharge pipes. The flow is then directed into the jet pump nozzles and thus to the lower plenum, which eventually refills and floods the reactor core from the bottom. The other is the low pressure core spray, which has a lower flow capacity but injects water to a pair of spargers located within the reactor vessel core shroud above the core. This flow then sprays down directly into the core from above.

The low pressure ECCS system initiation sequences have several steps. (This can vary from plant to plant, but the example of Browns Ferry will be used here.) For LPCI, the pumps start on either low-low-low reactor water level or on high drywell pressure combined with low reactor vessel pressure. Upon receipt of the start signal, the response depends on the availability of power. If normal AC power is <u>not</u> available, the four main RHR pumps are started essentially simultaneously, as soon as the diesel generators are capable of taking load - about 10 seconds after diesel start, if the diesels are not already running. In contrast, if normal AC power is available, the four pumps start in a seven-second timed sequence, to prevent overloading the auxiliary power source. In either case, it takes time for the pumps to get up to speed. Meanwhile, once reactor pressure has decreased to below 450 psig, which will take about 24 seconds, the inboard LPCI injection valves will automatically open. As reactor pressure continues to fall to 230 psig, the recirculation pump discharge valves are signaled to close, to direct flow to the jet pumps and thereby to the lower plenum of the reactor vessel. Flow will not begin until the pressure in the reactor vessel drops below the discharge pressure of the RHR pumps, which will take about 30 seconds, and will not reach full value until the recirculation pump discharge valves fully close, which will take approximately 30 seconds more.

Similarly, the low pressure core spray pumps start on either low-low-low reactor water level or on high drywell pressure combined with low reactor vessel pressure. If offsite power is available, the pumps are started in a seven-second timed sequence, just as are the LPCI pumps. If offsite power is not available, and the boards are powered by the diesel generators, the core spray pumps are started together, but seven seconds after power is available, so that they do not start at the same time as the LPCI pumps. Once reactor vessel pressure drops to 450 psig, the pump discharge valves open, allowing water to be sprayed over the core. Table 3.193-1, taken from the training manual, summarizes the operational sequence for a large break LOCA with no offsite power available.

Table 3.193-1
Fast Sequencing Scenario

Event	Time (seconds)
Design basis LOCA starts	0
Drywell high pressure and reactor low-low water level	~1
Scram, design-basis analysis assumes diesel-generators signaled to start, primary containment isolates, recirculation pumps trip	3
Low-low-low reactor water level.	~6-8
Diesel generators ready for load/If offsite power not available, start LPCI pumps.	13
LPCI pumps at speed. Signal all 4 core spray pumps to start	20
Reactor reaches 450 psig/Core spray and LPCI injection valves signaled to open	22
Core spray pumps at speed	25
Reactor reaches 230 psig/Signal recirculation pump discharge valves to close	26
Recirculation pump discharge valves begin to close	29
Core spray injection valves fully open	30
LPCI injection valves fully open	46
Recirculation pump discharge valves fully closed	62
Core effectively flooded	~108

If the diesel generators are already running, the LPCI pumps will start at low-low-low reactor water level, and the core spray pumps will start seven seconds later.

A similar table (Table 3.193-2) can be constructed for the situation where offsite power is available, and the diesel generators remain in standby. In this case, the four LPCI pumps and the four core spray pumps are sequenced on in four seven-second intervals, one LPCI pump and one core spray pump at a time.

Table 3.193-2
Slow Sequencing Scenario

Event	Time (seconds)
Design basis LOCA starts	0
Drywell high pressure and reactor low-low water level	~1
Scram, Diesel-generators signaled to start, primary containment isolates, recirculation pumps trip	3
Low-low-low reactor water level/First LPCI and core spray pumps auto start	~6-8
Second LPCI and core spray pumps auto start	13
Third LPCI and core spray pumps auto start	20
Reactor reaches 450 psig/Core spray and LPCI injection valves signaled to open	22
Reactor reaches 230 psig/Signal recirculation pump discharge valves to close	26
Fourth LPCI and core spray pumps auto start	27
Recirculation pump discharge valves begin to close	29
Core spray injection valves fully open	30
LPCI injection valves fully open	46
Recirculation pump discharge valves fully closed	62
Core effectively flooded	~108

The plant designer has some freedom in low pressure ECCS initiation timing in that there will be no flow into the reactor vessel until the vessel pressure drops to below the shutoff head of the ECCS pumps. Thus, the pump sequencing is not critical so long as all pumps are ready by the time the vessel pressure drops sufficiently to allow injection. The designer will generally design the initiation sequencing to limit the severity of the loading transient on the power supply boards. Although individual plants will vary, the two sequencing schemes described above should bound most designs.

Effect of Concerns: The first of the three concerns asserted that, with normal AC power available, the low pressure ECCS pumps would start earlier, and under such circumstances a significant quantity of entrained gas might be drawn into the pump suctions. As can be seen from the description above, this is not necessarily true - at least in the Browns Ferry example, the pumps are actually sequenced on faster when the diesel generators are supplying power.

However, the overall concern raised by this issue appears to be well taken, regardless of this detail. According to the MARK I Long-Term Program, the pool swell transient typically lasts on the order of 3 to 5 seconds. Some of the LPCI and Core Spray pumps will be signaled to start at 6 to 8 seconds after the start of the accident - very close to this same time frame. Although there will be relatively little flow when the pumps first start, the pump discharge valves will be opening about 22 seconds into the accident, and flow will increase rapidly thereafter.

The second of the three concerns asserted that the original AEOD evaluation[1817] calculated only the dissolved gas in the pump suction piping and should have included the entire suppression pool water inventory. It is certainly true that the entire inventory will be subjected to significant heating, and would be expected to release any dissolved gas. However, the amount of gas released into each cubic foot of water will be the same - if the bubbles remain suspended uniformly in the water, the amount of gas entering a pump suction with each cubic foot of water will not change. Most will be released in the initial heatup, as the reactor blows down. Eventually, these gas bubbles will concentrate and coalesce, but they are unlikely to do so in a downward direction. Moreover, the AEOD report[1817] concluded that the pumps were able to tolerate the 2% (by volume) air content "without a discernable loss in pump performance." There is no new information presented to invalidate this conclusion, but this source of entrained gas will be included in the analysis.

The third concern has to do with the swell exclusion zone (basically how large an area is affected by the blowdown through one of the downcomers) and the sizing of the suction strainers. The concern appears to be that the initial bubble formed during the air-clearing phase will extend to the 30-inch connecting tee, and gas rather than liquid will be drawn into the pipe. This can happen for two reasons. First, when the pumps are running, there will of course be flow into the suction piping. However, the pumps are likely running on minimum flow (if they have started at all) during the air-clearing phase of the transient, and the bubble drawn into the suction piping due to this reason would be limited in size. Second, the force of the blowdown could force some non-condensible gas directly into the suction piping, independent of any flow caused by pump operation.

Given the violent nature of the blowdown into the suppression pool, the first and third concerns do have some credibility. The basic questions are first, whether the design of the ECCS suction configuration will be able to keep significant quantities of entrained gas away from the various pump inlets, and second, whether the pool will have sufficiently settled down by the time the pumps are delivering significant flow.

Specifically, at least some of the pumps will be starting just as the air-clearing phase of the blowdown has most of the suppression pool "on the ceiling." The pump flow will just be that of the minimum flow lines (about 500 gpm) which return flow back to the suppression pool. The pumps require about 30 elevation feet of water (about 13 psi) for NPSH, which should not be a problem, since the blowdown will pressurize the suppression chamber to at least this level. However, if large air bubbles are drawn into such a pump, the result will be air binding, flow instability, high vibration, and ultimately impeller damage if the pump does not trip on high vibration or on electrical supply current instability.

Frequency Estimate

The design basis event for the large break ECCS is, as the originator of this issue stated, a large-break LOCA combined with a LOOP, plus an assumption of worst-case single failure. As was discussed above, there is some question as to whether the case with offsite power or the case without offsite power is the more limiting for this issue. Both will be considered.

Initiating Event Frequency: For the case where offsite power is available, the initiating event is a large break LOCA. Instead of using the "traditional" NUREG-1150[1081] value of 10^{-4} event/RY, a more modern value of 3×10^{-5} event/RY, based on the analysis of operating experience, will be used. (See Appendix J of Reference 1819) However, the effect of this choice will be explored with a sensitivity study - this is not intended as an endorsement of the more modern estimate. (Further discussion can be found in the "Other Considerations" below.)

The case where offsite power is not available (the design basis) is somewhat more complicated. The random likelihood of a large LOCA occurring simultaneously with a LOOP is very small, and the probability of a LOOP subsequent to a LOCA is relatively low. The probability value used in the Peach Bottom PRA [1081] was 2 x 10^{-4} (mean). Combined with the 10^{-4}/RY large LOCA-initiating event frequency in NUREG-1150,[1081] the combined LOCA-LOOP event would have a frequency on the order of 10^{-7}/RY or lower. However, a seismic event could cause both a LOOP and a LOCA. (Fire-initiated LOCAs are generally stuck-open SRVs and are not applicable to this issue.)

Such a seismically-induced combined LOCA-LOOP was included in the external events analysis PRA for the Peach Bottom plant.[1081] In the Peach Bottom seismically-induced large LOCA, the frequency was computed based on the failure of the supports of the recirculation pumps. (Failures of the piping were not included as a review of their capacities showed that they were significantly higher than the pump support failures, and thus would make a negligible contribution to the initiating event frequency.) However, an earthquake severe enough to topple a recirculation pump can be expected to break the ceramic insulators on the transmission lines, thereby causing a loss of offsite power. (The ceramic insulators' fragility is listed as 0.25g and the lowest ground motion interval considered in the large LOCA analysis is 0.23g.)

To estimate the frequency of a seismically-induced LOCA-LOOP event, the seismic event frequency and consequent large LOCA probability for each ground motion interval were multiplied, and the products summed to get an overall large LOCA frequency. Since the ground motion intervals were all at or above the ceramic insulator fragility, all of these LOCAs are expected to also result in a non-recoverable LOOP. The result was 6 x 10^{-5}/RY for the LLNL seismic hazard curve and 2.6 x 10^{-6}/RY for the EPRI seismic hazard curve.[1318] Although these estimates differ by about a factor of 20, they will bound most seismic studies. In this analysis, the EPRI curve was used as being more representative of a low seismicity site. The original LLNL curve, which was modified, was used in a sensitivity study below to examine the effect of higher seismicity.

Pump Failure Probability: The parameter of greatest significance for this issue is the probability of pump failure as a function of time after LOCA initiation. This probability can be broken down into two factors: (1) the probability of failure as a function of the volume percent of entrained air in the pump suction; and (2) the fraction of entrained air in the suppression pool volume as a function of time.

The entrained air comes from the three sources discussed earlier. During the initial portion of the blowdown, the drywell atmosphere is carried along with the steam through the downcomers and injected into the suppression pool, until the drywell free volume is essentially all steam and water vapor. Also, the heatup of the suppression pool water will cause some dissolved gas to come out of solution. Finally, during the initial blowdown, the suppression pool water is violently mixed with the air in the upper portion of the torus. Once the blowdown is complete, the water will fall back down relatively rapidly into the lower portion of the torus, but it may take some time for the entrained air bubbles to rise to the surface of the pool.

Although there may be some uncertainty in the amount of non-condensible gas which will be present, the total amount (number of moles) of gas will not have a direct effect on the total void fraction in the suppression chamber free volume. The total free volume is fixed, and the total liquid volume is also fixed since the liquid is essentially incompressible. Therefore, the total volume available to be occupied by air is also fixed. Adding more moles of air will only increase the pressure but not directly effect the total gas volume. (Pressure can cause second order effects, e.g., by driving some air back into solution, but this is not expected to be significant.)

Experimental Work: In the late-1970s, the GE performed a series of experimental tests on a full scale model of a MARK I containment.[1818] The test facility was only a portion of a full 360° torus but was otherwise full scale. Two of the tests simulated a large-break LOCA, a large steam break (Test M7), and a large liquid break (Test M8). The objective of these tests was to measure hydrodynamic loads and structural response, not air entrainment, but the tests nevertheless provided some insight for the purposes of this issue.

In these tests, after the initial blowdown, pool swell, and air-clearing, the system eventually reached a fairly stable condition in which steam exiting the downcomer formed a bubble at the downcomer exit, with steam condensing at the surface of the bubble. The situation was stable in the sense that the downcomer exit bubble presumably expanded until the bubble's surface area was sufficient for the rate of condensation at the bubble surface to match the mass flow of steam into the downcomer. (BWRs have TS limits on initial suppression pool temperature that are intended to ensure stable steam condensation after a blowdown.)

Two phenomena were observed in the tests: chugging and condensation oscillations. Chugging occurred during some of the tests simulating small steam breaks, which are not of significance for this issue. However, condensation oscillations, which were generated by the condensation process at the steam bubble surface, continued for an extended period of time and were observed in all tests, including those tests simulating the large steam and liquid breaks. Because of these condensation oscillations, a certain degree of turbulence will be present in the suppression pool throughout an actual LOCA event.

Although the instrumentation and measurements in these tests were geared toward structural impacts, there were some observations that have some significance for this issue. For example, in Test M8, the large liquid break, the weight percent of air was 3% in the south vent line three seconds into the test. By 15 seconds, it had dropped to 0.2%, and by 25 seconds it had dropped to 0.09%. However, at 40 seconds, the weight percent of air went back up to 0.4%. Thus, there is experimental evidence that air injection will continue at a low rate for some time after the pool swell, but the majority of the drywell atmosphere is injected into the pool very early.

Also, some visual observations were made during the tests. These observations were limited in the sense that the condensing vapor in the wetwell airspace tended to obscure the view, but it was noted in the report that the liquid surface exhibited standing waves that appeared to be correlated with the condensation oscillations associated with the downcomers. After the initial pool swell, standing waves of roughly 2 to 3 inches in amplitude were observed after 13 seconds from test initiation for the liquid break, and after 20 seconds for the steam break. Thus, in this time range, the experimental evidence seems to point to an agitated pool, but a pool with a reasonably well-defined surface.

Bubble Rise Phenomena: The GE experimental work did not record the parameters of most interest for this issue, and thus it is necessary to use some more general knowledge of such phenomena. According to the BWR Fundamentals training manual, the height of the suppression pool air and liquid space (i.e., minor diameter of the torus) is typically about 29 feet. After being forcibly thrown up "to the ceiling" by the initial blowdown, the suppression pool water will fall back in about one second, held back only by air drag. However, entrained air bubbles in the pool will take much longer to rise to the surface, because of the viscosity of the water.

Steam bubbles in subcooled water will break up and disperse quite rapidly, and the increased surface area of the smaller bubbles will lead to rapid condensation. Air bubbles, in contrast, may break up or coalesce, depending on size and surroundings, but will persist until they rise to the pool surface.

In theory, air bubbles will rise and achieve a terminal velocity governed by Stoke's law. Although experiments have shown that Stoke's law works reasonably well for individual bubbles small enough (under 2 mm) for the flow around them to be laminar, a number of effects alter the terminal velocity in practice. An extensive discussion of these effects is contained in "Liquid-Gas Systems," (Fair, J.R., Steimeyer, D.E., Penney, W.R., and Brink, J.A., in "*Chemical Engineers' Handbook*," Perry, R.H., and Chilton, C.H., Fifth Edition, McGraw-Hill, New York, 1973) which include the following:

- At a Reynolds number of about 100, a wobble occurs that causes bubbles to rise in a spiral or helical path.

- Above about 2mm, the bubbles change from spheres to ellipsoids, and above 1 cm, they become lens-shaped.

Both of these effects will tend to lengthen the time bubbles will remain in the suppression pool. In addition, "Liquid-Gas Systems," (Fair, J.R., Steimeyer, D.E., Penney, W.R., and Brink, J.A., in "*Chemical Engineers' Handbook*," Perry, R.H., and Chilton, C.H., Fifth Edition, McGraw-Hill, New York, 1973) discusses two more effects that occur when bubbles are produced "in clouds" and interact with each other. These two effects actually oppose each other:

– A "chimney" effect can develop in which the cloud of bubbles cause a significant upward current in the middle of the bubble stream, which will accelerate bubble rise.

– The proximity of bubbles to each other will hinder the downward flow of the liquid displaced by the bubbles, which will slow the rate of bubble rise.

Thus, the velocity of bubble rise is not easily calculated from theoretical principles, and empirical data must be used. Considerable experimental data can be found in: (1) Fair, J.R., Steimeyer, D.E., Penney, W.R., and Brink, J.A., "Liquid-Gas Systems," in Perry, R.H., and Chilton, C.H., "*Chemical Engineers' Handbook*," Fifth Edition, McGraw-Hill, New York, 1973; and (2) Griffith, P., "Two-Phase Flow," in Rohsenow, W.M., Hartnett, J.P., and Ganić, E.N., "*Handbook of Heat Transfer Fundamentals*," McGraw-Hill, New York. However, these references are intended for chemical engineering applications where bubble columns are intentionally designed to produce a large number of small bubbles, to maximize the interfacing surface area over which chemical reactions can occur. For bubble diameters of a few millimeters, these references predict a bubble rise velocity on the order of about 0.8 feet/second. It is unlikely that bubbles in a suppression pool will be quite this small.

However, CEN 420-P, Volume 1,[1820] describes experimental work done in support of small break LOCA analyses, which is likely to be a more realistic estimate for a suppression pool situation. This correlation gives a "best" estimate of about 3 feet/second at atmospheric pressure, with the data ranging from about 1.7 to 3.3 feet/second. For purposes of this issue, a best estimate of 3 feet/second will be used, but a sensitivity study will be done to see the effect of a rise velocity as low as 0.8 feet/second.

Effect of Turbulence: Bubble rise experiments are generally performed in still water, which will not be the case for a suppression pool right after the blowdown associated with a large-break LOCA. There will be considerable turbulence caused by condensation oscillations arising from steam condensation at the downcomer exits. This effect is not likely to cause significant change in the bulk bubble rise velocity, since this turbulence is equally likely to force a bubble up or sideways as to force it down, with little net effect. (This turbulence may cause some breakup of larger air bubbles into many smaller bubbles, however.)

However, there are likely to be residual macroscopic "swirling" currents induced by the initial blowdown and collapse of the pool swell, plus significant convection currents induced by the ongoing heating of the pool at the downcomer exits. These currents will force bubbles down in some areas, and up in other areas. The net effect on the rate of bubbles of entrained air being brought to the liquid surface (and thus leaving the liquid volume) will not be great, but these currents will reduce stratification of the bubbles in the pool, keeping the entrained air more well mixed.

Phase Separation: If every air bubble were the same size and rose at the same velocity in still water, the percentage of entrained air would drop linearly to zero at a time equal to the pool depth divided by the velocity of rise. With a pool depth of 14 to 15 feet and a bubble rise velocity of three feet per second, this would be about 5 seconds. If the water were completely still, any calculations based on the entire pool depth would be an overestimate, since the ECCS pumps take suction from the bottom of the pool, not at the pool surface. However, because of the presence of turbulence and currents in the pool, no credit will be taken for such stratification within the pool. Under this assumption, the opposite of completely still water, the bubbles are assumed to remain uniformly mixed in the water volume, and the void fraction can be readily estimated for these conditions.

Consider a pool of irregular shape and depth, with water volume V and surface area A, containing N bubbles rising with velocity v. If the pool is constantly being mixed such that the bubbles remain uniformly distributed over the volume V, the bubble density is then a constant (with respect to position) equal to N/V.

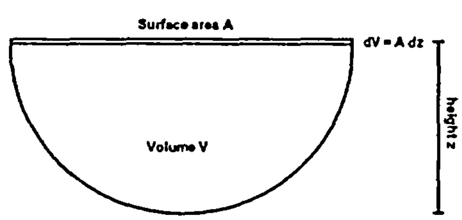

Figure 3.193-3

Consider a volume element dV at the surface of the pool, with area A and thickness dz (see Figure 3.193-3).

$$dV = Adz$$

The number of bubbles in this volume is:

$$dN = \left(\frac{N}{V}\right)dV = \frac{N}{V}Adz$$

The change in the number of bubbles in the volume V per unit time is:

$$\frac{dN}{dt} = -\frac{N}{V}A\frac{dz}{dt} = -\frac{N}{V}Av$$

where v is the bubble rise velocity. The integration of this equation is straightforward:

$$\frac{dN}{N} = -\left(\frac{Av}{V}\right)dt$$

$$N = N_0 e^{-\frac{Av}{V}(t-t_0)}$$

The number of bubbles then follows an exponential decay law, where N_0 is the number of bubbles at time t_0. Interestingly, the time constant is not directly related to the depth of the pool, but instead is related to the surface to volume ratio and the bubble rise velocity. The semicircular shape of the bottom of a Mark I suppression pool actually contributes to a more rapid loss of air bubbles as compared to a rectangular shape, since there is more surface area per unit volume in a pool with a semicircular bottom.

If the water depth z is approximately equal to the minor radius of the torus, the surface to volume ratio for this horizontal semi-cylindrical shape can be approximated by:

$$\left(\frac{A}{V}\right) = \frac{2z}{\frac{1}{2}\pi z^2} = \frac{4}{\pi z}$$

where z is 15 feet. For a relatively simplistic case where all bubbles have the same volume V_b and rise velocity v, the void fraction VF in the pool is:

$$VF(t) = \frac{NV_b}{V} = N_0\frac{V_b}{V}e^{-\frac{4v}{\pi z}(t-t_0)} = VF(t_0)e^{-\frac{4v}{\pi z}(t-t_0)}$$

The void fraction then begins at an initial value $VF(t_0)$ and drops exponentially with time.

As described above, the bubble rise velocity will be assumed to be 3 feet/second. The pool depth (and torus minor radius) will be assumed to be 15 feet. The "start" time t_0 will be assumed to be at the end of the blowdown-induced pool swell, which is 8 seconds. At this time t_0, the initial void fraction will be assumed to be 50%, corresponding to the air and water volumes being equal and completely mixed.

This does introduce some conservatism, in that at 50% void fraction, phase separation is due as much to falling droplets as it is to rising bubbles, and the two phases will begin to separate faster than this primitive model would predict, at least for a few seconds. The model's prediction is shown in Figure 3.193-4.

Effect of Entrained Air on Pumps: Cavitation has been cited as one of the most commonly occurring and damaging problems in liquid pump systems (Lobanoff, V.S., and Ross, R.R., "*Centrifugal Pumps: Design and Application*," Gulf Publishing Co., 1992.). In most nuclear engineering applications, the problem is related to insufficient net positive suction head (NPSH), where the local pressure at the eye of the impeller drops below the vapor pressure of the liquid being pumped, causing bubbles to form. When these bubbles pass through the impeller to a region of higher pressure, where the local pressure is higher than the vapor pressure of the liquid, the bubbles collapse rather violently, creating shock waves in the liquid. Even minor cavitation can produce noise, vibration, loss of head and capacity, and erosion of the impeller and casing surfaces. More severe cavitation can cause cracking of the impeller vanes and pump failure.

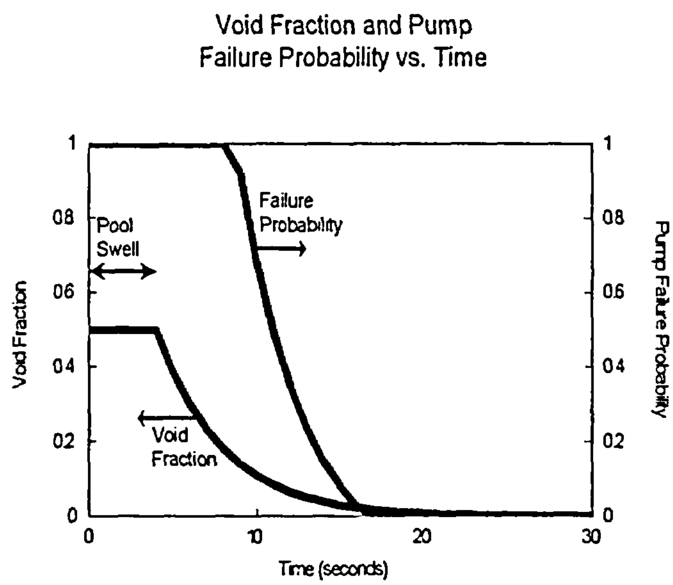

Figure 3.193-4

The specific situation envisioned by this issue is slightly different, in that the cavitation results from entrained air rather than from low pressure. Bubbles formed in this manner will not violently collapse as would bubbles filled with water vapor. In theory, a true "froth" consisting of extremely small bubbles of a non-condensible gas would only have the effect of reducing the density of the pumped liquid, resulting only in some loss of pumping efficiency. However, air bubbles tend to collect at the eye of the impeller, resulting in air binding. In additions, if larger bubbles occur, the result can include turbulence, imbalances in the impeller, severe vibration, and pump failure. A training manual ("*Predicting NPSH for Centrifugal Pumps*," *www.pump.zone.com/articles/00/dec/feature1.htm*) quoted by the originator of this issue makes the claim, "A centrifugal pump can handle 0.5% air by volume. At 6% air the results can be disastrous."

Pump Failure Probability: It seems reasonable to assume that the pump failure probability due to entrained air would be essentially unity if the pumps were to be started right at the point of violent pool swell, but that this failure probability contribution would drop fairly rapidly to essentially zero after 20 seconds or so, based on the visual observations in the GE tests. Because the pumps will be starting during this interval, it will be necessary to make some assumptions on how the pump unavailability varies with time.

NUREG/CR-2792[1821] specifically studied the effect of air and debris ingestion in RHR and containment spray pumps. This study concluded that "for air ingestion level less than about 2%, degradation is not a concern for flows near rated conditions; for ingestion levels in the neighborhood of 5%, performance is dependent on pump design; and for ingestion greater that about 15%, most pumps are fully degraded." It was assumed that the pump failure rate is zero below 2% void fraction, is unity above 15%, and rises linearly from zero to unity between 2% and 15%. This result can be combined with the void fraction estimates described earlier to give a pump failure probability as a function of initiation time, as shown in Figure 3.193-4.

It should be noted that NUREG/CR-2792[1821] also states that a pump can become air bound at very low flow rates, if operation continues over an extended period of time. Although the pumps will be operating at low flow under the conditions of this issue, this will not be for an extended period of time, and thus the full flow assumptions will be used.

Error Analysis Assumptions: The probability of pump failure due to entrained gas was estimated by using start timing for each group of pumps and inputting this time into the pump failure probability function, as illustrated in the second curve of Figure 3.193-4. In order to perform at least a first effort at an error analysis, the following variations were used:

- The end of pool swell, nominally at 4 seconds, was varied between 3 and 5 seconds, which is the interval given in the literature.
- The bubble rise velocity, nominally three feet per second, was varied between 2.2 and 3.8 feet/second, based on an examination of the data in CEN 420-P.[1820]
- The pump failure probability curve was varied by shifting entire function such that the "breakpoint" where the failure probability drops to zero moved from a void fraction of 0.02 to a void fraction of zero, and then shifting the function a symmetric amount in the opposite direction. (The rationale was that the failure probability due to entrained gas would have to go to zero at a zero void fraction.)
- The three parameters were each set at their two extremes and failure probabilities for each pump startup time were calculated.
- The resulting ranges at each startup time were assumed to be 95% intervals in a normal distribution.

In addition to this, the effect of bubble rise velocity was also explored with a sensitivity study.

ECCS Failure Probability: The hypothesis of this issue was that the blowdown into the suppression pool is of sufficient severity and duration to cause a loss of NPSH to the LPCI and core spray pumps because of the entrained gas. This would be a common mode failure of the entire low pressure ECCS. The first question is whether the pumps could survive this situation. If the pumps cavitate and the breakers trip, in theory the pumps could be re-started. However, emergency systems are generally not equipped with any more protective trips than are absolutely necessary.

Of course, if the pumps are damaged, there will be no recovery. It will be assumed for the purposes of issue screening that a pump will fail and not be recoverable if a significant quantity of entrained gas is drawn into its suction.

For the purposes of this issue, the success criteria used in the PRA[1081] for the Peach Bottom plant will be used. Specifically, the success criteria are that either one RHR pump (in LPCI mode) or two core spray pumps will provide sufficient cooling to avoid severe core damage.

As was described earlier, there is more than one possible pump start sequence, depending on specific plant design and depending on whether offsite power is available. The approach will be to use the Browns Ferry sequencing, and then reverse the slow and fast sequences and re-analyze them. This should bound the spectrum of plant designs.

Fast Sequence: An event tree was drawn for the fast sequence, which in the Browns Ferry design corresponds to a LOCA with offsite power unavailable. The initiating event frequency is a seismic event which induces failure of the ceramic insulators on the plant's transmission lines, and also breaks the lateral supports on a recirculation pump. The LOCA is caused by the tipping of the pump.

In this scenario, the diesel generators are likely to be running before the LOCA occurs. All four RHR pumps will start (in LPCI mode) on low-low-low reactor water level six seconds into the accident. All four core spray pumps will start seven seconds later.

Two assumptions are necessary to create an appropriate event tree. First, it is assumed that if a set of pumps does not fail due to entrained gas early in the accident, pumps which are sequenced on later in the transient also do not fail. That is, because the void fraction in the pump suction piping is assumed to be monotonically decreasing, sequences where early pumps do not fail and later pumps do fail are not allowed.

Second, it is assumed that sequences which do not contain at least one pump failure due to entrained gas are not to be included. This is because the parameter of interest for screening generic issues is the change in CDF due to entrained gas. Sequences that lead to core damage but which do not include failures due to entrained gas certainly exist, but are not developed here, since they would be there even if the entrained gas issue were completely fixed.

The event tree for the fast scenario is shown in Figure 3.193-5. (The very first sequence is not developed, since it does not contain any failures due to entrained gas.) The event tree is rather simple in that, if the RHR pumps cavitate, the core spray pumps can fail either due to entrained gas or due to other causes - the "V2" top event in the NUREG-1150[1081] analysis.

This event tree was analyzed using the SAPHIRE code, using a Monte Carlo analysis of 10,000 samples and an analysis cutoff of 10^{-10}. The results were as follows:

Table 3.193-3
CDF for Fast LOCA Sequences

Sequence	Point Estimate	Mean	5th percentile	Median	95th percentile
3	2.3×10^{-9}	1.4×10^{-9}	5.9×10^{-13}	5.5×10^{-11}	4.0×10^{-9}
4	6.1×10^{-7}	1.3×10^{-6}	4.5×10^{-10}	3.8×10^{-8}	2.8×10^{-6}

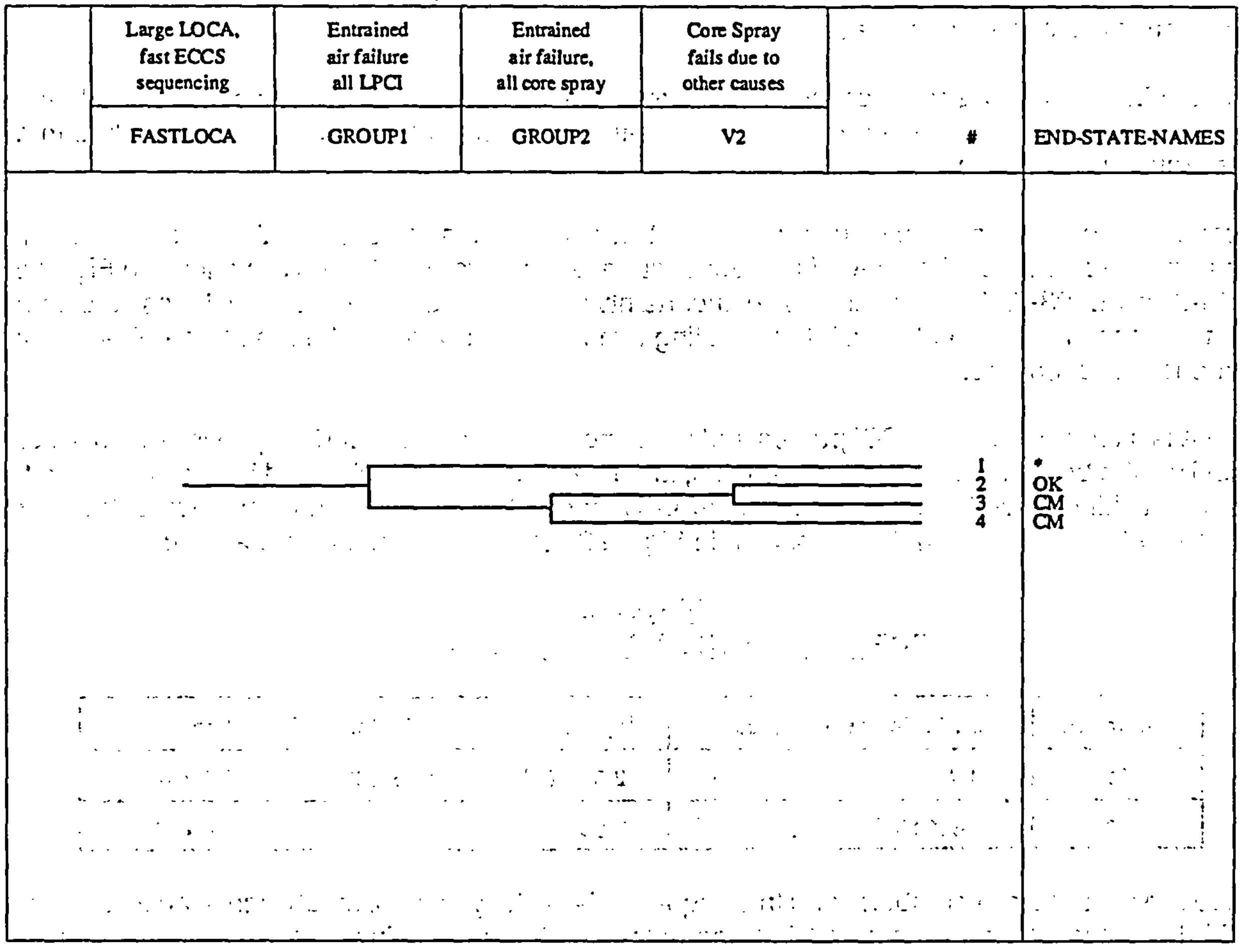

Figure 3.193-5

(Results in Table 3.193-3 and in subsequent tables were given to two significant figures for the convenience of the reader who wishes to follow the calculations, and were not intended to imply that these parameters were known to this accuracy, as the percentile range given in the table itself clearly shows.)

Clearly, this event tree is dominated by sequence number four, where all four LPCI pumps fail due to air entrainment with probability near unity, and the four core spray pumps, which are sequenced on in a group just four seconds later, also fail due to air entrainment with about a 24% probability.

<u>Slow Sequence</u>: The slow sequence corresponds to a LOCA with offsite power available. In this scenario, the pumps are sequenced on in four groups. Each group contains one RHR pump (in LPCI mode) and one core spray pump.

There are two possibilities in this scenario, depending on whether the first RHR pump sequenced to start injects into the intact or into the broken recirculation loop. That is, the pipe break will be in one of the two recirculation loops, and the break will divert injection flow from either RHR pumps 1 and 3, or pumps 2 and 4. (It is assumed that the plant does not use LPCI selection logic.)

Because the failure probability due to entrained gas will be different for the four pumps, two event trees were developed, one for each situation.

As in the fast scenario, sequences with no failures due to entrained gas were not developed, and sequences with a successful pump start for an early group but with an entrained gas failure in a later group were not allowed.

Because the four start sequencing groups do not turn on all trains of a system all at once, the event trees are more complex than that of the fast sequence. The two event trees are shown in Figures 3.193-6 and 3.193-7. Because there is no uncertainty in the number of recirculation loops, and both loops are assumed to be identical, the initiating event frequency is one-half the large pipe break frequency for each tree.

Case A is a case where the LPCI pumps that sequence on in Groups 1 and 3 inject into the reactor vessel via the broken recirculation loop. Thus, LPCI trains 1 and 3 are disabled by the LOCA itself, and the accident must be mitigated by either LPCI pump 3, LPCI pump 4, or any two of the four core spray trains. The event tree is shown in Figure 3.193-6. The results were as follows:

Table 3.193-4
CDF for Slow LOCA Sequences (Case A)

Sequence	Point Estimate	Mean	5th percentile	Median	95th percentile
11	1.7×10^{-9}	1.3×10^{-9}	2.5×10^{-11}	4.1×10^{-10}	5.1×10^{-9}
16	3.6×10^{-10}	6.1×10^{-10}	7.0×10^{-12}	1.6×10^{-10}	2.4×10^{-9}

These are clearly small numbers, and this sequence is unlikely to be of much significance, at least for this base case. It should be noted that Sequence 11, although low in absolute numbers, is much higher than a hand calculation first indicated. This is because the RHR pump in LPCI Train B and the Core Spray pump in Core Spray Train B are associated with some of the same components in the emergency service water system. If it were not for this common cause failure mechanism, Sequence 11 would be down in the 10^{-12} range. Also, it should be noted that the underlying PRA models used a cutoff of 10^{-10} for truncation when building the cut sets, and these numbers are close to this cutoff. Thus, some sequences may be missing, and these numbers may be underestimates. However, because these sequences are of relatively little significance in the total, this should not affect any conclusions.

Case B is a case where the LPCI pumps that sequence on in Groups 2 and 4 inject into the reactor vessel via the broken recirculation loop. Thus, LPCI trains 2 and 4 are disabled by the LOCA itself, and the accident must be mitigated by either LPCI pump 1, LPCI pump 3, or any two of the four core spray trains. The event tree is shown in Figure 3.193-7. The results for Case B are as follows:

Table 3.193-5
CDF for Slow LOCA Sequences (Case B)

Sequence	Point Estimate	Mean	5th percentile	Median	95th percentile
9	1.7×10^{-9}	1.3×10^{-9}	2.5×10^{-11}	4.1×10^{-10}	5.2×10^{-9}
13	2.7×10^{-9}	4.5×10^{-9}	2.6×10^{-11}	7.0×10^{-10}	1.6×10^{-8}

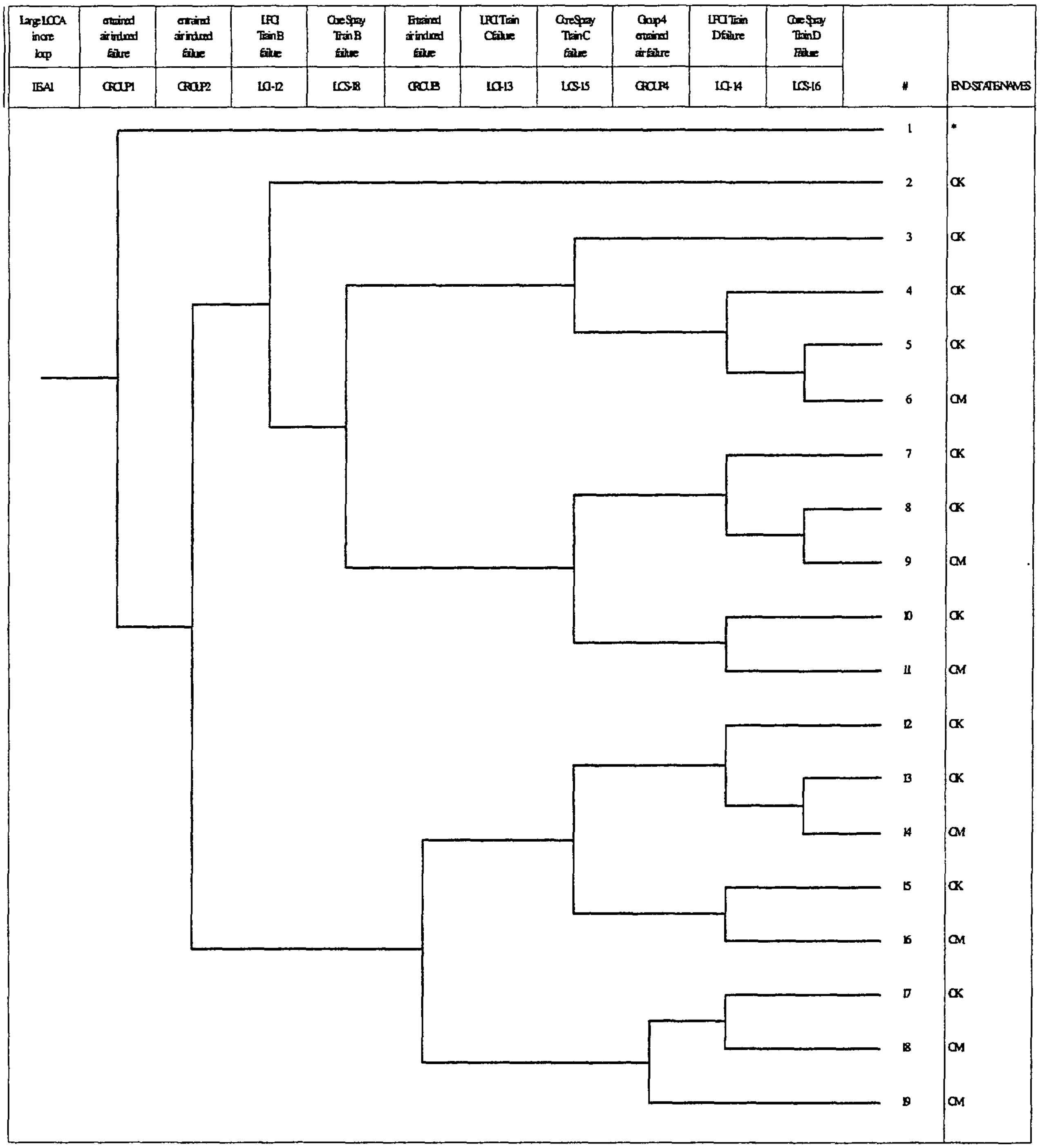

Figure 3.193-6

The numbers are very close to those of Case A. This is primarily because the first few pump groups have failure probabilities of essentially unity, either from gas entrainment or from flow diversion through the broken piping.

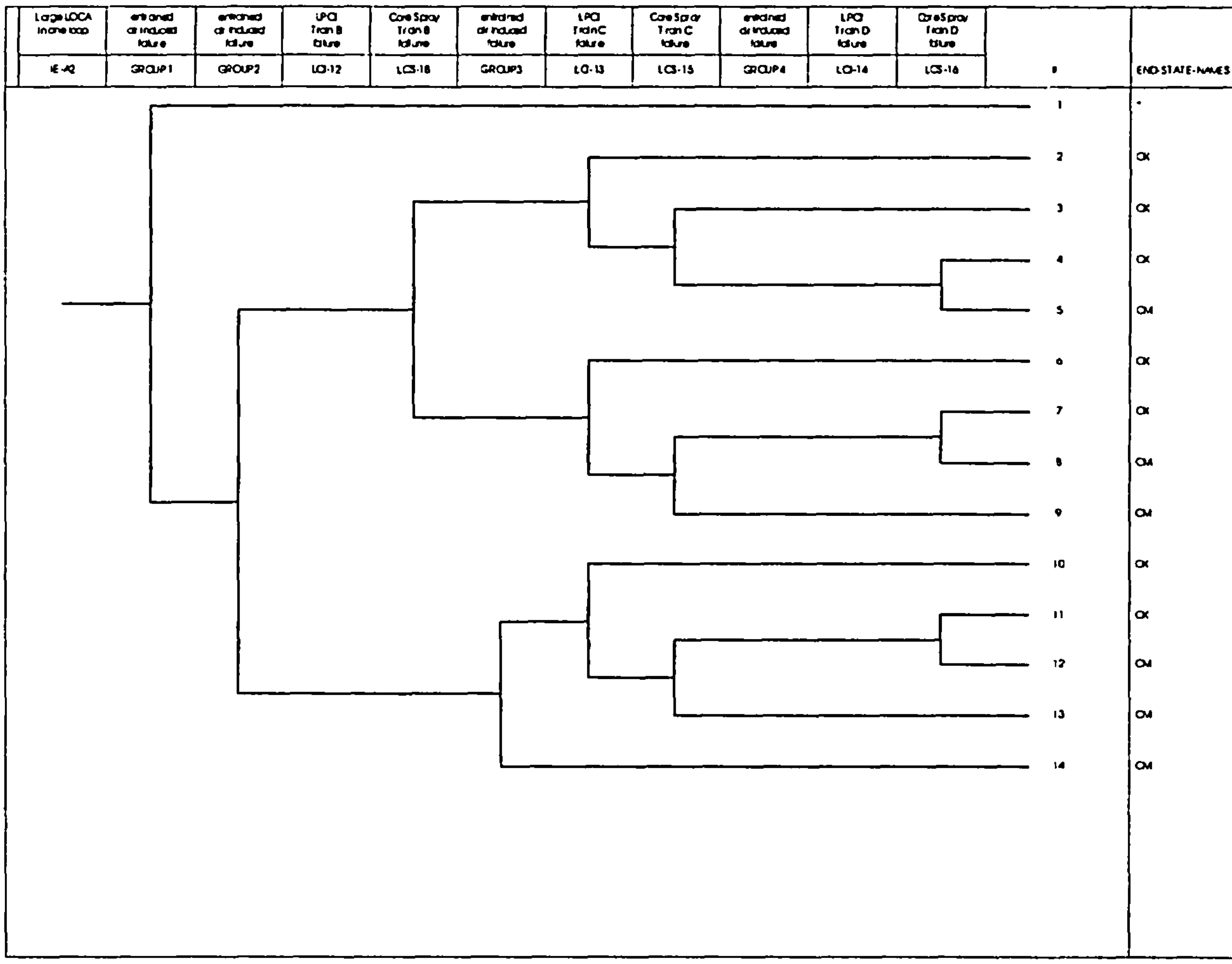

Figure 3.193-7

Combined Results, Base Case, Fast and Slow Sequences: The overall results, adding up the two fast sequences and four slow sequences, are as follows:

Table 3.193-6
CDF (Base Case, Total All Sequences)

	Point Estimate	Mean	5th Percentile	Median	95th Percentile
Combined CDF	6.2×10^{-7}	9.2×10^{-7}	2.7×10^{-9}	4.7×10^{-8}	2.8×10^{-6}

Of this total, about 98% is from sequence 4 of the fast LOCA sequence, in which the LPCI pumps are disabled by entrained gas, and seven seconds later the core spray pumps are disabled by entrained gas.

Sensitivity Studies: Four sensitivity studies were performed. All four use the model described above as the base case. The results are tabulated as follows in Table 3.193-7:

Table 3.193-7
CDF (Base Case and Sensitivities)

Case	Point Estimate	Mean	5th percentile	Median	95th percentile
Base case: Combined CDF	6.2×10^{-7}	9.2×10^{-7}	2.7×10^{-9}	4.7×10^{-8}	2.8×10^{-6}
First case: fast sequencing when offsite power is available	4.7×10^{-6}	8.0×10^{-6}	1.2×10^{-7}	2.4×10^{-6}	3.2×10^{-5}
Second case: high seismicity	1.4×10^{-5}	2.1×10^{-5}	1.7×10^{-6}	8.7×10^{-5}	6.2×10^{-5}
Third case: slow bubble rise	1.9×10^{-5}	1.8×10^{-5}	1.6×10^{-6}	9.1×10^{-6}	5.9×10^{-5}
Fourth case: original LOCA frequency	6.6×10^{-7}	9.9×10^{-7}	1.1×10^{-8}	1.0×10^{-7}	3.0×10^{-6}

The first case was done by reversing the initiating event frequencies for the fast and slow sequences, which is equivalent to a plant with pump initiation sequencing that is faster when offsite power is available - the case brought forth originally in this issue. For this case, the mean CDF rises to 8×10^{-6}, almost all of which comes from Sequence 4.

The second case uses the original NUREG-1150[1081] seismic frequency based on the original LLNL ground motion curve. This case corresponds to a plant wired like Browns Ferry, i.e., where the pumps are sequenced on more rapidly when using the diesel-generators, but located within a high seismic zone. Not surprisingly, over 99% of the CDF again comes from Sequence 4 of the fast LOCA sequence.

The third case uses the base case, but with the bubble rise frequency set to 0.8 feet/second, which corresponds to a suppression pool mixed with very small bubbles - intended to bound the phenomenological aspects of this issue. This case was done to investigate the effect of a slow bubble rise, as would be experienced if the bubbles were all one centimeter or less in diameter. In this case, the pump failure probability is essentially unity except for the fourth group, which has a failure probability of about 65%. (This may not be a physically realistic case, but it does imply that some care should be taken to keep the suppression pool water free of cleaning agents and other surfactants.)

The fourth case was done to illustrate the sensitivity of this model to the spontaneous LOCA frequency. Use of the original LOCA frequency of 10^{-4}/RY instead of the more modern estimate in NUREG/CR-5750,[1760] has only a minor effect on the overall CDF. This is because the base case is dominated by the seismic-induced LOCA sequences.

Consequence Estimate

For this issue, all of the sequences that result in severe core damage include failure of all four RHR pumps. These same pumps are also used for suppression pool cooling and for containment spray. Thus, each of these core damage sequences will also result in containment failure due to overpressure.

Cost Estimate

The LERF estimates are such that the cost will not affect the conclusion. Thus, a cost estimate was not performed for this issue.

Other Considerations

Effect of Pump Suction Configuration: The analysis assumes that the various LPCI and core spray pumps take suction directly from the suppression pool. The actual suction piping configuration varies from plant to plant. In the case of Browns Ferry, these pumps take suction from a large (typically 30-inch diameter) suction header pipe in the shape of a large ring that encircles, and is mounted below, the torus (see Figure 3.193-8). The suction header is connected to the bottom of the suppression pool by (typically) four 30-inch connecting lines ("tee's"). Each connecting line is equipped with a strainer to keep debris out of the suction header. As the originator of this issue pointed out, these strainers were recently re-sized to keep them from being plugged with paint flakes and other small debris. The four connecting pipes are located in "unused" portions of the suppression chamber so that they will not be directly subjected to the water jet issuing from the downcomers. ("Unused" is defined as outside of the swell exclusion zone of any downcomer pipe, as mentioned in the third concern of this issue.) Thus, it can be seen that the design is such that the suction header is somewhat decoupled from the phenomena associated with the pressure suppression function, for plants which are so equipped.

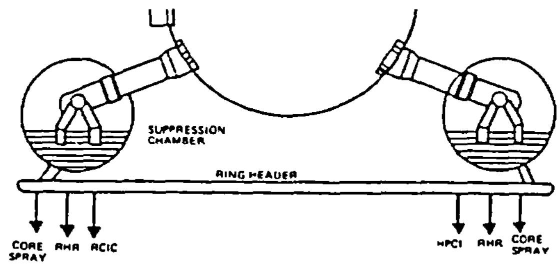

Figure 3.193-8

Nevertheless, there are two possibilities for entrained gas to be drawn into the pumps. First, once the first group of pumps start, any entrained gas present near the connecting tees will be drawn into the ring header, which is common to all the pumps, even those pumps which may be started later.

The second possibility is a function of the asymmetry of the blowdown, in that forcing a bubble into one of the four tees implies that an equal volume of displaced water must flow out of the other three tees. (If all four tees were impacted equally, there would be no bubble ingress.) For bubbles to be forced into the piping in this manner, one tee would have to experience forces significantly greater than the other three tees. This may well be possible.

Special ECCS Pumps: Some plants may have installed ECCS pumps which are especially designed to operate under adverse conditions, such as pumping suppression pool water which is already at or near saturation temperature. Such a pump might be able to survive the presence of significant entrained gas.

Other Containment Designs: Although this screening analysis was performed on a MARK I containment design, the phenomena of interest are also possible in the MARK II and III designs, and these designs should be included in any task action plan for this issue.

Other Suppression Pool Experiments: In the course of the discussion and review of this analysis, it was mentioned that some further experimental work on blowdown phenomena in suppression pools may have been performed overseas, possibly for the Mark II design. Accessing foreign experimental data is generally beyond the scope of a screening analysis such as this, but should be considered as part of any follow-up work.

Initiating LOCA Frequencies: Two frequencies for the large LOCA were considered in this report, the "traditional" frequency from WASH-1400[16] and NUREG-1150,[1081] and a newer, lower frequency described in NUREG/CR-5750.[1760] The two estimates only differ by about a factor of 3, and in any case, even the lower of the two leads to a conclusion that this issue should be studied further. However, it should be noted that still more work in this area is currently in progress, and it is not the intent of this screening analysis to either anticipate the outcome of this new work or to in any way endorse either of the two existing estimates.

Defense-in-depth: The postulated effect of entrained gas bubbles is to defeat a major portion of the low pressure ECCS. Even if the low initiating event frequency results in a low frequency for most of the accident sequences, there is a policy question regarding the wisdom of allowing such a failure, i.e., what is the purpose of maintaining the first group of pumps if there is a high likelihood of failure for this group? This consideration is tempered by the fact that: (a) the estimates used in this screening analysis contain some conservatism, and it is really not known for certain that the first group will fail; and (b) this really applies only to the very large break LOCA, which will violently entrain air in the suppression pool, and the rest of the LOCA break spectrum may not be affected.

Other Means of Mitigation: Given this operational event, the next question is, if LPCI and core spray are ineffective due to entrained gas, what other systems are available to supply coolant to the core? HPCI and RCIC are initially lined up to take suction from the condensate storage tank, but these two systems are turbine-driven, and will not be available since the large break in the primary system will depressurize the reactor, and sufficient steam pressure will not be available.

If offsite power is available, some coolant will be supplied by normal feedwater. However, this will be of limited value, for several reasons:

(1) Once the level drops to the low-low setpoint, the main steam isolation valves will be signaled to close. For those plants with turbine-driven main feedwater pumps, high pressure feedwater will be lost since there will be no steam for the feedwater turbines.

(2) The condensate and condensate booster pumps will continue to run, and are capable of pumping water through the feedwater pumps and to the reactor. The condensate boosters normally run with a discharge pressure of roughly 300 psig, and have a capacity comparable to LPCI. (Some plants have high-head condensate pumps and do not have condensate boosters, but these systems will have a similar performance.) Unlike LPCI, there will be considerable line losses since the flow will have to travel through the feedwater pumps and feedwater heater strings, plus a significant length of piping to the reactor. The flow of condensate will not be large until reactor pressure has dropped well below 300 psig. The primary system does depressurize quite rapidly, however. (Also, the situation will be more favorable in certain older plants which use motor-driven feedwater pumps.)

(3) The main condenser hotwell does not contain enough condensate to last more than about three minutes at full flow. Although it is possible to transfer coolant to the hotwell from the condensate storage tank, the transfer is not high capacity.

(4) The condensate and feedwater system supplies water to the reactor via the feedwater spargers, which are located in the vessel annulus above the jet pumps. If the pipe break which initiated the accident is in the recirculation outlet (i.e., recirculation pump suction) lines, the annulus will be drained, and much of the water sprayed in via the feedwater sparger will miss the jet pump inlets and go out the break. (Conversely, if the pipe break is in the recirculation pump discharge pipes, recirculation inlet lines or semicircular manifolds, the injected water will be much more effective.)

Thus, the condensate and feedwater system has the advantage of already being running when the pipe break occurs, and also will not need operator intervention, but may not be very effective and will certainly not be effective at high flow rates for very long. (Interestingly, the Peach Bottom PRA[1081] gave some credit for feedwater, but the Grand Gulf PRA[1081] did not.) If the LOCA is combined with a LOOP, this system will not be available at all. The only effect of this system is to stave off core melt in the short term.

For longer term coolant supply, a significant means of supplying water is the standby coolant supply system. Details of this system can vary from plant to plant, but every BWR has some means of lining up valves to supply raw water directly to the reactor core. This is commonly done by providing a cross-tie between service water and the RHR piping. However, this system must be lined up manually, and the reactor must be down to about 50 psig. Thus, the standby coolant supply is primarily a long term cooling system, and will not be available during the first minutes after the break, when there is great turmoil in the suppression chamber. For this system, the Peach Bottom PRA[1081] estimated a failure probability of about 25%.

Thus, using both main feedwater and the standby coolant supply, it may be possible to mitigate a large-break LOCA in those situations where offsite power remains available, and this possibility should be considered as part of any full technical assessment of this issue. However, for screening purposes, no credit will be given for this strategy.

The only other systems available for long term coolant supply include the condensate system using makeup to the hotwell from the condensate storage tank, and the control rod drive pumps, which take suction from the condensate storage tank. These are low capacity systems, effective only after many hours have elapsed and decay heat is low, and are not expected to be effective in the time frame envisioned in the scenario of this issue. Thus, no credit will be given for these systems.

Discussion

For the BWR/3 and GE designs after, BWRs are equipped with an ECCS which is both redundant and diverse. In most BWR PRAs, LOCA-initiated sequences generally are not principal contributors in the overall safety profile of the plant. This issue postulated a failure mechanism which, if it is indeed true, has the potential to defeat the entire low pressure ECCS and post-accident containment cooling as well.

Overall, the safety significance is dominated by the fast sequencing scenario and is a concern for the largest-break LOCA. The spaced-out pump startups in the slow sequencing scenario significantly reduces the air entrainment effect on safety postulated in the issue. The analysis indicated some importance even for the base case, but rises significantly for a BWR with fast sequencing when offsite power is available, and also for a BWR in a high seismic area. However, the various estimates given above include some conservatism, and should be understood as an importance measure, not as a best estimate. That is, if the postulated mechanism is true, these are estimates of what the safety significance would be.

It is suggested that any technical assessment include some effort to address the various points of conservatism within this analysis:

(1) This analysis assumed that non-condensible gas bubbles are uniformly mixed within the suppression pool. The actual situation, including stratification and how deeply bubbles will be driven into the pool, should be investigated.

(2) The number of plants with fast sequencing should be investigated along with the number in high seismic zones.

(3) The efficacy of ring headers and other pump suction piping configurations in isolating the pumps from suppression pool phenomena should be investigated.

(4) The ability of pumps to withstand entrained air, particularly for short periods of time, should be investigated.

CONCLUSION

Based on the LERF estimates given above, work on the issue continued to the technical assessment stage.[1824]

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ISSUE 194: IMPLICATIONS OF UPDATED PROBABILISTIC SEISMIC HAZARD ESTIMATES

DESCRIPTION

Historical Background

Beginning in the early-1980s, the NRC sponsored the development of a Probabilistic Seismic Hazard Analysis (PSHA) methodology by LLNL. For the purpose of conducting a systematic evaluation of the licensing criteria for older plants, a limited study of the seismic hazard at the sites where these plants are located was conducted in 1982 and documented in NUREG/CR-1582.[1834] In a 1982 letter, the USGS suggested that deterministic and probabilistic evaluations of seismic hazard should be made for the Eastern United States (EUS) to assess the likelihood of large earthquakes along the eastern seaboard. This led to the 1989 publication of the PSHA study of all 69 sites in the Central and Eastern United States (CEUS) by LLNL in NUREG/CR-5250.[1835] In conjunction with funding the LLNL study, NRC also recommended that the nuclear power industry conduct an independent study to present a coordinated utility position on PSHA estimates. The industry study of 56 CEUS sites was conducted by EPRI and the results were published in EPRI-NP-4726 in 1986.

A draft report on the trial implementation of the Senior Seismic Hazard Analysis Committee (SSHAC) guidance[1838] for the probabilistic seismic hazard assessment of the Watts Bar and Vogtle[1839] nuclear plants showed a higher probabilistic seismic hazard estimate for the Watts Bar site than the value obtained from NUREG-1488.[1836] The increase in the seismic hazard estimate was investigated in a follow-on study which identified the root causes to be a combination of characteristics of the Watts Bar site, such as the site-specific source zones characterization, and more generic ones, such as the modified ground motion model. Depending on whether new information becomes available, other sites could have similar conclusions, such as in the case of Vogtle, for which the mean estimates of the seismic hazard slightly decreased between the 1993 EUS and the 1998 Trial Implementation Plan (TIP) studies. This represented a new interpretation of new seismicity data and resulted in the identification of this issue.[1837]

Safety Significance

The safety concerns were: (1) Did the new data warrant concerns regarding the seismic design bases for nuclear power plants in the region around the Eastern Tennessee Seismic Zone (ETSZ)? and (2) Were other nuclear power plants in the region adversely affected?

ANALYSIS

Frequency Estimate

Large differences in the seismic hazard results between those from the LLNL study and the EPRI study led to the examination of the conflicting results. The staff decided to supplement the LLNL study by improving the elicitation of data and its associated uncertainty from the experts to better capture the uncertainty in our knowledge. The results of this study were published in NUREG-1488.[1836]

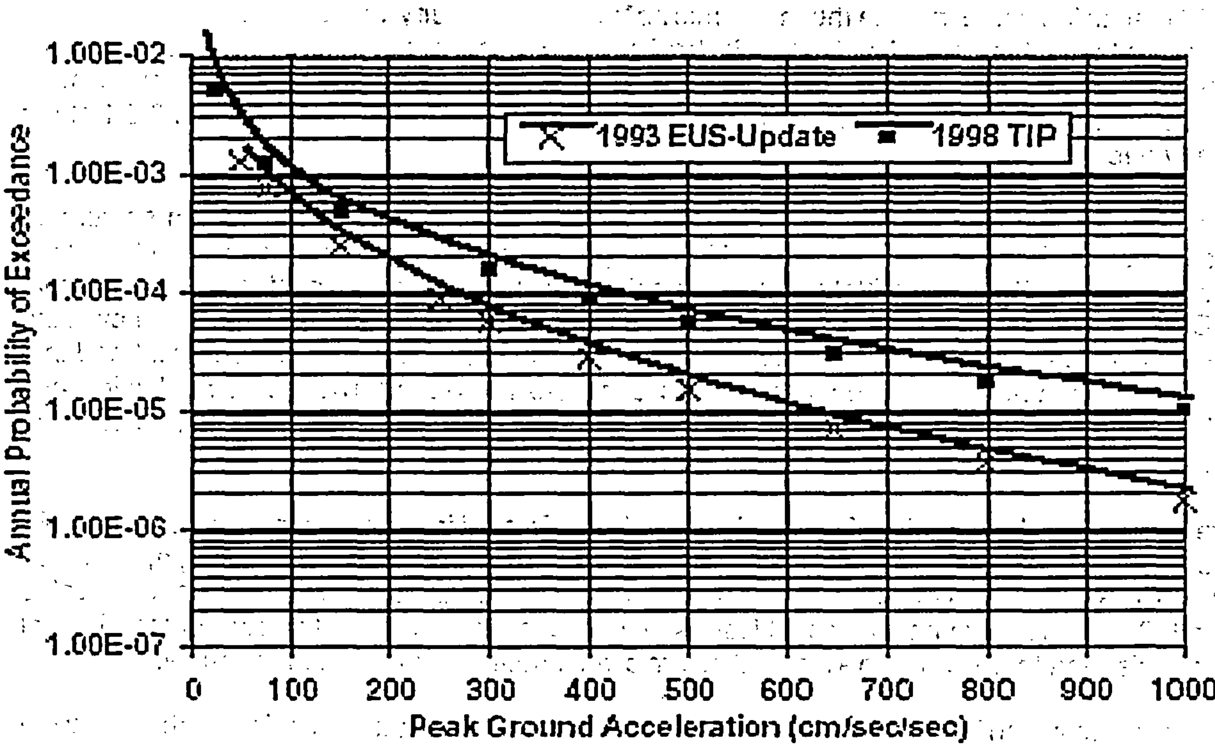
Although the PSHA results in NUREG-1488[1836] show that there is reasonable agreement on plant-specific SSEs, the LLNL seismic hazard estimates in the 10^{-4} to 10^{-6} range are systematically higher than the EPRI hazard results for this range. This is the range of seismic hazard that typically has the most influence on the contribution to seismic risk for nuclear power plants. In an attempt to better understand the reasons for the differences in the two methods, the SSHAC was established under the sponsorship of NRC, EPRI, and DOE in early-1993. The SSHAC report[1838] was published in April 1997 and stated: "Originally, some of the sponsors and participants proposed that one key objective should be to 'resolve' the differences between the LLNL and EPRI studies. However, the Committee quickly realized that the new project would be most useful if it were forward-looking rather than backward-looking - specifically, if it could pull together what is known about PSHA in order to recommend an improved methodology, rather than specifically attempting to figure out which of the two studies was 'correct,' or which specific problems with either study were most important in affecting the study's specific results."

In order to apply the SSHAC methodology, LLNL was contracted to perform a study[1839] (the TIP) of two trial sites (Watts Bar and Vogtle) in the Southeastern United States, a draft of which was completed in 1998. The TIP results for the Watts Bar site indicated that, at the mean annual frequency of 10^{-4}, the peak ground acceleration (PGA) value is about 0.45g, compared to a PGA of about 0.28g at the same mean annual frequency of 10^{-4} from NUREG-1488.[1836] In order to investigate the reasons for the difference in the results from the TIP and the earlier LLNL study, another study was conducted and documented in the draft report UCRL-ID 142039, "Comparison of the PSHA Results of the 1993-EUS-Update and the 1998-TIP Studies for Watts Bar," in March 2002. The introduction of the ETSZ, and to a lesser extent the change in the ground motion attenuation model, increased the potential for higher seismic hazard at sites in the proximity of the ETSZ. A comparison of the TIP and NUREG-1488[1836] hazard curves for the PGA values is shown in Figure 3.194-1 below.

At the reference annual frequency of 10^{-4}, the TIP results are about 1.6 times higher than the 1993 EUS-Update estimate. Sites with operating plants in the proximity of the ETSZ are Browns Ferry, Sequoyah, and Watts Bar. Based on the results for the Watts Bar site, there is a potential that the ETSZ could influence the seismic hazard at these other sites as well. The effect of changes in ground motion model, although secondary in nature, can increase the response spectrum shape in the high frequency range from 9 Hz to 50 Hz. A recent study[1840] also showed the increase of spectral ordinates in the high frequency end. Seismic input in the high frequency end of the response spectrum can cause relay chatter and other effects to vibration-sensitive components. The USGS seismic hazard maps for the Eastern Tennessee area also indicated a higher seismic hazard.

The assessment of seismic risk using seismic PRA models starts with a seismic hazard curve (e.g., frequency of exceedence versus PGA), as described above. Then, fragility curves (conditional frequency of failure versus PGA) for each structure, system, and component of interest must be derived. Finally, the fragility curves are convolved with the seismic hazard curve using event tree and/or fault tree logic models to calculate the frequency of various end states (e.g., CDF) - a fairly involved numerical integration. This calculation can be rather formidable - much more so than the usual internal events PRA, since a seismic event can both initiate an accident and also serve as a common mode failure mechanism for many components, structures, and systems in the plant.

If the change in the seismic hazard curve were a constant multiplicative factor, constant over the domain of the curve, the resulting change in seismic CDF would also be a simple multiplicative factor, since the proportional change would carry through the entire calculation. However, the TIP



Comparison of the Mean Seismic Hazard Estimates for the Watts Bar Site
Figure 3.194-1

curve does not differ from the original curve by a constant factor. This does not change the Boolean logic of a PRA, but does change the numerical integrations. Another complication is that many plants do not have a seismic PRA, but rather as part of their IPEEE, many licensees performed a seismic margins analysis (SMA). This results in no quantification of the seismic risk at these plants, though it does provide a determination that there are safe shutdown paths that meet a required review level earthquake (RLE) and also identifies any potential vulnerabilities associated with those paths. For these plants, the IPEEE typically does identify an overall plant high confidence of a low probability of failure (HCLPF) value, though this value may take credit for plant modifications to resolve the identified vulnerabilities, anomalies, outliers, etc.

Fortunately, an August 1999 paper by Robert P. Kennedy ("*Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations,*" Proceedings of the OECD-NEA Workshop on Seismic Risk, Tokyo, Japan,) presented an approximate method of estimating seismic risk using the plant HCLPF value. This method assumed that the seismic hazard curve can be approximated by an exponential curve and that the fragility curves can be approximated as being log-normally distributed. Both assumptions are reasonable approximations for the purposes of the screening of this issue. Using these assumptions, this method develops a closed form solution for the seismic risk which was developed for use in sensitivity studies such as this. This method was used to develop a sense of the change in the risk estimates, based on the different seismic hazard curves (i.e., LLNL 1993 vs. TIP 1998) for the Watts Bar site. As a caution, these are simplistic calculations that give a rough estimate of the seismic CDF. However, a reasonable estimate of the expected change in CDF resulting from the change to the latest seismic hazard estimate can be obtained by applying the same approach to both sets of seismic hazard information.

The TIP results indicated that the mean seismic hazard estimate for Watts Bar was about two times greater than that estimated in NUREG-1488.[1836] To compare the impact of this new seismic hazard information on CDF for Watts Bar, a simple calculation was carried out using the approximate method described above. The specific steps of the approach are identified in Section 6.2.1 of the Kennedy paper.

This calculation addressed only the seismic contribution. It did not address random equipment failures/unavailabilities or operator errors. However, it was noted from the NRC contractor's TER on the Watts Bar IPEEE submittal that "... non-seismic failures are not expected to be significant for WBN [Watts Bar Nuclear] because there seems to be sufficient diversity and redundancy in the equipment selected in the SSEL [safe shutdown equipment list] for the success paths ..." and that "... significant human action problems are not expected for WBN." Therefore, neglecting any contribution to the CDF from simultaneous random equipment failure or adverse human action in this simple calculation should not lead to erroneous results.

The results of the Watts Bar IPEEE seismic analysis, performed in accordance with the EPRI SMA methodology as described in EPRI-NP-6041-SL, "Nuclear Power Plant Seismic Margin," Revision 1, August 1991, indicated that the plant HCLPF value exceeded the review level earthquake value of 0.3g PGA. There were no significant issues identified in the staff's SER or contractor's TER of this analysis, and there were no identified seismic vulnerabilities, anomalies, or outliers.

The simple calculation included some assumptions regarding the plant's seismic capability and the logarithmic standard deviation of 0.4 that was recommended in the Kennedy paper was used. A lower logarithmic standard deviation would result in higher calculated CDF and change in CDF values. In addition, Watts Bar had identified two success paths that both exceed a HCLPF value of 0.3g PGA. Using the HCLPF Max/Min method rules, the plant HCLPF is equal to the greater of the HCLPF values for these two success paths. However, it was not clear from the SER or TER what precise HCLPF values were achieved for each success path; only that they both exceeded 0.3g PGA. Therefore, in this analysis both success paths were assumed to only just meet the 0.3g PGA and, thus, this capacity was also used to represent the plant HCLPF in the analysis. If a higher HCLPF value were used, lower CDF and change in CDF values would be calculated. With the plant HCLPF of 0.3g PGA and assuming the logarithmic standard deviation of 0.4, the simplistic approach was used to estimate the risk associated with seismic events for the different seismic hazard information.

Using this method and the LLNL seismic hazard information documented in NUREG-1488,[1836] the Watts Bar seismic CDF was estimated to be about 10^{-5}/RY. Using this approach and the new seismic hazard information from TIP, the Watts Bar seismic CDF estimate increases to about 4 x 10^{-5}/RY. This approach implicitly assumed no change in the spectrum shape from the IPEEE study. But the TIP uniform hazard spectrum, which is based on a 10^{-4} mean PGA value, has higher spectral acceleration values than the design SSE spectral acceleration values above about 7 Hz and the increase peaks at about 25 Hz. However, in the 1 to 7 Hz range, the spectral acceleration values are significantly below those from the SSE spectrum. In order to account for the effect of this difference in spectrum shape on the CDF, the Watts Bar plant HCLPF value (0.3g) was scaled to the spectral acceleration values at 5 and 10 Hz, and the scaling relationships for 5 and 10 Hz spectral ordinate from the TIP uniform hazard spectrum were used to determine the CDF values at 5 and 10 Hz. The resulting average CDF was 1.8 x 10^{-5}/year. Therefore, accounting for the TIP uniform hazard spectrum shape, there was an increase in CDF of about 0.8 x 10^{-5}/year.

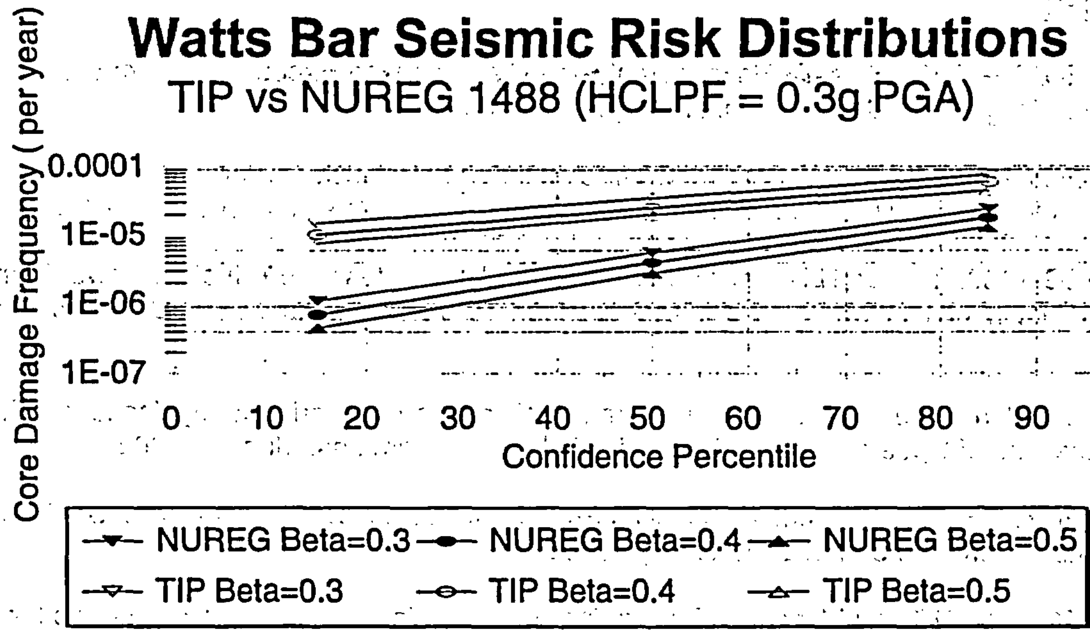


Figure 3.194-2

In order to determine the sensitivity of the estimated CDF for the Watts Bar site using the TIP seismic hazard curve, several CDF estimates were made using the mean, 15th, and 85th percentile hazards, with varying uncertainties (beta values). From Figure 3.194-2, it is apparent that the CDF values are not very sensitive to the percentile level of the hazard curve. This is because the HCLPF value is high and at the low end of the annual frequency of occurrence.

Other Considerations

This issue specifically addressed plants in the ETSZ. However, at the time of this analysis in 2003, the USGS had undertaken a nationwide effort of seismic hazard mapping under the National Earthquake Hazard Reduction Act. In early-2003, the USGS issued revised hazard maps using a methodology quite similar to the SHAAC approach and the NRC was conducting a study of the USGS methodology as a part of the 10-year seismic data base updating activity. This project was expected to lead to an assessment of seismic hazard at existing plant sites. At the end of the NRC study, a comprehensive perspective of the increase or decrease of plant seismic hazard and its effects on the SSE ground motion at all the EUS plants was expected to be available.

CONCLUSION

Based on the risk estimates associated with the spectrum shape for the Watts Bar site and Figure C5 of Management Directive 6.4, the issue regarding the adequacy of deterministic seismic design criteria for the licensing basis of plants in the ETSZ was excluded from further consideration. A generic study may be required to assess the significance for other plants, if the revised USGS results confirm the TIP results and show increases in the seismic hazard for more sites.[1841]

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1836. NUREG-1488, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains," U.S. Nuclear Regulatory Commission, (Draft) October 1993.

1837. Memorandum to J. Flack from D. Dorman, "Proposed Generic Safety Issue on the Implications of Updated Probabilistic Seismic Hazard Estimates," June 6, 2002.

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1839. NUREG/CR-6607, "Guidance for Performing Probabilistic Seismic Hazard Analysis for a Nuclear Plant Site: Example Application to the Southeastern United States," August 18, 1998.

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ISSUE 195: HYDROGEN COMBUSTION IN BWR PIPING

DESCRIPTION

Historical Background

The issue of potential hazards from combustible gases was raised[1822] after two events involving hydrogen combustion occurred within 2 months in late-2001 at two foreign BWRs. Both these events involved sudden rupture of the pipe segments of the RCS by detonation of radiolytic hydrogen. The first event occurred on November 7, 2001, at Hamaoka Unit 1 (BWR-4) and involved rupture of the RHR steam condensing line during a routine surveillance testing of the HPCI system. In the second event on December 14, 2001, at Brunsbüttel, a segment of the head spray line was destroyed. The two reactors were designed by NSSS vendors other than GE. Both these events were reportedly caused by ignition of combustible H_2-O_2 mixture generated by radiolysis of steam/water.

In the past 2 decades, additional events involving hydrogen combustion have occurred at other foreign BWRs. Furthermore, there have been relevant events at foreign non-LWRs. Some of these events also involved personnel injury.

A few hydrogen combustion events associated with the primary reactor coolant system have also been reported at US LWRs; however, there were no significant consequences because of the plant conditions and/or timely counteractions. Some events at US reactors ranged from small hydrogen fires, with no personnel injury, to spillage of primary reactor coolant. In a handful of events, personnel escaped without significant injury and/or contamination. For instance, an event at a US PWR involved hydrogen ignition in the high pressure injection line on the cold leg of the RCS due to welding activities in the vicinity. The plant licensee promptly reported this occurrence to another plant licensee where similar piping weld repairs were being performed, thus preventing a similar occurrence. In another event, personnel error caused hydrogen seepage into a plant's air system while the plant was in a refueling outage. This condition persisted for a couple of hours before the plant personnel realized their error.

Over the years, several fires involving generator hydrogen and the hydrogen storage systems have been reported at US and foreign reactors. In the early 1990s, the NRC had reviewed such events while studying GSI-106, "Piping and the Use of Highly Combustible Gases in Vital Areas." The scope of GSI-106 included evaluation of risk from: (1) the storage and distribution of hydrogen for the volume control tank (VCT) in the PWRs and the main electric generator in the BWRs and the PWRs; (2) other sources of hydrogen such as battery rooms, the waste gas system in PWRs and the Off-gas system in BWRs; and (3) small portable bottles of combustible gases used in maintenance, testing, and calibration. The risk from large storage facilities outside the reactor, auxiliary, and turbine buildings was addressed separately and was not within the scope of GSI-106. In the evaluation of GSI-195, it was presumed that, since the VCT and the generator are not located near the reactor and primary coolant system piping, the risk from hydrogen fires or explosions would not lead to pipe breaks resulting in the LOCAs, ATWS, and steam generator tube ruptures. Additionally, that scoping analysis did not consider the effect of hydrogen explosions on barrier walls and penetrations, such as doors between the turbine building and the adjoining reactor, control, and auxiliary buildings for the two BWR-3 and four BWR-4 considered therein.

On October 25, 1993, the NRC issued Generic Letter 93-06[1547] to inform US licensees of the technical findings from the NRC's resolution of GSI-106, with the expectation that the recipients would review the information for applicability to their facilities and consider actions.

An exhaustive review of foreign and US reactor operational experience review revealed a number of significant events as precursors with potential consequences on plant safety. These events affected both BWRs and PWRs.

The regulators of the two countries where the 2001 hydrogen explosion events occurred released reports which apprised other regulators of their event investigation, analyses, and lessons learned. Follow-up actions taken by the foreign regulatory agencies, the NRC, and the industry are summarized below.

NRC Generic Communications: After the two most recent foreign events, the NRC issued two Information Notices[1823] to inform the US licensees of the events and the associated safety concerns. Prior to this, the NRC had issued IE Bulletin No. 78-03[121] and Information Notice No. 89-44[1552] on the potential hazards of combustible gases. The issue of air/steam/gas-binding of pumps in safety systems has been examined in detail by the NRC and documented in the following AEOD reports: C404[637]; E218[1817]; E325[636]; E910[1830]; T515[1831]; and T927[1832]. Via the NRC communications, US licensees were notified of the potential hazards of entrapped gases in safety system piping and other components. The licensees were cautioned against accumulation of combustible gases to explosive levels, and were advised that should there be a possibility of the presence of a combustible mixture to take the necessary precautions to prevent hydrogen ignition, especially when conducting maintenance activities. Some of the US events and/or generic safety studies were also the subjects of the Nuclear Energy Agency/International Atomic Energy Agency Incident Reporting System (IRS) to share the safety concerns and potential risk with the worldwide nuclear community (e.g., IRS 0001023).

GE Nuclear Energy (GE-NE) Initiatives: On November 20, 2001, GE-NE issued Rapid Information Communication Service Information Letter (RICSIL) No. 85, "HPCI/RHR Steam Supply Line Rupture," to advise the GE BWR owners of the Hamaoka-1 event. This RICSIL contained a brief event description and the information publicly available at the time. It identified the piping systems susceptible to accumulation of non-condensible gases and recommended necessary actions. The RICSIL indicated that, in April 2002, the Hamaoka plants staff had established the cause of the pipe rupture as a result of the accumulation and ensuing detonation of radiolytic gases. This communique briefly discussed similarities between the two events and, based on its assessments of the available information, agreed with the parent utilities' determination of the root cause of the events being hydrogen explosion, as also confirmed by the estimates of the energy releases associated with the event that detonation of a stoichiometric mixture is the most plausible cause. On June 14, 2002, GE-NE issued Services Information Letter No. 643, "Potential for Radiolytic Gas Detonation." In addition to Hamaoka, this SIL advised the GE BWR and ABWR owners about the December 2001 event at Brunsbüttel.

A GE Pipe Rupture Task Force evaluated the two foreign events and concluded that probability of similar events in GE BWRs, while small, cannot be completely precluded. No plant design deficiencies were identified. GE-NE recommended that the GE BWR and ABWR owners consider the following: (1) review piping systems to identify any potential vulnerabilities for accumulation of radiolytic gases; (2) assess detonation potential of vulnerable piping; (3) consider design or system operation modification(s); and consider the potential for accumulation and detonation of radiolytic gases.

The Task Force concluded that there are no design deficiencies in the GE BWRs or ABWRs. GE-NE identified the susceptible piping configurations as those which: (1) are stagnant during normal plant operation; (2) are not continuously or periodically vented or purged; (3) are connected to the steam-filled areas of the NSSS; (4) are lines isolated from higher pressure systems by a potentially leaky valve; (5) can allow accumulation of non-condensible gases; and (6) have continuous steam condensation and drainage. For the combustible gases to detonate, the hydrogen content has to be greater than 15 v/o, with fluid temperature greater than 500° F.

GE-NE also evaluated consequences of potential hydrogen fires and subsequent pipe ruptures. In June 2002, it presented its findings to the staff. GE had estimated that the detonation over-pressure (i.e., the pressure developed during detonation) is dependent on the piping geometry, and from1000 psi can increase locally by a factor of 17 to170. GE also performed a risk assessment by considering H_2-O_2 detonation as the initiator of a small- to medium-break LOCA, and concluded that the incremental CDF for the GE BWRs is 10^{-6}, the base CDF being 2×10^{-5}.

BWR Owners' Group (BWROG) Initiatives: In 2002, the BWROG formed the Hydrogen Accumulation Committee to provide detailed guidance to BWR utilities for identification, disposition, and mitigation of potential radiolytic H_2-O_2 accumulation in plant piping and equipment. In November 2002, the BWROG provided the Committee's guidance document, "BWR Piping and Component Susceptibility to Hydrogen Detonation," (GE-NE-0000-0007-4008-01, Revision 0, Class I) to US utilities and the NRC. This communication indicated that one risk-significant characteristic – the presence of RHR steam condensing mode valves and piping – was not present at many US BWRs, and many had eliminated that feature. RHR steam isolation valve leakage either had not been a problem or had been corrected or mitigated. The Hydrogen Accumulation Committee sought input from GE and the foreign utilities, and provided detailed guidance to the BWR utilities for identification, disposition, and mitigation of potential radiolytic H_2-O_2 in plant piping and equipment. The BWROG surveyed the US utilities for the vulnerabilities, including those similar to the two foreign BWRs, and issued an interim status report on the licensees' responses.[1829] The Committee identified the significant plant equipment that was vulnerable to H_2-O_2 accumulation; surveyed its members to identify the plant areas with the greatest potential for H_2-O_2 accumulation, and actions taken to address this configuration; reviewed the recommendations in Generic Letter 91-18[1828] (including Revision 1) to ensure that operability with respect to hydrogen accumulations is properly addressed; and developed the guidance document (GE-NE-0000-0007-4008-01, Revision 0, Class I) for identifying equipment subject to hydrogen accumulation and potential rupture, as well as short- and long-term mitigation strategies. This guidance was endorsed by BWROG members.

The conclusions based on evaluation of hydrogen build-up rates and survivability of components and piping, as documented in GE-NE-0000-0007-4008-01, were as follows:

(1) when non-condensibles accumulate, the pipe temperature decreases to the saturation temperature of the steam partial pressure, which decreases with time;

(2) larger diameter pipes take longer for radiolytic gases to accumulate;

(3) condensate pots with piping configurations that result in temperature below 467°F need further analysis;

(3) carbon steel piping and higher operating temperatures and pressures are more susceptible than stainless steel piping or lower operating temperature and pressures; and

(4) larger diameter piping will generally fail from detonation when operating near reactor temperature pressure and temperature.

Subsequently, the BWROG conducted a survey of all the utilities regarding the action taken at each of the operating 34 BWRs to address the hydrogen accumulation and potential pipe rupture situations. On August 4, 2003, the BWROG forwarded the survey results to the NRC highlighting the following findings: (1) all had reviewed the available literature; (2) half had evaluated the susceptible piping; (3) some had completed and others were continuing risk assessment of plant equipment; (4) less than half the plants had performed physical walk-downs; (5) 20% had reviewed the plant drawings; and (6) half had identified potentially vulnerable equipment and are pursuing solutions to address them.

Safety Significance

Under some circumstances, an hydrogen explosion in the primary system piping and equipment could lead to an "unisolatable" LOCA. The effect on BWR plant safety of a hydrogen detonation, such as those discussed above, is to either cause a pipe break or damage an SRV. In either case, the effect is to cause a loss of coolant from the primary system, but the mechanistic effects are somewhat different, and the two effects will be treated separately. Additionally, there have been some instances of personnel injury and fatalities stemming from hydrogen explosions. These, however, have not posed significant risk to the public, but instead are of significance for occupational safety and health.

Regarding detonations which rupture pipes, all of the events which have happened thus far have been in locations where the break was isolated, thus limiting the loss of coolant inventory. However, a break in a location which cannot be isolated, or a failure to isolate, would result in a LOCA.

Based on the actual events, such a loss of coolant accident is not likely to be a large design-basis LOCA, since the stagnant "dead end" locations where the combustible gases can accumulate are not large pipes. However, it is quite conceivable that such a detonation-induced pipe break could result in an intermediate-break "S1" LOCA. In reality, of course, a smaller break would be expected to be more likely than a larger break, but for generic issue screening purposes, it will be assumed that a detonation-induced pipe break, if not isolated, will result in an intermediate-break LOCA.

Regarding detonations which occur in SRVs, the effect has been to damage the valve such that the valve opens and remains open, blowing down the primary system into the suppression pool. (In addition to the SRVs in the main steam system, there are safety or relief valves in the liquid-filled systems, such as LPCI and LPCS as well. Failure of these valves may also have potential for coolant loss. However, these valves are separated from the primary system by isolation valves, and failures of these valves will be included as part of the pipe burst accident sequences.) The inadvertent opening of an SRV (IORV) is normally treated as an anticipated transient, since this is not a rare event.

ANALYSIS

Frequency Estimate

The history of these hydrogen detonations suggest two slightly different initiating events. The first is a detonation which causes a pipe to burst. If the resulting leak is not isolated, this will cause a LOCA, as described above. The second initiating event is a detonation within an SRV or its associated inlet piping, which causes the valve to jam open. The resulting loss of coolant will not be isolatable. Initiating event frequencies will be estimated for both of these events.

Pipe Breaks: The severity of combustible gas detonations is likely to vary widely, from mild "pops" to events sufficiently violent to cause damage. The milder events may well not be detected, or may be attributed to other possible causes such as loose parts or water hammer. Thus, the frequency of detonations is difficult to estimate. However, the frequency of those events sufficiently severe to cause pipes to burst can be estimated more directly, since these events are not likely to be missed. Thus, the frequency of a detonation-induced pipe break can be estimated directly from the actual experience.

Based on a private communication with the IAEA, the overall BWR operating experience (as of mid-2003), foreign plus domestic, was 27,900 reactor-months, or 2,325 BWR-years. The event descriptions in the various databases were sometimes somewhat ambiguous, but there were two events which definitely caused pipes to burst, plus one more that may have done so. It will be assumed that three "pipe burst" events had taken place in 2325 RY, so the frequency of detonation-induced pipe breaks will be assumed to be approximately 1.3×10^{-3} event/RY. (This estimate was given to two significant figures only to aid in following the calculations. It was not intended to imply that the frequency is known to such accuracy, as will be shown below.)

For an uncertainty estimate, standard "counting" statistics will be used. The standard deviation of such an estimate is just the square root of the number of events, or 1.7 in 2325 RY.

SRV Failures: Again, there is some ambiguity in the event descriptions, but there was definitely one event where a detonation apparently caused an SRV to fail open and cause a system blowdown. There was a second event where a detonation appeared to have caused an SRV to fail open, plus at least one more event where a detonation apparently caused a blowdown in conjunction with ADS testing. Thus, it was assumed that three events have taken place. Such an event results directly in an intermediate-break "S1" LOCA, as described above. Again, three events in 2325 RY implied an initiating event frequency of 1.3×10^{-3} event/RY. However, because of the ambiguity in the number of events, a standard deviation of 2.7 was used this time to account for both the statistical uncertainty and the uncertainty in the number of events.

Failure to Isolate: The likelihood of the coolant leak not being isolated, either because of a location which has no isolation valve, or because of damage to or failure to close an isolation valve, is more problematic. In all of the known events, the leak was isolated. (This does not apply to the SRV failures.)

The isolation valves in question are generally check valves or motor-operated gate valves, and there are usually two in every fluid-carrying line that penetrates the primary system. Normally, the likelihood of isolation failure would be quite low. However, an examination of piping and instrumentation diagrams for some plants has shown that some locations do exist where a break could not be isolated, e.g., lengths of pipe on the primary system side of isolation valves. Moreover, there is also some possibility that the detonation itself could damage the isolation valves. Thus, it is known that this likelihood is not zero.

The fact that a number of events have happened with no instances of non-isolation, does allow some inferences to be drawn. Obviously, as the number of events with successful isolation goes up, the more confidence there is that the fraction of non-isolatable events is small. This can actually be quantified. For a confidence interval of 95%, if the number of events is *n*, and the fraction of events where isolation is not possible is *x*, then

$$x \leq 1 - \sqrt[n]{0.05}$$

That is, for *n* events with successful isolation, to 95% confidence, the likelihood of non-isolation is less than the limit given by this equation. For three pipe burst events, this works out to an upper limit of about 63%.

This is, of course, just an estimate of a high percentile, not an estimate of the actual likelihood of non-isolation. Given that all events were in isolatable locations, the data would give a "best" estimate of zero for this likelihood. However, an examination of some plant drawings has shown that there are locations where there is either no isolation valve or both isolation valves are normally left open or have leaked. Thus, from engineering knowledge, it is known that this likelihood is greater than zero.

Given the lack of any further information, it will be assumed that the likelihood of non-isolation is described by an exponential distribution. The equation for this distribution is:

$$f(x) = \lambda e^{-\lambda x}$$

The standard deviation λ will be chosen such that the integral of this distribution from zero to x = 0.63 is equal to 95%:

$$\int_0^x f(x)dx = \int_0^x \lambda e^{-\lambda x}dx = 0.95$$

A value of λ = 4.76 will set this integral equal to the desired 95% at x = 0.63, to match the 95th percentile point estimated from the data. This is, of course, a great deal of mathematical inference to be based on a rather meager three data points, and a "sanity check" is very much in order. For an exponential distribution such as this, the mean is just the reciprocal of the parameter λ:

$$\mu = \frac{1}{\lambda} = 0.21$$

Thus, the distribution has a maximum value at zero (i.e., zero non-isolatable piping), a mean of 21%, and a tail such that the 95th percentile is reached at a fraction of non-isolatable piping of 63%. This is, of course, an educated guess, not a rigorous inference from experimental data. These numbers will be used because, after an examination of some piping diagrams, they appear reasonable, particularly the 21% mean fraction of piping that is not isolatable. (The numbers are given to two significant figures for the benefit of those who wish to follow the calculation, and are not intended to imply any degree of accuracy.)

Core Damage Frequency: To estimate the core damage frequency associated with pipe breaks associated with these hydrogen detonations, the NUREG-1150[1081] PRA for the Peach Bottom plant was chosen, primarily because of its availability in the SAPHIRE code package. Two separate calculations were done, one for the SRV blowdown and one for a burst pipe that is not isolatable.

SRV Blowdown: The uncontrolled blowdown of the primary system through an SRV that has failed open is an anticipated transient, and has resulted from causes other than a hydrogen detonation. In terms of a standard probabilistic risk analysis, this scenario affects the safety profile of the plant in three different ways. First, as is the case for any transient, there is always some probability of severe core damage if enough systems fail. Second, an SRV blowdown is an intermediate-break LOCA. Third, the IORV event is a classic ATWS scenario.

The NUREG-1150[1081] Peach Bottom PRA "T3C" event tree is initiated by an IORV event, and is linked to subtrees for transient, LOCA, and ATWS scenarios. The SRV blowdown scenario for this generic issue was analyzed by setting the initiating event frequency to 1.3×10^{-3} event/RY, as discussed above, and calculating the various end state frequencies. The calculations were done using 10,000 samples and standard Monte Carlo uncertainty analysis techniques, and a truncation level of 10^{-10} event/RY.

Although the event trees contain sequences that end in plant damage states PDS-6 and PDS-7, which would involve large early releases, none of these sequences passed the 10^{-10} truncation threshold, and thus are not counted. The remainder of the sequences that led to core damage were as follows:

SRV Blowdown Scenario					
Sequence	**Point Estimate**	**Mean**	**5th percentile**	**Median**	**95th percentile**
All core damage sequences (CDF)	1.1×10^{-7}	1.5×10^{-7}	7.6×10^{-9}	7.3×10^{-8}	4.8×10^{-7}

There is no one highly-dominant sequence, but the majority of the dominant sequences involve the LOCA event trees. The low result is not surprising in that these detonation-induced SRV blowdowns are an increase of less than 1% to the IORV initiating event tree frequency (0.19/RY) already assumed in this PRA.

Burst Pipe: BWRs are generally well defended against LOCAs. Again, for this size LOCA, the coolant inventory loss is more than can be supplied by the RCIC system, but will be within the capacity of the HPCI system. However, after a few hours, the coolant leak (and the steam supply used to power HPCI) will depressurize the reactor, and the low-pressure systems will be needed to keep the reactor core covered.

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 above. The remainder of the event tree is exactly the same as that for the "S1" LOCA.

As in the original "S1" LOCA event tree, the principal contributor to the CDF was a sequence where all the low pressure injection systems fail. This is also the only sequence which results in a large early release of radioactivity; in the other sequences that lead to severe core damage, containment failure is avoided by containment venting. This sequence leads to Plant Damage State 1 (PDS-1). (The end state nomenclature is S1-V2V3V4NU11.)

The calculations were done using 10,000 samples and standard Monte Carlo uncertainty analysis techniques, with a truncation level of 10^{-10}. The results are as follows:

Burst Pipe Scenario					
Sequence	Point Estimate	Mean	5th percentile	Median	95th percentile
PDS-1 Sequences (Possible LERF)	1.5×10^{-7}	1.6×10^{-7}	1.0×10^{-9}	3.4×10^{-8}	6.7×10^{-7}
All other core damage sequences (CDF)	7.1×10^{-8}	7.1×10^{-8}	1.3×10^{-9}	2.9×10^{-8}	2.6×10^{-7}
Total:	2.1×10^{-7}	2.1×10^{-7}	3.0×10^{-9}	7.2×10^{-8}	7.8×10^{-7}

Again, as the table parameters themselves illustrate, the number of significant figures does not imply accuracy to this degree, but instead are provided as an aid in following the calculation. The most dominant sequence is a sequence in which a mis-calibration of the pressure sensors in the drywell defeats all of the low pressure ECCS systems, which may be somewhat plant-specific. If this sequence were not present, the PDS-1 end state frequency would drop by roughly a factor of 30, and the CDF would be reduced by about half.

The initiating event frequency for this "pipe burst" scenario is significant - addition of this sequence approximately doubles the "S1" intermediate LOCA frequency for the plant (normally 3×10^{-4}/RY). However, the LOCA sequences are relatively minor contributors to the overall CDF of the plant, because BWRs are so well-defended against loss of coolant events.

Large Early Release Frequency (LERF): The LERF is estimated based on the frequency of the sequence leading to PDS-1. All the other sequences result in core damage, but the containment integrity is preserved and the release of radioactivity to the environment is limited by venting of the containment from the wetwell airspace, which "scrubs" the release through the suppression pool.

PDS-1 can result in a spectrum of accident progression bins and source term groups. Some of these accident progression bins involve large early containment failure. To quote the description in the PRA, "There are no high RPV vessel breach scenarios because of the LOCA depressurizing the vessel. Since the drywell is flooded by water from the vessel, drywell melt-through is less likely in this case (only 0.36). There is some probability of overpressure failure or venting; but, the availability of containment heat removal in this sequence results in a high probability of no containment failure at all (0.536)."

The estimated frequency for PDS-1 for this issue is 1.6×10^{-7}/RY. If the mean probability of no containment failure is 0.536 for a PDS-1 event, then the probability of a large release is one minus this, or 0.464. Multiplying this by the estimated PDS-1 frequency, the product is an overall large early release frequency (LERF) of 7×10^{-8} large early releases per reactor-year.

Consequence Estimate

A rough estimate of the consequences was made, using the CRIC-ET Code[1795] and the Peach Bottom site. The results for a PDS-1 frequency of 1.6×10^{-7}/RY was on the order of 3 person-rem/year. A calculation based on a generic site population density might be higher than this, but this is well below the 100 person-rem/year cutoff in the MD 6.4 Handbook. Thus, this parameter is unlikely to be limiting.

Cost Estimate

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

Other Considerations

(1) The scope of this analysis was restricted to BWRs, based on the identifying document[1822] and because of a belief that PWRs would be less susceptible, since PWRs have primary systems that are primarily liquid-filled, and are generally operated with an excess of dissolved hydrogen with the explicit purpose of limiting the amount of radiolysis-generated oxygen. The operational experience review conducted in the course of this evaluation revealed that a significant number of events involving hydrogen detonation/explosion have also occurred at US and other foreign PWRs. It is recognized that near the end of cycle, a certain amount of hydrogen can accumulate in the vapor space of a pressurizer which, in the event of an overpressure and subsequent PORV actuation, can and will build up in the relief tank. The potential for rapid oxidation does exist there, as seen in some of the events reported. Relevant operating experience and practices should be reviewed by the NRC, and the implications of hydrogen explosions in PWR piping and components should be assessed as a possible new GSI.

(2) Although the significance of this issue to the safety of the public is low, a burst pipe or a damaged SRV could cause a plant shutdown and necessitate some cleanup and repair, with a resultant increase in occupational risk exposure. Thus, it may be in the economic interest of licensees to take some preventive measures.

(3) The CDFs associated with this issue are well below the thresholds in NRC Management Directive (MD) 6.4. However, the LERF associated with the pipe burst scenario is less than a factor of two below the MD 6.4 threshold for plants with an existing LERF above 10^{-5}. It is expected that detonations (and some pipe bursts) will continue to occur. However, if more pipe bursts occur such that the estimated frequency is higher than the 1.3×10^{-3} event/RY used in this analysis, or if there is operational experience that the likelihood of non-isolation of the break is higher than that estimated in this analysis, then the analysis should be reevaluated.

(4) There have been a number of hydrogen combustion events during maintenance and/or shutdown operations. These events were not included in the scope of this generic issue

because they were thought to be insignificant in terms of the health and safety of the public because the low pressure ECCS is operational during hot shutdown and hot standby, and the pressure is nearly atmospheric while in cold shutdown and refueling. However, such events can be very significant to the occupational safety of plant personnel.

(5) Several events at the foreign BWRs have shown that hydrogen combustion in the valve control lines resulted in excess pressure which, in turn, caused the compression of the central guide pins and damage to the pilot valves. The ensuing deformation of valve internals may cause an impairment of their opening function, and the damage to the pilot valves may lead to failure-to-close of individual safety and relief valves, resulting in a LOCA.

(6) The industry initiatives reported[1829] to the NRC, especially the BWROG efforts to survey the US BWR licensees for the actions taken in response to the pipe rupture events in non-U.S. BWRs, indicated that some of the US BWR licensees had not yet completed their reviews. As of August 2003, the following actions were reported:

- All plants had reviewed the available literature (e.g., RICSIL, SIL, Information Notice, WANO summary, and BWROG guidance document);

- Fifteen of the 16 plants with RHR-SCM piping and 18 of the 19 plants with RHR head spray piping had evaluated that piping. (Survey response answers "NA" are interpreted to mean that the piping is not present or is disconnected.) Remaining plant evaluations were ongoing, but were not complete;

- A risk category assessment of plant equipment had been completed or was in progress at all of the plants.

- Thirteen plants had completed or were performing physical walk-downs of plant equipment;

- Seven plants had reviewed plant drawings. Some of these plants may substitute the drawing reviews for a walk-down, some may conduct a walkdown at a later date.

- 17 plants have identified potentially vulnerable equipment and are pursuing appropriate solutions to address these configurations (e.g., procedure notes, procedure changes, equipment temperature monitoring, configuration analysis, and equipment modification, if necessary).

No new information had been received by the time this evaluation was completed in December 2003.

Suggestions

(1) Hydrogen explosions can potentially threaten plant safety by challenging the integrity of the safety systems, components and equipment, and/or endanger plant personnel safety. To avoid these explosions, the best course of action for licensees would be to prevent build-up of combustible levels of H_2-O_2 mixtures by frequent venting of piping and components that are stagnant and are not normally kept filled.

(2) There are many potential ignition sources at operating plants, and many more exist when routine maintenance activities are conducted during refueling outages. Therefore, it is not practical to eliminate all the likely sources of hydrogen ignition. In fact, the ignition energy for the H_2-O_2 combustible mixture is so low that the ignition sources become practically irrelevant. Therefore, the most prudent action for licensees to take is to prevent the accumulation of detonatable levels of H_2-O_2 mixture. As evidenced by reported events, licensees should take sufficient care during maintenance activities to ensure that no stagnant pipes and components with potentially combustible levels of H_2-O_2 mixtures exist.

(3) GE RTICSIL No. 85 identified and ranked in the reverse order of vulnerability (high-to-low) various piping and components determined to be vulnerable, and recommended certain actions. In "Bin 4-D" of the GE RTICSIL No. 85 table, it was indicated that for safety and relief valves, although hydrogen accumulation is possible, the detonation potential is mitigated by the water vapor content in excess of 5 v/o. Hence, no corrective action is required. However, based on a review of power reactor operational experience, there have been hydrogen explosion events involving rupture of SRVs. In some instances, the presence of a small amount of metal, e.g., inside the valve body, is known to have served as a catalyst and ignited the H_2-O_2 mixture resulting in valve rupture/leakage. One possible cost-effective fix would be to replace the metal segments with palladium (a hydrogen adsorber) or coat them with platinum as catalyst. This would enable the valve body to serve as a "mini recombiner" and prevent build-up of explosive levels of H_2-O_2 mixtures, as successfully done at the Finnish plants.

(4) No unmanageable loss of coolant occurred in the reported SRV failure events; however, if the leaky/ruptured valves had not been isolated or the leakage was large enough, the potential for a LOCA would have existed. Hence, a caution to licensees in this regard is warranted. One vehicle to accomplish this would be a generic communication in which staff review of relevant US and foreign operational experience and the associated risk implications of "unisolatable" breaks would be summarized. This generic communication could also be used to reemphasize good maintenance practices.

CONCLUSION

The detonation-induced SRV failure scenario has a mean CDF well below the cutoff in the MD 6.4 Handbook with no significant LERF and was dropped from further consideration. The CDF associated with detonation-induced pipe bursts was also well below the cutoff, and the LERF associated with this scenario was below the 10^{-7} cutoff. Therefore, it was concluded that there was insufficient justification for this issue to continue to the technical assessment stage. However, it was recommended that the above findings and suggestions be communicated to licensees.[1833]

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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
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
W	- Westinghouse Electric Corporation

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor BWR	Affected NSSS Vendor 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 BWR	Affected NSSS Vendor PWR	Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
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

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.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 BWR	Affected NSSS Vendor PWR	Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
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
80.	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR MARK I and II Containments	CONTINUE	All	NA		TBD	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

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			
128.	Electrical Power Reliability	NOTE 3(a)	All	All		04/29/91	04/29/91
130.	Essential Service Water Pump Failures at Multiplant Sites	NOTE 3(a)	NA	All		09/19/91	09/19/91
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
185.	Control of Recriticality Following Small-Break LOCA in PWRs	HIGH	All	All		TBD	TBD
186.	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants	CONTINUE	All	All		TBD	TBD
188.	Steam Generator Tube Leaks/Ruptures Concurrent With Containment Bypass	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
196.	Boral Degradation	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

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The 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 and qualitative factors. To the extent practical, estimates are quantitative.

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