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

AUGUST 1992

SUPPLEMENT 14 TO NUREG-0933
"A PRIORITIZATION OF GENERIC SAFETY ISSUES"
REVISION INSERTION INSTRUCTIONS

	Remove	Insert
Introduction	pp. 27 to 60, Rev. 13	pp. 27 to 60, Rev. 14
Section 1	pp. 1.II.B-1 to 14, Rev. 2	pp. 1.II.B-1 to 14, Rev. 3
	pp. 1.II.J.4-1	pp. 1.II.J.4-1, Rev. 1
Section 3	pp. 3.24-1	pp. 3.24-1 to 4, Rev. 1
	pp. 3.29-1 to 5	pp. 3.29-1 to 6, Rev. 1
	pp. 3.38-1	pp. 3.38-1 to 2, Rev. 1
	pp. 3.73-1	pp. 3.73-1 to 3, Rev. 1
	pp. 3.87-1 to 7	pp. 3.87-1 to 7, Rev. 1
	-	pp. 3.100-1 to 4
	-	pp. 3.123-1 to 4
	pp. 3.128-1	pp. 3.128-1 to 3, Rev. 1
	pp. 3.130-1 to 9	pp. 3.130-1 to 9, Rev. 1
	pp. 3.133-1 to 2	pp. 3.133-1 to 2, Rev. 1
	pp. 3.135-1 to 2, Rev. 1	pp. 3.135-1 to 2, Rev. 2
	-	pp. 3.138-1 to 6.
	pp. 3.142-1 to 9	pp. 3.142-1 to 9, Rev. 1
	-	pp. 3.150-1 to 7
	-	pp. 3.151-1 to 4
	-	pp. 3.156-1 to 6
Appendix B	pp. A-9 to 21, Rev. 5	pp. A-9 to 22, Rev. 6

9209170127 920831
PDR NUREG
0933 R PDR

TABLE II

LISTING OF ALL TMI ACTION PLAN ITEMS, TASK ACTION PLAN ITEMS, NEW GENERIC ISSUES, AND HUMAN FACTORS 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[98]
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

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	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	2	12/31/86	F-01
I.A.1.2	Shift Supervisor Administrative Duties	-	NRR/DHFS/LQB	I	2	12/31/86	
I.A.1.3	Shift Manning	-	NRR/DHFS/LQB	I	2	12/31/86	F-02
I.A.1.4	Long-Term Upgrading	Colmar	RES/DFO/HFBR	NOTE 3(a)	2	12/31/86	
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	5	12/31/87	F-03
I.A.2.1(2)	Training	-	NRR/DHFS/LQB	I	5	12/31/87	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	5	12/31/87	F-03
I.A.2.2	Training and Qualifications of Operations Personnel	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.2.3	Administration of Training Programs	-	NRR/DHFS/LQB	I	5	12/31/87	
I.A.2.4	NRR Participation in Inspector Training	Colmar	NRR/DHFS/LQB	LI (NOTE 3)	5	12/31/87	NA
I.A.2.5	Plant Drills	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	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)	5	12/31/87	NA
I.A.2.6(2)	Staff Review of NRR 80-117	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.2.6(3)	Revise 10 CFR 55	Colmar	NRR/DHFS/LQB	I.A.2.2	5	12/31/87	NA
I.A.2.6(4)	Operator Workshops	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.2.6(5)	Develop Inspection Procedures for Training Program	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.2.6(6)	Nuclear Power Fundamentals	Colmar	NRR/DHFS/LQB	DROP	5	12/31/87	NA
I.A.2.7	Accreditation of Training Institutions	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	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	5	12/31/86	
I.A.3.2	Operator Licensing Program Changes	Emrit	NRR/DHFS/OLB	NOTE 3(b)	5	12/31/86	NA
I.A.3.3	Requirements for Operator Fitness	Colmar	RES/DRAO/HFSB	NOTE 3(b)	5	12/31/86	NA
I.A.3.4	Licensing of Additional Operations Personnel	Thatcher	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/86	NA
I.A.3.5	Establish Statement of Understanding with INPO and DOE	Thatcher	NRR/DHFS/HFEB	LI (NOTE 3)	5	12/31/86	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)	5	06/30/88	NA
I.A.4.1(2)	Interim Changes in Training Simulators	Thatcher	NRR/DHFS/OLB	NOTE 3(a)	5	06/30/88	
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)	5	06/30/88	

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
I.A.4.2(2)	Upgrade Training Simulator Standards	Colmar	RES/DFO/HFBR	NOTE 3(a)	5	06/30/88	
I.A.4.2(3)	Regulatory Guide on Training Simulators	Colmar	RES/DFO/HFBR	NOTE 3(a)	5	06/30/88	
I.A.4.2(4)	Review Simulators for Conformance to Criteria	Colmar	NRR/DLPQ/LOLB	NOTE 3(a)	5	06/30/88	
I.A.4.3	Feasibility Study of Procurement of NRC Training Simulator	Colmar	RES/DAE/RSRB	LI (NOTE 3)	5	06/30/88	NA
I.A.4.4	Feasibility Study of NRC Engineering Computer	Colmar	RES/DAE/RSRB	LI (NOTE 3)	5	06/30/88	NA
I.B	SUPPORT PERSONNEL						
I.B.1	Management for Operations						
I.B.1.1	Organization and Management Long-Term Improvements	-	-	-			
I.B.1.1(1)	Prepare Draft Criteria	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	3	12/31/86	NA
I.B.1.1(2)	Prepare Commission Paper	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	3	12/31/86	NA
I.B.1.1(3)	Issue Requirements for the Upgrading of Management and Technical Resources	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	3	12/31/86	NA
I.B.1.1(4)	Review Responses to Determine Acceptability	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	3	12/31/86	NA
I.B.1.1(5)	Review Implementation of the Upgrading Activities	Colmar	OIE/DQASIP/ORPB	NOTE 3(b)	3	12/31/86	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	3	12/31/86	NA
I.B.1.1(7)	Issue Regulatory Guides 1.33 and 1.8	Colmar	NRR/DHFS/LQB	I.A.2.6(1), 75	3	12/31/86	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)	3	12/31/86	NA
I.B.1.2(2)	Review Near-Term Operating License Facilities	-	NRR/DHFS/LQB	NOTE 3(b)	3	12/31/86	NA
I.B.1.2(3)	Include Findings in the SER for Each Near-Term Operating License Facility	-	NRR/DL/ORAB	NOTE 3(b)	3	12/13/86	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)	3	12/31/86	NA
I.B.1.3(2)	Use Existing Enforcement Options to Accomplish Safest Shutdown Cooling	Sege	RES	LI (NOTE 3)	3	12/31/86	NA
I.B.1.3(3)	Use Non-Fiscal Approaches to Accomplish Safest Shutdown Cooling	Sege	RES	LI (NOTE 3)	3	12/31/86	NA
I.B.2	Inspection of Operating Reactors						
I.B.2.1	Revise OIE Inspection Program	-	-	-			
I.B.2.1(1)	Verify the Adequacy of Management and Procedural Controls and Staff Discipline	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(2)	Verify that Systems Required to Be Operable Are Properly Aligned	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	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)		11/30/83	NA
I.B.2.1(4)	Observe Surveillance Tests to Determine Whether Test Instruments Are Properly Calibrated	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
I.B.2.1(5)	Verify that Licensees Are Complying with Technical Specifications	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(6)	Observe Routine Maintenance	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	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)		11/30/83	NA
I.B.2.2	Resident Inspector at Operating Reactors	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)		11/30/83	NA
I.B.2.3	Regional Evaluations	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)		11/30/83	NA
I.B.2.4	Overview of Licensee Performance	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)		11/30/83	NA
I.C	OPERATING PROCEDURES						
I.C.1	Short-Term Accident Analysis and Procedures Revision	-	-	-			
I.C.1(1)	Small Break LOCAs	-	NRR	I	3	12/31/86	
I.C.1(2)	Inadequate Core Cooling	-	NRR	I	3	12/31/86	F-04
I.C.1(3)	Transients and Accidents	-	NRR	I	3	12/31/86	F-05
I.C.1(4)	Confirmatory Analyses of Selected Transients	Riggs	NRR/DSI/RSB	NOTE 3(b)	3	12/31/86	NA
I.C.2	Shift and Relief Turnover Procedures	-	NRR	I	3	12/31/86	
I.C.3	Shift Supervisor Responsibilities	-	NRR	I	3	12/31/86	
I.C.4	Control Room Access	-	NRR	I	3	12/31/86	
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	-	NRR/DL	I	3	12/31/86	F-06
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	-	NRR/DL	I	3	12/31/86	F-07
I.C.7	NSSS Vendor Review of Procedures	-	NRR/DHFS/PSRB	I	3	12/31/86	
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	-	NRR/DHFS/PSRB	I	3	12/31/86	
I.C.9	Long-Term Program Plan for Upgrading of Procedures	Riggs	NRR/DHFS/PSRB	NOTE 3(b)	3	12/31/86	NA
I.D	CONTROL ROOM DESIGN						
I.D.1	Control Room Design Reviews	-	NRR/DL	I	5	12/31/89	F-08
I.D.2	Plant Safety Parameter Display Console	-	NRR/DL	I	5	12/31/89	F-09
I.D.3	Safety System Status Monitoring	Thatcher	RES/DE/MEB	MEDIUM	5	12/31/89	
I.D.4	Control Room Design Standard	Thatcher	RES/DRPS/RHFB	NOTE 3(b)	5	12/31/89	NA
I.D.5	Improved Control Room Instrumentation Research	-	-	-			
I.D.5(1)	Operator-Process Communication	Thatcher	RES/DFO/HFBR	NOTE 3(b)	5	12/31/89	NA
I.D.5(2)	Plant Status and Post-Accident Monitoring	Thatcher	RES/DFO/HFBR	NOTE 3(a)	5	12/31/89	
I.D.5(3)	On-Line Reactor Surveillance System	Thatcher	RES/DE/MEB	NOTE 1	5	12/31/89	
I.D.5(4)	Process Monitoring Instrumentation	Thatcher	RES/DFO/ICBR	NOTE 3(b)	5	12/31/89	NA
I.D.5(5)	Disturbance Analysis Systems	Thatcher	RES/DRPS/RHFB	LI (NOTE 5)	5	12/31/89	NA
I.D.6	Technology Transfer Conference	Thatcher	RES/DFO/HFBR	LI (NOTE 3)	5	12/31/89	NA

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
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)	1	6/30/84	NA
I.E.2	Program Office Operational Data Evaluation	Matthews	NRR/DL/ORAB	LI (NOTE 3)	1	6/30/84	NA
I.E.3	Operational Safety Data Analysis	Matthews	RES/DRA/RRBR	LI (NOTE 3)	1	6/30/84	NA
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.5	Nuclear Plant Reliability Data System	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.6	Reporting Requirements	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.7	Foreign Sources	Matthews	IP	LI (NOTE 3)	1	6/30/84	NA
I.E.8	Human Error Rate Analysis	Matthews	RES/DFO/HFBR	LI (NOTE 3)	1	6/30/84	NA
I.F	QUALITY ASSURANCE						
I.F.1	Expand QA List	Pittman	RES/DRA/ARGIB	NOTE 3(b)	2	06/30/89	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	2	06/30/89	NA
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	2	06/30/89	NA
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	2	06/30/89	NA
I.F.2(4)	Establish Criteria for Determining QA Requirements for Specific Classes of Equipment	Pittman	OIE/DQASIP/QUAB	LOW	2	06/30/89	NA
I.F.2(5)	Establish Qualification Requirements for QA and QC Personnel	Pittman	OIE/DQASIP/QUAB	LOW	2	06/30/89	NA
I.F.2(6)	Increase the Size of Licensees' QA Staff	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	2	06/30/89	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	2	06/30/89	NA
I.F.2(8)	Compare NRC QA Requirements with Those of Other Agencies	Pittman	OIE/DQASIP/QUAB	LOW	2	06/30/89	NA
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	2	06/30/89	NA
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation	Pittman	OIE/DQASIP/QUAB	LOW	2	06/30/89	NA
I.F.2(11)	Define Role of QA in Design and Analysis Activities	Pittman	OIE/DQASIP/QUAB	LOW	2	06/30/89	NA
I.G	PREOPERATIONAL AND LOW-POWER TESTING						
I.G.1	Training Requirements	-	NRR/DHFS/PSRB	I	2	06/30/89	
I.G.2	Scope of Test Program	V'Molen	NRR/DHFS/PSRB	NOTE 3(a)	2	06/30/89	NA

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
II.A	SITING						
II.A.1	Siting Policy Reformulation	V'Molen	NRR/DE/SAB	NOTE 3(b)	1	12/31/84	NA
II.A.2	Site Evaluation of Existing Facilities	V'Molen	NRR/DE/SAB	V.A.1	1	12/31/84	NA
II.B	CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW						
II.B.1	Reactor Coolant System Vents		NRR/DL	I	3	12/31/91	F-10
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	-	NRR/DL	I	3	12/31/91	F-11
II.B.3	Post-Accident Sampling	-	NRR/DL	I	3	12/31/91	F-12
II.B.4	Training for Mitigating Core Damage	-	NRR/DL	I	3	12/31/91	F-13
II.B.5	Research on Phenomena Associated with Core Degradation and Fuel Melting	-	-	-			
II.B.5(1)	Behavior of Severely Damaged Fuel	V'Molen	RES/DSR/AEB	LI (NOTE 5)	3	12/31/91	NA
II.B.5(2)	Behavior of Core-Melt	V'Molen	RES/DSR/AEB	LI (NOTE 5)	3	12/31/91	NA
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structure	V'Molen	RES/DSR/AEB	LI (NOTE 5)	3	12/31/91	NA
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	Pittman	NRR/DST/RRAB	NOTE 3(a)	3	12/31/91	
II.B.7	Analysis of Hydrogen Control	Matthews	NRR/DSI/CSB	II.B.8	3	12/31/91	
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	V'Molen	RES/DRAO/RAMR	NOTE 3(a)	3	12/31/91	
II.C	RELIABILITY ENGINEERING AND RISK ASSESSMENT						
II.C.1	Interim Reliability Evaluation Program	Pittman	RES/DRAO/RRB	NOTE 3(b)	2	12/31/88	NA
II.C.2	Continuation of Interim Reliability Evaluation Program	Pittman	NRR/DST/RRAB	NOTE 3(b)	2	12/31/88	NA
II.C.3	Systems Interaction	Pittman	NRR/DST/GIB	A-17	2	12/31/88	NA
II.C.4	Reliability Engineering	Pittman	RES/DRPS/RHFB	NOTE 3(b)	2	12/31/89	NA
II.D	REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES						
II.D.1	Testing Requirements	-	NRR/DL	I	1	06/30/89	F-14
II.D.2	Research on Relief and Safety Valve Test Requirements	Riggs	RES	LOW	1	06/30/89	NA
II.D.3	Relief and Safety Valve Position Indication	-	NRR	I	1	06/30/89	
II.E	SYSTEM DESIGN						
II.E.1	Auxiliary Feedwater System						
II.E.1.1	Auxiliary Feedwater System Evaluation	-	NRR/DL	I	1	12/31/86	F-15
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	-	NRR/DL	I	1	12/31/86	F-16, F-17

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	Riggs	RES/DRA/RRBR	NOTE 3(a)	1	12/31/86	
II.E.2	Emergency Core Cooling System						
II.E.2.1	Reliance on ECCS	Riggs	NRR/DSI/RSB	II.K.3(17)	1	12/31/85	NA
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	Riggs	RES/DAE/RSRB	NOTE 3(b)	1	12/31/85	NA
II.E.2.3	Uncertainties in Performance Predictions	V'Molen	NRR/DSI/RSB	LOW	1	12/31/85	NA
II.E.3	Decay Heat Removal						
II.E.3.1	Reliability of Power Supplies for Natural Circulation	-	NRR/DL	I	1	06/30/91	
II.E.3.2	Systems Reliability	V'Molen	NRR/DST/GIB	A-45	1	06/30/91	NA
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements	V'Molen	NRR/DST/GIB	A-45	1	06/30/91	NA
II.E.3.4	Alternate Concepts Research	Riggs	RES/DAE/FBRB	NOTE 3(b)	1	06/30/91	NA
II.E.3.5	Regulatory Guide	Riggs	NRR/DST/GIB	A-45	1	06/30/91	NA
II.E.4	Containment Design						
II.E.4.1	Dedicated Penetrations	-	NRR/DL	I		06/30/88	F-18
II.E.4.2	Isolation Dependability	-	NRR/DL	I		06/30/88	F-19
II.E.4.3	Integrity Check	Milstead	RES/DRPS/RPSI	NOTE 3(b)		06/30/88	NA
II.E.4.4	Purging	-	-	-			
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging	Milstead	NRR/DSI/CSB	NOTE 3(c)		06/30/88	
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Letter	Milstead	NRR/DSI/CSB	NOTE 3(a)		06/30/88	
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability	Milstead	NRR/DSI/CSB	NOTE 3(a)		06/30/88	
II.E.4.4(4)	Evaluate Purging and Venting During Normal Operation	Milstead	NRR/DSI/CSB	NOTE 3(b)		06/30/88	NA
II.E.4.4(5)	Issue Modified Purging and Venting Requirement	Milstead	NRR/DSI/CSB	NOTE 3(b)		06/30/88	NA
II.E.5	Design Sensitivity of B&W Reactors						
II.E.5.1	Design Evaluation	Thatcher	NRR/DSI/RSB	NOTE 3(a)	1	12/31/84	
II.E.5.2	B&W Reactor Transient Response Task Force	Thatcher	NRR/DL/ORAB	NOTE 3(a)	1	12/31/84	
II.E.6	In Situ Testing of Valves						
II.E.6.1	Test Adequacy Study	Thatcher	RES/DE/EIB	NOTE 3(a)	1	06/30/89	
II.F	INSTRUMENTATION AND CONTROLS						
II.F.1	Additional Accident Monitoring Instrumentation	-	NRR/DL	I	2	06/30/89	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	2	06/30/89	F-26
II.F.3	Instruments for Monitoring Accident Conditions	V'Molen	RES/DFO/ICBR	NOTE 3(a)	2	06/30/89	
II.F.4	Study of Control and Protective Action Design Requirements	Thatcher	NRR/DSI/ICSB	DROP	2	06/30/89	NA
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	Thatcher	RES/DE	LI (NOTE 3)	2	06/30/89	NA

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
II.G	ELECTRICAL POWER						
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	-	NRR	I			
II.H	TMI-2 CLEANUP AND EXAMINATION						
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact	Matthews	NRR/TMIPO	NOTE 3(b)		11/30/83	NA
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	Milstead	RES/DRAA/AEB	HIGH		11/30/83	
II.H.3	Evaluate and Feed Back Information Obtained from TMI	Milstead	NRR/TMIPO	II.H.2		11/30/83	NA
II.H.4	Determine Impact of TMI on Socioeconomic and Real Property Values	Milstead	RES/DHSWM/SEBR	LI (NOTE 3)		11/30/83	NA
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)		11/30/83	NA
II.J.1.2	Modify Existing Vendor Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.2	Construction Inspection Program						
II.J.2.1	Reorient Construction Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.2.3	Assign Resident Inspectors to All Construction Sites	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	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		11/30/83	NA
II.J.3.2	Issue Regulatory Guide	Pittman	NRR/DHFS/LQB	I.B.1.1		11/30/83	NA
II.J.4	Revise Deficiency Reporting Requirements						
II.J.4.1	Revise Deficiency Reporting Requirements	Riani	AEOD/DSP/ROAB	NOTE 2(a)	1	12/31/91	NA

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
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	-
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" Indication 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	-

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
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	-
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

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
II.K.2(17)	Analysis of Potential Voiding in RCS During Anticipated Transients	Emrit	NRR	I		12/31/84	F-33
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
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

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
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
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

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
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
III.A	EMERGENCY PREPAREDNESS AND RADIATION EFFECTS						
III.A.1	Improve Licensee Emergency Preparedness - Short-Term						
III.A.1.1	Upgrade Emergency Preparedness	-	-	-	2	06/30/91	
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	I			
III.A.2.1(2)	Conduct Public Regional Meetings	-	RES	I			
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules	-	RES	I			

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
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
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

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
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
III.D.1.3(2)	Review and Revise SRP	Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems	Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Adsorber	Emrit	NRR/DSI/METB	NOTE 3(b)	1	12/31/88	NA
III.D.1.4	Radwaste System Design Features to Aid in Accident Recovery and Decontamination	Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.2	Public Radiation Protection Improvement						
III.D.2.1	Radiological Monitoring of Effluents	-	-	-			
III.D.2.1(1)	Evaluate the Feasibility and Perform a Value-Impact Analysis of Modifying Effluent-Monitoring Design Criteria	Emrit	NRR/DSI/METB	LOW	2	12/31/85	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	2	12/31/85	NA
III.D.2.1(3)	Revise Regulatory Guides	Emrit	NRR/DSI/METB	LOW	2	12/31/85	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)	2	12/31/85	NA
III.D.2.2(2)	Evaluate Data Collected at Quad Cities	Emrit	NRR/DSI/RAB	III.D.2.5	2	12/31/85	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	2	12/31/85	NA
III.D.2.2(4)	Revise SRP and Regulatory Guides	Emrit	NRR/DSI/RAB	III.D.2.5	2	12/31/85	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)	2	12/31/85	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)	2	12/31/85	NA
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction	Emrit	NRR/DE/EHEB	NOTE 3(b)	2	12/31/85	NA
III.D.2.3(4)	Prepare a Summary Assessment	Emrit	NRR/DE/EHEB	NOTE 3(b)	2	12/31/85	NA
III.D.2.4	Offsite Dose Measurements	-	-	-			
III.D.2.4(1)	Study Feasibility of Environmental Monitors	V'Molen	NRR/DSI/RAB	NOTE 3(b)	2	12/31/85	NA
III.D.2.4(2)	Place 50 TLDs Around Each Site	V'Molen	OIE/DRP/DRPB	LI (NOTE 3)	2	12/31/85	NA

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
III.D.2.5	Offsite Dose Calculation Manual	V'Molen	NRR/DSI/RAB	NOTE 3(b)	2	12/31/85	NA
III.D.2.6	Independent Radiological Measurements	V'Molen	OIE/DRP/ORPB	LI (NOTE 3)	2	12/31/85	NA
III.D.3	Worker Radiation Protection Improvement						
III.D.3.1	Radiation Protection Plans	V'Molen	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	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(2)	Issue a Regulatory Guide	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(3)	Develop Standard Performance Criteria	V'Molen	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	V'Molen	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		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			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	V'Molen	RES/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	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
III.D.3.5(3)	Revise 10 CFR 20	V'Molen	RES/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

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
IV.D	NRC STAFF TRAINING						
IV.D.1	NRC Staff Training	Emrit	ADM/MDTS	LI (NOTE 3)		11/30/83	NA
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

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	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
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	

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
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 Dynamic 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	
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	V'Molen	NRR/DSI/CSB	LOW		11/30/83	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
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)		11/30/83	
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	Emrit	NRR/DST/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	LOW		11/30/83	NA
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	Emrit	NRR/DST/GIB	NOTE 3(a)	1	6/30/85	
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

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
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)	1	12/31/87	
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	06/30/8d	
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		11/30/83	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	V'Molen	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
B-17	Criteria for Safety-Related Operator Actions	Milstead	RES/DRPS/RHFB	MEDIUM	2	12/31/86	
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/D[illegible]/CPB	NOTE 3(b)		6/30/85	NA
B-20	Standard Problem Analysis	-	RES/[illegible]/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		06/30/91	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

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
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	
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	V'Molen	RES/DE/EIB	MEDIUM		11/30/83	
B-56	Diesel Reliability	Milstead	RES/DRPS/RPSI	HIGH		11/30/83	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	6/30/85	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/DRAA/PRAB	MEDIUM		11/30/8	
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-64	Decommissioning of Reactors	Colmar	RES/DE/MEB	NOTE 2		11/30/83	
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

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
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
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	LOW		11/30/83	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	Colmar	NRR/DSI/ICSB	NOTE 4		11/30/83	NA
3.	Set Point Drift in Instrumentation	Emrit	NRR/DSRO/RSIB	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	V'Molen	NRR/DSI/CPB	NOTE 3(b)		11/30/83	NA
7.	Failures Due to Flow-Induced Vibrations	V'Molen	NRR/DSI/RSB	DROP	1	06/30/91	NA

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
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)	1	12/31/85	NA
15.	Radiation Effects on Reactor Vessel Supports	Emrit	NRR/DE/MTEB	HIGH	2	12/31/89	
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	V'Molen	NRR/DSI/RSB	NOTE 3(b)	1	12/31/84	NA
23.	Reactor Coolant Pump Seal Failures	Riggs	RES/DE/EIB	HIGH		11/30/83	
24.	Automatic ECCS Switchover to Recirculation	Milstead	NRR/DSIR/RPSIB	MEDIUM	1	12/31/91	
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	V'Molen	RES/DSIR/EIB	NOTE 3(b)	1	12/31/91	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	V'Molen	NRR/DSI/CPB, RSB	LOW	1	06/30/85	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	1	12/31/91	NA

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
39.	Potential for Unacceptable Interaction Between the CRD System and Non-Essential Control Air System	Pittman	NRR/DSI/ASB	25		11/30/83	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	V'Molen	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	
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	
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	V'Molen	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	MEDIUM	1	06/30/88	
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	V'Molen	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	-	-	-		-	

Action Pl Iss.	Item/ No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
67.2.1		Integrity of Steam Generator Tube Sleeves	Riggs	NRR/DE/MEB	135	3	06/30/91	NA
67.3.1		Steam Generator Overfill	Riggs	NRR/DST/GIB NRR/DSI/RSB	A-47, I.C.1	3	06/30/91	NA
67.3.2		Pressurized Thermal Shock	Riggs	NRR/DST/GIB	A-49	3	06/30/91	NA
67.3.3		Improved Accident Monitoring	Riggs	NRR/DSI/ICSB	NOTE 3(a)	3	06/30/91	A-17
67.3.4		Reactor Vessel Inventory Measurement	Riggs	NRR/DSI/CPB	II.F.2	3	06/30/91	NA
67.4.1		RCP Trip	Riggs	NRR/DSI/RSB	II.K.3(5)	3	06/30/91	G-01
67.4.2		Control Room Design Review	Riggs	NRR/DHFS/HFEB	I.D.1	3	06/30/91	F-08
67.4.3		Emergency Operating Procedures	Riggs	NRC/DHFS/PSRB	I.C.1	3	06/30/91	F-05
67.5.1		Reassessment of Radiological Consequences	Riggs	RES/DRPS/RPSI	LI (NOTE 5)	3	06/30/91	NA
67.5.2		Reevaluation of SGTR Design Basis	Riggs	RES/DRPS/RPSI	LI (NOTE 5)	3	06/30/91	NA
67.5.3		Secondary System Isolation	Riggs	NRR/DSI/RSB	DROP	3	06/30/91	NA
67.6.0		Organizational Responses	Riggs	OIE/DEPER/IRDB	III.A.3	3	06/30/91	NA
67.7.0		Improved Eddy Current Tests	Riggs	RES/DE/EIB	135	3	06/30/91	NA
67.8.0		Denting Criteria	Riggs	NRR/DE/MTEB	135	3	06/30/91	NA
67.9.0		Reactor Coolant System Pressure Control	Riggs	NRR/DSI/GIB NRR/DSI/ASB	A-45, I.C.1 (2,3)	3	06/30/91	NA
67.10.0		Supplemental Tube Inspections	Riggs	NRR/DL/ORAB	LI (NOTE 5)	3	06/30/91	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	B-43
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	LOW	1	06/30/90	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 2	1	12/31/91	
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 B-90,B-91 B-92,B-93
76.		Instrumentation and Control Power Interactions	Pittman	RES/DRA/ARGIB	NOTE 4		11/30/83	
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	Riggs	RES/DRA/ARGIB	NOTE 4		11/30/83	
79.		Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	Colmar	RES/DE/EIB	MEDIUM	1	12/31/84	
80.		Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments	V'Molen	NRR/DSI/RSB, ASB, CPB	LOW	1	06/30/91	NA

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
81.	Impact of Locked Doors and Barriers on Plant and Personnel Safety	Riggs	RES/DRA/ARGIB	DROP	2	06/30/90	NA
82.	Beyond Design Basis Accidents in Spent Fuel Pools	V'Molen	RES/DRPS/RPSI	NOTE 3(b)	1	06/30/89	NA
83.	Control Room Habitability	Emrit	RES/DRAA/SAIB	NOTE 1	1	12/31/86	
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)	1	12/31/91	
88.	Earthquakes and Emergency Planning	Riggs	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
89.	Stiff Pipe Clamps	Riggs	RES	NOTE 4		(later)	
90.	Technical Specifications for Anticipatory Trips	V'Molen	NRR/DSI/RSB, ICSB	LOW		12/31/84	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	V'Molen	NRR/DSI/RSB, CPB	LOW		12/31/84	NA
93.	Steam Binding of Auxiliary Feedwater Pumps	Pittman	RES/DRPS/RPSI	NOTE 3(a)		06/30/88	
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	[illegible]	RES/DRA/ARGIB	105		06/30/90	NA
97.	PWR Reactor Cavity Uncontrolled Exposures	V'Molen	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	
100.	Once-Through Steam Generator Level	Jackson	RES/DSIR/EIB	DROP	1	12/31/91	NA
101.	BWR Water Level Redundancy	V'Molen	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	HIGH	1	06/30/91	
106.	Piping and Use of Highly Combustible Gases in Vital Areas	Milstead	RES/DRPS	MEDIUM		12/31/87	
107.	Main Transformer Failures	Milstead	RES/DRA/ARGIB	LOW	1	06/30/91	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	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
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/DE/EIB	HIGH		12/31/87	
114.	Seismic-Induced Relay Chatter	Riggs	NRR/DSRO/SPEB	A-46	1	06/30/91	NA

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
115.	Enhancement of the Reliability of Westinghouse Solid State Protection System	Milstead	RES/DRPS/RPSI	NOTE 3(b)		06/30/89	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	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
119.	Piping Review Committee Recommendations	-	-	-			
119.1	Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads	Riggs	NRR/DE	RI (NOTE 3)	1	06/30/91	NA
119.2	Piping Damping Values	Riggs	NRR/DE	RI (NOTE 3)	1	06/30/91	NA
119.3	Decoupling the OBE from the SSE	Riggs	NRR/DE	RI (NOTE 5)	1	06/30/91	NA
119.4	BWR Piping Materials	Riggs	NRR/DE	RI (NOTE 5)	1	06/30/91	NA
119.5	Leak Detection Requirements	Riggs	NRR/DE	RI (NOTE 5)	1	06/30/91	NA
120.	On-Line Testability of Protection Systems	Milstead	RES/DRA/ARGIB	MEDIUM		06/30/91	
121.	Hydrogen Control for Large, Dry PWR Containments	Emrit	RES/DRA/RDB	HIGH		12/31/85	
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	V'Molen	NRR/DSRO/RSIB	124	3	06/30/91	NA
122.1.b	Recovery of Auxiliary Feedwater	V'Molen	NRR/DSRO/RSIB	124	3	06/30/91	NA
122.1.c.	Interruption of Auxiliary Feedwater Flow	V'Molen	NRR/DSRO/RSIB	124	3	06/30/91	NA
122.2	Initiating Feed-and-Bleed	V'Molen	NRR/DEST/SRXB	NOTE 3(b)	3	06/30/91	NA
122.3	Physical Security System Constraints	V'Molen	NRR/DSRO/SPEB	LOW	3	06/30/91	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		12/31/91	NA
124.	Auxiliary Feedwater System Reliability	Emrit	NRR/DEST/SRXB	NOTE 3(a)	3	06/30/91	
125.	Davis-Besse Loss of All Feedwater Event of June 9, 1985: Long-Term Actions	-	-	-			
125.I.1	Availability of the Shift Technical Advisor	V'Molen	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.I.2	PORV Reliability	-	-	-	6	12/31/89	
125.I.2.a	Need for a Test Program to Establish Reliability of the PORV	V'Molen	NRR/DSRO/SPEB	70	6	12/31/89	NA
125.I.2.b	Need for PORV Surveillance Tests to Confirm Operational Readiness	V'Molen	NRR/DSRO/SPEB	70	6	12/31/89	NA
125.I.2.c	Need for Additional Protection Against PORV Failure	V'Molen	NRR/DSRO/SPEB	DROP	6	12/31/89	NA
125.I.2.d	Capability of the PORV to Support Feed-and-Bleed	V'Molen	NRR/DSRO/SPEB	A-45	6	12/31/89	NA
125.I.3	SPDS Availability	Milstead	RES/DRA/ARGIB	NOTE 3(b)	6	12/31/89	NA
125.I.4	Plant-Specific Simulator	Riggs	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.I.5	Safety Systems Tested in All Conditions Required by DBA	Riggs	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.I.6	Valve Torque Limit and Bypass Switch Settings	V'Molen	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.I.7	Operator Training Adequacy	-	-	-			
125.I.7.a	Recover Failed Equipment	Pittman	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.I.7.b	Realistic Hands-On Training	V'Molen	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.I.8	Procedures and Staffing for Reporting to NRC Emergency Response Center	V'Molen	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.1	Need for Additional Actions on AFW Systems	-	-	-			

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
125.II.1.a	Two-Train AFW Unavailability	V'Molen	NRR/DSRO/SPEB	DROP	6	12/31/89	NA
125.II.1.b	Review Existing AFW Systems for Single Failure	V'Molen	NRR/DSRO/SPEB	124	6	12/31/89	NA
125.II.1.c	NUREG-0737 Reliability Improvements	V'Molen	NRR/DSRO/SPEB	DROP	6	12/31/89	NA
125.II.1.d	AFW/Steam and Feedwater Rupture Control System/ICS Interactions in B&W Plants	V'Molen	NRR/DSRO/SPEB	DROP	6	12/31/89	NA
125.II.2	Adequacy of Existing Maintenance Requirements for Safety-Related Systems	Riggs	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.3	Review Steam/Feedline Break Mitigation Systems for Single Failure	V'Molen	NRR/DSRO/SPEB	DROP	6	12/31/89	NA
125.II.4	Thermal Stress of OTSG Components	Riggs	NRR/DSRO/SPEB	DROP	6	12/31/89	NA
125.II.5	Thermal-Hydraulic Effects of Loss and Restoration of Feedwater on Primary System Components	Riggs	RES/DRA/ARGIB	DROP	6	12/31/89	
125.II.6	Reexamine PRA Estimates of Core Damage Risk from Loss of All Feedwater	V'Molen	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.7	Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break	V'Molen	RES/DRPS/RPSI	NOTE 3(b)	6	12/31/89	NA
125.II.8	Reassess Criteria for Feed-and-Bleed Initiation	V'Molen	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.9	Enhanced Feed-and-Bleed Capability	V'Molen	NRR/DSRO/SPEB	DROP	6	12/31/89	NA
125.II.10	Hierarchy of Impromptu Operator Actions	Riggs	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.11	Recovery of Main Feedwater as Alternative to Auxiliary Feedwater	Riggs	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.12	Adequacy of Training Regarding PORV Operation	Riggs	RES/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.13	Operator Job Aids	Pittman	NRR/DRA/ARGIB	DROP	6	12/31/89	NA
125.II.14	Remote Operation of Equipment Which Must Now Be Operated Locally	V'Molen	NRR/DSRO/SPEB	LOW	6	12/31/89	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)	1	12/31/91	
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)	1	12/31/91	
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 Pumps Inside Containment	Riggs	RES/DRA/ARGIB	NOTE 4		(later)	
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)	2	12/31/91	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	LOW		12/31/91	NA

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
139.	Thinning of Carbon Steel Piping in LWRs	Riggs	RES/DRA/ARGIB	RI (NOTE 3)		12/31/88	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	MEDIUM	1	12/31/91	
143.	Availability of Chilled Water Systems and Room Cooling	Milstead	RES/DRA/ARGIB	HIGH		06/30/91	
144.	Scram Without a Turbine/Generator Trip	Riggs	RES/DRA/ARGIB	NOTE 4		(later)	
145.	Improve Surveillance and Startup Testing Programs	Riggs	RES/DRA/ARGIB	NOTE 4		(later)	
146.	Support Flexibility of Equipment and Components	Riggs	RES/DRA/ARGIB	NOTE 4		(later)	
147.	Fire-Induced Alternate Shutdown Control Room Panel Interactions	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
148.	Smoke Control and Manual Fire-Fighting Effectiveness	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
149.	Adequacy of Fire Barriers	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
150.	Overpressurization of Containment Penetrations	Milstead	RES/DSIR/SAIB	DROP		12/31/91	NA
151.	Reliability of Anticipated Transient Without SCRAM Recirculation Pump Trip in BWRs	Milstead	RES/DSIR/RPSIB	MEDIUM	1	12/31/91	
152.	Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
153.	Loss of Essential Service Water in LWRs	Riggs	RES/DRA/ARGIB	HIGH		06/30/91	
154.	Adequacy of Emergency and Essential Lighting	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
155.	Generic Concerns Arising from TMI-2 Cleanup	-	-	-		-	-
155.1	More Realistic Source Term Assumptions	Emrit	RES/DSIR	NOTE 4		(later)	
155.2	Establish Licensing Requirements for Non-Operating Facilities	Emrit	RES/DSIR	NOTE 4		(later)	
155.3	Improve Design Requirements for Nuclear Facilities	Emrit	RES/DSIR	NOTE 4		(later)	
155.4	Improve Criticality Calculations	Emrit	RES/DSIR	NOTE 4		(later)	
155.5	More Realistic Severe Reactor Accident Scenario	Emrit	RES/DSIR	NOTE 4		(later)	
155.6	Improve Decontamination Regulations	Emrit	RES/DSIR	NOTE 4		(later)	
155.7	Improve Decommissioning Regulations	Emrit	RES/DSIR	NOTE 4		(later)	
156.	Systematic Evaluation Program	-	-	-		-	-
156.1.1	Settlement of Foundations and Buried Equipment	Chang	RES/DSIR	NOTE 4		(later)	
156.1.2	Dam Integrity and Site Flooding	Chen	RES/DSIR	NOTE 4		(later)	
156.1.3	Site Hydrology and Ability to Withstand Floods	Chen	RES/DSIR	NOTE 4		(later)	
156.1.4	Industrial Hazards	Ferrell	RES/DSIR	NOTE 4		(later)	
156.1.5	Tornado Missiles	Chen	RES/DSIR	NOTE 4		(later)	
156.1.6	Turbine Missiles	Emrit	RES/DSIR/RPSIB	DROP		12/31/91	NA
156.2.1	Severe Weather Effects on Structures	Chen	RES/DSIR	NOTE 4		(later)	
156.2.2	Design Codes, Criteria, and Load Combinations	Kirkwood	RES/DSIR	NOTE 4		(later)	
156.2.3	Containment Design and Inspection	Shaukat	RES/DSIR	NOTE 4		(later)	
156.2.4	Seismic Design of Structures, Systems, and Components	Chen	RES/DSIR	NOTE 4		(later)	
156.3.1.1	Shutdown Systems	Woods	RES/DSIR	NOTE 4		(later)	
156.3.1.2	Electrical Instrumentation and Control	Woods	RES/DSIR	NOTE 4		(later)	
156.3.2	Service and Cooling Water Systems	Su	RES/DSIR	NOTE 4		(later)	
156.3.3	Ventilation Systems	Burdick	RES/DSIR	NOTE 4		(later)	
156.3.4	Isolation of High and Low Pressure Systems	Burdick	RES/DSIR/RPSIB	DROP		12/31/91	NA
156.3.5	Automatic ECCS Switchover	Milstead	RES/DSIR/SAIB	24		12/31/91	NA
156.3.6.1	Emergency AC Power	Serkiz	RES/DSIR	NOTE 4		(later)	
156.3.6.2	Emergency DC Power	Rourk	RES/DSIR	NOTE 4		(later)	
156.3.8	Shared Systems	Emrit	RES/DSIR	NOTE 4		(later)	

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
156.4.1	RPS and ESFS Isolation	Emrit	RES/DSIR/RPSIB	142		12/31/91	NA
156.4.2	Testing of the RPS and ESFS	Chang	RES/DSIR	NOTE 4		(later)	
156.6.1	Pipe Break Effects on Systems and Components	Page	RES/DSIR	NOTE 4		(later)	
157.	Containment Performance	Shaperow	RES/DSIR/SAIB	NOTE 4		(later)	
158.	Performance of Power-Operated Valves Under Design Basis Conditions	Shaperow	RES	NOTE 4		(later)	
159.	Qualification of Safety-Related Pumps While Running on Minimum Flow	Shaperow	RES	NOTE 4		(later)	
160.	Spurious Actions of Instrumentation Upon Restoration of Power	Shaperow	RES	NOTE 4		(later)	
161.	Use of Non-Safety-Related Power Supplies in Safety-Related Circuits	Shaperow	RES	NOTE 4		(later)	
162.	Inadequate Technical Specifications for Shared Systems at Multiplant Sites When Once Unit is Shut Down	Shaperow	RES	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
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)	3	06/30/91	NA
HF4.2	Procedures Generation Package Effectiveness Evaluation	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	3	06/30/91	NA
HF4.3	Criteria for Safety-Related Operator Actions	Pittman	NRR/DHFT/HFIB	B-17	3	06/30/91	NA
HF4.4	Guidelines for Upgrading Other Procedures	Pittman	RES/DRPS/RHFB	HIGH	3	06/30/91	NA
HF4.5	Application of Automation and Artificial Intelligence	Pittman	NRR/DHFT/HFIB	HF5.2	3	06/30/91	NA

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

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority/ Status	Latest Revision	Latest Issuance Date	MPA No.
CH2	DESIGN						
CH2.1	Reactivity Accidents	-	-				
CH2.1A	Reactivity Transients	Emrit	RES/DSR/RPSB	LI (NOTE 5)		06/30/89	NA
CH2.2	Accidents at Low Power and at Zero Power	Emrit	RES/DRA/ARGIB	CH1.4		06/30/89	NA
CH2.3	Miltiple-Unit Protection	-	-				
CH2.3A	Control Room Habitability	Emrit	RES/DRA/ARGIB	83		06/30/89	NA
CH2.3B	Contamination Outside Control Room	Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.3C	Smoke Control	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH2.3D	Shared Shutdown Systems	Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.4	Fire Protection	-	-				
CH2.4A	Firefighting With Radiation Present	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH3	CONTAINMENT						
CH3.1	Containment Performance During Severe Accidents	-	-				
CH3.1A	Containment Performance	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH3.2	Filtered Venting	-	-				
CH3.2A	Filtered Venting	Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4	EMERGENCY PLANNING						
CH4.1	Size of the Emergency Planning Zones	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH4.2	Medical Services	Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH4.3	Ingestion Pathway Measures	-	-				
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 - Issue that is not a Generic Safety Issue but should be Assigned Resources for Completion

DROP	- Issue Dropped as a Generic Issue
EI	- Environmental Issue
GSI	- Generic Safety Issue
HIGH	- High Safety Priority
I	- TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737[98]
LI	- Licensing Issue
LOW	- Low Safety Priority
MEDIUM	- Medium Safety Priority
RI	- Regulatory Impact Issue
USI	- Unresolved Safety Issue

TABLE III (Continued)

ACTION ITEM/ISSUE GROUP	I	COVERED IN OTHER ISSUES	RESOLVED STAGES NOTE 1	NOTE 2	NOTE 3	USI	HIGH	MEDIUM	LOW	DROP	NOTE 4	NOTE 5	TOTAL
1. TMI ACTION PLAN ITEMS (369)													
(i) GSI	87	46	1	0	129	0	1	1	12	9	-	-	286
(ii) LI	-	0	-	-	74	-	-	-	-	-	-	9	83
2. TASK ACTION PLAN ITEMS (142)													
(i) USI	-	-	-	-	27	0	-	-	-	-	-	-	27
(ii) GSI	-	20	0	1	31	-	1	3	3	11	0	-	70
(iii) RI	-	-	-	-	6	-	-	-	-	-	-	1	7
(iv) LI	-	-	-	-	12	-	-	-	-	-	-	12	24
(v) EI	-	-	-	-	12	-	-	-	-	-	-	2	14
3. NEW GENERIC ISSUES (240)													
(i) GSI	-	52	1	1	46	0	7	6	10	53	48	-	224
(ii) RI	-	-	-	-	4	-	-	-	-	-	-	5	9
(iii) LI	-	-	-	-	3	-	-	-	-	-	-	4	7
4. HUMAN FACTORS ISSUES (27)													
(i) GSI	-	8	0	0	5	0	3	0	0	0	0	-	16
(ii) LI	-	-	-	-	3	-	-	-	-	-	-	8	11
5. CHERNOBYL ISSUES (32)													
(i) LI	-	2	-	-	7	-	-	-	-	-	-	23	32
TOTAL:	87	128	2	2	359	0	12	10	25	73	48	64	810

TASK II.B: CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW

The objective of this task was to enhance public safety and reduce individual and societal risk by developing and implementing a phased program to include, in safety reviews, consideration of core degradation and melting beyond the design basis.

ITEM II.B.1: REACTOR COOLANT SYSTEM VENTS

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-10 was established by DL for implementation purposes.

ITEM II.B.2: PLANT SHIELDING TO PROVIDE ACCESS TO VITAL AREAS AND PROTECT SAFETY EQUIPMENT FOR POST-ACCIDENT OPERATION

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-11 was established by DL for implementation purposes.

ITEM II.B.3: POST-ACCIDENT SAMPLING

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-12 was established by DL for implementation purposes.

ITEM II.B.4: TRAINING FOR MITIGATING CORE DAMAGE

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-13 was established by DL for implementation purposes.

ITEM II.B.5: RESEARCH ON PHENOMENA ASSOCIATED WITH CORE DEGRADATION AND FUEL MELTING

ITEM II.B.5(1): BEHAVIOR OF SEVERELY DAMAGED FUEL

Items II.B.5(1) and II.B.5(2) were combined and evaluated together.

DESCRIPTION

Historical Background

For a number of key severe accident sequences, there are critical phenomenological unknowns or uncertainties that impact containment integrity assessments and judgments regarding the desirability of certain mitigating features. The phenomena fall into three broad categories: (1) the behavior of severely damaged fuel, including oxidation and hydrogen generation; (2) the behavior of the core-melt in its interaction with water, concrete, and core-retention materials;

and (3) the effect of potential hydrogen burning and/or explosions on containment integrity. Steam explosions were also to be considered in this category. Previous work in these several areas received less attention since they related to accidents beyond the design basis. At the time this TMI Action Plan item[48] was raised, RES was conducting major programs to support the basis for rulemaking and to confirm certain licensing decisions. Complementary efforts conducted within NRR were to address specific licensing issues related to the subject research.

(1) Behavior of Severely Damaged Fuel

(a) In-pile Studies: Fuel behavior research was to include in-pile testing to help evaluate the effects of conditions leading to severe fuel damage. Such tests were being performed in the INEL Power Burst Facility (PBF) in FY 1983 and later in the ACRR at SNL and in the NRU reactor at Chalk River National Lab, Canada. In the PBF, RES was to perform a series of in-reactor fuel experiments to determine the effect of heating and cooling rates on damage to the bundle, rod fragmentation, distortion, and debris formation. Fission product release and hydrogen generation were also to be measured during the testing. Separate effects studies were to be conducted on rubble beds in the ACRR at SNL.

(b) Hydrogen Studies: The objective of this work was to increase the understanding of the formation of hydrogen in a reactor from metal-water reactions, radiolytic decomposition of coolant, and corrosion of metals, and to determine its consequences in terms of pressure-time histories and hydrogen deflagration or detonation. This work was also to include: (1) the preparation of a compendium of information related to hydrogen as it affects reactor safety; (2) analysis of radiolysis under accident conditions; (3) a review of hydrogen sampling and analysis methods; (4) a study of the effects of hydrogen embrittlement on reactor vessel materials; and (5) a review of means of handling accident-generated hydrogen, with recommendations on improving existing methods. Results of these studies were considered in the resolution of Issue A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment," and were not considered further in this issue.

(c) Studies of Postaccident Coolant Chemistry: The RES objective in this area was the development of a relationship between fission product release and fuel failure, and the improvement of postaccident sampling and analysis techniques. This was to be accomplished by the investigation of fission product release in a variety of fuel failure experiments.

(d) Modeling of Severe Fuel Damage: The effort in this area was the development of models for fuel rods operating beyond 2200°F that suffer a loss in geometry in order to compute extensive damage phenomena (such as eutectic liquid formation, fuel slumping, oxidation, and hydrogen generation, fission product release and interaction with the coolant, rubble-bed particle size, extent of fuel and clad melting, and flow blockage).

(2) Behavior of Core-Melt

The RES fuel melt research program was to develop a base and verified methodology for assessing the consequences and mitigation of fuel melt accidents. The program addressed the range of severe reactor accident phenomena from the time when extensive fuel damage and major core geometry changes have occurred until the containment has failed and/or the molten core materials have attained a semi-permanent configuration and further movement is terminated. Studies of improvements in containment design to reduce the risk of core-melt accidents were also included.

The program was composed of integrated tasks that included scoping, phenomenological and separate effects tests, and demonstration experiments that provided results for the development and verification of analytical models and codes. These codes and supporting data were then used for the analysis of thermal, mechanical, and radiological consequences of accidents and for decisions related to requirements of design features for mitigation and performance confirmation. The technical scope of the program included work in the following areas: fuel debris behavior; fuel interactions with structure and soil; radiological source term; fuel-coolant interactions; systems analysis codes; and mitigation features.

Safety Significance

The results of the research programs described above were expected to find broad application in areas such as PRA, accident analysis, siting, evacuation planning, emergency procedures, code development, etc. Thus, these programs would have considerable value just as licensing improvement efforts. However, the programs had sufficiently well-defined scopes to permit some estimates of direct safety significance. These programs were directed at a better understanding of severely damaged and molten cores. Once a core is in this state, any safety significance has to be in the area of minimizing radioactive releases and consequent dose to the public.

Possible Solutions

It was assumed that means would be devised to reduce the probability of containment failure and release of activity to the environment. Completely different approaches could be suggested after the results of the research programs were known.

The "classical" engineering approaches to handling degraded or melted cores are filtered vents to prevent containment overpressure and core retention devices (core catchers) to prevent containment basemat melt-through. These approaches were used for cost estimates, but the other priority parameters were not specific to these approaches.

PRIORITY DETERMINATION

Studies by PNL[64] considered only containment basemat melt-through. The approach presented here was expanded to include other aspects. The effect on a PWR with a dry containment was considered, based partly on the availability of information. It was not expected that the results for other containments or for BWRs would be greatly different, at least in the context of the uncertainty of such an analysis.

Frequency Estimate

Essentially, all core-melts are assumed to result in containment failure in WASH-1400.[16] To estimate the effect of being able to deal with a severely damaged core, this assumption was relaxed. The modes of containment failure for PWRs were defined as follows:

- α - Containment rupture due to a reactor vessel steam explosion.
- β - Containment failure due to inadequate isolation of openings and penetrations.
- γ - Containment failure due to hydrogen burning.
- δ - Containment failure due to overpressure.
- ε - Containment vessel melt-through.

Assuming that the research programs were successful in leading to engineering solutions, reductions in the frequency of the various failure modes were estimated as follows:

α	- 10%	(Little can be done about steam explosions)
β	- 0%	(This does not affect isolation failure)
γ,δ	- 90%	(Venting containment should be quite effective if methods are available for sizing the vent and determining what filtration is needed)
ε	- 90%	(Should be achievable if a core catcher can be designed)

Consequence Estimate

The consequences were straightforward in the sense that the consequences of each release category have been studied. However, the reduction in consequences was more difficult to assess since the release from a molten core in a tight containment is still not zero. Instead, it depends on the containment design leak rate, the efficiency of filtration of a containment relief vent, etc. To allow for this, it was assumed instead that the prevented releases corresponding to the α, γ, δ, and ε failure modes release activity corresponding to a PWR-9 release. The results of this calculation are summarized in Table II.B-1. For a new (forward-fit) plant (which was the most likely candidate for implementation), the public risk reduction was estimated to be 1,600 man-rem.

Table II.B-1

Release Category	Frequency* $(RY)^{-1}$	% Reduction** in Frequency	ΔF $(RY)^{-1}$	R (man-rem)	ΔFR
PWR-1	5.3×10^{-8}	10%	5.3×10^{-9}	4.9×10^{6}	2.6×10^{-2}
PWR-2	6.7×10^{-6}	90%	6.0×10^{-6}	4.8×10^{6}	2.9×10^{1}
PWR-3	2.6×10^{-6}	81%	2.1×10^{-6}	5.4×10^{6}	1.1×10^{1}
PWR-4	2.1×10^{-11}	--	--	2.7×10^{6}	--
PWR-5	4.9×10^{-8}	--	--	1.0×10^{6}	--
PWR-6	6.3×10^{-7}	90%	5.7×10^{-7}	1.4×10^{5}	8.0×10^{-2}
PWR-7	3.4×10^{-5}	90%	3.1×10^{-5}	2.3×10^{3}	7.1×10^{-2}
PWR-8	8.0×10^{-7}	--	--	7.5×10^{4}	--
PWR-9	4.0×10^{-4}	--	-3.9×10^{-5}	1.2×10^{2}	-4.7×10^{-3}
			TOTAL:		4.0×10^{1}

*Because the specific containment failure mode was of interest here, the frequencies above were "unsmoothed." This is in contrast to the calculations in WASH-1400[16] which assumed a 10% contribution in frequency from adjacent release categories.

**Release Category PWR-1 is made up entirely of α failures and thus was assigned a 10% reduction in frequency. Categories PWR-2, PWR-6, and PWR-7 are made up of γ, δ, and ε failures and were thus assigned 90%. Category PWR-3 contains both α and δ failures which results in a net assignment of 81%.

Cost Estimate

Industry Cost: PNL estimated[64] the cost of a core retention device at $1.4M for a forward-fit. SNL estimated[312] the cost of a filtered containment vent to be on the order of a few million dollars. Thus, the industry cost was projected to be $10M/reactor.

NRC Cost: PNL estimated[64] total NRC costs at $2.3M, assuming implementation at 134 reactors. In reality, implementation might take place at a far smaller number of plants due to considerations of containment type, backfit vs. forward fit, etc. However, even if only 10 plants were affected, the NRC cost would be insignificant compared to licensee costs. Therefore, NRC costs were neglected.

Value/Impact Assessment

Based on a potential risk reduction of 1,600 man-rem/reactor and a cost of $10M/reactor, the value/impact score was given by:

$$S = \frac{1{,}600 \text{ man-rem/reactor}}{\$10\text{M/reactor}}$$

$$= 160 \text{ man-rem/\$M}$$

CONCLUSION

Based on the factors considered above, this issue was given a high priority ranking. However, after further evaluation by the staff, the issue was determined to be clearly within the realm of severe accident research and was reclassified as a Licensing Issue.[1102] The issue was pursued[1381] as part of SARP Issue L2, "In-Vessel Core Melt Progression and Hydrogen Generation," documented in NUREG-1365.[1382]

ITEM II.B.5(2): BEHAVIOR OF CORE-MELT

This item was evaluated in Item II.B.5(1) above and determined to be a high priority. However, after further evaluation by the staff, the issue was determined to be clearly within the realm of severe accident research and was reclassified as a Licensing Issue.[1102] The issue was pursued[1381] as part of SARP Issue L2, "In-Vessel Core Melt Progression and Hydrogen Generation," documented in NUREG-1365.[1382]

ITEM II.B.5(3): EFFECT OF HYDROGEN BURNING AND EXPLOSIONS ON CONTAINMENT STRUCTURE

DESCRIPTION

Historical Background

TMI Action Plan Item II.B.5 called for research into the phenomena associated with severe core damage and core melting.[48] Item II.B.5(3) addressed the effect of hydrogen burns and/or explosions on containment integrity.

Safety Significance

Whereas Items II.B.5(1) and II.B.5(2) dealt with (among other things) the generation of hydrogen via radiolysis, metal-water interaction, interaction of a molten core with concrete, etc., Item II.B.5(3) was concerned with effects on the containment of the burning and/or detonation of this hydrogen. If the containment retains its integrity, even a severe accident resulting in a damaged or molten core produces relatively low offsite consequences. Item II.B.5(3) also included the effect of steam explosions. Again, the emphasis here was not in preventing the explosion but, instead, in maintaining containment integrity.

Possible Solution

Most of the work on Item II.B.5(3) was couched in terms of a stronger containment.

PRIORITY DETERMINATION

Item II.B.5(3) was, to a large extent, similar to Issue A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment." Issue A-48 was somewhat more general in that it included the effects of a hydrogen burn or detonation on containment penetrations and on safety systems located within the containment, not just the structural response of the containment. In addition, Issue A-48 included measures for control of the hydrogen burn and thus had preventive as well as mitigative aspects. However, even though Issue A-48 was expected to use the results of Item II.B.5(3), Item II.B.5(3) was not integrated into Issue A-48 because: (1) the scope of Issue A-48 was still under discussion; and (2) Item B.5(3) included steam explosions as well as hydrogen burns.

Frequency/Consequence Estimate

In WASH-1400,[16] the PWR sequences refer to steam explosion-induced containment failures as "α" failures. Containment failures induced by a hydrogen burn are called "γ" failures. Sequences including these two failure modes can be found in Release Categories PWR-1, PWR-2, and PWR-3. It was assumed that the possible solution would result in a 90% reduction in the probabilities of the sequences involving these two failure modes. The results were tabulated as follows:

Release Category	α Frequency(F) (per RY)	γ Frequency(F) (per RY)	Consequences(R) (man-rem)	0.9FR (man-rem/RY)
PWR-1	5.3×10^{-8}	--	4.9×10^{6}	0.23
PWR-2	--	7.0×10^{-7}	4.8×10^{6}	3.0
PWR-3	3.4×10^{-7}	--	5.4×10^{6}	1.7
PWR-7	-3.9×10^{-7}	-7.0×10^{-7}	2.3×10^{3}	-0.002
			TOTAL:	4.9

The PWR-7 category has a negative contribution because a molten core still gives some release, even if containment failure is prevented. Thus, it was assumed that the events which would have been α or γ failures instead lead to PWR-7 releases.

Over a 40-year plant lifetime, the risk reduction above corresponds to about 200 man-rem/reactor. This was calculated using WASH-1400[16] data for a PWR with a large, dry containment. BWR pressure-suppression containments and PWR ice-condenser containments have a much smaller free volume and thus are more susceptible to α and γ failures. Therefore, the risk for these plants could well be considerably higher.

Cost Estimate

Industry Cost: Without the results of research at the time of this evaluation, it was difficult to assess costs. A stronger containment could cost $15M, based on doubling the 3½ foot wall thickness of a (150 ft x 200 ft) structure. (Such structures cost roughly $1,000/cubic yard of concrete.)

NRC Cost: NRC costs were considered to be negligible.

Value/Impact Assessment

Based on an estimated risk reduction of 200 man-rem/reactor and a cost of $15M/reactor, the value/impact score was given by:

$$S = \frac{200 \text{ man-rem/reactor}}{\$15\text{M/reactor}}$$

$$= 13 \text{ man-rem/\$M}$$

CONCLUSION

The public risk estimate for this issue was significant even for dry containments. Because of the difficulty in determining a cost-effective solution, the issue was given a medium priority ranking. However, after further evaluation by the staff, the issue was determined to be clearly within the realm of severe accident research and was reclassified as a Licensing Issue.[1102] The issue was pursued[1381] as part of SARP Issue 13, "Hydrogen Transport and Combustion," documented in NUREG-1365.[1382]

ITEM II.B.6: RISK REDUCTION FOR OPERATING REACTORS AT SITES WITH HIGH POPULATION DENSITIES

DESCRIPTION

Historical Background

This TMI Action Plan item[48] involved the review of operating reactors in areas of high population density to determine what additional measures and/or design changes could be implemented that would further reduce the probability of a severe reactor accident, and would reduce the consequences of such an accident by reducing the amount of radioactive releases and/or by delaying any radioactive releases, and thereby provide additional time for evacuation near the sites.

Risk studies were completed in 1981 for the Zion and Limerick sites and in 1982 for Indian Point. Although risk assessments of other sites were conducted by other NRC programs e.g., National Reliability Evaluation Program (NREP), no further risk studies were envisioned as part of this issue. Further efforts directed towards this issue were review of the analyses and the possible implementation of site-specific fixes to reduce the risk at these sites. Special hearings were scheduled in FY 1982 to review possible design changes for Indian Point and follow-up work in connection with the accepted fixes was anticipated following these hearings.

Safety Significance

Concern existed over the potential for above-average societal risk due to accidents at reactor sites located near regions of high population densities.

Possible Solutions

As mentioned above, hearings were scheduled on possible fixes at the Indian Point site to reduce risk. The actual fixes that resulted from these hearings were unknown at the time of this evaluation. Nevertheless, it was assumed that fixes would be made to reduce the likelihood of the most dominant accident sequences contributing to the frequency of core-melt accidents.

PRIORITY DETERMINATION

Assumptions

Based on a review of similar RSSMAP and IREP analyses, it was assumed that two sequences contributed to a large portion (50%) of the likelihood of a core-melt. It was further assumed that it was possible to reduce the frequency of each sequence by a factor of 10.

Frequency/Consequence Estimate

Based on age and other related factors, it was believed that reactors in this category had an increased frequency of core-melt over the baseline plant (Oconee) by a factor of 5.5 and an increased exposure increase over the mean population density (340 persons per square mile) and release fractions by a factor of 3. This resulted in a revised baseline of the following:

$$\text{Core-Melt Frequency} = (5.5)\ (8.2 \times 10^{-5}/\text{RY}) = 4.5 \times 10^{-4}/\text{RY}$$

$$\text{Exposure Increase} = (3)\ (2.5 \times 10^{6}\ \text{man-rem}) = (7.5 \times 10^{6})\ \text{man-rem}$$

Assuming that the dominant sequences (50% of the frequency) could be reduced by a factor of 10, the revised core-melt frequency was $(0.55)(4.5 \times 10^{-4})/\text{RY} = 2.5 \times 10^{-4}/\text{RY}$.

The baseline public risk was $(4.5 \times 10^{-4}/\text{RY})(7.5 \times 10^{6}$ man-rem) or 3,380 man-rem/RY. The revised public risk was $(2.5 \times 10^{-4}/\text{RY})(7.5 \times 10^{6}$ man-rem) or 1,880 man-rem/RY. The resulting change in public risk was then 1,500 man-rem/RY resulting from the reduction in core-melt frequency of $2 \times 10^{-4}/\text{RY}$. Over the estimated 27 years of remaining plant life, this would result in a total risk reduction of 40,500 man-rem/reactor.

Cost Estimate

Industry Cost: Licensee costs were estimated to be $4M/reactor to implement the changes required to reduce the two dominant sequences.

NRC Cost: NRC costs were estimated to be $22,000.

Total Cost: Total implementation costs were $4.02M/reactor.

Value/Impact Assessment

Based on an estimated public risk reduction of 40,500 man-rem/reactor and a cost of $4.02M/reactor, the value/impact score was given by:

$$S = \frac{40{,}500\ \text{man-rem/reactor}}{\$4.02\text{M/reactor}}$$

$$= 10{,}000\ \text{man-rem/\$M}$$

Other Considerations

Another factor that was considered in this issue was the accident avoidance cost, estimated to be approximately $11M, which would result in a potential cost saving of $7M, considering the $4M implementation costs.

CONCLUSION

Based on the above value/impact score, this issue was given a high priority ranking. A staff review of PRAs submitted by the affected licensees was used to identify the strengths and weaknesses of the various plants and to assess the risk associated with their operation. A special adjudicatory proceeding was held from 1982 to 1983 during which time the issues regarding continued operation and risk of the Indian Point plants were heard. Following these hearings, the Commission concluded that neither shutdown of Indian Point Units 2 or 3 nor imposition of additional remedial actions beyond those already implemented by the licensees were warranted.[806]

The staff also reviewed the Zion PRA and concluded that the risk posed by the Zion plants was small. The dominant contributors to severe accidents at the Zion plants were examined and the staff recommended that: (1) the integrity of the two motor-operated gate valves in the RHR suction line from the RCS be checked each refueling outage; and (2) the diesel-driven containment spray pump be modified so that it could be capable of operating without AC power.[806] Thus, this item was RESOLVED and new requirements were established. DL/NRR was responsible for managing the implementation of the above recommendations.[806]

ITEM II.B.7: ANALYSIS OF HYDROGEN CONTROL

DESCRIPTION

The TMI-2 accident resulted in a metal-water reaction which involved hydrogen generation in excess of the amounts specified in 10 CFR 50.44. As a result, it became apparent to the NRC that additional hydrogen control and mitigation measures would have to be considered for all nuclear power plants.

The purpose of this TMI Action Plan item[48] was to establish the technical basis for the interim hydrogen control measures on small containment structures and to establish the basis for continued operation and licensing of plants, pending long-term resolution of the hydrogen control issue. The long-term resolution of this issue was accomplished by rulemaking as part of Item II.B.8. A final rule was published on December 2, 1981 requiring inerting of the small BWR MARK I and II containments. In addition, based on Commission guidance, interim hydrogen control systems were required as a licensing condition for the intermediate volume ice condenser and MARK III containments. A proposed rule was published on December 23, 1981 (Federal Register 46 FR 62281) which required these systems for the intermediate volume containments. Except for pending CP and ML applications, no additional requirements for hydrogen control or hydrogen analyses were imposed at that time for large, dry containments. However, the proposed rule required that dry containments be analyzed to determine their ability to accommodate the release of large quantities of hydrogen (75% metal-water reaction). Also, hydrogen control requirements were established as part of the final Near Term CP and ML Rule published on January 15, 1982.

CONCLUSION

Based on the accomplishments above, the basis for continued operation and licensing of plants with respect to the hydrogen control issue was established. Future work related to finalizing the proposed rule dealing with intermediate volume containments (Ice Condenser and MARK III) and large, dry containments continued as part of Item II.B.8.

ITEM II.B.8: RULEMAKING PROCEEDING ON DEGRADED CORE ACCIDENTS

DESCRIPTION

Historical Background

In the past, safety reviews concentrated on how to prevent a core from being damaged. Consequently, little attention was given to how a severely damaged core could be dealt with after damage occurred. Other subtasks within Task II.B were concerned with the study of the characteristics of degraded and melted cores (research programs) plus some immediate actions to be taken at plants in operation. Item II.B.8 envisioned both a short-term and a long-term rulemaking to establish policy, goals, and requirements to address accidents resulting in core damage greater than the existing design basis.

Item II.B.8 included an Advance Notice of Proposed Rulemaking and an Interim Rule. The Advance Notice was issued in December 2, 1980 (45 FR 65474). The Interim Rule was issued in two parts: the first was issued in effective form in October, 1981 (46 FR 58484) and the second was issued as a proposed rule on December 23, 1981 (46 FR 62281).

On January 4, 1982, the staff sent a policy paper, SECY-82-1,[309] to the Commission requesting reconsideration of the approach to long-term rulemaking. The events which prompted this request were as follows:

- The Commission had required more protection from severe accidents in some licensing actions (e.g., Sequoyah) than was envisioned in the TMI Action Plan.

- A rule was developed to specify additional requirements for pending CP and ML applications. Again, these requirements were somewhat more extensive than that envisioned in the TMI Action Plan.

- New PRAs indicated lower risk than was previously estimated for large, dry PWR containments.

- The safety of existing plants had been considerably improved by the modifications guided by NUREG-0737.[98]

- The industry initiated a program to study the costs and benefits of design features for mitigating severe accidents.

- An extensive research program to study damaged and melted core behavior was underway.

- A safety goal statement, based on PRA, was developed.

The substance of SECY-82-1[309] was that the uncertainty associ with long-term rulemaking was an inhibiting force on the industry. The paper recommended that, since new applications were to be standardized, licensing could proceed on these standardized designs using the information available. PRAs and the safety goal would be used to assess plant safety. If plants needed safety features beyond the existing requirements to meet the safety goal, they could be included. This approach would not need rulemaking specifically directed at severe accident mitigation.

The Commission directed[310] the staff to make several changes recommended in SECY-82-1.[309] The staff then submitted revised papers SECY-82-1A[311] and SECY-82-1B[1405] that incorporated the changes directed by the Commission, including ACRS input. The evaluation of this item included consideration of Item II.B.7.

Safety Significance

Most of the engineered safety features at nuclear power plants of the existing generation were intended to prevent severe core damage. Relatively little attention was given in the past to dealing with a severely damaged or melted core. Once a core is damaged, the containment will still prevent the release of large amounts of radioactive material. However, once the core melts, the containment is likely to fail (although the hazard to the public varies widely, depending on the way in which the containment fails).

The degraded-core accident rulemaking was intended to require means for dealing with a damaged core. This translated into preventing the release of radioactivity and providing means for recovering from the accident. Specific items to be considered included the following: use of filtered, vented containment; hydrogen control measures; core retention devices ("core catchers"); reexamination of design criteria for decay heat removal and other systems; post-accident recovery plans; criteria for locating highly radioactive systems; effects or accidents at multi-unit sites; and comprehensive review and evaluation of related guides and regulations.

PRIORITY DETERMINATION

The safety significance of this issue was essentially the same as that of the research programs described in the analyses of Items II.B.5(1) and II.B.5(2) above. Examination of the estimated frequency of core damage and/or core-melt, coupled with estimates of the potential effectiveness of engineering solutions (and their cost) led to the recommended high priority for Items II.B.5(1) and II.B.5(2). In the same manner, Item II.B.8 had the potential for a significant (and cost-effective) reduction in public risk. In addition, it should be noted that some of the plant modifications contemplated were far more expensive to backfit than to forward-fit. Unnecessary delay could have reduced the cost-effectiveness of the resolution to this issue.

CONCLUSION

Based on the above evaluation, this item was given a high priority ranking. Work performed by RES on the hydrogen control aspect of the issue resulted in a

Hydrogen Control Rule that was approved by the Commission and published in the Federal Register on January 25, 1985.[807] The severe accident portion of the issue was addressed in April 1983 by a Policy Statement that set forth the Commission's intentions for rulemakings and other regulatory actions for resolving safety issues related to reactor accidents more severe than design basis accidents (48 FR 16014). Certain severe accident technical issues identified under the discussion of long-term rulemaking were to be dealt with for future and existing plants through procedures and ongoing severe accident programs identified in the Policy Statement and described more fully in Chapter IV of NUREG-1070.[809] Thus, with the issuance of the rule on hydrogen control, this item was RESOLVED and new requirements were established.[808]

REFERENCES

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

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

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Revision 1) August 1980.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986.

98. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980.

309. SECY-82-1, "Severe Accident Rulemaking and Related Matters," January 4, 1982.

310. Memorandum for W. Dircks from S. Chilk, "Staff Requirements - Briefing on Status and Plan for Severe Accident Rulemaking (SECY-82-1)," January 29, 1982.

311. SECY-82-1A, "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation," July 16, 1982.

312. NUREG/CR-0165, "A Value-Impact Assessment of Alternate Containment Concepts," U.S. Nuclear Regulatory Commission, June 1978.

806. Memorandum for W. Dircks from H. Denton, "Closeout of TMI Action Plan, Task II.B.6, 'Risk Reduction for Operating Reactors at Sites With High Population Densities,'" September 25, 1985.

807. Memorandum for W. Dircks from R. Minogue, "Closeout of TMI Action Plan Task II.B.8 'Rulemaking Proceeding on Degraded Core Accidents - Hydrogen Control,'" July 19, 1985.

808. Memorandum for W. Dircks from H. Denton, "Close Out of TMI Action Plan, Task II.B.8," August 12, 1985.

809. NUREG-1070, "NRC Policy on Future Reactor Designs," U.S. Nuclear Regulatory Commission, July 1985.

1102. Memorandum for T. Speis from R. Houston, "Integration of Generic Issue Resolution," November 4, 1987.

1381. Memorandum for W. Minners from B. Sheron, "Update of Generic Issue Management Control System (GIMCS)," July 5, 1991.

1382. NUREG-1365, "Revised Severe Accident Research Program Plan," U.S. Nuclear Regulatory Commission, August 1989.

1405. SECY-82-1B, "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation," November 24, 1982.

TASK II.J.4: REVISE DEFICIENCY REPORTING REQUIREMENTS

The objective of this task was to clarify deficiency report requirements to obtain uniform reporting and earlier identification and correction of problems.

ITEM II.J.4.1: REVISE DEFICIENCY REPORTING REQUIREMENTS

DESCRIPTION

This TMI Action Plan[48] item called for the NRC to revise, as necessary, the event-reporting requirements of 10 CFR 21 to assure that all reportable items are reported promptly and that the information submitted is complete. Improvements were to be implemented by rule changes as appropriate and coordinated with those made under TMI Action Plan Item I.E.6. The reports received as a result of these rule changes will provide increased information on component failures that affect safety so that prompt and effective corrective action can be taken. The information will also be used as input to an augmented role of the NRC's vendor and construction inspection program.

CONCLUSION

This issue was originally classified as nearly-resolved based on changes to 10 CFR 50.55(e) and 10 CFR 21 proposed by OIE.[291,292] The issue was later RESOLVED with new requirements when amendments to 10 CFR 21 and 10 CFR 50.55(e) were issued.[1396] The staff's changes were presented to the Commission in SECY-91-150.[1397]

REFERENCES

48. NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident," U.S. Nuclear Regulatory Commission, May 1980, (Revision 1) August 1980.

291. Memorandum for E. Jordan, et al., from R. Bernero, "Proposed Rule Review Request - 10 CFR Part 21 and 10 CFR Part 50.55(e), Reporting of Defects and Non-Compliance," February 3, 1983.

292. Memorandum for R. Minogue from R. DeYoung, "Proposed Rule Amending 10 CFR Parts 50.55(e) and 21: RES Task Numbers RA 128-1 and RA 808-1," July 13, 1982.

1396. Federal Register Notice 56 FR 36081, "10 CFR Parts 21 and 50, Criteria and Procedures for the Reporting of Defects and Conditions of Construction Permits," July 31, 1991.

1397. SECY-91-150, "Proposed Amendments to 10 CFR Part 21, 'Reporting of Defects and Noncompliance' and 10 CFR 50.55(e), 'Conditions of Construction Permits,'" May 22, 1991.

ISSUE 24: AUTOMATIC ECCS SWITCHOVER TO RECIRCULATION

DESCRIPTION

Historical Background

This issue was raised by the staff following a review[28,29] of operating events that indicated a significant number of ECCS spurious actuations, particularly, the four events that occurred at Davis-Besse during 1980.

ECCS operation has two different phases in PWRs: injection and recirculation. The injection phase involves initial cooling of the reactor core and replenishment of the primary coolant following a LOCA. In this phase, the ECCS pumps typically take suction from the refueling water storage tank (RWST). The recirculation phase provides long-term cooling during the accident recovery period. The ECCS is realigned in the recirculation phase to take suction from the containment sump.

Switchover from injection to recirculation phase involves realignment of several valves and may be accomplished by: (1) manual operations to realign the valves; (2) fully automatic realignment of the valves; or (3) automatic realignment of some valves, followed by manual completion of the process (semi-automatic). Each option is vulnerable in varying degrees to human errors, hardware failures, and common cause failures.

Safety Significance

During a LOCA, ECCS pump suction must be switched from the RWST to the containment sump before RWST inventory is lost or loss of the ECCS pumps will occur. Switching to the sump early could adversely affect the accident because the containment sump may not have enough inventory to provide pump suction. The automatic and the semi-automatic switchovers reduce the risk of human error but have a slight increase in risk for inadvertent actuations. This issue affects PWRs only.

Possible Solutions

The two possible solutions to this issue are alternate cases requiring fully-automatic or semi-automatic switchover to the containment sump. The fully-automatic switchover could be implemented by a system that would monitor the water level in the RWST and, at a preset level, automatically realign the ECCS to take suction from the containment sump. The semi-automatic switchover could be implemented by a system that would involve automatic alignment of some valves and manual completion of the switchover process.

PRIORITY DETERMINATION

Assumptions

All operating or proposed PWRs may be affected by this issue and the Oconee 3 PRA was assumed to be representative of PWRs. LERs between 1987 and 1990 were used to calculate the potential risk from spurious actuations. The spurious actuation probability was then used for the automatic switchover and modified for the semi-automatic switchover.

The base case of the possible solutions assumes manual switchover. Some PWRs are already fully automatic and some are semi-automatic. It was assumed that all PWRs with manual switchover could benefit from the possible solutions; since some PWRs are already automatic, some (fixed) costs will be spread over fewer reactors than calculated.

Frequency Estimate

The issue to be addressed is the failure of an operator to open containment sump suction valves at the start of recirculation. New parameters were introduced to provide estimates of recirculation system unavailability corresponding to manual, automated, and semi-automated switchover options.[29] The new parameters were then updated for human error rate estimates given in NUREG/CR-4639.[1327] The updated parameters were then factored into the core-melt frequency.

The frequencies of the affected release categories were summed for each case to give the total core-melt frequency for the three cases considered.[64]

Base Case (Manual):	3.1×10^{-6}/RY
Semi-Automatic Switchover:	1.6×10^{-6}/RY
Fully-Automatic Switchover:	1.3×10^{-6}/RY

The adjusted case core-melt frequencies were calculated by substituting the adjusted probabilities into the failure scenarios which require sump suction valves to be opened for success.[64] Thus, the potential reduction in core-melt frequency was estimated to be 1.5×10^{-6}/RY and 1.8×10^{-6}/RY for the semi-automatic and the fully-automatic switchover options, respectively.

Consequence Estimate

Multiplying the affected release categories by the estimated public dose, the total affected public risk for the three cases were as follows:

Base Case (Manual):	7.5 man-rem/RY
Semi-Automatic Switchover:	3.2 man-rem/RY
Fully-Automatic Switchover:	3.0 man-rem/RY

Thus, the estimated risk reduction was 4.3 man-rem/RY and 4.5 man-rem/RY for the semi-automatic and fully-automatic switchover, respectively. Based on an average remaining operating life of 28.8 years for PWRs, this reduction was estimated to be 125 man-rem/reactor and 130 man-rem/reactor for the semi-automatic and fully automatic switchover, respectively.

Installing automatic or semi-automatic systems reduces human error. However, the estimated risk reduction from installing actuation systems that are less prone to human error was offset somewhat by an increased risk due to spurious actuations.

Cost Estimate

Industry Cost: The cost was estimated to be the same for both semi-automatic and fully-automatic switchover at all affected PWRs. The estimates for TS, maintenance procedure, and operating procedure changes were taken from NUREG/CR-4627.[961] The implementation costs were calculated as follows:

Design/QA	= 8 man-weeks
Install/Calibrate/Test Equipment	= 1 man-week
Safety Analysis	= 8 man-weeks
TS Changes	= 16 man-weeks
Training	= 8 man-weeks
Hardware (New Controller/Logic Module)	= $5,000
Revise Operating and Maintenance Procedures	= $7,800

Thus, the total estimated cost was $110,000/plant, based on 41 man-weeks at $2,270/week and a fixed cost of $12,800.

Operation and maintenance of the possible solutions were estimated to require an additional 1 man-week/RY. Over the average remaining operating life of 28.8 years, and at a discount rate of 5%, this cost was estimated to be $34,000/reactor.

NRC Cost: It was estimated that 1 man-year of contractor effort will be required to research potential design changes and prepare a regulatory analysis. A project manager will be required at 10% of the contractor cost. At an estimated cost of $100,000/man-year, the contractor and project manager cost was estimated to be $110,000.

Eight man-weeks will be required to review and evaluate each plant's design, safety analyses, QA documentation, TS changes, and procedure changes. With an assumed labor cost of $2,270/week, this cost is $18,000/reactor.

Total Cost: The total industry and NRC cost associated with the possible solutions to this issue is $272,000/reactor.

Value/Impact Assessment

Separate value/impact scores were calculated for the semi-automatic switchover and the fully-automatic switchover possible solutions.

(1) Semi-Automatic:

$$S = \frac{125 \text{ man-rem/reactor}}{\$0.272\text{M/reactor}}$$

$$= 460 \text{ man-rem/\$M}$$

(2) Fully-Automatic:

$$S = \frac{130 \text{ man-rem/reactor}}{\$0.272\text{M/reactor}}$$

$$= 478 \text{ man-rem/\$M}$$

Other Considerations

Since much of the work will be in radiation zones, a significant occupational dose will occur. The dose rate was assumed to be 2.5 millirem/hr for work outside containment.[64] The occupational dose was assumed to be the same for both possible solutions. The implementation dose was calculated at 0.5 man-rem/reactor and total operation and maintenance dose at 0.6 man-rem/reactor. This results in a total ORE of 1 man-rem/reactor.

CONCLUSION

Based on the value/impact score and the potential risk reduction for PWRs with manual switchover, this issue was given a MEDIUM priority ranking. Furthermore, since the uncertainties in the assumptions and analysis are very large, a more extensive study than is possible in a prioritization would be required to resolve this issue with reasonable confidence in the conclusion. Therefore, resources should be allocated to obtain more reliable estimates of equipment reliability, human error rates, and competing risks and, thereby, resolve this issue in a definitive manner. The resolution of this issue will address the concern of Issue 156.3.5.

REFERENCES

28. Memorandum for T. Novak from P. Check, "Spurious Automatic Switchover of ECCS from the Injection Mode to the Recirculation Mode," January 21, 1981.

29. Memorandum for T. Novak and J. Olshinski from A. Thadani, "Comparative Risk Assessment of ECCS Functional Switchover Options," April 1, 1981.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986.

961. NUREG/CR-4627, "Generic Cost Estimates," U.S. Nuclear Regulatory Commission, June 1986.

1327. NUREG/CR-4639, "Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR)," U.S. Nuclear Regulatory Commission, (Volume 1) February 1988, (Volume 2) September 1988, (Volume 3) November 1988, (Volume 4) June 1988, (Volume 5) June 1988.

ISSUE 29: BOLTING DEGRADATION OR FAILURE IN NUCLEAR POWER PLANTS

DESCRIPTION

Historical Background

Prior to 1981, the number of bolting-related incidents reported by licensees was on the increase. A large number of these were related to primary pressure boundary applications and major component support structures. As a result, there was concern for the integrity of the primary pressure boundary in operating plants and the reliability of the component support structures following a LOCA or earthquake. This issue was identified by the ACRS.[1384]

There are numerous bolting applications in nuclear power plants the most crucial of which are those constituting an integral part of the primary pressure boundary such as closure studs and bolts on reactor vessels, reactor coolant pumps, and steam generators. Failure of these bolts or studs could result in the loss of reactor coolant that could jeopardize the safe operation of the plants. Other bolting applications, such as component support and embedded anchor bolts or studs, are essential for withstanding transient loads created during abnormal or accident conditions. A report summarizing bolting failure experience was issued by DL/NRR.[184]

Safety Significance

At the time of this evaluation, there had been a total of 44 reported bolting incidents most of which were discovered either during refueling outages or scheduled ISI or maintenance/repair outages. These incidents had no immediate impact on public health and safety since they had not resulted in any accident. However, degradation or failure of such studs and bolts constitutes a reduction in the integrity of the primary pressure boundary.

Concern was compounded by the fact that there was no reliable NDE method to detect the cracking or degradation of bolts or studs resulting from the principal modes of failure: stress corrosion, fatigue, erosion corrosion, and boric acid corrosion. Visual examination was the only reliable method to discover degradation by boric acid corrosion or erosion corrosion. In almost all cases, this required disassembly of the component in order to inspect the bolts or studs. If there is no clear evidence of boric acid leakage to the surroundings, bolting degradation by boric acid corrosion can potentially be undetected until the bolts or studs completely fail. Under the existing ISI program, visual inspection of bolts was not a mandatory requirement and UT inspection was not required on pressure-retaining bolts or studs with diameters less than 2 inches. A major accident such as a LOCA could conceivably occur due to undetected extensive bolting failure of the primary pressure boundary.

Possible Solution

Because bolting has a wide range of application in nuclear power plants, there is no simple solution to the problem. Therefore, in order to minimize the potential bolting problems in new power plants, improvements in one or all of the following five areas could be recommended: design, materials, fabrication, installation, and ISI. For this analysis, the focus was placed on improving the efficiency and adequacy of ISI programs.

PRIORITY DETERMINATION

Frequency Estimate

Based on a review of the 44 bolting incidents reported by PWR licensees, the principal causes of bolting failure or degradation were classified as stress corrosion, fatigue, boric acid corrosion, erosion corrosion, and other types. A total of 19 bolting incidents were identified as resulting from stress corrosion, the most common cause of bolting failure. Boric acid corrosion was the second most common cause of bolting failure or degradation reported. A total of 12 bolting incidents resulting from boric acid corrosion occurred. The remaining 13 incidents were either fatigue, erosion corrosion, or other types. No bolting failures in BWRs were reported.

A total of 16 of the reported 44 incidents were related to primary pressure boundary bolting applications such as various closure studs in reactor vessels, pressurizers, steam generators, and hold-down bolts in various types of valves. A total of 13 bolting incidents related to component support structures, such as the column support or embedded anchor bolts or studs of steam generators, reactor vessels, reactor coolant pumps, and piping restraints, were reported. Although failure of such bolts or studs will not normally impair the normal operation of a plant, extensive failure of such bolts or studs could cause component damage or multiple piping failure under abnormal or accident conditions such as a LOCA or an earthquake. The 44 bolts or stud failures occurred in about 350 RY of experience. Thus, the frequency of corrision-initiated events was 44/350 event/RY or 1.3×10^{-1} event/RY.

Based on experience, there is a good chance that the corrosion will be discovered and the studs replaced before failure occurs. However, it was conservatively assumed that 10% of the bolts or studs will not be discovered before they fail and will result in a small break LOCA (S2). Therefore, the frequency (F) of corrosion-initiated events was estimated to be 1.3×10^{-2} S2 event/RY.

Twenty-nine of the reported incidents or 66% had a direct potential for causing a large-break LOCA due to bolting or stud failure in restaints for large piping, component supports, or steam generator manways when these hold-down devices have degraded to the point that they will not provide the necessary support following a water hammer or seismic event. However, even though the actual determination is complex, the S2 event was believed to be the most limiting.

Consequence Estimate

An S2 event can result in a wide spectrum of consequences, depending on whether or not the engineered safety features are required to function or whether they do function. Using WASH-1400[16] S2 sequences with the frequency estimated above, the release was determined to be 3×10^4 Curies/RY.

The total whole-body man-rem dose was obtained by using the CRAC Code[64] for the particular release category. A uniform population density of 340 people per square mile (which is average for U.S. domestic sites) and a typical (midwest plain) meteorology were assumed. Therefore, the estimated public risk was 3.5 x 10^3 man-rem/RY. For 43 plants with an average remaining lifetime of 30 years, the potential risk reduction was 4.5 x 10^5 man-rem.

Cost Estimate

Industry Cost: The proposed fix could vary considerably depending on the type and depth of solution sought. However, the most probable fix was to visually inspect bolts or studs using an improved UT inspection technique and a more frequent inspection schedule. This represented an increase in surveillance and would require extra effort during each plant refueling outage. Because of the wide variety of uses of studs and bolts for safety functions in nuclear plants, the actual cost would vary greatly.

Based on the information provided,[64] an interim and simple fix would be to inspect studs and bolts only on components that had been opened for inspection or maintenance during a refueling outage. This would require a minor increase in surveillance and would not require an extension of outage time. It was estimated that 3 man-weeks/plant of extra effort would be required per 18-month refueling schedule and 40 man-weeks/plant to allow for administrative overhead. At \$100,000/staff-year, the cost (C) of the increased surveillance over the 30-year life of a plant was given by:

$$C = \$\left[\frac{(3)(30/1.5) + 40}{52}\right](0.1)\text{M/reactor}$$

$$= \$0.2\text{M/reactor}$$

If, however, each plant was required to inspect 10% of the bolts or studs in primary system components per refueling outage, whether open for inspection or not, then based on an 18-month refueling schedule, each plant will have inspected 200% of its bolts or studs over its 30-year lifetime. (This inspection frequency should detect any bolt degradation that might occur.) This would represent an increase in surveillance and would require an extension of outage time by 1.5 working days. At \$300,000/day for replacement power, the total cost (C) over the plant life was given by:

$$C = \$(30/1.5)(0.3)(1.5)\text{M/reactor}$$

$$= \$9\text{M/reactor.}$$

NRC Cost: NRC costs per reactor were negligible in comparison to industry costs.

Value/Impact Assessment

(1) For inspecting bolts on disassembled components only, the value/impact score was given by:

$$S = \frac{4.5 \times 10^5 \text{ man-rem}}{\$(0.2)(43)\text{M}}$$

$$= 53{,}000 \text{ man-rem/\$M.}$$

(2) For inspecting 10% of the bolts, the value/impact score was given by:

$$S = \frac{4.5 \times 10^5 \text{ man-rem}}{\$(9.0)(43)M}$$

$$= 1,160 \text{ man-rem/\$M.}$$

Uncertainties

The uncertainties in the estimates of accident frequencies and consequences were such that, if they lowered the value/impact score by an order of magnitude, the score would still be above a threshold that would warrant resolution of the issue. As the cost estimates increase for specific solutions (particularly when plant shutdown or extended shutdowns are required), the value/impact scores decrease and could affect the priority ranking. If the cost estimates associated with inspecting 10% of the bolts are off by a large factor, the potential risk reduction would still be sufficiently high to maintain a high priority ranking.

Other Considerations

A secured reactor primary system pressure boundary, which depends on the integrity of the system's piping and components, is an integral part of the "defense-in-depth" concept embodied in the design of nuclear power plants to protect against a core-melt. Also, some safety system functions rely on a secured pressure boundary to prevent or mitigate the consequences of an event. Accordingly, inspection of 10% of the bolts per refueling outage (200% over the lifetime of the plant) provides assurance that the primary system pressure boundary will not be breached by failed bolts or studs.

When the averted costs of cleanup following a LOCA are considered, the value/impact scores calculated above become more favorable. It was estimated that the averted occupational dose of inspection versus reduction due to accident dose would fall between a PWR-8 or PWR-9 event and a PWR-1 to PWR-7 event. This represented an averted dose between 2,400 to 8,000 man-rem.

CONCLUSION

Based on the above value/impact scores, this issue was given a high priority ranking. In resolving the issue, the staff took into the consideration previous actions taken by the NRC to address the concerns related to threaded fasteners: Bulletin Nos. 82-02,[1129] 87-02,[1389] and 89-02[1388]; Information Notice Nos. 86-25,[1393] 89-22,[1390] 89-56,[1391] and 89-70[1392]; and Generic Letter Nos. 87-02[1387] and 88-05.[1386]

The staff's regulatory analysis, NUREG-1445,[1398] proved to be inconclusive regarding a mandatory program on safety-related bolting for operating plants. The staff's technical findings were documented in NUREG-1339[1395] which endorsed the recommendations of independent studies performed by the industry Joint Task Group on Bolting. This group was set up by AIF, EPRI, and the Materials Properties Council and its studies resulted in EPRI NP-5769, "Good Bolting Practices," and three EPRI video training tapes on "Pressure Boundary Bolting Problems."

The staff concluded that leakage of bolted pressure joints was possible, but catastrophic RCPB joint failure that could lead to significant accident sequences was highly unlikely. This conclusion was based on: (1) operating experience; (2) actions taken through bulletins, generic letters, and information notices; and (3) proposed industry actions. Generic Letter No. 91-17[1385] was issued to licensees to: (1) implement the industry bolting integrity program, as presented in the EPRI report and video tapes; and (2) continue actions in accordance with commitments made in response to NRC generic letters and bulletins. Thus, this issue was RESOLVED and no new requirements were established. However, in order to improve the review of future plants and significant modifications to operating plants, the staff recommended that a new SRP[11] Section be developed to codify existing guidance and industry recommendations.[1394]

REFERENCES

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

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

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986.

184. Memorandum for R. Vollmer from D. Eisenhut, "Transmittal of Report on Threaded Fastener Experience in Nuclear Power Plants," August 25, 1982.

1129. IE Bulletin No. 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants," U.S. Nuclear Regulatory Commission, June 2, 1982.

1384. Memorandum for W. Dircks from R. Fraley, "Bolt Failures in Nuclear Power Plants," October 20, 1981.

1385. NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants, "Generic Safety Issue 29, 'Bolting Degradation or Failure in Nuclear Power Plants,' (Generic Letter 91-17)," October 17, 1991.

1386. NRC Letter to All Licensees of Operating PWRs and Holders of Construction Permits for PWRs, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants (Generic Letter 88-05)," March 17, 1988.

1387. NRC Letter to All Holders of Operating Licenses Not Reviewed to Current Licensing Criteria on Seismic Qualification of Equipment, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46 (Generic Letter 87-02)," February 19, 1987.

1388. NRC Bulletin No. 89-02, "Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves or Valves of Similar Design," U.S. Nuclear Regulatory Commission, July 19, 1989.

1389. NRC Compliance Bulletin No. 87-02, "Fastener Testing to Determine Conformance with Applicable Material Specifications," U.S. Nuclear Regulatory Commission, November 6, 1987, (Supplement 1) April 22, 1988, (Supplement 2) June 10, 1988.

1398. NUREG-1445, "Regulatory analysis for the Resolution of Generic Safety Issue-29: Bolting Degradation or Failure in Nuclear Power Plants,"U.S. Nuclear Regulatory Commission, September 1991.

1390. NRC Information Notice No. 89-22, "Questionable Certification of Fasteners," U.S. Nuclear Regulatory Commission, March 3, 1989.

1391. NRC Information Notice No. 89-56, "Questionable Certification of Material Supplied to the Defense Department by Nuclear Suppliers," U.S. Nuclear Regulatory Commission, July 20, 1989, (Supplement 1) November 22, 1989, (Supplement 2) July 19, 1991.

1392. NRC Information Notice No. 89-70, "Possible Indications of Misrepresented Vendor Products," U.S. Nuclear Regulatory Commission October 11, 1989, (Supplement 1) April 26, 1990.

1393. IE Information Notice No. 86-25, "Traceability and Material Control of Material and Equipment, Particularly Fasteners," U.S. Nuclear Regulatory Commission, April 11, 1986.

1394. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Safety Issue 29, 'Bolting Degradation or Failure in Nuclear Power Plants,'" October 25, 1991.

1395. NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, June 1990.

1398. NUREG-1445, "Regulatory Analysis for the Resolution of Generic Safety Issue-29: Bolting Degradation or Failure in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, September 1991.

ISSUE 38: POTENTIAL RECIRCULATION SYSTEM FAILURE AS A CONSEQUENCE OF INGESTION OF CONTAINMENT PAINT FLAKES OR OTHER FINE DEBRIS

DESCRIPTION

Historical Background

This issue was identified[496] when AEOD expressed concerns about the use inside containment of a particular polymer coating that could flake off and fail when subjected to DBA conditions. In addition to the concern for paint flakes, AEOD also raised concerns about fibrous insulation and other debris that could pass through sump screens, but could not pass through the more restrictive clearances present in systems that take suction from the containment sump during the recirculation phase of accident mitigation.

Safety Significance

Potential safety concerns stemming from the presence of paint debris in the containment building during a LOCA include the following: (1) blockage of containment emergency sump debris screens; (2) blockage of containment building spray system nozzles and system flow passages associated with residual heat removal/safety injection systems and their equipment; and (3) degradation of ECCS performance by the entrainment of fine particles of paint debris. This issue is applicable to all plants.

Possible Solution

In the resolution of Issue A-43, the staff evaluated the performance of the containment emergency sump in providing a clean, reliable source of water during a LOCA and during long-term recirculation following a LOCA. Specifically, the evaluation included analysis of the transport of fine debris.

In its application to operate Comanche Peak Steam Electric Station, Units 1 and 2, Texas Utilities Electric Company (TUEC) performed an analysis in support of its request to amend its FSAR to eliminate the commitment that coatings inside the reactor containment building be qualified. This analysis considered the potential for, and effects of, debris blockage of the containment building emergency sumps. TUEC followed the guidance and methodology developed by the staff in the resolution of Issue A-43 and concluded that debris generated by the failure of all coatings inside the containment building under DBA conditions would not unacceptably degrade the performance of post-accident fluid systems. The staff's SER on the TUEC analysis was published in Supplement No. 9 to NUREG-0797.[1332]

CONCLUSION

The general concerns of sump blockage were addressed in the technical findings reported in NUREG-0897,[1057] the revisions to Regulatory Guide 1.82,[1058] SRP[11] Section 6.2.2, and Generic Letter 85-22.[1059] The TUEC analysis provided data[1057] on the significance of containment sump blockage caused by paint flakes or other fine debris. Thus, this issue was DROPPED from further consideration as a new and separate issue.

REFERENCES

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

496. Memorandum for H. Denton from C. Michelson, "Concerns Relating to the Integrity of a Polymer Coating for Surfaces Inside Containment (IE Draft Bulletin No. 80-21)," August 29, 1980.

1057. NUREG-0897, "Containment Emergency Sump Performance," U.S. Nuclear Regulatory Commission, (Revision 1) October 1985.

1058. Regulatory Guide 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission, (Revision 1) November 1985.

1059. NRC Letter to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage (Generic Letter 85-22)," December 3, 1985.

1332. NUREG-0797, "Safety Evaluation Report Related to the Operation of Comanche Peak Steam Electric Station, Units 1 and 2," U.S. Nuclear Regulatory Commission, (Supplement 9) March 1985.

ISSUE 73: DETACHED THERMAL SLEEVES

DESCRIPTION

Historical Background

During the period 1978 to 1980, there were reports of fatigue failure of thermal sleeve assemblies in the piping systems of both PWRs and BWRs. The BWR problem was addressed by GE in NEDO-21821 and was resolved with a staff SER[1314] and the publication of NUREG-0619.[742] Fatigue problems occurred subsequently in 1982 in PWRs designed by B&W and W. IE Information Notices No. 82-09[1311] and No. 82-30[1312] were issued by the staff to address the problems at B&W and W reactors, respectively. Status reports on the B&W pipe cracking problem were contained in SECY-82-186[1313] and SECY-82-186A[513] which described the problem of thermal sleeve cracking in the normal make-up/high pressure injection nozzles of four B&W plants. No notable events have been reported at PWRs designed by CE.

As a result of the fatigue failures in B&W reactors, a B&W Owners' Group Task Force was established to identify the cause of the failure and to recommend modifications to eliminate future failures. This Task Force submitted a report[514] to the NRC and the concern was resolved in Issue 69, "Make-up Nozzle Cracking in B&W Plants."

The concern regarding thermal sleeves in W-designed plants was raised by the staff[518] following remote video inspections that revealed pieces of metal at the bottom of a W reactor vessel at the Trojan Nuclear Plant; a metal fragment was also found between the lower core plate and the core support plate. All metal pieces were subsequently identified as part of the thermal sleeves initially installed in the safety injection accumulator piping nozzle connections to the reactor coolant system cold leg piping. Confirmation that the 10-inch thermal sleeves were missing from the four safety injection piping nozzle connections was obtained shortly thereafter. In response to the Trojan cracking and detachment of thermal sleeves, a plant-specific review was conducted and an SER[1315] was completed by the staff. In this SER, the staff established the basis for continued operation of the Trojan plant, subject to the findings of a staff generic study on W plants.

There have been five generations (0 through 4) of thermal sleeves used in W reactors. Only "Generation 3" thermal sleeves have been found to be susceptible to high-cycle stresses due to flow-induced vibrations because of the particular weld attachments used in that design. The vibrations caused fatigue failures at the thermal sleeve attachment welds and subsequent cracking and tearing away of the thermal sleeves resulted in the formation of loose parts moving into the reactor vessel. This issue applies to the design and operation of approximately 20 W plants that use "Generation 3" thermal sleeves.

Safety Significance

The safety significance of loose parts in the vessel is that there is the potential for flow blockage and/or cladding wear or destruction by parts wedged within the fuel assemblies. In addition, if the thermal sleeves become detached, or are removed and not replaced, there is an ancillary concern that the nozzle cumulative usage factor may be exceeded during the remaining lifetime of a plant.

Possible Solution

After an evaluation of the thermal sleeve problems at several W plants, the staff concluded[1316] that licensees could elect to either retain the "Generation 3" thermal sleeves or remove them. If the decision was to retain the sleeves, licensees would have to develop a program to inspect the attachment welds of these sleeves at each refueling outage. If licensees elected to remove the "Generation 3" thermal sleeves, they would have to submit revised TS to monitor the injection flow transients which occur at the affected nozzles and evaluate the cumulative fatigue usage factors.

CONCLUSION

This issue was resolved for BWRs with the publication of NUREG-0619;[742] for B&W reactors, the issue was resolved in Issue 69. No problems have been reported in CE reactors. For W reactors, a proposed resolution has been identified[1316] and, therefore, the issue was considered to be nearly-resolved.

REFERENCES

513. SECY-82-186A, "Make-up Nozzle Cracking in Babcox and Wilcox (B&W) Plants," July 23, 1982.

514. B&W Document No. 77-1140611-00, "177 Fuel Assembly Owners' Group Safe End Task Force Report on Generic Investigation of HPI/MU Nozzle Component Cracking," Babcock and Wilcox Company, 1983.

518. Memorandum for D. Eisenhut from J. Crews, "NRC Lead Responsibility for Possibly Detached Thermal Sleeves - Trojan Nuclear Plant - Docket No. 50-344," June 18, 1982.

742. NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," U.S. Nuclear Regulatory Commission, November 1980.

1311. IE Information Notice No. 82-09, "Cracking in Piping of Makeup Coolant Lines at B&W Plants," U.S. Nuclear Regulatory Commission, March 31, 1982.

1312. IE Information Notice No. 82-30, "Loss of Thermal Sleeves in Reactor Coolant System Piping at Certain Westinghouse PWR Power Plants," U.S. Nuclear Regulatory Commission, July 26, 1982.

1313. SECY-82-186, "Status of Make-up Nozzle Cracking in Babcock & Wilcox (B&W) Plants," May 7, 1982.

1314. Letter to R. Gridley (General Electric Company) from D. Eisenhut, "Safety Evaluation for the General Electric Topical Report NEDE-218121-02, 'BWR Feedwater Nozzle/Sparger Final Report, Supplement 2,'" January 14, 1980.

1315. Memorandum for D. Sternberg from R. Clark, "Degradation of Thermal Sleeves - Trojan Nuclear Plant," August 11, 1982.

1316. Memorandum for T. Novak, et al., from J. Knight, "Evaluation of Thermal Sleeve Problems in Westinghouse Plants," October 28, 1983.

ISSUE 87: FAILURE OF HPCI STEAM LINE WITHOUT ISOLATION

DESCRIPTION

Historical Background

The HPCI steam supply line has two containment isolation valves in series: one inside and one outside of the containment. Both are normally open in most plants; however, two plants were found to operate with the HPCI outboard isolation valve normally closed. A HPCI supply valve, located adjacent to the turbine, and the turbine stop valve are normally closed. This issue concerns a postulated break in the HPCI steam supply line and the uncertainty regarding the operability of the HPCI steam supply line isolation valves under those conditions.[824] A similar situation can occur in the RWCU system which has two normally open containment isolation valves that must remain open if the system is to function.[830]

The operation of the valves is tested periodically without steam. Due to flow limitations at the valve manufacturers' facilities, only the opening characteristics are tested under operating conditions. Therefore, the capability of the valves to close when exposed to the forces created by the flow resulting from a break downstream has not been demonstrated. However, there are reasons why the valves may operate under these accident conditions. The containment isolation valves are specified to open or close within 15 to 20 seconds. Calculations performed by Bechtel[826] indicated that the mass flow through the HPCI steam line isolation valves reduces from 1470 lbm/sec. at the time of a break to 328 lbm/sec. after 0.135 seconds and remains constant until the valve closes.

Isolation valves are selected by the A/E for each plant. This results in a diversity of valves and valve types from plant to plant and increases the difficulty of demonstrating valve operating capability. Some plants have "Y-type" globe valves while others have gate valves. One plant using globe valves for HPCI steam supply isolation had the valve inside containment positioned such that the steam flow exerted a force on the valve skirt in the closed position. This force is expected to reduce the closing torque requirement of the valve motor-operator and increase the probability that the valve will close when a large amount of steam is flowing through the valve. Also, some valve experts believe that the force required to open gate valves under pressure is greater than the force required to close the valves under flow.

Safety Significance

In Mark I containments, the HPCI steam line exits the drywell and enters the torus compartment where it typically traverses approximately a 75° arc before exiting to the HPCI pump room. In the four corners of the reactor building along the torus compartment are four triangular-shaped rooms which house the RHR/LPCI system, the RCIC system, and the core spray system. In some reactor buildings, there is a ventilation opening or a door, usually open, between the rooms housing the emergency core cooling pumps and the torus compartment. Given an unisolated break of the HPCI steam supply line in the torus compartment, the environment in the emergency core cooling pump rooms may exceed design limits. This places in jeopardy the other systems required to cool the core.

In Mark II containments, the emergency core cooling components are typically housed in individual rooms which are contained in the large, annular-shaped area about the suppression pool. The HPCI steam supply line exits the containment and is then routed down through two floors to the room containing the HPCI turbine and pump. Again, given an unisolated break of the HPCI steam supply line, other systems which may be required to cool the core may be placed in jeopardy.

Possible Solutions

A proposed solution to the HPCI problem was to require that the outboard HPCI isolation valve be normally closed. However, a small bypass line on those plants not having this feature would be required to prevent thermal shock and water hammer and to provide assurance that leaks in the line would be detected before they become breaks. If the HPCI supply valve were kept normally open -- currently it is kept normally closed -- the probability of not getting steam to the HPCI turbine when needed might not be significantly changed.

Another possible solution that would apply to valves in any system was a demonstration by test or the verification of use in other service applications that certified the operability of the valve under line rupture flow conditions. If the normal HPCI steam flow rate approximates that estimated for a break in the steam line, the valves might be tested by individually closing them when the HPCI turbine is in operation.

PRIORITY DETERMINATION

Frequency Estimate

In the Browns Ferry 1 IREP study,[367] the frequency of intermediate-size steam line breaks (in which size category the HPCI steam supply line is included) is 2×10^{-4}/RY. It was also assumed that a break is equally probable at any point in the steam lines of this size and category. The HPCI steam supply lines were estimated to constitute 23% of the steam lines in the intermediate-size category. Hence, the frequency of a HPCI steam supply line break was assumed to be 5×10^{-5}/RY.

The probability that both steam supply line isolation valves will fail to close was difficult to determine on a probabilistic basis because of the lack of engineering data. If one valve fails to perform its intended function because of conditions which exceed its design capability, it would be most probable that the second valve would also fail to function. As an upper bound calculation, it was assumed that the valve failure rate will be unity, given a line break, and that the dependency between valves was also unity. The lower bound was calculated by assuming that the valve design was adequate and that there were no failure dependencies between the valves. Thus, the frequency of both valves failing was obtained by taking the product of the independent failure frequency of both valves.

The major contribution to the accident scenario considered was the dependency between the unisolated steam line breaks and the low-pressure injection systems.

For both the upper and lower bound calculations, it was assumed that the dependence is unity, i.e., that the low-pressure injection systems will fail, given an unisolated line break.

If, during the accident condition described, the core is maintained covered by the feedwater system, the steam mass flow generated by decay heat should lower to a point that would permit the closing of an isolation valve. One means available would be electrically closing the isolation valve inside the containment; the other means available would be manually closing one of the isolation valves.

If the steam flow forces prevent the initial closure of the isolation valve, the motor control breakers will likely trip from the overcurrent condition before motor damage can occur. Further, the isolation valve inside containment will not have been exposed to the steam environment from the broken line. Resetting the motor control breaker would then permit energizing the valve motor and closing the isolation valve from the control room.

The second method available was to close one of the isolation valves by manual actuation of the hand crank. This would require suiting the operator in special garments and possibly using an airpack. Due to the expected high temperature in the torus compartment, the isolation valve inside the containment would be the valve most likely closed.

NEDO-24708A[827] analyzed an unisolatable 0.5 square-foot steam line break inside containment, which approximated a break of the 10 in. (0.55 ft^2) HPCI steam line from the time of the break up to the time that the low pressure systems would begin injection (225 seconds). The analysis also included the water injected by the RCIC, but this should be minimal.

The 0.5 square-foot line break model predicted that the system pressure will fall below 300 psia at approximately 210 seconds after the break occurs. The water level will still be above the core and the condensate and the condensate booster pumps can be used (for those systems having a turbine-driven feed pump) to supply feedwater to the reactor. For those feedwater systems having motor-driven feed pumps, the feedwater system can supply feedwater continuously following the reactor trip. With the feedwater system providing cooling water, the fuel will remain covered until a HPCI isolation valve is closed and the RHR system is restored to operation.

It was calculated that 12,500 gallons/hour of water at 94°F will be converted to steam at 212°F in absorbing the decay heat from the fuel. At this rate of consumption, a 500,000 gallon condensate tank could be emptied in 40 hours. In order to maintain adequate coolant for the extended time period, the vacuum must be restored in the condenser and the decay heat dissipated using the condenser. This will also necessitate using the auxiliary boiler to provide steam for the gland seals. Having the condensers available will reduce the steam pressure in the reactor, thus reducing the amount of steam that will be discharged through the broken HPCI steam supply line and decreasing the consumption of water from the condensate storage tank. This action will also lower the amount of heat and humidity being dumped into the torus compartment.

The probability of the loss of the feedwater during a 168-hour interval, the time assumed necessary to restore the RHR system following a HPCI steam supply line break, was calculated to be 0.03. This was based on the Browns Ferry IREP[367] frequency of transients that result in loss of feedwater (~1.4/RY). This equated to a mean time between failure of 5,570 hours. Assuming an exponential distribution, a failure rate of 1.8×10^{-4}/hour results.

Of concern were the operator actions needed to maintain the operation of the main feedwater system. Although this is an activity with which the operator should be very familiar, detecting that the HPCI is not providing make-up inventory may not be immediate. Further, the inventory in the hotwell must be maintained by flow from the condensate storage tank. To obtain an adequate flow, it may be necessary to reestablish the vacuum in the condensers. As reported in NUREG/CR-3933,[828] PRAs assign a probability of 0.1 for failure to recover the power conversion system in a short interval. In this accident, the time needed to make the necessary operating adjustments will not be as short as required for transients or small breaks in liquid coolant lines. In addition, approximately one-fourth or one-half of the make-up water requirements will be provided by one- or two-pump operation of the CRD hydraulic system. Thus, a human error probability of 0.05 was assigned. The total probability of failing to maintain coolant inventory with the feedwater systems for 168 hours was estimated to be (0.05 + 0.03) = 0.08. Thus, the frequency estimates were:

Upper Bound: $(5 \times 10^{-5})(1)(1)(0.08) = 4 \times 10^{-6}$ core-melt/RY
Lower Bound: $(5 \times 10^{-5})(10^{-3})(10^{-3})(1)(0.8) = 4 \times 10^{-12}$ core-melt/RY

Closing the outboard isolation valve and opening the supply valve was assumed to result in no net change in the unavailability of the HPCI and, therefore, the frequency of other accident sequences was unchanged. Closing the outboard isolation valve until the HPCI is commanded does not reduce the accident rate from breaks that occur when the HPCI is energized or go undetected prior to the HPCI being energized. With the inclusion of a bypass line to prevent thermal shock, this contribution was believed to be much smaller than the long-term exposure with the line pressurized. Hence, the remaining contribution was not considered to be significant.

The BNL estimate[829] of the frequency of a core-melt accident due to an unisolated break outside the containment in a six-inch RWCU line was 1.4×10^{-5}/RY. The study also conservatively assumed that the conditional probability for the isolation valves failing to close, given a line break, was 1.

Consequence Estimate

A break in the HPCI steam supply line would be a LOCA outside containment. This would be closely equivalent to the PWR Event V sequence identified in WASH-1400.[16] The consequences were obtained using the CRAC Code.[64] An average population of 340 persons per square mile (which is the average for U.S. domestic sites) was assumed from an exclusion area one-half mile about the reactor to a 50-mile radius about the reactor. Typical midwest site meteorology was assumed. Based on these assumptions, a release produces an exposure of 5×10^{6} man-rem. With upper and lower bound frequencies of 4×10^{-6} and 4×10^{-12} core-melt/RY, the upper and lower values of risk exposure were 20 man-rem/RY and 2×10^{-5} man-rem/RY, respectively. Based upon an average remaining life of 24 years for 24 BWRs having a HPCI system with open isolation valves, the risk posed by this issue has an upper bound of 11,500 man-rem and a lower bound of 1.1×10^{-2} man-rem. The consequences of the RWCU line break sequence would be 70 man-rem/RY and

40,000 man-rem total. Thus the maximum risk reduction associated with this issue was estimated to be 51,500 man-rem.

Cost Estimate

Industry Cost: Implementation of the proposed change to leave the outboard isolation valve closed was estimated to be 2.5 man-years. This included: (1) an engineering review of the logic for HPCI initiation to assure that the valve will be commanded open and will properly isolate if required; (2) preparation of changes to procedures (normal and emergency); (3) revision to operator training covering the change; (4) revision to the FSAR; (5) license amendments; and (6) hardware changes. No added maintenance costs were anticipated. No hardware costs were assessed to add a bypass line because it was believed that most reactors already had this feature. At an average cost of \$100,000/man-year, the total industry cost was estimated to be \$6.75M.

NRC Cost: The NRC cost was estimated to be 1 man-month/reactor or \$210,000 for all reactors. However, there was at least one reported instance in which the isolation valve could not be opened under pressure; this occurrence was reported in AEOD/T420.[825] If these valves would have to be modified to open under pressure, the costs would be much greater.

Performing qualification tests on a selected sample of RWCU isolation valves and actuators and demonstrating, by analyses, that the other valves and actuator combinations will perform satisfactorily were estimated to cost \$1M. If actuators have to be replaced, this would add to the costs.

Total Cost: The total industry and NRC cost associated with the possible solution was estimated to be \$7.96M.

Value/Impact Assessment

Based on a potential risk reduction of 51,500 man-rem and a cost of \$7.96M, the value/impact score was given by:

$$S = \frac{51{,}500 \text{ man-rem}}{\$7.96M}$$

$$\cong 6{,}500 \text{ man-rem/\$M}$$

Other Considerations

The occurrence of the analyzed event would result in the loss of one defense layer (containment). Other considerations, which in individual cases may reduce the risk associated with this issue, include the absence of ventilation openings or open doors between the torus compartment and the pump rooms. The absence of these openings reduces the common cause failure potential of the RHR/LPCI, RCIC, and core spray systems with the HPCI steam supply line break. Consideration should be given to reducing the risk if the isolation valves were selected based on the requirement to close under line break/steam mass flow conditions. This concern could be eliminated if it could be shown by test or from actual application that valve operation was verified under loads equivalent to line break conditions.

A similar situation exists for the RCIC system. Since the RCIC steam line is smaller than the HPCI line, the risk may not be as great but would still add substantially to the values estimated previously.

CONCLUSION

Based on both the RWCU and HPCI event sequences and the Event V consequences, this issue was given a high priority ranking. In resolving the issue, the staff conducted a two-phase valve test program: Phase 1 was reported in NUREG/CR-5406[1403] and Phase 2 was reported in NUREG/CR-5558.[1404] In addition, laboratory tests of DC-powered MOVs were conducted and reported in NUREG/CR-5720.

In general, it was found that many of the valves of concern in the issue did not have sufficient margin to close under the blowdown loads that would be encountered under the design basis conditions caused by a pipe break. The primary reasons for this were: (1) at the time the valves and operators were sized, the internal mechanisms and load paths of the MOVs were not well understood; and (2) the standard equation used by the industry to predict MOV stem loads does not adequately account for all of the force components resulting from the interaction of the blowdown flows on the valve internal parts.

The results of the Phase 1 test program were factored into the development of Generic Letter No. 89-10,[1217] thus providing licensees with the best guidance available at that time regarding how they should assure that their MOVs would perform their design basis function. The results of the Phase 2 test program were used in the development of Supplement 3 to Generic Letter No. 89-10,[1217] which provided licensees with further guidance. The staff also conducted training for inspectors and provided computer software to aid in identifying these problems on site. Thus, this issue was RESOLVED[1406] and requirements were issued to licensees in Generic Letter No. 89-10.[1217] The related ACRS concern for the design basis for valves that might be subjected to significant blowdown loads will be addressed in Issue 152.

REFERENCES

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

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986.

367. NUREG/CR-2802, "Interim Reliability Evaluation Program: Analysis of the Browns Ferry Unit 1 Nuclear Plant," U.S. Nuclear Regulatory Commission, August 1982, (Appendix A) August 1982, (Appendix B) August 1982, (Appendix C) August 1982.

824. Memorandum for T. Speis from R. Mattson, "Request for Prioritization of Generic Safety Issue - Failure of HPCI Steam Line Without Isolation," October 18, 1983.

825. Memorandum for K. Seyfrit from P. Lam, "Failure of an Isolation Valve of the Reactor Core Isolation Cooling System to Open Against Operating Reactor Pressure," August 23, 1984.

826. Letter to A. Schwencer (NRC) from J. Kemper (Philadelphia Electric Company), "Limerick Generating Station, Units 1 & 2, Request for Additional Information from NRC Equipment Qualification Branch (EQB)," February 27, 1984

827. NEDO-24708A, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," General Electric Company, December 1980.

828. NUREG/CR-3933, "Risk Related Reliability Requirements for BWR Safety-Important Systems with Emphasis on the Residual Heat Removal System," U.S. Nuclear Regulatory Commission, August 1984.

829. BNL A-3740, "An Evaluation of Unisolated LOCA Outside the Drywell in the Shoreham Nuclear Power Station," Brookhaven National Laboratory, June 1985.

830. Memorandum for W. Minners from A. Thadani, "Comments on Generic Issue No. 87 - Failure of HPCI Steam Line Without Isolation," June 28, 1985.

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

1403. NUREG/CR-5406, "BWR Reactor Water Cleanup System Flexible Wedge Gate Isolation Valve Qualification and High Energy Flow Interruption Test," U.S. Nuclear Regulatory Commission, (Volume 1) October 1989, (Volume 2) October 1989, (Volume 3) October 1989.

1404. NUREG/CR-5558, "Generic Issue 87: Flexible Wedge Gate Valve Test Program," U.S. Nuclear Regulatory Commission, January 1991.

1406. Memorandum for J. Taylor from E. Beckjord, "Technical Resolution of Generic Issue 87, 'Failure of HPCI Steam Line Without Isolation,'" December 9, 1991.

ISSUE 100: ONCE-THROUGH STEAM GENERATOR LEVEL

DESCRIPTION

Historical Background

Once-through steam generators (OTSGs) are a feature unique to B&W reactor designs. Main feedwater is injected from a header, located at approximately mid-elevation of the OTSG, into an annular downcomer region. As the feedwater is sprayed into the downcomer, the condensing action of the relatively cold (455°F) feedwater draws steam from the tube bundle through the aspirator ports in the inner shroud. This steam then heats the feedwater rapidly to saturation temperature (about 535°F) preventing thermal shocking of the shell.

In 1984, Crystal River 3 plant submitted a TS change request to raise the operating water level limit of the OTSGs to 100% of the operating range, which is six inches above the aspirator ports. Since most operating B&W plants do not have an upper OTSG level limit in their TS, the TS change was requested to give Crystal River 3 the same operational flexibility. This change request raised a concern regarding operating OTSGs with the water level above the aspirator ports.[1380]

Safety Significance

Permitting operation with a higher water level limit would allow less time for corrective operator action, if there were a transient that involved an increase in feedwater flow to one or more steam generators. If the increased feedwater flow continues (e.g., if the steam generator high-level detection system has failed, feedwater control valves have failed, or the main feedwater pump fails to trip), the steam generator water level may exceed the aspirator port level, thus preventing the preheating of the feedwater. If the increased thermal stresses on the tubes or shell wall are excessive, an SGTR or steam generator shell failure accident could occur creating a LOCA or an initial steam-side break in the case of a shell failure. This issue affects all B&W PWRs.

Possible Solution

The possible solution entails a detailed generic analysis to determine whether operating OTSGs at levels near or above the aspirator ports will introduce a significant safety problem at B&W plants. New plant-specific TS limits would have to be developed to preclude plant operations if OTSG water levels exceed a pre-set maximum level.

PRIORITY DETERMINATION

Assumptions

The reference plant selected for analysis was Oconee 3 because it is a representative B&W PWR. All ten of the operating or proposed B&W PWRs were considered in this analysis; TMI-2 was not included because it was shut down indefinitely. The average remaining operating life of the 10 B&W plants considered was 28.2 years, based on the original 40-year license. However, it was assumed that 75% of these plants will have their licenses extended for an additional 20 years and, therefore, the total remaining operating life of these plants was assumed to be 432 RY.

Frequency Estimate

The initiating transient of concern involves an increase in feedwater flow to one or more loops. If the increased feedwater flow continues, e.g., if the steam generator high-level detection system fails, feedwater control valves fail, or the main feedwater pump fails to trip, the steam generator water level may exceed the aspirator port level. The resultant backflow of feedwater through the aspirator ports may then result in a SGTR or steam generator shell failure accident. The accident sequence developed by PNL[64] to model the effects of the proposed solution produced the following results:

(1) The average frequency of increased feedwater flow for B&W plants was developed in Section 1.0 of NUREG/CR-3862[1186] to be 0.13 transient/RY

(2) The probability of failure on demand to reduce main feedwater flow (product of undetected failure of steam generator high-level trip and operator failure to terminate the overfeed event) is (0.047)(0.7) = 0.033

(3) The probability of an SGTR, given an overfill event, is 0.027

(4) The probability of steam generator shell failure (SGSF), given an overfill event, is 0.027

(5) The sum of the conditional probabilities of SGTR and SGSF is 0.054

(6) The probability of a failure on demand to mitigate the SGTR or SGSF (estimated from core-melt frequency from SGTR sequences divided by the SGTR initiating event frequency) is $(2.7 \times 10^{-6})/(8.6 \times 10^{-3}) = 3.14 \times 10^{-4}$

(7) The resulting base case accident sequence frequency was estimated to be $(0.13/RY)(0.033)(0.054)(3.14 \times 10^{-4}) = 7.27 \times 10^{-8}/RY$

The effects of an enhanced testing and inspection program for steam generator level instrumentation and feedwater controls was assumed to reduce the conditional probablity of failure to reduce main feedwater flow, given a feedwater overfeed event, to 0.011/demand. Therefore, the adjusted case accident frequency is $[(0.011)/(0.033](7.27 \times 10^{-8})/RY = 2.43 \times 10^{-8}/RY$. Thus, the reduction in core-melt frequency was estimated to be $(7.27 \times 10^{-8})/RY - (2.43 \times 10^{-8})/RY = 4.8 \times 10^{-8}/RY$.

Consequence Estimate

Containment failure probabilities and corresponding dose consequences were then combined with the accident frequencies to calculate public risk. The base case and adjusted case risk was calculated by PNL[64] to be 0.2 and 0.066 man-rem/RY, respectively. Based on a total operating life of 432 RY, the potential risk reduction for all affected plants was determined to be 56 man-rem.

Cost Estimate

Industry Cost: Licensees will have to prepare safety analyses to support revising plant-specific TS. The total cost for these safety analyses and TS preparation was estimated to be $4.1M; operation and maintenance costs associated with increased inspection and testing were estimated to be $4.3M. Thus, the total industry cost associated with the possible solution was estimated to be $8.4M.

NRC Cost: The total cost for development of a solution, support of implementation, and review of operation and maintenance was estimated to be $320,000, $91,000, and $490,000, respectively. Thus, the total NRC cost for the possible solution was estimated to be $0.9M.

Total Cost: The total industry and NRC cost associated with the possible solution was estimated to be $9.3M.

Value/Impact Assessment

Based on a potential risk reduction of 56 man-rem and a total cost of $9.3M for a possible solution, the value/impact score was given by:

$$S = \frac{56 \text{ man-rem}}{\$9.3\text{M}}$$

$$\approx 6 \text{ man-rem/\$M}$$

Other Considerations

The central concern in this issue is that operating OTSGs at a high water level could allow feedwater back through the aspirator ports onto steam generator tubes potentially affecting tube integrity. The effects on steam generator tube integrity from this particular event, however, are within the steam generator design bases. A partial list of these design bases is summarized below:

- 15,000 cycles of adding 40°F feedwater at 875 gpm when at hot standby conditions (normal condition)
- 500 cycles of adding 40°F feedwater at 875 gpm during loading conditions (normal condition)
- 500 cycles of adding 100°F feedwater at 875 gpm during loading conditions (normal condition)
- 7 cycles of adding 40°F feedwater at 1750 gpm during a steam line break (faulted condition)
- 280 cycles of adding 40°F feedwater at 1750 gpm with the flow initiated 30 seconds after a loss of main feedwater (faulted condition).

Given that main feedwater is normally about 465°F during loading conditions, the thermal effects of adding 40°F emergency feedwater would be substantially greater than those of spilling main feedwater through the aspirator ports onto the tubes. In addition, the aspirator ports are located near the middle of the tube sheet, which is a less stressed position than the location of the emergency feedwater nozzle, which is higher in the OTSG. This further indicates that the steam generator tubes are likely to withstand introducing 455°F main feedwater through the aspirator ports and onto the tubes.

CONCLUSION

OTSGs are designed to withstand over 15,000 cycles of injection of 40°F emergency feedwater. The consequences of operating with a water level above the aspirator ports (which would introduce 455°F water) are less severe than that assumed in this analysis and are within the OTSG design limits. The possible solution does not produce a significant reduction in public risk and the value/impact score is small. Therefore, this issue was DROPPED from further pursuit.

REFERENCES

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986.

1186. NUREG/CR-3862, "Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments," U.S. Nuclear Regulatory Commission, May 1985.

1380. Memorandum for W. Minners from B. Sheron, "Request for Prioritization of Potential Generic Issues," September 4, 1984.

ISSUE 123: DEFICIENCIES IN THE REGULATIONS GOVERNING DBA AND FAILURE CRITERION SUGGESTED BY THE DAVIS-BESSE INCIDENT OF JUNE 9, 1985

DESCRIPTION

Following the Davis-Besse event in June 1985, potential inadequacies of nuclear power plant design criteria and safety analyses were raised in a DST/NRR memorandum.[1323] These concerns were: (1) root causes of DBAs are not analyzed and may initiate or exacerbate a plant transient involving an initiating event; (2) allowable outage times (AOTs) and limiting conditions of operation (LCOs) may be inadequate since they are derived from potentially flawed DBA analyses; (3) high-probability common cause failures are not adequately addressed in licensing requirements; and (4) single human errors that may have a broader effect than active failures are not covered.

(1) Root Causes of DBAs: This concern addressed the possibility that auxiliary or support system failures may cause a plant transient or initiating event, as well as result in failure of one or more safety systems to perform their intended function. This concern was addressed, in part, in the resolution of Issues A-17, A-44, and A-45.

Issue A-17 specifically addressed: (1) events involving transients and loss of at least one redundant portion of any one of the systems required to respond to a transient; (2) initiating events and similar failures of redundant safety systems; and (3) degradation of safety systems by non-safety systems, as well as degradation of auxiliary support systems such as SSW, CCW, and AC/DC power. The results of studies[1232] indicated that the causes and effects of systems interactions were plant-specific in nature due to the differences in plant designs. In addition, it was demonstrated that plant-specific PRAs were effective tools for identifying vulnerabilities to systems interactions. Currently, licensees are required to perform IPEs that include an evaluation of common cause (dependent) failures, of which, systems interactions are a subset.[1222] The information and insights gained from the Issue A-17 studies have been provided to licensees to assist in the identification and evaluation of system interactions and other common cause failures. Licensees are expected to propose plant-specific procedure and/or hardware modifications, where appropriate, to reduce their vulnerabilities to such events. Consequently, vulnerabilities to the root causes of DBAs are being systematically identified and corrected, as determined by licensees, on a plant-specific basis in the IPE process.

Issue A-44 addressed the likelihood and duration of losses of offsite power, the redundancy and reliability of onsite emergency AC power sources (e.g., diesel generators), and the effects on plant risk of failures of all AC power sources. Support system failures were important aspects of these analyses, particularly DC instrumentation and control power supplies, instrument air supplies, and auxiliary cooling systems such as SSW and CCW. Resolution of this concern involved improving the reliability of onsite AC power systems and strengthening each plant's capability to cope with an extended loss of AC power.

Issue A-45 addressed potential improvements in the reliability of shutdown decay heat removal systems that are required to operate after a transient or initiating event and included support system failures and single-point vulnerabilities.

It was concluded in NUREG-1289[1326] that resolution of Issue A-45 could only be achieved on a plant-specific basis; this is being implemented as part of the IPE process. Licensees were directed to identify decay heat removal vulnerabilities and to determine if cost-effective solutions to these vulnerabilities could be achieved.

In addition to this concern being addressed in part by resolution of Issues A-17, A-44, and A-45, IPEs currently underway by each licensee are expected to search for vulnerabilities stemming from support system failures. It is expected that these IPEs, when completed, will contain dependency tables (or other similar methods of displaying dependencies) that identify dependencies between initiating events and mitigating functions or systems. It is also expected that licensees will move expeditiously to correct any identified vulnerabilities that warrant correction in accordance with Generic Letter 88-20.[1222]

(2) AOTs and LCOs: This concern relates to the fact that AOTs and LCOs may be derived from the results of DBA analyses; if the DBA analyses are inadequate, then the AOTs and LCOs may also be inadequate. Since it is not uncommon for a plant to have several components out of service at the same time, the potential exists for operation of a plant in a dangerous configuration in which two or more components that appear in the same accident sequence are out of service. The concern focused on outages for diverse components that are not necessarily in the same safety system, such as simultaneous outage of valves in the high pressure injection system (HPIS) and the low-pressure injection system (LPIS).

This concern deals with AOTs for components and the possibility that a plant may be operating one active component failure away from core damage. A large fraction of the potential core damage probability and public risk reduction associated with this concern would be associated with removing vulnerabilities associated with the component outages. AOTs and LCOs were addressed in Issue 117 where the approach to evaluating the change in core damage probability was to remove the test/maintenance unavailability from basic events in each cut set that contained multiple test/maintenance outage terms. This analysis assumed a scenario which precluded the possibility that a plant could be operating at full power with vital equipment in different ESF systems down for maintenance, and effectively removed the vulnerabilities associated with AOTs and LCOs on components in different ESF systems and in redundant divisions of each ESF system. Issue 117 was not pursued separately because its safety concern was addressed as part of the staff's Technical Specification Improvement Program (TSIP). In addition, as part of the Implementation of the Maintenance Rule,[1338] licensees should make an assessment of the total plant equipment that is out of service during power operation. This assessment is to ensure that the objective of preventing failures by performing maintenance is appropriately balanced against the objective of minimizing unavailability.

(3) High Probability Common Cause Failures: Issue A-17 addressed, among other things, the potential for common cause events involving systems/components that share physical connections or spatial configurations, or could cause operator errors that may result from operation disinformation or inhibition of an operator's ability to respond to a malfunction.[1233] An example that was addressed in the Issue A-17 analyses was a high-energy line break and the possibility that adverse environmental conditions resulting from such an event could induce failures in one or more safety systems designed to respond to the event. This is an example of the spatially-coupled system interaction. Other examples include

seismic events, fires, and floods that could affect the operability of equipment/systems located in close proximity to each other, interactions between normal offsite and emergency onsite AC power systems, (e.g., sharing common breakers or power distribution buses), and common support systems, cross-connects, and other functional dependencies.

As discussed previously, the staff concluded that plant-specific analyses were necessary to accurately identify, evaluate, and resolve (where appropriate) vulnerabilities to systems interactions. The plant-specific IPE program[1222] includes an assessment of common cause or dependent failures. Since systems interactions are a subset of common cause failures, this concern is covered in the performance of the IPEs.

Issue A-47 also addressed aspects of this concern, including single failures or multiple failures which could cause a malfunction in one or more control systems. Such malfunctions may result in an undesirable control system response or provide misleading information to an operator. The analyses[1248] in support of the resolution of Issue A-47 identified potential control system failures that could cause overpressure, overcooling, overheating, overfilling, or reactivity events. All of these events are covered in DBA analyses. Requirements were established that, in general, provide or enhance systems to protect against reactor vessel/steam generator overfill events and to prevent steam generator dryout, enhance procedures and provisions to verify the operability of these systems, and modify selected procedures to respond to small-break LOCAs. This concern is considered to be resolved.

(4) <u>Single Human Errors:</u> This concern relates to the possibility that a single human error could potentially result in a plant transient or initiating event and defeat one or more divisions of a safety system. No events of this type have been identified in plant operating experience, although the Davis-Besse incident was one that involved two human errors and a flawed Steam and Feedwater Rupture Control System. Therefore, it appears unlikely that significant vulnerabilities to single human errors exist in the industry.

Issues A-17, A-44, A-45, and A-47 addressed various aspects of this concern as contributors to system failures, including degradation of operator information that could lead to operator "blindness," incorrect operator actions, and human errors. The analyses performed in support of these issues considered, for the most part, the possibility that single operator action could defeat one or more divisions of an ESF system. In addition, in situations where operator actions are necessary but the integrity of the information in the control room may be questionable (such as following a station blackout), it was assumed that the operator would not respond correctly. This effectively addresses single human errors that may defeat an ESF that otherwise would be operable.

Single human errors may also initiate a plant transient. Instances can be found in LERs in which single human errors have resulted in plant shutdowns, such as maintenance errors during electrical switchgear work that result in main feedwater isolation or interruption of vital AC power sources. Maintenance errors on the non-nuclear side of a plant that resulted in turbine-generator trips have also occurred. However, to date, such failures have not resulted in the occurrence of a transient and simultaneous failure of ESF systems that are designed to respond to the transient. This is primarily because of the redundancy and diversity of plant systems, particularly ESF systems, that are designed to minimize the effects of single failures by maintaining separation of different divisions of vital plant equipment. This conclusion was supported by the results of a number of recent PRAs, including NSAC-60[889] and the PRAs prepared in support of NUREG-1150,[1081] in which no accident sequences initiated by single human errors were found to contribute significantly to core damage probability.

CONCLUSION

Since this issue was raised, all the safety concerns have been or will be adequately addressed in the resolution of Issues A-17, A-44, A-45, and A-47, the evaluation of Issue 117, the IPE program, and the Maintenance Rule. Thus, this issue was DROPPED as a new and separate issue.

REFERENCES

889. NSAC-60, "A Probabilistic Risk Assessment of Oconee Unit 3," Electric Power Research Institute, June 1984.

1081. NUREG-1150, "Reactor Risk Reference Document," U.S. Nuclear Regulatory Commission, December 1990.

1222. NRC Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR § 50.54(f), (Generic Letter No. 88-20)," November 23, 1988, (Supplement 1) August 29, 1989, (Supplement 2) April 4, 1990.

1232. NUREG-1174, "Evaluation of Systems Interactions in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, May 1989.

1233. NUREG-1229, "Regulatory Analysis for Resolution for USI A-17," U.S. Nuclear Regulatory Commission, August 1989.

1248. NUREG-1218, "Regulatory Analysis for Resolution of USI A-47," U.S. Nuclear Regulatory Commission, July 1989.

1323. Memorandum for W. Minners, et al., from F. Rowsome, "Generic Issue 123, 'Deficiencies in the Regulations Suggested by the Davis-Besse Incident,'" November 21, 1985.

1326. NUREG-1289, "Regulatory and Backfit Analysis: Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements," U.S. Nuclear Regulatory Commission, November 1988.

1338. <u>Federal Register</u> Notice 56 FR 31306, "10 CFR Part 50, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," July 10, 1991.

ISSUE 128: ELECTRICAL POWER RELIABILITY

DESCRIPTION

Following the NRR reorganization in November 1985, DSRO/EIB was responsible for resolving three issues that were directly related to onsite electrical systems: Issue 48, "LCO for Class 1E Vital Instrument Buses in Operating Reactors"; Issue 49, "Interlocks and LCOs for Class 1E Tie Breakers"; and Issue A-30, "Adequacy of Safety-Related DC Power Supplies." In an effort to provide a more integrated approach to resolving these three issues, DSRO formulated a program to combine the issues into one generic issue.[1001] In addressing this issue, the staff was to consider NRC endorsement of IEEE Standards 603 and 308 with possible revisions to the related Regulatory Guides. Any revisions to Regulatory Guides resulting from this issue could impact future plants.

Issue A-30 was initiated because of concerns regarding the reliability of nuclear plant DC battery systems and the ability of plants to safely shut down in the event of a common mode failure or multiple failures of redundant systems. This concern resulted in a study of the safety-related DC power supplies at operating nuclear power plants. The results of this study were presented in NUREG-0305[163] which recommended that a quantitative reliability assessment of DC power systems be performed to identify and provide a basis for any changes in licensing criteria.

A reliability assessment was performed and documented in NUREG-0666.[164] The dominant failure modes identified in this study involved: (1) the inability of batteries to provide sufficient power to Class 1E instrumentation and controls buses upon loss of AC power to the battery chargers; and (2) operational, test, or maintenance errors that could result in the loss of multiple DC divisions. Monitoring provisions and procedures for preventing these occurrences were developed by the staff and presented in NUREG reports, Information Notices, Bulletins, and STS. Two of these provisions were adopted in industry standards: IEEE-450, "IEEE Recommended Practice for Maintenance, Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Substations," and IEEE-946 "Recommended Practices for the Design of Safety-Related DC Auxiliary Power Systems for Nuclear Power Generating Stations."

Issue 48 was identified when it was found that some nuclear power plants lacked administrative controls or TS governing operational restrictions for their Class 1E 120V AC Vital Instrument Buses. These restrictions are required to ensure compliance with GDC 17, 21, 34, and 35 of 10 CFR 50, Appendix A. During repair or maintenance activities on bus power sources or inverters, one or more of the normal or alternate vital instrument bus power sources could be removed from service indefinitely. This condition could lead to the loss of more than one vital instrument bus in the event of a single failure or loss of offsite power.

Issue 49 was identified as a result of an incident at the Point Beach Nuclear Plant. The licensee reported to the NRC that a manually-operated tie-breaker between redundant safety buses, which had been closed during a plant outage in order to facilitate maintenance without interrupting power to affected systems, remained closed for a period of 5 weeks after the plant returned to operation. In the event of a loss of normal AC power, the diesel generator output breakers would have been prevented from closing, as a result of the tie-breakers being closed.

CONCLUSION

This issue was given a high priority ranking and resolution was pursued based on the separate evaluations of Issues 48, 49, and A-30.

To address Issue A-30, Generic Letter No. 91-06[1399] was issued to request licensees to respond to 9 questions that were developed to facilitate staff determination of licensee implementation of existing recommendations. These recommendations included provisions for monitoring DC systems, test procedures, and operating procedures. The recommendations were identified from a number of previous and ongoing actions including industry standards, INPO recommendations, STS, and existing licensing practices. In the generic letter, the option was provided for licensees to supply information as part of the IPE. The actions described in the letter were not considered to constitute a backfit but only involved information gathering, in accordance with 10 CFR 50.54(f). Follow-up NRC actions, if necessary, would be pursued on an individual plant basis.

Issues 48 and 49 were addressed in Generic Letter No. 91-11,[1400] issued to licensees to certify that they either have implemented TS or administrative controls conforming to the guidelines in the letter, or to justify why such controls may not be required. Any modification (e.g., procedural changes) performed to complete implementation would be considered a backfit to be performed as a compliance matter. This precluded evaluation of the need for controls as part of the IPE as an acceptable alternative to responding to Generic Letter No. 91-11.[1400] However, the option to further evaluate certain aspects (such as the optimum length of time for allowing outage of equipment) as part of the IPE was still provided.

The staff's technical findings were published in NUREG/CR-5414.[1401] Thus, this issue was RESOLVED and requirements were issued.[1402]

REFERENCES

163. NUREG-0305, "Technical Report on DC Power Supplies in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1977.

164. NUREG-0666, "A Probabilistic Safety Analysis of DC Power Supply Requirements for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, April 1981.

1001. Memorandum for H. Denton from T. Speis, "Integration of Electrical Power Issues into Proposed Generic Issue 128, 'Electrical Power Reliability,'" November 28, 1986.

1399. NRC Letter to All Holders of Operating Licensees, "Resolution of Generic Issue A-30, 'Adequacy of Safety-Related DC Power Supplies,' Pursuant to 10 CFR 50.54(f) (Generic Letter 91-06)," April 29, 1991.

1400. NRC Letter to All Holders of Operating Licenses, "Resolution of Generic Issues 48, 'LCOs for Class 1E Vital Instrument Buses,' and 49, "Interlocks and LCOs for Class 1E Tie Breakers' Pursuant to 10 CFR 50.54(f) (Generic Letter 91-11)," July 18, 1991.

1401. NUREG/CR-5414, "Technical Findings for Proposed Integrated Resolution of Generic Issue 128, Electric Power Reliability," U.S. Nuclear Regulatory Commission, November 1989.

1402. Memorandum for J. Taylor from E. Beckjord, "Resolution of GI-128, 'Electrical Power Reliability,'" September 12, 1991.

ISSUE 130: ESSENTIAL SERVICE WATER PUMP FAILURES AT MULTIPLANT SITES

DESCRIPTION

Historical Background

This issue was identified[958] when the staff found the Byron Unit 1 vulnerable to core-melt sequences in the absence of the availability of Byron Unit 2 which was not yet operational. Because of the licensing status of the multiplant configuration of Byron Units 1 and 2, and the more immediate need to make a third service water pump available to Byron Unit 1 via a crosstie with one of the two Byron Unit 2 essential service water (ESW) pumps, the Byron Unit 1 concern was classified as a plant-specific (not generic) issue. However, this plant-specific issue raised concerns for multiplant sites that have only two ESW pumps/plant with crosstie capabilities. The future operation of Byron Unit 2 would place both Byron units in this limited group of plants with multiplant configurations.

A limited survey[958] of W plants was conducted to help identify the generic applicability of multiplant configuration vulnerabilities with only 2 ESW pumps/plant. In the multiplant configuration identified (approximately 16 plants), all plants can share ESW pumps via crosstie between plants. It was stated[958] that B&W and CE plants would be surveyed to identify if similar multiplant configurations with 2 ESW pumps/plant and crosstie capabilities exist in the other NSSS vendors' designs. Based on the staff's limited survey, this issue had the potential to affect at least 16 PWR plants. A survey was recommended for single-unit plants to identify if similar ESW vulnerabilities existed.

Safety Significance

All ESW systems are front-line (supporting) safety systems. The design of the ESW support systems are highly plant-specific with plant-specfic equipment, crosstie capability, and ESW operability and functionability needs for successful (accident mitigation) operations. Because of the variability between ESW systems of different plant configurations, approximate generic modeling of the success criteria for the multiplant configurations with 2 ESW pumps/plant with crosstie capabilities was used to scope the safety significance of this issue. The assumed success criteria and systemic events leading to core-melt are discussed below.

The core-melt and radiological risk (consequences) determined by this evaluation pertain only to the generic model multiplant configuration with 2 ESW pumps/plant. However, as discussed herein, other plant configurations may also contain similar ESW system vulnerabilities.

Should the front-line ESW systems fail to provide adequate cooling capability to shut down a plant when subject to a loss of ESW, a core-melt accident could result in significant risk to the public.

Possible Solutions

The possible solutions to reduce the public risk from a loss of the ESW system were: (1) provide a third ESW pump/plant; (2) provide an additional swing pump that is shared between units: and (3) modify TS governing the LCO for the ESW pumps.

PRIORITY DETERMINATION

The service water cooling system is used to remove heat from essential and non-essential equipment. Under accident conditions, the non-essential heat loads are isolated and the ESW system provides cooling only to essential equipment for plant cooldown and post-accident operations. At multiplant sites, the ESW systems for each plant are crosstied with double isolation valves that are normally closed.

ESW Success Criteria: The success criteria for the ESW systems in providing adequate cooling capability during normal, accident, and post-accident conditions are plant/design specific. The ESW vulnerabilities will depend on the plant configurations, numbers and the capacities of the ESW pumps, and equipment ESW cooling dependencies. Because the success criteria may be as varied as the ESW systems, this generic evaluation assumed the following success criteria as a representative model for purposes of quantifying the systemic events leading to possible core-melt accidents. The generic criteria may apply only to multi-plant sites having 2 ESW pumps/plant with crosstie capabilities.

During normal operations, one ESW pump/plant provides adequate cooling to systems such as CCW, RCP motor coolers, and air-conditioning and ventilation systems. The second ESW pump/plant is assumed to be normally in a standby mode. Because of load shedding (isolation of non-essential equipment), one ESW pump/plant was assumed to be capable of handling the accident and cooldown heat loads. Typical equipment cooled by the ESW under these conditions are the CCW heat exchangers, containment spray heat exchangers, diesel generators, and auxiliary building ventilation coolers. With one plant in normal operation and the second plant already in the shutdown or refueling modes of operation, the criteria assume one ESW pump can provide adequate cooling to shutdown the operating plant through the crosstie connections, should the need arise.

Initiating Transient Event: The initiating events leading to core-melt assume the following: One plant "A" ESW pump (P_1) fails and the second ESW pump (P_2) is out of service during a TS allowed outage time (AOT) of 72 hours. The failure frequency of P_1 was estimated at approximately 10^{-1}/RY.[959] The unavailability of P_2 (normally in standby) from the AOT was approximately 10^{-2}/RY. Therefore, the initiating event that originates from plant A (T_a), due to the loss of service water in plant A, had a frequency of 10^{-3}/RY.

Plant B may be in operation or in the shutdown or refueling mode of operation. If a 0.7 capacity factor was assumed for both plants, the probability that both plants would be operating at the same time was 0.5 (product of capacity factors). Conversely, the probability that one plant is operating and the other plant is shutdown was also 0.5. Absent any TS requirements on the Plant B ESW pumps during shutdown or refueling modes, the status of Plant B ESW pumps (P_3,P_4) was uncertain. Therefore, as shown below, the unavailability (W_i) to meet the success criteria (N) is the product of the status mode probability and the conditional failure probability, given the status mode of the ESW pumps.

T_a (Initiating Events Frequency)	U (Plant B Status)	N (Number of Plant B Pumps Required)	ESW PUMPS (Status Mode)	ESW PUMPS (Unavailability)	W_i (Unavailability of N)
10^{-3}	Operating	2	P_3=R		
			P_4=AOT	$(10^{-2})(1.0)$	$W_1=10^{-2}$
	U_{bo}=0.5	2	P_3=R		
			P_4=SB	$(0.98)(7\times10^{-3})$	$W_2=(7\times10^{-3})$
10^{-3}	Shutdown	1	P_3=M	$(0.25)(1.0)$	$W_3=(0.25)$
			P_4=M		
	U_{br}=0.5	1	P_3=M		
			P_4=SB	$(0.25)(7\times10^{-3})$	$W_4=(2\times10^{-3})$
			P_3=R	-	-
			P_4=SB		
			P_3=R	-	-
			P_4=M		

AOT - Allowed Outage Time
M - Maintenance
R - Running
SB - Standby

Loss Of Service Water Transient Event Sequences: This section describes the loss of service water events for a two-unit multiplant configuration with 2 ESW pumps/plant, given the loss of service water initiating transient (T_a) in Plant A discussed earlier.

The control room operator is expected to trip the Plant A reactor and initiate local recovery actions to open the ESW crossties between Plant A and Plant B. After the Plant A reactor trip, the auxiliary feedwater system (L) would be demanded. If Plant B ESW pumps are available and the ESW is recovered by valve realignments (X, crosstie), it was assumed that the reactor (Plant A) can be cooled by steam generators using "L". If "L" is not successful (failure on demand), the operator would initiate HPI and cool the reactor by feed-and-bleed. Recovery of service water via "X" would also restore cooling to the CCW heat exchangers that cool the HPI pumps and other essential equipment

If the Plant B ESW pumps are available and ESW recovery by "X" is not made, the Plant A RCP seals may fail (S) due to loss of seal injection (charging pumps) cooling and RCP thermal barrier cooling (CCW). The RCP seal failure results in a LOCA. The ECCS pumps were assumed to fail because of lack of CCW heat exchanger cooling by the service water, resulting in a core-melt event.

If "L" fails on demand, the operator would initiate the HPI pumps and attempt to cool the reactor by feed-and-bleed. However, the HPI pumps, as described earlier, indirectly require ESW cooling and are assumed to fail. If L is successful, the pressure relief valves (if required) could either fail to open (P) and relieve the reactor pressure (overpressure failure of reactor), or fail to close (Q), given that they opened (LOCA). Given a LOCA, the HPI pumps were assumed to fail because the service water cooling to the CCW heat exchangers, which cool the HPI pumps, was not available.

If Plant B ESW pumps are not available (W_1) due to extended maintenance outage (M) or failure to start and run from a standby condition (SB), it was assumed that recovery of the ESW pumps cannot be obtained in sufficient time to preclude core-melt. In these cases, a successful crosstie (X) is not effective in reducing core melt.

The cut sets (systemic event sequences) for the above loss of service water transient in Plant A (T_a) were:

(1) Plant B Operating (U_{bo})

	$X(L+P+Q)$			3×10^{-4}		1.5×10^{-7}
T_aU_{bo}	XS	=	(5×10^{-4})	3×10^{-4}	=	1.5×10^{-7}
	(W_1+W_2)			2×10^{-2}		1.0×10^{-5}

(2) Plant B in Shutdown or Refueling (U_{br})

	$X(L+P+Q)$			3×10^{-4}		1.5×10^{-7}
T_aU_{br}	XS	=	(5×10^{-4})	3×10^{-4}	=	1.5×10^{-7}
	(W_3+W_4)			2.5×10^{-1}		1.3×10^{-4}

The base case frequencies and probabilities for the cut sets shown above were:

$T_a = 10^{-3}/RY$	$W_3 = 2.5 \times 10^{-1}$
$U_{bo} = 5 \times 10^{-1}$	$W_4 = 2 \times 10^{-3}$
$U_{br} = 5 \times 10^{-1}$	$X = 3 \times 10^{-2}$
$W_1 = 10^{-2}$	$S = 10^{-2}$
$W_2 = 7 \times 10^{-3}$	

$L = 10^{-2}$ to 10^{-5}, depending on plant-specific design and ESW cooling needs

$P = (10^{-3}/\text{demand})(10^{-1}\ \text{demand}/L) = 10^{-4}$

$Q = (10^{-2}/\text{demand})(10^{-1}\ \text{demand}/L) = 10^{-3}$

Frequency Estimate

Based on the success criteria and examination of the above base case core-melt frequency estimates, a dominant core-melt frequency of approximately 10^{-4}/RY for the multiplant units with 2 ESW pumps/plant can occur with one plant operating and the other plant shut down (refueling).

Based on engineering judgment, at least one of the ESW pumps in the shutdown plant should be kept running. In addition, the RHR and diesel generator TS operability requirements for Modes 5 and 6 indicated (indirectly) that the ESW pumps should be operable in Modes 5 and 6. However, by possible valving alignments (plant-specific), the RHR system and diesel generators could be cooled by the adjoining operating plant's ESW pumps. Therefore, lacking specific operability requirements on the ESW pumps when the plant is in Modes 5 or 6, the operability of the shutdown plant's ESW pumps was not assured. If only one of the two ESW pumps is out for maintenance and the other pump is in standby, the core-melt frequency for the operating plant was approximately 10^{-6}/RY from T_a. If at least one ESW pump is running (simultaneous multiple failures of running pumps in both plants was considered unlikely) in the shutdown plant, the core-melt frequency of the operating plant from T_a was negligible.

Based on the above, TS requirements on ESW pumps while plants are in Modes 5 and 6 may provide a reduction in core-melt frequency of approximately 10^{-4}/RY for the operational plant at a two-unit multiplant site. When both plants are operating, the dominant core-melt frequency from an ESW transient (T) was estimated at 10^{-5}/RY. Improvements in valve realignments (crosstie) procedures were not believed to contribute significantly to core-melt frequency, but the resolution of this issue should reexamine the need for TS or procedures for these crosstie operations. It also appeared that changes to the ESW TS in Modes 1, 2, 3, and 4 would not provide significant reductions in core-melt frequency.

An additional ESW swing pump between plants or a third ESW pump/plant was estimated to provide at least an order of magnitude reduction in core-melt frequency. Therefore, the reduction in core-melt frequency from the addition of an ESW pump was estimated at approximately 10^{-5}/RY.

Consequence Estimate

As shown above, the two-unit multiplant configurations with only 2 ESW pumps/unit may have a core-melt frequency reduction potential (CM) on the order of 10^{-5}/RY when both units are running, or 10^{-4}/RY when one unit is running and the other is shut down. Because the indicated remedies for each dominant core-melt frequency were significantly different in scope and costs to implement, the risks were calculated separately. In each case, however, the estimated core-melt frequency was predicated on the potential unavailability of the ESW pumps in the adjoining unit of the multiplant configuration. The crosstie configurations and capability of the plant operators to realign the valves in the crosstie configurations were not estimated to be as significant an impediment to success in reducing core-melt frequency.

It was also estimated that recovery of the ESW pumps out of service cannot be assured in time to preclude a core-melt. Equipment such as the screen wash pumps (non-safety grade) might provide alternate means of service water cooling.

However, alternate equipment and its use in these situations will be highly plant-specific.

With the ESW system unavailable for direct or indirect cooling of all emergency core cooling systems and containment cooling systems, the containment was estimated to be as likely to fail by overpressurization (WASH-1400,[16] Category 2) as by basemat melt-through (WASH-1400,[16] Category 6), the timing of the release being dependent on progress and timing of the core-melt. Potential containment failures similar to the WASH-1400,[16] Category 4 (failure to isolate containment) were estimated to be of lower probability and, therefore, of lesser significance.

Given the above, the risk (consequences) was calculated as a product of the core-melt frequency, the release (dose) per category type release, the probability of the category type release, and the number of remaining reactor years of plant life. The conditional public dose per category type release was based on the fission product inventory of a 1120 MWe PWR, meteorology typical of the Byron site, and a surrounding uniform population density of 340 persons per square mile over a 50-mile radius from the plant site, with an exclusion radius of one-half mile from the plant.

Public Risk Parameters

Plant A Operating	Core-Melt Freq. (CM/RY)	Release Category (WASH-1400)[16]	Prob. of Release Category	Dose per Release Category (man-rem)	Remaining Plant Life	Public Risk (man-rem/reactor)
Plant B	1.3×10^{-4}	2	0.5	4.8×10^{6}	30	9,360
Shutdown	1.3×10^{-4}	6	0.5	1.5×10^{5}	30	300
					TOTAL:	9,700
Plant B	10^{-5}	2	0.5	4.8×10^{6}	30	720
Operating	10^{-5}	6	0.5	1.5×10^{5}	30	35
					TOTAL:	755

The estimated risk reduction that may result from installing a third ESW pump/plant, or an ESW swing pump per 2-unit multiplant configuration, was 755 man-rem/plant when both plants are in operation.

When one plant is in operation and the other plant is shut down (refueling), the estimated risk reduction from improved TS LCOs in Modes 5 and 6 was 9,700 man-rem/plant for the operating plant.

Cost Estimate

Three cost estimates were provided for this issue. The first considered the costs associated with the addition of a third pump per plant in a multiplant configuration. The estimated cost of the third pump/plant was also considered applicable to the cost of a swing pump between the 2 plants. In this second option, the cost of the swing pump can be shared between the 2 plants. This significantly lowers the per plant costs in a multiplant configuration. The third option involved modified TS on the LCOs for the ESW pumps. This analysis addressed TS LCOs on the ESW pumps in Modes 5 and 6. However, the TS for all modes of operation should be reviewed for adequacy and updated accordingly. It was also expected that Options 1 and 2 stated above might require additional TS.

Industry Cost: Based on estimates provided,[960] the cost of an additional service water pump/plant was approximately $15M, assuming an additional pump-house is not needed and that the work can be performed during a 60-day scheduled outage (no replacement power cost). The $15M/ESW pump included the following: direct cost (pump, piping, valve, and labor) estimated at $6M; indirect cost (engineering, temporary construction, and construction management) estimated to be approximately equal to the direct cost ($6M); and an additional cost ($3M) equivalent to 25% of direct and indirect costs to cover contingencies and operations and maintenance.

The industry cost to prepare the TS was estimated to be $16,000/plant[961] and included 8 man-weeks of licensee technical, legal, management, and committee input.

The total estimated industry cost/plant for each of the three options were:

(1)	Additional ESW Pump Plus TS	= $15M
(2)	Additional Swing Pump	= $7.5M
(3)	TS Modifications	= $0.016M

For Options 1 and 2, the TS costs were negligible when compared to the associated pump costs.

NRC Cost: The NRC cost included the cost to review and develop a solution(s) for the issue and the cost of reviewing plant-specific TS. The review and development of the solution(s) were estimated to require one staff-year of NRC time and approximately one man-year of contractor assistance. At a cost of $100,000/man-year, this amounted to $200,000 for all plants or $12,500/plant when distributed over at least 16 plants.

The NRC cost per plant was based on cost estimates given in NUREG/CR-4627[961] and included 6 staff-weeks of technical effort and three weeks for management and legal reviews and concurrences. Based on a rate of $50.00/staff-hour, the NRC costs were estimated at $18,000/plant per TS change. Considering that two Federal Register notices might be required ($800), the total NRC cost was estimated to be approximately $19,000/plant. The total NRC cost, including the generic review costs distributed over the affected plants and the plant-specific TS costs, amounted to a total NRC cost of $32,000/plant. The above NRC costs were applicable to each of the three options discussed in this analysis.

Total Cost: The estimated total industry and NRC cost/plant for the above three conditions and options were approximately $15M, $7.5M, and $0.05M, respectively.

Value/Impact Assessment

Three value/impact assessments were calculated for this issue.

(1) Additional ESW Pump/Plant: $S = \frac{755 \text{ man-rem/reactor}}{\$15M\text{/reactor}}$

$= 50$ man-rem/\$M

(2) Additional Swing Pump: $S = \frac{755 \text{ man-rem/reactor}}{\$7.5M\text{/reactor}}$

$= 100$ man-rem/\$M

(3) Modified TS/Modes 5, 6: $S = \frac{9{,}700 \text{ man-rem/reactor}}{\$0.05M\text{/reactor}}$

$= 2 \times 10^5$ man-rem/\$M

Other Considerations

This issue was evaluated based on approximate generic success criteria for 2-Unit configurations with 2 ESW pumps/unit and crosstie capabilities between the units. In actual plant configurations, the success criteria and shared use of ESW and other equipment are highly plant-specific. Because of various ESW pump capacities, some plants with more than 2 F pumps/plant might also have vulnerable ESW systems. Single unit designs should be reviewed for potential ESW vulnerabilities.

Because of the large variations in ESW designs and success criteria, there are large uncertainties in a limited generic analysis such as this one. Further, a more careful analysis that includes additional sequences (valve faults, etc.) may show greater (or lesser) ESW plant-specific vulnerabilities and public risk.

The possible solutions may also vary from plant to plant. However, this issue identified the need to evaluate possible ESW vulnerabilities in all modes of plant operations for single and multiplant configuations.

The need for requirements on crosstie operations and ESW TS in Modes 5 and 6 was identified in this evaluation as potentially significant in reducing public risk and was determined to be potentially cost-effective. In this regard, it was recommended that resolution of this issue be coordinated with the Technical Specifications Branch, DOEA/NRR.

CONCLUSION

Based on the evaluation and other considerations described above, this issue was given a high priority ranking. In resolving the issue, the staff addressed the loss of essential service water at 7 multiplant sites. The affected units have similar ESW system designs with two trains per unit: one pump per train with a crosstie between units. The issue was resolved with TS and emergency procedures improvements issued in Generic Letter No. 91-13.[1368] The staff's technical findings and regulatory analysis were published in NUREG/CR-5526[1408] and NUREG-1421,[1409] respectively. Thus, this issue was RESOLVED and requirements were issued.[1410]

REFERENCES

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

958. Memorandum for T. Speis from T. Novak, "Need For Oversight Guidance - Byron 2 - Pump Service Water Issue and Related Generic Issues," May 6, 1986.

959. EGG-EA-5524, "Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants from January 1, 1972, to September 30, 1980," Idaho National Engineering Laboratory, September 1981.

960. Letter to D. Ericson (Sandia National Laboratories) from J. Mulligan (United Engineers and Constructors), "Decay Heat Removal Systems Evaluations Feasibility and Cost Evaluations of Special Issues Related to Decay Heat Removal," January 20, 1986.

961. NUREG/CR-4627, "Generic Cost Estimates," U.S. Nuclear Regulatory Commission, June 1986.

1368. NRC Letter to Licensees and Applicants of the Following Pressurized-Water Reactor Nuclear Power Plants: (1) Braidwood Units 1 and 2; (2) Byron Units 1 and 2; (3) Catawba Units 1 and 2; (4) Comanche Peak Units 1 and 2; (5) Cook Units 1 and 2; (6) Diablo Canyon Units 1 and 2; (7) McGuire Units 1 and 2, "Request for Information Related to the Resolution of Generic Issue 130, 'Essential Service Water System Failures at Multi-Unit Sites,' Pursuant to 10 CFR 50.54(f) - Generic Letter 91-13," September 19, 1991.

1408. NUREG/CR-5526, "Analysis of Risk Reduction Measures Applied to Shared Essential Service Water Systems at Multi-Unit Sites," U.S. Nuclear Regulatory Commission, June 1991.

1409. NUREG-1421, "Regulatory Analysis for the Resolution of Generic Issue 130: Essential Service Water System Failures at Multi-Unit Sites," U.S. Nuclear Regulatory Commission, June 1991.

1410. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Issue 130, 'Essential Service Water System Failures at Multi-Unit Sites,'" September 23, 1991.

ISSUE 133: UPDATE POLICY STATEMENT ON NUCLEAR PLANT STAFF WORKING HOURS

DESCRIPTION

Historical Background

In IE Circular No. 80-02,[975] the concern of overtime work for licensee staff who perform safety-related functions was discussed and limits on maximum working hours were recommended. In July 1980, a letter[976] was issued to OLs and CPs with interim criteria for shift staffing, including restrictions on overtime. These criteria were superseded by the NRC requirements issued in NUREG-0737,[98] Item I.A.1.3.

In February 1982, NRC issued a policy statement[977] on "Nuclear Power Plant Staff Working Hours." Based on public comments, this policy statement was revised and reissued[978] in June 1982. Generic Letter 82-12[979] transmitted this latter version of the policy statement to OLs and CPs along with instructions to revise TS administrative procedures to conform to the policy statement. Guidance on incorporating limits on overtime into the TS was later issued in Generic Letters 82-16[980] and 83-02[981] to PWRs and BWRs, respectively. In March 1983, Generic Letter 83-14[982] was issued to clarify the definition of "Key Maintenance Personnel" stated in Generic Letter 82-12. In September 1985, the staff was directed[983] to update the policy statement on "Nuclear Power Plant Staff Working Hours." Since the NRC policy was stated in several documents, revision of NRC's policy guidance on limits on overtime and shift scheduling was needed to consolidate the guidance into a single document.

The existing policy statement and implementing documents were considered adequate from a safety perspective in that the amount of overtime worked by nuclear power plant personnel was not identified as an actual contributor to reportable events, nor did it degrade the safety of plant operations. However, one specific area of guidance relating to the use of 12-hour shifts was absent from the policy statement. The staff reviewed and approved on a case-by-case basis licensee programs for routine 12-hour shifts (e.g., Oconee and Callaway).

The proposed policy statement was intended to achieve the following: (1) update and clarify NRC's policy on shift scheduling for both routine 8-hour and 12-hour shifts; (2) establish control of overtime hours worked by nuclear power plant personnel who perform safety-related functions; and (3) clarify what action NRC will take in instances where it is determined that fatigue from excessive working hours has degraded personnel performance and thereby contributed to unsafe nuclear power plant operation.

The proposed policy statement was unchanged from existing practice, with respect to administrative procedures, to prevent personnel who perform safety-related functions from working in a fatigued condition during normal operations no more than 16 hours in a 24-hour period, 24 hours in a 48-hour period, or 72 hours in any 7-day period. Overtime on an individual basis was to be controlled with management approval of deviations from recommended limits on working hours.

The staff believed that a revised policy statement on shift and scheduling and hours of work would eliminate licensee confusion resulting from multiple policy and requirement documents and would clearly identify licensee management's responsibility to assure that nuclear power plant staff fatigue resulting from excessive working hours did not adversely affect public health or safety. Additionally, the revised policy statement would benefit the NRC staff in conducting reviews of licensee programs and in monitoring licensee implementation.

CONCLUSION

This issue was classified as a Licensing Issue that was resolved[1349] with the issuance of NRC Information Notice No. 91-36.[1350]

REFERENCES

98. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

975. IE Circular No. 80-02, "Nuclear Power Plant Staff Work Hours," U.S. Nuclear Regulatory Commission, February 1, 1980.

976. NRC Letter to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, "Interim Criteria for Shift Staffing," July 31, 1980.

977. Federal Register Notice 47 FR 7352, "Nuclear Power Plant Staff Working Hours," February 18, 1982.

978. Federal Register Notice 47 FR 23836, "Nuclear Power Plant Staff Working Hours," June 1, 1982.

979. NRC Letter to All Licensees of Operating Plants, Applicants for an Operating License, and Holders of Construction Permits, "Nuclear Power Plant Staff Working Hours (Generic Letter No. 82-12)," June 15, 1982.

980. NRC Letter to All Pressurized Power Reactor Licensees, "NUREG-0737 Technical Specifications (Generic Letter No. 82-16)," September 20, 1982.

981. NRC Letter to All Boiling Water Reactor Licensees, "NUREG-0737 Technical Specifications (Generic Letter No. 83-02)," January 10, 1983.

982. NRC Letter to All Licensees of Operating Plants, Applicants for Operating Licenses, and Holders of Construction Permits, "Definition of 'Key Maintenance Personnel,' (Clarification of Generic Letter 82-12) (Generic Letter 83-14)," March 7, 1983.

983. Memorandum for W. Dircks from J. Hoyle, "Updating NRC Policy Statements," September 30, 1985.

1349. Memorandum for J. Roe from J. Wermiel, "Closure of Generic Issue No. 133, 'Update Policy on Nuclear Plant Staff Working Hours,'" July 10, 1991.

1350. NRC Information Notice No. 91-36, "Nuclear Plant Staff Working Hours," June 10, 1991.

ISSUE 135: STEAM GENERATOR AND STEAM LINE OVERFILL

DESCRIPTION

Steam generator overfill and its consequences have received staff and industry attention because of the frequency and severity of overfill events. Over the years, a number of issues have been raised concerning steam generator overfill including Issue 66, "Steam Generator Requirements," and Issue 67, "Steam Generator Staff Actions." In order to provide an integrated work plan for the resolution of these issues, Issue 135 was initiated[1075] and assigned a medium priority ranking based on the separate evaluation of Issue 67.7.0, "Improved Eddy Current Tests."

Resolution of Issue 135 was expected to provide a better understanding of steam generator and secondary steam integrity, including the effects of water hammer on secondary system components and piping as well as the resultant radiological consequences. The work scope for resolving this issue was divided into four tasks which called for the following staff actions:

Task 1: (a) survey the code requirements and industry practice for eddy current testing procedures; (b) assess the capability of current methods to detect steam generator tube degradation; (c) review existing ASME Section II requirements on eddy current testing procedures and determine its adequacy for use as a standard for inspection of steam generator tubes; and (d) develop written recommendations for regulatory guidance and/or requirements, including possible endorsement of ASME Section II requirements on eddy current testing procedures for development of a draft regulatory guide.

Task 2: Review the results and conclusions of studies on SGTR and propose specific modifications to SRP[11] Section 15.6.3 including tube integrity, operator action time, and offsite dose limits. Develop a regulatory analysis supporting the SRP changes including a risk analysis and a cost benefit of the proposed SRP changes.

Task 3: Reassess the following concerns in Issue 67 for potential inclusion in an integrated resolution: reassessment of radiological consequences, reevaluation of design basis SGTR, supplemental tube inspections, integrity of steam generator tube sleeves, denting criteria, improved accident monitoring, reactor vessel inventory measurement, RCP trip, control room design review, EOPs, organizational responses, and RCS pressure control.

Task 4: Review the effects of water hammer, overfill, and water carryover on the secondary system and connecting systems and develop proposals for mitigating the consequences. Consider the effects of sagging due to water weight, operability of valves, and other components when subjected to two-phase flow of liquid.

The coordination of results of the different tasks was to provide a basis for the staff to develop a position on offsite dose, operator action time, and tube integrity. Water hammer mitigation studies were to be carried out to give the staff a better understanding for developing positions on water hammer in main steam lines and operability of valves and other components.

CONCLUSION

This issue was given a medium priority ranking and pursued by the staff. It was found that SGTR and steam line overfill events pose a relatively low public risk, as previously indicated in NUREG-0844[681]; comparable risk results for SGTR events were also published in NUREG-1150.[1081] The staff technical findings were published in NUREG/CR-4893.[1411] Thus, this issue was RESOLVED and no new requirements were established.[1337]

REFERENCES

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

681. NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, A-5 Regarding Steam Generator Tube Integrity," U.S. Nuclear Regulatory Commission, September 1988.

1075. Memorandum for K. Kniel from R. Bosnak, "Integration of NUREG-0933 Issues," May 27, 1986.

1081. NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Volume 1) December 1990, (Volume 2) December 1990, (Volume 3) January 1991.

1337. Memorandum for J. Taylor from E. Beckjord, "Proposed Resolution and Closeout of Generic Issue 135, 'Steam Generator and Steam Line Overfill Issues,'" March 29, 1991.

1411. NUREG/CR-4893, "Technical Findings Report for Generic Issue 135," U.S. Nuclear Regulatory Commission, May 1991.

ISSUE 138: DEINERTING OF BWR MARK I AND MARK II CONTAINMENTS DURING POWER OPERATIONS UPON DISCOVERY OF RCS LEAKAGE OR A TRAIN OF A SAFETY SYSTEM INOPERABLE

DESCRIPTION

Historical Background

The issue of deinerting upon discovery of RCS leakage was identified[1414] by DL/NRR, based on data collected by OIE. The related but separate concern of deinerting with one train of a safety system inoperable was also raised.

BWR MARK I and II containments are inerted with nitrogen during normal operations to protect against the build-up of a potentially explosive H_2-O_2 mixture, in the event of a LOCA or core damage event. Following a LOCA, H_2 is evolved within the containment from zircaloy-water reactions and H_2 and O_2 are also produced by radiolysis of the coolant. Core damage or melting would add an additional large quantity of H_2 as a result of metal/water reaction with fuel cladding and core structural materials. Plant atmosphere systems are designed to maintain containment O_2 concentration to less than 5% by volume, or the H_2 concentration to less than 4% by volume, to ensure that a combustible gas mixture does not form.

RCS leakage outside of TS limits requires licensees to identify, isolate, and repair the leak to avoid or mitigate the consequences of a LOCA. These steps require plant personnel entry into the containment. In accordance with plant TS, operators typically reduce power, deinert the containment, and allow personnel entry to identify potential RCS leaks. However, deinerting under leak conditions leaves the containment vulnerable to H_2-O_2 concentration build-up, if the leak progresses to a LOCA or core damage event. This was the primary concern in this issue. A secondary concern was that TS also allow licensees to operate with a deinerted containment for up to 24 hours with one train of a safety system inoperable.

The OIE data consisted of 13 RCS leak event reports in which the containment was deinerted to allow corrective action by plant personnel. These events occurred between 1981 and 1986. Existing NRC guidance for this issue included GDC 41 (Containment Atmosphere Cleanup), from 10 CFR 50 Appendix A, and Section 3.6.6.4 of the GE STS for BWR/5 designs.

Safety Significance

With either of the above concerns, the possibility of early gross containment failure with energetic source term release could be significantly increased, thereby increasing public risk. The issue affected 33 BWRs with MARK I or II containments.

Possible Solution

A possible solution to this issue was to revise plant TS to require a reactor to be brought to cold shutdown, prior to deinerting the containment, when either unidentified leakage in the containment building or inoperability of a safety system is noted.

PRIORITY DETERMINATION

Frequency/Consequence Estimate

The baseline risk assessment used for this issue was the Millstone 1 PRA which was based on the assumption that the containment would always be inerted unless the reactor is in a cold shutdown condition; the PRA provided an assessment of risks associated with accident sequences at full power. In addition, the Millstone 1 PRA took into account TS allowing operations with one train of the safety system inoperable. The following two cases were assessed for this issue.

Case 1 A deinerted containment during a shutdown for an unidentified RCS leak.

Case 2 A deinerted containment with one train of a safety system inoperable.

For this issue, potential public risk reduction results from the change in containment failure mode afforded by not permitting deinerting during power operations under off-normal conditions. The assumed resolution of the issue will not result in a reduction of core-melt frequency.

To analyze the issue, an average consequence factor (C) was determined using the following equation:

$$R = (f_{cm})(\bar{C}) = \Sigma(f_{cmn})(C_n)$$

where R = Risk (man-rem/RY)

f_{cm} = Core-melt frequency (event/RY)

$\bar{C}$ = Average Consequence Factor (man-rem/event)

f_{cmn} = Total core-melt frequency for the nth release category (event/RY)

C_n = Consequence Factor for the nth release category (man-rem/event)

The following values were used from the Millstone 1 PRA and NUREG/CR-2800:[64]

$f_{cm} = 3.09 \times 10^{-4}$/RY	$f_{cm_1} = 1 \times 10^{-6}$/RY	$C_1 = 5.4 \times 10^6$ man-rem
	$f_{cm_2} = 8 \times 10^{-6}$/RY	$C_2 = 7.1 \times 10^6$ man-rem
	$f_{cm_3} = 1 \times 10^{-4}$/RY	$C_3 = 5.1 \times 10^6$ man-rem
	$f_{cm_4} = 2 \times 10^{-4}$/RY	$C_4 = 6.1 \times 10^5$ man-rem

Therefore, the value of $\bar{C}$ was 2.25×10^6 man-rem/event.

From the Millstone 1 PRA, the only dominant LOCA sequence is found to be the small-break LOCA ($SB_{(B)}$). The frequency of $SB_{(B)}$($fSB_{(B)}$) is 3×10^{-6} event/RY.

In analyzing the first concern, i.e., deinerting the containment with an unknown leakage into the containment building, two different schools of thought concerning the effect of detected leakage upon LOCA expected frequency were addressed. Employing the leak-before-break theory, detection of leakage within the containment building would not be associated with an increase in LOCA frequency. Conversely, it could be assumed that leakage from a through-wall pipe crack would increase the probability that the pipe would break causing a LOCA. As a result, the unknown leakage concern was analyzed from both perspectives as Case 1.

It was assumed that, whenever the containment building is deinerted, any core-melt event will result in containment failure due to H_2 burn with a probability of 1. TS permit licensees to deinert the containment building of BWR MARK I and II designs, 24 hours prior to a scheduled shutdown. LERs gathered over the 5-year period revealed 9 events in which the plants were deinerted before reaching a hot shutdown condition and one event in which the plant was deinerted before shutdown with one train of ECCS inoperable. Of the 9 leakage events, 8 were found to be valve stem, body packing, or seal failures and one was determined to be a through-wall crack in a RCS line. Assuming that the containment was deinerted for 24 hours during plant operation for each of these events and that during the five-year period 29 plants operated with an assumed power production factor (i.e., historical fraction of the calendar-year that plants operate at or near full power) of 0.7, the following fractions of plant operating history were calculated. Each represents the fraction of the operating history that a plant experiences a deinerted condition, as a result of the particular set of circumstances:

V_a	- deinerted due to unknown leakage (9 events)	$= 2.4 \times 10^{-4}$
V_b	- deinerted due to unknown leakage actually due to through-wall pipe leak (1 event)	$= 2.7 \times 10^{-5}$
V_c	- deinerted due to unavailability of one train of a safety system (1 event)	$= 2.7 \times 10^{-5}$

Assuming that, if the containment is deinerted and a core damage event occurs, the conditional probability of containment failure (F_{con}) due to H_2 explosion is 1, the conditional probability of containment failure becomes the probability that the containment is deinerted (V_n). The consequence of containment failure due to H_2 explosion in the Millstone 1 PRA is best represented by the consequence factor (C_2) for a Category 2 release, i.e., early large containment failure. Therefore, the risk during periods while the plant is deinerted in preparation for a shutdown with an unknown leakage, assuming that detection of leakage does not increase the probability of a pipe break (LOCA), was calculated as follows:

$$r = (f_{cm})(V_a)(F_{con})(C_2)$$

$$= (3.1 \times 10^{-4})(2.4 \times 10^{-4})(1)(7.1 \times 10^{6}) \text{ man-rem/RY}$$
$$= 0.528 \text{ man-rem/RY}$$

This was the base case risk. If plants are not permitted to deinert during periods of unknown leakage in the containment building, the adjusted case risk would be:

$$r' = (f_{cm})(V_a)(\bar{C})$$
$$= (3.1 \times 10^{-4})(2.4 \times 10^{-4})(2.25 \times 10^{6}) \text{ man-rem/RY}$$
$$= 0.167 \text{ man-rem/RY}$$

The change in risk (Δr) is ($r-r'$) or 0.361 man-rem/RY. When applied to the 33 affected plants with an average remaining life of 18.6 yrs, the total risk reduction attainable by not permitting plants to deinert when an unidentified source of leakage in the containment building exists was estimated to be 222 man-rem.

If it is assumed that the probability of a LOCA (in this case $SB_{(B)}$) is increased by two orders of magnitude if a through-wall leak exists, the potential risk reduction afforded by not permitting plants to deinert when an unidentified source of leakage in the containment building exists is calculated as follows:

$$r = (f_{cm})(V_a - V_b)(F_{con})(C_2) + (f_{cm}')(V_b)(F_{con})(C_2)$$

where $fSB_{(B)}' = fSB_{(B)} \times 100 = 3.1 \times 10^{-4}$/RY

and $f_{cm}' = f_{cm} - fSB_{(B)} + fSB_{(B)}' = 6.08 \times 10^{-4}$/RY

Thus, the base case risk (r) is 0.585 man-rem/RY.

If deinerting is not allowed when there is an unknown leakage in the containment building, the adjusted case risk is calculated as follows:

$$r' = (f_{cm})(V_a - V_b)(\bar{C}) + (f_{cm}')(V_b)(\bar{C}) = 0.186 \text{ man-rem/RY}$$

Thus, the change in risk due to resolution of this aspect of the issue is (0.585 - 0.186) man-rem/RY = 0.399 man-rem/RY.

When applied to the 33 affected plants with an average remaining life of 18.6 years, the potential risk reduction due to resolution, if it is assumed that through-wall leakage increases the probability of LOCA by two orders of magnitude, is 245 man-rem.

The second case, i.e. containment deinerting prior to shutdown with a train of a safety system inoperable, was analyzed as follows.

The LER data from the 5-year study period revealed only one instance of deinerting prior to shutdown with a train of a safety system (LPS in this case) inoperable. This results in a fraction of operating history for this condition $V_c = 2.7 \times 10^{-5}$ as shown earlier. It was conservatively assumed that unavailability of one train of any safety system will increase the core-melt frequency by one order of magnitude, i.e., $f_{cm}^{z} = 3.1 \times 10^{-3}$/RY.

The base case risk (r_z) was calculated by the following relationship:

$$r_z = (f_{cm}^{z})(V_c)(F_{con})(C_2)$$
$$= (3.1 \times 10^{-3})(2.7 \times 10^{-5})(1)(7.1 \times 10^{6}) \text{ man-rem/RY}$$
$$= 0.594 \text{ man-rem/RY}$$

If resolution of the issue does not permit deinerting with one train of a safety system inoperable, the adjusted case risk (r_z') is calculated from the following relationship:

$$\begin{aligned} r_z' &= (f_{cm}{}^{z})(V_c)(\bar{C}) \\ &= (3.1 \times 10^{-3})(2.7 \times 10^{-5})(2.25 \times 10^{6}) \text{ man-rem/RY} \\ &= 0.188 \text{ man-rem/RY} \end{aligned}$$

Thus, the change in plant risk for this case due to resolution of the issue is ($r_z - r_z'$) = (0.594 - 0.188) man-rem/RY = 0.406 man-rem/RY. When applied to the affected population of 33 plants over the average remaining life of 18.6 years, the potential risk reduction attainable by not permitting deinerting with one train of any safety system inoperable is 249 man-rem.

Cost Estimate

Industry Cost: Industry costs included preparation and implementation of TS changes for the affected plants and replacement power costs that would be incurred because TS eliminating deinerting prior to shutdown under off-normal conditions will lengthen plant outages by up to 24 hours per shutdown. At $18,000/plant, the total industry cost for simple TS changes was estimated to be $594,000.[961]

Assuming that each time a plant must shut down for an unidentified leak in the containment building and/or a safety system train inoperable not allowing the 24-hour deinerting period will add one day to the plant outage, the average replacement power cost was estimated to be $300,000/day for each instance. Ratioing the 9 containment leakage events and the 1 loss of a safety system train event from the 5-year survey of LERs, it was determined that for the 33 affected plants over their remaining life of 18.6 years, 38 leakage events and 4 safety system train events would be expected. This equates to an industry replacement power cost of $12.6M ($11.4M for leakage events and $1.2M for loss of a safety system train events).

Thus, the total estimated industry costs were $12M and $1.8M for the leakage and safety system inoperability aspects of the issue, respectively.

NRC Cost: Resolution of either or both parts of this issue will require the issuance of a backfit order and the development, review, and approval of a revised TS for each of the affected plants. Development and approval of the resolution of the issue was estimated to require a staff effort of $100,000 and a technical assistance contractual effort of $250,000, for a total of $350,000. Imposition and implementation of the resolution of the issue, i.e., review and approval of a simple TS change, were estimated to be $11,000/plant, and $363,000 for the 33 affected plants.[961] Thus, the total NRC cost was estimated to be $713,000.

Total Cost: For Case 1, the total industry and NRC cost associated with the possible solution is $(12 + 0.713)M for leakage events and loss of a safety system train events. For Case 2, the total industry cost is $(1.8 + 0.713)M.

Value/Impact Assessment

Separate value/impact scores were calculated for each case.

(1) Case 1 - Leakage Only

(a) Based on no increased probability of LOCA,

$$S = \frac{222 \text{ man-rem}}{\$12.7M} = 17.5 \text{ man rem/\$M}$$

(b) Based on two orders of magnitude increase in LOCA probability,

$$S = \frac{245 \text{ man-rem}}{\$12.7M} = 19.3 \text{ man-rem/\$M}$$

(2) Case 2 - Safety System Train Inoperable

$$S = \frac{249 \text{ man-rem}}{\$2.5M} = 99.6 \text{ man-rem/\$M}$$

Other Considerations

The remaining life of the plants used to calculate the risk, cost, and value/impact scores was based on the assumption that the total operating life of the existing operating plants was limited to 40 years. The potential for license extension was also considered with the assumption that 75% of existing operating plants would apply for license extensions of 20 years.

The additional risk reduction increment from license extensions would not result in a total potential risk reduction of more than 500 man-rem for either aspect of this issue. Since both the risk reduction estimates and the licensee costs estimates were a direct function of remaining plant operating life, the value/impact scores remain essentially unchanged by consideration of license extension.

CONCLUSION

Whether considered for the remaining licensed lifetime of BWR MARK I and II plants or for additional operating life through license extension, the value/impact scores and potential risk reduction for both the leakage and safety system train inoperability aspects of the issue fall into the LOW priority category.

REFERENCES

1414. Memorandum for K. Kniel from G. Lainas, "Proposed Generic Issue Deinerting Upon Discovery of Reactor Coolant System Leakage," August 1, 1986.

961. NUREG/CR-4627, "Generic Cost Estimates," U.S. Nuclear Regulatory Commission, June 1986.

ISSUE 142: LEAKAGE THROUGH ELECTRICAL ISOLATORS IN INSTRUMENTATION CIRCUITS

DESCRIPTION

Historical Background

Electronic isolators are used to maintain electrical separation between safety and non-safety-related electrical systems in nuclear power plants, preventing malfunctions in the non-safety systems from degrading performance of safety-related circuits. Isolators are primarily used where signals from Class-1E safety-related systems are transmitted to non-Class 1E control or display equipment.

There are a number of devices which may qualify as electrical isolators in a nuclear power plant, including fiber optic and photo-electric couplers, transformer-modulated isolators, current transformers, amplifiers, circuit breakers, and relays. These isolators are designed and tested to prevent the maximum credible fault applied in the transverse mode on the non-Class 1E side of the isolator from degrading the performance of the safety-related circuit (Class-1E side) below an acceptable level.

This issue was identified[1270] by the staff in June 1987 and arose from observations made during SPDS evaluation tests that, for electrical transients below the maximum credible level, a relatively high level of noise could pass through certain types of isolation devices and be transmitted to safety-related circuitry.[1269] In some cases, the amount of energy that can pass through the isolator may be sufficient to damage or seriously degrade the performance of Class 1E components, while, in other cases, electrically-generated noise on the circuit may cause the isolation device to give a false output.

Safety Significance

Recent observations have shown instances in which isolation devices subjected to failure voltages and/or currents less than maximum credible fault levels passed significant levels of voltage or current, but the same devices performed acceptably at maximum credible levels. The safety system on the Class 1E side of the isolation device may be affected by the passage of small levels of electrical energy, depending upon the design and function of the safety system.

In the event that safety systems are affected by less than maximum credible faults on the non-Class 1E side of isolators, the effects can range from degradation to failure of single or multiple trains of safety systems resulting in failure on demand or inadvertent operation. In one recorded incident, a voltage transient induced by a power line fault caused a false indication that the turbine-generator output breaker had tripped, resulting in a reactor scram.

Possible Solution

The assumed solution to this issue would require the staff to determine the extent to which potentially susceptible isolators are used in nuclear power plants and to identify the systems in which they are used. An NRC bulletin to all licensees to provide input on these questions would be necessary. Assuming that the staff determines from the licensee responses to the proposed bulletin that a potential problem exists, a research program consisting of two major objectives would have to be initiated to develop the solution to this issue. The first objective would be to develop test procedures and acceptance criteria for isolators that licensees could use to determine the adequacy of installed isolators. The second objective would involve development of appropriate hardware fixes that could resolve the issue.

Electrical hardware currently exists either to reduce the amount of energy that may leak through electrical barriers provided by various types of isolation devices, or to minimize the consequences of any unwanted signals that may leak through the isolator. Some of these devices are described below.

Surge arresters, also called lightning arresters, provide an effective means of eliminating high voltage transients from a circuit. These devices are simply connected from the conductor directly to ground, preferably as close as possible to the device to be protected. The arresters function by simply shunting to ground any voltage spikes above a certain level.

Filter chokes and capacitors can greatly attenuate high frequency electrical noise. These components create an impedance to the passage of electrical energy proportionate to the frequency of the signal and are especially effective against radio frequency noise. Filter chokes (or reactors) also function as current limiters in AC circuits and thus offer additional protection from overload currents.

At power frequencies, power conditioners can be employed to eliminate all unwanted signals. Power line conditioners function by rectifying an AC signal into DC and then reconverting power through an invertor into a clean, noise-free AC signal. These devices prevent notches, spikes, radio frequency, brownouts, and overload power at the input terminals from degrading the quality of power at the protected output.

The final step in the solution to this issue would be the issuance of a generic letter to licensees with the following guidelines for: (1) inspection and testing of all electrical isolation devices between Class 1E and non-Class 1E systems; (2) repair/replacement of isolators that fail the tests, including description of acceptable hardware fixes to the isolators; and (3) implementation of an annual program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems.

PRIORITY DETERMINATION

Assumptions

A total of 90 PWRs and 44 BWRs are potentially affected by this issue. The expected average remaining lives of these plants are 28.8 and 27.4 years for PWRs and BWRs, respectively.

Frequency Estimate

There are several sources of uncertainty associated with this issue, the most important of which are: (1) the extent to which potentially susceptible isolators are used at nuclear power plants; (2) the amount of electrical energy leakage through isolation devices that could compromise the function of Class 1E system components; and (3) the number of components in which such compromises would be critical. While a recent study[1269] indicated that a safety problem may exist due to energy leakage through electronic devices, no definitive research has been conducted to date to indicate the character and magnitude of the associated safety concerns. As a result, a sensitivity analysis was performed to bound the potential public risk reduction associated with this issue. Estimates of the upper and lower bounds were developed as well as a third case that represents the "best estimate" based on the available information.

The Oconee 3 and Grand Gulf 1 PRA studies were used as representative of PWRs and BWRs, respectively.[64] The parameters affected by this issue are those involving control circuitry failures and functional failure of ESF actuation systems. These components may be directly affected by energy leakage through isolation devices that are intended to protect them from signals originating in connected non-Class 1E systems. It is also possible that sensors in the Class 1E safety systems may be affected by the electrical energy leakage from the non-Class 1E system. These sensors may include valve position, temperature, and pressure sensors that alert plant operators to take a particular action. In this case, plant operators may be misled into not taking appropriate actions when required. For this reason, operator error terms are also included as potentially affected parameters. The affected parameters in the Oconee 3 and Grand Gulf 1 PRAs were identified and modified to model the three sensitivity cases.

Best Estimate: All of the affected control circuitry failure, ESF actuation functional failure, and operator error terms were multiplied by a factor of two (assumed) to account for the potential additional failures associated with electrical isolators. A factor of two was assumed based on engineering judgment and the findings of previous prioritization analyses.

Upper Bound: All of the affected control circuitry failure, ESF actuation functional failure, and operator error terms were multiplied by a factor of ten (assumed) to account for the potential additional failures associated with electrical isolators. A factor of 10 was likewise assumed based on judgment and previous analytical experience.

Lower Bound: The control circuitry and ESF actuation functional failures were multiplied by a factor of 1.4. This is based on an assumed factor of two increase in only the probability of fuse failures which are included in the control circuitry unavailability values. No effect on the operator error terms were assumed in this case.

It is noted that varying all the control circuitry, ESF function failure, and operator error terms is a conservative approach. Logic dictates that not all the terms would be effected at the same time and that a plant-specific detailed evaluation would probably result in a reduced sensitivity. After the failure terms were modified, they were combined with the remaining unaffected portions of the parameter unavailabilities to calculate the revised unavailabilities. The affected cut-set elements and their base case and adjusted case unavailability values are shown in Table 3.142-1.

In performing the risk analysis, it was assumed that the isolator failures were not considered as potential causes of failure in the original Oconee and Grand Gulf PRAs. (This assumption may also introduce additional conservatism.)

Since the base case is intended to represent the situation in which isolator failures are considered as possible causes of safety system failures and the adjusted case represents the situation after the resolution is implemented, the modified parameter values are used in the base case and the adjusted case represents the original Oconee and Grand Gulf parameter values. The base case and adjusted case values of the affected parameters were then incorporated in the Oconee 3 and Grand Gulf 1 PRAs to derive the estimated core-melt frequency and the associated public risk reduction. Based on the data in Table 3.142-1, the following core-melt frequency reduction was estimated for the representative PWR and BWR.

Sensitivity Case	Core-Melt Frequency Reduction PWR	BWR
Best Estimate	2.55×10^{-5}/RY	7.98×10^{-6}/RY
Lower Bound	5.37×10^{-6}/RY	2.07×10^{-6}/RY
Upper Bound	4.35×10^{-4}/RY	1.17×10^{-4}/RY

Utilizing generic release categories and containment failure modes, the public risk reduction was estimated to be as follows:

Sensitivity Case	Public Risk Reduction (man-rem/RY) PWR	BWR
Best Estimate	57	53
Lower Bound	13	14
Upper Bound	1,016	789

Based on the public risk reduction estimates presented before for the representative PWR and BWR and the three sensitivity cases, the following public risk reduction was estimated (weighted average over all affected PWRs and BWRs and their remaining lives):

Best Estimate = 1,580 man-rem/plant
Lower Bound = 378 man-rem/plant
Upper Bound = 26,752 man-rem/plant

Cost Estimate

Industry Cost: It was assumed that the proposed generic letter would contain the following guidelines applicable to all affected plants: (1) inspection and testing of all electrical isolation devices between Class 1E and non-Class 1E systems; (2) replacement of failed or unacceptable isolators, including descriptions of acceptable hardware fixes to the isolators; and (3) implementation of an annual program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems.

The initial testing and inspection program at each plant is estimated to require approximately 4 man-weeks for planning and 8 man-weeks for review and evaluation of the data, preparation of the final response to the generic letter, and preparation of a safety analysis. Industry cost to conduct the initial test program is highly uncertain because there are unknown numbers of affected systems and susceptible isolators at each plant. For the purposes of this analysis, the number of potentially affected isolators was estimated using

TABLE 3.142-1

Base Case and Adjusted Case Values of Affected Parameters

Parameter	Adjusted Case[a]	Base Case 1[b]	Base Case 2[c]	Base Case 3[d]
Grand Gulf				
H	0.0212	0.0225	0.0217	0.0329
HACT, RACT	0.00123	0.00223	0.00163	0.0102
R	0.0512	0.0530	0.0518	0.067
L	0.0213	0.0226	0.0218	0.033
LRACT, BCACT	0.00123	0.00223	0.00163	0.0102
LA2, LB2	0.0140	0.0151	0.0144	0.0240
LB1	0.0134	0.0138	0.0135	0.017
LC	0.0215	0.0230	0.0220	0.035
VGA1, VGB1	0.0148	0.0156	0.0150	0.022
[illegible]	0.0236	0.0270	0.0238	0.0553
[illegible]	0.0144	0.0150	0.0146	0.0198
[illegible]	0.00123	0.00223	0.00163	0.0102
SS4, [illegible]	0.0205	0.0223	0.0209	0.0361
[illegible]	0.0140	0.0151	0.0144	0.0239
[illegible]	0.00123	0.00223	0.00163	0.0102
[illegible]	0.00803	0.0091	0.00813	0.0173
[illegible]	0.0033	0.0064	0.0033	0.0296
[illegible]	0.0315	0.0333	0.0321	0.0477
Oconee				
B, C	0.0033	0.0043	0.0037	0.0121
D, E	0.0231	0.0354	0.0249	0.1334
CONST1	0.0002	0.00048	0.0003	0.0007
CONST2	0.0006	0.00125	0.00083	0.0123
A1, C1	0.0098	0.0163	0.0124	0.0683
B1	0.0349	0.0502	0.0710	0.1718
G1	0.0135	0.0172	0.0150	0.046
RCSRBCM	0.00003	0.00007	0.00003	0.00032
WXCM	0.003	0.006	0.003	0.03
D.E	0.00049	0.00121	0.0006	0.0178
W.X	0.00009	0.00025	0.0001	0.00451
B.W, C.X	0.00003	0.00006	0.00004	0.00081
D.X, E.W	0.00021	0.0006	0.00029	0.00895
B.D, E.C	0.00006	0.0001	0.00008	0.0016

NOTES: (a) Original Oconee 3 and Grand Gulf 1 PRA values
(b) Best estimate
(c) Lower bound case
(d) Upper bound case

the number of safety system components in the Oconee and Grand Gulf PRAs with functional and/or control circuitry failure terms. Accordingly, 46 isolators for BWRs and 78 isolators for PWRs were estimated. Assuming a two-man team can test 10 isolators per day, labor requirements for the initial test/inspection required by the generic letter were estimated at 10 man-days/plant for PWRs and 16 man-days/plant for BWRs.

Furthermore, isolators that fail the initial tests must be replaced or repaired. It was conservatively assumed that 25% of the tested isolators will fail the tests; this results in 12 failures at PWRs and 20 failures at BWRs. The cost to purchase, install, test, and perform adequate QC of acceptable replacement isolators was estimated at $10,000/isolator. This included approximately 2 man-days/isolator for replacement. Thus, the total isolator replacement costs were estimated to be $120,000/plant and $200,000/plant for PWRs and BWRs, respectively.

Assuming a cost of $2,270/man-week, the total implementation cost (including hardware) was estimated to be $156,000/plant and $239,000/plant for PWRs and BWRs, respectively.

The generic letter was assumed to include a requirement for annual testing and inspection of all electronic isolators. The industry labor requirements for this activity were estimated to be 1 man-wk/RY for test planning (this is significantly lower than the 4 man-wks for planning the initial test program), plus 10 man-days/RY to conduct the tests at PWRs and 16 man-days/RY to conduct the tests at BWRs. An additional 1 man-wk/RY at all plants to review the test results and prepare a report for the NRC was also included. This resulted in estimated labor requirements of 4 man-wks/RY and 5.2 man-wks/RY for PWRs and BWRs, respectively.

Furthermore, the annual testing program is likely to determine that there are additional failed or suspect isolators that require replacement. For the purposes of this analysis, it was assumed that all the remaining isolators (i.e., other than those that were replaced as a result of the initial test program) will eventually be replaced with acceptable components. The number of remaining isolators to be replaced at PWRs was estimated to be 38 (i.e., 46 - 12) over a 28.8 year period or 1.2/RY. At BWRs, the annual replacement rate is equal to 58 (i.e., 78 - 20) over a 27.4 year period or 2.1/RY. The annual replacement costs at each plant were thus estimated to be $12,000/RY and $21,000/RY for PWRs and BWRs, respectively.

At $2270/man-week, the total cost of maintenance and operation (including hardware) of the possible solution at each plant was estimated to be $21,000/RY and $33,000/RY for PWRs and BWRs, respectively. Using a 5% discount rate, the present worth of the cost associated with plant maintenance and operation for PWRs and BWRs was estimated to be $1[illegible],600/RY and $18,300/RY, respectively.

NRC Cost: It was assumed that the NRC's first activity involves issuance of a bulletin to determine the extent to which potentially susceptible isolators are used in nuclear power plants and to identify the systems in which they are used. It was estimated that 2 man-weeks ($4,000) would be required to prepare the bulletin. Licensee responses to the bulletin must then be reviewed and analyzed by the staff. To perform this review and analysis, it was estimated that 6 man-months ($50,000) of technical support would be needed. The total cost of this activity was estimated to be $54,000.

Assuming that, after analyzing licensee responses, NRC concludes the issue warrants further attention, the second activity involves a research program that would develop the details of the final resolution to this issue. This program would involve two major objectives. First, test procedures and acceptance criteria for isolators would be developed for licensee use in determining the adequacy of their installed isolators. It was estimated that a $50,000 subcontract plus $10,000 for NRC contract support will be needed to accomplish this objective. Second, appropriate hardware fixes would be developed that could resolve the issue. Safety and cost analyses to determine the cost-effectiveness of the proposed hardware fixes would also be necessary. An estimated $150,000 subcontract plus $20,000 for NRC contract support would be needed to accomplish this activity. Thus, the total cost of this activity was estimated to be $230,000.

The next step is to prepare and issue a generic letter to all licensees. Approximately 4 man-wks ($10,000) are estimated to be required to prepare and issue the letter. It was estimated that 6 man-months of staff time would be required to review and evaluate each licensee response. (This is equivalent to a $55,000 subcontract and $10,000 for NRC contract support.) Thus, the total estimated cost for this effort is $75,000.

Based on the above estimates, the total NRC cost for development of the possible solution is $355,000. Averaging this cost over the 134 affected plants results in an NRC cost of $2,650/plant for development.

It was assumed that the staff will review the implementation of the requirements in the generic letter, review the test procedures, review plant-specific implementation plans, and prepare a safety evaluation. The NRC cost for this review was estimated to be 4 man-wks/plant. At $2,270/man-wk, this cost is $9,080/plant.

Furthermore, an additional 0.5 man-wk/RY of NRC effort will be required for an annual review of the operation and maintenance of the solution. Summing this cost over the remaining lives of the affected plants at $2,270/man-wk results in an NRC cost of $32,200/plant. Using a 5% discount rate, the present worth of this review is $17,900/plant.

Therefore, the total NRC cost for the development and implementation of the possible solution was estimated to be approximately $30,000/plant.

Total Cost: The total cost of implementation of the proposed solution was estimated to be $0.6M/plant.

Value/Impact Assessment

Based on the above estimates, the following value/impact scores were calculated for the three cases considered.

Best Estimate: $$S = \frac{1,580 \text{ man-rem/plant}}{\$0.6\text{M/plant}}$$

$$= 2,633 \text{ man-rem/\$M}$$

Lower Bound: $S = \frac{378 \text{ man-rem/plant}}{\$0.6\text{M/plant}}$

$= 630$ man-rem/\$M

Upper Bound: $S = \frac{26{,}752 \text{ man-rem/plant}}{\$0.6\text{M/plant}}$

$= 44{,}587$ man-rem/\$M

Other Considerations

Implementation of the possible solution was assumed to include repair, replacement, and testing of potentially susceptible isolators. This results in labor estimates of 34 man-days/plant for PWRs and 56 man-days/plant for BWRs in radiation zones. Radiation fields of 25 millirem/hr are assumed to exist inside containment where most of the isolators are located. Utilizing a 75% efficiency factor for labor in radiation zones, the occupational dose increase for implementation of the possible solution was estimated to be 9.1 man-rem/plant and 14.9 man-rem/plant for PWRs and BWRs, respectively.

Licensee labor requirements in radiation zones for operation and maintenance of the possible solution includes:

	PWRs (man-days/RY)	BWRs (man-days/RY)
Annual Test Program	10	16
Replacement of Isolators	2.4	4.2
Total:	12.4	20.2

Again, utilizing a 75% efficiency factor for labor in radiation zones and radiation fields of 25 millirem/hr results in an estimated increase in occupational exposure of 3.3 man-rem/RY and 5.4 man-rem/RY for PWRs and BWRs, respectively. Summing these values over the remaining lives of the affected plants (28.8 years for PWRs and 27.4 years for BWRs) results in an increase in ORE of approximately 95 man-rem/plant and 148 man-rem/plant for PWRs and BWRs, respectively.

CONCLUSION

The best estimate of public risk reduction associated with preventing leakage through electrical isolators is significant and indicated a high priority ranking. However, the calculation of risk reduction included a number of conservative assumptions which were also noted during the peer review process. Use of conservative assumptions where real data does not exist will always result in overprediction of potential risk reduction.

In acknowledgement of the conservatisms in the analysis and the peer review comments, a MEDIUM priority ranking was assigned to this issue. This ranking was consistent with the qualitative judgments from the peer review process and was further supported by NRR's stated intention to process a research request to initiate an electrical isolator testing program to improve the current state of knowledge concerning isolator characteristics at less than maximum credible fault levels. Resolution of the issue will also address the safety concern of Issue 156.4.1.

REFERENCES

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986.

98. NUREG-0737, "Clarification of TMI Action Plan Requirements," U.S. Nuclear Regulatory Commission, November 1980, (Supplement 1) January 1983.

1269. NUREG/CR-3453, "Electronic Isolators Used in Safety Systems of U.S. Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1986.

1270. Memorandum for B. Morris from B. Sheron, "Pr sed Generic Issue on Leakage Through Electrical Isolators," June 23, 1987.

ISSUE 150: OVERPRESSURIZATION OF CONTAINMENT PENETRATIONS

DESCRIPTION

Historical Background

This issue was identified[1330] by DSIR/RES and addressed the concern for overpressurization of containment piping penetrations following a containment isolation and subsequent heat-up.

Containment isolation at all nuclear power plants ensures that radioactive materials are contained if an accident or inadvertent release of such materials occurs. Isolation is provided for all piping systems that penetrate the containment. Double barriers are provided to ensure that no single failure of an active component can result in a loss of this isolation function. Typically, this double barrier system is provided by isolation valves inside and outside containment. When containment isolation is required because of an accident or inadvertent release of radioactive materials, these valves are closed to prevent leakage of radioactive materials to the environment.

Safety Significance

Overpressurization of the containment piping penetrations could potentially occur during an accident involving a significant increase in the containment temperature. This might occur when water that is trapped between the inner and outer containment isolation valves is heated and expands. Theoretically, heating a constant volume of water from 100°F and 100 psia to 200°F would increase the pressure to 3000 psia. This pressure increase could fail the penetration or the isolation valves and could provide a direct flow path to the environment from the potentially contaminated containment atmosphere. The pressure increase is mitigated somewhat by the penetration itself expanding because of the temperature increase, as well as the possibility that the isolation valves will not be leak-tight and thus will not pressurize fully.

Possible Solution

A possible solution to this issue is to provide a mechanism for preventing water from becoming trapped or for relieving the pressure that could build up in the piping systems between the inner and outer containment isolation valves. Licensees would need to perform thermal and structural analyses of the penetration systems to determine which penetrations, if any, are susceptible to such failure. A pressure relief system would be needed to prevent the pressure increase from failing the penetration. This pressure relief system could consist of the following:

(a) Check valves inside the reactor building instead of the inner containment isolation valves. These valves would prevent water from becoming trapped between the two isolation valves but is only viable for penetrations with flow into containment.

(b) A method to provide pressure relief, such as a rupture disk or safety valve. A storage tank might be needed to contain blowdown liquid or vapor that would be forced through the pressure relief equipment when the equipment is operated.

PRIORITY DETERMINATION

Assumptions

This issue does not directly impact the potential for a core damage accident but addresses a plant's ability to contain radioactive materials that might be released during a core damage accident. Thus, the only concern is the probability of containment failure resulting from the failure of containment isolation. Therefore, to estimate the potential public risk reduction, the effect of the possible solution on the probability of containment failure, assuming a core damage event has occurred, was evaluated.

There are 90 PWRs and 44 BWRs affected by this issue with average remaining lives of 28.8 and 27.4 years, respectively. Oconee 3 and Grand Gulf 1 were used as the reference PWR and BWR, respectively.

Frequency Estimate

The Oconee 3 PRA[54] addressed failure of, or leakage through, containment penetrations as a potential failure mode of the containment. For Oconee 3, the probability of containment penetration leakage (β), given the occurrence of a core damage accident, was estimated to be 7.3×10^{-3}. This probability was used as the base case value for Oconee 3.

The Grand Gulf 1 PRA[54] does not explicitly evaluate containment penetration leakage because none of the accident sequences involving containment isolation failure were found to be among the dominant sequences from a public risk perspective. However, the Grand Gulf 1 PRA was based on WASH-1400[16] which did assess the conditional probability of containment isolation failure (β). Therefore, the base case value used in this analysis was based on the WASH-1400[16] analysis.

The probability of containment isolation failure is dependent upon the specific core damage sequence that occurs before containment failure.[16] As a result, the containment isolation failure probability depends on the prior success or failure of the various engineered safeguards functions. Based on information in Appendix V of WASH-1400,[16] a value of 4×10^{-2} was selected as a representative probability for containment isolation failure. This value represents the weighted average of the range of possible values based on the number of observations in WASH-1400.[16]

The release categories associated with containment isolation failure for Oconee 3 are PWR-4 and PWR-5. For Grand Gulf 1, the affected release categories were not explicitly stated in NUREG/CR-2800[64] or NUREG/CR-1659.[54] It was noted in WASH-1400[16] and a recent PRA[889] of Oconee 3 that relatively high containment leakage, attainable from failure of containment isolation, would prevent failure of the containment building from potential overpressure caused by hydrogen gas explosions. Consideration of the containment isolation failure mode was therefore incorporated into accident sequences involving containment overpressure events caused by gas generation, as shown in Appendix B of NUREG/CR-2800.[64] A new base case risk value was developed for all BWR accident sequences that involve containment overpressure events. To accomplish this, the core damage sequences presented in NUREG/CR-2800[64] that could result in containment overpressure were modified to incorporate the base case containment isolation failure probability, rather than the containment overpressure probability. This had the effect of creating a set of new accident sequences that included containment isolation failure events.

The adjusted case values of the affected parameters were estimated by adding to the base case values the probability of failure of the penetration system (failure of one or more penetrations) that would arise from overpressurization. A conservative approach was taken to develop a new containment leakage sequence of events that incorporated the potential for overpressurization. This sequence consisted of the following events: (1) containment isolation is successful; (2) water becomes trapped between inner and outer isolation valves; (3) containment heating causes heating and expansion of the water between the isolation valves; and (4) the water expansion causes the penetration to fail, such that a leakage path occurs between the containment atmosphere and the environment. Based on this sequence of events and using the rare event approximation, the additional probability of containment system failure is the product of the following terms:

- N = number of penetrations that are susceptible to overpressurization
- P[1] = probability water becomes trapped between isolation valves
- P[2] = probability inboard and outboard isolation valves are leak-tight
- P[3] = probability penetration overpressurizes to rupture, given that the penetration is leak-tight and full of water.
- P[N] = probabilty that the peretration fails in a manner that results in a leakage path from the containment atmosphere to the environment, given that the penetration ruptures due to overpressurization.

N depends on the types of penetrations and isolation valves at each plant. Only liquid penetrations are susceptible to this type of failure and penetrations provided with check valves are not. To determine the value for N, the description of containment penetrations given in NSAC-60[889] was assumed to represent both PWRs and BWRs. A total of 62 penetrations were listed, of which, 36 were provided with check valves or were not liquid-carrying lines and were not susceptible to this containment failure sequence. Therefore, the value of N is 26.

No information was available to calculate P[1]. Containment isolation valve closures are timed such that one valve closes slightly sooner than the other to prevent water from becoming trapped between the valves. Therefore, this event could be caused by failure of the containment isolation system control logic or circuitry to function as intended, or by failure of the valve to close when intended. P[1] was assigned a value of 0.50.

In this analysis, P[2] was set equal to unity. Thus, no credit was taken for the fact that penetration overpressurization would not occur if one of the isolation valves were not leak-tight. It has been estimated that there is approximately a 30% chance that one of these valves is not leak-tight.[1331]

To assist in estimating P[3], a simplified engineering analysis was performed by DSIR/RES to determine the stress and strain that a penetration would experience, assuming that it is leak-tight and full of water. For the purposes of the analysis, a typical penetration was approximated as a 2-inch diameter, 12 inch long cylinder fabricated of steel and having a yield point of 30 ksi. Assuming that the water was initially at 100°F and its temperature increased to 200°F as a result of an accident, it was calculated that the hoop stress would exceed the yield point. However, the volume of the cylinder would only have to increase 2.6% to accommodate the expansion of the water. This corresponds to a plastic strain of 1.3% in the diameter of the penetration, conservatively assuming no plastic strain in the axial direction. If the water were heated to 300°F, the diametric plastic strain needed to accommodate the expansion of the water would be 3.8%, again conservatively assuming no plastic strain in the axial direction. These values of strain are far below the values that would be expected to cause rupture of the penetration. Also, as expansion of the penetration volume occurs due to plastic deformation, the pressure of the trapped water decreases, further decreasing the likelihood of rupture. Using the above information, but considering that there is some probability that the material used to fabricate the penetration has an undetected flaw, the value of P[3] was estimated as 1×10^{-4}.

For there to be a leak path that satisfies the definition of P[4], there must be a failure inside and outside the containment. One possibility is that the penetration rupture "runs" past the containment vessel. The other possibilities involve failures of both containment isolation valves, or the containment penetration and one isolation valve. Such failures must be simultaneous since the failure of one component relieves the pressure and eliminates the possibility of sequential failures. It was estimated that the value of P[4] is 0.1.

Based on the above estimates, the additional probability of containment system isolation failure is the product of N, P[1], P[2], P[3], and P[4], and is approximately $(26)(0.5)(1.0)(1 \times 10^{-4})(0.1) = 1.3 \times 10^{-4}$.

Consequence Estimate

Incorporating the values into the Oconee 3 and Grand Gulf 1 PRAs results in a potential public risk reduction of 1.3×10^{-2} man rem/RY and 3.0×10^{-3} man-rem/RY, respectively. Thus, the total potential public risk reduction is about 40 man-rem for all 134 affected plants.

Cost Estimate

Industry Cost: To implement the possible solution, licensees will be required to perform analyses to determine if certain penetrations are vulnerable to over-pressurization following containment heat-up. These analyses were estimated to require 4 man-weeks/plant.

In addition, safety analyses and QA-related activities are needed because of the installation of hardware inside containment that requires considerable attention to QA in designing the penetrations' pressure relief systems. Two man-weeks of labor per penetration were estimated for the design and safety analyses. The number of vulnerable penetrations was assumed to be 20% of the total penetrations without check valves, or 7 penetrations. Therefore, at 2 man-weeks/penetration, 14 man-weeks/plant would be required.

Installing new hardware within containment at operating plants (i.e., backfit) will require about 1 man-week of labor in radiation zones; plants under construction (i.e., forward-fit) would not require labor in radiation zones. The material costs were estimated to be $900/penetration and $6,300/plant, including labor for pipefitters, welders, radiation monitoring staff, and instrument technicians. For forward-fit plants, the hardware costs remained the same, but labor costs would be reduced because personnel would not be working in radiation zones. Therefore, a 50% reduction in labor requirements was estimated i.e., 0.5 man-week/plant.

Based on the above estimates, the total labor required is 19 man-weeks/plant for backfit plants and 18.5 man-weeks/plant for forward-fit plants. Therefore, at $2,270/man-week, the estimated industry labor cost was $43,000/backfit-plant and $42,000/forward-fit plant. With a total of 71 backfit plants and 63 forward-fit plants, the total estimated industry implementation cost was $6.5M.

It was estimated that 28 man-hrs/RY will be required to conduct periodic (monthly) testing of the pressure relief system. At $2,270/man-week, this cost is $1,589/RY. For the 134 plants, the total estimated cost for operation and maintenance is $6M. Using a 5% discount rate, the present worth of the recurring costs associated with plant maintenance and operation is $3.3M.

<u>NRC Cost</u>: It was estimated that 5 man-months will be required for the staff to develop acceptable methods, data, and acceptance criteria for licensees to use when evaluating the vulnerability of penetrations to the overpressure phenomenon analyzed in this issue. At $2,270/man-week, the total cost for this development is $54,000.

About 2 man-weeks/plant were estimated for reviewing and evaluating licensee calculations of the stresses within the penetrations and for reviewing the design, safety analyses, and QA documentation for the penetration pressure relief systems. At a cost of $4,500/plant, the total cost for this effort was estimated to be $600,000.

After implementation, the NRC will have to inspect the operation and maintenance of the penetration isolation systems. One man-hr/RY was estimated as sufficient for NRC review of each system. Therefore, the annual labor requirement is 0.18 man-week/RY for seven such systems. At $2,270/man-week, the total cost for the inspection of the 134 affected plants is $1.5M. At a 5% discount rate, this cost is $880,000.

<u>Total Cost</u>: The total estimated industry and NRC cost associated with the possible solution to this issue is $11.3M.

Value/Impact Assessment

Based on an estimated public risk reduction of 40 man-rem and a cost of \$11.3M for the possible solution, the value/impact score was given by:

$$S = \frac{40 \text{ man-rem}}{\$11.3\text{M}}$$

$$= 3.5 \text{ man-rem/\$M}$$

Other Considerations

(1) For backfit plants, an estimated 5 man-hr/penetration of labor in a radiation zone would be required and, assuming 7 penetrations/plant, the total ORE would be 35 man-hr/plant. The dose rate was assumed to be 25 millirem/hr, which is representative of the dose rate inside containment during reactor shutdowns. The implementation dose was therefore estimated to be about 0.9 man-rem/plant. For the 71 backfit plants, the ORE is greater than the total averted public dose.

(2) Routine testing and inspection of the penetration pressure relief systems were assumed to occur once per 30 days, similar to testing of the containment isolation valves. The testing was assumed to be performed by a 2-man team and to last about 10 minutes. Assuming 7 penetrations to be tested, the total operation and maintenance dose was estimated to be about 0.7 man-rem/RY.

(3) The public risk reduction estimated for this issue was overestimated for several reasons. First, no credit was taken for the protection from overpressure that would be provided if one of the isolation valves were not leak-tight. Investigation of a previous containment issue indicated that there was approximately a 30% chance that one of the valves would not be leak-tight.

(4) The costs were estimated assuming that the containment isolation systems were located away from the containment building wall. This is not the case for many of the isolation valves which are located adjacent to the containment walls. This assumption tends to minimize the costs because they would be clearly higher when the containment structure must be modified to accommodate the pressure relief system. As a result, the cost estimates provided above were believed to be low.

CONCLUSION

The estimated public risk associated with overpressurization of containment penetrations is not significant. Based on the value/impact assessment and the staff's simplified engineering analysis, this issue was placed in the DROP category.

REFERENCES

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

54. NUREG/CR-1659, "Reactor Safety Study Methodology Applications Program," U.S. Nuclear Regulatory Commission, (Volume 1) April 1981, (Volume 2) May 1981, (Volume 3) June 1982, (Volume 4) November 1981.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986.

889. NSAC-60, "A Probabilistic Risk Assessment of Oconee Unit 3," Electric Power Research Institute, June 1984.

1330. Memorandum for T. King from W. Minners, "Overpressurization of Containment Penetrations," March 16, 1989.

1331. NUREG/CR-4220, "Reliability Analysis of Containment Isolation Systems," U.S. Nuclear Regulatory Commission, June 1985.

ISSUE 151: RELIABILITY OF ANTICIPATED TRANSIENT WITHOUT SCRAM RECIRCULATION PUMP TRIP IN BWRs

DESCRIPTION

Historical Background

This issue was identified in a DSIR/RES memorandum[1329] which addressed the concern for the reliability of breakers used to trip the recirculation pumps at high pressure or low water level signals during ATWS mitigation in BWRs. A staff review of BWRs that experienced failures of breakers in the recirculation pump trip (RPT) system was documented in AEOD/E804.[1328]

If a plant transient requiring a reactor scram occurs and the scram function does not occur, then an ATWS event exists. To lessen the effects of an ATWS event, negative reactivity must be added to the reactor core by tripping the recirculation pumps. Negative reactivity is added as a result of the ensuing steam voiding in the core area as the core flow decreases, thereby decreasing the power generation and limiting the power or pressure disturbance.

Plants equipped with GE AKF-25 circuit breakers have experienced failures of the field breakers in the RPT system which were caused by binding of the trip latch mechanism and misadjustment of the breakers' mechanical linkage. GE issued a service information letter which attributed the circuit breaker failures to misadjustment or lubrication problems and suggested corrective actions and maintenance practices to improve the breakers' performance. In addition, Information Notice No. 87-12 was issued by the NRC to all BWR licensees to alert them of potential problems with these circuit breakers.

Safety Significance

A RPT breaker is included in the design of plants to automatically trip the recirculation pump on high vessel pressure or low reactor water level during an ATWS event. If the RPT breaker fails to trip on demand, the reactor could experience continued power generation resulting in high suppression pool temperature. This issue affects BWRs only.

Possible Solution

A possible solution is based on the corrective actions implemented at the Pilgrim Nuclear Power Station and involves installing a redundant ATWS trip signal that would interrupt current to the recirculation pumps. Specifically, a new trip coil initiated by an ATWS signal would be installed in each recirculation pump motor-generator set drive motor. During an ATWS event, both the recirculation pump field breaker and the motor-generator set supply breaker would receive trip signals, if either high vessel pressure or low reactor water level was reached. Thus, the reliability of the RPT system would be increased and the potential for reaching an unacceptable suppression pool temperature during an ATWS event would be diminished.

PRIORITY DETERMINATION

Frequency Estimate

Several designs are currently used in accomplishing the RPT function; however, the GE AKF-25 breaker is used primarily in BWR/3 and BWR/4 designs. Pilgrim (BWR/3) was excluded as an affected plant because it had already implemented the proposed solution. There are 25 BWR/3 and BWR/4 plants with an average remaining life of 26.5 years affected by this issue:

Plant Type	Number of Plants	Average Life Expectancy (yr)
BWR/3	6	15.4
BWR/4	19	30.0

The issue affects a plant's ability to render the reactor subcritical following an ATWS event. Since the reactor subcriticality analysis[64] of Grand Gulf 1 (BWR/6) is analogous to the WASH-1400[16] analysis of Peach Bottom (BWR/3), the Grand Gulf 1 analysis was used to quantify the influence of the solution on accident frequency and consequence. Thus, the accident sequence affected by this issue is a scram followed by a failure to render the reactor subcritical and is depicted as $T_{23}C$. The transient-initiating event (T_{23}) has a frequency of 7 events/RY.[64]

Failure of reactor subcriticality (Event C) has been probabilistically modelled as the product of the following: (1) failure of the RPS; and (2) failure of the RPT or failure of the operator to take the appropriate actions to shut down the reactor, given RPS failure.[64] From WASH-1400,[16] the failure rate of the RPS was given as 7.7×10^{-6}/demand. The operator error, which was estimated to be 0.1, dominated Item 2 above. The RPT circuit breaker failure rate was given as 3×10^{-3}/demand.

To derive the base case value for event C, the RPT failure rate was modified to 5×10^{-2}/demand, which reflects the lower reliability of the GE AKF-25 circuit breaker.[1328] Therefore, the base case value for event C is approximately $(7.7 \times 10^{-6})[0.1 + (5 \times 10^{-2})]$/demand or 1.16×10^{-6}/demand.

The adjusted case value for Event C assumes the installation of a redundant ATWS RPT signal. Therefore, the event (recirculation pump fails to trip) requires failure of both RPT subsystems. Assuming that the RPT subsystems are independent and using the GE AKF-25 circuit breaker reliability value, the RPT failure frequency is $(5 \times 10^{-2})(5 \times 10^{-2})$/demand or 2.5×10^{-3}/demand. This value does not take credit for potential increases in reliability that could result from improved test and maintenance programs or from changing to a more reliable RPT circuit breaker. However, the estimate also does not consider the potential decrease in RPT system reliability due to common cause failure mechanisms. Thus the adjusted case value of Event C is about $(7.7 \times 10^{-6}) \times [0.1 + (2.5 \times 10^{-3})]$/demand or 7.9×10^{-7}/demand.

Therefore, the $T_{23}C$ accident sequence frequency is 8.1×10^{-6}/RY for the base case and 5.5×10^{-6}/RY for the adjusted case. The total reduction in accident frequency is 2.6×10^{-6}/RY.

Consequence Estimate

Accident sequence $T_{23}C$ falls into the BWR-2 release category (7.1×10^6 man-rem/event).[64] The total public dose associated with the base case and adjusted case is 57.4 man-rem/RY and 39.2 man-rem/RY, respectively. Thus, the estimated public dose reduction from implementing the possible solution is 18.2 man-rem/RY and the total risk reduction for 25 reactors with an average remaining life of 26.5 years is 12,000 man-rem.

Cost Estimate

Industry Cost: The cost to implement the possible solution will vary from plant to plant. The following Pilgrim 1 actual costs were used to estimate the industry cost: (1) engineering = \$390,000; (2) labor = \$66,000; (3) hardware = \$10,000. Thus, the implementation cost was estimated to be about \$466,000/plant for a total industry cost of about \$11M (excluding Pilgrim).

It is expected that the installed redundant RPT subsystem will only be operated for testing purposes and operation costs are thus negligible. In addition, the testing and maintenance procedures for the redundant RPT subsystem will be very similar to existing RPT subsystems and, therefore, should require very little additional development time. Thus, testing and maintenance are each estimated to require 0.5 man-week/RY, resulting in a plant cost of \$2,270/RY and a total industry cost of about \$1.5M. Thus, the total industry implementation, operation, and maintanance cost is \$12.5M.

NRC Cost: Development of the solution is estimated to require one man-year of contractor labor, at a cost of \$100,000/man-year, to complete an evaluation of the solution and any potential alternatives (e.g., enhanced test/maintenance or replacing the degraded RPT breakers with more reliable models). This study would also need to include a preliminary review of plant designs to determine the technical feasibility of the proposed modifications. Development of the solution would also include issuing an NRC generic letter to the affected plants, which is estimated to cost about \$11,000.[961]

Review of the proposed plant modifications was estimated to take 5 man-weeks/plant for a total NRC review cost of \$280,000 for the 25 affected plants. Thus, the total NRC cost for development and review is \$3[illegible],000.

Total Cost: The total industry and NRC cost associated with the possible solution is approximately \$13M.

Value/Impact Assessment

Based on an estimated public risk reduction of 12,000 man-rem and a cost of \$13M associated with the possible solution, the value/impact score is given by:

$$S = \frac{12,000 \text{ man-rem}}{\$13M}$$

$$\approx 923 \text{ man-rem}/\$M$$

Other Considerations

The scram frequency estimate used in the above calculations is considerably greater than that reflected in recent performance indicator reports. In addition, the RPS failure rate was originally developed for WASH-1400[16] prior to the ATWS rulemaking and is also quite outdated. As a result, the RPS reliability goal from the ATWS rulemaking proceedings was utilized as a conservative value and the risk reduction calculations were repeated using: (1) a SCRAM frequency of 3.1/RY, derived from data in the 1988 AEOD Annual Report and Part 1 of the Third Quarter 1990 AEOD report, "Performance Indicators for Operating Commercial Nuclear Power Reactors"; and (2) an RPS failure rate of 3×10^{-5}/demand from the ATWS rulemaking proceeding.[704] These estimates resulted in a potential risk reduction of 20,800 man-rem and a value/impact score of 1600 man-rem/$M.

CONCLUSION

Based on the potential public risk reduction associated with this issue and the other considerations above, the issue was given a MEDIUM priority ranking.

REFERENCES

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

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986.

704. NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1978, (Vol. 2) April 1978, (Vol. 3) December 1978, (Vol. 4) March 1980.

961. NUREG/CR-4627, "Generic Cost Estimates," U.S. Nuclear Regulatory Commission, June 1986.

1328. AEOD/E804, "Reliability of Non-Safety Related Field Breakers During ATWS Events," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, July 26, 1988.

1329. Memorandum for T. King from K. Kniel, "Request for Prioritization of New Generic Safety Issue 'Reliability of Recirculation Pump Trip (RPT) During an ATWS,'" March 17, 1989.

ISSUE 156: SYSTEMATIC EVALUATION PROGRAM

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

In SECY-84-133, the staff presented the 27 SEP issues to the Commission as part of a proposal for an Integrated Safety Assessment Program (ISAP). The intent of the ISAP was to review safety issues for a specific plant in an integrated manner. Two SEP plants participated in the ISAP pilot efforts. Following the review of these two pilot plants, ISAP was discontinued.

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

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

The staff completed this effort and placed each SEP issue into one of the following categories: (1) issues that had been completely resolved (i.e., necessary corrective actions had been identified by the staff, transmitted to licensees, and implemented by licensees); (2) issues that were of such low safety significance so as to require no further regulatory action; (3) issues that were unresolved, but for which the staff had identified existing regulatory programs that cover the scope of the technical concerns and whose implementation would resolve the specific SEP issue (such as IPE and IPEEE); and (4) issues that were unresolved and regulatory actions to resolve the issues had not been identified. The 27 SEP issues and applicable regulatory programs were summarized and presented in SECY-90-343.[1351] The staff concluded that the 22 SEP issues in Categories 3 and 4 remained unresolved for purposes of justifying the adequacy of the CLB for some portion of the 41 older, non-SEP plants. The following is an evaluation of these 22 issues.

ISSUE 156.1.1: SETTLEMENT OF FOUNDATIONS AND BURIED EQUIPMENT

This issue is being prioritized.

ISSUE 156.1.2: DAM INTEGRITY AND SITE FLOODING

This issue is being prioritized.

ISSUE 156.1.3: SITE HYDROLOGY AND ABILITY TO WITHSTAND FLOODS

This issue is being prioritized.

ISSUE 156.1.4: INDUSTRIAL HAZARDS

This issue is being prioritized.

ISSUE 156.1.5: TORNADO MISSILES

This issue is being prioritized.

ISSUE 156.1.6: TURBINE MISSILES

DESCRIPTION

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

As a result of turbine disc failures at two nuclear plants and a number of non-nuclear plants prior to 1973, the staff believed that high energy missiles could be generated from steam turbines with the potential for causing failures in safety-related systems. The two areas of concern are: (1) failures at design overspeed because of degraded disc material, poor in-service inspection (ISI) of flaws, or chemistry conditions leading to stress corrosion cracking

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

CONCLUSION

The safety concerns of this issue were addressed in the evaluation of Issue A-37, "Turbine Missiles," which focused primarily on plants licensed prior to November 1976; SRP[11] requirements for turbine design were issued for use by CP applicants after this date. Based on the historical failure rate of turbines used in the evaluation, Issue A-37 was determined to have little safety significance. No new data were provided in SECY-90-343[1351] that would change this conclusion. Therefore, this issue was DROPPED from further consideration.

ISSUE 156.2.1: SEVERE WEATHER EFFECTS ON STRUCTURES

This issue is being prioritized.

ISSUE 156.2.2: DESIGN CODES, CRITERIA, AND LOAD COMBINATIONS

This issue is being prioritized.

ISSUE 156.2.3: CONTAINMENT DESIGN AND INSPECTION

This issue is being prioritized.

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

This issue is being prioritized.

ISSUE 156.3.1.1: SHUTDOWN SYSTEMS

This issue is being prioritized.

ISSUE 156.3.1.2: ELECTRICAL INSTRUMENTATION AND CONTROL

This issue is being prioritized.

ISSUE 156.3.2: SERVICE AND COOLING WATER SYSTEMS

This issue is being prioritized.

ISSUE 156.3.3: VENTILATION SYSTEMS

This issue is being prioritized.

ISSUE 156.3.4: ISOLATION OF HIGH AND LOW PRESSURE SYSTEMS

DESCRIPTION

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

Issue 105, "Interfacing Systems LOCA in LWRs," is currently being resolved and is concerned with the possible breach of those interfacing boundaries that are created by a series of pressure isolation valves (PIVs) and the consequences of failure of a boundary by mechanical failure, human error, or external event. Thus, Issue 105 covers all interfacing systems, including those identified in Issue 156.3.4. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 are affected by this issue.

CONCLUSION

The safety concern of Issue 156.3.4 is being addressed in the resolution of Issue 105, "Interfacing Systems LOCA in LWRs." Therefore, Issue 156.3.4 was DROPPED from further pursuit as a new and separate issue.

ISSUE 156.3.5: AUTOMATIC ECCS SWITCHOVER

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351]

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

CONCLUSION

Issue 24 is currently scheduled for resolution and is directed at studying the merits of manual, automatic, and semi-automatic ECCS switchover to recirculation. All 41 plants affected by this issue will be considered in the resolution of Issue 24. Thus, Issue 156.3.5 will be covered in Issue 24.

ISSUE 156.3.6.1: EMERGENCY AC POWER

This issue is being prioritized.

ISSUE 156.3.6.2: EMERGENCY DC POWER

This issue is being prioritized.

ISSUE 156.3.8: SHARED SYSTEMS

This issue is being prioritized.

ISSUE 156.4.1: RPS AND ESFS ISOLATION

DESCRIPTION

This issue is one of the three Category 4 issues identified by NRR in SECY-90-343.[1351] The safety concern is that, in the event of non-safety system failures, the lack of isolation devices could result in the propagation of faults to safety systems and common cause failures may result. In its study, the staff found that approximately 39 plants at 28 sites were not required to meet IEEE 279-1971[397] and have not been reviewed for this safety concern since the time of their licensing.

Non-safety systems generally receive control signals from the reactor protection system (RPS) and engineered safety features (ESF) sensor current loops. The non-safety circuits are required to be isolated to ensure the independence of the RPS and ESF channels. Requirements for the d[illegible] and qualification of isolation devices are quite specific. Evaluation [illegible] the quality of isolation devices is not the safety issue of concern; rather, the issue is the existence of isolation devices which will preclude the propagation of non-safety system faults to safety systems.

CONCLUSION

The safety concerns of leakage through electrical isolators in instrumentation circuits and electrical isolation in plants not required to meet IEEE 279-1971[397] are currently being addressed in the resolution of Issue 142, "Leakage Through Electrical Isolators in Instrumentation Circuits." Therefore, Issue 156.4.1 will be covered in the resolution of Issue 142.

ISSUE 156.4.2: TESTING OF THE RPS AND ESFS

This issue is being prioritized.

ISSUE 156.6.1: PIPE BREAK EFFECTS ON SYSTEMS AND COMPONENTS

This issue is being prioritized.

REFERENCES

397. IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations (ANSI N42.7-1972)," The Institute of Electrical and Electronics Engineers, Inc., 1971.

1351. SECY-90-343, "Status of the Staff Program to Determine How the Lessons Learned from the Systematic Evaluation Program Have Been Factored into the Licensing Bases of Operating Plants," October 4, 1990.

APPENDIX B

APPLICABILITY OF NUREG-0933 ISSUES TO OPERATING AND FUTURE PLANTS

This appendix contains a listing of those safety issues that are applicable to operating plants as well as future plants. The priority designations for all issues are consistent with those listed in Table II of the Introduction. This listing includes: issues that have been resolved with new requirements [NOTE 3(a)]; USI, HIGH and MEDIUM priority issues that are under development; nearly-resolved issues (NOTES 1 and 2) whose impact is not yet known, and issues that are scheduled for prioritization (NOTE 4). 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.

Legend

NOTES:
1 - Possible Resolution Identified for Evaluation
2 - Resolution Available (Documented in NUREG, NRC Memorandum, SER or equivalent)
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

B&W	- Babcock & Wilcox Company
CE	- Combustion Engineering Company
GE	- General Electric Company
HIGH	- High Safety Priority
I	- Resolved TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737[98]
MEDIUM	- Medium Safety Priority
MPA	- Multiplant Action
NA	- Not Applicable
TBD	- To Be Determined
USI	- Unresolved Safety Issue
W	- Westinghouse Electric Corporation

Action Plan Item/ Issue No	Title	Safety Priority/ Status	Affected NSSS Vendor BWR	PWR	Operating Plants- MPA No.	Operating Plants- Effective Date	Future Plants- Effective Date
	TMI ACTION PLAN ITEMS						
I.A	OPERATING PERSONNEL						
I.A.1	Operating Personnel and Staffing						
I.A.1.1	Shift Technical Advisor	I	All	All	F-01	9/13/79	9/27/79
I.A.1.2	Shift Supervisor Administrative Duties	I	All	All		9/13/79	9/27/79
I.A.1.3	Shift Manning	I	All	All	F-02	7/31/80	5/26/80
I.A.1.4	Long-Term Upgrading	NOTE 3(a)	All	All		4/28/83	4/28/83
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	3/28/80	3/28/80
I.A.2.1(2)	Training	I	All	All	F-03	3/28/80	3/28/80
I.A.2.1(3)	Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	I	All	All	F-03	3/28/80	3/28/80
I.A.2.3	Administration of Training Programs	I	All	All		3/28/80	3/28/80
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		-	-
I.A.3	Licensing and Requalification of Operating Personnel						
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	I	All	All		3/28/80	3/28/80
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		4/-/81	3/28/81
I.A.4.2	Long-Term Training Simulator Upgrade	-					
I.A.4.2(1)	Research on Training Simulators	NOTE 3(a)	All	All		4/-/87	4/-/87
I.A.4.2(2)	Upgrade Training Simulator Standards	NOTE 3(a)	All	All		4/-/81	4/-/81
I.A.4.2(3)	Regulatory Guide on Training Simulators	NOTE 3(a)	All	All		4/-/81	4/-/81
I.A.4.2(4)	Review Simulators for Conformance to Criteria	NOTE 3(a)	All	All		3/25/87	[illegible]/25/87
I.C	OPERATING PROCEDURES						
I.C.1	Short-Term Accident Analysis and Procedures Revision	-					
I.C.1(1)	Small Break LOCAs	I	All	All		9/13/79	9/13/79
I.C.1(2)	Inadequate Core Cooling	I	All	All	F-04	9/13/79	9/13/79
I.C.1(3)	Transients and Accidents	I	All	All	F-05	9/13/79	9/27/79
I.C.2	Shift and Relief Turnover Procedures	I	All	All		9/13/79	9/27/79
I.C.3	Shift Supervisor Responsibilities	I	All	All		9/13/79	9/27/79
I.C.4	Control Room Access	I	All	All		9/13/79	9/27/79
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	I	All	All	F-06	5/7/90	6/26/80

Appendix B (Continued)

Action Plan Item/ Issue No.	Title	Safety Priority/ Status	Affected NSSS Vendor BWR	PWR	Operating Plants- MPA No	Operating Plants- Effective Date	Future Plants- Effective Date
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	6/26/80
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	I	All	All		NA	6/26/80
I.C.9	Long-Term Program Plan for Upgrading of Procedures	NOTE 3(a)	All	All		9/13/79	6/-/85
I.D	CONTROL ROOM DESIGN						
I.D.1	Control Room Design Reviews	I	All	All	F-08	6/26/80	6/26/80
I.D.2	Plant Safety Parameter Display Console	I	All	All	F-09	6/26/80	6/26/80
I.D.3	Safety System Status Monitoring	MEDIUM	All	All		-	-
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.D.5(3)	On-Line Reactor Surveillance System	NOTE 3	All	All			
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	7/-/81
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	NOTE 3(a)	All	All		NA	7/-/81
I.F.2(6)	Increase the Size of Licensees' QA Staff	NOTE 3(a)	All	All		NA	7/-/81
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	NOTE 3(a)	All	All		NA	7/-/81
I.G	PREOPERATIONAL AND LOW-POWER TESTING						
I.G.1	Training Requirements	I	All	All		NA	6/26/80
I.G.2	Scope of Test Program	NOTE 3(a)	All	All		NA	7/-/81
II.B	CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW						
II.B.1	Reactor Coolant System Vents	I	All	All	F-10	9/13/79	9/27/79
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	I	All	All	F-11	9/13/79	9/27/79
II.B.3	Post-Accident Sampling	I	All	All	F-12	9/13/79	9/27/79
II.B.4	Training for Mitigating Core Damage	I	All	All	F-13	3/28/80	3/28/80
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

Action Plan Item/ Issue No.	Title	Safety Priority/ Status	Affected NSSS Vendor BWR	PWR	Operating Plants-MPA No.	Operating Plants-Effective Date	Future Plants-Effective Date
II.D	REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES						
II.D.1	Testing Requirements	I	All	All	F-14	9/13/79	9/27/79
II.D.3	Relief and Safety Valve Position Indication	I	All	All		7/21/79	9/27/79
II.E	SYSTEM DESIGN						
II.E.1	Auxiliary Feedwater System						
II.E.1.1	Auxiliary Feedwater System Evaluation	I	NA	All	F-15	3/10/80	3/10/80
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	I	NA	All	F-16, F-17	9/13/79	9/27/79
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	NOTE 3(a)	All	All		NA	7/-/81
II.E.3	Decay Heat Removal						
II.E.[illegible]	Reliability of Power Supplies for Natural Circulation	I	NA	All		9/13/79	9/27/79
II.E.4	Containment Design						
II.E.4.1	Dedicated Penetrations	I	All	All	F-18	9/13/79	9/27/79
II.E.4.2	Isolation Dependability	I	All	All	F-19	9/13/79	9/27/79
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		9/27/79	NA
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		TBD	TBD
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	9/13/79	9/27/79
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate [illegible] Cooling	I	All	All	F-26	7/2/79	9/27/79
II.F.3	Instruments for Monitoring Accident Conditions	NOTE 3(a)	All	All		NA	12/-/80

Action Plan Item/ Issue No.	Title	Safety Priority/ Status	Affected NSSS Vendor BWR	PWR	Operating Plants-MPA No.	Operating Plants-Effective Date	Future Plants-Effective Date
II.G	ELECTRICAL POWER						
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	I	NA	All		9/13/79	9/27/79
II.H	TMI-2 CLEANUP AND EXAMINATION						
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	HIGH	NA	B&W		5/-/80	NA
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		7/31/91	7/31/91
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		3/31/80	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		3/31/80	NA
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	NOTE 3(a)	NA	All		3/31/80	NA
II.K.1(4)	Review Operating Procedures and Training Instructions	NOTE 3(a)	All	All		3/31/80	NA
II.K.1(5)	Safety-Related Valve Position Description	NOTE 3(a)	All	All		3/31/80	3/31/80
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	NOTE 3(a)	All	All		3/31/80	NA
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	NOTE 3(a)	NA	B&W		3/31/80	NA
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	NOTE 3(a)	NA	B&W		3/31/80	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		3/31/80	NA
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	NOTE 3(a)	All	All		3/31/80	3/31/80
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		3/31/80	NA

Action Plan Item/ Issue No.	Title	Safety Priority/ Status	Affected NSSS Vendor BWR	Affected NSSS Vendor PWR	Operating Plants- MPA No.	Operating Plants- Effective Date	Future Plants- Effective Date
II.G	ELECTRICAL POWER						
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	I	NA	All		9/13/79	9/27/79
II.H	TMI-2 CLEANUP AND EXAMINATION						
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	HIGH	NA	B&W		5/-/80	NA
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	NO & 3(a)	All	All		7/31/91	7/31/91
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		3/31/80	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		3/31/80	NA
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	NOTE 3(a)	NA	All		3/31/80	NA
II.K.1(4)	Review Operating Procedures and Training Instructions	NOTE 3(a)	All	All		3/31/80	NA
II.K.1(5)	Safety-Related Valve Position Description	NOTE 3(a)	All	All		3/31/80	3/31/80
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	NOTE 3(a)	All	All		3/31/80	NA
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	NOTE 3(a)	NA	B&W		3/31/80	NA
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	NOTE 3(a)	NA	B&W		3/31/80	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		3/31/80	NA
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	NOTE 3(a)	All	All		3/31/80	3/31/80
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		3/31/80	NA

Action Plan Item/ Issue No.	Title	Safety Priority/ Status	Affected NSSS Vendor BWR	PWR	Operating Plants- MPA No.	Operating Plants- Effective Date	Future Plants- Effective Date
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		1/1/81	1/1/81
II.K.1(14)	Review Operating Modes and Procedure to Deal with Significant Amounts of Hydrogen	NOTE 3(a)	GE	CE, W		3/31/80	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		3/31/80	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		3/31/80	3/31/80
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		3/31/80	3/31/80
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		3/31/80	3/31/80
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	NOTE 3(a)	All	NA		3/31/80	3/31/80
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		1/1/81	1/1/82
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	

Action Plan Item/ Issue No.	Title	Safety Priority/ Status	Affected NSSS Vendor BWR	Affected NSSS Vendor PWR	Operating Plants- MPA No.	Operating Plants- Effective Date	Future Plants- Effective Date
II.K.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	1/1/81	1/1/81
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	I	NA	B&W	F-28	1/1/81	1/1/81
II.K.2(11)	Operator Training and Drilling	I	NA	B&W	F-29	1/1/81	1/1/81
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	1/1/81	1/1/81
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	I	NA	B&W	F-31	1/1/81	1/1/81
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes After Primary System Voiding	I	NA	B&W		6/1/80	6/1/80
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA With Loss of Offsite Power	I	NA	B&W	F-32	6/1/80	6/1/80
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	1/1/81	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	1/1/81	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	7/1/81	7/1/81
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	I	NA	All	F-37	1/1/81	1/1/81
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	I	All	All	F-38	4/1/80	4/1/80
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	I	NA	All	F-39, G-01	1/1/81	1/1/81
II.K.3(7)	Evaluation of PORV Opening Probability During Overpressure Transient	I	NA	B&W		1/1/81	1/1/81
II.K.3(9)	Proportional Integral Derivative Controller Modification	I	NA	W	F-40	7/1/80	7/1/80
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	7/1/80	7/1/80
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	I	GE	NA	F-43	10/1/80	10/1/80
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	I	GE	NA	F-44	1/1/81	NA
II.K.3(15)	Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	I	GE	NA	F-45	1/1/81	1/1/81
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification	I	GE	NA	F-46	1/1/81	1/1/81
II.K.3(17)	Report on Outage of ECC Systems - Licensee Report and Technical Specification Changes	I	GE	NA	F-47	1/1/81	1/1/81

Appendix B (Continued)

Action Plan Item/ Issue No.	Title	Safety Priority/ Status	Affected NSSS Vendor BWR	Affected NSSS Vendor PWR	Operating Plants-MPA No.	Operating Plants-Effective Date	Future Plants-Effective Date
II.K.3(18)	Modification of ADS Logic - Feasibility Study and Modification for Increased Diversity for Some Event Sequences	I	GE	NA	F-48	1/1/81	1/1/81
II.K.3(19)	Interlock on Recirculation Pump Loops	I	GE	NA	F-49	1/1/81	NA
II.K.3(20)	Loss of Service Water for Big Rock Point	I	GE	NA		1/1/81	NA
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level - Design and Modification	I	GE	NA	F-50	1/1/81	1/1/81
II.K.3(22)	Automatic Switchover of RCIC System Suction - Verify Procedures and Modify Design	I	GE	NA	F-51	1/1/81	1/1/81
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	I	GE	NA	F-52	1/1/82	1/1/82
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	I	GE	NA	F-53	1/1/82	1/1/82
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	I	GE	NA	F-54	10/1/80	10/1/80
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	I	GE	NA	F-55	1/1/82	1/1/82
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	I	GE	NA	F-56	4/1/81	NA
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	I	All	All	F-57	1/1/83	1/1/83
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	I	All	All	F-58	1/1/83	1/1/83
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	I	GE	NA	F-59	1/1/81	1/1/81
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	I	GE	NA	F-60	1/1/81	1/1/81
II.K.3(46)	Response to List of Concerns from ACRS Consultant	I	GE	NA	F-61	7/1/80	7/1/80
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	I	GE	NA	F-62	10/1/80	NA
III.A	EMERGENCY PREPAREDNESS AND RADIATION EFFECTS						
III.A.1	Improve Licensee Emergency Preparedness - Short Term						
III.A.1.1	Upgrade Emergency Preparedness	-					
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	I	All	All		10/10/79	8/19/80
III.A.1.2	Upgrade Licensee Emergency Support Facilities	-					
III.A.1.2(1)	Technical Support Center	I	All	All	F-63	9/13/79	9/27/79
III.A.1.2(2)	On-Site Operational Support Center	I	All	All	F-64	9/13/79	9/27/79
III.A.1.2(3)	Near-Site Emergency Operations Facility	I	All	All	F-65	9/13/79	9/27/79
III.A.2	Improving Licensee Emergency Preparedness-Long Term						
III.A.2.1	Amend 10 CFR 50 and 10 CFR 50, Appendix E	-					
III.A.2.1(1)	Publish Proposed Amendments to the Rules	I	All	All			
III.A.2.1(2)	Conduct Public Regional Meetings	I	All	All			

Action Plan Item/ Issue No.	Title	Safety Priority/ Status	Affected NSSS Vendor BWR	PWR	Operating Plants- MPA No.	Operating Plants- Effective Date	Future Plants- Effective Date
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules	I	All	All			
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	I	All	All	F-67		
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		7/2/79	9/27/79
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	9/13/79	9/27/79
III.D.3.3(2)	Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	NOTE 3(a)	All	All		9/13/79	9/27/79
III.D.3.3(3)	Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation-Monitoring Instruments	NOTE 3(a)	All	All		9/13/79	9/27/79
III.D.3.3(4)	Issue a Regulatory Guide	NOTE 3(a)	All	All		9/13/79	9/27/79
III.D.3.4	Control Room Habitability	I	All	All	F-70	5/7/80	6/26/80

TASK ACTION PLAN ITEMS

Action Plan Item/ Issue No.	Title	Safety Priority/ Status	BWR	PWR	Operating Plants- MPA No.	Operating Plants- Effective Date	Future Plants- Effective Date
A-1	Water Hammer (former USI)	NOTE 3(a)	All	All		NA	3/15/84
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems (former USI)	NOTE 3(a)	NA	All	D-10	1/-/81	1/-/81
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	NOTE 3(a)	NA	W		4/17/85	4/17/85
A-4	CE Steam Generator Tube Integrity (former USI)	NOTE 3(a)	NA	CE		4/17/85	4/17/85
A-5	B&W Steam Generator Tube Integrity (former USI)	NOTE 3(a)	NA	B&W		4/17/85	4/17/85
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	8/-/82	8/-/82
A-8	Mark II Containment Pool Dynamic Loads - Long Term Program (former USI)	NOTE 3(a)	GE	NA		8/-/81	8/-/81
A-9	ATWS (former USI)	NOTE 3(a)	All	All		6/26/84	6/26/84
A-10	BWR Feedwater Nozzle Cracking (former USI)	NOTE 3(a)	All	NA	B-25	11/-/80	11/-/80

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
	NEW GENERIC ISSUES						
2.	Failure of Protective Devices on Essential Equipment	NOTE 4	All	All		TBD	TBD
15.	Radiation Effects on Reactor Vessel Supports	HIGH	All	All		TBD	TBD
23.	Reactor Coolant Pump Seal Failures	HIGH	All	All		TBD	TBD
24.	Automatic Emergency Core Cooling System Switch to Recirculation	MEDIUM	NA	All		TBD	TBD
25.	Automatic Air Header Dump on BWR Scram System	NOTE 3(a)	All	NA		1/9/81	1/9/81
40.	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	NOTE 3(a)	All	NA	B-65	8/31/81	8/31/81
41.	BWR Scram Discharge Volume Systems	NOTE 3(a)	All	NA	B-58	12/9/80	NA
43.	Reliability of Air Systems	NOTE 3(a)	All	All		8/8/88	NA
45.	Inoperability of Instrumentation Due to Extreme Cold Weather	NOTE 3(a)	All	All		NA	9/1/83
51.	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	NOTE 3(a)	All	All		07/18/89	07/18/89
57.	Effects of Fire Protection System Actuation on Safety-Related Equipment	MEDIUM	All	All		TBD	TBD
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 2	NA	W		TBD	TBD
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
76.	Instrumentation and Control Power Interactions	NOTE 4	All	All		TBD	TBD
78.	Monitoring of Fatigue Transient Limits for Reactor Coolant System	NOTE 4	All	All		TBD	TBD
79.	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	MEDIUM	NA	B&W		TBD	TBD
83.	Control Room Habitability	NOTE 1	All	All		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 4	All	All		TBD	TBD
93.	Steam Binding of Auxiliary Feedwater Pumps	NOTE 3(a)	NA	All		TBD	TBD

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
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		10/17/88	NA
103.	Design for Probable Maximum Precipitation	NOTE 3(a)	All	All		10/19/89	10/19/89
105.	Interfacing Systems LOCA at BWRs	HIGH	All	NA		TBD	TBD
106.	Piping and Use of Highly Combustible Gases in Vital Areas	MEDIUM	All	All		TBD	TBD
110.	Equipment Protective Devices on Engineered Safety Features	NOTE 4	All	All		TBD	TBD
113.	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	HIGH	All	All		TBD	TBD
118.	Tendon Anchorage Failure	NOTE 4	All	All		TBD	TBD
120.	On-Line Testability of Protection Systems	MEDIUM	All	All		TBD	TBD
121.	Hydrogen Control for Large, Dry PWR Containments	HIGH	All	All		TBD	TBD
124.	Auxiliary Feedwater System Reliability	NOTE 3(a)	All	All		TBD	TBD
128.	Electrical Power Reliability	NOTE 3(a)	All	All		04/29/91	04/29/91
130.	Essential Service Water Pump Failures at Multiplant Sites	NOTE 3(a)	NA	All		09/19/91	09/19/91
132.	RHR Pumps Inside Containment	NOTE 4	All	All		TBD	TBD
142.	Leakage Through Electrical Isolators in Instrumentation Circuits	MEDIUM	All	All		TBD	TBD
143.	Availability of Chilled Water Systems	HIGH	All	All		TBD	TBD
144.	Scram Without a Turbine/Generator Trip	NOTE 4	All	All		TBD	TBD
145.	Improve Surveillance and Startup Testing Programs	NOTE 4	All	All		TBD	TBD
146.	Support Flexibility of Equipment and Components	NOTE 4	All	All		TBD	TBD
147.	Fire-Induced Alternate Shutdown Control Room Panel Interactions	NOTE 4	All	All		TBD	TBD
148.	Smoke Control and Manual Fire-Fighting Effectiveness	NOTE 4	All	All		TBD	TBD
149.	Adequacy of Fire Barriers	NOTE 4	All	All		TBD	TBD
151.	Reliability of Recirculation Pump Trip During an ATWS	MEDIUM	All	NA		TBD	TBD
152.	Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads	NOTE 4	All	All		TBD	TBD
153.	Loss of Essential Service Water in LWRs	HIGH	All	All		TBD	TBD
154.	Adequacy of Emergency and Essential Lighting	NOTE 4	All	All		TBD	TBD
155.	Generic Concerns Arising from TMI-2 Cleanup	-	-	-		-	-
155.1	More Realistic Source Term Assumptions	NOTE 4	All	All		TBD	TBD
155.2	Establish Licensing Requirements for Non-Operating Facilities	NOTE 4	All	All		TBD	TBD
155.3	Improve Design Requirements for Nuclear Facilities	NOTE 4	All	All		TBD	TBD
155.4	Improve Criticality Calculations	NOTE 4	All	All		TBD	TBD
155.5	More Realistic Severe Reactor Accident Scenario	NOTE 4	All	All		TBD	TBD

Action Plan Item/ Issue No.	Title	Safety Priority/ Status	Affected NSSS Vendor BWR	PWR	Operating Plants- MPA No.	Operating Plants- Effective Date	Future Plants- Effective Date
155.6	Improve Decontamination Regulations	NOTE 4	All	All		TBD	TBD
155.7	Improve Decommissioning Regulations	NOTE 4	All	All		TBD	TBD
156.	Systematic Evaluation Program	-	-	-		-	-
156.1.1	Settlement of Foundations and Buried Equipment	NOTE 4	All	All		TBD	TBD
156.1.2	Dam Integrity and Site Flooding	NOTE 4	All	All		TBD	TBD
156.1.3	Site Hydrology and Ability to Withstand Floods	NOTE 4	All	All		TBD	TBD
156.1.4	Industrial Hazards	NOTE 4	All	All		TBD	TBD
156.1.5	Tornado Missile	NOTE 4	All	All		TBD	TBD
156.2.1	Severe Weather Effects on Structures	NOTE 4	All	All		TBD	TBD
156.2.2	Design Codes, Criteria, and Load Combinations	NOTE 4	All	All		TBD	TBD
156.2.3	Containment Design and Inspection	NOTE 4	All	All		TBD	TBD
156.2.4	Seismic Design of Structures, Systems, and Components	NOTE 4	All	All		TBD	TBD
156.3.1.1	Shutdown Systems	NOTE 4	All	All		TBD	TBD
156.3.1.2	Electrical Instrumentation and Controls	NOTE 4	All	All		TBD	TBD
156.3.2	Service and Cooling Water Systems	NOTE 4	All	All		TBD	TBD
156.3.3	Ventilation Systems	NOTE 4	All	All		TBD	TBD
156.3.6.1	Emergency AC Power	NOTE 4	All	All		TBD	TBD
156.3.6.2	Emergency DC Power	NOTE 4	All	All		TBD	TBD
156.3.8	Shared Systems	NOTE 4	All	All		TBD	TBD
156.4.2	Testing of the RPS and ESFS	NOTE 4	All	All		TBD	TBD
156.6.1	Pipe Break Effects on Systems and Components	NOTE 4	All	All		TBD	TBD
157.	Containment Performance	NOTE 4	All	All		TBD	TBD
158.	Performance of Power-Operated Valves Under Design Basis Conditions	NOTE 4	All	All		TBD	TBD
159.	Qualification of Safety-Related Pumps While Running on Minimum Flow	NOTE 4	All	All		TBD	TBD
160.	Spurious Actions of Instrumentation Upon Restoration of Power	NOTE 4	All	All		TBD	TBD
161.	Use of Non-Safety-Related Power Supplies in Safety-Related Circuits	NOTE 4	All	All		TBD	TBD
162.	Inadequate Technical Specifications for Shared Systems at Multiplant Sites When One Unit Is Shut Down	NOTE 4	All	All		TBD	TBD

HUMAN FACTORS ISSUES

HF1	STAFFING AND QUALIFICATIONS						
HF1.1	Shift Staffing	NOTE 3(a)	All	All		TBD	TBD
HF4	PROCEDURES						
HF4.4	Guidelines for Upgrading Other Procedures	HIGH	All	All		TBD	TBD

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
HF5	MAN-MACHINE INTERFACE						
HF5.1	Local Control Stations	HIGH	All	All		TBD	TBD
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	HIGH	All	All		TBD	TBD

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 14

2. TITLE AND SUBTITLE

A Prioritization of Generic Safety Issues

3. DATE REPORT PUBLISHED

MONTH	YEAR
August	1992

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

R. Emrit, R. Riggs, W. Milstead, J. Pittman

6. TYPE OF REPORT

7. PERIOD COVERED (Inclusive Dates)

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 Safety Issue Resolution
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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 Safety Issue Resolution
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The report presents the 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, and DROP and have been assigned on the basis of risk significance estimates, the ratio of risk to costs and other impacts estimated to result if resolutions of the safety issues were implemented, and the consideration of uncertainties and other quantitative or 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
risk

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page)

Unclassified

(This Report)

Unclassified

15. NUMBER OF PAGES

16. PRICE

THIS DOCUMENT WAS PRINTED USING RECYCLED PAPER