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

JUNE 2003

SUPPLEMENT 27 TO NUREG-0933
"A PRIORITIZATION OF GENERIC SAFETY ISSUES"

REVISION INSERTION INSTRUCTIONS

	Remove	Insert
Introduction:	pp. 29 to 68, Rev. 26	pp. 29 to 68, Rev. 27
Section 3:	pp. 3.80-1 to 8, Rev. 2 pp. 3.82-1 to 6, Rev. 1 pp. 3.83-1 to 4, Rev. 2 -	pp. 3.80-1 to 17, Rev. 3 pp. 3.82-1 to 6, Rev. 2 pp. 3.83-1 to 3, Rev. 3 pp. 3.192-1 to 8
References:	pp. R-1 to R-121, Rev. 16	pp. R-1 to R-123, Rev. 17
Appendix B:	pp. A.B-1 to 13, Rev. 17	pp. A.B-1 to 13, Rev. 18

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
75.	(Cont.)						B-90, B-91, B-92, B-93
76.	Instrumentation and Control Power Interactions	Zimmerman	RES/DSIR/EIB	DROP	3	06/30/95	NA
77.	Flooding of Safety Equipment Compartments by Back-flow Through Floor Drains	Colmar	RES/DE/EIB	A-17		12/31/87	NA
78.	Monitoring of Fatigue Transient Limits for Reactor Coolant System	Rourk	RES/DET/GSIB	NOTE 3(b)	3	12/31/97	
79.	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	Colmar	RES/DSIR/EIB	NOTE 3(b)	3	06/30/95	NA
80.	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments	Vandermolen	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	Vandermolen	RES/DRPS/RPSI	NOTE 3(b)	2	06/30/03	NA
83.	Control Room Habitability	Emrit	RES/DST/AEB	NOTE 3(b)	3	06/30/03	NA
84.	CE PORVs	Riggs	RES/DSIR/RPSI	NOTE 3(b)	2	06/30/90	NA
85.	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	Milstead	NRR/DSI/CSB	DROP	2	06/30/91	NA
86.	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	06/30/88	B-84
87.	Failure of HPCI Steam Line Without Isolation	Pittman	RES/DSIR/EIB	NOTE 3(a)	2	06/30/95	
88.	Earthquakes and Emergency Planning	Riggs	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
89.	Stiff Pipe Clamps	Chang	RES/DSIR/EIB	LOW	2	06/30/95	NA
90.	Technical Specifications for Anticipatory Trips	Vandermolen	NRR/DSI/RSB, ICSB	DROP	2	12/31/98	NA
91.	Main Crankshaft Failures in Transamerica DeLaval Emergency Diesel Generators	Emrit	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
92.	Fuel Crumbling During LOCA	Vandermolen	NRR/DSI/RSB, CPB	DROP	1	12/31/98	NA
93.	Steam Binding of Auxiliary Feedwater Pumps	Pittman	RES/DRPS/RPSI	NOTE 3(a)		06/30/88	B-98
94.	Additional Low Temperature Overpressure Protection for Light Water Reactors	Pittman	RES/DSIR/RPSI	NOTE 3(a)		06/30/90	
95.	Loss of Effective Volume for Containment Recirculation Spray	Milstead	RES/DRA/ARGIB	NOTE 3(b)		06/30/90	NA
96.	RHR Suction Valve Testing	Milstead	RES/DRA/ARGIB	105		06/30/90	NA
97.	PWR Reactor Cavity Uncontrolled Exposures	Vandermolen	NRR/DSI/RAB	III.D.3.1		06/30/85	NA
98.	CRD Accumulator Check Valve Leakage	Pittman	NRR/DSI/ASB	DROP		06/30/85	NA
99.	RCS/RHR Suction Line Valve Interlock on PWRs	Pittman	RES/DRPS/RPSI	NOTE 3(a)	3	06/30/91	L-817
100.	Once-Through Steam Generator Level	Jackson	RES/DSIR/EIB	DROP	1	06/30/95	NA

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

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

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	of All Feedwater						
125.II.7	Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break	Vandermolen	RES/DRPS/RPSI	NOTE 3(b)	7	12/31/98	NA
125.II.8	Reassess Criteria for Feed-and-Bleed Initiation	Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.9	Enhanced Feed-and-Bleed Capability	Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.10	Hierarchy of Impromptu Operator Actions	Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.11	Recovery of Main Feedwater as Alternative to Auxiliary Feedwater	Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.12	Adequacy of Training Regarding PORV Operation	Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.13	Operator Job Aids	Pittman	NRR/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.14	Remote Operation of Equipment Which Must Now Be Operated Locally	Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
126.	Reliability of PWR Main Steam Safety Valves	Riggs	RES/DRA/ARGIB	LI (NOTE 3)		06/30/88	NA
127.	Maintenance and Testing of Manual Valves in Safety-Related Systems	Pittman	RES/DRA/ARGIB	LOW		12/31/87	NA
128.	Electrical Power Reliability	Emrit	RES/DSIR/EIB	NOTE 3(a)	2	06/30/95	
129.	Valve Interlocks to Prevent Vessel Drainage During Shutdown Cooling	Milstead	RES/DRA/ARGIB	DROP		06/30/90	NA
130.	Essential Service Water Pump Failures at Multiplant Sites	Riggs	RES/DSIR/RPSIB	NOTE 3(a)	2	12/31/95	
131.	Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse-Designed Plants	Riggs	RES/DRA/ARGIB	S	1	06/30/91	NA
132.	RHR System Inside Containment	Su	RES/DSIR/SAIB	DROP	1	12/31/95	NA
133.	Update Policy Statement on Nuclear Plant Staff Working Hours	Pittman	NRR/DLPQ/LHFB	LI (NOTE 3)	1	12/31/91	NA
134.	Rule on Degree and Experience Requirement	Pittman	RES/DRA/RDB	NOTE 3(b)		12/31/89	NA
135.	Steam Generator and Steam Line Overfill	Emrit	RES/DSIR/EIB	NOTE 3(b)	3	06/30/95	NA
136.	Storage and Use of Large Quantities of Cryogenic Combustibles On Site	Milstead	RES/DRA/ARGIB	LI (NOTE 3)		06/30/88	NA
137.	Refueling Cavity Seal Failure	Milstead	RES/DRA/ARGIB	DROP		06/30/90	NA
138.	Deinerting of BWR Mark I and II Containments During Power Operations Upon Discovery of RCS Leakage or a Train of a Safety System Inoperable	Milstead	RES/DSIR/SAIB	DROP	2	12/31/98	NA
139.	Thinning of Carbon Steel Piping in LWRs	Riggs	RES/DRA/ARGIB	RI (NOTE 3)	1	06/30/95	NA
140.	Fission Product Removal Systems	Riggs	RES/DRA/ARGIB	DROP		06/30/90	NA
141.	Large-Break LOCA With Consequential SGTR	Riggs	RES/DRA/ARGIB	DROP		06/30/90	NA
142.	Leakage Through Electrical Isolators in Instrumentation Circuits	Milstead	RES/DSIR/EIB	NOTE 3(b)	4	12/31/97	NA
143.	Availability of Chilled Water Systems and Room Cooling	Milstead	RES/DRA/ARGIB	NOTE 3(b)	2	06/30/95	NA
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 Engineer	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	Milstead	RES/DSIR/SAIB	LI (NOTE 3)	1	06/30/94	NA
148.	Smoke Control and Manual Fire-Fighting Effectiveness	Basdekas	RES/DSIR/RPSIB	LI (NOTE 3)	1	06/30/00	NA
149.	Adequacy of Fire Barriers	Emrit	RES/DSIR/EIB	DROP	2	12/31/98	NA
150.	Overpressurization of Containment Penetrations	Milstead	RES/DSIR/SAIB	DROP	1	06/30/95	NA
151.	Reliability of Anticipated Transient Without SCRAM Recirculation Pump Trip in BWRs	Milstead	RES/DSIR/SAIB	NOTE 3(b)	2	06/30/95	NA
152.	Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads	Emrit	RES/DSIR/EIB	DROP	3	06/30/01	NA
153.	Loss of Essential Service Water in LWRs	Riggs	RES/DRA/ARGIB	NOTE 3(b)	2	12/31/95	NA
154.	Adequacy of Emergency and Essential Lighting	Woods	RES/DSIR/SAIB	DROP	2	12/31/98	NA
155.	Generic Concerns Arising from TMI-2 Cleanup	-	-	-			
155.1	More Realistic Source Term Assumptions	Emrit	RES/DST/AEB	NOTE 3(a)	2	06/30/95	NA
155.2	Establish Licensing Requirements for Non-Operating Facilities	Emrit	RES/DSIR/EIB	RI (NOTE 5)	2	06/30/95	NA
155.3	Improve Design Requirements for Nuclear Facilities	Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.4	Improve Criticality Calculations	Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.5	More Realistic Severe Reactor Accident Scenario	Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.6	Improve Decontamination Regulations	Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.7	Improve Decommissioning Regulations	Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
156.	Systematic Evaluation Program	-	-	-			
156.1.1	Settlement of Foundations and Buried Equipment	Chang	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.1.2	Dam Integrity and Site Flooding	Chen	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.1.3	Site Hydrology and Ability to Withstand Floods	Chen	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.1.4	Industrial Hazards	Ferrell	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.1.5	Tornado Missiles	Chen	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.1.6	Turbine Missiles	Emrit	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.2.1	Severe Weather Effects on Structures	Chen	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.2.2	Design Codes, Criteria, and Load Combinations	Kirkwood	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.2.3	Containment Design and Inspection	Shaukat	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.2.4	Seismic Design of Structures, Systems, and Components	Chen	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.1.1	Shutdown Systems	Woods	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.1.2	Electrical Instrumentation and Controls	Woods	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.2	Service and Cooling Water Systems	Su	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.3	Ventilation Systems	Burdick	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.4	Isolation of High and Low Pressure Systems	Burdick	RES/DSIR/SAIB	DROP	7	06/30/01	NA
156.3.5	Automatic ECCS Switchover	Milstead	RES/DSIR/SAIB	24	7	06/30/01	NA
156.3.6.1	Emergency AC Power	Emrit	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.3.6.2	Emergency DC Power	Rourk	RES/DSIR/EIB	DROP	7	06/30/01	NA

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
156.3.8	Shared Systems	Emrit	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.4.1	RPS and ESFS Isolation	Emrit	RES/DSIR/EIB	142	7	06/30/01	NA
156.4.2	Testing of the RPS and ESFS	Chang	RES/DSIR/SAIB	120	7	06/30/01	NA
156.6.1	Pipe Break Effects on Systems and Components	Page	RES/DET/GSIB	HIGH	7	06/30/01	
157.	Containment Performance	Shaperow	RES/DSIR/SAIB	NOTE 3(b)		06/30/95	NA
158.	Performance of Power-Operated Valves Under Design Basis Conditions	Hrabal	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
159.	Qualification of Safety-Related Pumps While Running on Minimum Flow	Su	RES/DSIR/SAIB	DROP	1	06/30/95	NA
160.	Spurious Actions of Instrumentation Upon Restoration of Power	Rourk	RES/DSIR/EIB	DROP	1	06/30/95	NA
161.	Use of Non-Safety-Related Power Supplies in Safety-Related Circuits	Rourk	RES/DSIR/EIB	DROP	1	06/30/95	NA
162.	Inadequate Technical Specifications for Shared Systems at Multiplant Sites When One Unit Is Shut Down	Cheh	RES/DSIR/SAIB	DROP	1	06/30/95	NA
163.	Multiple Steam Generator Tube Leakage	Coffman	RES/DET/GSIB	HIGH		12/31/97	
164.	Neutron Fluence in Reactor Vessel	Emrit	RES/DSIR/EIB	DROP	1	06/30/95	NA
165.	Safety and Safety/Relief Valve Reliability	Hrabal	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
166.	Adequacy of Fatigue Life of Metal Components	Emrit	NRR/DE/EMEB	NOTE 3(b)	2	12/31/97	NA
167.	Hydrogen Storage Facility Separation	Burdick	RES/DSIR/SAIB	LOW	1	06/30/95	NA
168.	Environmental Qualification of Electrical Equipment	Emrit	NRR/DSSA/SPLB	HIGH	2	12/31/98	
169.	BWR MSIV Common Mode Failure Due to Loss of Accumulator Pressure	Emrit	RES/DET/GSIB	DROP	1	06/30/00	NA
170.	Fuel Damage Criteria for High Burnup Fuel	Emrit	RES/DET/GSIB	NOTE 3(b)	2	06/30/01	NA
171.	ESF Failure from LOOP Subsequent to a LOCA	Rourk	RES/DET/GSIB	NOTE 3(b)	1	12/31/98	NA
172.	Multiple System Responses Program	Emrit	RES/DET/GSIB	NOTE 3(b)	2	06/30/02	NA
173.	Spent Fuel Storage Pool	-	-				
173.A	Operating Facilities	Emrit	RES/DET/GSIB	NOTE 3(b)	4	06/30/02	NA
173.B	Permanently Shutdown Facilities	Emrit	RES/DET/GSIB	NOTE 3(b)	4	06/30/02	NA
174.	Fastener Gaging Practices	-	-				
174.A	SONGS Employees' Concern	Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
174.B	Johnson Gage Company Concern	Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
175.	Nuclear Power Plant Shift Staffing	Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
176.	Loss of Fill-Oil in Rosemount Transmitters	Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
177.	Vehicle Intrusion at TMI	Emrit	RES/DET/GSIB	NOTE 3(a)	1	06/30/00	NA
178.	Effect of Hurricane Andrew on Turkey Point	Emrit	RES/DET/GSIB	LI (NOTE 3)	2	06/30/00	
179.	Core Performance	Emrit	RES/DET/GSIB	LI (NOTE 5)	1	06/30/00	
180.	Notice of Enforcement Discretion	Emrit	RES/DET/GSIB	LI (NOTE 3)	1	06/30/00	
181.	Fire Protection	Emrit	RES/DET/GSIB	LI (NOTE 5)	1	06/30/00	
182.	General Electric Extended Power Uprate	Emrit	RES/DET/GSIB	RI (NOTE 5)	1	06/30/00	

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Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
183.	Cycle-Specific Parameter Limits in Technical Specifications	Emrit	RES/DET/GSIB	RI (NOTE 3)	2	06/30/00	
184.	Endangered Species	Emrit	RES/DET/GSIB	EI (NOTE 5)	1	06/30/00	
185.	Control of Recriticality Following Small-Break LOCA in PWRs	Vandermolen	RES/DSARE/REAHFB	HIGH		06/30/01	
186.	Potential Risk and Consequences of Heavy Load Drops	Lloyd	RES/DSARE/REAHFB	NOTE 4		(Later)	
187.	The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump in Nuclear Power Plants	Vandermolen	RES/DSARE/REAHFB	DROP		06/30/01	NA
188.	Steam Generator Tube Leaks/Ruptures Concurrent with Containment Bypass	VanderMolen	RES/DSARE/REAHFB	Continue		06/30/02	
189.	Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Combustion During A Severe Accident	VanderMolen	RES/DSARE/REAHFB	Continue		06/30/02	
190.	Fatigue Evaluation of Metal Components for 60-Year Plant Life	Shaukat	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
191.	Assessment of Debris Accumulation on PWR Sump Performance	Marshall	RES/DET/GSIB	HIGH	1	12/31/98	
192.	Secondary Containment Drawdown Time	VanderMolen	RES/DSARE/REAHFB	DROP		06/30/03	NA
193.	BWR ECCS Suction Concerns	Vandermolen	RES/DSARE/REAHFB	NOTE 4		(Later)	
194.	Implications of Updated Probabilistic Seismic Hazard Estimates	TBD	RES/DSARE/REAHFB	NOTE 4		(Later)	
195.	Hydrogen Combustion in Foreign BWR Piping	TBD	RES/DSARE/REAHFB	NOTE 4		(Later)	
			HUMAN FACTORS ISSUES				
HF1	STAFFING AND QUALIFICATIONS						
HF1.1	Shift Staffing	Pittman	RES/DRPS/RHFB	NOTE 3(a)	2	06/30/89	
HF1.2	Engineering Expertise on Shift	Pittman	NRR/DHFT/HFIB	NOTE 3(b)	2	06/30/89	
HF1.3	Guidance on Limits and Conditions of Shift Work	Pittman	NRR/DHFT/HFIB	NOTE 3(b)	2	06/30/89	
HF2	TRAINING						
HF2.1	Evaluate Industry Training	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF2.2	Evaluate INPO Accreditation	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF2.3	Revise SRP Section 13.2	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF3	OPERATOR LICENSING EXAMINATIONS						
HF3.1	Develop Job Knowledge Catalog	Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
HF3.2	Develop License Examination Handbook	Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF3.3	Develop Criteria for Nuclear Power Plant Simulators	Pittman	NRR/DHFT/HFIB	I.A.4.2(4)	2	12/31/87	NA
HF3.4	Examination Requirements	Pittman	NRR/DHFT/HFIB	I.A.2.6(1)	2	12/31/87	NA
HF3.5	Develop Computerized Exam System	Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF4	PROCEDURES						
HF4.1	Inspection Procedure for Upgraded Emergency Operating Procedures	Pittman	NRR/DLPQ/LHFB	NOTE 3(b)	6	06/30/95	NA
HF4.2	Procedures Generation Package Effectiveness Evaluation	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	6	06/30/95	NA
HF4.3	Criteria for Safety-Related Operator Actions	Pittman	NRR/DHFT/HFIB	B-17	6	06/30/95	NA
HF4.4	Guidelines for Upgrading Other Procedures	Pittman	RES/DRPS/RHFB	NOTE 3(b)	6	06/30/95	NA
HF4.5	Application of Automation and Artificial Intelligence	Pittman	NRR/DHFT/HFIB	HF5.2	6	06/30/95	NA
HF5	MAN-MACHINE INTERFACE						
HF5.1	Local Control Stations	Pittman	RES/DRPS/RHFB	NOTE 3(b)	4	06/30/95	NA
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	Pittman	RES/DRPS/RHFB	NOTE 3(b)	4	06/30/95	NA
HF5.3	Evaluation of Operational Aid Systems	Pittman	NRR/DHFT/HFIB	HF5.2	4	06/30/95	NA
HF5.4	Computers and Computer Displays	Pittman	NRR/DHFT/HFIB	HF5.2	4	06/30/95	NA
HF6	MANAGEMENT AND ORGANIZATION						
HF6.1	Develop Regulatory Position on Management and Organization	Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF6.2	Regulatory Position on Management and Organization at Operating Reactors	Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF7	HUMAN RELIABILITY						
HF7.1	Human Error Data Acquisition	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.2	Human Error Data Storage and Retrieval	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.3	Reliability Evaluation Specialist Aids	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.4	Safety Event Analysis Results Applications	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF8	Maintenance and Surveillance Program	Pittman	NRR/DLPQ/LPEB	NOTE 3(b)	2	06/30/88	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	CHERNOBYL ISSUES						
CH1	ADMINISTRATIVE CONTROLS AND OPERATIONAL PRACTICES						
CH1.1	Administrative Controls to Ensure That Procedures Are Followed and That Procedures Are Adequate	-	-				
CH1.1A	Symptom-Based EOPs	Emrit	NRR/DLPQ/LHFB	LI (NOTE 5)		06/30/89	NA
CH1.1B	Procedure Violations	Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.2	Approval of Tests and Other Unusual Operations	-	-				
CH1.2A	Test, Change, and Experiment Review Guidelines	Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.2B	NRC Testing Requirements	Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.3	Bypassing Safety Systems	-	-				
CH1.3A	Revise Regulatory Guide 1.47	Emrit	RES/DE/EMEB	LI (NOTE 5)		06/30/89	NA
CH1.4	Availability of Engineered Safety Features	-	-				
CH1.4A	Engineered Safety Feature Availability	Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.4B	Technical Specifications Bases	Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.4C	Low Power and Shutdown	Emrit	RES/DSR/PRAB	LI (NOTE 5)		06/30/89	NA
CH1.5	Operating Staff Attitudes Toward Safety	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH1.6	Management Systems	-	-				
CH1.6A	Assessment of NRC Requirements on Management	Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.7	Accident Management	-	-				
CH1.7A	Accident Management	Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH2	DESIGN						
CH2.1	Reactivity Accidents	-	-				
CH2.1A	Reactivity Transients	Emrit	RES/DSR/RPSB	LI (NOTE 5)		06/30/89	NA
CH2.2	Accidents at Low Power and at Zero Power	Emrit	RES/DRA/ARGIB	CH1.4		06/30/89	NA
CH2.3	Miltiple-Unit Protection	-	-				
CH2.3A	Control Room Habitability	Emrit	RES/DRA/ARGIB	83		06/30/89	NA
CH2.3B	Contamination Outside Control Room	Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.3C	Smoke Control	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH2.3D	Shared Shutdown Systems	Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.4	Fire Protection	-	-				
CH2.4A	Firefighting With Radiation Present	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH3	CONTAINMENT						
CH3.1	Containment Performance During Severe Accidents	-	-				
CH3.1A	Containment Performance	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH3.2	Filtered Venting	-	-				

Table II (Continued)

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

TABLE III

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

Legend

NOTES:	1 - Possible Resolution Identified for Evaluation
	2 - Resolution Available
	3 - Resolution Resulted in either the Establishment of New Requirements or No New Requirements
	4 - Issues to be Prioritized in the Future
	5 - Issues that are not GSIs but Should be Assigned Resources for Completion
DROP	- GSI Dropped from Further Pursuit
EI	- Environmental Issue
GSI	- Generic Safety Issue
HIGH	- High Safety Priority
I	- TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737
LI	- Licensing Issue
LOW	- Low Safety Priority
MEDIUM	- Medium Safety Priority
RI	- Regulatory Impact Issue
USI	- Unresolved Safety Issue
Continue	- As defined in NRC Management Directive 6.4

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 (275)														
GSI	-	54	0	0	82	0	6	0	4	97	2	4	-	249
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	416	0	6	0	16	121	2	4	64	845

ISSUE 80: PIPE BREAK EFFECTS ON CONTROL ROD DRIVE HYDRAULIC LINES IN THE DRYWELLS OF BWR MARK I AND II CONTAINMENTS

DESCRIPTION

Historical Background

This issue was identified by the ACRS in 1978 during the operating license reviews of some BWRs. The ACRS posed questions concerning the likelihood and effects of a LOCA which could cause interactions with the CRD hydraulic lines in such a way as to prevent rod insertion, creating the potential for recriticality when the core is reflooded.[537] The staff investigated this potential problem and concluded that the existing SRP[11] criteria were adequate to assure integrity of the CRD hydraulic lines.[538] These criteria assume conservative failure stresses and break locations in coolant pipes and require examination of the effects of pipe whip and jet impingement on essential safety components (including the CRD hydraulic lines) for approximately 100 breaks.

The ACRS discussed this conclusion with the staff during its 273rd meeting on January 6, 1983, but remained concerned about MARK I and II containments, which are smaller and more congested than the MARK III containments upon which the staff's analysis was concentrated.[539] Thus, the issue remained open for the MARK I and II containments.

Following an analysis of the issue in January 1984, the issue was given a LOW-priority ranking (based on Appendix C of NUREG-0933). It was later concluded in NUREG/CR-5382[1563] that consideration of a 20-year license renewal period could change the ranking of the issue to medium priority. However, further evaluation, using the conversion factor of $2,000/man-rem approved[1689] by the Commission in September 1995, resulted in the issue being placed in the DROP category.

During site visits associated with Issue 156.6.1, "Pipe Break Effects on Systems and Components," some new piping configurations were discovered that were not considered in the original evaluation of Issue 80. Thus, in March 1998, during a periodic review of LOW-priority GSIs, NRR indicated[1810] that the priority of Issue 80 should be reassessed in light of the concerns of Issue 156.6.1. As a result, a study[1811] was conducted by RES to determine the safety significance of the issue and the findings were used in this assessment.

Safety Significance

Recriticality during the course of an accident has no direct effect on the health and safety of the public. However, failure to insert a significant number of control rods could pose two separate safety problems. First, when the core is reflooded by cold emergency core cooling water, the reactor will undergo a cold water reactivity transient if the core is not subcritical. The cold water can insert considerable positive reactivity, which means that portions of the core where control rods failed to insert can return to a significant power level and may even overshoot to power levels considerably higher than those experienced during normal operation. Secondly, the recirculation phase of emergency core cooling is sized to carry away decay heat. If fission heat is not shut off, the ECCS may not be sufficient to remove this extra energy, resulting in coolant boil-off, core-melt, and potential containment failure.

Possible Solutions

It may be possible to reduce any safety concerns to acceptable levels by performing more frequent or enhanced inspections of those lengths of primary system piping that could impact the CRD hydraulic lines. If this is not possible, the installation of some type of guard structure may be justified.

EVALUATION

A BWR control rod is scrammed by applying pressure from an accumulator or from the reactor vessel to the volume below the CRD piston and venting the volume above the piston to the scram discharge volume which is near atmospheric pressure. If the insert line is either blocked or broken, a ball check valve built into the CRD (for all BWR/3 and later designs) will admit reactor water to the volume under the piston. (See Figure 80-1.) Thus, the insert line is necessary for scram only when the reactor pressure is low, e.g., during reactor startup.

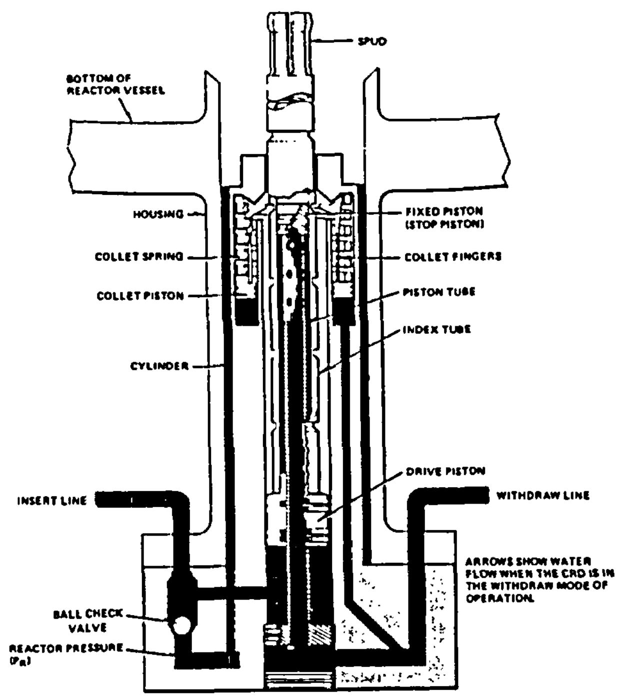

Figure 80-1
BWR Control Rod Drive

Breaking the withdraw line will open the volume above the piston to atmospheric pressure and thus cause (not prevent) a scram. The only way to prevent a scram by mechanical damage to the CRD lines is to crimp the withdraw line shut. Breaking or crimping an insert line will prevent a scram only at low reactor pressure at which time the high energy coolant lines, which are to provide the crimping force, are also at low pressure and the reactor is also at very low power. CRD hydraulic lines originate at the CRD flanges. They are routed up from these flanges, curve 90°, and travel horizontally between the CRD housings. The lines are divided into two banks which exit the area under the vessel in two penetrations of the reactor support pedestal placed 180° apart. After traversing the drywell area, the lines exit the containment via two containment penetrations and are then routed to the two banks of hydraulic control units.

In the area under the reactor vessel, there is only one high-energy line, a two-inch lower vessel head drain which is one input to the RWCU system. This line is not considered a significant hazard to the CRD lines for several reasons:

(1) The CRD lines are routed below a set of I-beams. (The CRD housing support is attached to hanger rods which descend from these beams). Thus, the CRD lines are well shielded from the drain line which is above the I-beams.

(2) Breakage of this drain line would be a small LOCA. Normally, the reactor would continue to run, with the only problems being loss of some RWCU flow and a steam-feed flow mismatch. The reactor would not scram until the drywell pressure rose to the scram setpoint. This does not isolate the reactor and main feedwater would continue. Although some rods might fail to insert, and the resulting fission heat would have to be accommodated, the core would not uncover, and there would be no fuel melting.

(3) Even if main feedwater were lost, HPCI has the capacity to handle a 2-inch break (double-ended) with enough extra flow to supply about 40 bundles operating at average power. Again, the core would not uncover.

(4) If HPCI is insufficient, ADS can vent about 38% of rated steam flow. Thus, unless more than 38% of the rods fail to insert, ADS should be able to depressurize the vessel to the point where the high-capacity low pressure ECCS would keep the core flooded.

In any of these small-break scenarios, there would be no fuel melting because the core would not uncover, and there would be no reflood-induced reactivity transient. Depending on the number of control rods that fail to insert, steam production might exceed the turbine bypass capacity, or the MSIVs might close. In such a case, the heat sink provided by the RHR system would likely be insufficient to accommodate the extra heat, and the containment would eventually overpressurize and fail. This would not result directly in a major release of radioactivity, because there would be no severe fuel damage. In theory, the ECCS systems would eventually deplete the suppression pool and the core would eventually uncover. This situation would be alleviated by the fact that, as the suppression pool depletes, the standby liquid control system would become more effective because the concentration of sodium pentaborate in the coolant would increase as coolant boiled off, and fission heat would diminish. Alternatively, the standby coolant supply system could be used to augment the coolant supply.

In the area between the reactor support pedestal and the drywell wall, the situation is different. Here, the CRD lines pass near the reactor coolant piping and headers. The recirculation piping exits the vessel from two nozzles located near the bottom of the annulus and travels down through the general area where the CRD lines are located to the recirculation pumps which are at a still lower elevation. Flow from the pumps travels through two pipes up to two semi-circular manifolds, which again are in the general area of the CRD lines. Each manifold then supplies driving flow to the jet pumps through a series of risers, one riser for every two jet pumps. The CRD hydraulic lines cross this area under the manifolds. The usual practice is to route each bank in an array of six horizontal rows of hydraulic lines.

The rest of the vessel piping (feedwater, etc.) is located considerably higher in the drywell. This other piping is not considered a significant hazard because of its distance from the CRD lines and the rather narrow annular gap through which any missiles or jets would have to pass. Thus, concentration was placed on the recirculation piping. Given a break in the recirculation system, an estimate of the probability of crimping or sealing a line completely shut was needed. The best that could be done was to attempt to bound the true probability.

It should be noted that the outcome of the accident under consideration is relatively insensitive to scram timing, so long as the rods are successfully inserted. A small LOCA will not cause a reactor scram until either the water level drops to the scram setpoint or the drywell pressure rises to its setpoint. A large LOCA will depressurize the reactor and stop the fission chain reaction by high voiding of the moderator and the rods need not be inserted until the blowdown is complete. Thus, the interest was in complete rather than partial obstruction of the CRD lines, since partial obstruction would only delay, not prevent, the scram.

No credit was taken for the possibility that non-inserted rods might be widely dispersed and thus may not lead to recriticality. This was not as conservative as it first appeared. The CRD lines are not necessarily routed in such a manner as to disperse the drives they control, and blockage of adjacent lines may well inhibit scram in adjacent CRDs. (Two adjacent control rods can achieve criticality if withdrawn under cold conditions in a BWR.) Finally, insert and withdrawal lines were considered equally, since a large LOCA could depressurize the reactor before a rod with a crimped insert line is completely inserted. (This was in fact quite conservative.) The SLCS is normally capable of borating the moderator to 600 ppm of natural boron (referenced to cold water density) plus a 25% safety margin. This concentration would render the core up to 5% subcritical with all control rods fully removed at cold, xenon-free conditions at the most reactive point in core life. However, following a large LOCA, the SLCS effectiveness is reduced by the diluting effect of the suppression pool, which normally contains about 7½ vessel inventories. Thus, the SLCS can realistically borate only to about 88 ppm. Based on calculations done for ATWS, this would reduce power to roughly 75% of rated (with no rod insertion) but would not shut the reactor down.

Several effects help bring power down.[541] First, existing xenon, augmented by xenon increase, holds power down for roughly 24 hours after the accident. Second, the recirculation pumps are no longer providing forced flow through the core, which tends to bring power down by allowing more voiding. Finally, unless the pipe break area is small enough to limit leakage to less than ECCS injection, water level will drop to ⅔ of the core height, which will greatly reduce moderator density in the upper third of the core. Nevertheless, the core must eventually be brought to cold shutdown by means of the SLCS. Over the long term, this would not be difficult, since more sodium pentaborate mixture could be added to the SLCS so long as the secondary containment remained accessible. It was assumed that the SLCS would be ultimately used to render the core sub-critical over a span of several days.

An examination of the sequence of events was performed. A CRD line can be crimped completely shut by the impact of a missile or energetic fluid jet, if the circumstances are right. First, the line could be caught between the impacting mass and an opposing surface and be flattened shut. Second, if the impact occurred near a point of support for the line, the line could be severed and the stub bent over at a right angle. The line might then be flattened shut at the point of minimum radius of the bend. Finally, a sufficiently energetic impact theoretically could seal the line with only the inertia of the opposite side of the tube providing an opposing force.

In a study of design drawings and field walkdowns of three plants (Browns Ferry 3, Quad Cities 2, and Vermont Yankee) completed as part of the evaluation of Issue 156.6.1, it was found that the break of an RHR return line could also impact the CRD lines, in addition to the recirculation lines. With the exception of BWR/6 plants, the RHR systems in all BWRs are connected to the recirculation system. (In the BWR/6 design, the RHR system returns water to the RCS via a feedwater line or, in LPCI mode, directly into the core bypass region.) The RHR return lines range in size from 16 to 20 inches and connect to, and are unisolatable from, the recirculation lines. Based on rough measurements of MARK I plant drawings, the combined length of the unisolatable portions of the RHR lines (extending out to second isolation valves) was assumed to be 20% of the length of the recirculation lines.

The piping configuration for the three plants reviewed were broken down into two groups, depending on the plant configuration, and the calculations for each group were done separately considering three failure scenarios: **pipe whip; fluid jet impingement**; and **piping fragments.**

Group I: Browns Ferry 3 and Vermont Yankee
Group II: Quad Cities 2

Group II was created to characterize those plants in which a recirculation discharge line was believed to be in very close proximity to one-half of the CRD insert and withdraw lines. (See Figure 80-2.)

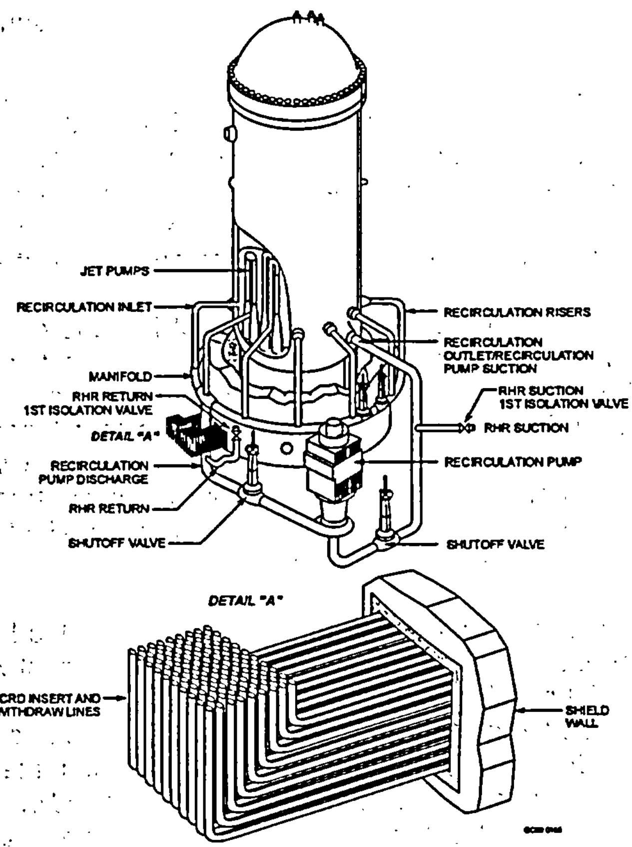

Figure 80-2
Group II plant piping layout

<u>Frequency Estimate</u>

<u>Pipe Whip</u>: In this scenario, a recirculation line breaks in such a manner that the whipping pipe strikes one bank of CRD hydraulic lines. It was assumed that the impact would block the entire bank, either by flattening the lines or by breaking the lines and bending them sharply. The CRD lines are located under the two semicircular recirculation manifolds. Thus, they are vulnerable to pipe whip primarily from the manifolds but also from the vertical recirculation pipes carrying flow to and from the recirculation pumps.

The frequency of a large break somewhere in the recirculation system has a mean distribution of **10^{-4}** event/RY. This number was modified to account for several spatial effects, based on the study of design drawings and the system walkdowns mentioned above[1811]:

- Break Location - Pipe whip restraints are located every 30° around the split manifold, except for two 60° intervals located at the ends of the two semicircles. To be a hazard to the CRD lines, the pipe break must be in the interval which spans the CRD lines. Therefore, a factor of **0.05** was used, which was the length of pipe in one 60° interval divided by the total length of recirculation piping.

- Vertical Piping - The CRD lines may be routed close enough to a recirculation pump suction or discharge line to be affected by breaks in these lines. This was conservatively accounted for by introducing a factor of **2.**

- Direction of Whip - The pipe break is as likely to cause the pipe to move sideways or away from the CRD lines as toward them. For this, a factor of **0.25** was assumed.

- Two CRD Line Banks - To account for the fact that there are two sets of lines 180° apart, a factor of **2** was used.

- Extent of Whip - Pipes are not expected to whip more than one pipe diameter at the maximum. In addition, although CRD line routing is done in the field, the fact that insulation has to be installed on recirculation lines gives assurance of at least a foot or so of clearance between the recirculation piping and the CRD lines. The probability that the pipe will whip far enough to hit the CRD lines was assumed to be **0.1**.

Multiplying the above numbers, the frequency of the partial (10%) core-melt scenario was estimated to be **(10^{-4})(0.05)(2)(0.25)(2)(0.1)** event/RY or **5 x 10^{-7}** event/RY. (In this and in subsequent calculations, the number of significant figures shown are not intended to imply that the various parameters are known to that accuracy. Instead, the extra figures are given to aid the reader in following the calculations. The uncertainties in these figures will be assessed quantitatively in the "uncertainties and sensitivities" section below.)

When the core is reflooded, about half the core will undergo a cold water reactivity transient. Cladding failure is not a concern here, since it was assumed that every fuel rod in the core would be perforated. Instead, it was necessary to examine the effect of the transient on the fuel matrix itself. The rod drop accident (licensing basis) inserts ~1.3% ΔK in about 0.6 seconds. Reflooding the reactor will insert about 8% ΔK, when filled with cold water (with xenon present). However, it takes about 30 seconds to refill the vessel from the bottom to the top of the core. Thus, the reactivity insertion rate is about a factor of eight below that of the rod drop accident and the rod drop accident is more limiting.

The licensing basis calculations for a control rod drop accident predict a peak fuel rod enthalpy of about 220 calories/gram when the inserted reactivity is 1.3% ΔK.[540] However, the rod drop accident initial conditions include an initial enthalpy of 20 calories/gram (540°F), whereas the cold water reflood transient under consideration here starts with fuel enthalpies as high as 85 calories/gram (2200°F). Thus, since the reactivity insertion rate in the reflood transient is less than the reactivity insertion rate in the rod drop accident, the rod drop accident enthalpy increase (ΔH) can be added to the initial enthalpy of the reflood transient and it can be concluded that the peak enthalpy achieved in the reflood transient will be less than 285 calories/gram.

This peak enthalpy corresponds to a point about 20% into the interval between onset of fuel melting (269.4 calories/gram) and complete melting (336.8 calories/gram). Therefore, we will bound the radiological effects of the reflood reactivity transient by assuming that the radioactive release due to this transient is at most 20% of a core-melt release in those fuel bundles where the associated control rods do not scram. Since only half of the control rods fail to scram, the release is bounded by one-half of 20%, or 10% of a full core-melt.

It should be noted that this estimate, which was used in the original analysis, is rather conservative. First, the assumed reactivity insertion rate was about a factor of eight higher than realistic. Second, the ΔH calculations do not take credit for moderator feedback; more realistic calculations have predicted ΔH values on the order of 100 calories/gram.[540] Finally, the duration of the hypothetical partially-molten state is very brief. Thus, it is doubtful that the reflood reactivity transient would directly cause this much fuel melting.

However, even if there is less fuel melting caused directly by a reactivity transient when the core is reflooded, it is likely that there will be at least some severe fuel damage in the region where the control rods do not insert. As a shutdown core is reflooded, individual fuel rods, now at a high surface temperature, will first experience film boiling and then "quench" as the cladding temperature drops and the rod transitions into nucleate boiling. However, if the control rods are not inserted, the linear heat generation rate in the fuel rods will greatly increase as the moderator returns and fission heat is generated in addition to decay heat. Even if the fission power is only a few percent of rated power, this would more than double the linear heat generation rate assumed in the ECCS analysis, and it is not likely that cladding temperatures will remain below 2200°F.

Finally, it should be noted that the amount of severe fuel damage will change the source term for purposes of calculating man-rem in a Level III PRA analysis. However, it will be shown later that the evaluation of this issue is governed by the Large Early Release Frequency (LERF), where the exact extent of fuel damage is of less importance given that there is at least some fuel melting.

After core reflood, fission power will continue at a low rate in the core.[541] The recirculation phase of ECCS may not be sufficient to remove this energy and the containment would then fail due to overpressure. Thus, the radioactivity released by the reactivity excursion would escape to the atmosphere in the manner of a BWR-2 release but with one-tenth its magnitude. In addition, the gap activity from the fuel which did not undergo a reactivity transient, and which would otherwise have been trapped within containment, would be released. There is no BWR release category for this situation, but the consequences of this release can be bounded by those of a PWR-8 release.

With the containment open and steam escaping to the atmosphere, the suppression pool will eventually be depleted of water. If the standby coolant supply system fails (for which a probability of **0.015** was assumed), there would be no liquid water supply for the ECCS and the entire core would melt. For this, a full BWR-2 release was assumed. The frequency for the full core-melt scenario was calculated to be **(0.015)(5 x 10^{-7})** event/RY or **7.5 x 10^{-9}** event/RY. Theoretically, the partial core-melt frequency should be reduced by a factor of (1 - 0.015), or 0.985, to account for those events that progress to a full core-melt. However, this difference produces an error that is <2% and will be neglected here. (The automated calculations used in the uncertainty studies described below will include this correction.)

<u>Group I Plants</u>: The analysis for the recirculation lines was expanded to include the RHR lines with the following assumptions: (1) the large-break LOCA frequency for the recirculation system is **10^{-4}** event/RY; (2) the ratio of the unisolatable portion of the RHR piping length to the total RHR/recirculation piping length is **0.2**; (3) only about one-third **(0.33)** of the RHR length of piping is near the CRD lines; (2) pipe whip may be towards or away from the CRD line bundle (or sideways), so that the probability of motion towards the bundle is **0.25**; (3) the probability that a pipe would whip far enough to contact a CRD line bundle is **0.1**; and (4) the scenario results in a 10% core-melt. Combining the result for the recirculation system from above, the frequency of a partial (10%) core-melt is given by the sum of **5 x 10^{-7}** event/RY (from the recirculation line break) and **(10^{-4})(0.2)(0.33)(0.25)(0.1)** event/RY (from the RHR line break). The result is a frequency estimate of or **6.6 x 10^{-7}** event/RY.

Again, if the standby coolant supply system is assumed to fail (for which a probability of **0.015** was assumed), there would be no liquid water supply for the ECCS and the entire core will melt. For this, a full BWR-2 release was assumed. The frequency of the full core-melt scenario was calculated to be **7.5 x 10^{-9}** event/RY + **(0.015)(1.6 x 10^{-7})** event/RY or **10^{-8}** event/RY.

<u>Group II Plants</u>: The analysis for the recirculation lines was expanded to include the RHR lines with the following assumptions: (1) the CDF increase for RHR piping is the same as calculated for Group I plants (**1.6 x 10^{-7}** event/RY); (2) the frequency of a recirculation line break is the same (**10^{-4}**

event/RY); (3) there is an additional contribution to CDF resulting from the recirculation piping being in close proximity to the CRD bundles; (4) the probability is **0.05** that, given a recirculation line pipe break, the break would be in the discharge line sector; (5) because pipe whip may be towards or parallel to the CRD line bundles that straddle it, there is a probability of **0.5** that the motion will be towards the bundles; (6) the probability that a pipe would whip far enough to contact a CRD line bundle is **1**; and (7) the scenario results in a 10% core-melt. Therefore, the frequency of a partial (10%) core-melt is given by the sum of **5×10^{-7}** event/RY (from the recirculation line break), **1.6×10^{-7}** event/RY (from the RHR line break), and **$(10^{-4})(0.05)(0.5)(1)$** event/RY (from the recirculation lines in close proximity). This results in a frequency estimate of **3.16×10^{-6}** event/RY.

Again, if the standby coolant supply system is assumed to fail (for which a probability of **0.015** was assumed), there would be no liquid water supply for the ECCS and the entire core will melt. For this, a full BWR-2 release was assumed. The frequency of the full core-melt scenario was calculated to be **7.5×10^{-9}** event/RY + **$(0.015)(3.16 \times 10^{-6})$** event/RY or **$4.74 \times 10^{-8}$** event/RY.

Fluid Jet Impingement: A fluid jet driven by a 1000 psi pressure cannot directly flatten a tube which contains 1000 psi fluid. However, impingement of such a jet will cause severe vibration of CRD lines. The lines may flatten as they repeatedly hit each other or hit any other structures (e.g., supports) which are within their vibrational amplitude. In reality, one would expect these lines to be more likely to rupture than to flatten. Nevertheless, flattening is possible and was assumed here.

The hazard to the CRD lines depends on their arrangement and distance from the pipe break. A typical practice in routing CRD hydraulic lines is to arrange the lines in six horizontal rows. In such an arrangement, lines located within the matrix would be shielded from some of the force of an external fluid jet. Thus, if the CRD lines are located close to the pipe break, the jet would be concentrated and might penetrate into the CRD lines matrix with sufficient force to cause vibratory flattening. Conversely, if the lines are located at some distance from the break, the jet would be more diffuse and less likely to penetrate past the first row of lines but will also, because of this same dispersion, impinge on a wider area and thus affect more of the outside row.

It was assumed that the break (and the jet) are 22-inches in diameter, which is the diameter of the recirculation manifold. (This is based on judgment. It is possible, of course, for the jet area to be any size from near zero to the equivalent of two pipe diameters, if the break is circumferential. If the break is longitudinal, the length of the break could theoretically extend the length of the manifold. A jet of one pipe diameter seems a reasonable first assumption.) To cover both the near and far cases, it was assumed that the entire top row of lines is flattened and, in addition, a 22-inch (transverse) span is flattened to a depth of all six rows. For a 1000 MWe plant with 185 control rods, this means that 43 rods would fail to insert; this corresponds to **23%** of the core.

The above was based on the assumption that the CRD lines are arranged in a matrix 6 rows high and with a pitch of two inches. In such a case, the matrix would be 62 inches wide. The probability of a break in the recirculation manifold being above this span is about **1.7%**.

The event tree is similar to that of a pipe whip: a recirculation line breaks (**10^{-4}/RY**), the break is above the CRD lines (**0.017**), and the fluid jet is directed downward (**0.25**). The result is that 23% of the core would experience a reactivity transient and continued steam production would eventually rupture the containment (20% of a BWR-2 release in the uncontrolled fuel plus a PWR-8 release). However, priority parameters calculated from these figures must be **doubled** to account for the presence of two banks of CRD lines and **doubled** again to account for the presence of vertical recirculation piping. Thus, the frequency of a partial or 4.6% core-melt (20% of 23%) was estimated to be **(10^{-4}) (0.017)(2)(2)(0.25)** event/RY or **1.7 x 10^{-6}** event/RY.

If the standby coolant supply system is assumed to fail **(0.015)**, the ECCS would eventually run out of water and the entire core would melt (BWR-2 release). The frequency for this full core-melt scenario was calculated to be **(0.015)(1.7 x 10^{-6})** event/RY or **2.55 x 10^{-8}** event/RY.

Group I Plants: The analysis for the recirculation lines was expanded to include the RHR lines with the following assumptions: (1) the large-break LOCA frequency for the recirculation system is **10^{-4}** event/RY; (2) the ratio of the unisolatable portion of the RHR piping length to the total RHR/recirculation piping length is **0.2**; (3) only about one-third **(0.33)** of the RHR length of piping is near the CRD lines; (4) the probability that the jet direction is towards the CRD lines is **0.1**; and (5) the scenario would result in a 4.6% core-melt. Combining the result for the recirculation system from above, the frequency of a partial (4.6%) core-melt was estimated to be **1.7 x 10^{-6}** event/RY + **(10^{-4})(0.2)(0.33)(0.1)** event/RY or **2.36 x 10^{-6}** event/RY.

If the standby coolant supply system is assumed to fail **(0.015)**, the ECCS would eventually run out of water and the entire core would melt (BWR-2 release). The frequency for this full core-melt scenario was calculated to be **2.55 x 10^{-8}** event/RY + **(0.015)(2.36 x 10^{-6})** event/RY) or **3.54 x 10^{-8}** event/RY.

Group II Plants: The analysis for the recirculation lines was expanded to include the RHR lines with the following assumptions: (1) the core-melt frequency increase for RHR piping and recirculation line breaks are the same as calculated for Group I **(6.6 x 10^{-7}** event/RY), plus other additions; (2) the probability is **0.05** that, given a recirculation line break, the break would be in the discharge line sector; (3) the portion of the recirculation discharge line directly adjacent to the CRD bundle is **0.5**; (4) the probability is **0.5** that the jet direction is towards the CRD lines; and (5) the scenario would result in a 4.6% core-melt. Therefore, the frequency of a partial (4.6%) core-melt is given by **1.7 x 10^{-6}** event/RY + **6.6 x 10^{-7}** event/RY + **(10^{-4})(0.05)(0.5)(0.5)** event/RY or **3.61 x 10^{-6}** event/RY.

If the standby coolant supply system is assumed to fail **(0.015)**, the ECCS would eventually run out of water and the entire core will melt (BWR-2 release). The frequency for this full core-melt scenario was calculated to be **2.55 x 10^{-8}** event/RY + **(0.015)(1.91 x 10^{-6})** event/RY or **5.41 x 10^{-8}** event/RY.

Pipe Fragments: The original analysis included the effects of pipe fragments on the CRD lines. Based on the additional insights gained during the evaluation of Issue 156.6.1, the failure modes of large reactor coolant piping were thermal fatigue or intergranular stress corrosion cracking which generally occurred in the region of circumferential welds. This suggested that clean breaks with the production of fragments is almost impossible. For the sake of completeness, the effect of pipe fragments will be discussed, but these sequences will not be included in the final analysis.

The hazard from pipe fragments is different from that of a fluid jet. First, because a solid object can concentrate its impact in a small area, it can block a CRD line directly by denting the line. Second, solid objects will retain this full impact over a great distance, as opposed to the diffusion of a fluid jet. On the other hand, a solid object cannot flatten a CRD line within the matrix without breaking the lines in the rows above.

The original analysis assumed that a section of recirculation manifold with a span equal to a pipe diameter (22 inches) suddenly breaks into fragments. To estimate the number of CRD lines which could be dented shut, it was further assumed that the lines are located immediately adjacent to the manifold. The pipe fragments, which at close range would act like one solid mass, would then impact a 22-inch span of the top row of CRD lines. Since these lines may well be all withdrawal lines, it was assumed that eleven control rods would fail to insert.

The accident sequence starts out with a large LOCA (**10^{-4}/RY**). The break must be over the CRD lines (**0.017**) and pointed down (**0.25**). The result is that 6% of the core would return to criticality after a mild reactivity excursion (20% of a BWR-2 release per fuel bundle) and the containment eventually would be overpressurized (75,000 man-rem from gap activity). This equates to a 1.2% partial core-melt. Again, the resultant figures must be multiplied by **four** to account for vertical pipes and two CRD banks. The frequency of this partial (1.2%) core-melt scenario is (10^{-4})(0.017)(0.25)(2)(2) event/RY or **1.7×10^{-6}** event/RY.

If the standby coolant supply system is assumed to fail (**0.015**), the entire core would melt (BWR-2 release). The frequency for this full core-melt scenario was calculated to be (**0.015**)(**1.7×10^{-6}**) event/RY or **2.55×10^{-8}** event/RY. Once again, these sequences are shown in Table 80-1 for comparison purposes only and were not included in the final analysis.

Table 80-1
Core-Melt Frequency Summary
Group I and Group II Plants

FAILURE MODE	GROUP I		GROUP II	
	Partial Core-Melt (Event/RY)	Full Core-Melt (Event/RY)	Partial Core-Melt (Event/RY)	Full Core-Melt (Event/RY)
Pipe Whip	6.60×10^{-7}	1.00×10^{-8}	3.16×10^{-6}	4.74×10^{-8}
Fluid Jet Impingement	2.36×10^{-6}	3.54×10^{-8}	3.61×10^{-6}	5.41×10^{-8}
Pipe Fragments	[Not included]	[Not included]	[Not included]	[Not included]
TOTAL:	**3.0×10^{-6}**	**4.5×10^{-8}**	**6.8×10^{-6}**	**1.0×10^{-7}**

Other Considerations

Uncertainties and Sensitivities: Many of the parameters involved in the estimates above are not "standard" PRA unavailabilities and, thus, do not have a commonly accepted distribution with mean estimates and error bounds. Nevertheless, an uncertainty analysis was performed although, in the absence of better numbers, the following judgment was used to estimate error bounds in some parameters:

Initiating event - large break LOCA	The "classic" distribution from NUREG-1150[1081] was used - a lognormal distribution, mean of 10^{-4}/RY, with a lognormal error factor of 10
Standby coolant supply unavailability	A lognormal distribution with an error factor of 10 was used, based on NUREG-1150,[1081] but using a mean from the original analysis. The effect of this will be examined in the sensitivity studies below.

Geometric factors, including the likelihood of a break being located above the CRD lines, or being between the two supports that bracket the CRD lines	In the original analysis, these parameters are based on a length of vulnerable circular manifold divided by the total length of primary system piping. This quotient was then multiplied by a factor of two (for two CRD banks) and another factor of two to account for an assumed equal length of vulnerable piping in the vertical runs. Since modern automated event tree analysis requires split fractions that are less than or equal to unity, the two factors of two were combined with the original quotient into just one parameter. Because this is still basically a ratio of lengths of pipe, the uncertainty distribution was assumed to be normal (rather than lognormal), centered on the point estimate in the analysis. For error bounds, the 5th and 95th percentiles were set at zero and at double the point estimate, based purely on judgment.
Direction, including direction of whip and direction of fluid jet	Depending on whether the pipe is within or outside of the CRD tube array, these parameters were either 50% or 25%. Based mostly on judgment (but partly on some piping diagrams), a normal distribution was used, with the 5th and 95th percentile limits set at ±0.2. Thus, the limits were at 0.30 to 0.80 and 0.05 to 0.45, respectively.
Extent of pipe whip	The analysis assumed a likelihood of 0.1 of the CRD lines being impacted by a whipping pipe. For this parameter, an exponential distribution with mean of 0.1 was used.
RHR fraction	This is the ratio of unisolable RHR piping to the total length of primary system piping. The original analysis estimated 0.2 for this parameter. For the uncertainty analysis, a normal distribution was used, with the 5th and 95th percentile limits set at 0.1 and 0.3, based on judgment.
RHR piping location	This is the fraction of RHR piping which is located near the CRD bundles, for the Group I plants. The analysis described above estimated this parameter to be 0.33. For the uncertainty analysis, a normal distribution was assumed, with the 5th and 95th percentile bounds set at zero and 0.66.
Fraction of recirculation piping located within the CRD bundles	This is the fractional length of piping located physically within the CRD bundles, for Group II plants. The analysis above used 0.05 (i.e., 5%). For the uncertainty analysis, a normal distribution was assumed, with the 5th and 95th percentile bounds set at 0.02 and 0.08.

The uncertainty analysis was constructed based on the above parameters, and distributions were calculated for the partial and full core-melt frequencies using 10,000 samples. For the original analysis, the results are shown in Table 80-2. Again, as the ranges in the Table 80-2 clearly indicate, the number of significant figures shown are not intended to imply that these results have high uncertainty, but instead are provided to assist the reader in following the calculations.

As can be seen, the means are not significantly higher than the point estimates. The distributions are not symmetric, as can be seen by how far the medians differ from the means. This is not surprising considering that the initiating event and the standby coolant supply unavailability are assumed to have log-normal distributions, but the geometric and directional parameters are assumed to have linear normal distributions. Moreover, some of the parameters were assigned 5th percentile bounds at zero, which "chops off" the lower 5% of the distribution and tends to lower the tail of the distributions of the products.

Table 80-2
Core-Melt Frequency (Event/RY) Uncertainties
Original Analysis

Event	End State	Point Estimate	Mean	5th percentile	95th percentile	Median
Fluid jet	4.7% core-melt	1.7E-6	1.8E-6	2.8E-8	7.0E-6	5.3E-7
	Full core-melt	2.6E-8	2.6E-8	7.1E-11	1.0E-7	<1.0E-8
Fragmentation	1.2% core-melt	1.7E-6	1.8E-6	2.8E-8	7.0E-6	5.3E-7
	Full core-melt	2.6E-8	2.6E-8	<1.0E-8	1.0E-7	<1.0E-8
Pipe whip	10% core-melt	4.9E-7	4.9E-7	<1.0E-8	2.0E-6	8.8E-8
	Full core-melt	<1.0E-8	<1.0E-8	<1.0E-8	2.5E-8	<1.0E-8

Starting with the original analysis, a series of changes and sensitivities were performed, the first of which was the removal of the contribution of fragmentation. The results are shown in Table 80-3.

Table 80-3
Core-Melt Frequencies (Event/RY)
Original Analysis With and Without Fragmentation Contribution

	End State	Point Estimate	Mean	5th percentile	95th percentile	Median
Original analysis	1.2%	1.7E-6	1.8E-6	2.8E-8	7.0E-6	5.3E-7
	4.7%	1.7E-6	1.7E-6	2.6E-8	7.0E-6	5.1E-7
	10%	5.0E-7	5.3E-7	<1.0E-8	2.1E-6	8.7E-8
	Full	5.8E-8	6.6E-8	<1.0E-8	2.3E-7	7.0E-7
Original analysis (no fragmentation)	4.7%	1.7E-6	1.7E-6	2.5E-8	7.0E-6	5.1E-7
	10%	5.0E-7	5.3E-7	<1.0E-8	2.1E-6	8.7E-8
	Full	3.3E-8	3.7E-8	<1.0E-8	1.3E-7	<1.0E-8

Here, the various states are summed by end state, and the "full core-melt " rows are the sums of the contributions of the pipe whip, fragmentation, and fluid jet scenarios. Although the point estimates for the full core-melt states are the sums of the individual full core-melt frequencies from the fluid jet, fragmentation, and pipe whip event trees, the means and limits are the result of adding up the three sequences 10,000 times while varying the initiating event frequency and split fractions about their distributions, and then forming a distribution for the sum.

Using the original analysis with the fragmentation contribution removed as a base, the sequences were modified to cover the Group I and Group II plants. The results are shown in Table 80-4.

Table 80-4
Core-Melt Frequency (Event/RY) Uncertainties
Original Analysis, Group I Plants, and Group II plants

	End State	Point Estimate	Mean	5th percentile	95th percentile	Median
Original analysis, no fragmentation	4.7%	1.7E-6	1.7E-6	2.5E-8	7.0E-6	5.1E-7
	10%	5.0E-7	5.3E-7	1.9E-9	2.1E-6	8.7E-8
	Full	3.3E-8	3.7E-8	<1.0E-8	1.3E-7	<1.0E-8
Group I plants	4.7%	2.3E-6	2.4E-6	6.1E-8	9.4E-6	8.0E-7
	10%	6.5E-7	7.0E-7	<1.0E-8	2.8E-6	1.2E-7
	Full	4.5E-8	5.2E-8	<1.0E-8	1.8E-7	<1.0E-8
Group II plants	4.7%	4.7E-6	4.7E-6	1.5E-7	1.8E-5	1.7E-6
	10%	3.1E-6	3.1E-6	9.2E-8	1.2E-5	1.0E-6
	Full	1.2E-7	1.3E-7	<1.0E-8	4.7E-7	1.7E-8

As can be seen from an examination of Table 80-4, the means do not vary significantly from the point estimates.

In addition to the calculations described in Table 80-4, two sensitivity studies were performed. The first was to examine possible double-counting of the vertical runs of RHR and recirculation piping. In the original analysis, the fraction of primary system piping physically located such that a break could threaten the CRD hydraulic lines was estimated by examining the layout of the split manifold, and then doubling the result to account for vertical piping runs for which no layout information was available. This is, in effect, an assumption that a vertical run of either RHR or recirculation piping, equal in length to the length of threatening pipe in the split manifold, is located close enough to pose a hazard to the CRD lines. This is a reasonable estimate for most plants, if no other information is available. However, the analysis of the Group I and II plants added vertical piping contributions to the original analysis. For Group II plants especially, if the analysis has added the contributions of vertical pipes known to be right in the middle of the CRD line bundles, it is known with equal certainty that these vertical pipes are <u>not</u> located in any other nearby location, and the original accounting for vertical piping runs should be removed.

The second sensitivity has to do with the availability of the standby coolant supply. This is not a stand-alone system. Although individual plants vary, every modern BWR has some means of pumping water from the ultimate heat sink into the reactor if the suppression pool is not available. Typically, this is done by providing a valved-out link between RHR service water and the RHR suction lines. Use of standby coolant supply requires a number of manual actions on the part of the operator. The original analysis for this issue used an unavailability for standby coolant supply of 1.5%, based on WASH-1400[16]-era analyses. The NUREG-1150[1081] Peach Bottom PRA performed a much more extensive analysis of the equipment and actions associated with standby coolant supply, and calculated a much higher unavailability (a mean of about 17%). For this screening analysis, the effect of increasing the unavailability of standby coolant supply is not to change the likelihood of an accident, but instead to change the end state from a partial core-melt to a full core-melt. The results of the two sensitivities are shown in Table 80-5.

Table 80-5
Core-Melt Frequency (Event/RY) Sensitivity Studies

	End State	Point Estimate	Mean	5th percentile	95th percentile	Median
Group II plants	4.7%	4.7E-6	4.7E-6	1.5E-7	1.8E-5	1.7E-6
	10%	3.1E-6	3.1E-6	9.2E-8	1.2E-5	1.0E-6
	Full	1.2E-7	1.3E-7	<1.0E-8	4.7E-7	1.7E-8
Group II plants (original vertical pipe contribution removed)	4.7%	3.9E-6	3.9E-6	1.3E-7	1.4E-5	1.4E-6
	10%	2.9E-6	2.8E-6	8.3E-8	1.1E-5	9.3E-7
	Full	1.0E-7	1.1E-7	<1.0E-8	4.0E-7	1.4E-8
Group II plants (no extra vertical pipe, modern standby coolant supply unavailability)	4.7%	3.3E-6	3.1E-6	9.3E-8	1.2E-5	1.1E-6
	10%	2.4E-6	2.3E-6	5.8E-8	8.9E-6	7.3E-7
	Full	1.2E-6	1.6E-6	2.3E-8	6.3E-6	4.4E-7

As can be seen from Table 80-5, removing the double-counting of vertical piping reduces the various core damage frequencies by about 20%, and an updated treatment of standby coolant supply increases the full core-melt frequency by an order of magnitude. This last sensitivity calculation, with the double-counting removed and the updated standby coolant supply, is the "best" estimate for this generic issue for the Group II plants - the most vulnerable group.

Containment Response: In any of these scenarios, even if the entire core is not damaged, the reactor core is not subcritical, and fission heat production continues. The RHR system is sized to remove decay heat. (For example, the Browns Ferry RHR has four heat exchangers rated at 70 million BTU/ hour each, which corresponds to about 2.5% of the reactor's rated thermal power of 3293 MW - equivalent to decay heat about 10 minutes after shutdown.) Obviously, if fission heat production continues with 23% of the rods failing to insert (as in the fluid jet scenario), and the standby liquid control system unable to shut the reactor down, the RHR system will not be able to accommodate the extra heat and the containment will overpressurize. Thus, any of these end states, even those involving partial core damage, will result in containment failure and a large early release.

Another perspective regarding containment response can be gained by examining the suppression pool inventory. Again, using the Browns Ferry plant as an example, the suppression pool inventory is 135,000 ft^3 (maximum), which is about 8.4 million pounds of water. Normal feedwater flow at full power is about 13.4 million pounds per hour. If fission power were to continue at about 10% of rated due to rods failing to scram, the entire suppression pool inventory would be boiled off in about 6.3 hours (not including the existing reactor water inventory, nor including the effect of residual heat removal, both of which would stretch the time somewhat).

End States: The end states in this screening analysis, i.e., 1.2%, 4.7%, 10% and 100% core-melt, are subject to considerable uncertainty. These numbers would be of significance if this calculation were carried out to PRA Level III consequences (e.g., man-rem/RY), as was done in the original analysis. However, a screening decision can be made based on LERF and thus the uncertainty in the degree of core damage was not explored.

Large Early Release Frequency (LERF): An uncertainty study was also performed for the LERFs for the various scenarios. The results are shown in Table 80-6.

Table 80-6
Large Early Release Frequencies (Event/RY)

	Point Estimate	Mean	5th Percentile	95th Percentile	Median
Plants with Original Analysis Piping Configuration (no fragmentation)	2.2E-6	2.4E-6	4.9E-8	9.5E-6	7.7E-7
Group I Plants	3.0E-6	3.3E-6	8.7E-8	1.3E-5	1.1E-6
Group II Plants	6.8E-6	7.0E-6	2.2E-7	2.6E-5	2.6E-6

It should be noted that some of these large early releases are much larger than others, since the total LERF includes sequences that breach containment, but only melt a small part of the reactor core. Nevertheless, the numbers are significant in that any increase in LERF greater than 10^{-6}/RY passes the screening tests documented in Figure C4 of the NRC Management Directive 6.4 Handbook.

It should also be noted that, for the specific case of Group II plants and using the NUREG-1150[1081] unavailabilities for standby coolant supply, the full core damage frequency is 1.2×10^{-6} event/RY, and all of these sequences lead to containment overpressurization and failure.

Basic Assumptions: There are several mechanistic or phenomenological postulates in the analysis that were not addressed in the uncertainty analysis because they were postulated to be true by the generic issue itself. Like an importance measure calculation, screening of a generic issue assumes these to be true, and then attempts to estimate their risk significance. Nevertheless, the task action plan for the issue should include an investigation of the validity of the following assumptions.

(1) Can a whipping pipe or other moving mass crush a CRD line completely shut? A CRD line pressurized to 1000 psi will resist denting. It is straightforward to show that the imposed force must be at least 1500 pounds per linear inch just to overcome the internal pressure, with no credit for the stiffness of the stainless steel tube wall, and assuming that the tube is in contact with a stationary support on the opposite side from the impacting mass.

(2) A fluid jet driven by 1000 psi cannot directly flatten a tube filled with 1000 psi fluid. The analysis assumes that the CRD lines will strike each other when exposed to the jet, and will flatten by repeated impacts.

(3) If a CRD line were crimped shut, would the internal fluid pressure be sufficient to overcome the stiffness of the stainless steel and partially re-open the tube, to the point where the associated control rod would eventually be inserted?

(4) The pipe whip analysis for the Group II plants assumes that the whipping pipe will bend the much-smaller CRD hydraulic lines to the point where the small lines will develop "kinks" which will close off all flow. The configuration of the CRD lines is such that the lines have a 90-degree bend or elbow near the point of impact, and are not likely to have significant lateral support - the impacting large pipe can bend the smaller lines without stretching them.

Experience suggests that this failure mode is quite credible, but the rather large number of CRD lines may add up to a significant resistive force, even though the individual lines may be relatively weak. An investigation of the force needed to "kink" a significant number of lines would be of considerable interest.

An investigation of these assumptions, either by calculation or by experiment, could add significant confidence to the resolution of the issue.

Early BWR Designs: As was described earlier, the control rod drive mechanism for all BWR/3 and later designs incorporates a ball check valve which prevents a broken insert line from interfering with a scram. There are still two operating plants, both of the BWR/2 product line, for which this may not be true. However, the BWR/2 design uses an ECCS for large-break LOCAs which is based on a high volume core spray - the core is not re-flooded after a large line break located below the level of the core. Thus, the accident scenarios associated with the issue do not apply to the BWR/2 design.

Old vs. New Analysis: An obvious question is, why has the conclusion changed from the original analysis? This is in spite of the fact that the pipe fragmentation sequences have been removed. An examination of the table will show two reasons. First and most obvious, the more vulnerable piping configurations in the Group I and Group II plants were not known when the original analysis was performed in 1984. These piping configurations are obviously of greater concern.

A second reason is more subtle. When the original analysis was performed, the generic issue screening criteria were based only on either core damage frequency or man-rem/RY. In 2002, a new set of criteria were added which were based on LERF. This particular generic issue involves a partial core-melt, and thus a relatively low source term and low public risk, but a high likelihood of containment failure, because only a small amount of fission heat will overwhelm the capacity of the RHR heat exchangers. Thus, the LERF criterion becomes limiting.

CONCLUSION

Applying the criteria of NRC Management Directive 6.4, Figure C4, the potential changes in the large early release frequencies (ΔLERF) place the issue in the category where work on a technical assessment should continue.[1809] The consideration of uncertainties in the analysis corroborates this conclusion.

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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 is 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 are 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, provide 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 last 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

No generic solution to this potential problem has yet been identified. Several possibilities exist, however. The first possibility is 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 x 10^{-6}/RY. Adding the tornado-induced accident frequency to this, we get a frequency for this class of accidents of 2.2 x 10^{-6}/RY.

Consequence Estimate

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 x 10^{-6}	7.4 x 10^{6}	16.3
BWR-3	7.8 x 10^{-8}	6.5 x 10^{6}	0.5
BWR-4	1.4 x 10^{-6}	1.1 x 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 this potential problem has yet been settled upon. However, any hardware addition would probably have to be seismic Category I and, thus, costs are unlikely to be less than $1M/reactor. NRC costs will 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:

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

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

Other Considerations

It should be noted that a low seismic probability will drop the above estimates to about 200 man-rem/reactor and 200 man-rem/$M. This will not change the final conclusion. In any case, this analysis was based on a specific pool design which was picked 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]

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ISSUE 83: CONTROL ROOM HABITABILITY

DESCRIPTION

Historical Background

On August 18, 1982, the ACRS issued a letter[671] to the Commission which: (1) identified deficiencies in the maintenance and testing of engineered safety features designed to maintain control room habitability; (2) provided examples of design and installation errors, including inadvertent degradation of control room leak tightness; and (3) cited a shortage of NRC and licensee personnel knowledgeable about HVAC systems and nuclear air-cleaning technology. These ACRS concerns encompassed both plant licensing review and operations/inspection activities. In January 1983, the staff responded[436] to the ACRS concerns and recommended increased training of NRC and licensee personnel in inspection and testing of control room habitability systems. The staff also provided a profile of control room HVAC system component failures based on an analysis of LERs from 1977 through mid-1982. On April 28, 1983, NRR and OIE representatives met with the ACRS Subcommittee on Reactor Radiological Effects to discuss the staff response.

In May 1983, the ACRS issued a letter[673] to the EDO which expressed continuing concerns about control room habitability and provided both general and specific comments and recommendations for further staff evaluation. In July 1983, NRR transmitted to the EDO a joint NRR/OIE proposal[674] for evaluating the ACRS comments and recommendations and the adequacy of the control room habitability licensing review process and criteria. In August 1983, the EDO indicated agreement[675] with the proposal and directed NRR to coordinate with OIE and the NRC Regional Offices to complete the program and submit a report to the EDO by June 1, 1984. In September 1983, NRR established[676] a Control Room Habitability Working Group and a Steering Group for conducting and guiding the proposed review. Other generic issues that addressed related concerns were B-36, B-66, and III.D.3.4.

Safety Significance

Loss of control room habitability following an accident release of external airborne toxic or radioactive material or smoke can impair or cause loss of the control room operators' capability to safely control the reactor and could lead to a core-damage accident. Use of the remote shutdown station outside the control room following such events is unreliable since this station has no emergency habitability or radiation protection provisions similar to the control room.

Possible Solution

The Control Room Habitability Work Group was expected to identify any recommended actions that would correct significant deficiencies in control room habitability design, installation, test, or maintenance.

CONCLUSION

In June 1984, NRR provided a report[678] to the EDO along with its plans for implementing the recommendations of the report, including a survey of several operating plants. Based on the ongoing staff work, it was concluded that a solution had been identified and a schedule[679] for the resolution of the issue was developed by DSI/NRR.

PNL completed a report[677] entitled "A Probabilistic Examination of Nuclear Power Plant Control Room Habitability During Various Accident Scenarios," and the findings of the survey of operating plants were published in NUREG/CR-4960.[1371] As a result of these studies, it was recognized that the methodology used to evaluate control room habitability system design needed improvement. Accordingly, the staff initiated activities to develop: (1) improved methods for calculating control room dose and exposure levels; (2) improved meteorological models for use in control room habitability calculations; and (3) revised exposure limits to toxic gases for control room operators.

The results of the improved methods were documented in NUREG/CR-5669[247] and NUREG/CR-6210[249] and the HABIT Code was developed to provide an integrated code package for evaluating control room habitability. NUREG-1465,[1465] published with the resolution of Issue 155.1, will provide updated source term information for the evaluation of control room designs. As recommended[296] by the ACRS, the staff was expected to consider NIOSH recommendations for toxic chemicals in its revision of Regulatory Guide 1.78.[1373] Thus, this issue was RESOLVED with no new requirements.[335] Consideration of a license renewal period of 20 years would not have changed this conclusion.

However, in June 2003, NRC Generic Letter 2003-01[1812] was issued to address findings at U.S. nuclear power plants which suggested that licensees may not have been meeting the control room licensing and design bases, and applicable regulatory requirements, and existing TS surveillance requirements may not have been adequate. The affected licensees were requested to submit information demonstrating that their control rooms complied with existing licensing and design bases, and applicable regulatory requirements, and that suitable design, maintenance, and testing control measures were in place.

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ISSUE 192 : SECONDARY CONTAINMENT DRAWDOWN TIME

DESCRIPTION

Historical Background

This issue was raised[1789] by NRR and addresses the adequacy of the calculations, testing, and acceptance criteria related to the creation of a vacuum in the reactor building of a BWR, following an engineered safeguards actuation signal.

The time required to attain a vacuum in the reactor building is commonly referred to as the "drawdown time." The vacuum is necessary to ensure that any air leakage flows into the building so that any radiological contamination in the building air is processed by the appropriate safety systems, before being released to the environment. Guidelines for including the drawdown time in offsite and control room dose calculations are specified in BTP CSB 6.3 of SRP[11] 6.2.3, "Secondary Containment Functional Design." Independent calculations showed that plants could potentially exceed the limits for offsite and control room doses.

Safety Significance

The Standby Gas Treatment System (SGTS) provides a means for minimizing the release of radioactive material from the containment to the environment by filtering and exhausting the atmosphere from any or all zones of the reactor building during containment isolation conditions. The SGTS is classified as an Engineered Safety System. The design basis for the system is to prevent any uncontrolled release due to a design basis LOCA (during power operation), or due to a fuel handling accident (during refueling conditions, when the primary containment is open and the secondary containment is the only containment).

During normal operation, the reactor building is heated, cooled, and ventilated by a circulating air system, which generally exhausts from the reactor building roof with minimal or no filtration. This reactor building HVAC system is shut down and isolated when the secondary containment is isolated and connected to the SGTS. The SGTS will initiate automatically on reactor zone high radiation, refueling zone high radiation, low reactor water level, or high drywell pressure. In addition to this primary function, the SGTS also has other uses of interest:

- It is used to process exhaust gases from the gland seal condenser of the HPCI turbine. Normally, the signals which initiate HPCI (e.g., low-low reactor water level) will also initiate the SGTS.
- It is used to test secondary containment integrity.
- It is used to purge air from the drywell and suppression pool air space when necessary (e.g., prior to personnel entry).

The size and location of the SGTS is site-specific. However, the SGTS will normally consist of more than one train, and be capable of performing its function with one train out of service. (A single-unit site will generally have two trains, but a multiple-unit site, where there may be one reactor building housing two primary containments, may have a three-train shared system.)

Each SGTS train generally consists of a (shared) suction duct system, a moisture separator and heater to keep humidity within limits, then a set of particulate filters and charcoal adsorbers plus a blower. The train will discharge to the plant stack, to provide an elevated release. The system is design to remove particulates and iodine. Unlike the offgas system, there is no holdup pipe to allow the noble gases to decay before release.

If the SGTS system is used to exhaust just a few individually-isolated zones, it is possible to draw a significant vacuum. In order to prevent structural damage, there is also a standby gas treatment vacuum relief system, which will bleed outside air into each zone of the reactor building to prevent the outside pressure from exceeding the inside pressure by more than a certain amount (e.g., ½ inch water gauge).

There are several limiting conditions for operation and surveillance requirements in the TS regarding SGTS operability.[706] These generally include running the system monthly plus verifying each refueling cycle (or every 18 months) that the flow rate through the system and pressure drop across the various filters are within specification.

In addition, there is generally a surveillance requirement that secondary containment integrity be demonstrated every 24 hours by verifying that the secondary containment interior pressure is at least 1/4 inch water gauge vacuum. This is checked daily, and the negative pressure is maintained by the normal HVAC system. Once per 18 months, it is verified that one train of the SGTS can draw down the secondary containment to 1/4 inch water gauge within a set time (120 seconds in the STS). This test verifies both SGTS system efficacy and leak tightness of the secondary containment with the normal HVAC isolated. Some older TS did not include a time limit. [See Appendix A to Facility Operating License DPR-33, Technical Specifications and Bases for Browns Ferry Nuclear Plant Unit 1, Limestone County, Alabama, Tennessee Valley Authority, Docket No. 50-259, Amendment 50, September 15, 1981.]

The three specific safety concerns raised[1789] by NRR were:

(1) Calculations for reactor facilities (primarily Brunswick, Cooper, and BWR/4 plants and earlier) are performed using a single volume to represent the secondary containment. This doesn't account for the compartmentalization of the building and different heat sources in different compartments. Some compartments, perhaps compartments with sources of radioactivity, may not depressurize as fast as others and may be potential leakage paths.

(2) Reactor facilities (primarily Brunswick, Cooper, and BWR/4 plants and earlier) measure the vacuum in only one location in the secondary containment. This location may not be in the most conservative location (the last area to reach the desired vacuum).

(3) The criterion used, 0.25 in-water vacuum, only accounts for pressure distribution around the building due to wind. It does not account for the difference in inside temperature during a cold test and during a LOCA accident.

Possible Solutions

The documentation presented with the issue did not give an explicit recommended solution. Presumably, the solution would be to re-calculate the drawdown time with more accurate models. The objective was to ensure that, if an appropriate pressure reduction is achieved at the point where the pressure is being measured, an appropriate pressure reduction is being achieved throughout the entire secondary containment.

If the drawdown time were then shown to be too long, a modification to the SGTS would be required. This could involve the installation of more ducting to provide multiple suction points for the SGT, and/or actions to minimize inleakage to the secondary containment, especially at points near the SGTS intake(s), where a leak could prevent more remote areas from being drawn down.

In theory, a fix might involve an upgrade to the flow capacity of the SGTS system itself. It should be noted, however, that the SGTS flow is normally adjusted to be within a certain range under the assumption that, if design SGTS flow does not achieve the desired pressure reduction, there are leaks to be fixed. Simply increasing SGTS flow might reduce pressure in areas near the SGTS intake, but might not achieve the required pressure reduction everywhere.

ASSESSMENT

Frequency Estimate

Large-Break LOCA: A large-break LOCA will cause widespread failure of the cladding integrity due to departure from nucleate boiling, resulting in the release of gap activity to the primary coolant. The accident will also release this primary coolant to the containment atmosphere. Smaller-break LOCAs will also release primary coolant to the primary containment atmosphere, but will not necessarily release gap activity from the fuel rods and thus were not included here. The "classic" large LOCA frequency of 10^{-4} event/RY was assumed.[16]

Fuel Handling Accident: The other design basis accident for the SGTS system is a fuel handling accident where a fuel assembly is mechanically damaged and gap activity is released. This accident is not normally modeled in modern PRAs because it is generally not a significant contributor to a plant's total risk profile. It was addressed many years ago in WASH-1400[16] (Appendix I, p. I-100) which estimated a frequency of 10^{-4} event/RY. (This is the frequency of events in which gap activity is actually released; it is not the frequency of all events in which a fuel assembly is dropped or otherwise mishandled.)

Consequence Estimate

Large-Break LOCA: The essence of Concerns 1 and 2 was that the calculations and measurement techniques used to measure the efficacy of the system may have been too primitive. The secondary containment is not a single volume, but the calculations may model the secondary containment as one large volume, and the vacuum may be measured at just one point. The practical effect of this is that it may take longer than expected for the SGTS system to draw the secondary containment down to the required vacuum and some compartments (e.g., a compartment containing a leak from the primary containment but located such that the pathway to the SGTS system intake is long) may never achieve the required vacuum at all. The essence of the third concern was that the 0.25 inch water gauge vacuum criterion may not have been sufficient, since it was based only on overcoming external wind conditions.

In either case, to evaluate the risk significance of the issue, it was necessary to estimate the "worth" of the SGTS system in terms of averted public dose, given that a design-basis LOCA has occurred.

Ideally, the risk worth of the SGTS system could be calculated by using the source term for a successfully mitigated LOCA with the containment losing inventory at the design leakage rate. The calculation would then be done with and without the SGTS system. The SGTS system should greatly reduce the public dose because of its filtration of the air flow and because the SGTS discharge air is routed to the plant stack, resulting in an elevated release.

Unfortunately, few modern PRAs model a mitigated LOCA since such events, which do not result in a severely damaged core, are not risk significant. The only readily-available probabilistic analysis which included mitigated LOCA sequences was the original WASH-1400[16] calculation. The release category of interest was:

BWR-5 This category approximates a BWR design basis accident (large pipe break) in which only the activity initially contained within the gap between the fuel pellet and cladding would be released into containment. The core would not melt, and containment leakage would be small. It is assumed that the minimum required engineered safeguards would function satisfactorily. The release would be filtered and pass through the elevated stack.

If there were a similar BWR release category in which the SGTS was not functioning, a simple comparison would give the risk worth of the SGTS system. Such a release category was not used in WASH-1400.[16] However, there is an analogous category for the PWR analysis:

PWR-9 This category approximates a PWR design basis accident (large pipe break), in which only the activity initially contained within the gap between the fuel pellet and cladding would be released into the containment. The core would not melt.

Can this category, which applies to a completely different reactor design, be used for comparison? The radioactive inventories of the two designs are very similar and it was reasonable to assume that the gap activity releases were similar. A comparison of some containment parameters[1790] was summarized in Table 3.192-1.

Although the PWR containment is much larger, having a free volume almost a factor of six greater than the combined BWR drywell and suppression chamber free volume, the design leak rate for the PWR is one-fifth of the BWR rate. The two effects almost cancel and the leak rate for the PWR containment is only 20% larger than that of the BWR primary containment.

As listed in the Introduction, the man-rem associated with the two release categories of interest were as follows:

BWR-5	20 man-rem
PWR-9	120 man-rem

Table 3.192-1

WASH-1400[16] BWR		WASH-1400[16] PWR	
Core thermal power	3293 MWt	Core thermal power	2441 MWt
Drywell free volume	175,000 cubic feet	Free volume	1.8 x 10^6 cubic feet
Suppression chamber air volume	127,700 cubic feet		
Drywell design pressure	56 psig	Design pressure	60 psig
Suppression chamber design pressure	52 psig		
Design leak rate	0.5% per day	Design leak rate	0.1% per day
Calculated leak rate at design pressure	1500 cubic feet/day	Calculated leak rate at design pressure	1800 cubic feet/day

The WASH-1400[16] source terms were calculated using a "representative" reactor core power of 3200 MWt (Appendix VI, Section 3.2, p. 3-1), and the man-rem figures above assumed this core power. Thus, it was not necessary to adjust for core power differences.

Other than a minor difference in leak rate, the difference in consequences between these two release categories is presumably due primarily to the presence of the secondary containment and SGTS. (Some other effects will be addressed below in the discussion of uncertainties.) Thus, this reduction is estimated to be :

$$(120 \text{ man-rem})[(1500 \text{ ft}^3/\text{day})/(1800 \text{ ft}^3/\text{day})] - 20 \text{ man-rem} = 80 \text{ man-rem}$$

It was conservatively assumed that this entire reduction is due to the SGTS system. In actuality, if the SGTS system were not functioning, there would still be some retention of radioactive material in the secondary containment structure. This estimate of 80 man-rem is somewhat of an overestimate of the worth of the SGTS system by itself.

This factor of five reduction (from 100 to 20 man-rem) is significant. The 20 man-rem is likely due primarily to noble gases, since they, unlike particulates and iodine, will not be removed by the SGTS system. Even the noble gases will have fewer health effects, because the SGTS system will release them at an elevated location.

Fuel Handling Accident: A fuel handling accident releases gap activity due to mechanical damage to the fuel pins, such as could happen if a fuel assembly were dropped during refueling or during normal fuel pool operations. Although the release would be modest, the accident would not take place within a closed primary containment, and the secondary containment would be the only containment. The SGTS system would then preclude an "uncontrolled" release to the environment.

Unfortunately, there is no readily-available calculation of the consequences of such a release. However, it is possible to put an upper bound by adapting the PWR-8 release, which is a mitigated large-break LOCA where the containment fails to isolate. This is extremely conservative, in that the PWR-8 release category deals with fuel which was in full power operation just moments earlier, and also includes the release of all the primary coolant activity in the primary system. The fuel involved in a fuel handling accident would have had time for some radionuclides to decay away, and would not involve any primary coolant activity. Moreover, a fuel handling accident would be expected to occur with the fuel assemblies fully submerged, and any activity released would be scrubbed by the overlaying water. Thus, this upper bound may be conservative by two or more orders of magnitude.

As documented in the Introduction, the consequences calculated for a PWR-8 release are 75,000 man-rem; this is for an entire reactor core. The reactor of interest here consists of 764 fuel assemblies (NUREG/CR-5640,[1790] Table 8.3-1, page 8-17). Thus, the normalized release for one fuel assembly is about 100 man-rem. It was assumed that the SGTS is capable of reducing this to zero.

Cost Estimate

A cost estimate was not performed. The results of the value assessment are such that a cost estimate is not necessary.

Screening Assessment

The screening criteria of CDF and large early release frequency (LERF) were not applicable to this issue. The SGTS is a mitigative system and does not affect CDF or LERF. The applicable criterion was averted offsite man-rem/year, based on the absolute value of the system's risk worth. The risk reduction for the large-break LOCA is the product of the frequency of large-break LOCAs multiplied by this risk worth:

$$(10^{-4}\ \text{LOCAs/RY})(80\ \text{man-rem/LOCA}) = 0.008\ \text{man-rem/RY}$$

Similarly, the risk reduction associated with the fuel handling accident is given by:

$$(10^{-4}\ \text{event/RY})(100\ \text{man-rem/event}) = 0.01\ \text{man-rem/RY}$$

The two risk reductions add up to 0.018 man-rem/RY. Summing the BWR/2, BWR/3, and BWR/4 plants, there were 27 reactors affected by this issue. This implied a total value of 0.5 man-rem/year. Comparing with Figure C6 of NRC Management Directive Handbook 6.4, this issue warranted exclusion from further consideration, regardless of cost.

Uncertainties

(1) At the time of this evaluation in June 2002, there was some ongoing research regarding whether the large-break LOCA frequency of 10^{-4}/RY was too high. However, lowering this frequency would not change the conclusion.

(2) It is possible that some intermediate-LOCA events could also result in significant gap activity being released into the primary containment atmosphere. However, even including the small ("S1") LOCAs would increase the frequency by a factor of ten, which would not be enough to change the conclusion.

(3) As was discussed above, the estimate of change in man-rem estimated above includes more than just the effect of the SGTS system. In addition to this, the first two concerns raised in the issue have to do with transient conditions: Can the SGTS draw a vacuum in the secondary containment in time? Presumably, once the secondary containment reaches the desired subatmospheric pressure, the SGTS system would still be efficacious and the above estimate of change in man-rem would be conservative. This also would have no effect on the conclusion.

(4) The analysis approximated the consequences of a mitigated LOCA in a BWR with no credit for the SGTS system by using a PWR-9 release, although the assumed leakage from primary containment differs by about 20%. In addition to this difference, an actual release pathway in such circumstances in a BWR large LOCA sequence would at least partially be from the suppression pool airspace, where the release would have been scrubbed by the suppression pool water. Also, some of the particulates and aerosols would be removed by plateout in the reactor building. Finally, unlike a BWR release, a PWR release would be entirely at ground level. Thus, the approximation of a BWR release with no SGTS system by a PWR-9 release was conservative. This also would have no effect on the conclusion.

(5) In the calculations above, a core power of 3200 Mwt was assumed. In actuality, most of the affected reactors are not licensed to this high a power, so the consequences were somewhat overestimated.

(6) The estimate of the fuel handling accident consequences were extremely conservative. This is appropriate for screening a generic issue where conservatism is normally included and does not affect the conclusion. However, the value estimated here is not appropriate for use as a best estimate for other purposes.

Other Considerations

(1) This issue appeared to have no effect on the ability of the SGTS to accommodate the exhaust from the HPCI gland seal. Moreover, the HPCI system is not used to mitigate a large LOCA.

(2) Similarly, this issue does not affect the ability of the SGTS to test secondary containment integrity.

(3) The analysis above dealt with a design basis LOCA event, not a severe (i.e., core-melt) accident. In theory, the presence of the SGTS could help mitigate the radiological effects of a core-melt. In practice, the SGTS will not have the capacity to make a significant difference in the release associated with a core-melt, even if upgraded. Moreover, in most severe accident sequences, the core would not actually melt immediately. Generally, severe core damage would not occur until the existing SGTS is able to achieve drawdown in any case.

Moreover, if a core-melt event were to occur in a BWR, one strategy for dealing with the situation would be to intentionally vent the suppression pool air space to the outside, thereby preventing containment failure due to overpressure, but also using the suppression pool to scrub the release. In such an event, SGTS drawdown time is unlikely to be of much concern.

(4) Another effect of the presence of the SGTS is to reduce personnel exposure within the plant area, allowing more freedom of movement for plant personnel to take mitigative actions. This is rather limited in that there is no effect inside the secondary containment. In addition, the control room has its own separate ventilation system with a filtered intake. Thus, the effect of the SGTS is limited to other areas, such as outdoors near the reactor building. There was no simple way to quantify this effect, but it was unlikely to be major.

CONCLUSION

The low potential risk reduction associated with this issue implied that it was very unlikely that a backfit requirement could be imposed under 10 CFR 50.109. However, the originator of the issue has made a valid point in that SRP[11] 6.2.3 does state in acceptance Criterion 3a that "The secondary containment depressurization and filtration systems should ... be capable of maintaining a uniform negative pressure throughout the secondary containment, as well as other areas served by the systems." In addition, Criterion 1g states that "heat loads generated within the secondary containment (e.g., equipment heat loads) should be considered." In hindsight, it appeared that these acceptance criteria were not fully used in the reviews of individual BWR TS. Based on the above estimates of averted offsite man-rem, this issue was DROPPED from further consideration.[1813]

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

Appendix B (Continued)

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

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
II.K.1(4)	Review Operating Procedures and Training Instructions	NOTE 3(a)	All	All		03/31/80	NA
II.K.1(5)	Safety-Related Valve Position Description	NOTE 3(a)	All	All		03/31/80	03/31/80
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	NOTE 3(a)	All	All		03/31/80	NA
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	NOTE 3(a)	NA	B&W		03/31/80	NA
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	NOTE 3(a)	NA	B&W		03/31/80	NA
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred out of Containment Inadvertently	NOTE 3(a)	All	All		03/31/80	NA
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	NOTE 3(a)	All	All		03/31/80	03/31/80
II.K.1(11)	Make All Operating and Maintenance Personnel Aware of the Seriousness and Consequences of the Erroneous Actions Leading up to, and in Early Phases of, the TMI-2 Accident	NOTE 3(a)	All	All		03/31/80	NA
II.K.1(12)	One Hour Notification Requirement and Continuous Communications Channels	NOTE 3(a)	All	All			NA
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	NOTE 3(a)	All	All		01/01/81	01/01/81
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	NOTE 3(a)	GE	CE, W		03/31/80	NA
II.K.1(15)	For Facilities with Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW	NOTE 3(a)	NA	CE, W		NA	
II.K.1(16)	Implement Procedures That Identify PRZ PORV "Open" Indications and That Direct Operator to Close Manually at "Reset" Setpoint	NOTE 3(a)	NA	CE, W		NA	
II.K.1(17)	Trip PZR Level Bistable so That PZR Low Pressure Will Initiate Safety Injection	NOTE 3(a)	NA	W			
II.K.1(18)	Develop Procedures and Train Operators on Methods of Establishing and Maintaining Natural Circulation	NOTE 3(a)	NA	B&W		NA	
II.K.1(19)	Describe Design and Procedure Modifications to Reduce Likelihood of Automatic PZR PORV Actuation in Transients	NOTE 3(a)	NA	B&W		03/31/80	NA
II.K.1(20)	Provide Procedures and Training to Operators for Prompt Manual Reactor Trip for LOFW, TT, MSIV Closure, LOOP, LOSG Level, and LO PZR Level	NOTE 3(a)	NA	B&W		03/31/80	03/31/80

Appendix B (Continued)

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

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		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level - Design and Modification	I	GE	NA	F-50	01/01/81	01/01/81
II.K.3(22)	Automatic Switchover of RCIC System Suction - Verify Procedures and Modify Design	I	GE	NA	F-51	01/01/81	01/01/81
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	I	GE	NA	F-52	01/01/82	01/01/82
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	I	GE	NA	F-53	01/01/82	01/01/82
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	I	GE	NA	F-54	10/01/80	10/01/80
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	I	GE	NA	F-55	01/01/82	01/01/82
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	I	GE	NA	F-56	04/01/81	NA
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	I	All	All	F-57	01/01/83	01/01/83
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	I	All	All	F-58	01/01/83	01/01/83
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	I	GE	NA	F-59	01/01/81	01/01/81
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	I	GE	NA	F-60	01/01/81	01/01/81
II.K.3(46)	Response to List of Concerns from ACRS Consultant	I	GE	NA	F-61	07/01/80	07/01/80
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	I	GE	NA	F-62	10/01/80	NA
<u>III.A</u>	<u>EMERGENCY PREPAREDNESS AND RADIATION EFFECTS</u>						
<u>III.A.1</u>	<u>Improve Licensee Emergency Preparedness - Short Term</u>						
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
<u>III.A.2</u>	<u>Improving Licensee Emergency Preparedness-Long Term</u>						
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

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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			
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

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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			
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 Iand 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

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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			
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
168.	Environmental Qualification of Electrical Equipment	HIGH	All	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	NOTE 4	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	NOTE 4	All	NA		TBD	TBD
194.	Implications of Updated Probabilistic Seismic Hazard Estimates	NOTE 4	All	All		TBD	TBD
195.	Hydrogen Combustion in Foreign BWR Piping	NOTE 4	All	NA		TBD	TBD
		HUMAN FACTORS ISSUES					
HF1	STAFFING AND QUALIFICATIONS						
HF.1.1	Shift Staffing	NOTE 3(a)	All	All		01/--/84	01/--/84

NRC FORM 335
(2-89)
NRCM 1102,
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-0933
Supplement 27

2. TITLE AND SUBTITLE

A Prioritization of Generic Safety Issues

3. DATE REPORT PUBLISHED

MONTH	YEAR
December	2003

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

R. Emrit, et al.

6. TYPE OF REPORT

7. PERIOD COVERED *(Inclusive Dates)*

07-01-2002 to 06-30-2003

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

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.

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

generic safety issues

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)

unclassified

(This Report)

unclassified

15. NUMBER OF PAGES

16. PRICE

NRC FORM 335 (2-89)

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