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SUPPLEMENT 26 TO NUREG-0933 "A PRIORITIZATION OF GENERIC SAFETY ISSUES"

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TABLE II

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

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

Legend

NOTES:	1 - Possible Resolution Identified for Evaluation
	2 - Resolution Available (Documented in NUREG, NRC Memorandum, SER, or equivalent)
	3 - Resolution Resulted in either: (a) The Establishment of New Regulatory Requirements (By Rule, SRP Change, or equivalent) or (b) No New Requirements
	4 - Issue to be Prioritized in the Future
	5 - Issue that is not a Generic Safety Issue but should be Assigned Resources for Completion
HIGH	- High Safety Priority
MEDIUM	- Medium Safety Priority
LOW	- Low Safety Priority
DROP	- Issue Dropped as a Generic Issue
EI	- Environmental Issue
I	- Resolved TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737
LI	- Licensing Issue
MPA	- Multiplant Action
NA	- Not Applicable
RI	- Regulatory Impact Issue
S	- Issue Covered in an NRC Program Outside the Scope of This Document
USI	- Unresolved Safety Issue
Continue	- As defined in NRC Management Directive 6.4

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
C-8	Main Steam Line Leakage Control Systems	Milstead	RES/DRPS/RPSI	NOTE 3(b)	1	06/30/90	NA
C-9	RHR Heat Exchanger Tube Failures	V'Molen	NRR/DSI/RSB	DROP		11/30/83	NA
C-10	Effective Operation of Containment Sprays in a LOCA	Emrit	NRR/DSI/AEB	NOTE 3(a)		11/30/83	NA
C-11	Assessment of Failure and Reliability of Pumps and Valves	Emrit	NRR/DE/MEB	NOTE 3(b)		12/31/85	NA
C-12	Primary System Vibration Assessment	Thatcher	NRR/DE/MEB	NOTE 3(b)		11/30/83	NA
C-13	Non-Random Failures	Emrit	NRR/DST/GIB	A-17	1	06/30/91	NA
C-14	Storm Surge Model for Coastal Sites	Emrit	NRR/DE/EHEB	LI (NOTE 3)		06/30/88	NA
C-15	NUREG Report for Liquid Tank Failure Analysis	-	NRR/DE/EHEB	LI (NOTE 3)		11/30/83	NA
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	Emrit	NRR/DSI/METB	NOTE 3(a)		11/30/83	NA
D-1	Advisability of a Seismic Scram	Thatcher	RES/DET/MSEB	DROP	1	12/31/98	NA
D-2	Emergency Core Cooling System Capability for Future Plants	Emrit	RES/DRA/ARGIB	DROP		12/31/88	NA
D-3	Control Rod Drop Accident	Emrit	NRR/DSI/CPB	NOTE 3(b)		11/30/83	NA

NEW GENERIC ISSUES

Action Plan Item/ Issue No.	Title	Priority Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
1.	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
2.	Failure of Protective Devices on Essential Equipment	Diab	RES/DSIR/EIB	DROP	2	06/30/95	NA
3.	Set Point Drift in Instrumentation	Emrit	NRR/DSIR/RPSIB	NOTE 3(b)	1	06/30/86	NA
4.	End-of-Life and Maintenance Criteria	Thatcher	NRR/DE/EQB	NOTE 3(b)		11/30/83	NA
5.	Design Check and Audit of Balance-of-Plant Equipment	Pittman	NRR/DSI/ASB	I.F.1		11/30/83	NA
6	Separation of Control Rod from Its Drive and BWR High Rod Worth Events	Vandermolen	NRR/DSI/CPB	NOTE 3(b)	1	12/31/94	NA
7.	Failures Due to Flow-Induced Vibrations	Vandermolen	NRR/DSI/RSB	DROP	1	06/30/91	NA
8.	Inadvertent Actuation of Safety Injection in PWRs	Colmar	NRR/DSI/RSB	I.C.1		11/30/83	NA
9.	Reevaluation of Reactor Coolant Pump Trip Criteria	Emrit	NRR/DSI/RSB	II.K.3(5)		11/30/83	NA
10.	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges	Riggs	NRR/DSI/ICSB	DROP		11/30/83	NA
11.	Turbine Disc Cracking	Pittman	NRR/DE/MTEB	A-37		11/30/83	NA
12.	BWR Jet Pump Integrity	Sege	NRR/DE/MTEB, MEB	NOTE 3(b)	1	12/31/84	NA
13.	Small Break LOCA from Extended Overheating of Pressurizer Heaters	Riani	NRR/DSI/RSB	DROP		11/30/83	NA
14.	PWR Pipe Cracks	Emrit	NRR/DE/MTEB	NOTE 3(b)	2	12/31/94	NA
15.	Radiation Effects on Reactor Vessel Supports	Emrit	RES/DET/EMMEB	NOTE 3(b)	3	06/30/96	NA

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

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

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

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

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

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

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

Table II (Continued)

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

Table II (Continued)

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

CHERNOBYL ISSUES

Table II (Continued)

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

Table II (Continued)

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

TABLE III

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

Legend

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

TABLE III

TABLE III (Continued)

ACTION ITEM/ISSUE GROUP	I	S	RESOLVED STAGES			USI	HIGH	MEDIUM	LOW	DROP	CONT.	NOTE 4	NOTE 5	TOTAL
			NOTE 1	NOTE 2	NOTE 3									
TMI ACTION PLAN ITEM (369)														
GSI	84	46	0	0	135	0	0	0	12	9	-	-	-	286
LI	-	0	-	-	75	-	-	-	-	-	-	-	8	83
TASK ACTION PLAN ITEMS (142)														
USI	-	-	-	-	27	0	-	-	-	-	-	-	-	27
GSI	-	20	0	0	36	-	0	0	0	14	-	-	-	70
RI	-	-	-	-	6	-	-	-	-	-	-	-	1	7
LI	-	-	-	-	11	-	-	-	-	-	-	-	12	23
EI	-	-	-	-	13	-	-	-	-	-	-	-	2	15
NEW GENERIC ISSUES (274)														
GSI	-	54	0	0	82	0	6	0	4	97	2	3	-	248
RI	-	1	-	-	5	-	-	-	-	1	-	-	5	12
LI	-	1	-	-	8	-	-	-	-	-	-	-	4	13
EI	-	-	-	-	-	-	-	-	-	-	-	-	1	1
HUMAN FACTORS ISSUES (27)														
GSI	-	8	0	0	8	0	0	0	0	0	-	-	-	16
LI	-	-	-	-	3	-	-	-	-	-	-	-	8	11
CHERNOBYL ISSUES (32)														
LI	-	2	-	-	7	-	-	-	-	-	-	-	23	32
TOTAL:	84	132	0	0	416	0	6	0	16	121	2	3	64	844

SECTION 2

TASK ACTION PLAN ITEMS

This section contains all Task Action Plan Items documented in NUREG-0371[2] and NUREG-0471[3] as well as all USIs documented in other NRC publications. Items A-1 through A-41 are listed in NUREG-0371[2] and all items with prefixes "B," "C," and "D" are listed in NUREG-0471.[3] USIs identified after publication of NUREG-0371[2] and NUREG-0471[3] are listed in the following documents: NUREG-0510[186] (A-42 through A-44); NUREG-0705[44] (A-45 through A-48); and NUREG-1090[1788] (A-49). A total of 142 items are listed in this section.

The Generic Issues Tracking System (GITS) Report[38] issued on December 17, 1981, provided a status report on the majority of the 142 items as well as their classification into four categories: Environmental, Licensing Improvement, Safety, and USI. The safety issues identified in the GITS Report[38] provided the basis for all prioritization work contained in this section. The lead responsibility and a summary of the findings for each item listed in this section can be found in Table II of the Introduction.

ITEM A-1: WATER HAMMER

DESCRIPTION

The issue was raised after the occurrence of various incidents of water hammer that involved steam generator feedrings and piping, emergency core cooling systems, RHR systems, containment spray, service water, feedwater, and steam lines. The incidents were attributed to such causes as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Most of the damage was relatively minor and involved pipe hangers and restraints. However, there were several incidents which resulted in piping and valve damage. This item was originally identified in NUREG-0371[2] and was later declared a USI in NUREG-0510.[186]

No water hammer incident resulted in the release of radioactivity outside of plants. However, because of the continuing incidence of water hammer events, the number of phenomena, and the potential safety significance of the systems involved, the staff believed that systematic review procedures needed to be developed to ensure that water hammer would be given appropriate consideration in CP and OL reviews, and in the review of operating reactors.

CONCLUSION

This USI was RESOLVED with the publication of NUREG-0927, Rev. 1,[698] and the following SRP[11] Sections: *3.9.3, Rev. 1; 3.9.4, Rev. 2; 5.4.6, Rev. 3; 5.4.7, Rev. 3; 6.3, Rev. 2; 9.2.1, Rev. 3; 9.2.2, Rev. 2; 10.3, Rev. 3; and 10.4.7, Rev. 3.* The revised SRP Sections will be used only for the review of "custom plant" CP applications and for standard plant applications docketed after the issuance of these revised SRP Sections (which are intended for referencing in CP applications). Thus, this USI will affect all future plants only.

REFERENCES

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

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

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

698. NUREG-0927, Revision 1, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1984.

ITEM A-2: ASYMMETRIC BLOWDOWN LOADS ON REACTOR PRIMARY COOLANT SYSTEMS

DESCRIPTION

On May 7, 1975, the NRC was informed by VEPCO that an asymmetric loading on the reactor vessel supports resulting from a postulated reactor coolant pipe rupture at a specific location (e.g., the vessel nozzle) had not been considered by W or S&W in the original design of the reactor vessel support systems for North Anna Units 1 and 2. This item was originally identified in NUREG-0371[2] and was later declared a USI in NUREG-0510.[186]

In a postulated event at the vessel nozzle, asymmetric LOCA loading could result from forces induced on the reactor internals by transient differential pressures across the core barrel and by forces on the vessel due to transient differential pressures in the reactor cavity. With the advent of more sophisticated computer codes and the accompanying more-detailed analytical models, it became apparent to W that such differential pressures, although of short duration, could place a significant load on the reactor vessel supports, thereby affecting their integrity. This issue was determined by the NRC to have generic implications for all PWRs.

CONCLUSION

This USI was RESOLVED with the publication of NUREG-0609[699] and affected all operating and future PWRs. For operating PWRs, MPA D-10 was established by DL/NRR for implementation purposes. Generic Letter 84-04[700] was also issued by the staff.

REFERENCES

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

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

699. NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," U.S. Nuclear Regulatory Commission, January 1981.

700. NRC Letter to All Operating PWR Licenses, Construction Permit Holders, and Applicants for Construction Permits, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops (Generic Letter 84-04)," February 1, 1984.

ITEM A-3: WESTINGHOUSE STEAM GENERATOR TUBE INTEGRITY

DESCRIPTION

Prior to 1978, operating experience with PWR steam generators was characterized by extensive corrosion and mechanically-induced degradation of the steam generator tubes, frequent plant shutdowns to repair primary-to-secondary leaks, and two SGTR events (Point Beach 1 in 1975 and Surry 2 in 1976). In 1978, Task Action Plans for Items A-3, A-4, and A-5 were established in NUREG-0371[2] to evaluate the safety significance of degradation in W, CE, and B&W steam generators, respectively. These items were later declared USIs in NUREG-0510[186] and were combined into one effort because many problems with PWR steam generators supplied by the three vendors were similar. Thus, an integrated program was developed for the resolution of USIs A-3, A-4, and A-5.

After SGTR events at Prairie Island 1 in 1979 and at Ginna in 1982, the staff initiated an integrated program to evaluate a number of recommendations stemming from the early USI effort and from lessons learned as a result of the SGTR events. The objective of the integrated program was to complete resolution of USIs A-3, A-4, and A-5, including identification of new requirements that could be imposed on OL applicants and licensees and identification of further efforts that should be undertaken by NRC. The results of this program were documented in NUREG-0844.[681]

The staff's risk analysis, as described in Section 3 of NUREG-0844,[681] indicated that SGTR events beyond the design basis did not constitute a significant fraction of the early and latent cancer fatality risks associated with reactor events at a given site. Furthermore, the risk assessment indicated that the increment in risk associated with SGTR events was a small fraction of the accidental and latent cancer fatality risks to which the general public is routinely exposed. These findings reflected not just the effectiveness of NRC regulatory guidance and TS requirements, but very importantly also reflected industry efforts to improve steam generator reliability which was of significant economic importance to the industry, in addition to providing added assurance of public health and safety.

The risk estimates documented in NUREG-0844[681] were based on consequence calculations that employed population distributions, protective actions, and meteorological assumptions equivalent to those presented in the Byron final environmental statement (NUREG-0848).[1135] The staff completed a comparative analysis which confirmed that the risk from SGTR-related causes did not exceed the Commission's safety goals on early or latent fatalities. Early fatality risks were estimated to be less than 10% of the safety goal, and the latent fatality risks were found to be a very small fraction of the safety goal.

In view of the relatively low risk estimates associated with SGTR events, the staff concluded that new generic requirements that had initially been proposed as part of the USI program were not warranted. However, the staff found in its value-impact analysis that a number of these proposals, as a group, were effective measures for significantly reducing the incidence of tube degradation, the frequency of SGTRs and the corresponding potential for significant non-core-melt release, and occupational exposures, and were consistent with good operating and engineering practice. As a group, these actions were considered to be effective measures for mitigating the consequences of SGTRs. Adoption of these actions by licensees would further reduce public risk (by as much as

70%) and provide added assurance that risk would continue to be small. These actions were designated as staff-recommended actions.

CONCLUSION

As part of the steam generator USI program, the staff issued Generic Letter 85-02[1136] to all PWR licensees and applicants to inform them of the staff-recommended actions and to request a description of their overall programs for ensuring steam generator tube integrity and SGTR mitigation. The staff's assessment of the licensee and applicant responses to Generic Letter 85-02[1136] was provided to the Commission in SECY-86-97[1137] in March 1986. The staff concluded on the basis of this assessment that the large majority of the licensees and applicants were following programs, practices, and/or procedures that were partially to fully consistent with, or equivalent to, the staff-recommended actions.

Following the North Anna 1 SGTR event on July 15, 1987, NRC Bulletin No. 88-02[1138] was issued requesting that licensees and OL applicants perform specified inspections and analyses to determine whether their plants were susceptible to the failure mechanism that led to the North Anna event, and that they implement corrective actions, if necessary.

The Commission's current regulations (10 CFR Part 50, Appendices A and B; 10 CFR 50.55a; 10 CFR 50.109; and 10 CFR Part 100) provide the staff with sufficient authority to ensure that licensees implement programs relating to steam generator tube integrity that provide adequate protection to public health and safety. The staff will continue to monitor steam generator experience as an indicator of the effectiveness of licensee programs for ensuring steam generator tube integrity. As exemplified by Bulletin 88-02,[1138] the staff may impose additional requirements (pursuant to applicable regulations) to continue to ensure that licensees adequately implement effective programs where such action is determined to be necessary on the basis of operating experience, or as a result of ongoing staff studies. Thus, as stated in SECY-88-272,[1139] USIs A-3, A-4, and A-5 were RESOLVED and requirements were established.

REFERENCES

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

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

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.

1135. NUREG-0848, "Final Environmental Statement Related to the Operation of Byron Station Units 1 and 2," U.S. Nuclear Regulatory Commission, April 1982.

1136. NRC Letter to All PWR Licencees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, and Ft. St. Vrain, "Staff Recommended Actions Stemming from NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity (Generic Letter 85-02)," April 17, 1985.

1137. SECY-86-97, "Steam Generator USI Program - Utility Responses to Staff Recommendations in Generic Letter 85-02," March 24, 1986.

1138. NRC Bulletin No. 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," February 5, 1988.

1139. SECY-88-272, "Technical Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," September 27, 1988.

ITEM A-4: CE STEAM GENERATOR TUBE INTEGRITY

DESCRIPTION

This item was originally identified in NUREG-0371[2] and was later declared a USI in NUREG-0510.[186] (See Item A-3 for further details.)

CONCLUSION

This item was RESOLVED and requirements were established. (See Item A-3 for further details.)

REFERENCES

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

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

ITEM A-5: B&W STEAM GENERATOR TUBE INTEGRITY

DESCRIPTION

This item was originally identified in NUREG-0371[2] and was later declared a USI in NUREG-0510.[186] (See Item A-3 for further details.)

CONCLUSION

This item was RESOLVED and requirements were established. (See Item A-3 for further details.)

REFERENCES

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

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

ITEM A-6: MARK I SHORT-TERM PROGRAM

DESCRIPTION

During the conduct of a large scale testing program for an advanced design BWR pressure suppression containment system (MARK III), new suppression pool hydrodynamic loads associated with a postulated LOCA were identified which had not been explicitly included in the original design of the MARK I containment systems. These additional loads result from the dynamic effects of drywell air and steam being rapidly forced into the suppression pool (Torus) during a postulated LOCA event. Consequently, it was determined that a reassessment of the MARK I containment system design would be required. This item was originally identified in NUREG 0371[2] and was later declared a USI in NUREG-0510.[186]

CONCLUSION

This USI was RESOLVED with the publication of NUREG-0408.[701] All plant-unique analyses and required equipment modifications were reviewed and accepted by the staff and appropriate TS changes were made by the affected licensees.

REFERENCES

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

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

701. NUREG-0408, "MARK I Containment Short-Term Program Safety Evaluation Report," U.S. Nuclear Regulatory Commission, December 1977.

ITEM A-7: MARK I LONG-TERM PROGRAM

DESCRIPTION

During testing for an advanced BWR containment system design (MARK III), suppression pool hydrodynamic loads were identified which had not been considered in the original design of the MARK I containment system. To address this issue, a MARK I Owners' Group was formed and the assessment was divided into a short-term and a long-term program. The results of the NRC staff's review of the MARK I Containment Short-Term Program were documented in NUREG-0408.[701] The long-term program was conducted to provide a generic basis to define suppression pool hydrodynamic loads and the related structural acceptance criteria, such that a comprehensive reassessment of each MARK I containment system could be performed.

A series of experimental and analytical programs were conducted by the MARK I Owners' Group to provide the necessary bases for the generic load definition and structural assessment techniques. The generic methods proposed by the MARK I Owners' Group, as modified by the NRC staff's requirements, were to be used to perform plant-unique analyses which would identify the plant modifications, if any, needed to restore the originally intended margin of safety in the MARK I containment designs. This item was originally identified in NUREG-0371[2] and was later declared a USI in NUREG-0510.[186]

CONCLUSION

This USI was RESOLVED with the issuance of Supplement 1 to NUREG-0661[702] and SRP[11] Section *6.2.1.1C*. For operating BWRs, MPA D-01 was established by DL/NRR for implementation purposes.

REFERENCES

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

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

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

701. NUREG-0408, "Mark I Containment Short-Term Program Safety Evaluation Report," U.S. Nuclear Regulatory Commission, December 1977.

702. NUREG-0661, "Mark I Containment Long Term Program Safety Evaluation Report, Resolution of Generic Technical Activity A-7," U.S. Nuclear Regulatory Commission, July 1980, (Supplement 1) August 1982.

ITEM A-8: MARK II CONTAINMENT POOL DYNAMIC LOADS LONG-TERM PROGRAM

DESCRIPTION

As a result of the GE testing program for the MARK III pressure-suppression containment program, new containment loads associated with a postulated LOCA were identified in 1975 which had not been explicitly included in the original design of MARK I and MARK II containments. These loads resulted from the dynamic effects of drywell air and steam being rapidly forced into the suppression pool during a postulated LOCA event. Other pool dynamic loads previously unaccounted for resulted from the actuation of SRVs in the MARK II containment. The review and evaluation of the MARK I loads were addressed in USI A-7, and SRV loads for all suppression-type containments were addressed in USI A-39. This item was originally identified in NUREG-0371[2] and was later declared a USI in NUREG-0510.[186]

CONCLUSION

This USI was RESOLVED with the issuance of NUREG-0808[703] and SRP[11] Section *6.2.1.1C*.

REFERENCES

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

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

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

703. NUREG-0808, "MARK II Containment Program Evaluation and Acceptance Criteria," U.S. Nuclear Regulatory Commission, August 1981.

ITEM A-9: ATWS

DESCRIPTION

The technical report on ATWS for water-cooled reactors (WASH-1270)[751] discussed the probability of an ATWS event as well as an appropriate safety objective for the event. After several years of discussions with vendors and evaluations of vendor models and analyses, the staff published a status report on each vendor analysis in 1975 . This report included detailed guidelines on analysis models and ATWS safety objectives. This item was originally identified in NUREG-0371[2] and was later declared a USI in NUREG-0510.[186]

CONCLUSION

The staff's technical findings were published in Volume 4 of NUREG-0460[704] and the USI was RESOLVED with the publication of a final rule.[724,725]

REFERENCES

2. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, March 1980.

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

704. NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," U.S. Nuclear Regulatory Commission, March 1980.

724. Memorandum for W. Dircks, et al., from S. Chilk, "Staff Requirements - Affirmation/Discussion and Vote, 11:30 a.m., Friday, June 1, 1984, Commissioners' Conference Room, D.C. Office (Open to Public Attendance)," June 1, 1984.

725. Federal Register Notice 49 FR 26036, "10 CFR Part 50, Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," June 26, 1984.

751. WASH-1270, "Anticipated Transients Without Scram for Water-Cooled Reactors," U.S. Nuclear Regulatory Commission, September 1973.

ITEM A-10: BWR FEEDWATER NOZZLE CRACKING

DESCRIPTION

Inspections of operating BWRs conducted up to April 1978 revealed cracks in the feedwater nozzles of 20 reactor vessels. Most of these BWRs contained 4 nozzles with diameters ranging from 10 in. to 12 in. Although most cracks ranged from 1/2 in. to 3/4 in. in depth (including cladding), one crack penetrated the cladding into the base metal for a total depth of approximately 1.5 in. It was determined that cracking was due to high-cycle fatigue caused by fluctuations in water temperature within the vessel in the nozzle region. These fluctuations occurred during periods of low feedwater temperature when flow is unsteady and intermittent. Once initiated, the cracks enlarge from high pressure and thermal cycling associated with startups and shutdowns. This item was originally identified in NUREG-0371[2] and was later declared a USI in NUREG-0510.[186]

CONCLUSION

This issue was RESOLVED with the issuance of NUREG-0619[742] and MPA B-25 was established by DL/NRR for implementation purposes.

REFERENCES

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

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

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

ITEM A-11: REACTOR VESSEL MATERIALS TOUGHNESS

DESCRIPTION

Because of the remote possibility of failure of nuclear reactor pressure vessels designed to the ASME Boiler and Pressure Vessel Code, the design of nuclear facilities does not provide protection against reactor vessel failure. Prevention of reactor vessel failure depends primarily on maintaining the reactor vessel material fracture toughness at levels that will resist brittle fracture during plant operation. At service times and operating conditions typical of existing operating plants, reactor vessel fracture toughness properties provide adequate margins of safety against vessel failure; however, as plants accumulate more and more service time, neutron irradiation reduces the material fracture toughness and initial safety margins. This item was originally identified in NUREG-0371[2] and was later declared a USI in NUREG-0510.[186]

CONCLUSION

This USI was RESOLVED with the issuance of NUREG-0744,[743] Revision 1, which was later transmitted to all licensees with Generic Letter 82-26.[744]

REFERENCES

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

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

743. NUREG-0744, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue," U.S. Nuclear Regulatory Commission, (Rev. 1) October 1982.

744. NRC Letter to All Power Reactor Licensees (Except Ft. St. Vrain), "NUREG-0744 Rev. 1; Generic Letter No. 82-26 - Pressure Vessel Material Fracture Toughness," November 12, 1982.

ITEM A-12: FRACTURE TOUGHNESS OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORTS

DESCRIPTION

During the course of the licensing action for North Anna Units 1 and 2, a number of questions were raised as to the potential for lamellar tearing and low fracture toughness of the steam generator and RCP support materials for these facilities. Two different steel specifications (ASTM A36 and ASTM A572) covered most of the material used for these supports. Toughness tests, not originally specified and not in the relevant ASTM specifications, were made at various temperatures. The toughness of the A36 steel was found to be adequate, but the toughness of the A572 steel was relatively poor at a temperature of 80°F. In the case of North Anna Units 1 and 2, the applicant agreed to raise the temperature of the A572 beams in the steam generator supports to a minimum temperature of 225°F, prior to reactor coolant system pressurization to levels above 1,000 psig. Auxiliary electrical heat was supplied as necessary to supplement the heat derived from the reactor coolant loop to obtain the required operating temperature of the support materials. Concerns regarding the supports at North Anna were applicable to all PWRs. This item was originally identified in NUREG-0371[2] and was later declared a USI in NUREG-0510.[186]

CONCLUSION

This USI was RESOLVED with the publication of NUREG-0577,[388] Revision 1. The resolution contained no backfit requirements and applied only to new construction after issuance of SRP[11] Section 5.3.4.

REFERENCES

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

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

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

388. NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports," U.S. Nuclear Regulatory Commission, (Rev. 1) October 1983.

ITEM A-13: SNUBBER OPERABILITY ASSURANCE

DESCRIPTION

Historical Background

In May 1978, the ACRS and the staff expressed concern over the substantial number of LERs related to the malfunction of snubbers, the most frequent of which were: (1) seal leakage in hydraulic snubbers; and (2) high rejection rate during functional testing of snubbers. In reviewing these LERs, the staff noted differences in the numbers of snubbers utilized in systems of similar configurations and questioned the methodology used for determining the need for snubbers in any given system. As a result of these concerns and the desire to provide a significant increase in assurance of the health and safety of the public, snubber operability assurance was given a priority Category A designation and included in NUREG-0371.[2]

Safety Significance

Snubbers are utilized primarily as seismic and pipe whip restraints at nuclear power plants. Their safety function is to operate as rigid supports for restraining the motion of attached systems or components under rapidly applied load conditions such as earthquakes, pipe breaks, and severe hydraulic transients.

Possible Solutions

The solutions proposed in NUREG-0371[2] were as follows: (1) evaluation of industry practice associated with snubber qualification testing, design and analysis procedures, selection and specification criteria, and pre-service and in-service inspection programs; and (2) development of TS, SRP[11] revisions, and Regulatory Guides to assure a high level of snubber operability.

CONCLUSION

In 1980, the staff addressed the operation of snubbers with revisions to STS 3/4.7.9; SRP[11] Section 3.9.3 was later revised in 1981. A draft regulatory guide (Task SC-708-4)[12] on the qualification and acceptance tests for snubbers used in systems important to safety was issued by the staff, but was later withdrawn when it was determined that there were no plans for its use in the licensing process.[9] Thus, with the SRP revision, this issue was RESOLVED and requirements were issued.

REFERENCES

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

9. Federal Register Notice 54 FR 16030, "Draft Regulatory Guide; Withdrawal," April 20, 1989.

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.

12. Draft Regulatory Guide and Value/Impact Statement, Task SC 708-4, "Qualification and Acceptance Tests for Snubbers Used in Systems Important to Safety," U.S. Nuclear Regulatory Commission, February 1981.

ITEM A-14: FLAW DETECTION

DESCRIPTION

Historical Background

After the 1970 issuance of inspection requirements in Section XI of the ASME Boiler and Pressure Vessel Code,[14] the staff recognized the need to quantify the uncertainty in the existing inspection requirement techniques. Also, the staff recognized its responsibility to upgrade these requirements, if necessary, when improvements in inspection techniques became commercially available.

This item was identified in NUREG-0371[2] and consisted of quantifying and upgrading the reliability of existing ISI techniques and of developing, evaluating, and validating improved techniques for flaw detection and evaluation during ISI of primary system components. The results were to be used in improving ASME Code Section XI[14] inspection provisions, and preparing Regulatory Guides as needed. At the time of the evaluation of this issue in 1983, a major part of the NRC effort on the issue was being carried out under the RES program on NDE (RES Long Range Research Plan, Program 6.3). This program resulted in the issuance of a new Regulatory Guide 1.150[141] and preparation of improved piping inspection provisions which were to be incorporated into Section XI[14] of the ASME Code.

Safety Significance

As part of the defense-in-depth approach, components and structures are inspected in order to detect and repair flaws well before they reach a critical size and lead to undesirable consequences ranging from small leaks to a large LOCA.

It was believed that improvements in flaw detection reliability and capability could contribute to reducing the risk associated with specific safety issues that were open at the time this issue was evaluated: Issue 15, "Radiation Effects on Reactor Vessel Supports," and Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants." This issue was also part of USIs A-3, A-4, and A-5 which addressed steam generator tube integrity. It was believed that improved flaw detection would provide a longer-term contribution to the resolution of USI A-12, "Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports," and USI A-49, "Pressurized Thermal Shock." Resolution of this issue was expected to provide the experimental basis for the technical position of a planned Regulatory Guide on ISI of austenitic stainless steel piping.

Possible Solution

If successfully resolved, this issue would quantify: (1) the uncertainties concerning the smallest size defect which could reliably be detected by required inspection techniques; and (2) the dimensions of identified defects. Thus, the uncertainty in the resolution of other safety issues could be reduced and, possibly, could allow for modifications of some inspection requirements.

CONCLUSION

This item was not a safety issue by itself but was only amenable to risk reduction value/impact assessment by reference to other issues. This was an ongoing task that was sponsored by RES, with the results of development efforts on the part of both NRC and industry (EPRI) to be used for improving the inspection provisions of Section XI[14] of the ASME Code and for providing the technical basis for Regulatory Guides related to ISI. These efforts were to be closely monitored by the users of the results to ensure that they were directly applied to the resolution of the safety issues identified above. Since this item was largely an RES program which served a number of generic issues, it was DROPPED from further consideration as a separate issue.

REFERENCES

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

14. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers, 1974.

141. Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," U.S. Nuclear Regulatory Commission, June 1981, (Rev. 1) February 1983.

ITEM A-15: PRIMARY COOLANT SYSTEM DECONTAMINATION AND STEAM GENERATOR CHEMICAL CLEANING

DESCRIPTION

Historical Background

Operation of a LWR results in slow corrosion of the interior metal surfaces of the primary coolant system. The resulting corrosion products circulate through the reactor core and are activated by neutron flux from the fissioning reactor fuel. While some of these activated corrosion products are removed by the reactor's water chemistry system, a small amount is continually deposited or plated out on the primary coolant system's internal surfaces. Once activated corrosion products are deposited or plated out, they are not removed by the reactor water cleanup system and continue to accumulate. As a direct result of this accumulation, radiation levels in the vicinity of the primary system rise, thus inhibiting or complicating routine inspection and maintenance of the primary system. This issue was identified in NUREG-0371[2] and, at the time of its evaluation in 1983, technical activities in pursuit of a solution were in progress by groups that were sponsored by government and private industry prior to 1977.

Safety Significance

Decontamination of primary coolant systems and steam generators is not a safety issue related to the health and safety of the general public but rather to the health and safety of workers in nuclear power plants. Annual occupational radiation doses from the operation and maintenance of nuclear reactors tend to increase with increasing reactor age.[78,79] Much of this increase is due to the continued deposition of highly activated corrosion products, such as Co-60 and Co-58, in various locations in the primary coolant system. In 1979, the average occupational collective radiation dose per operating PWR was 510 man-rem and the corresponding figure per operating BWR was 733 man-rem.[78] Approximately 80% of the occupational radiation dose resulted from inspection and maintenance activities[78] which were mostly related to the primary coolant system.

Possible Solutions

Periodic removal of activated corrosion products would reduce occupational exposure due to maintenance and inspection activities. Two methods were proposed for decontaminating reactors intended to be returned to service: (1) strong solution decontamination, such as NS-1 (Dow Chemical); and (2) weak or dilute contamination solutions. Weak solutions are typically CAN-DECON (London Nuclear Limited), LOMI (Central Electric Generating Board), hydrogen peroxide/citric acid, and hydrazine/EDTA. Decontamination factors range from 2 to 5.[80] Weak solution decontamination was utilized at Nine Mile Point 1, Vermont Yankee, and Brunswick Units 1 and 2, while strong solutions were utilized at Peach Bottom Units 2 and 3. In these plants, decontamination solutions were used to decontaminate systems and components.

CONCLUSION

This issue was RESOLVED with the publication of decontamination criteria in NUREG/CR-2963.[489]

REFERENCES

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

78. NUREG/CR-1496, "Nuclear Power Plant Operating Experience - 1979," U.S. Nuclear Regulatory Commission, May 1981.

79. NUREG-0109, "Occupational Radiation Exposure at Light Water Cooled Power Reactors 1969-1975," U.S. Nuclear Regulatory Commission, August1976.

80. DOE/ET/34204-43, "Dilute Chemical Decontamination Program - Final Report," U.S. Department of Energy, August 1981.

489. NUREG/CR-2963, "Planning Guidance for Nuclear Power Plant Decontamination," U.S. Nuclear Regulatory Commission, June 1983.

ITEM A-16: STEAM EFFECTS ON BWR CORE SPRAY DISTRIBUTION

DESCRIPTION

Historical Background

Prior to May 1978, tests conducted by GE showed that the presence of steam and/or increased pressure in and above the upper core region of BWRs could adversely affect the distribution of flow from certain types of core spray nozzles. These nozzles are arranged to distribute water over the top of the reactor core in the event of a LOCA.

These new test data were collected from a reactor core spray system with a single nozzle spraying downward. However, spray flow in most domestic BWR core spray systems comes from many nozzles spraying approximately horizontally over the core from a sparger (or spargers) surrounding the core. Therefore, the degree of applicability of the new data to domestic BWRs was not known. As a result, this issue was included in NUREG-0371[2] to provide results which could be verified as being applicable to the size and design of each BWR in operation. In order to justify the preliminary acceptability of core spray cooling designs in operation, GE presented test results and calculations that were based on the separability of hydrodynamic phenomena (droplet-to-droplet interaction where spray patterns from two or more nozzles intersect) and thermal phenomena (steam condensation).

Safety Significance

If BWRs are to strictly conform to the post-LOCA requirements established by 10 CFR 50.46 to ensure the health and safety of the public, then their core spray systems must supply a specified minimum amount of coolant to each fuel bundle in their respective reactor cores. Therefore, core spray assumed in the LOCA analyses must be actually supplied in the post-LOCA steam environment. This issue was a topic in the SEP for both Millstone 1 and Dresden 2.

Possible Solution

The solution recommended in NUREG-0371[2] called for: (1) a series of tests on operating BWR/6 core spray distribution systems; and (2) a full-scale test of a 30° sector of a BWR/6 upper plenum, complete with spargers. Test results were to be reviewed by the NRC for acceptability of the analytical and experimental techniques used to determine the safety margin present in core spray distributions for all BWRs in operation and under construction.

CONCLUSION

Test results issued by GE in August 1979[40] compared favorably with the pre-test prediction, within defined acceptance limits, and confirmed the capability of the methodology to handle steam environment effects on spray performance. The results substantiated the key assumption of separability of thermodynamic and hydrodynamic effects. These test results were reviewed by the NRC and determined to "constitute an adequate confirmation of the GE spray distribution methodology for BWR/6-type spargers."[100] However, the NRC required additional tests to be performed to confirm the design methodology for other sparger designs. As a result, a test program was initiated to provide core spray distribution data in a steam environment for a 30-degree sector of the BWR/4 and BWR/5 design. In March 1981, test results for this BWR design were published in NUREG/CR-1707.[39] These data demonstrated the applicability of the core spray methodology in this design which had nozzle types and sparger evaluations that were different from the BWR/6 design tested in 1979.

The BWR/1 core spray design was reviewed by DSI/NRR in 1979 and found to be acceptable. Following the review of GE test data for the BWR/3 core spray design, DSI/NRR concluded in March 1983 that the core spray distribution adequacy was not a safety concern for all BWR/3 reactors.[427]

MPA D-12 was established by DL/NRR for the review of the BWR/2 core spray system design, and for the preparation of an SER for each of the two domestic reactors of this design: Oyster Creek and Nine Mile Point, Unit 1.[392,427] Based on the plant-specific reviews that were undertaken by the NRC, this issue was RESOLVED with no new requirements for licensees.

REFERENCES

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

39. NUREG/CR-1707, "BWR Refill-Reflood Program, Task 4.2 - Core Spray Distribution Final Report," U.S. Nuclear Regulatory Commission, March 1981.

40. NEDO-24712, "Core Spray Design Methodology Confirmation Tests," General Electric Company, August 1979.

100. Letter to General Electric Company from R. Tedesco, "Acceptance for Referencing Topical Report, NEDO-24712: Core Spray Design Methodology Confirmation Tests," January 1981.

392. Memorandum for J. Funches from R. Mattson, "Request for Approval to Work on Low Priority Generic Safety Issues," November 5, 1982.

427. Memorandum for T. Speis from R. Mattson, "Close-out of TAP-A-16, Steam Effects on BWR Core Spray Distribution (TACS-40066)," March 29, 1983.

ITEM A-17: SYSTEMS INTERACTIONS IN NUCLEAR POWER PLANTS

DESCRIPTION

Nuclear power plants contain many structures, systems, and components (SSCs), some of which are safety-related. Certain SSCs are designed to interact to perform their intended functions. These "systems interactions" are usually well recognized and, therefore, are accounted for in the evaluation of plant safety by designers and in plant safety assessments. However, prior to the time this issue was identified in 1978, a number of significant plant-specific events had occurred that involved unintended or unrecognized dependencies among the SSCs. Some of these events involved subtle dependencies between safety-related SCCs and other SCCs, while other events involved subtle dependencies between redundant safety-related SSCs that were believed to be independent. This issue was originally identified in NUREG-0371[2] and was later declared a USI in NUREG-0510.[186]

CONCLUSION

The purpose of this issue was to investigate the potential that unrecognized, subtle dependencies among SSCs have remained hidden and that they could lead to safety-significant events. The term used to describe these unrecognized, subtle dependencies is adverse systems interactions (ASIs). In resolving this issue, the staff did not recommend that licensees conduct further broad searches specifically to identify all ASIs because such searches had not proved to be cost-effective in the past, and there was no guarantee after such studies that all ASIs had been uncovered. Rather, in its study, the staff concluded that certain more specific actions, together with other ongoing activities, could reduce the risk from ASIs.

The staff concluded from its investigations that the following actions should be taken:

(1) Issuance of a generic letter that included: (a) the bases for resolution of USI A-17; and (b) a summary of information relevant to existing operating experience reviews.

(2) Recognition that the IPE Program already included the evaluation of internal flooding and the insights from USI A-17 were to be referred to in the IPE guidance documents. If licensee action regarding flooding and water intrusion was implemented as proposed, there would be no further action on Issue 77 which was integrated into the resolution of USI A-17.

(3) Recognition that the USI A-46 implementation was expected to address seismically-induced systems interactions to verify that components and systems needed to safely shut down a plant were protected, given a loss of offsite power. (New plants, not covered by USI A-46, were reviewed to existing requirements that addressed seismically-induced systems interactions.)

(4) Communication of information regarding ASIs for staff review of PRAs and for staff evaluation of electric power supplies as part of Issue 128.

(5) Identification and definition of concerns related to USI A-17 and other programs that had not been specifically addressed in this or other generic issues. The staff established the

Multiple System Responses Program (MSRP),[1237] the objective of which was to define the concerns with sufficient specificity to permit them to be evaluated as potential GSIs.

(6) *Development of an SRP for future plants that would include guidance regarding protection* from internal flooding and water intrusion events.

The staff's technical findings were published in NUREG-1174[1232] and the regulatory analysis associated with the resolution of this issue was published in NUREG-1229.[1233] The Commission was informed of the staff's resolution in SECY-89-230[1234] and Generic Letter 89-18[1235] was later issued to licensees. Thus, this issue was RESOLVED with no new or revised requirements for licensees.[1236]

REFERENCES

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

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

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 of USI A-17," U.S. Nuclear Regulatory Commission, August 1989.

1234. SECY-89-230, "Unresolved Safety Issue A-17, 'Systems Interactions in Nuclear Power Plants,'" August 1, 1989.

1235. NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants, "Resolution of Unresolved Safety Issue A-17, 'Systems Interactions in Nuclear Power Plants' (Generic Letter 89-18)," September 6, 1989.

1236. Federal Register Notice 54 FR 34836, "Issuance and Availability of NUREG-1174, 'Evaluation of Systems Interactions in Nuclear Power Plants: Technical Findings Related to Unresolved Safety Issue A-17,' and NUREG-1229, 'Regulatory Analysis for Resolution of USI A-17, - Systems Interactions in Nuclear Power Plants,'" August 22, 1989.

1237. NUREG/CR-5420, "Multiple System Responses Program - Identification of Concerns Related to a Number of Specific Regulatory Issues," U.S. Nuclear Regulatory Commission, October 1989.

ITEM A-18: PIPE RUPTURE DESIGN CRITERIA

DESCRIPTION

Historical Background

A major objective of this NUREG-0371[2] item was the development of consistent criteria for application in licensing processes. Additional research programs to implement licensing positions were to be conducted under separate issues. The problems specific to this issue were as follows:

(1) Existing design criteria for the postulation of pipe breaks and protection therefrom had been developed over a period of time and lacked consistency when applied inside and outside containment. Regulatory Guide 1.46,[18] issued in 1973, which addressed pipe breaks inside containment, was based on the concept of a limited number of design basis breaks. Section 3.6 of the SRP,[11] issued in 1975, which addressed pipe breaks outside containment, combined limited design basis breaks for mechanistic protection and unlimited breaks for non-mechanistic protection. At the time this issue was identified in 1978, staff efforts toward documentation of the rationale and engineering justification for existing pipe break criteria was ongoing. These efforts were expected to assist in focusing on areas requiring first attention and providing a valuable document for both public and staff use as bases for testimony before the ACRS and hearing boards. Work in this area was completed prior to 1983.

(2) An evaluation of the pipe break exclusion concept in the containment penetration area of both PWRs and BWRs was required. The need to specify the extent of break exclusion regions, criteria for the use of guard pipes, and adequacy of design requirements for piping systems in break exclusion regions were topics for which improved guidance were to be developed.

(3) The development of postulated pipe rupture criteria and the trend towards more conservative seismic criteria placed increased emphasis on piping system design to withstand these dynamic events. However, these criteria had also resulted in systems which were significantly more rigid. These more rigidly designed systems in the plants that were not in operating in 1978 had resulted in calculated stresses for normal operation which, although still within code limits, were significantly higher than in earlier plants. In addition, dynamic event devices, such as snubbers and pipe-whip restraints which had been added in increased numbers, had the potential for deleterious interaction with the piping system during its normal operation. It was believed that a balance in piping system design for both normal and abnormal situations should be achieved to ensure that consideration is given to the effects of abnormal situation design criteria on normal operation.

The evaluation of this issue included consideration of Item B-16.[2]

Possible Solutions

At the time of the evaluation of this issue in 1983, a study of the effects of abnormal loading scenario design criteria on normal operation had been completed. Determining licensing positions and the consequences of implementing the results of this issue were not considered in this evaluation. Item B-6[2] more directly addressed: (1) the safety consequences of combining unusual dynamic events and normal plant operating conditions; and (2) the option of limiting the number of dynamic event devices.

The criteria used for designing and constructing containment penetrations were to be evaluated in this issue. Guidelines for limiting the extent of break exclusion areas, criteria for the use of guard pipes, and the adequacy of design requirements for piping systems in break exclusion areas were of concern. The consequences of implementing the resultant guidelines was expected to differ for various plant types and piping systems. It was assumed that the resolution would, in general, limit the number of break exclusion areas. It was further assumed that this limitation would affect only 60% of all forward-fit PWRs and BWRs.

PRIORITY DETERMINATION

Frequency/Consequence Estimate

The reduction in public risk was determined to be negligible (≤100 man-rem)[64] and limiting the extent of break exclusion areas did not increase or decrease the probability of a pipe rupture.

Cost Estimate

Industry Cost: It was estimated that only 60% of all forward-fit plants (43 PWRs and 20 BWRs) would be affected by limitations on break exclusion areas. Thus, the total number of affected plants was 38. The average remaining life of these affected plants was (38)(30) RY or 1,140 RY.

Labor included: (1) implementation of criteria for defining pipe break and crack locations and configurations; (2) implementation of criteria dealing with special features, such as augmented ISI or use of postulated event devices; and (3) the review of analysis results, including jet-thrust and impingement forcing functions, pipe-whip dynamic effects, and design adequacy of systems to ensure that function is not impaired as a result of pipe-whip or jet impingement loadings.

It was assumed that labor included the time required to analyze lines located outside the break exclusion regions and that analysis procedures, computer codes, applicable transient data, etc., were readily available. It was also assumed that only 50% of the 12 welds under investigation needed analysis (i.e., those excluded either already fell into an analyzed line or did not fall into a high energy/high stress area which required analysis). The total industry cost for implementing the possible solution was estimated to be $2.07M.[64]

Industry operation and maintenance costs associated with the solution would result in cost savings to the industry due to fewer ISI periods when weld design locations are shifted from a break exclusion area. Based on a labor decrease of 2.2 man-hours/RY at a cost of $2,270/man-week, this cost saving was $125/RY. The total industry cost savings that would result from reduced operation and maintenance at all affected plants were ($125/RY x 1,140 RY) or $143,000.

NRC Cost: It was assumed that NRC would provide the criteria to limit the extent of break exclusion regions for plant types and piping systems. Independent plant reviews with respect to new SRP[11] regulations would then be conducted. At the time this issue was evaluated in 1983, the resclution had been completed. Therefore, based on an implementation estimate of 3 man-weeks/plant, the total NRC cost was estimated to be ($6,810/plant)(38 plants) or $259,000.

NRC costs for reviewing piping systems were not expected to change. However, a review of the consequences of imposing limitations on break exclusion areas would result in NRC costs of approximately \$191/RY. Thus, the total NRC cost to support operation and maintenance was estimated to be (1,140 RY x \$191/RY) or \$220,000.

Total Cost: Summing all costs outlined above, the total cost associated with the possible solution was estimated to be \$[2.07 + 0.143 + 0.259 + 0.22]M or approximately \$2.7M.

Value/Impact Assessment

Based on an estimated public risk reduction of less than 100 man-rem and a cost of \$2.7M for a possible solution, the value/impact score was given by:

$$S \leq \frac{100 \text{ man-rem}}{\$2.7\text{M}}$$

$$\leq 37 \text{ man-rem/\$M}$$

Other Considerations

(1) Implementation Occupational Risk Increase

Implementation of the solution was estimated to occur during plant design stages. Therefore, any alterations made in break exclusion areas would occur before plant operation and startup. Thus, there was no occupational risk increase from implementation of the solution in the affected plants.

(2) Operation and Maintenance Occupational Risk Decrease

When a line is excluded from a break exclusion area, associated welds would no longer require a 100% volumetric inspection every 10 years. Instead, ISI of these welds would be scheduled once during the lifetime of a plant (i.e., 25% of welds would be inspected every 10 years).

Implementation of the possible solution was estimated to reduce operation and maintenance time in radiation zones by 2.2 man-hours/RY. Based on an average expected dose rate of 0.1 rem/hour for ISI, the total occupational risk reduction was estimated to be:

$$(2.2 \text{ man-hour/RY})(0.1 \text{ rem/hour})(1{,}140 \text{ RY}) = 251 \text{ man-rem}$$

(3) Accident Avoidance Occupational Risk Decrease

Implementation of the solution would not change the frequency of a core-melt accident. Thus, there was no occupational risk reduction associated with the solution.

Summing up the above three factors, the total occupational risk decrease was 251 man-rem. Inclusion of this factor in the value/impact score calculation would produce a value/impact score of $S \leq 130$ man-rem/\$M.

CONCLUSION

Based on the estimated public risk reduction and the value/impact score, this issue was DROPPED from further consideration. Consideration of occupational risk decrease did not affect this conclusion.

REFERENCES

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

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

18. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," U.S. Nuclear Regulatory Commission, May 1973.

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, (Supplement 5) July 1996.

ITEM A-19: DIGITAL COMPUTER PROTECTION SYSTEM

DESCRIPTION

At the time this issue was identified in NUREG-0371,[2] trends in the design of nuclear power plants showed an increase in the use of digital computer technology in safety-related instrumentation and control systems. The first application of this technology was Arkansas Nuclear One, Unit 2 (ANO-2), where digital computers were used in the initiating logic for two reactor trip parameters. After the ANO-2 application, other digital computers, such as core protection calculators, were installed by licensees to provide reactor trip signals.

Since digital technology is considerably different from analog technology, the criteria appropriate for the safety review of digital computer-based systems are different from those used for analog-based systems. Thus, in this issue, the staff identified the need to standardize the safety review of reactor protection systems that incorporated digital computers. It was believed that the results of such standardization would be: (1) the definition of the staff's requirements for the design, development, and qualification of digital computers for use by applicants; and (2) an SRP[11] that would define uniform and consistent guidelines for the conduct of the staff's safety review.

CONCLUSION

In 1982, ANS and IEEE jointly approved the standard ANSI/IEEE-ANS-7-4.3.2[1324] which established a method for designing, verifying, and implementing software, and validating computer systems used in the safety-related systems of nuclear power plants.[1237] In 1985, the NRC issued Regulatory Guide 1.152[1325] which endorsed the method in ANSI/IEEE-ANS-7-4.3.2-1982.[1324] At the time this issue was evaluated in 1991, the staff was conducting a research program to investigate the use of digital computer safety systems at nuclear power plants.[1286] In particular, specific licensing needs in the area of microcomputer and Artificial Intelligence Systems had been identified and were to be addressed. The desired end product of the research effort was a regulatory guide for the design, development, acceptance testing, and periodic functional verification of Class 1E computer safety systems, and an SRP[11] addendum providing review guidance for digital computer systems in nuclear power plant safety systems (by referencing Regulatory Guide 1.152[1325] and the new regulatory guide).

Since this issue addressed the use of alternative (i.e., digital instead of analog) technology for nuclear power plant safety systems, it was not intended that the use of digital technology would result in a change in the safety of existing nuclear power plants. Thus, the issue addressed the staff's efforts in improving its capability to make independent assessments of safety and was classified as a Licensing Issue.

REFERENCES

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

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

1237. NUREG/CR-5420, "Multiple System Responses Program - Identification of Concerns Related to A Number of Specific Regulatory Issues," U.S. Nuclear Regulatory Commission, October 1989.

1286. Memorandum for M. Virgilio from S. Newberry, "Proposed Research Programs to Support SICB Regulation Needs," April 26, 1990.

1324. ANSI/IEEE-ANS-7-4.3.2-1982, "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations," American Nuclear Society, July 6, 1982.

1325. Regulatory Guide 1.152, "Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, November 1985.

ITEM A-20: IMPACTS OF THE COAL FUEL CYCLE

DESCRIPTION

At the time this issue was identified in NUREG-0371,[2] compliance with NEPA required that alternatives to a proposed Federal action be considered, and that required alternatives be balanced against the base case in terms of their associated environmental impacts. NRC had established, through its rulemaking authority, a generic description and evaluation of the environmental impacts of the uranium fuel cycle in WASH-1248,[456] NUREG-0116,[457] and NUREG-0216.[458] Based on these studies, a summary table, Table S-3, had been prepared and promulgated as regulation in 10 CFR Part 51.20(e).

In 1978, a coal-fired plant was considered the only realistic alternative to a nuclear power plant. Existing treatment of the coal alternative was aimed essentially at economics and public health impacts; it was relatively incomplete in other areas of impact. It was believed that the comparison of the coal alternative to a proposed nuclear facility would be significantly improved, if a study were conducted for the coal fuel alternative that augmented the work that had been done by ANL in the area of health effects. Such a study would provide a comprehensive summary which evaluated the environmental effects of the coal fuel cycle in a form directly comparable to that for the uranium fuel cycle. In the absence of such a generic treatment of the effects of using coal for generating electric power, it was necessary for the staff to develop an analysis *de novo* for each licensing action, to present this individual analysis in detail in the EIS, and to defend it throughout the hearing process. It was believed that this repetitive staff effort could be avoided by preparing a generic statement suitable to support rulemaking proceedings. After the rulemaking procedure, such a statement would have the force of law necessary to avoid repetitive staff effort.

A thorough analysis of alternatives to a proposed nuclear power plant required an evaluation of the environmental effects of the coal fuel cycle to the same extent as the nuclear cycle. The environmental effects of the coal fuel cycle had long been recognized as being significant. There were deleterious effects to human health due to burning coal, but there were other significant socioeconomic and other environmental impacts at each stage of the cycle. For example, mining coal exacts a penalty in human health and safety, may require modification of large areas of land use requiring expensive reclamation and habitat restoration, and frequently produces polluting liquid and solid mine wastes. Environmental, social, economic, and health effects also accompany the transportation, storage, treatment, combustion, and waste management and disposal aspects of the fuel cycle. Failure to treat these factors had been criticized by ASLB and the ASLAB in the past, necessitating increased staff efforts in this direction.

CONCLUSION

This issue addressed the staff's efforts in improving its capability to make independent assessments of safety and, therefore, was considered a Licensing Issue. The issue had been covered extensively in NUREG-0252,[459] NUREG/CR-1060,[460] and NUREG-0332,[461] and further work on the subject had been discussed with personnel of the National Academy of Sciences who had expressed the view that adequate scientific bases for analyzing impacts of coal burning did not exist. It was thought that a workshop could be arranged to determine what the questions were and how they could be resolved. Definitive answers required an extensive program over a period of years and the role of the NRC in carrying out such a program was expected to be determined by the Commission.[412] The results of this issue were expected to be used in Item B-72.[2]

REFERENCES

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

412. Memorandum for T. Speis from J. Funches, "Prioritization of Generic Issues - Environmental and Licensing Improvements," February 24, 1983.

456. WASH-1248, "Environmental Survey of the Uranium Fuel Cycle," U.S. Atomic Energy Commission, April 1974.

457. NUREG-0116, "Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," U.S. Nuclear Regulatory Commission, October 1976.

458. NUREG-0216, "Public Comments on the Environmental Survey of the Reprocessing and Waste Management Portions of the LWR Fuel Cycle," U.S. Nuclear Regulatory Commission, March 1977.

459. NUREG-0252, "The Environmental Effects of Using Coal for Generating Electricity," U.S. Nuclear Regulatory Commission, June 1977.

460. NUREG/CR-1060, "Activities, Effects, and Impacts of the Coal Fuel Cycle for a 1,000 MWe Electric Power Generating Plant," U.S. Nuclear Regulatory Commission, February 1980.

461. NUREG-0332, "Health Effects Attributable to Coal and Nuclear Fuel Cycle Alternatives," U.S. Nuclear Regulatory Commission, November 1977.

ISSUE 172: MULTIPLE SYSTEM RESPONSES PROGRAM

In resolving GSIs over the years, the staff generally found it necessary to make assumptions and establish limitations on the scope of the issues. As a result of its review of the resolution of some GSIs, the ACRS expressed concerns that the assumptions and limitations on the scope of the issues, the lack of thorough coordination among issues, and the inconsistent assumptions for related issues may have resulted in some potentially significant safety concerns not being addressed. Specifically, these concerns were raised in ACRS meetings during the resolution of Issues A-17, A-46, and A-47. To address these concerns, RES initiated the Multiple System Responses Program (MSRP) program in 1986.

The purpose of the MSRP was to gather and review documentation (correspondence, meeting minutes, etc.) for the issues and other programs of interest and, from this documentation, describe potential safety concerns that were identified or expressed by the ACRS or NRC staff. The issues selected for the MSRP were A-17, A-46, and A-47. Issues that involved concerns similar to those addressed in the resolution of these three issues were also considered and included: (1) equipment qualification (10 CFR 50.49); (2) fire protection rules (10 CFR 50.48 and 10 CFR 50, Appendix R); and (3) related guidelines and reviews implemented based on the SRP.[11] In the MSRP, evaluations or judgments were not made regarding the validity of the concerns; rather, the concerns were examined, documented, and potential safety issues were defined as specifically as possible. The results of this effort were documented in NUREG/CR-5420.[1237]

In NUREG/CR-5420,[1237] related concerns were grouped into defined potential safety issues and information was provided to assist the staff in evaluating them. This grouping was based on the following criteria: (1) concerns that had the same initiator (e.g., seismic event, flooding/moisture intrusion, fires); (2) concerns that related to a particular class of failures or failure modes (e.g., degradation of component performance rather than "failure," or common cause failures); (3) concerns that related to a particular group of components or systems (e.g., non-safety-related control system and safety-related protection system dependencies); (4) concerns that already existed as GSIs; and (5) concerns that were unrelated to other concerns or that were being evaluated through separate research activities and should be separate issues. Applying these criteria to the identified concerns yielded 21 potential safety issues.

Of the 21 MSRP concerns, the staff concluded that eleven were to be covered in the IPE or IPEEE Programs. The remaining ten concerns were dropped from further consideration as new and separate issues because eight were included in the scope of existing generic issues or other ongoing NRC programs, one (Item 4) had negligible risk reduction potential, and one (Item 9) was deemed to be a compliance concern. This conclusion was reached after several meetings between the ACRS and the staff and an extensive review[1581] of the ACRS concerns by the staff. A comprehensive report[1580] on the staff's findings was submitted to the ACRS. The following is a summary of the staff's findings:

IPE/IPEEE Programs

(1) Common Cause Failures Related to Human Errors (IPE)
(2) Non-Safety-Related Control System/Safety-Related Protection System Dependencies (IPE)

(13) Effects of Fire Suppression System Actuation on Non-Safety-Related and Safety-Related Equipment (IPEEE)
(14) Effects of Flooding and/or Moisture Intrusion on Non-Safety-Related and Safety-Related Equipment (IPE/IPEEE)
(15) Seismically-Induced Spatial and Functional Interactions ((IPEEE)
(16) Seismically-Induced Fires (IPEEE)
(17) Seismically-Induced Fire Suppression System Actuations (IPEEE)
(18) Seismically-Induced Flooding (IPEEE)
(19) Seismically-Induced Relay Chatter (IPEEE)
(20) Evaluation of Earthquake Magnitudes Greater Than the Safe Shutdown Earthquake (IPEEE)
(21) Effects of Hydrogen Line Ruptures (IPEEE)

DROP

(3) Failure Modes of Digital Computer Control Systems
(4) Specific Scenarios Not Considered in USI A-47
(5) Effects of Degradation of HVAC Equipment on Control and Protection Systems
(6) Failure Modes Resulting from Degraded Electric Power Sources
(7) Failure Modes Resulting from Degraded Compressed Air Systems
(8) Potential Effects of Untimely Component Operation
(9) Propagation of Environments Associated with DBEs
(10) Evaluation of Heat, Smoke, and Water Propagation Effects Resulting from Fires
(11) Synergistic Effects of Harsh Environmental Conditions
(12) Environmental Qualification of Seals, Gaskets, Packing, and Lubricating Fluids Associated with Mechanical Equipment

Based on the ongoing work to address the safety concerns, the issue was considered nearly-resolved in December 1995, but was later given a high priority ranking in SECY-98-166.[1718] The MSRP was considered resolved at the conclusion of the IPE/IPEEE Programs when a summary report was issued[1806] on how the above eleven concerns were addressed. The staff's evaluations of the above 21 concerns are presented below:

(1) COMMON CAUSE FAILURES RELATED TO HUMAN ERRORS

DESCRIPTION

CCF resulting from human error include operator acts of commission or omission that could be initiating events or could affect redundant safety-related trains needed to mitigate the events. Other human errors that could initiate CCF include: (1) manufacturing errors in components that affect redundant trains; and (2) installation, maintenance, or testing errors that are repeated on redundant trains. Since personnel are always intimately involved in all phases of nuclear power plant planning, operation, testing, and maintenance, there is the potential for human errors which may contribute or lead to systems interaction events or CCF. This concern was identified as Item 7.4.1 in NUREG/CR-5420.[1237]

CONCLUSION

While existing PRAs have identified human error possibilities to some extent, they are principally limited to errors of omission. The identification or the modeling of errors of commission is in the developmental stages and will continue to require further work. Efforts to increase understanding and preclude the occurrence of this type of human error will continue to be a priority research activity. With the use of NUREG/CR-5455,[1582] the staff has been following the investigations of events at operating plants in recent years that involved human performance. In conducting control room design reviews, the staff uses the criteria documented in NUREG-0711[1583] and NUREG/CR-5908.[1584]

The staff will continue the present approach of reducing human errors of all types through regulatory review, inspection, research, and the development of regulatory guidance based upon systematic application of human engineering principles, rather than attempting to identify and correct specific human errors that may lead to CCF. Additionally, potential CCFs resulting from human errors of omission in operation, maintenance, or testing are to be considered on a plant-specific basis by licensees in their IPEs. (CCFs resulting from human errors in installation and manufacturing of components are generally not explicitly considered in PRAs and hence would not be explicitly considered in the IPE process.)

The staff's approach will reduce the likelihood of human errors, including those that have not been identified thus far. The staff believes that the potentially significant generic issues associated with CCFs related to human errors are currently being addressed by this approach. Therefore, based on the existing IPE Program, this concern was not pursued as a new and separate issue.

(2) NON-SAFETY-RELATED CONTROL SYSTEM/SAFETY-RELATED PROTECTION SYSTEM DEPENDENCIES

DESCRIPTION

Multiple failures in non-safety-related control systems may have an adverse impact on safety-related protection systems as a result of potential unrecognized dependencies between control and protection systems. There is concern that plant-specific implementation of the regulations regarding separation and independence of control and protection systems may be inadequate. This concern was expressed by the ACRS during their review of the resolution of Issue A-47 and was identified as Item 7.4.2 in NUREG/CR-5420.[1237]

CONCLUSION

The resolution of Issue A-17 stated that "[m]ethods are available (and some are under development) for searching out systems interactions on a plant-specific basis. Studies conducted by utilities and national laboratories indicate that a full-scope plant search takes considerable time and money. Even then, there is not a high degree of assurance that all, or even most, adverse systems interactions will be discovered." Thus, the staff concluded that the cost of a systematic search of systems interactions, such as non-safety-related control system/safety-related protection system dependencies, would produce very little safety benefit.

The summary of NUREG/CR-5420[1237] states that this issue "does not question regulations but addresses plant-specific implementation." As such, the licensees' IPE process should provide a framework for evaluating interdependence between safety-related and non-safety-related systems and identify potential sources of vulnerabilities. Continued notices, letters, and bulletins addressing identified problems of this nature should aid in the identification and resolution at those plants where these or similar weaknesses may exist. Therefore, based on the existing IPE Program, this concern was not pursued as a new and separate issue.

(3) FAILURE MODES OF DIGITAL COMPUTER CONTROL SYSTEMS

DESCRIPTION

Two areas of concern were identified for digital computer control systems. The first is the potential for interactions between computerized non-safety-related control systems and safety-related protection systems. Use of computerized control systems presents the potential for complex or unexpected failure modes that might impact protection systems. The second area of concern is the use of digital control systems for safety-related purposes. The first OL application including this type of equipment for safety-related purposes (although on a small scale) was ANO Unit 2, where digital computers are used for the initiating logic for two reactor trip parameters. Several utilities are implementing core protection calculators (CPC), which are digital components, to provide trip signals. This concern was identified as Item 7.4.3 in NUREG/CR-5420.[1237]

This ACRS concern was based on the potential failure of digital computer control systems which may affect the safe shutdown capability of a plant. It applies primarily to the adequacy of NRC regulations and the NRC's capability to review designs for such equipment.

CONCLUSION

For the review and evaluation of digital instrumentation and control systems (including the interface design and the software to drive them), methods and technical bases for guidelines and criteria are being developed in the ongoing NRC research on human-system interface. The many research issues include the potential for interactions between computerized non-safety related control systems and safety-related protection systems. The research also addresses the use of digital instrumentation and control systems for safety-related purposes. Additional work is being initiated with the National Academies of Sciences and Engineering under a study titled "Study and Workshop on Application of Digital Instrumentation and Control Systems to Nuclear Power Plants," to identify the important safety and reliability issues associated with the use of digital instrumentation and control systems, and to address what approach and criteria should be applied to ensure safe application and effective regulation of digital instrumentation and control systems.

In addition, potential failure modes and interactions in computer systems are being considered in the NRR review of digital systems in operating plants and advanced reactors. Based on the ongoing work, this concern was dropped from further consideration as a new and separate issue.

(4) SPECIFIC SCENARIOS NOT CONSIDERED IN USI A-47

DESCRIPTION

The staff identified two scenarios of concern that were not evaluated during the review of Issue 47: (1) scram without turbine trip, including return to criticality resulting from overcooling the primary system; and (2) steam generator overfill resulting from SGTR leading to an MSLB and more SGTRs that would involve the blowdown of more than one steam generator. The other potential cause of steam generator overfill (excessive feedwater flow due to control system failure) and its consequences were analyzed in the resolution of Issue A-47. This concern was identified as Item 7.4.4 in NUREG/CR-5420.[1237]

CONCLUSION

The first scenario was addressed in Issue 144 which was given a low priority ranking. The second scenario, along with other concerns, was addressed in Issue 135 which was given a medium priority ranking and resolved by the staff. In NUREG/CR-4893,[1411] the staff's technical findings report for Issue 135, it was stated that for steam generator overfill resulting from SGTR "[a]nalyses for several plants on the increase in stress levels due to deadweight loading resulting from filling the steam lines indicate that, while in some cases the spring hangers may be loaded slightly beyond specification, they will not fail. The stress levels in the main steam line will remain within ASME Code limits in all cases. The NRC staff has concluded that the probability of failure of the main steam line is not increased by the deadweight loading. Further, because the water in the steam lines is essentially at saturation temperature and pressure, the potential for failure due to condensation-induced water hammer is considered insignificant ... there is no evidence of steam line failure from overstress, and dynamic loading from water hammer is not considered to be a problem."

Since steam generator overfill resulting from an SGTR is not likely to lead to an MSLB, an SGTR caused by an SGTR-induced MSLB and associated mechanical and thermal shock are also not very likely. Based on this low probability event, this concern was dropped from further consideration as a new and separate issue. Consideration of a 20-year license renewal period would not change this conclusion.

(5) EFFECTS OF DEGRADATION OF HVAC EQUIPMENT ON CONTROL AND PROTECTION SYSTEMS

DESCRIPTION

Instrumentation systems generally require a carefully controlled environment to function properly. Loss or degradation (i.e., partial loss) of either safety or non-safety-related HVAC systems could result in the failure of systems necessary to achieve and maintain safe shutdown. HVAC degradation can have a direct impact on safety-related equipment or an indirect impact through interactions with non-safety-related components. The possibility for HVAC degradation to have an undesirable impact on safety-related protection systems may not have been given adequate attention. This concern was identified as Item 7.4.5 in NUREG/CR-5420.[1237]

CONCLUSION

The concern for the effects of loss of HVAC/chilled water systems on safety-related systems and components was addressed in the resolution of Issue 143. In the regulatory analysis for the resolution for this issue documented in NUREG/CR-6084,[1550] it was indicated that the reduction in annual CDF by eliminating (or decreasing) the dependence of safety systems on HVAC and room cooling was only on the order of 10^{-6}/RY, and all three proposed resolution strategies exceeded the $1,000/man-rem cost-effectiveness ratio. Therefore, the staff did not recommend any new requirements in the resolution of Issue 143.

Although the effects of degradation (such as decrease in efficiency) of HVAC/chilled water systems were not considered in Issue 143, and only the effects of loss of HVAC/chilled water systems on safety-related systems and components were considered, Issue 143 did provide a worst-case scenario that enveloped the concerns of Item 7.4.5. This conclusion was based on the following: (1) the effects of degradation (partial loss) of HVAC/chilled water systems on systems and components will be less severe compared to those from the total loss of HVAC/chilled water systems; and (2) the indirect impact of HVAC degradation on safety-related equipment through interactions with non-safety-related components will lead to the same end results as the direct impact of loss of HVAC/chilled water systems on safety-related equipment. Therefore, the ACRS concerns were bounded by Issue 143 and were dropped from further consideration as a new and separate issue.

(6) FAILURE MODES RESULTING FROM DEGRADED ELECTRIC POWER SOURCES

DESCRIPTION

Electric power system degradation (i.e., undervoltage, overvoltage, underfrequency, overfrequency) has the potential for affecting multiple trains of safety-related equipment although it is not clear what failure modes could result from these types of events. The ACRS believed that, although Issue A-47 addressed sudden complete loss of electrical power, it did not address the effects of electric power system degradation on safety-related equipment. This concern was identified as Item 7.4.6 in NUREG/CR-5420.[1237]

CONCLUSION

The concern for electrical power reliability was addressed in the resolution of Issue 128 which was established to integrate the resolution of 3 separate safety issues: 48, "Limiting Conditions for Operations (LCOs) for Class 1E Vital Instrument Buses"; 49, "Interlocks and LCOs for Class 1E Tie Breakers"; and A-30, "Adequacy of Safety-Related DC Power Supplies." However, the resolution of Issue 128 did not specifically address "degradation" of electrical power systems and its consequences. Issue A-35, "Adequacy of Offsite Power Systems," did address the concern for the vulnerability of safety-related equipment to sustained degraded voltage from offsite power sources. It also addressed the concern relating to a rapid rate of frequency decay of the offsite power system.

The concerns regarding the performance of MOVs under degraded electric power sources, among other things, were addressed in the resolution of Issue II.E.6.1, "In Situ Testing of Valves - Test Adequacy Study," and resulted in the issuance of Generic Letter 89-10[1217] which required licensees to establish programs to ensure the operability of MOVs in safety-related systems. In the resolution of Issue 158, "Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions," currently being resolved with a high priority, the staff will investigate the performance of safety-related, power-operated valves such as SOVs, AOVs, and HOVs under design basis conditions. Thus, Item 7.4.6 is being addressed for power-operated valves.

Lastly, there was an extensive inspection program initiated by NRR in the late 1980s entitled Electrical Distribution System Functional Inspections (EDSFI) where all operating plants were reviewed and inspected regarding the design, operation, maintenance, and testing of their electrical distribution systems; both offsite and onsite electrical power systems were included. A number of information notices were issued as a result of this inspection program and an EDSFI data bank is being maintained by RSIB/NRR. RES, in consultation with NRR, will consider the information in the

EDSFI data bank and other pertinent operational experiences, to determine the effects on component operation by degraded input power and if further NRC action is appropriate.
With the completed and ongoing programs described above, this concern was dropped from further consideration as a new and separate issue.

(7) FAILURE MODES RESULTING FROM DEGRADED COMPRESSED AIR SYSTEMS

DESCRIPTION

Compressed air system degradation has the potential to affect multiple trains of safety-related equipment. Air system degradation includes: (1) gradual loss of air pressure; and (2) air underpressurization or overpressurization outside the design operating pressure range of the associated equipment dependent upon this system. It is not clear what failure modes could result from these types of events. Although Issue A-47 addressed sudden complete loss of air pressure, it did not specifically investigate the effects of compressed air system degradation on safety-related equipment. This concern was identified as Item 7.4.7 in NUREG/CR-5420.[1237]

CONCLUSION

Issue 43, "Reliability of Air Systems," which was resolved with the issuance of Generic Letter 88-14,[1141] addressed, to a large extent, the ACRS concern on air system reliability. However, the ACRS stated[1579] that "we do not consider the resolution of Generic Issue 43 as adequate. We support what has been proposed or done by the staff and the industry as described in the resolution package for Generic Issue 43, but further work is needed to show that the gradual loss of air pressure issue is not a safety problem for any plant."

In AEOD/C701,[1078] five recommendations to address air systems problems were made. Recommendation 5 stated that "[a]ll operating plants should be required to perform gradual loss of instrument air system pressure tests." CRGR considered the five recommendations while deliberating on the issuance of Generic Letter 88-14[1141] and concluded that licensees should implement four of the five recommendations. Recommendation 5, pertaining to slow bleed-down testing, was not supported by CRGR because it was believed that the other four recommendations would be effective in correcting the problems.

The issuance of Generic Letter 88-14[1141] resulted in major utility efforts in which dozens of air system problems that had the potential to compromise public health and safety were found and corrected. In addition, AEOD now believes that the importance of the slow bleed-down test recommendation has actually diminished because of the efforts that many licensees have made to find and correct other air system problems and the aggressive industry initiatives to improve the reliability of air-operated equipment. Evidence of these activities are: (1) INPO and EPRI/NSAC issued reports encouraging utilities to take actions to correct problems noted in NUREG-1275,[1079] Vol. 2; (2) EPRI/NMAC issued maintenance guides on air systems and SOVs; (3) the Air Operated Valve Users' Group was formed and members meet on a regular basis to exchange information and promote reliable equipment operation; and (4) there is an ongoing process to establish an ASME O&M performance guide/standard for air systems.

The slow bleed-down test will require the determination of the range of credible blowdown rates, and the performance of sequential testing of individual branches of the air distribution system to avoid creating a challenge to plant safety. In addition, to fully implement the slow bleed-down test recommendation could require expenditure of disproportionate amounts of resources and may also result in increased risk due to the introduction of unnecessary challenges to plant safety. AEOD is monitoring improvements in plant performance pursuant to Generic Letter 88-14.[1141] Based on the above actions that have been taken, this concern was dropped from further consideration as a new and separate issue.

(8) POTENTIAL EFFECTS OF UNTIMELY COMPONENT OPERATION

DESCRIPTION

This concern addressed the effects of components potentially changing state or actuating in an unanticipated sequence from spurious signals. This scenario can potentially cause damage to safety-related equipment. This concern was identified as Item 7.4.8 in NUREG/CR-5420.[1237]

CONCLUSION

The staff reviewed existing programs and found that this concern has been adequately addressed by existing generic issues and other NRC programs. This review involved an evaluation of operational events studied under the Accident Sequence Precursors Program which indicated that the major cause of untimely equipment operation is human error which will be reduced by the application of human engineering principles (See Item 7.4.1). In addition, the only effects from the untimely operation of equipment in many of the events are spurious reactor, generator, or turbine trip. The remaining events involve accident sequences which are within the scope of existing generic issues, or involve accident sequences which are within the design basis of plants, such as loss of one out of two redundant ESF trains. Consequently, the staff believed that the potential effects of untimely component operation have been adequately addressed by existing generic issues and other NRC programs and this concern was dropped from further consideration as a new and separate issue.

(9) PROPAGATION OF ENVIRONMENTS ASSOCIATED WITH DBEs

DESCRIPTION

A harsh environment results from certain DBEs (i.e., MSLB, HELB, or LOCA). Equipment exposed to such environments must be qualified to withstand the severe conditions (e.g., the combined effects of high temperature, pressure, humidity/moisture, radiation, and submergence). The actual zone of influence for a particular environment can be larger than the zone used in the analysis if the harsh environment propagates by some unknown or unrecognized path (e.g., open floor drains) into another zone. The following scenario was to be considered:

> Steam from an MSLB could travel from where it occurs into another area or zone. This could result in higher temperature, higher pressure, or higher humidity in the other zone. Equipment required for safe shutdown in this area may not be qualified to operate in such a harsh environment. Licensees may not have considered such pathways as HVAC ducts and electrical conduits to propagate harsh environments when performing their environmental qualification analyses.

This concern was identified as Item 7.4.9 in NUREG/CR-5420.[1237]

CONCLUSION

10 CFR 50.49 requires that the DBE environmental conditions (e.g., the time-dependent temperature, pressure, humidity, radiation, chemicals, submergence, etc.) be specified in the qualification file at locations where equipment important to safety must perform and this equipment, in turn, must be qualified to these DBE environmental conditions. The staff considered the scenario described above to be an issue of compliance with 10 CFR 50.49 and this concern was dropped from further consideration as a new and separate issue.

(10) EVALUATION OF HEAT, SMOKE, AND WATER PROPAGATION EFFECTS RESULTING FROM FIRES

DESCRIPTION

Fire can damage one train of equipment in one fire zone while a redundant train could potentially be damaged in one of the following ways:

(1) Heat, smoke, and water may propagate (e.g., through HVAC ducts or electrical conduit) into a second fire zone and damage a redundant train of equipment.

(2) A random failure, not related to the fire, could damage a redundant train.

(3) Multiple non-safety-related control systems could be damaged by the fire and their failure could affect safety-related protection equipment for a redundant train in a second zone.

A fire can cause unintended operation of equipment due to hot shorts, open circuits, and shorts to ground. Consequently, components could be energized or de-energized, valves could fail open or closed, pumps could continue to run or fail to run, and electrical breakers could fail open or closed. This concern was identified as Item 7.4.10 in NUREG/CR-5420.[1237]

CONCLUSION

The concern of water propagation effects resulting from fire was partially addressed in the resolution of Issue 57. For operating and future plants having a greater reliance on advanced digital instrumentation and control (I&C) systems, there is a separate ongoing RES program to investigate the effects of smoke (SNL/FIN W6051) together with synergistic effects from temperature, moisture/humidity, electromagnetic interference/radio frequency interference (EMI/RFI), etc., (ORNL/FIN L1798, ORNL/FIN L1951) on these systems. This study will involve identifying all plausible environmental stressors associated with the advanced digital I&C systems, collecting reliability data for components that are unique for the advanced digital I&C systems, and prioritizing these environmental stressors (including the synergistic effects) based on their risk significance (BNL/FIN L1908). The results of this study will be incorporated into an ORNL program on Qualification of Advanced Instrumentation and Control Systems (See initial results in NUREG/CR-5904[1668] and NUREG/CR-5941[1669]). Based on the above actions that have been taken, this concern was dropped from further consideration as a new and separate issue.

(11) SYNERGISTIC EFFECTS OF HARSH ENVIRONMENTAL CONDITIONS

DESCRIPTION

A synergistic effect is one in which the presence of simultaneous combined environmental conditions has a greater impact on equipment than the sum of the individual environmental conditions taken independently or sequentially. The ACRS contends that a lack of regulatory guidance for analyzing synergistic effects makes it difficult to assess what licensees have done in this area and, therefore, some equipment important to safety may not be adequately qualified for the actual environments. This concern was not combined with other concerns because it relates to a specific part of the environmental qualification (EQ) issue, namely, synergistic environmental effects. This concern was identified as Item 7.4.11 in NUREG/CR-5420.[1237]

CONCLUSION

10 CFR 50.49(e)(7) states that synergistic effects must be considered when these effects are believed to have a significant effect on equipment performance. The staff believed that, although regulatory guidance for analyzing synergistic effects is currently lacking, there is sufficient ongoing staff action to evaluate and resolve existing EQ concerns and to identify and resolve any other EQ issues that may exist. RES is currently working with NRR on the planned actions of the EQ 10 CFR 50.49 Task Action Plan (EQ-TAP) where the adequacy of existing EQ standards and regulations for operating reactors is to be evaluated. The EQ-TAP stated that "[a]lthough this TAP describes planned actions, it should be recognized that this is an evolving issue and the actions, as described, may be modified as additional information is obtained through further research and review of industry operating experience." The RES program plan for the EQ-TAP will include synergistic effects. Thus, the concerns of NUREG/CR-5420,[1237] Item 7.4.11 will be included in the EQ-TAP and additional guidance will be issued if appropriate. Therefore, this concern was dropped from further consideration as a new and separate issue.

(12) ENVIRONMENTAL QUALIFICATION OF SEALS, GASKETS, PACKING, AND LUBRICATING FLUIDS ASSOCIATED WITH MECHANICAL EQUIPMENT

DESCRIPTION

Sub-components (seals, gaskets, packing materials, and lubricating fluids, etc.) in some mechanical equipment may not be adequately qualified to normal harsh environments due to the lack of concerted industry equipment qualification programs on mechanical equipment and NRC review. This is possible because currently no specific NRC guidelines equivalent to 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," exist for mechanical equipment. This concern was identified as Item 7.4.12 in NUREG/CR-5420.[1237]

CONCLUSION

Previously-identified generic issues addressed the operability and reliability of PORVs, MOVs, and other power-operated valves. Specifically, Generic Letter 89-10[1217] was issued for Issue II.E.6.1; Generic Letter 90-06[1290] was issued for Issue 70; and Issue 158, "Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions," is currently being resolved with a high priority and will address the operability and reliability of AOVs, HOVs and SOVs.

The EPRI-sponsored reliability-centered maintenance program (RCM) and the associated Users' Group have been in existence for some time and are now well-represented by the nuclear utilities. This maintenance program encompasses equipment and components (includes non-metallic parts which is the focus of concern of this issue), and regularly identifies and replaces unqualified or degraded components and sub-components. The Users' Group members meet on a regular basis (with participation from the NRC staff) to exchange information on RCM and promote reliability of equipment and components.

In addition, an ASME Standard on environmental qualification of mechanical equipment (QME) is scheduled for issuance. This document will help to address the concerns of this item for future plants and for replacements at operating plants. Based on the above actions that have been taken, this concern was dropped from further consideration as a new and separate issue.

(13) EFFECTS OF FIRE SUPPRESSION SYSTEM ACTUATION ON NON-SAFETY-RELATED AND SAFETY-RELATED EQUIPMENT

DESCRIPTION

Fire suppression system actuation events can have an adverse effect on safety-related components either through direct contact with suppression agents or through indirect interactions with non-safety-related components. This concern was identified as Item 7.4.13 in NUREG/CR-5420.[1237]

CONCLUSION

This concern was addressed in the resolution of Issue 57 and will be considered by licensees on a plant-specific basis during implementation of the IPEEE Program. Supplement 4 to Generic Letter 88-20[1222] and NUREG-1407[1354] provided procedural and submittal guidance for the IPEEE Program. As stated in NUREG-1407[1354] for internal fires, some fire issues identified in NUREG/CR-5088[1211] such as seismic/fire interaction, effects of fire suppressants on safety equipment, and control system interactions, should be addressed in the IPEEE. Based on the existing IPEEE Program, this concern was not pursued as a new and separate issue.

(14) EFFECTS OF FLOODING AND/OR MOISTURE INTRUSION ON NON-SAFETY-RELATED AND SAFETY-RELATED EQUIPMENT

DESCRIPTION

Flooding and/or water intrusion events can affect safety-related equipment either directly or indirectly through flooding or moisture intrusion of multiple trains of non-safety-related equipment.

This type of event can result from external flooding events, tank and pipe ruptures, actuations of fire suppression system, or backflow through part of the plant drainage system. This concern was identified as Item 7.4.14 in NUREG/CR-5420.[1237]

CONCLUSION

The purpose of this concern was to determine whether additional regulations or more detailed requirements would result in a significant improvement in public health and safety. However, there is no evidence that this safety concern could be resolved in this manner. Instead, if a potential safety problem exists, it would appear to be a result of plant-specific vulnerabilities.

The IPE submittal guidance (Generic Letter 88-20[1222] and NUREG-1335[1587]) includes consideration of moisture intrusion and internal flooding. The concern for external flooding and/or moisture intrusion resulting from external events is being addressed in the IPEEE Program. Thus, the IPE/IPEEE process should detect plant-specific vulnerabilities identified in the ACRS concern. Based on the existing IPEand IPEEE Programs, this concern was not pursued as a new and separate issue.

(15) SEISMICALLY-INDUCED SPATIAL AND FUNCTIONAL INTERACTIONS

DESCRIPTION

Seismic events have the potential to cause multiple failures of safety-related systems through spatial and functional interactions. In particular, additional analyses may be necessary to ensure the following:

(1) small piping (e.g., air, instrument, and water lines) is properly evaluated to prevent small pipe ruptures that may disable essential plant shutdown systems;

(2) non-seismically qualified structures, systems and components cannot cause small piping failures from direct impact;

(3) seismic activity will not adversely affect safety-related protection systems via multiple non-safety-related control system failures and/or functional interactions (excluding direct impact); and

(4) indirect effects of seismic activity such as dust generation cannot affect essential plant shutdown systems.

The ACRS expressed concern that not all of the potential seismically-induced system interactions that could adversely affect safe shutdown of a plant have been thoroughly identified and investigated. This concern was identified as Item 7.4.15 in NUREG/CR-5420.[1237]

CONCLUSION

The procedural and submittal guidance document[1354] for the IPEEE Program states that, for seismic review, plant walkdowns must be performed consistent with the intent of the guidelines described in Sections 5 and 8 and Appendices D and I of the EPRI Seismic Margins Methodology

(EPRI NP-6041). EPRI NP-6041 in turn states that seismic systems interactions reviews should be one of the items performed during a plant walkdown and guidelines on how to perform these reviews are provided. These guidelines address the concern for seismically-induced spatial interactions; it is expected that implementation of the IPEEE Program will identify any vulnerabilities to seismically-induced functional interactions. Thus, licensee evaluations of their plants for vulnerabilities to seismic events as part of the IPEEE Program are sufficient to address the ACRS concern. Based on the existing IPEEE Program, this concern was not pursued as a new and separate issue.

(16) SEISMICALLY-INDUCED FIRES

DESCRIPTION

Seismically-induced fires have the potential to cause multiple failures of safety-related systems. The occurrence of a seismic event could create fires in multiple locations, simultaneously degrade fire suppression capability (because fire suppression systems are not seismically-qualified), and, therefore, prevent mitigation of fire damage to multiple safety-related systems. The ACRS expressed concern that seismically-induced fires were not adequately addressed in the resolution of Issue A-46, other seismic requirements, or fire protection regulations. This concern was identified as Item 7.4.16 in NUREG/CR-5420.[1237]

CONCLUSION

In resolving Issue 57, the staff considered the results of the PRA analyses for 4 operating plants (1 GE, 1 B&W, and 2 W plants) and these are summarized below.

The mean CDF from Issue 57 root causes for these 4 plants are in the range of 7.3 x 10^{-6}/RY to 5.6 x 10^{-5}/RY. The dominant risk contributors were found to be: (1) seismic-induced fire plus seismic-induced suppressant diversion, i.e., the unsuppressed fire and/or the diverted suppressant incapacitate safety-related equipment needed to mitigate effects of the seismic event; and (2) seismic-induced actuation of the fire protection systems (i.e., the released suppressant damages safety-related equipment needed to mitigate the effects of the seismic event) which are both being addressed by IPEEE (See Supplement 4 to Generic Letter 88-20[1222] and NUREG-1407[1354]). After subtracting these two dominant risk contributors, the mean CDF of remaining contributors is less than 10^{-5}/RY. Therefore, the staff recommended that, after considering credit for the IPEEE, generic backfit was not justifiable for Issue 57 and no new requirements were recommended.

Thus, the ACRS concern will be considered by licensees on a plant-specific basis during implementation of the IPEEE Program and this concern was not pursued as a new and separate issue.

(17) SEISMICALLY-INDUCED FIRE SUPPRESSION SYSTEM ACTUATIONS

DESCRIPTION

Seismic events can potentially cause multiple fire suppression system actuations which, in turn, can cause failures of redundant trains of safety-related systems. Analyses currently required by fire protection regulations generally only examine inadvertent actuations of fire suppression systems as single, independent events whereas a seismic event could cause multiple actuations of fire suppression systems in various areas. This concern was identified as Item 7.4.17 in NUREG/CR-5420.[1237]

CONCLUSION

As described in Item 7.4.16 above, the ACRS concern was addressed in the resolution of Issue 57 and will be considered by licensees on a plant-specific basis during implementation of the IPEEE Program. Therefore, this concern was not pursued as a new and separate issue.

(18) SEISMICALLY-INDUCED FLOODING

DESCRIPTION

Seismically-induced flooding events can potentially cause multiple failures of safety-related systems. The ACRS expressed several concerns related to seismically-induced flooding. First, although the ACRS believes that an SSE will likely not cause large-diameter piping to rupture, the ACRS feels that the seismic adequacy of smaller-diameter piping has not been adequately proven. Rupture of small piping could provide flood sources that could potentially affect multiple safety-related components simultaneously. Second, non-seismically qualified tanks are a potential source of flooding that the ACRS believes has not been adequately addressed. This concern was identified as Item 7.4.18 in NUREG/CR-5420.[1237]

CONCLUSION

Licensee evaluations of their plants for vulnerabilities to seismic events as part of the implementation of the IPEEE Program (Supplement 4 to Generic Letter 88-20[1222] and NUREG-1407[1354]) will address the ACRS concern. Therefore, this concern was not pursued as a new and separate issue.

(19) SEISMICALLY-INDUCED RELAY CHATTER

DESCRIPTION

Essential relays must operate during and after an SSE and must meet either one of the following conditions: (1) remain functional without contact chattering; (2) be seismically-qualified; or (3) licensees must show that contact chatter of the relay(s) is acceptable. It is possible that contact chatter of relays not required to operate during seismic events may produce some unanalyzed faulting mode that may impact the operability of equipment required to mitigate the event. This concern was identified as Item 7.4.19 in NUREG/CR-5420.[1237]

CONCLUSION

Licensee evaluations of their plants for vulnerabilities to seismic events as part of the implementation of the IPEEE Program (Supplement 4 to Generic Letter 88-20[1222] and NUREG-1407[1354]) will address the ACRS concern. Therefore, this concern was not pursued as a new and separate issue.

(20) EVALUATION OF EARTHQUAKE MAGNITUDES GREATER THAN THE SAFE SHUTDOWN EARTHQUAKE

DESCRIPTION

The ACRS expressed concern that adequate seismic margins may not have been included in the design of some safety-related equipment. In this context, seismic margin is defined as the capability of a plant to sustain an earthquake larger than its SSE. This concern was identified as Item 7.4.20 in NUREG/CR-5420.[1237]

CONCLUSION

Licensee evaluation of their plants for vulnerabilities to seismic events as part of the implementation of the IPEEE Program (Supplement 4 to Generic Letter 88-20[1222] and NUREG-1407[1354]) will address the ACRS concern. Therefore, this concern was not pursued as a new and separate issue.

(21) EFFECTS OF HYDROGEN LINE RUPTURES

DESCRIPTION

H_2 is used in electrical generators at nuclear plants to reduce windage losses and as a heat transfer agent. It is also used in some tanks (e.g., volume control tanks) as a cover gas. Leaks or breaks in H_2 supply piping could result in the accumulation of a combustible mixture of air and H_2 in vital areas, resulting in a fire and/or an explosion. This concern was identified as Item 7.4.21 in NUREG/CR-5420[1237] and addressed the potential for H_2 line ruptures to occur in the auxiliary building. Resulting fires and/or explosions could damage vital safety-related systems of the plant.

CONCLUSION

This concern was addressed in the resolution of Issue 106, "Piping and Use of Highly Combustible Gases in Vital Areas." The staff's technical findings and regulatory analysis were reported in NUREG/CR-5759[1544] and NUREG-1364,[1545] respectively. Generic Letter 93-06[1547] was issued to licensees and referred to new information developed in the resolution of Issue 106. This information was expected to be useful to licensees in performing their IPEEEs. Based on the above actions that have been taken, this concern was not pursued as a new and separate issue.

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1411. NUREG/CR-4893, "Technical Findings Report for Generic Issue 135, Steam Generator and Steam Line Overfill Issues," U.S. Nuclear Regulatory Commission, May 1991.

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1550. NUREG/CR-6084, "Value Impact Analysis of Generic Issue 143, 'Availability of Heating, Ventilation, Air Conditioning (HVAC) and Chilled Water Systems,'" U.S. Nuclear Regulatory Commission, November 1993.

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ISSUE 173: SPENT FUEL STORAGE POOL

In November 1992, two engineers who had previously worked under contract for the Pennsylvania Power and Light Company (PP&L) filed a report contending that the design of the Susquehanna station failed to meet regulatory requirements with respect to sustained loss of the cooling function to the SFP that mechanistically results from a LOCA or a LOOP. PP&L and the engineers each made a series of additional submittals to the NRC and participated in public meetings with the NRC to describe their respective positions on a number of technical and licensing issues. In order to inform the nuclear power industry of the issues, NRC issued IN 93-83 on October 7, 1993. The staff evaluated the issues as they related to Susquehanna, using a probabilistic safety assessment, a deterministic engineering assessment and a licensing basis analysis, and issued an SER on June 19, 1995.

A generic action plan[1623] was developed with two parts: (1) Part A, which encompassed the staff's review of generic issues relating to the SFP at operating reactor facilities; and (2) Part B, which included applicable issues from the Part A review and concerns from the Dresden-1 special inspection,[1601] particular to permanently shutdown facilities with stored, irradiated fuel to establish evaluation criteria for spent fuel pools at permanently shutdown facilities. Part B was included after the special inspection at Dresden-1 determined that problems in implementing the facility's decommissioning plan combined with certain SFP design features created the potential for a substantial loss of SFP water inventory. Dresden-1, which is permanently shutdown, experienced containment flooding due to freeze damage to the service water system on January 25, 1994, and the licensee for Dresden-1 reported a similar threat to SFP integrity. This licensee report resulted in the special inspections[1601] of La Crosse, Humboldt Bay, Rancho Seco, Trojan, San Onofre-1, Yankee Rowe, and Indian Point-1. The two parts of this issue were evaluated separately.

ISSUE 173.A: OPERATING FACILITIES

DESCRIPTION

Historical Background

The principal concerns included in Part A of the generic action plan[1623] involved the potential for a sustained loss of SFP cooling capability, which was identified through the report filed with the NRC relating to Susquehanna, and the potential for a substantial loss of SFP coolant inventory, which was given renewed emphasis following the Dresden-1 special inspection. Postulated adverse conditions that may develop following a LOCA or a sustained loss of power to SFP cooling system components could prevent restoration of SFP decay heat removal. The heat and water vapor added to the building atmosphere by subsequent SFP boiling could cause failure of accident mitigation or other safety equipment and an associated increase in the consequences of the initiating event. Incomplete administrative controls combined with certain design features, particularly at the oldest facilities, may create the potential for a substantial loss of SFP coolant inventory and the associated consequences, which include high local radiation levels due to loss of shielding, unmonitored release of radiologically contaminated coolant, and inadequate cooling of stored fuel.

The action plan was intended to encompass SFP issues identified through a 1994 special inspection at Dresden-1, the staff's review of loss of SFP cooling concerns at Susquehanna, and other SFP concerns identified as part of this plan. Specific review areas identified through implementation of this action plan include plant design features and administrative controls that affect the probability of spent fuel pool boiling, adverse environmental effects on essential equipment due to boiling, significant loss of spent fuel pool coolant inventory, adverse radiological conditions, unplanned spent fuel pool reactivity changes, undetected spent fuel pool events, and adverse effects of control system actuations. This issue was identified in an NRR memorandum[1601] to RES in February 1996.

Safety Significance

The postulated events do not pose an undue risk to the public based on the availability of design features that help protect stored irradiated fuel, protect essential reactor safety systems, and prevent development of adverse radiological conditions. These design features include the provision of diverse means of cooling, the strong structural design of the spent fuel pool, the absence of drainage paths from the pool, the anti-syphon protection on piping within the spent fuel pool, the availability of multiple sources of make-up water, spent fuel pool instrumentation with control room annunciation, the maintenance of a substantial shutdown reactivity margin in the pool, radiation shielding provided by coolant inventory, and spent fuel pool water purification systems. Additionally, the relatively slow evolution of these events in the spent fuel pool resulting from the initial large cooling water inventory creates significant opportunity for operator recovery prior to experiencing adverse conditions or consequences.

Possible Solutions

Specific actions include: (1) determination of the safety significance of identified concerns; (2) determination of the facilities where the concerns may be applicable; (3) evaluation of the adequacy of present SFP designs; (4) evaluation of the adequacy of current NRC guidance for SFP designs; and (5) evaluation of the need for generic actions to address significant issues at operating and permanently shutdown facilities. Based on findings from these review areas and their risk significance, the staff will develop criteria for specific spent fuel pool operations for potential use in formulating generic communications, revisions of regulatory guidance, and other appropriate regulatory actions.

CONCLUSION

This issue was considered nearly-resolved[1731] since a solution had been identified and resolution was in progress with an approved Action Plan. It was later given a HIGH priority ranking in SECY-98-166.[1718]

In pursuing a resolution to this issue, the staff performed a comprehensive study of the Susquehanna SFP. The results of the special inspection of Dresden-1, after rupture of the SWS occurred inside containment, were transmitted to licensees in IN 94-38.[1624] The identification of concerns for evaluation and review of existing guidance were completed along with on-site safety assessments of spent fuel storage at Brunswick, Monticello, Comanche Peak, and Ginna. The assessment team concluded that the potential for a sustained loss of SFP cooling or a significant loss of SFP coolant inventory at the sites visited was remote, based on certain design features and operational controls. The team found that other concerns within the scope of the action plan review were much less significant in terms of risk at the plants visited. An FSAR-based review was undertaken to identify facilities whose design was not well represented by any of the facilities reviewed through on-site assessments. As a result, approximately 26 concerns were identified in the major review areas; additional concerns associated with the Millstone-1 SFP (adequacy of SFP cooling during refueling with a full core off-load) were included. Each concern was to be addressed on the basis of a qualitative safety assessment. The concern for SFP criticality control (Boraflex degradation) was pursued through issuance of an information notice and a planned generic letter.

Following reports[1693,1694] to the Commission on its findings, the staff committed to complete regulatory analyses associated with plant-specific backfits, implement plant-specific backfits, and complete revisions to Regulatory Guide 1.13[1697] and SRP[11] Sections 9.1.1 and 9.1.3. The regulatory analyses were pursued by NRR under the proposed rulemaking on shutdown and fuel storage pool operation. In July 1997, the staff's proposed rule was presented to the Commission in SECY-97-168[1695] following which, the Commission directed[1696] the staff not to issue the proposed rule. After performing plant-specific evaluations and considering a license renewal period of 20 years, the issue was RESOLVED with no new or revised requirements.[1807]

ISSUE 173.B: PERMANENTLY SHUTDOWN FACILITIES

DESCRIPTION

Historical Background

The staff issued Bulletin 94-01[1625] requesting all holders of licenses for nuclear power reactors that were permanently shut down with spent fuel in the spent fuel pool to take actions to ensure the quality of the SFP coolant, the ability to maintain an adequate coolant inventory for cooling and shielding, and the necessary support systems were not degraded. In order to evaluate the management controls and SFP activities at permanently shutdown reactors, the NRC initiated a series of special team inspections at permanently shutdown facilities with stored, irradiated fuel in the SFP. This Part B effort was expected to use the results of Part A activities to establish evaluation criteria for SFPs at permanently shutdown plants to support rulemaking and other generic activities initiated by NRR. This issue was identified in an NRR memorandum[1601] to RES in February 1996.

Safety Significance

The postulated events involving a loss of cooling do not pose undue risk to the public because of the low residual decay heat in the spent fuel at permanently shutdown reactors and the associated long period of time available for recovery. Concerns involving maintenance of the coolant quality and ability to control coolant inventory were addressed through the special inspection activities. Therefore, continued facility operation was justified.

Possible Solution

Specific actions included in Part B of the generic action plan[1623] were: (1) the determination of significant identified concerns from Part A applicable to permanently shutdown facilities; and (2) the evaluation and implementation of additional requirements specifically applicable to permanently shutdown facilities with stored, irradiated fuel.

CONCLUSION

This issue was considered nearly-resolved[1731] since a solution had been identified and resolution is in progress with an approved Action Plan. The staff determined that all significant identified concerns from Part A applicable to permanently shutdown facilities were encompassed by the special inspection activities which showed no significant deficiencies other than at Dresden-1. In response to the Dresden-1 Special Inspection findings, NRR proceeded with issuance of a decommissioning action plan. Thus, this issue was RESOLVED with no new requirements.

REFERENCES

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1601. Memorandum to C. Serpan from A. Chaffee, "Nuclear Reactor Regulation (NRR) Input into Research NUREG-0933 (WITS Item 9400213)," February 13, 1996.

1623. Memorandum to A. Thadani from G. Holahan, "Task Action Plan for Spent Fuel Storage Pool Safety," October 13, 1994.

1624. NRC Information Notice 94-38, "Results of a Special NRC Inspection at Dresden Nuclear Power Station Unit 1 Following a Rupture of Service Water Inside Containment," May 27, 1994.

1625. NRC Bulletin 94-01, "Potential Fuel Pool Draindown Caused by Inadequate Maintenance Practices at Dresden Unit 1," April 14, 1994.

1693. Memorandum to Chairman Jackson, et al., from J. Taylor, "Report on Survey of Refueling Practices," May 21, 1996.

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1695. SECY-97-168, "Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation," July 30, 1997.

1696. Memorandum to L. Callan from J. Hoyle, "Staff Requirements - SECY-97-168 - Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation," December 11, 1997.

1697. Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," U.S. Nuclear Regulatory Commission, (Rev. 1) December 1975, (Draft Rev. 2) December 1981.

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

1731. Memorandum for W. Russell from D. Morrison, "Prioritization of the NRR Action Plans Submitted to RES on February 13, 1996," June 24, 1996.

1807. Memorandum to W. Travers from S. Collins, "Resolution of Generic Safety Issue (GSI) 173A, 'Spent Fuel Storage Pool for Operating Facilities,'" December 19, 2001.

ISSUE 188: STEAM GENERATOR TUBE LEAKS OR RUPTURES, CONCURRENT WITH CONTAINMENT BYPASS FROM MAIN STEAM LINE OR FEEDWATER LINE BREACHES

DESCRIPTION

Historical Background

This issue was identified when it was believed[1799] that the validity of steam generator (SG) tube leak and rupture analyses could be affected by resonance vibrations in steam generator tubes during steam line break depressurization. The concern is that an unisolable secondary system opening outside containment coupled with multiple steam generator tube leaks or ruptures could result in releases in excess of 10 CFR Part 100. The related technical issues include the ability to correctly predict SG secondary side thermal-hydraulic behavior, physical loadings, component response, resonance vibrations within the tube bundles, eddy current testing, iodine spiking, operator response, and risk. The issue is related to Issue163, "Multiple Steam Generator Tube Leakage."

Safety Significance

The issue raised the following two potentially risk-significant events that are not fully addressed as design basis accidents in FSARs, industry analyses, the SRP,[11] or staff reviews:

(1) Operating experience and design information suggested that the potential existed for a line breach to significantly increase SG leakage, because resonant vibration of SG tubes from a secondary side blowdown could cause increased tube leakage.

(2) Significant SG tube leakage could lead to secondary system breaches from a variety of causes. The resulting SG secondary side blowdown could further increase tube leakage due to resonance vibration within the affected SG tube bundle.

Such leakages, concurrent with containment bypass, might cause offsite radiation doses in excess of 10 CFR Part 100.

Main steam line break and steam generator tube rupture (SGTR) are both included as design basis accidents in Chapter 15 of most FSARs and the SRP,[11] and are addressed as accident initiators in most plant-specific PRAs. However, these accident initiators are generally assumed to occur independently unless there is severe core damage. Moreover, a SGTR is assumed to occur spontaneously in just one tube. This issue addresses the possibility of a causal relationship: a main steam or feedwater line break in an unisolable portion of the secondary system is postulated to cause a number of SG tubes to leak or rupture. Conversely, significant SG tube leakage or rupture is postulated to cause an unisolable secondary side breach which then may exacerbate the leakage.

Consequences of such an accident scenario are significant because primary coolant could be lost to the environment through the leaking or ruptured SG tubes and out the break in the secondary system. Given that the secondary side opening is outside containment but not isolable, the release of radioactivity could be above 10 CFR Part 100 limits, depending upon the iodine spiking factor and the duration of blowdown. Further, the escaping coolant will not be returned to the containment sump. There is a high probability that the ECCS will successfully mitigate a LOCA during the injection phase. However, when the refueling water storage tank (RWST) is depleted, it may not be possible to use the recirculation mode, possibly resulting in core damage. Because the release path is open to the environment outside of the containment, the release of radioactivity from the postulated core damage event could have significant risk impacts.

The issue also includes the safety concerns of increased risk from degraded operator performance because of environmental conditions that can occur during the event. Eddy current testing and iodine spiking issues were not originally identified but were included in this evaluation to provide more complete bases for understanding the safety concerns.

PRIORITY DETERMINATION

The accident scenario of concern consists of two events: (1) a non-isolable secondary system break or rupture that is outside containment; and (2) a coupling of this break with the rupture of, or significantly increased leakage from, affected SG tubes.

Non-Isolable Main Steam Line Break Outside Containment: Main steam line breaks (or equivalent ruptures in attached piping or equipment) may be caused by a combination of stresses from restriction of pipe thermal expansion by pipe supports, weld defects, lack of pipe stress relief, age-related erosion/corrosion, vibration-induced cyclic fatigue, or repeated safety valve operation causing fatigue cycles to the piping and tubes and increasing the likelihood of a safety valve sticking open. Relatively large steam line breaks have occurred outside the containment, upstream of the MSIV, during hot functional testing at Robinson 2 and Turkey Point 3. These resulted in collateral valve, piping, and equipment damage; blowdown of the affected SGs; and excessive cooldown of the RCS. In addition, large amplitude vibrations of components and structures, water hammers, and sonic booms that affected operator communication and actions were observed. The Turkey Point 3 event involved SG re-pressurization shortly after the initial blowdown as a result of collateral damage.

Other Secondary System Breaks: It is also possible to initiate the accident scenario of interest with breaks in other parts of the secondary system such as a main feedwater line, steam line supplying steam-driven auxiliary feedwater, or other steam supply lines. These would be considered within the scope of this generic issue. Main and auxiliary feedwater systems generally have check valves located inside containment, which may also fail during the event. Steam supply lines other than main steam will have their own isolation valves, and because of their smaller diameter, rupture of these lines may not cause as severe a blowdown transient. However, a smaller opening may create resonance vibrations in the affected SG that would continue for a longer period of time.

Steam Generator Tube Cracks and Test Data: PWR SG tube cracks are caused by such common-mode failure mechanisms as outside diameter stress corrosion cracking, primary water stress corrosion cracking, fretting and wear, high cycle fatigue cracking, denting, pitting, and wastage. Plant TS require that a 3% sample of SG tubes undergo NDE periodically. The percentage of tubes inspected increases as more indications are found. Existing regulatory guidance would require tubes with greater than 40% through-wall cracks to be repaired or plugged.

Eddy current testing has a variable probability of detection that depends on: the type of probe; crack width, depth, length, and orientation; background interference; and human error. While crack depth and length are the most important factors in determining SG tube integrity, accurate crack sizing by non-destructive means (eddy current, ultrasonics, etc.) remains challenging. Therefore, operation will likely occur with some degree of tube degradation at all times.

The NRC has approved several alternate repair criteria allowing small cracks to remain in service under certain conditions. Under the alternate repair criteria in Generic Letter 95-05[1804] for outside diameter stress corrosion cracks in intersections between tubes and tube support plates (TSPs), the industry must leak and burst test tube samples. However, the tubes are rigidly held in place during testing to avoid bending that would increase crack size. Tubes are tested under static conditions not subject to vibration and TSP movement that could be encountered during a main steam line break from differential pressure loadings and from vibrations at their lowest natural frequencies. Leak tests are not required to be performed at operating temperatures.

Resonance Vibrations: Resonance vibrations caused by a line break may develop in the SG internals through pressure pulses in the two-phase fluid and from pipe movement. Free span sections of tubes, portions of TSPs, and the U-tube assembly would vibrate from excitation frequencies emanating from the break. The tube/TSP movement from pressure pulses, resonance vibration, and potential steam chugging from possible recriticalities could destroy links between existing micro and macro cracks in SG tubes. Further, there has not been an integrated study of actual damage done to adjacent SG tubes following SGTRs, from steam line breaks, or from SG dry outs.

Neither resonance vibrations nor cross-flow forces can be calculated by the one-dimensional, RELAP thermal-hydraulic code. EPRI has developed multi-dimensional two-phase flow codes that are applicable only to steady-state conditions. The ACRS Ad Hoc DPO Subcommittee on SG integrity issues concluded[1800] that:

> "... *thermal-hydraulic codes usually employed by the staff for safety analyses are poorly suited to address the issues raised by this contention. The Subcommittee urges that investigation of this issue be completed expeditiously.*" (p. 10)

NRR's reviews in this area were consistent with the ACRS conclusion, since NRR has not relied upon licensee justifications based on such codes for SG secondary side analyses.

Tube Sheet Cladding Separation: Tube sheet cladding separation by the flow divider and cracks in first row tube welds and cladding may have occurred due to excessive primary-to-secondary tube sheet differential pressures during the primary system hydro at Robinson 2. The differential pressure across the tube sheet at Turkey Point 3 during its cold hydro was what could be expected from high head safety injection during main steam line break or stuck-open safety or atmospheric dump valve events, but this also caused cladding separation. Tube, tube sheet, and cladding stresses due to differential primary-to-secondary pressure and vibrations have not been modeled in an integrated risk assessment of a main steam line break.

Analysis and Understanding: The Ad Hoc DPO Subcommittee recommended[1800] that:

> "*Risk analyses that the staff considers need to account for progression of damage to steam generator tubes in a more rigorous way.*" They "... *found that the staff did not have a technically defensible understanding of these processes to assess adequately the potential for progression of damage to steam generator tubes. Bending and flexion of the tubes produce conditions regarding crack growth, tube*

leakage, and tube burst outside the range of analyses and experiments done by the staff." (p. 46) They concluded that the contention, "*Depressurization of the reactor coolant system during a main steam line break will produce shock waves and violent, sympathetic vibrations that will cause cracks to form, to grow and to unplug, leading to much higher leakage from the primary-to-secondary sides of the reactor coolant system than has been considered by the NRC staff... has merit and deserves investigation.*" (p. 10) The Subcommittee concluded that "*... there is an imperative for the staff to act expeditiously to develop a much better understanding of the dynamic processes associated with depressurization and how the processes could lead to damage progression.*" (p. 46) "*Similarly, the Ad Hoc Subcommittee did not feel that the staff had developed an adequate understanding of how movement of the tube support plates during an event could damage the tubes and augment leakage from the primary side to the secondary side of the reactor coolant system. The staff needs to develop an understanding of how tube support plate movement could lead to unplugging of cracks occluded by corrosion products in the annular space between the tube support plate and the tubes.*" (p. 46) Also, "*... the Ad Hoc Subcommittee has concluded that the staff has not adopted a technically defensible position on the choice of the iodine spiking factor to be used in the analysis of design basis accidents for compliance with the requirements of 10 CFR Part 100 or General Design Criterion (GDC) 19.*" (p. 48)

Operator Actions: The NRC has used estimates as low as 10^{-3} as the probability of the failure to depressurize and cool down the RCS in risk analyses of these containment bypass scenarios. The human error contribution to the estimated increment to core damage frequencies per year in these scenarios ranged from 29% to 93%. Operators have to identify the ruptured SG in order to isolate it, while primary and secondary temperature and pressure changes mask the diagnostic evidence they need to do so. There have been 10 SGTRs (or significant leaks) in U.S. PWRs from 1975 to 2000. Human performance weaknesses, such as mis-diagnoses, substantial delays in isolating the faulted steam generator, and delayed initiation of the residual heat removal system, have been identified in these events.[1801,1802] The events also involved unnecessary radiation releases, lack of RCS subcooled margin, excessive RCS cooldown rates, and overfilling the SG because of human or procedural problems.

The probability value can be significantly higher than 10^{-3} when performance shaping factors are incorporated for SGTRs concurrent with containment bypass based on operator performance as well as simulator experience. While one risk analysis that addressed a stuck open relief valve has a success path involving gagging the valve, this may be unrealistic given potential galling of the internals, steam release at the valve location, and the high radiation field at the valve created by a large tube leak. Additional complications would add to operator burdens. These include high noise levels preventing normal communications; RCS cooldown with potential recriticality; actions to recover RWST inventory; many radiation alarms, unexpected high radiation areas in the turbine building, and atmospheric releases; fire alarms and fires from steam and shrapnel from the break; and emergency communications with local, state, and Federal governments diverting operations personnel before the technical support center is manned or additional operations personnel arrive on site. The Halden Control Room Staffing study found poor operator performance in one of two simulations of a SG leak with a failed open SG safety relief valve, as well as simulations where crew size was decreased to attend to other duties.[1803] A model exists based on this simulation, but it has not been used in a sensitivity study to more accurately predict a probability of failure to depressurize and cool down the RCS under these circumstances.

The Ad Hoc DPO Subcommittee concluded[1800] that:

> "... *the [human performance] failure probabilities can rise from 10^{-3} to ~1, depending on the number of failed steam generator tubes.*" They also said that "*Risk evaluations should also include examination of the mechanisms for damage progression, which has not been observed in steam generator tube rupture accidents to date, but may occur as a result of dynamic processes during main steamline break depressurizations of the reactor coolant system. The effects of the dynamic events on operator performance both with respect to the time available for required responses and the level of operator distraction need to be evaluated.*" (p. 20) "*In all cases, the staff needs to develop defensible analyses of the uncertainties in its risk assessments, including uncertainties in its assessments of human error probabilities. As the staff develops a better understanding of the dynamic processes associated with depressurization during a main steamline break, it may want to revisit estimates of operator error probability in light of the considerable operator distraction that might occur during such events.*" (p. 47)

CONCLUSION

The staff found that the accident scenarios were credible, and that the issue could not be addressed by the enforcement of existing regulations. Therefore, it was concluded that a technical assessment should be performed on the issue, in accordance with NRC Management Directive 6.4.

REFERENCES

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1800. Letter to W. Travers from D. Powers, "Differing Professional Opinion on Steam Generator Tube Integrity," February 1, 2001.

1801 Letter to W. F. Conway, (Arizona Public Service Company) from J. B. Martin (NRC), "NRC Inspection Report 50-529/93-14," April 16, 1993.

1802. Letter to A. A. Blind, (Consolidated Edison Company of New York, Inc.) from H. J. Miller (NRC), "NRC Augmented Inspection Team - Steam Generator Tube Failure - Report No. 05000247/2000-002," April 28, 2000.

1803. NUREG/IA-0137, "A Study of Control Room Staffing Levels for Advanced Reactors," U.S. Nuclear Regulatory Commission, November 2000.

1804. NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," August 3, 1995.

1805. Memorandum to A. Thadani from N. Chokshi, "Initial Screening of Candidate Generic Issue 188, 'Steam Generator Tube Leaks or Ruptures, Concurrent with Containment Bypass from Main Steam Line or Feedwater Line Breaches," May21, 2001.

ISSUE 189: SUSCEPTIBILITY OF ICE CONDENSER AND MARK III CONTAINMENTS TO EARLY FAILURE FROM HYDROGEN COMBUSTION DURING A SEVERE ACCIDENT

DESCRIPTION

Historical Background

This generic issue was proposed[1791] in response to SECY-00-198[1792] which explored means of making 10 CFR 50.44 risk-informed. As a part of this effort, the paper recommended that safety enhancements that have the potential to pass the backfit test be assessed for mandatory application through the generic issue program.

Safety Significance

Since the last revision of 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors," in 1987, there have been significant advances in the understanding of the risk associated with the production and combustion of hydrogen (and other combustible gases) during reactor accidents. The work discussed in SECY-00-198[1792] was actually an investigation of relaxation of a number of requirements.

For the majority of PWRs with large dry or sub-atmospheric containments, direct containment heating (DCH) is the dominant mode of containment failure (a separate issue that was resolved by plant-specific comparison of DCH loads versus containment strengths), and the containment loads associated with hydrogen combustion are non-threatening.

However, it was discovered in the study associated with NUREG/CR-6427[1793] that, for ice condenser containments, the early containment failure probability is dominated by non-DCH hydrogen combustion events. This is not a surprising result, given the relatively low containment free volume and low containment strength in these designs. These containments rely on the pressure-suppression capability of their ice beds, and, for a design-basis accident, where the pressure is a result of the release of steam from blowdown of the primary (or secondary) system, an ability to withstand high internal pressures is not needed.

In a beyond-design-basis accident, where the core is severely damaged, significant quantities of hydrogen gas can be released. This hydrogen is generated by the exothermic chemical reaction of water and steam with metal (especially the Zircaloy cladding), and (to some extent) by radiolysis of water, where gamma rays actually split water molecules into hydrogen and oxygen.

To deal with large quantities of hydrogen, these containments are equipped with AC-powered igniters, which are intended to control hydrogen concentrations in the containment atmosphere by initiating limited "burns" before a large quantity accumulates. In essence, the igniters prevent the hydrogen (or any other combustible gas) from accumulating in large quantities and then suddenly burning (or detonating) all at once, which would pose a threat to containment integrity.

For most accident sequences, the hydrogen igniters can deal with the potential threat from combustible gas buildup. The situation of interest for this generic issue only occurs during accident sequences associated with station blackouts, where the igniter systems are not available because they are AC-powered.

Thus, this does not affect the frequency of severe accidents, but does affect the likelihood of a significant release of radioactive material to the environment should such an accident occur.

The issue also applies to BWR MARK III containments, because they also have a relatively low free volume and low strength, comparable to those of the PWR ice condenser designs. The MARK I and MARK II designs are also pressure-suppression designs, but are operated with the containment "inerted," i.e., the drywell and the air space above the suppression pool are flooded with nitrogen gas, and a nitrogen makeup system maintains oxygen level below a set limit by maintaining a slight positive nitrogen pressure within the primary containment. The low oxygen concentration is sufficient to accommodate the hydrogen threat (except possibly for long-term radiolysis). In contrast, the MARK III designs are equipped with hydrogen igniters just as are the PWR ice condenser designs, and are similarly potentially vulnerable in an accident sequence associated with station blackout.

Possible Solution

The solution is to provide an independent power supply for the igniter systems for the subject containments. The igniters are, essentially, diesel engine glow plugs. If necessary, they could be powered by storage batteries or by a portable generator.

PRIORITY DETERMINATION

The two containment types, ice condenser and MARK III, will be examined separately. In each case, the objective is to calculate plausible estimates of risk parameters that represent the particular class of plants in question. These estimates are for prioritization purposes only, and are not intended to represent the best the state of the art can produce.

In addition to the generic estimate calculated here, an independent calculation has been performed[1794] by Energy Research, Inc. (ERI). The ERI study arose out of an investigation of possible risk-informed alternative approaches to 10 CFR 50.44, the same project that generated this generic issue. The ERI study is based on the IPE and IPEEE studies for Catawba and Grand Gulf. Although the ERI study is more plant-specific, it also avoids some of the more debatable assumptions that were necessary in the generic analysis presented here.

PWR Ice Condenser

We will examine the ice condenser plants first. The strategy will be to start with the NUREG-1150[1081] Sequoyah Level II PRA, which should be reasonably representative and also has the advantage of being readily available, and modifying it in two ways. First, use plant damage state frequencies that are more generically representative, and second, change the probability of containment failure caused by hydrogen combustion to a value consistent with more modern investigations.

Frequency Estimate

The severe accident frequency of interest is the frequency of severe accidents associated with station blackout. Fortunately, this frequency is routinely calculated in PRAs, including the

NUREG-1150[1081] PRA and NUREG/CR-4551[1795] for the Sequoyah plant (the only NUREG-1150 PRA for a PWR with an ice condenser containment). However, internal-events PRAs such as the NUREG-1150[1081] Sequoyah study do not give the complete picture. Although these studies include station blackouts initiated by both plant-centered and grid-initiated losses of offsite power, external events are not included. In most external event studies, the principal accident sequence leading to severe core damage comes from a station blackout. In seismically-initiated sequences, the seismic event damages the ceramic insulators in the transmission lines, effectively disconnecting the plant from offsite power, and also increases the likelihood of a failure of onsite power. Similarly, the fire-initiated sequences may involve a fire in the electrical switchgear, again causing a total loss of AC power.

The following table summarizes estimates of this parameter from several sources:

Site	NUREG-1150 Slow SBO	NUREG-1150 Fast SBO	IPE CDF	IPE SBO CDF	IPEEE Fire CDF	IPEEE Seismic CDF	IPEEE External CDF	Total IPE/IPEEE CDF
Sequoyah	4.58E-6	9.26E-6	1.70E-4	5.32E-6	1.6E-5	[Margin]	[1.6E-5]	[1.86E-4]
Watts Bar			8.00E-5	1.73E-5	7.0E-6	[Margin]	[7.0E-6]	[8.70E-5]
Catawba			5.80E-5	6.00E-7	4.7E-6	1.6E-5	2.1E-5	6.01E-5
McGuire			4.00E-5	9.32E-6	2.3E-7	1.1E-5	1.1E-5	5.1E-5
DC Cook			6.26E-5	1.13E-6	3.8E-6	3.2E-6	7.0E-6	7.0E-5
"Average"				6.73E-6	6.34E-6	1.01E-6		
	From CRIC-ET database[1796]		From IPE database		From NUREG/CR-6427[1793] (Table 7.5)			

(The significant figures presented in this table are given for the convenience of the reader who wishes to duplicate the calculations, and are not intended to imply that these estimates are known to two or three significant figure accuracy.)

As can be seen from the IPE SBO column, the internal-events SBO-initiated CDF ranges over the decade from 10^{-6} to 10^{-5}. The fire- and seismically-initiated CDFs, which generally involve loss of all AC power, are in the same range. The row labeled "average" is a simple arithmetic mean average over the five sites, and is intended to provide a point estimate representative of this class of plants, recognizing that individual plants vary.

Of course, the fire and seismic initiator CDFs do not consist exclusively of sequences involving loss of all AC power, and the specifics of this breakdown will be plant-specific. To get a generically-representative number, it will be necessary to make some assumptions, recognizing that the result will be, at best, a rough estimate. The NUREG-1150[1081] PRA for Sequoyah did not address external events. Thus, we will base these assumptions on the fire and seismic analyses of the NUREG-1150 Surry PRA (NUREG/CR-4551,[1795] Vol. 3, Rev. 1, Parts 1 and 3),[1795] which have the advantage of readily-available and abundant documentation. (Surry is not an ice condenser plant, but containment design should not greatly affect the frequency and course of fire and seismically initiated sequences.) This "hybridization" or use of one PRAs results in another PRA, results in, at best, a very rough approximation. However, it will be shown later that the conclusion is not greatly affected by this approximation.

In the Surry fire analysis, the principal fire-initiated plant damage states were associated with four locations:

PDS for Surry Fire Initiators (NUREG/CR-4551,[1795] Table 2.2-4, pp. 2 to 14)	
Emergency Switchgear Room	54.3%
Auxiliary Building	20.0%
Cable Vault and Tunnel	13.0%
Control Room	12.7%

Fires in the emergency switchgear room, control room or auxiliary building are not likely to disable the igniters. Even if such a fire disabled emergency power, normal power would be available. However, it will be assumed that fires in the cable vault and tunnel will also disable the igniters, and thus 13% of the fire frequency will be added to the internal SBO frequency.

The Surry seismic analysis can be used in a more straightforward manner, since the four seismic groups explicitly list station blackout.

Plant Damage States for Seismic Initiators (NUREG/CR-4551,[1795] Table 2.2-6, pp. 2.16 to 2.17)			
Group	**Description**	**LLNL-based fraction of seismic CDF**	**EPRI-based fraction of seismic CDF**
EQ 1	Loss of Station Power (no SBO)	47.1%	53.7%
EQ 2	SBO	41.1%	33.7%
EQ 3	LOCAs	11.9%	12.5%

Here, we will use the EPRI-based estimate of 33.7%, as being more in line with modern analyses.

Large Early Release Frequency (LERF) Estimate

According to the studies presented in NUREG/CR-6427,[1793] the likelihood of early containment failure due to uncontrolled post-accident hydrogen combustion is significantly higher than the figure used in the NUREG-1150[1081] PRA for Sequoyah. Table 7.3 of NUREG/CR-6427[1793] gives a non-DCH failure probability for both fast and slow station blackout sequences of 0.9021, which is essentially all due to hydrogen combustion. The non-DCH failure probability is given as zero for all other core damage initiators, presumably due to the availability of AC power for the igniters. Therefore, it can be assumed that providing an alternative power supply for the igniters would lower the total containment failure probability by about 0.9. With this, it is possible to estimate the change in large early release frequency (ΔLERF) associated with the issue:

	CDF	SBO Fraction	SBO CDF	Change in Containment Failure Probability	ΔLERF
Internal			6.73E-6	0.90	6.06E-6
Fire	6.34E-6	13%	8.24E-7	0.90	7.42E-7
Seismic	1.01E-6	33.7%	3.40E-7	0.90	3.06E-6

Again, the significant figures are given for convenience in following these calculations, and are not intended to imply a high accuracy in the estimates.

The screening threshold for LERF given in Management Directive 6.4 (Appendix C, Figure C4) is any change in LERF greater than 10^{-6}/RY, regardless of the initial LERF. Thus, for ice condenser plants, this issue passes this screening criterion. It should be noted that the criterion is met even without the external events.

Recoverability: The analysis above does not distinguish between recoverable and non-recoverable station blackout. This leads to some conservatism in the result, since the existing igniter system will become available if AC power is recovered after core melt, but before hydrogen ignition. It should be noted, however, that the efficacy of the igniters in preventing large scale burns depends on their availability early, before combustible gases have time to accumulate in large quantities. Once this accumulation occurs, turning on the igniters may be counterproductive.

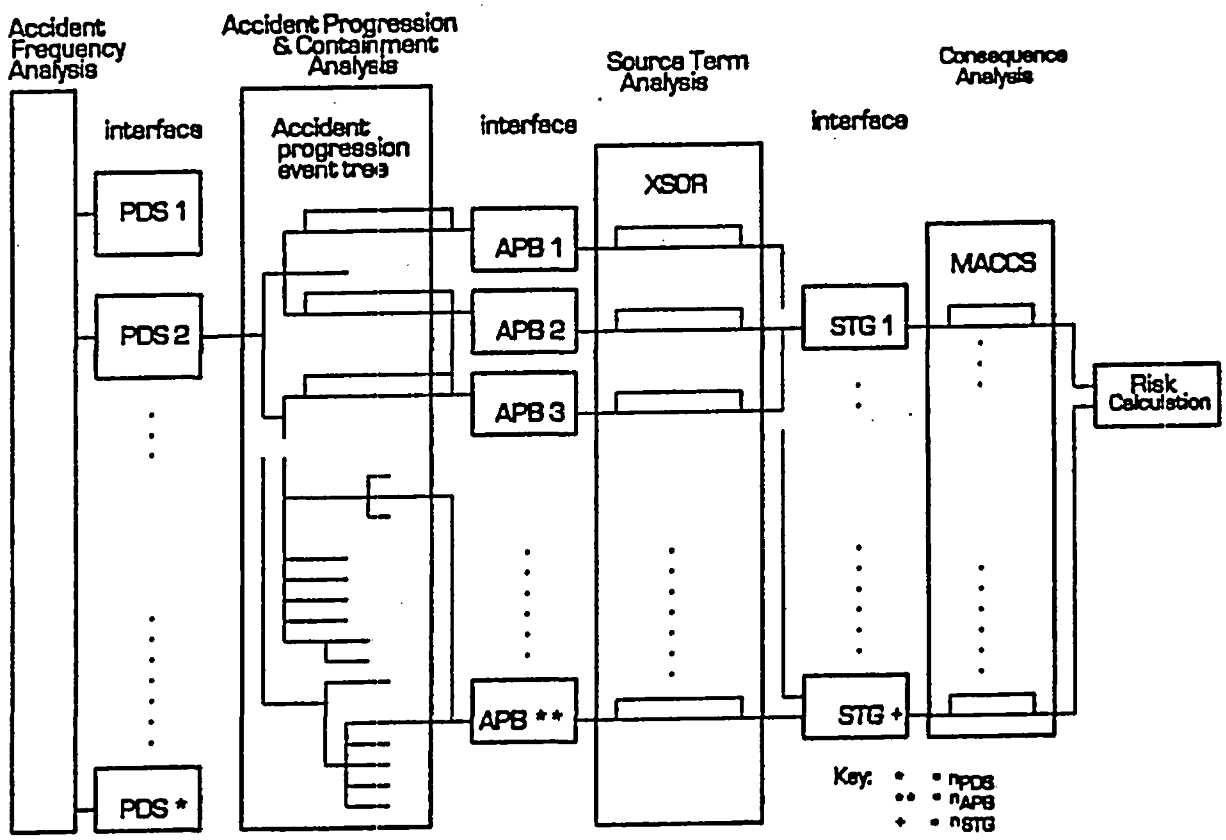
Hybridization: The various core damage frequencies and associated changes in LERF are based on a hybridization of several PRAs. Moreover, the estimates of the station blackout portion of the seismic and fire CDFs are, at best, educated guesses. Nevertheless, if the change in containment failure probability is 90%, most of the IPE SBO core damage frequencies are high enough for the ΔLERF to pass the screening criterion even without the hybridization or addition of external events. The conclusion that this issue passes the screening criterion is reasonably robust.

Consequence Estimate

Estimating the risk to the population from these accident sequences is not as straightforward as estimating LERF. In the integrated risk analysis for the NUREG-1150[1081] PRAs, the accident frequency analysis ("front end" analysis) produces an overall CDF, and also a set of plant damage states, each with its own frequency. For the Sequoyah PRA, the plant damage states are:

PDS Index	Plant Damage State (PDS)
1	Slow Station Blackout
2	Fast Station Blackout
3	LOCA
4	Event V (interfacing systems LOCA)
5	Transient
6	ATWS
7	Steam Generator Tube Rupture

The sequences of interest here are in plant damage states 1 and 2. However, these plant damage states do not correlate one-to-one with a consequence analysis. A description of the integrated risk analysis can be found in Reference 6, from which the following figure is taken:



In the integrated risk analysis, the accident progression event tree analysis (a very extensive set of calculations) is used to calculate a set of accident progression bin frequencies from each PDS. The set of accident progression bins is then input into a partitioning analysis (also very extensive) to calculate source term groups and associated frequencies. Actual consequences (e.g., man-rem) are then calculated for each source term, and the total risk is calculated by multiplying each consequence by its source term frequency, and summing the products.

It is not practical to calculate the risk associated with this issue with a hand calculation. Instead, a sensitivity analysis computer code, the Computational Risk Integration and Conditional Evaluation Tool (CRIC-ET), was used.[1796]

In order to use this code, it was necessary to "split" the generic station blackout frequency estimated above into "slow SBO" and "fast SBO." The IPE and IPEEE averages do not make this distinction, and thus some approximations must again be made. The three components, internal, seismic, and fire, were handled separately:

Internal - The internal SBO frequency contribution, based on the IPE average, was subdivided into slow and fast based on the proportions in the Sequoyah NUREG-1150[1081] PRA:

	NUREG-1150[1081] SBO CDF	Fractional Contribution	IPE-based SBO CDF	Proportioned SBO CDF
Slow	4.58E-6	33.1%	6.73E-6	2.23E-6
Fast	9.26E-6	66.9%		4.50E-6
Total	1.38E-5	100%	6.73E-6	6.73E-6

Seismic - The seismic SBO contribution (33.7% of the total seismically-initiated CDF, as discussed under LERF above) was assumed to be entirely in the slow category. (Generally, the seismic event causes the station blackout and destroys the condensate storage tank, and eventually the steam generators dry out.)

Fire - The fire SBO contribution (13.0%of the fire-initiated CDF) was assumed to be entirely in the fast category. (Fires in the cable vault are likely to fail everything at once.)

Several other assumptions were necessary:

The other PDS frequencies were set to zero so that the analysis would only include the SBO plant damage states.

The sequences ending in no containment failure were re-directed to the early containment failure accident progression bin, to account for the high susceptibility of the containment to failure due to hydrogen combustion, as estimated in NUREG/CR-6427.[1793] This is a slight overestimate, in that the containment failure probability due to hydrogen combustion is 90% rather than 100%, but the CRIC-ET code does not have this flexibility.

A corrected consequence file for Sequoyah was used to correct a known error.[1797] The results of the calculation of population dose within 50 miles of a reactor, using 200 samples and the usual limited Latin Hypercube technique, were:

5th percentile -	3.86×10^{-3} man-rem
95th percentile -	20.3 man-rem
Median -	2.24 man-rem
Mean -	6.43 man-rem

Again, as is obvious from the distribution, the two decimal places are not significant and are given only for purposes of reproducing the calculation. The error bounds reflect only the uncertainty associated with the Level II analysis, and do not include the uncertainty associated with the generic station blackout frequency or split of this frequency into the fast and slow SBO plant damage states.

Generic Population Distribution: The man-rem/RY figure above is based on the NUREG-1150[1081] model which is specific to the Sequoyah site. For generic issue calculations, such figures are generally based on a uniform population density of 340 persons/square-mile and a typical central Midwest plains meteorology. It is not practical to re-run the consequence analysis for the generic site but, as a first approximation, the risk figures can be re-normalized to the generic population.

Interpolating between the 30- and 100-mile radius population figures given in NUREG/CR-4551[1795] (Volume 5, Rev. 1, Part 1, Page 4.2), the Sequoyah population density for a 50- mile radius is approximately 159 persons/square-mile. Thus, to get a generic risk figure, the 6.43 man-rem/RY (mean) figure should be multiplied by 340/159. This gives a generic estimate of 13.73 man-rem/RY.

Aggregated Risk Figure: There are nine reactors with an ice condenser containment. Thus, the aggregated risk figure is 13.73 man-rem/RY times 9 reactors or 124 man-rem/year.

The screening threshold for averted offsite risk given in Management Directive 6.4 (Appendix C, Figure C6) is an averted offsite man-rem/year greater than 100, if the cost/benefit ratio is favorable (i.e., less than $2,000/man-rem).

Cost Estimate

A separate cost investigation will not be performed here. The ERI study[1794] concluded that the *proposed fix is cost-beneficial. Therefore, it will be assumed here that the cost/benefit ration is less* than $2,000/man-rem, and the issue passes the screening threshold for risk.

Other Considerations

Hybrid Models: The split of the generic station blackout frequency into the fast and slow station blackout plant damage states, as described above, is questionable at best, since it is based on a hybridization of several PRAs. Because of this, a sensitivity analysis was done to investigate how big an effect this was. First, the entire station blackout frequency was assigned to the slow SBO PDS and a mean man-rem/RY was calculated. Then, the entire frequency was assigned to the fast SBO PDS, and the calculation repeated. The results were:

Split	Mean Risk (man-rem/RY)
All in the slow SBO PDS	5.38
All in the fast SBO PDS	6.94
"Best guess" proportioned	6.43

Based on these results, it seems safe to conclude that the results are not very sensitive to how the frequency is split between the two plant damage states.

Recoverable Station Blackout: The Sequoyah analysis, as modeled in CRIC-ET, does not distinguish between recoverable and non-recoverable station blackout. As was the case in the estimate of LERF, this leads to some conservatism in the result, since the existing igniter system will become available if AC power is recovered after core melt, but before hydrogen ignition. Once again, however, the efficacy of the igniters in preventing large scale burns depends on their availability early, before combustible gases have time to accumulate in large quantities. Once this accumulation occurs, a late initiation of the igniter systems may not have the desired result.

ERI study: The ERI study[1794] estimated a risk of 3 man-rem/RY using the Catawba site and a more sophisticated methodology, which is about a factor of two less than the estimate presented here. In the context of PRA studies, a factor of two is very good agreement.

BWR MARK III Containments

The strategy for MARK III BWR containments is similar to that for ice condensers. The NUREG-1150[1081] Level II model for the Grand Gulf plant will be used, but will be modified to be more generic and to include a higher probability for containment failure due to hydrogen combustion.

The NUREG-1150[1081] Level II model for Grand Gulf is described in detail in NUREG/CR-4551[1795] (Vol. 6, Rev. 1, Part 1). The general approach, using plant damage states, accident progression bins, and source term groups, is similar to that discussed above for the Sequoyah model. However, the individual plant damage states are defined differently.

The Grand Gulf model consists of twelve plant damage states. PDS 1 through 8 are associated with station blackout, PDS 9 and 10 are associated with ATWS, and PDS 11 and 12 are associated with non-ATWS transient-initiated sequences. Although the total CDF (as estimated in NUREG-1150[1081]) is rather low (about 4 x 10^{-6}/RY), about 97% of this CDF comes from the station blackout sequences NUREG/CR-4551[1795] (Vol. 6, Rev. 1, Part 1, Table 2.2-3).

Of the eight station blackout plant damage states, the first six are recoverable station blackouts, in which severe core damage occurs, but AC power is recovered in time for the "miscellaneous systems" - containment venting, standby gas treatment, containment isolation, and the hydrogen igniters - to be effective. (This explicit modeling avoids the problems with treating recoverable station blackouts in the ice condensor plants, discussed earlier.) Adding backup power to the hydrogen igniters will not affect the sequences in these plant damage states.

Thus, the plant damage states of interest are PDS 7, non-recoverable fast SBO, and PDS 8, non-recoverable slow SBO. These two plant damage states represent 11% and 2% of the total station blackout frequency, respectively (NUREG/CR-4551,[1795] Vol. 6, Rev. 1, Part 1).

Frequency Estimate

The NUREG-1150[1081] estimate of CDF for Grand Gulf is 4 x 10^{-6}/RY, which is somewhat lower than the Grand Gulf IPE estimate of 1.72 x 10^{-5}/RY. Again, it is necessary to find a more generic number. For the IPEs' CDFs and, specifically, the IPE SBO CDFs, these figures are tabulated in the IPE Database.

As in the analysis of ice condenser plants, the fire-induced accident sequences are also significant. These are available from the IPEEE program, in NUREG-1742[1798] (Volume 2, Table 3.2).

Seismically-induced sequences are also a concern. However, there are no PRAs available for any plant with a MARK III containment. All four MARK III plants were analyzed with a seismic margins approach in the IPEEE program. Thus, once again it will be necessary to use a bit of improvisation.

The Grand Gulf and River Bend sites are in areas of low seismicity, and thus it is not anticipated that seismic sequences would be a significant contributor. The Clinton and Perry plants are located in areas of moderate seismicity, and thus may be of more concern. Given that there are no appropriate PRAs, the only recourse is to find a similar plant. The LaSalle plant is a reasonable choice, although it is a BWR/5 model with a Mark II containment, because the reactor systems (not containment systems) are similar, and the site is in the same general area (Great Lakes). The LaSalle seismic CDF, based on an existing simplified seismic PRA, is 7.6 x 10^{-7}/RY, as reported in NUREG-1742[1798] (Volume 2, Table 2.1). Although the use of this number is highly questionable at best, the seismic contribution is expected to be relatively minor compared to the other contributors, and thus more uncertainty can be tolerated. The CDF figures are as follows:

Site	NUREG-1150 Non-recoverable Fast SBO CDF	NUREG-1150 Non-recoverable Slow SBO CDF	IPE CDF	IPE SBO CDF	IPEEE Fire CDF	IPEEE Seismic CDF
Clinton			2.66E-5	9.80E-6	3.64E-6	SMA
Grand Gulf	4.3E-7 (11%)	6.6E-8 (2%)	1.72E-5	7.46E-6	8.89E-6	SMA
Perry			1.30E-5	2.25E-6	3.27E-5	SMA
River Bend			1.55E-5	1.35E-5	2.25E-5	SMA
LaSalle						7.6E-7
"Average"				8.25E-6	1.69E-5	7.6E-7
	From CRIC-ET database[1796]		From IPE database		From NUREG/CR-1742[1796] (Vol. 2, Table 3.2)	

Large Early Release Frequency (LERF) Estimate

To get non-recoverable station blackout frequencies, it will be assumed that the same percentage of the total station blackout frequency is non-recoverable as was the case in the NUREG-1150[1081] model, which is 13% (11% fast SBO plus 2% slow SBO). The generic estimate for the total non-recoverable SBO CDF is then:

$$[(8.25 \times 10^{-6} + 1.69 \times 10^{-5} + 7.6 \times 10^{-7}) \times 13\%] \text{ event/RY} = 3.37 \times 10^{-6} \text{ event/RY}$$

The response of the MARK III containments to an uncontrolled hydrogen containment is expected to be similar to that of an ice condenser containment. Thus, the change in large early release frequency (ΔLERF) will be approximately 90% of the CDF associated with unrecoverable station blackout:

$$\Delta\text{LERF} = 3.37 \times 10^{-6} \times 90\% = 3 \times 10^{-6} \text{ event/RY}$$

This is above the screening threshold given in Management Directive 6.4 (Appendix C, Figure C4), regardless of the initial LERF.

Other Considerations

As was the case with ice condenser containments, this generic estimate, the various CDFs and associated changes in LERF are based on a hybridization of several PRAs. Moreover, the estimates of the station blackout portion of the seismic and fire CDFs are, at best, educated guesses, and the fire contribution is the largest contributor. However, if the fire and seismic portions were not included, the ΔLERF would still be about 9.7×10^{-7} event/RY, very close to the cutoff of 10^{-6} event/RY.

If it is postulated that hydrogen combustion without igniters will result in containment failure 90% of the time, the robustness of the conclusion depends primarily on the SBO CDFs taken from the IPE submittals for the four plants, the assumption that about 13% will be non-recoverable blackouts, and an assumption that there will be at least a small contribution from external events. Even though there are many approximations in the estimates calculated above, these points seem reasonable.

Consequence Estimate

The MARK III containment has two air spaces, the drywell free volume and the wetwell airspace above the suppression pool. Combustible gases generated in the vessel prior to vessel breach may be vented by the safety/relief valves and tailpipes through the suppression pool to the wetwell airspace. After vessel breach, combustible gases may accumulate in the drywell airspace, and may be forced through the weir wall to the wetwell airspace. Combustion may occur in either airspace. Both airspaces are equipped with igniters.

In the NUREG-1150[1081] Grand Gulf analysis, the automatic depressurization system is not operable in a station blackout, and the vessel remains at high pressure. Moreover, depressurization of the vessel would have allowed the operators to use the firewater system to inject coolant. Thus, in the sequences of interest here, the vessel is likely to remain at high pressure until failure occurs at the bottom head.

The drywell is generally stronger than the wetwell. In most, but not all, cases, overpressurization will fail the containment in the wetwell airspace, which will cause radioactive releases to pass through (and be scrubbed by) the suppression pool. The accident progression event trees and source term analyses must account for all of this. A complete description can be found in NUREG/CR-4551[1795] (Volume 6, Rev. 1, Part 1).

To use the Grand Gulf model in the CRIC-ET code, the following assumptions were made:

> 11% of the generic internal SBO CDF frequency will be placed into PDS7 (non-recoverable fast blackout), and 2% will be placed into PDS8 (non-recoverable slow blackout), the proportions used in the Grand Gulf model.
>
> The same 11%/2% split applies to the fire CDF frequency. Most dominant fire scenarios result in a plant transient, generally involving loss of electrical buses due to the fire (See NUREG/CR-4551,[1795] Volume 4, Rev. 1, Part 1, §3.3.2.3). There is no easy way to estimate the fraction of these which involve non-recoverable station blackouts, so the fractions used in the internal events analysis will be used.
>
> All of the seismic sequences are slow, non-recoverable blackouts.

As in the calculation for the ice condenser containments, several other assumptions were necessary:

> The other PDS frequencies were set to zero, so that the analysis would only include the non-recoverable station blackout plant damage states.
>
> The sequences ending in no containment failure ("characteristic 6" in the Grand Gulf model - see NUREG/CR-4551[1795] (Volume 6, Rev. 1, Part 1, Table 2.4-1) were re-directed to the "rupture before vessel breach" accident progression bin, to account for the assumed high susceptibility of the containment to fail due to hydrogen combustion. This is a slight overestimate, since the model presumed that the igniters were not available in PDS 7 and

8 in any case.

The results of the calculation of population dose within 50 miles per reactor, using 250 samples and the usual limited Latin Hypercube technique, were:

5th percentile	1.23×10^{-2} man-rem
95th percentile	1.35 man-rem
Median	0.136 man-rem
Mean	0.363 man-rem

Again, as is obvious from the distribution, the two decimal places are not significant and are given only for purposes of reproducing the calculation. The error bounds reflect only the uncertainty associated with the Level II analysis, and do not include the uncertainty associated with the generic station blackout frequency or split of this frequency into the non-recoverable fast and slow SBO plant damage states.

Generic Population Distribution: The man-rem/RY figure is based on the NUREG-1150[1081] model which is specific to the Grand Gulf site. For generic issue calculations, such figures are generally based on a uniform population density of 340 persons/square-mile and a typical central Midwest plains meteorology. It is not currently practical to re-run the consequence analysis for the generic site, but as a first approximation, the risk figures can be re-normalized to the generic population. Interpolating between the 30- and 100-mile radius population figures given in NUREG/CR-4551[1795] (Volume 6, Rev. 1, Part 1, Page 4.3) the Grand Gulf population density for a 50-mile radius is approximately 39.3 persons/square-mile, much less than the generic figure. Thus, to get a generic risk figure, the 0.363 man-rem/RY figure should be multiplied by 340/39.3, which gives a generic estimate of 3.14 man-rem/RY.

Aggregated Risk Figure: There are only four reactors with a MARK III containment. Thus, the aggregated risk figure is 3.14 man-rem/RY times 4 reactors or 12.6 man-rem/RY.

Screening Threshold: The screening threshold for averted offsite risk given in Management Directive 6.4 (Appendix C, Figure C6) is an averted offsite man-rem/year greater than 100, if the cost/benefit ratio is less than $2,000/man-rem. Thus, this criterion is not met for MARK III plants, regardless of cost.

Other Considerations

Hybrid Models: The split of the generic station blackout frequency into the fast and slow station blackout plant damage states, as described above, is questionable at best, since it is based on a hybridization of several PRAs. Because of this, a sensitivity analysis was done to investigate how big an effect this was. First, the entire station blackout frequency was assigned to the slow SBO PDS and a mean man-rem/RY was calculated. Then, the entire frequency was assigned to the fast SBO PDS, and the calculation repeated. The results were:

Split	Mean Risk (man-rem/RY)
All in the slow SBO PDS	0.386
All in the fast SBO PDS	0.341
"Best guess" proportioned	0.363

Based on these results, it seems safe to conclude that the results are not very sensitive to how the frequency is split between the two plant damage states.

Re-Direction of Sequences Ending in No Containment Failure: A sensitivity analysis was performed to test the re-direction of the sequences that did not result in containment failure in the original model into failure before vessel breach. As was stated previously, the original model should have already accounted for the unavailability of the hydrogen igniters, so this was expected to be a minor effect. The sensitivity analysis calculated a population risk of 0.360 man-rem instead of 0.363 man-rem, which confirms the expectation.

ERI Study: The ERI study[1794] estimated a risk of 1.3 man-rem/RY for Grand Gulf. This is roughly a factor of four larger than the estimate calculated here. In the context of PRA calculations, this is reasonable agreement. It should be noted that quadrupling the generic risk estimates would not change the conclusion.

CONCLUSION

Based on the change in large early containment failure frequency (LERF) for both PWR ice condenser and BWR Mark III containment designs and on the change in risk (as measured by mann-rem/ year) for the ice condenser designs, this issue passes the screening criteria and should go on to the technical assessment stage.

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

This appendix contains a listing of those residual GSIs that are applicable to operating and future reactor plants and includes: issues that have been resolved with requirements [I, NOTE 3(a)]; USI, HIGH- and MEDIUM-priority issues scheduled for resolution; nearly-resolved issues scheduled for resolution (NOTES 1 and 2); and issues that are scheduled for prioritization (NOTE 4). The priority designations for all issues are consistent with those listed in Table II of the Introduction. In accordance with 10 CFR 52.47(a)(1)(iv), any future application for design certification must contain proposed technical resolutions for the issues in this listing that are designated USI, HIGH, MEDIUM, NOTE 1, and NOTE 2. (In July 1998, the priority categories NOTES 1 and 2 were eliminated and all GSIs in these categories were given a HIGH priority ranking.[17,18]) Also included in this listing are those GSIs that were either prioritized or resolved with no impact on operating reactor plants but contain recommendations for future reactor plants (NOTE 6).

Legend

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

Appendix B (Continued)

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

Appendix B (Continued)

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

Appendix B (Continued)

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

Appendix B (Continued)

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

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

Appendix B (Continued)

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

Appendix B (Continued)

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

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants- MPA No	Operating Plants - Effective Date	Future Plants- Effective Date
			BWR	PWR			
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level - Design and Modification	I	GE	NA	F-50	01/01/81	01/01/81
II.K.3(22)	Automatic Switchover of RCIC System Suction - Verify Procedures and Modify Design	I	GE	NA	F-51	01/01/81	01/01/81
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	I	GE	NA	F-52	01/01/82	01/01/82
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	I	GE	NA	F-53	01/01/82	01/01/82
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	I	GE	NA	F-54	10/01/80	10/01/80
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	I	GE	NA	F-55	01/01/82	01/01/82
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	I	GE	NA	F-56	04/01/81	NA
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	I	All	All	F-57	01/01/83	01/01/83
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	I	All	All	F-58	01/01/83	01/01/83
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	I	GE	NA	F-59	01/01/81	01/01/81
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	I	GE	NA	F-60	01/01/81	01/01/81
II.K.3(46)	Response to List of Concerns from ACRS Consultant	I	GE	NA	F-61	07/01/80	07/01/80
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	I	GE	NA	F-62	10/01/80	NA
III.A	EMERGENCY PREPAREDNESS AND RADIATION EFFECTS						
III.A.1	Improve Licensee Emergency Preparedness - Short Term						
III.A.1.1	Upgrade Emergency Preparedness	-	-	-	-	-	-
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	I	All	All		10/10/79	08/19/80
III.A.1.2	Upgrade Licensee Emergency Support Facilities	-	-	-	-	-	-
III.A.1.2(1)	Technical Support Center	I	All	All	F-63	09/13/79	09/27/79
III.A.1.2(2)	On-Site Operational Support Center	I	All	All	F-64	09/13/79	09/27/79
III.A.1.2(3)	Near-Site Emergency Operations Facility	I	All	All	F-65	09/13/79	09/27/79
III.A.2	Improving Licensee Emergency Preparedness-Long Term						
III.A.2.1	Amend 10 CFR 50 and 10 CFR 50, Appendix E	-	-	-	-	-	-
III.A.2.1(1)	Publish Proposed Amendments to the Rules	NOTE 3(a)	All	All			
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	I	All	All	F-67		

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Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
III.A.2.2	Development of Guidance and Criteria	I	All	All	F-68		
III.A.3	Improving NRC Emergency Preparedness						
III.A.3.3	Communications	-	-	-	-	-	-
III.A.3.3(1)	Install Direct Dedicated Telephone Lines	NOTE 3(a)	All	All			
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems	NOTE 3(a)	All	All			
III.D	RADIATION PROTECTION						
III.D.1	Radiation Source Control						
III.D.1.1	Primary Coolant Sources Outside the Containment Structure	-	-	-	-	-	-
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining to Reducing Leakage from Operating Systems	I	All	All		07/02/79	09/27/79
III.D.3	Worker Radiation Protection Improvement						
III.D.3.3	Inplant Radiation Monitoring	-	-	-	-	-	-
III.D.3.3(1)	Issue Letter Requiring Improved Radiation Sampling Instrumentation	I	All	All	F-69	09/13/79	09/27/79
III.D.3.3(2)	Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	NOTE 3(a)	All	All		09/13/79	09/27/79
III.D.3.3(3)	Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation-Monitoring Instruments	NOTE 3(a)	All	All		09/13/79	09/27/79
III.D.3.3(4)	Issue a Regulatory Guide	NOTE 3(a)	All	All		09/13/79	09/27/79
III.D.3.4	Control Room Habitability	I	All	All	F-70	05/07/80	06/26/80
	TASK ACTION PLAN ITEMS						
A-1	Water Hammer (former USI)	NOTE 3(a)	All	All		NA	03/15/84
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems (former USI)	NOTE 3(a)	NA	All	D-10	01/--/81	01/--/81
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	NOTE 3(a)	NA	W		04/17/85	04/17/85
A-4	CE Steam Generator Tube Integrity (former USI)	NOTE 3(a)	NA	CE		04/17/85	04/17/85
A-5	B&W Steam Generator Tube Integrity (former USI)	NOTE 3(a)	NA	B&W		04/17/85	04/17/85
A-6	Mark I Short-Term Program (former USI)	NOTE 3(a)	GE	NA		12/--/77	NA
A-7	Mark I Long-Term Program (former USI)	NOTE 3(a)	GE	NA	D-01	08/--/82	08/--/82
A-8	Mark II Containment Pool Dyanmic Loads - Long Term Program (former USI)	NOTE 3(a)	GE	NA		08/--/81	08/--/81
A-9	ATWS (former USI)	NOTE 3(a)	All	All		06/26/84	06/26/84

Appendix B (Continued)

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

Appendix B (Continued)

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

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
	Sites						
155	Generic Concerns Arising from TMI-2 Cleanup	-	-	-	-	-	-
155.1	More Realistic Source Term Assumptions	NOTE 3(a)	All	All	NA	NA	02/--/95
156	Systematic Evaluation Program	-	-	-	-	-	-
156.6.1	Pipe Break Effects on Systems and Components	HIGH	All	All		TBD	TBD
163	Multiple Steam Generator Tube Leakage	HIGH	NA	All		TBD	TBD
168	Environmental Qualification of Electrical Equipment	HIGH	All	All		TBD	TBD
177	Vehicle Intrusion at TMI	NOTE 3(a)	All	All		08/01/94	08/01/94
185	Control of Recriticality Following Small-Break LOCA in PWRs	HIGH	All	All		TBD	TBD
186	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants	NOTE 4	All	All		TBD	TBD
188	Steam Generator Tube Leaks/Ruptures Concurrent With Containment Bypass	CONT.	All	All		TBD	TBD
189	Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Combustion During A Severe Accident	CONT.	All	All		TBD	TBD
191	Assessment of Debris Accumulation on PWR Sump Performance	HIGH	NA	All		TBD	TBD
192	Secondary Containment Drawdown Time	NOTE 4	All	NA		TBD	TBD
193	BWR ECCS Suction Concerns	NOTE 4	All	NA		TBD	TBD
194	Implications of Updated Probabilistic Seismic Hazard Estimates	NOTE 4	All	All		TBD	TBD

HUMAN FACTORS ISSUES

HF1	STAFFING AND QUALIFICATIONS						
HF.1.1	Shift Staffing	NOTE 3(a)	All	All		01/--/84	01/--/84

APPENDIX F

NUCLEAR MATERIAL SAFETY AND SAFEGUARDS GSIs

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

TABLE F.1

LISTING OF NMSS GSIs

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

Legend

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

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

Issue No.	Title	Priority Engineer	LeadOffice/ Division/Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date
NMSS-0010	Troxler Gauge Source Rod Weld Failures	Ramsey	NMSS/IMNS	NOTE 3(b	1	06/30/2002
NMSS-0011	Spent Fuel Dry Cask Weld Cracks	Ramsey	NMSS/SFPO	NOTE 3(b)		12/31/1998
NMSS-0012	Inadequate Transportation Packaging Puncture Tests	Ramsey	NMSS/SFPO	NOTE 3(b)	1	06/30/2000
NMSS-0013	Use of Different Dose Equivalent Models to Show Compliance	Ramsey	NMSS/IMNS	NOTE 3(b)	1	06/30/2000
NMSS-0014	Surety Estimates for Groundwater Restoration at In-Situ Leach Fields	Ramsey	NMSS/DWM	MEDIUM		12/31/1998
NMSS-0015	Adequacy of 10 CFR 150 Criticality Requirements	Ramsey	NMSS/DWM	NOTE 3(b)	1	06/30/2000
NMSS-0016	Adequacy of 0.05 Weight Percent Limit in 10 CFR 40	Ramsey	NMSS/IMNS	MEDIUM		12/31/1998
NMSS-0017	Misleading Marketing Information to General Licensees	C. Mattsen	NMSS/IMNS	NOTE 3(a)		06/30/2001
NMSS-0018	Problems Encountered When Manually Editing Treatment Planning Data on Nucletron MicroSelectron-HDR Model 105.999	B. Ayres	NMSS/IMNS	NOTE 3(b)		06/30/2001
NMSS-0019	Control Unit Failures of Classic Nucletron HDR Units	B. Ayres	NMSS/IMNS	NOTE 3(b)		06/30/2001
NMSS-0020	Leaking Pools	M. Sitek	NMSS/IMNS	DROP		06/30/2001
NMSS-0021	Unlikely Events	M. Sitek	NMSS/IMNS	DROP		06/30/2001
NMSS-0022	Gamma Stereotactic Radiosurgery	M. Sitek	NMSS/IMNS	DROP		06/30/2001

NMSS-0010: TROXLER GAUGE SOURCE ROD WELD FAILURES

DESCRIPTION

This issue was identified[1709] by NMSS after it was reported in June 1997 that the source from a Troxler moisture density gauge broke off the source rod and was left at a temporary job site. Prior to this event, there had been 6 known disconnects and 57 additional devices with cracked welds since 1996, and NRC Information Notice 96-52[1713] had been issued to alert portable gauge licensees and vendors to the potential for cracks to develop in the insertion rod of Troxler Model 3400 portable moisture density gauges. If not detected early, the cracks may propagate, eventually leading to complete failure of the insertion rod and release of the contained radioactive material.

CONCLUSION

In July 1997, NMSS and representatives from the state of North Carolina met with Troxler to discuss the continuing problem of cracked and broken source rods. Between July 1997 and April 1998, the staff worked with the state of North Carolina on a Consent Order to Troxler which required Troxler to issue a customer bulletin, conduct accelerated device inspections, revise procedures, and perform additional tests. It is expected that the customer bulletin will address the problem and that Troxler will request their customers to have their gauges inspected. The issue was given a medium priority ranking[1709] and was later closed out after an NRC study showed that the gauge failure rate was low and Troxler had corrected its design.[1808]

REFERENCES

1709. Memorandum to J. Craig from F. Combs, "Submittal of New Generic Issues for Tracking in the Generic Issues Management and Control System (GIMCS)," June 4, 1998.

1713. NRC Information Notice 96-52, "Cracked Insertion Rods on Troxler Model 3400 Series Portable Moisture Density Gauges," September 26, 1996.

1808. Memorandum to T. King from D. Cool, "NMSS Input for First Quarter FY-2002 Update of the Generic Issue Management Control System," January 16, 2002.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

The report presents the safety priority rankings for generic safety issues related to nuclear power plants. The purpose of these rankings is to assist in the timely and efficient allocation of NRC resources for the resolution of those safety issues that have a significant potential for reducing risk. The safety priority rankings are HIGH, MEDIUM, LOW, 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 resolution of the safety issues were implemented, and the consideration of uncertainties and other quantitative and qualitative factors. To the extent practical, estimates are quantitative.

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