July 2010

SUPPLEMENT 33 TO NUREG-0933, "RESOLUTION OF GENERIC SAFETY ISSUES"

REVISION INSERTION INSTRUCTIONS

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U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG-0933
Supplement 33

Resolution of Generic Safety Issues

Supplement 33

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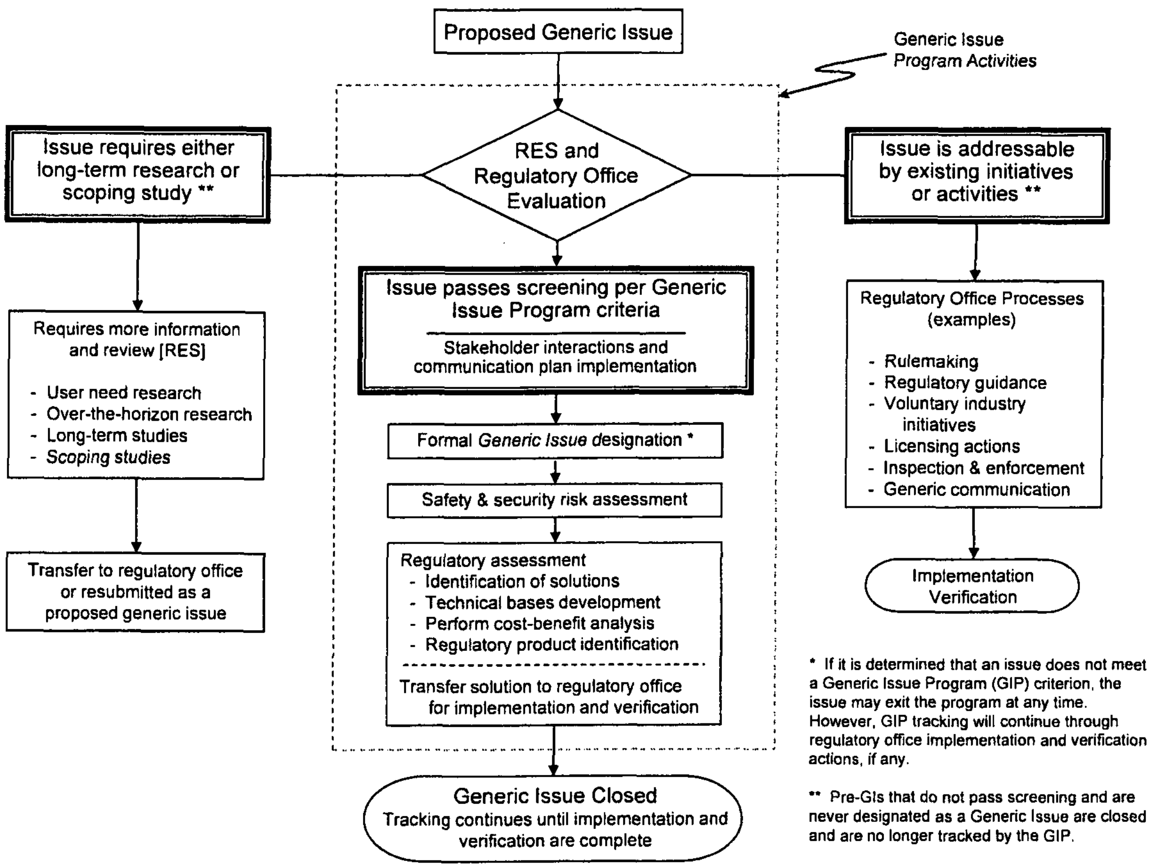
INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) has identified by its assessment of plant operations certain issues involving public health and safety, the common defense and security, or the environment that could affect multiple entities under NRC jurisdiction. Under the Generic Issues Program (GIP), resolution of these generic issues (GIs) is documented and tracked. In addition, GIP tracks and reports the GI status and resolutions to Congress and the public. The resolution of these issues may involve new or revised rules, new or revised guidance, or revised interpretation of rules or guidance that affect nuclear power plant licensees, nuclear material certificate holders, or holders of other regulatory approvals. Congress requires that the NRC maintain this program (see Section 210 of the Energy Reorganization Act of 1974 (Public Law 95-209)).

A generic issue is (1) a well-defined, discrete, technical or security issue, (2) the risk or safety significance of which can be adequately determined, and that (3) applies to two or more facilities or licensees and certificate holders or holders of other regulatory approvals (including design certification rules), (4) affects public health and safety, the common defense and security, or the environment, (5) is not already being processed under an existing program or process, (6) cannot be readily addressed through other regulatory programs and processes, existing regulations, policies, guidance, or voluntary industry initiatives, and (7) can be resolved by new or revised regulation, policy, or guidance or by voluntary industry initiatives. NRC staff or members of the public may propose a GI when issues are identified that indicate or suggest there might be weaknesses in NRC rules and regulations to ensure public health and safety and security for nuclear matters.

The agency's GIP process for resolving GIs is described in Management Directive 6.4, "Generic Issues Program," dated November 17, 2009, and SECY-07-0022, "Status Report on Proposed Improvements to the Generic Issues Program," dated January 30, 2007. These documents provide recent program improvement initiatives. This process includes five distinct possible stages: identification, acceptance review, screening, safety/risk assessment, and regulatory assessment. During each stage, the NRC staff determines whether or not the issue needs more information and if the issue should proceed to the next stage, or recommends that the issue exit the GIP. When issues exit the GIP, the possible outcomes include: no action, further research, transfer to appropriate regulatory programs, or possible industry initiative. In any case, the GIP provides feedback about the outcome at each stage to the person proposing the GI (requestor) and the appropriate regulatory office. Issues that proceed through all five stages result in regulatory solutions that are provided to regulatory offices for implementation and verification. The following figure presents the GIP in perspective with other regulatory programs and processes. Historical GIP procedures are documented in Appendix G of this report.



Progress in resolving GIs that the NRC has identified for regulation and guidance development is published quarterly in the Generic Issue Management Control System (GIMCS), which is available in the NRC's Public Document Room or from the Publicly Available Records component of the Agencywide Documents Access and Management System (ADAMS). Furthermore, this report (NUREG-0933) publishes the resolutions of all resolved generic safety issues and the partial assessments of all remaining unresolved GIs. Table II of NUREG-0933 presents a list of all GIs. In addition, Table III of this report summarizes and tabulates by group the results of the resolution of all issues contained in this report. GIs identified since the previous publication of NUREG-0933 are identified in the quarterly GIMCS reports.

TABLE II

LIST OF ALL THREE MILE ISLAND NUCLEAR PLANT 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. The "Status/Safety Priority Ranking" column notes those issues covered in other issues described in this document. For example, a notation of "I.A.2.2" in the Status/Safety Priority Ranking column for item I.A.2.6(3) means that item I.A.2.6(3) is covered in item I.A.2.2. For those issues covered in programs not described in this document, the Status/Safety Priority Ranking column includes the notation "S." For resolved issues that resulted in new requirements for operating plants, the appropriate multiplant licensing action number is given in the "MPA No." column. (The multiplant licensing action numbering system is not related to the numbering systems used to identify the prioritized issues.) This table is maintained primarily for historical purposes.

Legend

ACTIVE	Generic issue that involves actions under the GIP
DROP	Issue dropped from further pursuit as a generic issue
EI	Environmental issue
I	Resolved TMI Action Plan item with implementation of resolution mandated by NUREG-0737
LI	Licensing issue
LOW	Low safety priority (discontinued December 4, 2001)
MEDIUM	Medium safety priority (discontinued December 4, 2001)
MPA	Multiplant action
NA	Not applicable
NOTE:	1 Possible resolution identified for evaluation (discontinued July 6, 1998) 2 Resolution available (documented in NUREG, NRC memorandum, safety evaluation report, or equivalent) (discontinued July 6, 1998) 3 Resolution resulted in either: (a) the establishment of new regulatory requirements (by rule, Standard Review Plan change, or equivalent), or (b) no new requirements 4 Issue to be prioritized in the future (discontinued June 30, 2010) 5 Issue that is not a generic safety issue but should be assigned resources for completion (discontinued June 30, 2010)
ROI	Regulatory office implementation: A formal GI for which RES actions of safety/risk assessment or regulatory assessment are complete and remaining actions reside with program offices (e.g., regulatory compliance, reactor oversight process, rulemaking, further research, coordination with industry initiatives)
RI	Regulatory impact issue
S	Issue covered in an NRC program outside the scope of this document
USI	Unresolved safety issue

Table II

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

Table II (continued)

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

Table II (continued)

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

Table II (continued)

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

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
I.F.2(6)	Increase the Size of Licensees' QA Staff	J. Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	4	12/31/98	NA
I.F.2(7)	Clarify that the QA Program Is a Condition of the Construction Permit and Operating License	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA
I.F.2(8)	Compare NRC QA Requirements with Those of Other Agencies	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	J. Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	4	12/31/98	NA
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/30/98	NA
I.F.2(11)	Define Role of QA in Design and Analysis Activities	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/30/98	NA
<u>I.G</u>	<u>PREOPERATIONAL AND LOW-POWER TESTING</u>						
I.G.1	Training Requirements	-	NRR/DHFS/PSRB	I	3	12/31/97	
I.G.2	Scope of Test Program	H. Vandermolen	NRR/DHFS/PSRB	NOTE 3(a)	3	12/31/97	NA
<u>II.A</u>	<u>SITING</u>						
II.A.1	Siting Policy Reformulation	H. Vandermolen	NRR/DE/SAB	NOTE 3(b)	2	12/31/97	NA
II.A.2	Site Evaluation of Existing Facilities	H. Vandermolen	NRR/DE/SAB	V.A.1	2	12/31/97	NA
<u>II.B</u>	<u>CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW</u>						
II.B.1	Reactor Coolant System Vents	-	NRR/DL	I	4	12/31/97	F-10
II.B.2	Plant Shielding to Provide Access to Vital Areas and	-	NRR/DL	I	4	12/31/97	F-11
	Protect Safety Equipment for Post-Accident Operation	-	NRR/DL	I	4	12/31/97	F-12
II.B.3	Post-Accident Sampling						
II.B.4	Training for Mitigating Core Damage	-	NRR/DL	I	4	12/31/97	F-13
<u>II.B.5</u>	<u>Research on Phenomena Associated with Core Degradation and Fuel Melting</u>						
II.B.5(1)	Behavior of Severely Damaged Fuel	H. Vandermolen	RES/DSR/AEB	LI (NOTE 5)	4	12/31/97	NA
II.B.5(2)	Behavior of Core-Melt	H. Vandermolen	RES/DSR/AEB	LI (NOTE 5)	4	12/31/97	NA
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structure	H. Vandermolen	RES/DSR/AEB	LI (NOTE 5)	4	12/31/97	NA
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	J. Pittman	NRR/DST/RRAB	NOTE 3(a)	4	12/31/97	
II.B.7	Analysis of Hydrogen Control	P. Matthews	NRR/DSI/CSB	II.B.8	4	12/31/97	

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	H. Vandermolen	RES/DRAO/RAMR	NOTE 3(a)	4	12/31/97	
II.C	RELIABILITY ENGINEERING AND RISK ASSESSMENT						
II.C.1	Interim Reliability Evaluation Program	J. Pittman	RES/DRAO/RRB	NOTE 3(b)	3	12/31/97	NA
II.C.2	Continuation of Interim Reliability Evaluation Program	J. Pittman	NRR/DST/RRAB	NOTE 3(b)	3	12/31/97	NA
II.C.3	Systems Interaction	J. Pittman	NRR/DST/GIB	A-17	3	12/31/97	NA
II.C.4	Reliability Engineering	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	3	12/31/97	NA
II.D	REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES						
II.D.1	Testing Requirements	-	NRR/DL	I	3	12/31/98	F-14
II.D.2	Research on Relief and Safety Valve Test Requirements	R. Riggs	RES	DROP	3	12/31/98	NA
II.D.3	Relief and Safety Valve Position Indication	-	NRR	I	3	12/31/98	
II.E	SYSTEM DESIGN						
II.E.1	Auxiliary Feedwater System						
II.E.1.1	Auxiliary Feedwater System Evaluation	-	NRR/DL	I	2	12/31/97	F-15
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	-	NRR/DL	I	2	12/31/97	F-16, F-17
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	R. Riggs	RES/DRA/RRBR	NOTE 3(a)	2	12/31/97	
II.E.2	Emergency Core Cooling System						
II.E.2.1	Reliance on ECCS	R. Riggs	NRR/DSI/RSB	II.K.3(17)	3	12/31/98	NA
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	R. Riggs	RES/DAE/RSRB	NOTE 3(b)	3	12/31/98	NA
II.E.2.3	Uncertainties in Performance Predictions	H. Vandermolen	NRR/DSI/RSB	DROP	3	12/31/98	NA
II.E.3	Decay Heat Removal						
II.E.3.1	Reliability of Power Supplies for Natural Circulation	-	NRR/DL	I	2	12/31/97	
II.E.3.2	Systems Reliability	H. Vandermolen	NRR/DST/GIB	A-45	2	12/31/97	NA
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements	H. Vandermolen	NRR/DST/GIB	A-45	2	12/31/97	NA
II.E.3.4	Alternate Concepts Research	R. Riggs	RES/DAE/FBRB	NOTE 3(b)	2	12/31/97	NA
II.E.3.5	Regulatory Guide	R. Riggs	NRR/DST/GIB	A-45	2	12/31/97	NA
II.E.4	Containment Design						
II.E.4.1	Dedicated Penetrations	-	NRR/DL	I	2	12/31/97	F-18
II.E.4.2	Isolation Dependability	-	NRR/DL	I	2	12/31/97	F-19

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.E.4.3	Integrity Check	W. Milstead	RES/DRPS/RPSI	NOTE 3(b)	2	12/31/97	NA
II.E.4.4	Purging						
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging	W. Milstead	NRR/DSI/CSB	NOTE 3(a)	2	12/31/97	
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Letter	W. Milstead	NRR/DSI/CSB	NOTE 3(a)	2	12/31/97	
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability	W. Milstead	NRR/DSI/CSB	NOTE 3(a)	2	12/31/97	
II.E.4.4(4)	Evaluate Purging and Venting during Normal Operation	W. Milstead	NRR/DSI/CSB	NOTE 3(b)	2	12/31/97	NA
II.E.4.4(5)	Issue Modified Purging and Venting Requirement	W. Milstead	NRR/DSI/CSB	NOTE 3(b)	2	12/31/97	NA
<u>II.E.5</u>	<u>Design Sensitivity of B&W Reactors</u>						
II.E.5.1	Design Evaluation	D. Thatcher	NRR/DSI/RSB	NOTE 3(a)	2	12/31/98	
II.E.5.2	B&W Reactor Transient Response Task Force	D. Thatcher	NRR/DL/ORAB	NOTE 3(a)	2	12/31/98	
<u>II.E.6</u>	<u>In Situ Testing of Valves</u>						
II.E.6.1	Test Adequacy Study	D. Thatcher	RES/DE/EIB	NOTE 3(a)	2	12/31/98	
<u>II.F</u>	<u>INSTRUMENTATION AND CONTROLS</u>						
II.F.1	Additional Accident Monitoring Instrumentation	-	NRR/DL	I	3	12/31/98	F-20, F-21, F-22, F-23, F-24, F-25
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	-	NRR/DL	I	3	12/31/98	F-26
II.F.3	Instruments for Monitoring Accident Conditions	H. Vandermolen	RES/DFO/ICBR	NOTE 3(a)	3	12/31/98	
II.F.4	Study of Control and Protective Action Design Requirements	D. Thatcher	NRR/DSI/ICSB	DROP	3	12/31/98	NA
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	D. Thatcher	RES/DE	LI (NOTE 3)	3	12/31/98	NA
<u>II.G</u>	<u>ELECTRICAL POWER</u>						
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	-	NRR	I	1	12/31/98	NA
<u>II.H</u>	<u>TMI-2 CLEANUP AND EXAMINATION</u>						
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental	P. Matthews	NRR/TMIPO	NOTE 3(b)	3	12/31/98	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Impact						
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	W. Milstead	RES/DRAA/AEB	NOTE 3(b)	3	12/31/98	NA
II.H.3	Evaluate and Feed Back Information Obtained from TMI	W. Milstead	NRR/TMIPO	II.H.2	3	12/31/98	NA
II.H.4	Determine Impact of TMI on Socioeconomic and Real Property Values	W. Milstead	RES/DHSWM/SEBR	LI (NOTE 3)	3	12/31/98	NA
II.J	GENERAL IMPLICATIONS OF TMI FOR DESIGN AND CONSTRUCTION ACTIVITIES						
II.J.1	Vendor Inspection Program						
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.1.2	Modify Existing Vendor Inspection Program	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.2	Construction Inspection Program						
II.J.2.1	Reorient Construction Inspection Program	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.2.3	Assign Resident Inspectors to All Construction Sites	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.3	Management for Design and Construction						
II.J.3.1	Organization and Staffing to Oversee Design and Construction	J. Pittman	NRR/DHFS/LQB	I.B.1.1	1	12/31/98	NA
II.J.3.2	Issue Regulatory Guide	J. Pittman	NRR/DHFS/LQB	I.B.1.1	1	12/31/98	NA
II.J.4	Revise Deficiency Reporting Requirements						
II.J.4.1	Revise Deficiency Reporting Requirements	L. Riani	AEOD/DSP/ROAB	NOTE 3(a)	3	12/31/98	NA
II.K	MEASURES TO MITIGATE SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS AND LOSS-OF-FEEDWATER ACCIDENTS						
II.K.1	IE Bulletins						
II.K.1(1)	Review TMI-2 PNs and Detailed Chronology of the TMI-2 Accident	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(2)	Review Transients Similar to TMI-2 That Have	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Occurred at Other Facilities and NRC Evaluation of Davis-Besse Event						
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(4)	Review Operating Procedures and Training Instructions	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(5)	Safety-Related Valve Position Description	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred Out of Containment Inadvertently	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(11)	Make All Operating and Maintenance Personnel Aware of the Seriousness and Consequences of the Erroneous Actions Leading up to, and in Early Phases of, the TMI-2 Accident	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(12)	One-Hour Notification Requirement and Continuous Communications Channels	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(15)	For Facilities with Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(16)	Implement Procedures That Identify PRZ PORV "Open" Indications and That Direct Operator to Close Manually at "Reset" Setpoint	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(17)	Trip PZR Level Bistable So That PZR Low Pressure Will Initiate Safety Injection	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(18)	Develop Procedures and Train Operators on Methods of Establishing and Maintaining Natural Circulation	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(19)	Describe Design and Procedure Modifications to	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Reduce Likelihood of Automatic PZR PORV Actuation in Transients						
II.K.1(20)	Provide Procedures and Training to Operators for Prompt Manual Reactor Trip for LOFW, TT, MSIV Closure, LOOP, LOSG Level, and LO PZR Level	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor Trip for LOFW, TT, or Significant Decrease in SG Level	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(24)	Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses between Reactor Trip and RCP Trip	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(25)	Develop Operator Action Guidelines	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
<u>II.K.2</u>	<u>Commission Orders on B&W Plants</u>						
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System	R. Emrit	NRR/DSI	NOTE 3(a)	-	12/31/84	
II.K.2(2)	Procedures and Training to Initiate and Control AFW Independent of Integrated Control System	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	R. Emrit	NRR/DSI	NOTE 3(a)	-	12/31/84	
II.K.2(4)	Small-Break LOCA Analysis, Procedures and Operator Training	R. Emrit	NRR/DHFS/OLB	NOTE 3(a)	-	12/31/84	
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	R. Emrit	NRR	NOTE 3(a)	-	12/31/84	
II.K.2(6)	Reevaluate Analysis for Dual-Level Setpoint Control	R. Emrit	NRR/DSI	NOTE 3(a)	-	12/31/84	
II.K.2(7)	Reevaluate Transient of September 24, 1977	R. Emrit	NRR/DSI	NOTE 3(a)	-	12/31/84	
II.K.2(8)	Continued Upgrading of AFW System	R. Emrit	NRR	II.E.1.1, II.E.1.2	-	12/31/84	NA
II.K.2(9)	Analysis and Upgrading of Integrated Control System	R. Emrit	NRR	I	-	12/31/84	F-27
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	R. Emrit	NRR	I	-	12/31/84	F-28
II.K.2(11)	Operator Training and Drilling	R. Emrit	NRR	I	-	12/31/84	F-29
II.K.2(12)	Transient Analysis and Procedures for Management of Small Breaks	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.K.2(13)	Thermal-Mechanical Report on Effect of HPI on Vessel Integrity for Small-Break LOCA with No AFW	R. Emrit	NRR	I	-	12/31/84	F-30
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	R. Emrit	NRR	I	-	12/31/84	F-31
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes after Primary System Voiding	R. Emrit	NRR	I	-	12/31/84	
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA with Loss of Offsite Power	R. Emrit	NRR	I	-	12/31/84	F-32
II.K.2(17)	Analysis of Potential Voiding in RCS during Anticipated Transients	R. Emrit	NRR	I	-	12/31/84	F-33
II.K.2(18)	Analysis of Loss of Feedwater and Other Anticipated Transients	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.2(19)	Benchmark Analysis of Sequential AFW Flow to Once-Through Steam Generator	R. Emrit	NRR	I	-	12/31/84	F-34
II.K.2(20)	Analysis of Steam Response to Small-Break LOCA That Causes System Pressure to Exceed PORV Setpoint	R. Emrit	NRR	I	-	12/31/84	F-35
II.K.2(21)	LOFT L3-1 Predictions	R. Emrit	NRR/DSI	NOTE 3(a)	-	12/31/84	
<u>II.K.3</u>	<u>Final Recommendations of Bulletins and Orders Task Force</u>						
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	R. Emrit	NRR	I	-	12/31/84	F-36
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	R. Emrit	NRR	I	-	12/31/84	F-37
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	R. Emrit	NRR	I	-	12/31/84	F-38
II.K.3(4)	Review and Upgrade Reliability and Redundancy of Non-Safety Equipment for Small-Break LOCA Mitigation	R. Emrit	NRR	II.C.1, II.C.2, II.C.3	-	12/31/84	NA
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	R. Emrit	NRR	I	-	12/31/84	F-39, G-01
II.K.3(6)	Instrumentation to Verify Natural Circulation	R. Emrit	NRR/DSI	I.C.1(3), II.F.2, II.F.3	-	12/31/84	NA
II.K.3(7)	Evaluation of PORV Opening Probability during Overpressure Transient	R. Emrit	NRR	I	-	12/31/84	
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs	R. Emrit	NRR/DST/GIB	II.C.1, II.E.3.3	-	12/31/84	NA
II.K.3(9)	Proportional Integral Derivative Controller	R. Emrit	NRR	I	-	12/31/84	F-40

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Modification						
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	R. Emrit	NRR	I	-	12/31/84	F-41
II.K.3(11)	Control Use of PORV Supplied by Control Components, Inc., Until Further Review Complete	R. Emrit	NRR	I	-	12/31/84	
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	R. Emrit	NRR	I	-	12/31/84	F-42
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	R. Emrit	NRR	I	-	12/31/84	F-43
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	R. Emrit	NRR	I	-	12/31/84	F-44
II.K.3(15)	Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	R. Emrit	NRR	I	-	12/31/84	F-45
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves—Feasibility Study and System Modification	R. Emrit	NRR	I	-	12/31/84	F-46
II.K.3(17)	Report on Outage of ECC Systems—Licensee Report and Technical Specification Changes	R. Emrit	NRR	I	-	12/31/84	F-47
II.K.3(18)	Modification of ADS Logic—Feasibility Study and Modification for Increased Diversity for Some Event Sequences	R. Emrit	NRR	I	-	12/31/84	F-48
II.K.3(19)	Interlock on Recirculation Pump Loops	R. Emrit	NRR	I	-	12/31/84	F-49
II.K.3(20)	Loss of Service Water for Big Rock Point	R. Emrit	NRR	I	-	12/31/84	
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level—Design and Modification	R. Emrit	NRR	I	-	12/31/84	F-50
II.K.3(22)	Automatic Switchover of RCIC System Suction—Verify Procedures and Modify Design	R. Emrit	NRR	I	-	12/31/84	F-51
II.K.3(23)	Central Water Level Recording	R. Emrit	NRR	I.D.2, III.A.1.2(1), III.A.3.4	-	12/31/84	NA
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	R. Emrit	NRR	I	-	12/31/84	F-52
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	R. Emrit	NRR	I	-	12/31/84	F-53
II.K.3(26)	Study Effect on RHR Reliability of Its Use for Fuel Pool Cooling	R. Emrit	NRR/DSI	II.E.2.1	-	12/31/84	NA
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	R. Emrit	NRR	I	-	12/31/84	F-54
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	R. Emrit	NRR	I	-	12/31/84	F-55
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	R. Emrit	NRR	I	-	12/31/84	F-56

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	R. Emrit	NRR	I	-	12/31/84	F-57
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	R. Emrit	NRR	I	-	12/31/84	F-58
II.K.3(32)	Provide Experimental Verification of Two-Phase Natural Circulation Models	R. Emrit	NRR/DSI	II.E.2.2	-	12/31/84	NA
II.K.3(33)	Evaluate Elimination of PORV Function	R. Emrit	NRR	II.C.1	-	12/31/84	NA
II.K.3(34)	Relap-4 Model Development	R. Emrit	NRR/DSI	II.E.2.2	-	12/31/84	NA
II.K.3(35)	Evaluation of Effects of Core Flood Tank Injection on Small-Break LOCAs	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.3(36)	Additional Staff Audit Calculations of B&W Small-Break LOCA Analyses	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.3(37)	Analysis of B&W Response to Isolated Small-Break LOCA	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.3(38)	Analysis of Plant Response to a Small-Break LOCA in the Pressurizer Spray Line	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.3(39)	Evaluation of Effects of Water Slugs in Piping Caused by HPI and CFT Flows	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.3(40)	Evaluation of RCP Seal Damage and Leakage during a Small-Break LOCA	R. Emrit	NRR	II.K.2(16)	-	12/31/84	NA
II.K.3(41)	Submit Predictions for LOFT Test L3-6 with RCPs Running	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.3(42)	Submit Requested Information on the Effects of Non-Condensible Gases	R. Emrit	NRR	I.C.1(3)	-	12/31/84	NA
II.K.3(43)	Evaluation of Mechanical Effects of Slug Flow on Steam Generator Tubes	R. Emrit	NRR	II.K.2(15)	-	12/31/84	NA
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	R. Emrit	NRR	I	-	12/31/84	F-59
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	R. Emrit	NRR	I	-	12/31/84	F-60
II.K.3(46)	Response to List of Concerns from ACRS Consultant	R. Emrit	NRR	I	-	12/31/84	F-61
II.K.3(47)	Test Program for Small-Break LOCA Model Verification Pretest Prediction, Test Program, and Model Verification	R. Emrit	NRR	I.C.1(3), II.E.2.2	-	12/31/84	NA
II.K.3(48)	Assess Change in Safety Reliability as a Result of Implementing B&OTF Recommendations	R. Emrit	NRR	II.C.1, II.C.2	-	12/31/84	NA
II.K.3(49)	Review of Procedures (NRC)	R. Emrit	NRR/DHFS/PSRB	I.C.8, I.C.9	-	12/31/84	NA
II.K.3(50)	Review of Procedures (NSSS Vendors)	R. Emrit	NRR/DHFS/PSRB	I.C.7,	-	12/31/84 I.C.9	NA
II.K.3(51)	Symptom-Based Emergency Procedures	R. Emrit	NRR/DHFS/PSRB	I.C.9	-	12/31/84	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.K.3(52)	Operator Awareness of Revised Emergency Procedures	R. Emrit	NRR	I.B.1.1, I.C.2, I.C.5	-	12/31/84	NA
II.K.3(53)	Two Operators in Control Room	R. Emrit	NRR	I.A.1.3	-	12/31/84	NA
II.K.3(54)	Simulator Upgrade for Small-Break LOCAs	R. Emrit	NRR	I.A.4.1(2)	-	12/31/84	NA
II.K.3(55)	Operator Monitoring of Control Board	R. Emrit	NRR	I.C.1(3), I.D.2, I.D.3	-	12/31/84	NA
II.K.3(56)	Simulator Training Requirements	R. Emrit	NRR/DHFS/OLB	I.A.2.6(3), I.A.3.1	-	12/31/84	NA
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	R. Emrit	NRR	I	-	12/31/84	F-62
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						
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						
III.A.1.3(1)	Workers	R. Riggs	OIE/DEPER/EPB	NOTE 3(b)	2	06/30/91	NA
III.A.1.3(2)	Public	R. Riggs	OIE/DEPER/EPB	NOTE 3(b)	2	06/30/91	NA
III.A.2	Improving Licensee Emergency Preparedness—Long-Term						
III.A.2.1	Amend 10 CFR 50 and 10 CFR 50, Appendix E						
III.A.2.1(1)	Publish Proposed Amendments to the Rules	-	RES	NOTE 3(a)	-	12/31/94	NA
III.A.2.1(2)	Conduct Public Regional Meetings	-	RES	NOTE 3(b)	-	12/31/94	NA
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules	-	RES	NOTE 3(b)	-	12/31/94	NA
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	-	OIE	I	-		F-67
III.A.2.2	Development of Guidance and Criteria	-	NRR/DL	I	-		F-68
III.A.3	Improving NRC Emergency Preparedness						

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
III.A.3.1	NRC Role in Responding to Nuclear Emergencies						
III.A.3.1(1)	Define NRC Role in Emergency Situations	R. Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(2)	Revise and Upgrade Plans and Procedures for the NRC Emergency Operations Center	R. Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(3)	Revise Manual Chapter 0502, Other Agency Procedures, and NUREG-0610	R. Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(4)	Prepare Commission Paper	R. Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.1(5)	Revise Implementing Procedures and Instructions for Regional Offices	R. Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.2	Improve Operations Centers	R. Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.3	Communications						
III.A.3.3(1)	Install Direct Dedicated Telephone Lines	J. Pittman	OIE/DEPER/IRDB	NOTE 3(a)	1	06/30/85	NA
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems	J. Pittman	OIE/DEPER/IRDB	NOTE 3(a)	1	06/30/85	NA
III.A.3.4	Nuclear Data Link	D. Thatcher	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	
III.A.3.5	Training, Drills, and Tests	J. Pittman	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA
III.A.3.6	Interaction of NRC and Other Agencies						
III.A.3.6(1)	International	J. Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	06/30/85	NA
III.A.3.6(2)	Federal	J. Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	06/30/85	NA
III.A.3.6(3)	State and Local	J. Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	06/30/85	NA
III.B	EMERGENCY PREPAREDNESS OF STATE AND LOCAL GOVERNMENTS						
III.B.1	Transfer of Responsibilities to FEMA	W. Milstead	OIE/DEPER/IRDB	NOTE 3(b)	-	11/30/83	NA
III.B.2	Implementation of NRC and FEMA Responsibilities						
III.B.2(1)	The Licensing Process	W. Milstead	OIE/DEPER/IRDB	NOTE 3(b)	-	11/30/83	NA
III.B.2(2)	Federal Guidance	W. Milstead	OIE/DEPER/IRDB	NOTE 3(b)	-	11/30/83	NA
III.C	PUBLIC INFORMATION						
III.C.1	Have Information Available for the News Media and the Public						
III.C.1(1)	Review Publicly Available Documents	J. Pittman	PA	LI (NOTE 3)	-	11/30/83	NA
III.C.1(2)	Recommend Publication of Additional Information	J. Pittman	PA	LI (NOTE 3)	-	11/30/83	NA
III.C.1(3)	Program of Seminars for News Media Personnel	J. Pittman	PA	LI (NOTE 3)	-	11/30/83	NA
III.C.2	Develop Policy and Provide Training for Interfacing with the News Media						
III.C.2(1)	Develop Policy and Procedures for Dealing with Briefing	J. Pittman	PA	LI (NOTE 3)	-	11/30/83	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Requests						
III.C.2(2)	Provide Training for Members of the Technical Staff	J. Pittman	PA	LI (NOTE 3)	-	11/30/83	NA
III.D	RADIATION PROTECTION						
III.D.1	Radiation Source Control						
III.D.1.1	Primary Coolant Sources Outside the Containment Structure						
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining to Reducing Leakage from Operating Systems	-	NRR	I	1	12/31/88	
III.D.1.1(2)	Review Information on Provisions for Leak Detection	R. Emrit	RES/DRA/ARGIB	DROP	1	12/31/88	
III.D.1.1(3)	Develop Proposed System Acceptance Criteria	R. Emrit	RES/DRA/ARGIB	DROP	1	12/31/88	
III.D.1.2	Radioactive Gas Management	R. Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3	Ventilation System and Radioiodine Adsorber Criteria	-	-	-			
III.D.1.3(1)	Decide Whether Licensees Should Perform Studies and Make Modifications	R. Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3(2)	Review and Revise SRP	R. Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems	R. Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Adsorber	R. Emrit	NRR/DSI/METB	NOTE 3(b)	1	12/31/88	NA
III.D.1.4	Radwaste System Design Features to Aid in Accident Recovery and Decontamination	R. Emrit	NRR/DSI/METB	DROP	1	12/31/88	NA
III.D.2	Public Radiation Protection Improvement						
III.D.2.1	Radiological Monitoring of Effluents						
III.D.2.1(1)	Evaluate the Feasibility and Perform a Value-Impact Analysis of Modifying Effluent-Monitoring Design Criteria	R. Emrit	NRR/DSI/METB	LOW	3	12/31/98	NA
III.D.2.1(2)	Study the Feasibility of Requiring the Development of Effective Means for Monitoring and Sampling Noble Gases and Radioiodine Released to the Atmosphere	R. Emrit	NRR/DSI/METB	LOW	3	12/31/98	NA
III.D.2.1(3)	Revise Regulatory Guides	R. Emrit	NRR/DSI/METB	LOW	3	12/31/98	NA
III.D.2.2	Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis						
III.D.2.2(1)	Perform Study of Radioiodine, Carbon-14, and Tritium Behavior	R. Emrit	NRR/DSI/RAB	NOTE 3(b)	3	12/31/98	NA
III.D.2.2(2)	Evaluate Data Collected at Quad Cities	R. Emrit	NRR/DSI/RAB	III.D.2.5	3	12/31/98	NA
III.D.2.2(3)	Determine the Distribution of the Chemical Species of Radioiodine in Air-Water-Steam Mixtures	R. Emrit	NRR/DSI/RAB	III.D.2.5	3	12/31/98	NA
III.D.2.2(4)	Revise SRP and Regulatory Guides	R. Emrit	NRR/DSI/RAB	III.D.2.5	3	12/31/98	NA
III.D.2.3	Liquid Pathway Radiological Control						

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
III.D.2.3(1)	Develop Procedures to Discriminate Between Sites/Plants	R. Emrit	NRR/DE/EHEB	NOTE 3(b)	3	12/31/98	NA
III.D.2.3(2)	Discriminate Between Sites and Plants That Require Consideration of Liquid Pathway Interdiction Techniques	R. Emrit	NRR/DE/EHEB	NOTE 3(b)	3	12/31/98	NA
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction	R. Emrit	NRR/DE/EHEB	NOTE 3(b)	3	12/31/98	NA
III.D.2.3(4)	Prepare a Summary Assessment	R. Emrit	NRR/DE/EHEB	NOTE 3(b)	3	12/31/98	NA
III.D.2.4	Offsite Dose Measurements						
III.D.2.4(1)	Study Feasibility of Environmental Monitors	H. Vandermolen	NRR/DSI/RAB	NOTE 3(b)	3	12/31/98	NA
III.D.2.4(2)	Place 50 TLDs Around Each Site	H. Vandermolen	OIE/DRP/ORPB	LI (NOTE 3)	3	12/31/98	NA
III.D.2.5	Offsite Dose Calculation Manual	H. Vandermolen	NRR/DSI/RAB	NOTE 3(b)	3	12/31/98	NA
III.D.2.6	Independent Radiological Measurements	H. Vandermolen	OIE/DRP/ORPB	LI (NOTE 3)	3	12/31/98	NA
<u>III.D.3</u>	<u>Worker Radiation Protection Improvement</u>						
III.D.3.1	Radiation Protection Plans	H. Vandermolen	NRR/DSI/RAB	NOTE 3(b)	3	12/31/87	NA
III.D.3.2	Health Physics Improvements						
III.D.3.2(1)	Amend 10 CFR 20	H. Vandermolen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(2)	Issue a Regulatory Guide	H. Vandermolen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(3)	Develop Standard Performance Criteria	H. Vandermolen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(4)	Develop Method for Testing and Certifying Air-Purifying Respirators	H. Vandermolen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.3	In-plant Radiation Monitoring						
III.D.3.3(1)	Issue Letter Requiring Improved Radiation Sampling Instrumentation	-	NRR/DL	I	2	12/31/86	F-69
III.D.3.3(2)	Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	-	NRR	NOTE 3(a)	2	12/31/86	NA
III.D.3.3(3)	Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation-Monitoring Instruments	-	RES	NOTE 3(a)	2	12/31/86	NA
III.D.3.3(4)	Issue a Regulatory Guide	-	RES	NOTE 3(a)	2	12/31/86	NA
III.D.3.4	Control Room Habitability	-	NRR/DL	I	2	12/31/86	F-70
III.D.3.5	Radiation Worker Exposure						
III.D.3.5(1)	Develop Format for Data To Be Collected by Utilities Regarding Total Radiation Exposure to Workers	H. Vandermolen	DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
III.D.3.5(2)	Investigative Methods of Obtaining Employee Health Data by Nonlegislative Means	H. Vandermolen	DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
III.D.3.5(3)	Revise 10 CFR 20	H. Vandermolen	DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
<u>IV.A</u>	<u>STRENGTHEN ENFORCEMENT PROCESS</u>						
IV.A.1	Seek Legislative Authority	R. Emrit	GC	LI (NOTE 3)	-	11/30/83	NA
IV.A.2	Revise Enforcement Policy	R. Emrit	OIE/ES	LI (NOTE 3)	-	11/30/83	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
IV.B	ISSUANCE OF INSTRUCTIONS AND INFORMATION TO LICENSEES						
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	R. Emrit	OIE/DEPER	LI (NOTE 3)	-	11/30/83	NA
IV.C	EXTEND LESSONS LEARNED TO LICENSED ACTIVITIES OTHER THAN POWER REACTORS						
IV.C.1	Extend Lessons Learned from TMI to Other NRC Programs	R. Emrit	NMSS/WM	NOTE 3(b)	-	11/30/83	NA
IV.D	NRC STAFF TRAINING						
IV.D.1	NRC Staff Training	R. Emrit	ADM/MDTS	LI (NOTE 3)	-	11/30/83	NA
IV.E	SAFETY DECISION-MAKING						
IV.E.1	Expand Research on Quantification of Safety Decisionmaking	R. Colmar	RES/DRA/RABR	LI (NOTE 3)	2	12/31/86	NA
IV.E.2	Plan for Early Resolution of Safety Issues	R. Emrit	NRR/DST/SPEB	LI (NOTE 3)	2	12/31/86	NA
IV.E.3	Plan for Resolving Issues at the CP Stage	R. Colmar	RES/DRA/RABR	LI (NOTE 5)	2	12/31/86	NA
IV.E.4	Resolve Generic Issues by Rulemaking	R. Colmar	RES/DRA/RABR	LI (NOTE 3)	2	12/31/86	NA
IV.E.5	Assess Currently Operating Reactors	P. Matthews	NRR/DL/SEPB	NOTE 3(b)	2	12/31/86	NA
IV.F	FINANCIAL DISINCENTIVES TO SAFETY						
IV.F.1	Increased OIE Scrutiny of the Power-Ascension Test Program	D. Thatcher	OIE/DQASIP	NOTE 3(b)	1	12/31/86	NA
IV.F.2	Evaluate the Impacts of Financial Disincentives to the Safety of Nuclear Power Plants	P. Matthews	SP	NOTE 3(b)	1	12/31/86	NA
IV.G	IMPROVE SAFETY RULEMAKING PROCEDURES						
IV.G.1	Develop a Public Agenda for Rulemaking	R. Emrit	ADM/RPB	LI (NOTE 3)	1	12/31/86	NA
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules	W. Milstead	RES/DRA/RABR	LI (NOTE 3)	1	12/31/86	NA
IV.G.3	Improve Rulemaking Procedures	W. Milstead	RES/DRA/RABR	LI (NOTE 3)	1	12/31/86	NA
IV.G.4	Study Alternatives for Improved Rulemaking Process	W. Milstead	RES/DRA/RABR	LI (NOTE 3)	1	12/31/86	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
IV.H	NRC PARTICIPATION IN THE RADIATION POLICY COUNCIL						
IV.H.1	NRC Participation in the Radiation Policy Council	G. Sege	RES/DHSWM/HEBR	LI (NOTE 3)	-	11/30/83	NA
V.A	DEVELOPMENT OF SAFETY POLICY						
V.A.1	Develop NRC Policy Statement on Safety	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.B	POSSIBLE ELIMINATION OF NONSAFETY RESPONSIBILITIES						
V.B.1	Study and Recommend, as Appropriate, Elimination of Nonsafety Responsibilities	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.C	ADVISORY COMMITTEES						
V.C.1	Strengthen the Role of Advisory Committee on Reactor Safeguards	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.C.2	Study Need for Additional Advisory Committees	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.C.3	Study the Need to Establish an Independent Nuclear Safety Board	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.D	LICENSING PROCESS						
V.D.1	Improve Public and Intervenor Participation in the Hearing Process	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.D.2	Study Construction-during-Adjudication Rules	R. Emrit	GC	LI (NOTE 5)	-	12/31/86	NA
V.D.3	Reexamine Commission Role in Adjudication	R. Emrit	GC	LI (NOTE 5)	-	12/31/86	NA
V.D.4	Study the Reform of the Licensing Process	R. Emrit	GC	LI (NOTE 5)	-	12/31/86	NA
V.E	LEGISLATIVE NEEDS						
V.E.1	Study the Need for TMI-Related Legislation	R. Emrit	GC	LI (NOTE 5)	-	12/31/86	NA
V.F	ORGANIZATION AND MANAGEMENT						
V.F.1	Study NRC Top Management Structure and Process	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.F.2	Reexamine Organization and Functions of the NRC Offices	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.F.3	Revise Delegations of Authority to Staff	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
V.F.4	Clarify and Strengthen the Respective Roles of Chairman, Commission, and Executive Director for Operations	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.F.5	Authority to Delegate Emergency Response Functions to a Single Commissioner	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.G	CONSOLIDATION OF NRC LOCATIONS						
V.G.1	Achieve Single Location, Long-Term	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
V.G.2	Achieve Single Location, Interim	R. Emrit	GC	LI (NOTE 3)	-	12/31/86	NA
	TASK ACTION PLAN ITEMS						
A-1	Water Hammer (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	D-10
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	R. Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	12/31/88	
A-4	CE Steam Generator Tube Integrity (former USI)	R. Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	12/31/88	
A-5	B&W Steam Generator Tube Integrity (former USI)	R. Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	12/31/88	
A-6	Mark I Short-Term Program (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-7	Mark I Long-Term Program (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	D-01
A-8	Mark II Containment Pool Dynamic Loads Long-Term Program (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-9	ATWS (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-10	BWR Feedwater Nozzle Cracking (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-25
A-11	Reactor Vessel Materials Toughness (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-13	Snubber Operability Assurance	R. Emrit	NRR/DE/MEB	NOTE 3(a)	1	06/30/91	B-17, B-22
A-14	Flaw Detection	P. Matthews	NRR/DE/MTEB	DROP	-	11/30/83	NA
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning	J. Pittman	NRR/DE/CHEB	NOTE 3(b)	-	11/30/83	NA
A-16	Steam Effects on BWR Core Spray Distribution	R. Emrit	NRR/DSI/CPB	NOTE 3(a)	-	11/30/83	D-12
A-17	Systems Interactions in Nuclear Power Plants (former USI)	R. Emrit	RES/DSIR/EIB	NOTE 3(b)	1	12/31/89	NA
A-18	Pipe Rupture Design Criteria	R. Emrit	NRR/DE/MEB	DROP	-	11/30/83	NA
A-19	Digital Computer Protection System	W. Milstead	RES/DSR/HFB	LI (NOTE 5)	1	06/30/91	NA
A-20	Impacts of the Coal Fuel Cycle	-	NRR/DE/EHEB	LI (NOTE 5)	-	11/30/83	NA
A-21	Main Steamline Break Inside Containment—Evaluation of Environmental Conditions for Equipment Qualification	H. Vandermolen	NRR/DSI/CSB	DROP	1	12/31/98	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
A-22	PWR Main Steamline Break—Core, Reactor Vessel, and Containment Building Response	H. Vandermolen	NRR/DSI/CSB	DROP	-	11/30/83	NA
A-23	Containment Leak Testing	P. Matthews	NRR/DSI/CSB	RI (NOTE 5)	-	11/30/83	
A-24	Qualification of Class 1E Safety-Related Equipment (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-60
A-25	Non-Safety Loads on Class 1E Power Sources	D. Thatcher	NRR/DSI/PSB	NOTE 3(a)	-	11/30/83	
A-26	Reactor Vessel Pressure Transient Protection (former (USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-04
A-27	Reload Applications	-	NRR/DSI/CPB	LI (NOTE 5)	-	11/30/83	NA
A-28	Increase in Spent Fuel Pool Storage Capacity	R. Colmar	NRR/DE/SGEB	NOTE 3(a)	-	11/30/83	
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	R. Colmar	RES/DRPS/RPSI	NOTE 3(b)	1	12/31/89	NA
A-30	Adequacy of Safety-Related DC Power Supplies	G. Sege	NRR/DSI/PSB	128	1	12/31/86	NA
A-31	RHR Shutdown Requirements (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-32	Missile Effects	J. Pittman	NRR/DE/MTEB	A-37, A-38, B-68	-	11/30/83	NA
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	H. Vandermolen	NRR/DSI/ICSB	II.F.3	-	11/30/83	NA
A-35	Adequacy of Offsite Power Systems	R. Emrit	NRR/DSI/PSB	NOTE 3(a)	1	12/31/94	B-23
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	R. Emrit	NRR/DSI/GIB	NOTE 3(a)	2	06/30/04	C-10, C-15
A-37	Turbine Missiles	J. Pittman	NRR/DE/MTEB	DROP	-	11/30/83	NA
A-38	Tornado Missiles	G. Sege	NRR/DSI/ASB	DROP	3	06/30/00	NA
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-40	Seismic Design Criteria (former USI)	R. Emrit	RES/DSIR/EIB	NOTE 3(a)	1	12/31/89	NA
A-41	Long-Term Seismic Program	L. Riani	NRR/DE/MEB	NOTE 3(b)	1	12/31/84	NA
A-42	Pipe Cracks in Boiling Water Reactors (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-05
A-43	Containment Emergency Sump Performance (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	12/31/87	
A-44	Station Blackout (former USI)	R. Emrit	RES/DRPS/RPSI	NOTE 3(a)	1	06/30/88	
A-45	Shutdown Decay Heat Removal Requirements (former USI)	R. Emrit	RES/DRPS/RPSI	NOTE 3(b)	1	12/31/88	NA
A-46	Seismic Qualification of Equipment in Operating Plants (former USI)	R. Emrit	NRR/DSRO/EIB	NOTE 3(a)	2	06/30/00	
A-47	Safety Implications of Control Systems (former USI)	R. Emrit	RES/DSIR/EIB	NOTE 3(a)	1	12/31/89	
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	R. Emrit	NRR/DSIR/SAIB	NOTE 3(a)	1	06/30/89	
A-49	Pressurized Thermal Shock (former USI)	R. Emrit	NRR/DSRO/RSIB	NOTE 3(a)	1	12/31/87	A-21
B-1	Environmental Technical Specifications	-	NRR/DE/EHEB	EI (NOTE 3)	-	11/30/83	NA
B-2	Forecasting Electricity Demand	-	NRR	EI (NOTE 3)	-	11/30/83	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
B-3	Event Categorization	-	NRR/DSI/RSB	LI (NOTE 3)	-	11/30/83	NA
B-4	ECCS Reliability	R. Emrit	NRR/DSI/RSB	II.E.3.2	-	11/30/83	NA
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	D. Thatcher	RES/DE/EIB	NOTE 3(b)	1	06/30/88	NA
B-6	Loads, Load Combinations, Stress Limits	J. Pittman	NRR/DSRO/EIB	119.1	-	12/31/87	NA
B-7	Secondary Accident Consequence Modeling	-	NRR/DSI/AEB	LI (NOTE 3)	-	11/30/83	NA
B-8	Locking Out of ECCS Power Operated Valves	R. Riggs	NRR/DSI/RSB	DROP	1	12/31/94	NA
B-9	Electrical Cable Penetrations of Containment	R. Emrit	NRR/DSI/PSB	NOTE 3(b)	-	11/30/83	NA
B-10	Behavior of BWR Mark III Containments	H. Vandermolen	NRR/DSI/CSB	NOTE 3(a)	1	12/31/84	NA
B-11	Subcompartment Standard Problems	-	NRR/DSI/CSB	LI (NOTE 5)	-	11/30/83	NA
B-12	Containment Cooling Requirements (Non-LOCA)	R. Emrit	NRR/DSI/CSB	NOTE 3(b)	1	12/31/86	NA
B-13	Marviken Test Data Evaluation	-	NRR/DSI/CSB	LI (NOTE 5)	-	11/30/83	NA
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA	R. Emrit	NRR/DST/GIB	A-48	-	11/30/83	NA
B-15	CONTEMPT Computer Code Maintenance	-	NRR/DSI/CSB	LI (NOTE 3)	-	11/30/83	NA
B-16	Protection against Postulated Piping Failures in Fluid Systems Outside Containment	R. Emrit	NRR/DE/MEB	A-18	-	11/30/83	NA
B-17	Criteria for Safety-Related Operator Actions	W. Milstead	RES/DST/CIHFB	NOTE 3(b)	3	06/30/00	
B-18	Vortex Suppression Requirements for Containment Sumps	R. Emrit	NRR/DST/GIB	A-43	-	11/30/83	NA
B-19	Thermal-Hydraulic Stability	L. Riani	NRR/DSI/CPB	NOTE 3(b)	-	06/30/85	NA
B-20	Standard Problem Analysis	-	RES/DAE/AMBR	LI (NOTE 5)	-	11/30/83	
B-21	Core Physics	-	NRR/DSI/CPB	LI (NOTE 3)	-	11/30/83	NA
B-22	LWR Fuel	R. Emrit	RES/DSIR/RPSIB	DROP	2	06/30/95	NA
B-23	LMFBR Fuel	-	NRR/DSI/CPB	LI (NOTE 3)	-	11/30/83	NA
B-24	Seismic Qualification of Electrical and Mechanical Equipment	R. Emrit	NRR	A-46	-	11/30/83	NA
B-25	Piping Benchmark Problems	-	NRR/DE/MEB	LI (NOTE 5)	-	11/30/83	
B-26	Structural Integrity of Containment Penetrations	R. Riggs	NRR/DE/MTEB	NOTE 3(b)	1	12/31/84	NA
B-27	Implementation and Use of Subsection NF	-	NRR/DE/MEB	LI (NOTE 5)	-	11/30/83	
B-28	Radionuclide/Sediment Transport Program	-	NRR/DE/EHEB	EI (NOTE 3)	-	11/30/83	NA
B-29	Effectiveness of Ultimate Heat Sinks	J. Pittman	NRR/DE/EHEB	LI (NOTE 3)	1	06/30/91	NA
B-30	Design Basis Floods and Probability	-	NRR/DE/EHEB	LI (NOTE 5)	-	11/30/83	
B-31	Dam Failure Model	W. Milstead	NRR/DE/SGEB	LI (NOTE 3)	1	06/30/89	NA
B-32	Ice Effects on Safety-Related Water Supplies	J. Pittman	NRR/DE/EHEB	153	1	06/30/91	NA
B-33	Dose Assessment Methodology	-	NRR/DSI/RAB	LI (NOTE 3)	-	11/30/83	NA
B-34	Occupational Radiation Exposure Reduction	R. Emrit	NRR/DSI/RAB	III.D.3.1	-	11/30/83	NA
B-35	Confirmation of Appendix I Models for Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors	-	NRR/DSI/METB	LI (NOTE 5)	-	11/30/83	
B-36	Develop Design, Testing, and Maintenance Criteria for	R. Emrit	NRR/DSI/METB	NOTE 3(a)	-	11/30/83	

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems						
B-37	Chemical Discharges to Receiving Waters	-	NRR/DE/EHEB	EI (NOTE 5)	-	11/30/83	
B-38	Reconnaissance-Level Investigations	-	NRR/DE/EHEB	EI (NOTE 3)	-	11/30/83	NA
B-39	Transmission Lines	-	NRR/DE/EHEB	EI (NOTE 3)	-	11/30/83	NA
B-40	Effects of Power Plant Entrainment on Plankton	-	NRR/DE/EHEB	EI (NOTE 3)	-	11/30/83	NA
B-41	Impacts on Fisheries	-	NRR/DE/EHEB	EI (NOTE 3)	-	11/30/83	NA
B-42	Socioeconomic Environmental Impacts	-	NRR/DE/SAB	EI (NOTE 3)	-	11/30/83	NA
B-43	Value of Aerial Photographs for Site Evaluation	-	NRR/DE/EHEB	EI (NOTE 5)	-	11/30/83	
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants	-	NRR/DE/SAB	EI (NOTE 3)	-	11/30/83	NA
B-45	Need for Power—Energy Conservation	-	NRR/DE/SAB	EI (NOTE 3)	-	11/30/83	NA
B-46	Cost of Alternatives in Environmental Design	-	NRR/DE/SAB	EI (NOTE 3)	-	11/30/83	NA
B-47	Inservice Inspection of Supports—Classes 1, 2, 3, and MC Components	L. Riani	NRR/DE/MTEB	DROP	-	11/30/83	NA
B-48	BWR Control Rod Drive Mechanical Failures	R. Emrit	NRR/DE/MTEB	NOTE 3(b)	-	11/30/83	
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments	-	NRR	LI (NOTE 5)	-	11/30/83	
B-50	Post-Operating Basis Earthquake Inspection	L. Riani	NRR/DE/SGEB	RI (NOTE 3)	1	06/30/85	NA
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components	R. Emrit	NRR/DE/MEB	A-40	-	11/30/83	NA
B-52	Fuel Assembly Seismic and LOCA Responses	R. Emrit	NRR/DST/GIB	A-2	-	11/30/83	NA
B-53	Load Break Switch	G. Sege	NRR/DSI/PSB	RI (NOTE 3)	-	11/30/83	
B-54	Ice Condenser Containments	W. Milstead	NRR/DSI/CSB	NOTE 3(b)	1	12/31/84	NA
B-55	Improved Reliability of Target Rock Safety Relief Valves	H. Vandermolen	NRR/DE/EMEB	NOTE 3(b)	1	06/30/00	
B-56	Diesel Reliability	W. Milstead	RES/DRPS/RPSI	NOTE 3(a)	2	06/30/95	D-19
B-57	Station Blackout	R. Emrit	NRR/DST/GIB	A-44	-	11/30/83	
B-58	Passive Mechanical Failures	L. Riani	NRR/DE/EQB	NOTE 3(b)	1	12/31/85	NA
B-59	(N-1) Loop Operation in BWRs and PWRs	L. Riani	NRR/DSI/RSB	RI (NOTE 3)	1	06/30/85	E-04, E-05
B-60	Loose Parts Monitoring Systems	R. Emrit	NRR/DSI/CPB	NOTE 3(b)	1	12/31/84	NA
B-61	Allowable ECCS Equipment Outage Periods	J. Pittman	RES/DST/PRAB	NOTE 3(b)	1	06/30/00	
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions	-	NRR/DSI/CPB	LI (NOTE 3)	-	11/30/83	NA
B-63	Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary	R. Emrit	NRR/DE/MEB	NOTE 3(a)	-	11/30/83	B-45
B-64	Decommissioning of Reactors	L. Riani	RES/DE/MEB	NOTE 3(a)	2	06/30/95	NA
B-65	Iodine Spiking	W. Milstead	NRR/DSI/AEB	DROP	2	12/31/84	NA
B-66	Control Room Infiltration Measurements	P. Matthews	NRR/DSI/AEB	NOTE 3(a)	-	11/30/83	

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
B-67	Effluent and Process Monitoring Instrumentation	L. Riani	NRR/DSI/METB	III.D.2.1	-	11/30/83	NA
B-68	Pump Overspeed during LOCA	L. Riani	NRR/DSI/ASB	DROP	-	11/30/83	NA
B-69	ECCS Leakage Ex-Containment	L. Riani	NRR/DSI/METB	III.D.1.1(1)	-	11/30/83	NA
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps	R. Emrit	NRR/DSI/PSB	NOTE 3(b)	-	11/30/83	
B-71	Incident Response	L. Riani	NRR	III.A.3.1	-	11/30/83	NA
B-72	Health Effects and Life Shortening from Uranium and Coal Fuel Cycles	-	NRR/DSI/RAB	LI (NOTE 5)	-	11/30/83	NA
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel	D. Thatcher	NRR/DE/MEB	C-12	-	11/30/83	NA
C-1	Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	W. Milstead	NRR/DE/EQB	NOTE 3(a)	-	11/30/83	
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure	R. Emrit	NRR/DSI/CSB	NOTE 3(b)	-	11/30/83	NA
C-3	Insulation Usage within Containment	R. Emrit	NRR/DST/GIB	A-43	1	06/30/91	NA
C-4	Statistical Methods for ECCS Analysis	R. Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-5	Decay Heat Update	R. Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-6	LOCA Heat Sources	R. Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-7	PWR System Piping	R. Emrit	NRR/DE/MTEB	NOTE 3(b)	-	11/30/83	NA
C-8	Main Steam Line Leakage Control Systems	W. Milstead	RES/DRPS/RPSI	NOTE 3(b)	1	06/30/90	NA
C-9	RHR Heat Exchanger Tube Failures	H. Vandermolen	NRR/DSI/RSB	DROP	-	11/30/83	NA
C-10	Effective Operation of Containment Sprays in a LOCA	R. Emrit	NRR/DSI/AEB	NOTE 3(a)	-	11/30/83	NA
C-11	Assessment of Failure and Reliability of Pumps and Valves	R. Emrit	NRR/DE/MEB	NOTE 3(b)	-	12/31/85	NA
C-12	Primary System Vibration Assessment	D. Thatcher	NRR/DE/MEB	NOTE 3(b)	-	11/30/83	NA
C-13	Non-Random Failures	R. Emrit	NRR/DST/GIB	A-17	1	06/30/91	NA
C-14	Storm Surge Model for Coastal Sites	R. Emrit	NRR/DE/EHEB	LI (NOTE 3)	-	06/30/88	NA
C-15	NUREG Report for Liquid Tank Failure Analysis	-	NRR/DE/EHEB	LI (NOTE 3)	-	11/30/83	NA
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	-	NRR/DE/EHEB	EI (NOTE 3)	-	11/30/83	NA
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	R. Emrit	NRR/DSI/METB	NOTE 3(a)	-	11/30/83	NA
D-1	Advisability of a Seismic Scram	D. Thatcher	RES/DET/MSEB	DROP	1	12/31/98	NA
D-2	Emergency Core Cooling System Capability for Future Plants	R. Emrit	RES/DRA/ARGIB	DROP	-	12/31/88	NA
D-3	Control Rod Drop Accident	R. Emrit	NRR/DSI/CPB	NOTE 3(b)	-	11/30/83	NA

NEW GENERIC ISSUES

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
1	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems	R. Emrit	NRR/DSI/METB	DROP	-	11/30/83	NA
2	Failure of Protective Devices on Essential Equipment	S. Diab	RES/DSIR/EIB	DROP	2	06/30/95	NA
3	Set Point Drift in Instrumentation	R. Emrit	NRR/DSIR/RPSIB	NOTE 3(b)	1	06/30/86	NA
4	End-of-Life and Maintenance Criteria	D. Thatcher	NRR/DE/EQB	NOTE 3(b)	-	11/30/83	NA
5	Design Check and Audit of Balance-of-Plant Equipment	J. Pittman	NRR/DSI/ASB	I.F.1	-	11/30/83	NA
6	Separation of Control Rod from Its Drive and BWR High Rod Worth Events	H. Vandermolen	NRR/DSI/CPB	NOTE 3(b)	1	12/31/94	NA
7	Failures Due to Flow-Induced Vibrations	H. Vandermolen	NRR/DSI/RSB	DROP	1	06/30/91	NA
8	Inadvertent Actuation of Safety Injection in PWRs	L. Riani	NRR/DSI/RSB	I.C.1	-	11/30/83	NA
9	Reevaluation of Reactor Coolant Pump Trip Criteria	R. Emrit	NRR/DSI/RSB	II.K.3(5)	-	11/30/83	NA
10	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges	R. Riggs	NRR/DSI/ICSB	DROP	-	11/30/83	NA
11	Turbine Disc Cracking	J. Pittman	NRR/DE/MTEB	A-37	-	11/30/83	NA
12	BWR Jet Pump Integrity	G. Sege	NRR/DE/MTEB, MEB	NOTE 3(b)	1	12/31/84	NA
13	Small-Break LOCA from Extended Overheating of Pressurizer Heaters	L. Riani	NRR/DSI/RSB	DROP	-	11/30/83	NA
14	PWR Pipe Cracks	R. Emrit	NRR/DE/MTEB	NOTE 3(b)	2	12/31/94	NA
15	Radiation Effects on Reactor Vessel Supports	R. Emrit	RES/DET/EMMEB	NOTE 3(b)	3	06/30/96	NA
16	BWR Main Steam Isolation Valve Leakage Control Systems	W. Milstead	NRR/DSI/ASB	C-8	-	11/30/83	NA
17	Loss of Offsite Power Subsequent to a LOCA	L. Riani	NRR/DSI/PSB, ICSB	DROP	-	11/30/83	NA
18	Steam Line Break with Consequential Small LOCA	R. Riggs	NRR/DSI/RSB	I.C.1	-	11/30/83	NA
19	Safety Implications of Nonsafety Instrument and Control Power Supply Bus	G. Sege	NRR/DST/GIB	A-47	-	11/30/83	NA
20	Effects of Electromagnetic Pulse on Nuclear Power Plants	D. Thatcher	NRR/DSI/ICSB	NOTE 3(b)	1	06/30/84	NA
21	Vibration Qualification of Equipment	R. Riggs	NRR/DE/EIB	DROP	2	06/30/91	NA
22	Inadvertent Boron Dilution Events	H. Vandermolen	NRR/DSI/RSB	NOTE 3(b)	2	12/31/94	NA
23	Reactor Coolant Pump Seal Failures	R. Riggs	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
24	Automatic ECCS Switchover to Recirculation	W. Milstead	RES/DET/GSIB	NOTE 3(b)	3	12/31/95	NA
25	Automatic Air Header Dump on BWR Scram System	W. Milstead	NRR/DSI/RSB	NOTE 3(a)	-	11/30/83	
26	Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power	R. Emrit	NRR/DSI/ASB	17	-	11/30/83	NA
27	Manual vs. Automated Actions	J. Pittman	NRR/DSI/RSB	B-17	-	11/30/83	NA
28	Pressurized Thermal Shock	R. Emrit	NRR/DST/GIB	A-49	-	11/30/83	NA
29	Bolting Degradation or Failure in Nuclear Power Plants	H. Vandermolen	RES/DSIR/EIB	NOTE 3(b)	2	06/30/95	NA
30	Potential Generator Missiles—Generator Rotor	J. Pittman	NRR/DE/MEB	DROP	1	12/31/85	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Retaining Rings						
31	Natural Circulation Cooldown	R. Riggs	NRR/DSI/RSB	I.C.1	-	11/30/83	NA
32	Flow Blockage in Essential Equipment Caused by Corbicula	R. Emrit	NRR/DSI/ASB	51	-	11/30/83	NA
33	Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power	J. Pittman	NRR/DSI/ICSB	A-47	-	11/30/83	NA
34	RCS Leak	R. Riggs	NRR/DHFS/PSRB	DROP	1	06/30/84	NA
35	Degradation of Internal Appurtenances in LWRs	H. Vandermolen	NRR/DSI/CPB, RSB	DROP	2	12/31/98	NA
36	Loss of Service Water	L. Riani	NRR/DSI/ASB, AEB, RSB	NOTE 3(b)	3	06/30/91	NA
37	Steam Generator Overfill and Combined Primary and Secondary Blowdown	L. Riani	NRR/DST/GIB, NRR/DSI/RSB	A-47, I.C.1(2)	1	06/30/85	NA
38	Potential Recirculation System Failure as a Consequence of Ingestion of Containment Paint Flakes or Other Fine Debris	R. Emrit	RES/DSIR/RPSIB	DROP	2	06/30/95	NA
39	Potential for Unacceptable Interaction between the CRD System and Non-Essential Control Air System	J. Pittman	NRR/DSI/ASB	25	1	06/30/95	NA
40	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	L. Riani	NRR/DSI/ASB	NOTE 3(a)	1	06/30/84	B-65
41	BWR Scram Discharge Volume Systems	H. Vandermolen	NRR/DSI/RSB	NOTE 3(a)	-	11/30/83	B-58
42	Combination Primary/Secondary System LOCA	R. Riggs	NRR/DSI/RSB	I.C.1	1	06/30/85	NA
43	Reliability of Air Systems	W. Milstead	RES/DSIR/RPSI	NOTE 3(a)	2	12/31/88	B-107
44	Failure of Saltwater Cooling System	W. Milstead	NRR/DSI/ASB	43	1	12/31/88	NA
45	Inoperability of Instrumentation Due to Extreme Cold Weather	W. Milstead	NRR/DSI/ICSB	NOTE 3(a)	2	06/30/91	
46	Loss of 125 Volt DC Bus	G. Sege	NRR/DSI/PSB	76	-	11/30/83	NA
47	Loss of Offsite Power	D. Thatcher	NRR/DSI/RSB, ASB	NOTE 3(b)	-	11/30/83	
48	LCO for Class 1E Vital Instrument Buses in Operating Reactors	G. Sege	NRR/DSI/PSB	128	1	12/31/86	NA
49	Interlocks and LCOs for Redundant Class 1E Tie-Breakers	G. Sege	NRR/DSI/PSB	128	3	06/30/91	NA
50	Reactor Vessel Level Instrumentation in BWRs	D. Thatcher	NRR/DSI/RSB, ICSB	NOTE 3(b)	1	12/31/84	NA
51	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	R. Emrit	RES/DE/EIB	NOTE 3(a)	1	12/31/89	L-913
52	SSW Flow Blockage by Blue Mussels	R. Emrit	NRR/DSI/ASB	51	-	11/30/83	NA
53	Consequences of a Postulated Flow Blockage Incident in a BWR	H. Vandermolen	NRR/DSI/CPB, RSB	DROP	1	12/31/84	NA
54	Valve Operator-Related Events Occurring during 1978, 1979, and 1980	L. Riani	NRR/DE/MEB	II.E.6.1	1	06/30/85	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
55	Failure of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	R. Emrit	NRR/DSI/PSB	DROP	2	06/30/91	NA
56	Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event	L. Riani	NRR/DHFS/HFEB	A-47, I.D.1	-	11/30/83	NA
57	Effects of Fire Protection System Actuation on Safety-Related Equipment	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)	3	06/30/95	NA
58	Inadvertent Containment Flooding	G. Sege	NRR/DSI/ASB, CSB	DROP	-	11/30/83	
59	Technical Specification Requirements for Plant Shutdown When Equipment for Safe Shutdown is Degraded or Inoperable	R. Emrit	NRR/DST/TSIP	RI (NOTE 5)	1	06/30/85	NA
60	Lamellar Tearing of Reactor Systems Structural Supports	L. Riani	NRR/DST/GIB	A-12	-	11/30/83	NA
61	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments	W. Milstead	NRR/DSI/CSB	NOTE 3(b)	2	12/31/86	NA
62	Reactor Systems Bolting Applications	R. Riggs	RES/DSIR/EIB	29	1	12/31/88	NA
63	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis	J. Pittman	RES/DRA/ARGIB	DROP	1	06/30/90	NA
64	Identification of Protection System Instrument Sensing Lines	D. Thatcher	NRR/DSI/ICSB	NOTE 3(b)	-	11/30/83	
65	Probability of Core-Melt Due to Component Cooling Water System Failures	H. Vandermolen	NRR/DSI/ASB	23	1	12/31/86	NA
66	Steam Generator Requirements	R. Riggs	NRR/DEST/EMTB	NOTE 3(b)	2	12/31/88	NA
<u>67</u>	<u>Steam Generator Staff Actions</u>						
67.2.1	Integrity of Steam Generator Tube Sleeves	R. Riggs	NRR/DE/MEB	135	4	06/30/94	NA
67.3.1	Steam Generator Overfill	R. Riggs	NRR/DST/GIB, NRR/DSI/RSB	A-47, I.C.1	4	06/30/94	NA
67.3.2	Pressurized Thermal Shock	R. Riggs	NRR/DST/GIB	A-49	4	06/30/94	NA
67.3.3	Improved Accident Monitoring	R. Riggs	NRR/DSI/ICSB	NOTE 3(a)	4	06/30/94	A-17
67.3.4	Reactor Vessel Inventory Measurement	R. Riggs	NRR/DSI/CPB	II.F.2	4	06/30/94	NA
67.4.1	RCP Trip	R. Riggs	NRR/DSI/RSB	II.K.3(5)	4	06/30/94	G-01
67.4.2	Control Room Design Review	R. Riggs	NRR/DHFS/HFEB	I.D.1	4	06/30/94	F-08
67.4.3	Emergency Operating Procedures	R. Riggs	NRC/DHFS/PSRB	I.C.1	4	06/30/94	F-05
67.5.1	Reassessment of Radiological Consequences	R. Riggs	RES/DRPS/RPSI	LI (NOTE 3)	4	06/30/94	NA
67.5.2	Reevaluation of SGTR Design Basis	R. Riggs	RES/DRPS/RPSI	LI (67.5.1)	4	06/30/94	NA
67.5.3	Secondary System Isolation	R. Riggs	NRR/DSI/RSB	DROP	4	06/30/94	NA
67.6.0	Organizational Responses	R. Riggs	OIE/DEPER/IRDB	III.A.3	4	06/30/94	NA
67.7.0	Improved Eddy Current Tests	R. Riggs	RES/DE/EIB	135	4	06/30/94	NA
67.8.0	Denting Criteria	R. Riggs	NRR/DE/MTEB	135	4	06/30/94	NA
67.9.0	Reactor Coolant System Pressure Control	R. Riggs	NRR/DSI/GIB,	A-45,	4	06/30/94	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
67.10.0	Supplemental Tube Inspections	R. Riggs	NRR/DSI/RSB NRR/DL/ORAB	I.C.1 (2,3) LI (NOTE 5)	4	06/30/94	NA
68	Postulated Loss of Auxiliary Feedwater System Resulting from Turbine-Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	J. Pittman	NRR/DSI/ASB	124	3	06/30/91	NA
69	Make-up Nozzle Cracking in B&W Plants	R. Colmar	NRR/DE/MEB, MTEB	NOTE 3(b)	1	12/31/84	B43
70	PORV and Block Valve Reliability	R. Riggs	RES/DE/EIB	NOTE 3(a)	3	06/30/91	
71	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety	J. Pittman	RES/DRA/ARGIB	DROP	3	06/30/01	NA
72	Control Rod Drive Guide Tube Support Pin Failures	R. Riggs	RES	DROP	1	06/30/91	NA
73	Detached Thermal Sleeves	R. Emrit	RES/DSIR/EIB	NOTE 3(a)	3	06/30/95	NA
74	Reactor Coolant Activity Limits for Operating Reactors	W. Milstead	NRR/DSI/AEB	DROP	1	06/30/86	NA
75	Generic Implications of ATWS Events at the Salem Nuclear Plant	R. Emrit	RES/DRA/ARGIB	NOTE 3(a)	1	06/30/90	B-76, B-77, B-78, B-79, B-80, B-81, B-82, B-85 B-86, B-87, B-88, B-89, B-90, B-91, B-92, B-93
76	Instrumentation and Control Power Interactions	R. Zimmerman	RES/DSIR/EIB	DROP	3	06/30/95	NA
77	Flooding of Safety Equipment Compartments by Backflow Through Floor Drains	L. Riani	RES/DE/EIB	A-17	-	12/31/87	NA
78	Monitoring of Fatigue Transient Limits for Reactor Coolant System	C. Rourk	RES/DET/GSIB	NOTE 3(b)	3	12/31/97	
79	Unanalyzed Reactor Vessel Thermal Stress during Natural Convection Cooldown	L. Riani	RES/DSIR/EIB	NOTE 3(b)	3	06/30/95	NA
80	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments	H. Vandermolen	RES/DSARE/REAHFB	NOTE 3(b)	4	06/30/06	NA
81	Impact of Locked Doors and Barriers on Plant and	C. Rourk	RES/DSIR/EIB	LOW	4	06/30/95	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Personnel Safety						
82	Beyond Design-Basis Accidents in Spent Fuel Pools	H. Vandermolen	RES/DRPS/RPSI	NOTE 3(b)	3	06/30/04	NA
83	Control Room Habitability	R. Emrit	RES/DST/AEB	NOTE 3(b)	3	06/30/03	NA
84	CE PORVs	R. Riggs	RES/DSIR/RPSI	NOTE 3(b)	2	06/30/90	NA
85	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	W. Milstead	NRR/DSI/CSB	DROP	2	06/30/91	NA
86	Long-Range Plan for Dealing with Stress-Corrosion Cracking in BWR Piping	R. Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	06/30/88	B-84
87	Failure of HPCI Steam Line without Isolation	J. Pittman	RES/DSIR/EIB	NOTE 3(a)	2	06/30/95	
88	Earthquakes and Emergency Planning	R. Riggs	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
89	Stiff Pipe Clamps	T.Y. Chang	RES/DSIR/EIB	MEDIUM	2	06/30/95	NA
90	Technical Specifications for Anticipatory Trips	H. Vandermolen	NRR/DSI/RSB, ICSB	DROP	2	12/31/98	NA
91	Main Crankshaft Failures in Transamerica Delaval Emergency Diesel Generators	R. Emrit	RES/DRA/ARGIB	NOTE 3(b)	-	12/31/87	NA
92	Fuel Crumbling during LOCA	H. Vandermolen	NRR/DSI/RSB, CPB	DROP	1	12/31/98	NA
93	Steam Binding of Auxiliary Feedwater Pumps	J. Pittman	RES/DRPS/RPSI	NOTE 3(a)	-	06/30/88	B-98
94	Additional Low Temperature Overpressure Protection for Light-Water Reactors	J. Pittman	RES/DSIR/RPSI	NOTE 3(a)	-	06/30/90	
95	Loss of Effective Volume for Containment Recirculation Spray	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)	-	06/30/90	NA
96	RHR Suction Valve Testing	W. Milstead	RES/DRA/ARGIB	105	-	06/30/90	NA
97	PWR Reactor Cavity Uncontrolled Exposures	H. Vandermolen	NRR/DSI/RAB	III.D.3.1	-	06/30/85	NA
98	CRD Accumulator Check Valve Leakage	J. Pittman	NRR/DSI/ASB	DROP	-	06/30/85	NA
99	RCS/RHR Suction Line Valve Interlock on PWRs	J. Pittman	RES/DRPS/RPSI	NOTE 3(a)	3	06/30/91	L-817
100	Once-Through Steam Generator Level	J. Jackson	RES/DSIR/EIB	DROP	1	06/30/95	NA
101	BWR Water Level Redundancy	H. Vandermolen	RES/DE/EIB	NOTE 3(b)	1	06/30/89	NA
102	Human Error in Events Involving Wrong Unit or Wrong Train	R. Emrit	NRR/DLPQ/LPEB	NOTE 3(b)	2	12/31/88	NA
103	Design for Probable Maximum Precipitation	R. Emrit	RES/DE/EIB	NOTE 3(a)	1	12/31/89	NA
104	Reduction of Boron Dilution Requirements	J. Pittman	RES/DRA/ARGIB	DROP	-	12/31/88	NA
105	Interfacing Systems LOCA at LWRs	W. Milstead	RES/DE/EIB	NOTE 3(b)	4	06/30/95	NA
106	Piping and Use of Highly Combustible Gases in Vital Areas	W. Milstead	RES/DRPS	NOTE 3(b)	2	06/30/95	NA
107	Main Transformer Failures	W. Milstead	RES/DRA/ARGIB	DROP	3	06/30/00	NA
108	BWR Suppression Pool Temperature Limits	L. Riani	NRR/DSI/CSB	RI (NOTE 3)	-	06/30/85	NA
109	Reactor Vessel Closure Failure	R. Riggs	RES/DRA/ARGIB	DROP	-	06/30/90	NA
110	Equipment Protective Devices on Engineered Safety Features	S. Diab	RES/DSIR/EIB	DROP	1	06/30/95	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
111	Stress-Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments	R. Riggs	NRR/DE/MTEB	LI (NOTE 5)	1	06/30/91	NA
112	Westinghouse RPS Surveillance Frequencies and Out-of-Service Times	J. Pittman	NRR/DSI/ICSB	RI (NOTE 3)	-	12/31/85	NA
113	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	R. Riggs	RES/DSIR/EIB	NOTE 3(b)	2	06/30/95	NA
114	Seismic-Induced Relay Chatter	R. Riggs	NRR/DSRO/SPEB	A-46	1	06/30/91	NA
115	Enhancement of the Reliability of Westinghouse Solid State Protection System	W. Milstead	RES/DRPS/RPSI	NOTE 3(b)	2	06/30/00	NA
116	Accident Management	J. Pittman	RES/DRA/ARGIB	S	-	06/30/91	NA
117	Allowable Time for Diverse Simultaneous Equipment Outages	J. Pittman	RES/DRA/ARGIB	DROP	-	06/30/90	NA
118	Tendon Anchorage Failure	S. Shaukat	RES/DSIR/EIB	NOTE 3(a)	1	06/30/95	NA
<u>119</u>	<u>Piping Review Committee Recommendations</u>						
119.1	Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads	R. Riggs	NRR/DE	RI (NOTE 3)	3	12/31/97	NA
119.2	Piping Damping Values	R. Riggs	NRR/DE	RI (DROP)	3	12/31/97	NA
119.3	Decoupling the OBE from the SSE	R. Riggs	NRR/DE	RI (S)	3	12/31/97	NA
119.4	BWR Piping Materials	R. Riggs	NRR/DE	RI (NOTE 5)	3	12/31/97	NA
119.5	Leak Detection Requirements	R. Riggs	NRR/DE	RI (NOTE 5)	3	12/31/97	NA
120	On-Line Testability of Protection Systems	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)	2	06/30/95	NA
121	Hydrogen Control for Large, Dry PWR Containments	R. Emrit	RES/DSIR/SAIB	NOTE 3(b)	2	06/30/95	NA
<u>122</u>	<u>Davis-Besse Loss of All Feedwater Event of June 9, 1985: Short-Term Actions</u>						
122.1	Potential Inability to Remove Reactor Decay Heat						
122.1.a	Failure of Isolation Valves in Closed Position	H. Vandermolen	NRR/DSRO/RSIB	124	4	12/31/98	NA
122.1.b	Recovery of Auxiliary Feedwater	H. Vandermolen	NRR/DSRO/RSIB	124	4	12/31/98	NA
122.1.c.	Interruption of Auxiliary Feedwater Flow	H. Vandermolen	NRR/DSRO/RSIB	124	4	12/31/98	NA
122.2	Initiating Feed-and-Bleed	H. Vandermolen	NRR/DEST/SRXB	NOTE 3(b)	4	12/31/98	NA
122.3	Physical Security System Constraints	H. Vandermolen	NRR/DSRO/SPEB	DROP	4	12/31/98	NA
123	Deficiencies in the Regulations Governing DBA and Single-Failure Criteria Suggested by the Davis-Besse Event of June 9, 1985	W. Milstead	RES/DSIR/SAIB	DROP	1	06/30/95	NA
124	Auxiliary Feedwater System Reliability	R. Emrit	NRR/DEST/SRXB	NOTE 3(a)	3	06/30/91	

Table II (continued)

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

Table II (continued)

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

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
144	Scram without a Turbine/Generator Trip	C. Hrabal	RES/DSIR/EIB	DROP	2	12/31/98	NA
145	Actions to Reduce Common-Cause Failures	D. Rasmuson	RES/DST/PRAB	NOTE 3(b)	3	06/30/00	NA
146	Support Flexibility of Equipment and Components	T. Y. Chang	RES/DSIR/EIB	NOTE 3(b)	2	06/30/95	NA
147	Fire-Induced Alternate Shutdown/Control Room Panel Interactions	W. Milstead	RES/DSIR/SAIB	LI (NOTE 3)	1	06/30/94	NA
148	Smoke Control and Manual Fire-Fighting Effectiveness	D. Basdekas	RES/DSIR/RPSIB	LI (NOTE 3)	1	06/30/00	NA
149	Adequacy of Fire Barriers	R. Emrit	RES/DSIR/EIB	DROP	2	12/31/98	NA
150	Overpressurization of Containment Penetrations	W. Milstead	RES/DSIR/SAIB	DROP	1	06/30/95	NA
151	Reliability of Anticipated Transient without SCRAM Recirculation Pump Trip in BWRs	W. Milstead	RES/DSIR/SAIB	NOTE 3(b)	2	06/30/95	NA
152	Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads	R. Emrit	RES/DSIR/EIB	DROP	3	06/30/01	NA
153	Loss of Essential Service Water in LWRs	R. Riggs	RES/DRA/ARGIB	NOTE 3(b)	2	12/31/95	NA
154	Adequacy of Emergency and Essential Lighting	R. Woods	RES/DSIR/SAIB	DROP	2	12/31/98	NA
<u>155</u>	<u>Generic Concerns Arising from TMI-2 Cleanup</u>						
155.1	More Realistic Source Term Assumptions	R. Emrit	RES/DST/AEB	NOTE 3(a)	2	06/30/95	NA
155.2	Establish Licensing Requirements for Non-Operating Facilities	R. Emrit	RES/DSIR/EIB	RI (NOTE 5)	2	06/30/95	NA
155.3	Improve Design Requirements for Nuclear Facilities	R. Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.4	Improve Criticality Calculations	R. Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.5	More Realistic Severe Reactor Accident Scenario	R. Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.6	Improve Decontamination Regulations	R. Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
155.7	Improve Decommissioning Regulations	R. Emrit	RES/DSIR/EIB	DROP	2	06/30/95	NA
<u>156</u>	<u>Systematic Evaluation Program</u>						
156.1.1	Settlement of Foundations and Buried Equipment	T.Y. Chang	RES/DSIR/EIB	DROP	8	06/30/08	NA
156.1.2	Dam Integrity and Site Flooding	J. Chen	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.1.3	Site Hydrology and Ability to Withstand Floods	J. Chen	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.1.4	Industrial Hazards	C. Ferrell	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.1.5	Tornado Missiles	J. Chen	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.1.6	Turbine Missiles	R. Emrit	RES/DSIR/EIB	DROP	8	06/30/08	NA
156.2.1	Severe Weather Effects on Structures	J. Chen	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.2.2	Design Codes, Criteria, and Load Combinations	R. Kirkwood	RES/DSIR/EIB	DROP	8	06/30/08	NA
156.2.3	Containment Design and Inspection	S. Shaukat	RES/DSIR/EIB	DROP	8	06/30/08	NA
156.2.4	Seismic Design of Structures, Systems, and Components	J. Chen	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.3.1.1	Shutdown Systems	R. Woods	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.3.1.2	Electrical Instrumentation and Controls	R. Woods	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.3.2	Service and Cooling Water Systems	N. Su	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.3.3	Ventilation Systems	G. Burdick	RES/DSIR/SAIB	DROP	8	06/30/08	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
156.3.4	Isolation of High- and Low-Pressure Systems	G. Burdick	RES/DSIR/SAIB	DROP	8	06/30/08	NA
156.3.5	Automatic ECCS Switchover	W. Milstead	RES/DSIR/SAIB	24	8	06/30/08	NA
156.3.6.1	Emergency AC Power	R. Emrit	RES/DSIR/EIB	DROP	8	06/30/08	NA
156.3.6.2	Emergency DC Power	C. Rourk	RES/DSIR/EIB	DROP	8	06/30/08	NA
156.3.8	Shared Systems	R. Emrit	RES/DSIR/EIB	DROP	8	06/30/08	NA
156.4.1	RPS and ESFS Isolation	R. Emrit	RES/DSIR/EIB	142	8	06/30/08	NA
156.4.2	Testing of the RPS and ESFS	T.Y. Chang	RES/DSIR/SAIB	120	8	06/30/08	NA
156.6.1	Pipe Break Effects on Systems and Components	H. Vandermolen	RES/DRA/OEGIB	NOTE 3(b)	8	06/30/08	NA
157	Containment Performance	J. Shaperow	RES/DSIR/SAIB	NOTE 3(b)	-	06/30/95	NA
158	Performance of Power-Operated Valves Under Design Basis Conditions	C. Hrabal	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
159	Qualification of Safety-Related Pumps While Running on Minimum Flow	N. Su	RES/DSIR/SAIB	DROP	1	06/30/95	NA
160	Spurious Actions of Instrumentation Upon Restoration of Power	C. Rourk	RES/DSIR/EIB	DROP	1	06/30/95	NA
161	Use of Non-Safety-Related Power Supplies in Safety-Related Circuits	C. Rourk	RES/DSIR/EIB	DROP	1	06/30/95	NA
162	Inadequate Technical Specifications for Shared Systems at Multiplant Sites When One Unit Is Shut Down	U. Cheh	RES/DSIR/SAIB	DROP	1	06/30/95	NA
163	Multiple Steam Generator Tube Leakage	E. Murphy	NRR/DCI/CSG	NOTE 3(b)	2	06/30/10	
164	Neutron Fluence in Reactor Vessel	R. Emrit	RES/DSIR/EIB	DROP	1	06/30/95	NA
165	Safety and Safety/Relief Valve Reliability	C. Hrabal	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
166	Adequacy of Fatigue Life of Metal Components	R. Emrit	NRR/DE/EMEB	NOTE 3(b)	2	12/31/97	NA
167	Hydrogen Storage Facility Separation	G. Burdick	RES/DSIR/SAIB	LOW	1	06/30/95	NA
168	Environmental Qualification of Electrical Equipment	R. Emrit	NRR/DSSA/SPLB	NOTE 3(b)	3	06/30/04	NA
169	BWR MSIV Common Mode Failure Due to Loss of Accumulator Pressure	R. Emrit	RES/DET/GSIB	DROP	1	06/30/00	NA
170	Fuel Damage Criteria for High Burnup Fuel	R. Emrit	RES/DET/GSIB	NOTE 3(b)	2	06/30/01	NA
171	ESF Failure from LOOP Subsequent to a LOCA	C. Rourk	RES/DET/GSIB	NOTE 3(b)	1	12/31/98	NA
172	Multiple System Responses Program	R. Emrit	RES/DET/GSIB	NOTE 3(b)	2	06/30/02	NA
<u>173</u>	<u>Spent Fuel Storage Pool</u>						
173.A	Operating Facilities	R. Emrit	RES/DET/GSIB	NOTE 3(b)	4	06/30/02	NA
173.B	Permanently Shutdown Facilities	R. Emrit	RES/DET/GSIB	NOTE 3(b)	4	06/30/02	NA
<u>174</u>	<u>Fastener Gaging Practices</u>						
174.A	SONGS Employees' Concern	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
174.B	Johnson Gage Company Concern	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
175	Nuclear Power Plant Shift Staffing	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
176	Loss of Fill-Oil in Rosemount Transmitters	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
177	Vehicle Intrusion at TMI	R. Emrit	RES/DET/GSIB	NOTE 3(a)	1	06/30/00	NA
178	Effect of Hurricane Andrew on Turkey Point	R. Emrit	RES/DET/GSIB	LI (NOTE 3)	2	06/30/00	
179	Core Performance	R. Emrit	RES/DET/GSIB	LI (NOTE 5)	1	06/30/00	
180	Notice of Enforcement Discretion	R. Emrit	RES/DET/GSIB	LI (NOTE 3)	1	06/30/00	
181	Fire Protection	R. Emrit	RES/DET/GSIB	LI (NOTE 5)	1	06/30/00	
182	General Electric Extended Power Uprate	R. Emrit	RES/DET/GSIB	RI (NOTE 5)	1	06/30/00	
183	Cycle-Specific Parameter Limits in Technical Specifications	R. Emrit	RES/DET/GSIB	RI (NOTE 3)	2	06/30/00	
184	Endangered Species	R. Emrit	RES/DET/GSIB	EI (NOTE 5)	1	06/30/00	
185	Control of Recriticality Following Small-Break LOCA in PWRs	H. Vandermolen	RES/DSARE/REAHFB	NOTE 3(b)	1	06/30/06	NA
186	Potential Risk and Consequences of Heavy Load Drops	S. Jones	NRR/DSS/SBP	ACTIVE	-	06/30/04	
187	The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump in Nuclear Power Plants	H. Vandermolen	RES/DSARE/REAHFB	DROP	-	06/30/01	NA
188	Steam Generator Tube Leaks/Ruptures Concurrent with Containment Bypass	H. Vandermolen	RES/DSARE/REAHFB	NOTE 3(b)	1	06/30/06	NA
189	Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Combustion during a Severe Accident	S. Jones	NRR/DSS/SBP	ROI	1	06/30/08	
190	Fatigue Evaluation of Metal Components for 60-Year Plant Life	S. Shaukat	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
191	Assessment of Debris Accumulation on PWR Sump Performance	M. Scott	NRR/DSS/SSIB	ROI	2	06/30/08	
192	Secondary Containment Drawdown Time	H. Vandermolen	RES/DSARE/REAHFB	DROP	-	06/30/03	NA
193	BWR ECCS Suction Concerns	J. Lane	RES/DRA/OEGIB	ACTIVE	-	06/30/04	
194	Implications of Updated Probabilistic Seismic Hazard Estimates	D. Harrison	NRR/DSSA/SPSB	DROP	-	06/30/04	NA
195	Hydrogen Combustion in Foreign BWR Piping	H. Vandermolen	RES/DSARE/REAHFB	DROP	-	06/30/04	NA
196	Boral Degradation	H. Vandermolen	RES/DSARE/ARREB	NOTE 3(b)	1	06/30/07	NA
197	Iodine Spiking Phenomena	H. Vandermolen	RES/DSARE/ARREB	DROP	-	06/30/06	NA
198	Hydrogen Combustion in PWR Piping	H. Vandermolen	RES/DRASP/OERA	DROP	-	06/30/07	NA
199	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States	J. Kauffman	RES/DRA/OEGIB	ACTIVE	-	06/30/08	
200	Tin Whiskers	C. Antonescu	RES/DRASP/OERA	DROP	-	06/30/07	NA
201	Small-Break LOCA and Loss of Offsite Power Scenario	A. Salomon	RES/DRASP/OERA	DROP	-	06/30/07	NA
202	Spent Fuel Pool Leakage Limits	T. Mitts	RES/DRASP/OERA	DROP	-	06/30/07	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
203	Potential Safety Issues with Cranes that Lift Spent Fuel Casks	T. Mitts	RES/DRASP/OERA	DROP	-	06/30/07	NA
	HUMAN FACTORS ISSUES						
HF1	STAFFING AND QUALIFICATIONS						
HF1.1	Shift Staffing	J. Pittman	RES/DRPS/RHFB	NOTE 3(a)	2	06/30/89	
HF1.2	Engineering Expertise on Shift	J. Pittman	NRR/DHFT/HFIB	NOTE 3(b)	2	06/30/89	NA
HF1.3	Guidance on Limits and Conditions of Shift Work	J. Pittman	NRR/DHFT/HFIB	NOTE 3(b)	2	06/30/89	NA
HF2	TRAINING						
HF2.1	Evaluate Industry Training	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF2.2	Evaluate INPO Accreditation	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF2.3	Revise SRP Section 13.2	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF3	OPERATOR LICENSING EXAMINATIONS						
HF3.1	Develop Job Knowledge Catalog	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF3.2	Develop License Examination Handbook	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF3.3	Develop Criteria for Nuclear Power Plant Simulators	J. Pittman	NRR/DHFT/HFIB	I.A.4.2(4)	2	12/31/87	NA
HF3.4	Examination Requirements	J. Pittman	NRR/DHFT/HFIB	I.A.2.6(1)	2	12/31/87	NA
HF3.5	Develop Computerized Exam System	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF4	PROCEDURES						
HF4.1	Inspection Procedure for Upgraded Emergency Operating Procedures	J. Pittman	NRR/DLPQ/LHFB	NOTE 3(b)	6	06/30/95	NA
HF4.2	Procedures Generation Package Effectiveness Evaluation	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	6	06/30/95	NA
HF4.3	Criteria for Safety-Related Operator Actions	J. Pittman	NRR/DHFT/HFIB	B-17	6	06/30/95	NA
HF4.4	Guidelines for Upgrading Other Procedures	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	6	06/30/95	NA
HF4.5	Application of Automation and Artificial Intelligence	J. Pittman	NRR/DHFT/HFIB	HF5.2	6	06/30/95	NA
HF5	MAN-MACHINE INTERFACE						
HF5.1	Local Control Stations	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	4	06/30/95	NA
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	4	06/30/95	NA
HF5.3	Evaluation of Operational Aid Systems	J. Pittman	NRR/DHFT/HFIB	HF5.2	4	06/30/95	NA
HF5.4	Computers and Computer Displays	J. Pittman	NRR/DHFT/HFIB	HF5.2	4	06/30/95	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
HF6	MANAGEMENT AND ORGANIZATION						
HF6.1	Develop Regulatory Position on Management and Organization	J. Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF6.2	Regulatory Position on Management and Organization at Operating Reactors	J. Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF7	HUMAN RELIABILITY						
HF7.1	Human Error Data Acquisition	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.2	Human Error Data Storage and Retrieval	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.3	Reliability Evaluation Specialist Aids	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.4	Safety Event Analysis Results Applications	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF8	Maintenance and Surveillance Program	J. Pittman	NRR/DLPQ/LPEB	NOTE 3(b)	2	06/30/88	NA
	CHERNOBYL ISSUES						
CH1	ADMINISTRATIVE CONTROLS AND OPERATIONAL PRACTICES						
CH1.1	Administrative Controls To Ensure That Procedures Are Followed and That Procedures Are Adequate						
CH1.1A	Symptom-Based EOPs	R. Emrit	NRR/DLPQ/LHFB	LI (NOTE 5)	-	06/30/89	NA
CH1.1B	Procedure Violations	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)	-	06/30/89	NA
CH1.2	Approval of Tests and Other Unusual Operations						
CH1.2A	Test, Change, and Experiment Review Guidelines	R. Emrit	NRR/DOEA/OTSB	LI (NOTE 5)	-	06/30/89	NA
CH1.2B	NRC Testing Requirements	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)	-	06/30/89	NA
CH1.3	Bypassing Safety Systems						
CH1.3A	Revise Regulatory Guide 1.47	R. Emrit	RES/DE/EMEB	LI (NOTE 5)	-	06/30/89	NA
CH1.4	Availability of Engineered Safety Features						
CH1.4A	Engineered Safety Feature Availability	R. Emrit	NRR/DOEA/OTSB	LI (NOTE 5)	-	06/30/89	NA
CH1.4B	Technical Specifications Bases	R. Emrit	NRR/DOEA/OTSB	LI (NOTE 5)	-	06/30/89	NA
CH1.4C	Low Power and Shutdown	R. Emrit	RES/DSR/PRAB	LI (NOTE 5)	-	06/30/89	NA
CH1.5	Operating Staff Attitudes Toward Safety	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)	-	06/30/89	NA

Table II (continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Status/Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
CH1.6	Management Systems						
CH1.6A	Assessment of NRC Requirements on Management	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.7	Accident Management						
CH1.7A	Accident Management	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH2	DESIGN						
CH2.1	Reactivity Accidents						
CH2.1A	Reactivity Transients	R. Emrit	RES/DSR/RPSB	LI (NOTE 5)		06/30/89	NA
CH2.2	Accidents at Low Power and at Zero Power	R. Emrit	RES/DRA/ARGIB	CH1.4		06/30/89	NA
CH2.3	Multiple-Unit Protection						
CH2.3A	Control Room Habitability	R. Emrit	RES/DRA/ARGIB	83		06/30/89	NA
CH2.3B	Contamination Outside Control Room	R. Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.3C	Smoke Control	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH2.3D	Shared Shutdown Systems	R. Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.4	Fire Protection						
CH2.4A	Firefighting with Radiation Present	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH3	CONTAINMENT						
CH3.1	Containment Performance during Severe Accidents						
CH3.1A	Containment Performance	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH3.2	Filtered Venting						
CH3.2A	Filtered Venting	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4	EMERGENCY PLANNING						
CH4.1	Size of the Emergency Planning Zones	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH4.2	Medical Services	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH4.3	Ingestion Pathway Measures						
CH4.3A	Ingestion Pathway Protective Measures	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4.4	Decontamination and Relocation						
CH4.4A	Decontamination	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA

Table II (continued)

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

TABLE III

SUMMARY OF THE STATUS OF ALL GENERIC SAFETY ISSUES

Legend

ACTIVE	Generic issue that involves actions under the GIP
DROP	Issue dropped from further pursuit as a generic issue
EI	Environmental issue
GSI	Generic safety issue
I	Resolved Three Mile Island Action Plan item with implementation of resolution mandated by NUREG-0737
LI	Licensing issue
LOW	Low safety priority (discontinued December 4, 2001)
MEDIUM	Medium safety priority (discontinued December 4, 2001)
NOTE 3(a)	Resolution resulted in establishment of new regulatory requirements (by rule, SRP change, or equivalent)
NOTE 3(b)	Resolution resulted in no new requirements
NOTE 5	Issue that is not a generic safety issue but should be assigned resources for completion (discontinued June 30, 2010)
RI	Regulatory impact issue
ROI	Regulatory office implementation: A formal GI for which RES actions of safety/risk assessment or regulatory assessment are complete and remaining actions reside with program offices (e.g., regulatory compliance, reactor oversight process, rulemaking, further research, coordination with industry initiatives)
S	Issue covered in an NRC program outside the scope of this document
USI	Unresolved safety issue

TABLE III (continued)

ACTION ITEM / ISSUE GROUP	I	S	LEGACY GIP PROCESS			ACTIVE	ROI	ACTIONS COMPLETED			Total
			NOTE 5	MEDIUM	LOW			NOTE 3(a)	NOTE 3(b)	DROP	
TMI ACTION PLAN ITEM (369)											
GSI	84	46	-	-	10	-	-	66	69	11	286
LI	-	-	8	-	-	-	-	75		-	83
TASK ACTION PLAN ITEMS (142)											
USI	-	-	-	-	-	-	-	27		-	27
GSI	-	20	-	-	-	-	-	36		14	70
RI	-	-	1	-	-	-	-	6		-	7
LI	-	-	12	-	-	-	-	11		-	23
EI	-	-	2	-	-	-	-	13		-	15
NEW GENERIC ISSUES (283)											
GSI	-	54	-	1	3	3	2	23	66	105	257
RI	-	1	5	-	-	-	-	5		1	12
LI	-	1	4	-	-	-	-	8		-	13
EI	-	-	1	-	-	-	-	-		-	1
HUMAN FACTORS ISSUES (27)											
GSI	-	8	-	-	-	-	-	1	7	-	16
LI	-	-	8	-	-	-	-	3		-	11
CHERNOBYL ISSUES (32)											
LI	-	2	23	-	-	-	-	7		-	32
TOTAL:	**84**	**132**	**64**	**1**	**13**	**3**	**2**	**423**		**131**	**853**

ISSUE 163: MULTIPLE STEAM GENERATOR TUBE LEAKAGE

Issue Identification

The NRC identified[1031] this issue in June 1992 to address an NRC staff member's concern, given in a DPO dated December 3, 1991,[1936] and modified March 27, 1992,[1937] about the potential for a main steamline break (MSLB) accident to cause significant primary-to-secondary leakage that could damage the reactor core. The DPO was prompted by widespread outer-diameter stress-corrosion cracking (ODSCC) at the steam generator tube support plates (TSPs) at the Trojan Nuclear Power Plant, which the DPO author claimed could not be reliably detected, and by the staff's approval of alternate repair criteria (ARC) that would allow many tubes known to contain such cracks to remain in service.

In accordance with NRC Management Directive 6.4, "Generic Issues Program," dated November 17, 2009,[1858] the staff screened the issue and classified it as GSI-163 on June 16, 1992.[1031] The principal assertion addressed by GSI-163 was the potential for multiple steam generator (SG) tube leaks during an MSLB that cannot be isolated outside containment to lead to core damage that could result from the loss of all primary system coolant and safety injection fluid from the refueling water storage tank.

The intent of GSI-163 was to address the adequacy of regulatory requirements relating to the management of SG tube integrity to ensure that all tubes will continue to exhibit acceptable structural margins against burst or rupture under normal operating conditions, as well as during postulated design-basis accidents (DBAs) (including MSLB), and that leakage from one or multiple tubes during postulated DBAs will be limited to very small amounts, consistent with the applicable regulations for offsite and control room doses. In contrast, any actions needed to address containment bypass scenarios due to tube failure during severe accidents would likely involve changes to accident management procedures and, perhaps, hardware modifications not involving the steam generators and, therefore, were outside the scope of GSI-163. Similarly, iodine spiking and radiological assessment issues were outside the scope of GSI-163. DPO issues outside the scope of GSI-163 were managed under the SG Action Plan umbrella.

Importance to Safety

The SG tubes function as an integral part of the reactor coolant pressure boundary (RCPB) and, in addition, isolate radioactive fission products in the primary reactor coolant from the secondary coolant and the environment. Thus, the SG tubing serves a containment function as well as an RCPB function. SG tube leakage (i.e., primary-to-secondary leakage) or ruptures have a number of potential safety implications, including those associated with allowing fission products in the primary coolant to escape into the environment through the secondary system. In the event of an MSLB accident or stuck open SG safety valve, leakage of primary coolant through the tubes could contaminate the flow out of the ruptured steamline or safety valve, respectively. In addition, leakage of primary coolant through the SG tubing could deplete the inventory of water available for long-term cooling of the core in the event of an accident.

Regulatory Framework for Ensuring Steam Generator Tube Integrity

Title 10 of the *Code of Federal Regulations* (10 CFR), "Energy," establishes the fundamental regulatory requirements for the integrity of the SG tubes. Specifically, the general design criteria

(GDC) in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," state that the RCPB—

- shall have "an extremely low probability of abnormal leakage...and gross rupture" (GDC 14, "Reactor Coolant Pressure Boundary")
- "shall be designed with sufficient margin" (GDCs 15, "Reactor Coolant System Design," and 31, "Fracture Prevention of Reactor Coolant Pressure Boundary")
- shall be of "the highest quality standards practical" (GDC 30, "Quality of Reactor Coolant Pressure Boundary")
- shall be designed to permit "periodic inspection and testing...to assess...structural and leak-tight integrity" (GDC 32, "Inspection of Reactor Coolant Pressure Boundary")

To this end, 10 CFR 50.55a, "Codes and Standards," specifies that components that are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code).[1938] In 10 CFR 50.55a, the NRC further requires that, throughout the service life of a PWR facility, ASME Code Class 1 components meet the requirements (except for the design and access provisions and preservice examination requirements) in Section XI, "Rules for Inservice Inspection [ISI] of Nuclear Power Plant Components," of the ASME Code, to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code. Section XI requirements pertaining to ISI of SG tubing are augmented by additional requirements in the plant technical specifications (TS).

As part of the plant licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated DBAs, such as an SG tube rupture (SGTR) and MSLB. These analyses consider primary-to-secondary leakage that may occur during these events and must show that the offsite radiological consequences do not exceed the applicable limits of 10 CFR 50.67, "Accident Source Term," or 10 CFR Part 100, "Reactor Site Criteria," for offsite doses, GDC 19, "Control Room," criteria for control room operator doses (or some fraction thereof as appropriate to the accident), or the NRC-approved licensing basis (e.g., a small fraction of these limits).

Operating experience has proven that SG tubing is subject to a variety of mechanically and corrosion-induced degradation mechanisms that may impair the structural and leakage integrity of the SG tubing. The licensee's plant TS require the implementation of SG tube surveillance programs to ensure that tubes are repaired, or removed from service by plugging the tube ends, before the structural or leakage integrity of the tubes is impaired. The TS include a generally applicable depth-based tube repair limit, typically 40 percent of the nominal tube wall thickness, beyond which the tubes must be repaired or plugged. This depth-based tube repair limit is intended to ensure that tubes accepted for continued service will not leak and will retain safety factors against burst consistent with the design basis (i.e., the stress limits in the ASME Code, Section III[1938]) with allowance for flaw depth measurement uncertainty and for incremental flaw growth before the next scheduled inspection. The plant TS also include a limit on operational primary-to-secondary leakage, typically 150 gallons per day, beyond which the plant must be promptly shutdown.

Prioritization and Regulatory Assessment

The NRC gave the issue a HIGH priority ranking in 1997.[1091] The NRC originally planned to develop a rule involving a more flexible and more effective regulatory framework for SG tube surveillance and maintenance activities (compared with TS requirements existing at that time) that would allow a degradation-specific management approach. The staff discontinued this effort in 1997 after a regulatory analysis indicated that rulemaking was unnecessary. With Commission approval, the staff began to develop a generic letter requesting that all PWR licensees submit proposed changes to their plant TS that would ensure SG tube integrity is maintained. This generic letter initiative included a draft regulatory guide and sample TS incorporating a programmatic, performance based strategy for ensuring SG tube integrity.

On December 1, 1997, the industry informed the NRC staff of an industry initiative, Nuclear Energy Institute (NEI) 97-06, "Steam Generator Tube Integrity Guidelines,"[1939] which paralleled the draft regulatory guide and which all PWR licensees had committed (among themselves) to implement. NEI 97-06[1939] provided a programmatic, performance-based approach to ensuring SG tube integrity. With Commission approval, the staff put the generic letter initiative on hold and worked with the industry to identify revised TS that would be aligned with the NEI 97-06[1939] initiative and that would ensure all PWR licensees implement programs to ensure that SG tube integrity will be maintained. This effort was completed in May 2005 with the NRC staff's approval of Technical Specification Task Force (TSTF)-449, Revision 4, "Steam Generator Tube Integrity," dated April 14, 2005,[1897] which included a new standard TS template governing SG tube integrity. In response to NRC Generic Letter 2006-01, "Steam Generator Tube Integrity and Associated Technical Specifications," dated January 20, 2006,[1901] all PWR licensees submitted license amendment applications to change their TS in accordance with TSTF-449.[1897]

The nature of the DPO evolved considerably in the years after 1991, adding additional concerns related to alternate tube repair criteria, iodine spiking assumptions for radiological analysis, severe accidents, and many other concerns. The staff prepared a DPO consideration document and provided it to the NRC's Executive Director for Operations (EDO) on September 1, 1999. At the EDO's request, the ACRS served as an equivalent ad hoc panel to review the DPO issues. The ACRS met with the DPO author and other members of the NRC staff and reviewed the documentation related to the DPO issues. The ACRS issued NUREG-1740, "Voltage-Based Alternative Repair Criteria,"[1898] on February 1, 2001, documenting its conclusions and recommendations. By memorandum[1899] dated May 11, 2001, the Office of Nuclear Reactor Regulation and the Office of Nuclear Regulatory Research developed a joint action plan to address the conclusions and recommendations in the ACRS report. This action plan and resolution of GSI-163 was later incorporated into the NRC's Steam Generator Action Plan (SGAP).[1899, 1940] The status of the SGAP was presented to the Commission in SECY-03-0080, "Steam Generator Tube Integrity (SGTI)—Plans for Revising the Associated Regulatory Framework," dated May 16, 2003,[1900] and discussed at a Commission meeting on May 19, 2003. In a memorandum[1941] to the DPO author dated March 5, 2001, the EDO stated that the NRC concluded that the concerns raised in the DPO were dispositioned and the DPO closed on the basis of the following three points:

(1) the ACRS ad hoc subcommittee's finding that the ARC and condition monitoring program can adequately protect public health and safety

(2) the ACRS ad hoc subcommittee's conclusion that no immediate regulatory actions were necessary

(3) the NRC staff's development of an SGAP[1899, 1940] to address the conclusions and recommendations in the ACRS ad hoc subcommittee's report

By memorandum from B. Sheron to L. Reyes dated July 5, 2007,[1942] GSI-163 was closed in the Generic Issues Program and was transferred to the Office of Nuclear Reactor Regulation for regulatory office implementation.

As of September 30, 2007, new performance-based TS requirements were in place at all U.S. PWRs. These requirements were the culmination of years of work between the NRC staff and the industry to develop a generic template for new TS requirements incorporating a programmatic, performance-based approach for ensuring SG tube integrity (70 FR 24126; May 6, 2005).[1943] Each PWR licensee adopted the new TS requirements voluntarily, consistent with the generic template, and not as the result of an NRC backfit. These requirements are intended to ensure that all tubes exhibit adequate structural margins against burst or rupture for the spectrum of normal operating and DBA conditions, consistent with the original design basis. These requirements are also intended to ensure that total leakage from tubes at a plant will not exceed values assumed in licensing-basis accident analyses even if no tubes actually rupture under these conditions. In addition, licensees are required to periodically demonstrate that these structural margin and accident leakage criteria are satisfied for all tubes or, if not satisfied, to report the occurrence in accordance with 10 CFR 50.72, "Immediate Notification Requirements for Operating Nuclear Power Reactors," and 10 CFR 50.73, "Licensee Event Report System."

New Technical Specifications Requirements for Ensuring Steam Generator Tube Integrity

As discussed above, NRC requirements for the ISI and repair of SG tubes are contained in the plant TS. Until recently, these TS requirements were entirely prescriptive in nature, consisting of specified sampling plans for tube inspection, specified inspection intervals, and flaw acceptance limits (termed "tube repair limits") beyond which the tube must be removed from service by plugging or must be repaired. The TS defined the SGs to be operable when the facility met these requirements.

Although these requirements were intended to ensure SG tube integrity in accordance with the plant design and licensing bases (including the applicable regulations in 10 CFR Part 50), operating experience has shown that these earlier requirements did not necessarily ensure that facilities would meet this objective. For example, the required minimum tube inspection sample sizes and eddy current test (ECT) flaw detection performance were sometimes insufficient to ensure the timely detection of flaws before the desired margins against burst and the desired degree of leak tightness were compromised. In addition, ECT measurement uncertainties and flaw growth rates sometimes exceeded those allowed for by the tube repair criteria. Also, when flaws were detected by ISI and were determined to exceed the tube repair criteria (dictating plugging or repair of the affected tubes), there was no requirement to demonstrate that the affected tubes retained the desired margins against burst and leakage integrity at the time these flaws were detected and plugged or repaired. Thus, implementation of the surveillance requirements alone did not necessarily ensure that the scope, frequency, and methods of inspection would be sufficient to ensure SG tube integrity. These earlier requirements did not directly ensure that the objective of GSI-163 was being met.

As such, licensees experiencing significant degradation problems frequently found it necessary to implement measures beyond the minimum TS requirements in order to ensure the maintenance of adequate tube integrity consistent with the plant design and licensing bases. Until the 1990s, these measures tended to be ad hoc and licensee-specific. In the meantime, the industry and the NRC staff began initiatives to improve the effectiveness and consistency of the utility programs to ensure SG tube integrity. NEI 97-06[1939] provided general, high-level guidelines for a programmatic, performance-based approach for ensuring SG tube integrity. NEI 97-06[1939] references a number of detailed guideline documents from the Electric Power Research Institute for programmatic details concerning SG tube inspections, SG tube integrity assessment, in situ pressure testing, and monitoring of operational primary-to-secondary leakage. The NEI 97-06[1939] approach was inspired by, and is similar to, an approach developed by the NRC staff in a draft regulatory guide, "Steam Generator Tube Integrity," published as DG-1074[1944] in December 1998.

The new TS requirements[1943] address the previous lack of a direct relationship between the TS surveillance requirements and SG tube integrity. The new TS requirements require implementation of an SG program that focuses directly on maintaining tube integrity and periodically verifying that the program continues to be successful in meeting this goal. This required SG program addresses the central objective of GSI-163 in that it is intended to ensure that all SG tubes will exhibit acceptable structural margins against burst or rupture under normal operating conditions, as well as during postulated DBAs (including MSLB), and that leakage from one or multiple tubes during postulated DBAs (including MSLB), will be limited to very small amounts, consistent with the applicable regulations for offsite and control room dose.

Overview

New performance-based TS requirements [1943] include a new limiting condition for operation (LCO) that tube integrity shall be maintained with an associated surveillance requirement and that tube integrity shall be verified in accordance with the SG program. The key elements of the SG program are defined in the TS administrative controls, which specify that an SG program shall be established and implemented to ensure that SG tube integrity is maintained. The TS do not provide specific details on how this objective is to be met; it is the licensee's responsibility to ensure that the program will meet the stated objective. Industry guidelines in NEI 97-06[1939] and other guidance referenced therein provide a resource to utilities for meeting this objective. However, the TS do define a general programmatic framework for the SG program, which must include the following elements:

- performance criteria for SG tube integrity
- provisions for condition monitoring
- provisions for tube repair criteria
- provisions for SG tube inspections
- provisions for monitoring primary-to-secondary leakage

The TS define three different types of performance criteria for evaluating SG tube integrity:

(1) structural integrity criteria
(2) accident-induced leakage (primary-to-secondary) criteria
(3) operational primary-to-secondary leakage criterion

The condition of the tubes relative to the structural integrity criteria and the accident-induced leakage criteria is evaluated periodically, based on inservice inspection results, in situ pressure tests, or other means before the plugging of tubes to confirm that these criteria are met for all tubes. This periodic evaluation is termed a condition monitoring assessment and is performed during each plant outage during which the SG tubes are inspected, plugged, or repaired. The operational leakage criterion corresponds to the TS LCO limit for primary-to-secondary leakage. Primary-to-secondary leakage is monitored while the plant is operating. Should this leakage exceed the TS LCO limit, the plant must be shutdown in accordance with the TS. The structural integrity criteria define the minimum factors of safety against burst or plastic collapse that must be maintained for all tubes under normal operating and DBA loading conditions. These safety factor criteria were developed to be consistent with the safety factors that are ensured by the stress limits in ASME Code, Section III[1938] (i.e., the design basis). These safety factor criteria include, for example, a safety factor of 3 against burst under normal steady state full-power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to design-basis accident primary-to-secondary pressure differentials.

Even if all tubes exhibit safety factors in accordance with the structural integrity performance criteria, tubes with localized flaws can leak under normal operating and accident conditions, without burst or collapse. The central DPO concern[1936, 1940] was that such leakage from multiple tubes may lead to significant radiological releases or core melt. The accident-induced leakage criteria address this concern by limiting the allowable total accident-induced leakage in each SG (as determined during condition monitoring assessments) to values assumed in the licensing basis accident analyses to demonstrate that offsite and control-room doses meet applicable regulatory requirements. The accident-induced leakage criteria values are a small fraction of the values associated with a ruptured tube or values that affect peak clad temperature and the likelihood of core melt.

Given the TS LCO operational leakage limit, a separate performance criterion for operational leakage is unnecessary for ensuring prompt shutdown if the limit is exceeded. However, operational leakage is an indicator of tube integrity performance, although it is not a direct indicator. It is the only indicator that can be monitored while the plant is operating. Maintaining leakage within the limit provides added assurance that the plant is meeting structural and accident leakage performance criteria. Thus, inclusion of the TS leakage limit among the set of tube integrity performance criteria is appropriate from the standpoint of completeness of the performance criteria.

The new TS require that the SG program include periodic tube inspections. This includes a new performance-based requirement that the scope, methods, and intervals of the inspections ensure the maintenance of SG tube integrity until the next inspection. This performance-based requirement complements the requirement for condition monitoring in ensuring that tube integrity is maintained. The requirement for condition monitoring is backward looking in that it is intended to confirm that tube integrity has been maintained before the time the assessment is performed. The inspection requirement, by contrast, is forward looking, as it is intended to ensure that tube inspections, in conjunction with plugging of tubes, are performed so as to ensure that the plant will continue to meet the performance criteria until the next SG inspection. Tube inspections would be followed again by condition monitoring at the next SG inspection to confirm that the performance criteria were in fact met, and so on.

The new TS performance-based requirements are supplemented by a number of prescriptive requirements relating to minimum sample sizes for tube inspections, maximum allowable inspection intervals, and tube repair criteria. Even though the new TS compel implementation of a performance-based program (including inspections and plugging) that ensures tube integrity, the prescriptive requirements pertaining to inspection sample sizes and inspection intervals provide added assurance of tube integrity should new or unexpected degradation mechanisms or changes in previously observed flaw growth rates occur. The tube repair criteria provide added assurance that degraded tubes will be plugged or repaired before the integrity of these tubes is impaired.

For the tube repair criteria, the new TS retain the standard depth-based limit of 40 percent of the nominal tube wall thickness. In addition, any plant-specific requirements pertaining to the use of alternate repair criteria in the old TS have been carried over to the new TS.

Verification

The NRC regional offices conduct periodic inspections (typically during each outage inspection) to assess the effectiveness of licensee programs for ensuring tube integrity in accordance with the technical specifications. These regional inspections are performed in accordance with the NRC Inspection Manual, Inspection Procedure 71111.08, "Inservice Inspection Activities," dated November 9, 2009.[1945]

Failure to meet any of the TS tube integrity performance criteria is reportable pursuant to 10 CFR 50.72 and 50.73 in accordance with guidelines in NUREG-1022, Revision 2, "Event Reporting Guidelines: 10 CFR 50.72 and 50.73," issued September 2004.[1946] In addition, the NRC regional office would follow up on such an occurrence, as appropriate, consistent with the NRC Reactor Oversight Program[1947] and the risk significance of the occurrence.

Finally, the new TS requirements include a requirement that the following information be submitted to the NRC within 180 days of each SG inspection:

- a description of the inspections performed
- the results of these inspections
- the active degradation mechanisms found
- the number of tubes plugged or repaired
- the results of the condition monitoring assessments (vis-à-vis the tube integrity performance criteria)

The NRC staff reviews these reports for the purposes of monitoring SG tube degradation trends and assessing the effectiveness of licensee programs. These reviews, like the regional office inspection reports, are documented and publicly available.

Effectiveness—Steam Generator Program

Although the new TS requirements have only been in place since 2005–2007 (depending on the plant), all PWR licensees have been implementing the basic performance-based elements of these requirements since 1999–2000 following their commitment to the industry's NEI 97-06[1939] initiative. The NEI 97-06 initiative was an evolutionary change in licensee programs for ensuring tube integrity, because the effectiveness of these programs has been constantly evolving and improving since the 1970s. Industry guidelines relating to secondary water chemistry control and inservice inspection have been available since this period and have been frequently updated to reflect research findings, technology developments, and operating experience. In the late 1980s, licensees became sensitized to the need to monitor operational primary-to-secondary leakage on as close to a real-time basis as possible to provide added assurance of plant shutdown before rupture of a leaking tube. Industry guidelines for monitoring and responding to operational primary-to-secondary leakage have been available since the mid-1990s. Another trend dating from the 1970s was an ever-increasing awareness among licensees of the need for their SG programs to address tube integrity in addition to satisfying TS surveillance requirements. Industry guidelines for tube integrity assessment became available in the mid-1990s and led to improved consistency, rigor, and completeness of licensee tube integrity assessments.

In parallel with these SG programmatic improvements, tube integrity reliability appears to have improved significantly since the 1970s. This is evidenced by the sharply declining trends in frequency of SGTR and of forced shutdowns because of SG leakage.[1948] The use of tubing that is more resistant to stress-corrosion cracking (i.e., thermally treated alloy 600 and 690 tubing in lieu of the mill annealed (MA) alloy 600 tubing used in SGs manufactured through the late 1970s) in new (post-1970s) and replacement SGs has been responsible for some of this improvement. However, even plants with alloy 600 MA tubing have experienced sharply improved performance trends in forced outage and SGTR frequencies. The improving trends for the plants with alloy 600 MA tubing are due to a variety of factors relating to tube integrity management programs. These include more effective secondary water chemistry programs and steps taken to control copper and impurity ingress from the feed system. These improvements in water chemistry programs, however, are not the only reason for the improving tube integrity trends, because even with the improved water chemistry programs, plants with alloy 600 MA tubing have continued to experience extensive degradation, including stress-corrosion cracking. As a result, it is clear that improved, more effective inspection programs and tube integrity management have played very important roles in reducing the frequency of forced outages because of SG leakage and SGTRs.

Even with the improved SG programs, operating experience provided examples of tube flaws that were not detected by inservice inspection. These flaws were later discovered to not satisfy the required structural and accident leakage integrity margins. There have been three such occurrences from 2000 to 2009:

- Indian Point 2—SGTR event in February 2000.[1949] This represented a failure to meet structural and leakage integrity performance criteria.

- Comanche Peak 1—Failure to meet structural and leakage integrity performance criteria in Fall 2002, as determined by in situ pressure testing during condition monitoring.[1950]

- Oconee 2—Failure to meet structural integrity performance criteria in fall 2002, as determined by in situ pressure testing during condition monitoring.[1951]

Another occurrence, at Crystal River 3 in 2003, involved an apparent failure to satisfy the accident leakage criterion.[1952] The initial finding that the accident leakage rate exceeded the performance criteria was based on use of a leakage calculation model that was overly conservative. In 2005, the NRC staff approved a more realistic, but still conservative, leakage model than that used in the 2003 calculation.[1953]

Of these three occurrences, only the tube that ultimately ruptured under normal operating conditions at Indian Point would likely have ruptured had an MSLB event occurred during a several-month period preceding the SGTR event. This experience indicates that the frequency at which tubes may be vulnerable to rupture (or leakage from multiple tubes comparable to a ruptured tube) under MSLB is well within the conditional probability value of 0.05 assumed in NRC risk studies.[681, 1954]

On the basis of the above, the staff concludes that SG program improvements in the areas of inservice inspection and tube integrity management and assessment have contributed significantly to improved SG tube integrity performance. Improved water chemistry practices and the increasing number of PWRs with SGs of improved design and more stress corrosion cracking resistant tubing have also contributed to this trend.

Disposition of ACRS (DPO Review Panel) Recommendations

An ACRS ad hoc subcommittee served as the NRC DPO review panel for the DPO, documenting its conclusions and recommendations[1800, 1898] in February 2001. This section addresses the subcommittee's conclusions and recommendations as they relate to the adequacy of NRC requirements to ensure that tube structural and leakage integrity will be maintained such that there is reasonable assurance that public health and safety will continue to be maintained.

Voltage-Based Alternate Repair Criteria Issues

Background on Voltage-Based Tube Repair Limits

The DPO concerns were first prompted by the finding of intergranular attack (IGA) and ODSCC at the tube-to-TSP intersections at the Trojan nuclear power plant in 1991, the challenges that were encountered in reliably detecting such flaws, and consideration being given at the time to allowing some tubes with greater than 40 percent through-wall flaws to remain in service. At Trojan, and subsequently at many other PWRs with Westinghouse-designed SGs, ECT inspections identified hundreds of indications at the tube-to-TSP intersections. Examination of tube specimens removed from the field (i.e., pulled tube specimens) identified the degradation mechanism as stress-corrosion cracking initiating from the ODSCC, with varying degrees of general IGA. These examinations showed the ODSCC IGA to be confined to within the 0.75-inch thickness of the TSPs. Burst testing of these specimens revealed the failure mode to be axial.

ECT techniques were not capable of accurately sizing the depth of the ODSCC IGA flaws relative to the applicable TS tube repair limit of 40 percent of the nominal tube wall thickness. For this reason, it was necessary to assume that all detectable ODSCC/IGA indications exceeded the 40-percent tube repair limit, thus necessitating the plugging or repair of all affected tubes. However, the number of affected tubes at each plant ranged from hundreds to, sometimes, thousands of tubes. This had significant economic implications for the industry. Plugging such a large number of tubes would potentially significantly shorten the useful life of the SGs, after which SG replacement would be necessary. Depending on the plant, the useful SG lifetime could potentially expire before replacement SGs were available. Sleeve repairs at each TSP intersection did not appear to offer a practical, cost-effective alternative. For this reason, around 1990 the industry began to investigate alternative approaches to ensuring the integrity of tubing affected by ODSCC IGA at the TSPs.

The 40-percent, depth-based tube repair limit is intended to ensure that tubes accepted for continued service will not leak and will retain safety factors against burst consistent with the design basis (i.e., the stress limits in ASME Code, Section III[1938]) with allowance for flaw depth measurement uncertainty and for incremental flaw growth before the next scheduled inspection. These safety factors include a factor of 3 relative to normal operating pressure differential (between primary system and secondary system pressures) and 1.4 relative to postulated accident pressure differentials. The 40-percent limit was developed with the conservative assumption that degradation results in uniform thinning of the tube wall thickness in both the axial and circumferential directions. Burst testing of pulled tube samples with ODSCC IGA flaws showed the degrading effect of these flaws on tube burst pressure to be significantly less than is the case for tubes that are uniformly thinned to the same depth. This result is explained by the limited axial extent of the flaws (i.e., less than the thickness of the TSP (0.75 inches)), the nonuniformity of the depth profile, and the often segmented rather than continuous nature of the cracks. In some cases, crack segments could penetrate up to 100 percent through the tube wall while maintaining structural safety margins consistent with the design basis.

It was also observed from burst and leak tests performed on the pulled tube samples that those ODSCC IGA indications that had exhibited low-voltage ECT signals in the field tended to exhibit high burst strengths and low potential for leakage compared to indications exhibiting higher voltage responses. This observation led the industry to develop a database from pulled tube specimens and lab specimens correlating voltage response of the ODSCC IGA indications with burst strength, probability of leakage (POL) under MSLB differential pressure, and leak rate (given that leakage occurs) under MSLB differential pressure. This database was used as the basis for developing voltage-based ARC. Statistical/mathematical models were developed for each of these correlations. The burst and leak rate correlations were represented by a mean regression curve and an associated variability distribution to capture the scatter or variability of the data. The POL correlation was modeled as a log-logistic function with an associated uncertainty distribution. Separate sets of correlations were developed for SGs with 7/8-inch diameter tubing and 3/4-inch diameter tubing respectively.

The NRC approved the voltage-based ARC on an interim basis for Trojan in 1992, and subsequently for other plants. In 1995, the staff issued Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," dated August 3, 1995, with guidance for submitting applications for permanent voltage-based ARC amendments.[1804] Over the next several years, the NRC approved voltage-based ARC TS amendments for 27 units. However, because of subsequent SG replacements at many of these plants, only three units continue to have TS that allow implementation of the voltage-based ARC.

The supporting databases for the burst and leakage correlations are periodically updated as additional data become available. The conditional leak rate correlation for 7/8-inch diameter tubing has been, and continues to be, weak. For this reason, use of the linear regression fit of the conditional leak rate correlation is subject to demonstrating that the fit is valid at the 5-percent level with the "p-value" test. If this condition is not satisfied, the linear regression fit is assumed to be constant with voltage.

When implementing the voltage-based ARC, an upper limit on voltage is established such as to provide a factor of 1.4 against burst under MSLB conditions. (The TSP constrains radial expansion of the tube under normal operating conditions, ensuring that the factor-of-3 criterion for normal operating conditions is met.) For MSLB, the TSP is conservatively assumed to be displaced axially by hydraulic blowdown loads. Thus, the TSP is assumed not to constrain radial expansion and burst under MSLB conditions. This voltage limit is deterministically based, corresponding to the voltage in the burst pressure versus voltage correlation where the lower 95-percent prediction interval burst pressure equals 1.4 times the MSLB differential pressure. This voltage is adjusted downward on a plant-specific basis to allow for voltage growth between inspections and for voltage measurement variability. The voltage growth value is a generic or plant specific value, whichever is larger. Plant-specific values are based on the average value observed during the most recent one or two inspection intervals. The voltage measurement variability is an upper 95-percent cumulative probability estimate based on industry data.

Given the scatter and variability of the burst and leakage correlations, the growth rate distribution, and the voltage measurement variability distribution, there remains a probability that an indication at the upper voltage limit may burst at a pressure less than 1.4 times MSLB pressure. For this reason, it must also be demonstrated that the conditional probability of one or more tubes bursting under MSLB conditions, from among the entire population of indications projected to exist at the next scheduled inspection, is less than 0.01 (operational assessment). This forward projection (using a Monte Carlo sampling method) is performed assuming that the probability of detection (POD) for ODSCC IGA flaws during the current, or most recent, inspection is 0.6, independent of voltage amplitude. This conditional probability criterion was developed [681] to ensure that implementation of the voltage-based ARC would not significantly increase risk. In addition, a similar analysis is done during each inspection based on the as-found indications (without consideration of voltage growth) to confirm that the conditional probability criterion was met during the prior period of operation (condition monitoring assessment).

All tubes with bobbin coil indications exceeding the upper voltage criterion must be plugged. In addition, lower limits on voltages of 1 volt for 3/4-inch diameter tubing and 2 volts for 7/8-inch diameter tubing have been established for conservatism. Tubes with bobbin indications higher than the lower voltage limit, but less than or equal to the upper voltage limit, may be left in service if rotating probe inspections do not confirm the bobbin coil indication.

When implementing the voltage-based ARC, the licensee must also demonstrate that leakage under MSLB conditions will not exceed values assumed in the licensing-basis accident analyses. The MSLB leakage assessment is performed in a similar manner as the conditional probability of burst analyses, except that the Monte Carlo sampling is performed on the POL and leak rate correlations instead of the burst correlation. This analysis yields a probability distribution of leak rates. The MSLB leak rate is the upper 95-percent percentile value from the distribution, evaluated at an upper 95-percent confidence bound.

ACRS Ad Hoc Subcommittee's Conclusions

The ACRS ad hoc subcommittee's conclusions supported the technical adequacy of the voltage-based ARC[1804] subject to two recommendations described later. Specific conclusions included the following:

- "There is a need for ARCs." The subcommittee did not focus on the economic benefits of ARCs (from avoided tube plugging and repairs and extended SG life), but rather on the need for different plugging criteria to address the different types of degradation being encountered in the field. The subcommittee noted that ODSCC in the tube at the TSP intersections is difficult to detect and characterize relative to the standard 40-percent, depth-based repair criterion. The subcommittee noted the conservatism of the standard 40-percent, depth-based criterion for this type of degradation and the attractiveness of voltage-based ARCs for this type of degradation, especially if supplemented by characterizations that ensure flaws producing the signal meet explicit and implicit assumptions about the possible growth and behavior of flaws.

  The staff notes that the voltage-based ARC includes specific requirements for verifying that these assumptions continue to be valid. For example, the assumptions that the ODSCC has a predominant axial orientation and that it is confined to within the thickness of the TSP is verified by laboratory examinations of representative tube samples, which are periodically removed from the SGs, and by rotating coil inspection of all tube-to-TSP intersections with bobbin coil responses exceeding 1 volt for ¾-inch diameter tubing and 2 volts for 7/8-inch diameter tubing.[1804] As discussed later in response to the subcommittee's recommendation that the staff should develop a program to monitor the predictions of flaw growth for systematic deviations from expectations, the staff believes that any systematic deviations from expectations in flaw growth will be identified and addressed in the staff review of the reports submitted after each outage during which the voltage-based ARC is implemented.

- "Plants will be operated with flaws in the SG tubes and this need not be risk significant." The subcommittee noted that, provided risk is managed properly, it is acceptable to operate plants with known, small flaws as well as undetected flaws in the SG tubes. As discussed in the previous section, the staff notes that the new technical specifications ensure low risk by requiring implementation of an SG program that ensures that all tubes satisfy the performance criteria for structural and leakage integrity. The staff also notes that the performance criteria associated with implementation of voltage-based ARCs differ somewhat from those in the new generic TS (which are applicable when not implementing the voltage-based ARC). The ARC-specific performance criteria include a conditional probability criterion for induced tube ruptures to ensure that the conditional probability for induced ruptures is within values assumed in past risk assessments.

  The subcommittee also noted that additional, defense-in-depth management of risk can be achieved by restricting known flaws in the tubes to those unlikely to grow significantly during an operating cycle. The staff agrees, noting there have been cases in which preventive plugging of tubes not in violation of the voltage-based repair criteria was performed to prevent high-voltage growth from occurring during the next operating cycle.

- "The general features of the procedures that the staff has established to limit the number and size of flaws left in operating SG tubes are adequate." The subcommittee found no fault with the concept of voltage-based ARC and found the voltage repair criterion of 1 volt for ¾-inch diameter tubing and 2 volts for 7/8-inch diameter tubing to be conservative. The subcommittee did not attempt to reach conclusions about occasions when the staff granted exceptions to these criteria, except to note that these exemptions should have been accompanied by more complete risk analyses. The staff notes that the 1- and 2-volt criteria are lower threshold limits and that all indications below these limits are acceptable.[1804] However, the voltage-based ARC includes higher upper bound voltage threshold limits, which are determined in accordance with the voltage-based ARC methodology in Generic Letter 95-05.[1804] This methodology is based on satisfying the voltage-based ARC performance criteria, including the criterion on conditional probability of induced ruptures during MSLB, with allowance for voltage measurement variability and voltage growth rate distribution. As noted by the subcommittee, the staff approved an increase in the lower voltage threshold limit to 3 volts for three plants (with ¾-inch diameter tubing) where a number of tubes were expanded against the tube support plates for purposes of limiting axial support plate deflection under MSLB conditions.

These changes are no longer in effect, because these plants have undergone SG replacement and the provisions for implementing voltage-based ARCs have been eliminated from the TS for these plants. Under these changes, the licensees were required to demonstrate that the conditional probability of burst criterion continued to be met. Thus, the staff believes there were no risk implications associated with the 3-volt criterion.

- "The general features of the condition monitoring program are adequate." The subcommittee found the general approach used to assess the probabilities of leakage and tube burst to be conservative. The subcommittee felt that the development of empirical correlations of burst pressure and leakage with voltage amplitude are technically defensible. The subcommittee found no evidence that the supporting databases were flawed in any nonconservative, systematic way. The subcommittee felt that the constant POD assumption in the voltage-based ARC methodology approved by the staff could potentially deter technical improvements, but acknowledged that the staff would consider approving alternative POD assumptions that recognize that POD can depend on flaw size (with a sufficient technical justification). In fact, the NRC staff has approved an alternative[1955] to the constant POD model that replaces the POD parameter with a parameter known as the "probability of prior cycle detection." This empirical, plant-specific parameter is voltage dependant and relates the total number of indications found during a given inspection in a given voltage bin to the subset of these indications that were also detected during the previous inspection.

The subcommittee concluded that the condition monitoring program that licensees adopt in conjunction with the ARC, although not perfect, can produce a better understanding of the conditions and vulnerabilities of steam generators and afford additional protection to the public than has been possible in the past. The staff agrees with this conclusion and notes that the voltage-based ARC was an important step that contributed to the ultimate development of the performance-based strategies in DG-1074,[1944] NEI 97-06,[1939] and the new TS for ensuring SG tube integrity.

ACRS Ad Hoc Subcommittee's Recommendations

The following two recommendations accompanied the above conclusions by the ACRS ad hoc subcommittee:

(1) ACRS ad hoc subcommittee recommendation:[1898] "The databases for 7/8-inch diameter tubes need to be greatly improved to be useful."

The subcommittee observed that the correlation of leakage with voltage for the 7/8-inch diameter tubes does not correspond well with that for 3/4-inch diameter tubes. The subcommittee could identify no mechanistic reasons why this should be the case. The subcommittee felt that the poor correspondence may reflect stochastic scatter and the limited size of the database. Therefore, the subcommittee felt that the staff should consider requiring a near-term expansion of the database.

The staff evaluated this recommendation under item number 3.7 of the SGAP.[1899, 1940] The staff's findings are documented[1956] and include the following:

A. Evaluation of the leakage data has not led to a conclusive explanation for the poor correlation of the 7/8-inch diameter tube leakage data compared with ¾-inch diameter tube leakage data.

B. The poor correlation notwithstanding, the methodology for assessing leak rate is conservative for the following reasons:

   a. Pre-pull voltage responses are used for the correlations. If the crack tears as a result of the tube pull operation, the measured voltage is expected to be higher than if the tube were not damaged.

   b. The leak rate analysis yields a probability density function of total leak rate (using Monte Carlo sampling of the input parameter distributions and leak rate distributions as a function of voltage) for a given population of voltage responses. This probability density function is evaluated at the upper 95th percentile value at an upper 95-percent confidence bound vis-à-vis the applicable performance criterion for accident leakage.

   c. If a statistical correlation between leak rate and voltage cannot be demonstrated to within criteria specified in Generic Letter 95-05,[1804] Generic Letter 95-05[1804] specifies that leakage shall be treated as independent of voltage, which is conservative (because most indications left in service are relatively low-voltage indications, which tend to leak less than the mean).

On the basis of the above, the staff concluded[1956] that item number 3.7 (the leakage correlation issue) is adequately addressed and is, therefore, closed. In addition, the staff stated that it would continue to assess the leakage correlations as more data are added to the database. The ACRS reviewed these findings.[1862] The ACRS continues to believe that the leakage correlation for 7/8-inch diameter tubing should not be used, which is contrary to the staff's position, as stated above. As previously noted, the voltage-based ARC, including the leakage correlation, continues to be used at one plant with 7/8-inch diameter tubes (as of February 2009) and is approved for use at two additional plants (but not currently implemented). However, the ACRS stated that it agrees with the staff that the choice of a 2-volt limit for 7/8-inch diameter tubes is conservative with respect to the risk posed and that item number 3.7 should be closed.

(2) ACRS ad hoc subcommittee recommendation:[1898] "The staff should establish a program to monitor the predictions of flaw growth for systematic deviations from expectations."

One step in the voltage-based ARC methodology is the prediction of the change in the voltage distribution over an operating cycle. The subcommittee noted that this is done assuming a linear change in the distribution with time. The subcommittee noted that this is inconsistent with behavior of stress corrosion cracks observed in NRC research. These studies show that cracks grow slowly until they interlink, after which it is possible for flaws to grow very quickly. Flaw growth, then, is inherently nonlinear and can be treated as linear with time only in a bounding manner. The subcommittee stated that, even then, stochastic variability means that occasionally individual cracks can violate even very conservative linear bounds. Thus, the subcommittee found that it will be important for the staff to be vigilant in monitoring the implementation of the ARC to watch for such systematic errors in the crack growth predictions.

The staff evaluated this recommendation under item number 3.8 of the SGAP.[1899, 1940] The staff's findings are documented[1957] and are summarized in the following paragraphs.

In accordance with GL 95-05,[1804] licensees submit information related to the structural and leakage integrity of the tubes within 90 days (the 90-day report) of completion of the steam generator tube inspections. The information submitted includes the actual voltage distribution and the projected voltage distribution for the next operating cycle. It also includes the tube burst probability and calculated leakage under main steamline break differential pressure conditions. The projected voltage distribution with the resultant tube burst probability and leakage estimates account for flaw growth.

The staff routinely reviews these 90-day reports and compares the tube burst probability and leakage to the criteria specified in GL 95-05.[1804] In addition, the staff compares the predicted values to actual values. If the predicted values are conservative, the flaw growth distribution used in the prediction is typically considered to be within expectations. If the predicted values are not conservative when compared to the actual values, the staff evaluates the root cause and ensures appropriate corrective actions are taken by the licensee.

In summary, the staff concluded[1957] that any systematic deviations from expectations in flaw growth will be detected and addressed in the staff review of the 90-day reports. The staff also concluded that crack growth rates will continue to be adequately monitored as part of the implementation of the voltage-based ARC and considers SGAP item number 3.8 to be closed.[1958]

Damage Progression Issues

The ACRS ad hoc subcommittee recommended: "Risk analyses that the staff considers need to account for progression of damage in a more rigorous way."[1898] This recommendation stemmed from a DPO concern that dynamic loads induced in steam generator tubes by an MSLB or other secondary-side breaches would lead to growth of cracks and increased steam generator tube leakage or ruptures outside the range of analyses and experiments performed by the NRC staff. In addition, an MSLB may impose dynamic loads on the TSPs beyond simply those associated with differential pressure loads, and these loads could be transferred to the tubes. The subcommittee noted that this concern affects any consideration of SG tube integrity and is not unique to use of voltage-based ARCs. The staff opened a new generic issue, GSI-188, "Steam Generator Tube Leaks or Ruptures Concurrent with Containment Bypass from Main Steam Line or Feedwater Line Breaches," in part to address this concern. This work was performed under item number 3.1 of the SGAP[1899, 1940] and was completed. Key conclusions of the staff in resolution of GSI-188 included:[1870]

- Dynamic loads and resonance vibrations following an MSLB are low and have little impact on growth of existing cracks beyond the effects of differential pressure stress alone.
- Dynamic loads from an MSLB or feedwater line break do not affect the structural integrity of tubes in service and do not lead to additional leakage or ruptures beyond what would be determined using differential pressure loads alone.
- Therefore, the principal assertion of GSI-188 is closed, and no changes to existing regulations and guidance are recommended.
- The dynamic load effects from an MSLB or feedwater line break need not be taken into account in evaluating the potential for multiple tube ruptures under GSI-163.

The ACRS reviewed the technical basis for these findings[1862] and concluded that item number 3.1 of the SGAP is appropriately closed out. Confirmatory information requested by the ACRS[1862] was subsequently provided to the ACRS.[1870]

Jet Impingement Issue

The ACRS ad hoc subcommittee considered a DPO concern that particulate-laden fluids flowing from a cracked SG tube can pierce adjacent tubes. The staff evaluated this concern as item number 3.2 of the SGAP.[1899, 1940] This item addressed both MSLB and severe accident conditions. In its review of the DPO concerns,[1898] the ACRS ad hoc subcommittee concluded that the staff had undertaken adequate research (under item number 3.2 of the SGAP) to address this issue. The subcommittee stated that, although it is necessary to carry this research to an appropriate conclusion, early results suggest that damage progression by the jet cutting mechanism is not likely.

Item number 3.2 has been completed,[1959] and the detailed results of this study for MSLB conditions are documented.[1960] The study was based on tests that provided a conservative simulation of an MSLB to determine the susceptibility of SG tubes to erosive damage from impacting jets of superheated steam leaking from adjacent tubes. This study showed that the likelihood of failure propagation by jet erosion is low under these conditions.

The detailed results for severe accident conditions are documented.[1961] Erosion tests were conducted in a high-temperature, high-velocity erosion rig using micron-sized nickel and aluminum oxide particles mixed in a high-temperature gas. The erosion results, together with analytical models for crack opening area and jet velocities, were used to estimate the erosive effects of superheated steam with entrained aerosols from the core during severe accidents. It was determined that failure of an adjacent tube by jet impingement would take more than 10 hours after the subject crack had undergone significant crack opening displacement by creep at high temperature. However, once the system has reached these high temperatures, failure of some primary system component, including unflawed SG tubes, would be expected to occur in less than 1 hour. Thus, jet impingement is very unlikely to contribute in any significant way to severe accident risk.

The ACRS agreed with the staff's conclusion that the probability of damage progression via jet cutting of adjacent SG tubes is low and need not be considered in accident analyses.[1862] The ACRS also agreed that SGAP item number 3.2 should be closed.

Crack Unplugging Issue

The ACRS ad hoc subcommittee considered a DPO concern that forces involved with MSLB blowdown and leakage through cracks can cause cracks plugged with corrosion products to leak. In addition, the DPO was concerned that corrosion products in the annular gap between the tubes and TSP holes can be expelled, allowing otherwise occluded cracks to leak. The subcommittee stated that it found no evidence that the "unplugging" of cracks is a damage progression mechanism of concern.[1898] The subcommittee made no recommendations concerning any followup study of this issue, and no such work has been included as part of the SGAP. The staff does not believe such work is necessary. Models used to predict leak rate under accident conditions tend to be mechanistic models (based in part on crack geometry) that have been benchmarked against test data (from pulled tube specimens and laboratory specimens) or empirical models such as that used for the voltage-based ARC. In both cases, the test data are expected to reasonably reflect the leakage that would be expected for cracks in the free span under actual accident conditions.

Risk Issues Pertaining to Tube Ruptures or Leakage during MSLB

A central concern of the DPO[1936] was that MSLB can lead to primary-to-secondary leakage of tube rupture proportions sufficient to deplete the reactor water storage tank inventory via emergency core cooling system injection lost to the secondary side of the SGs (and therefore not available for recirculation from the containment sump), thereby leading to core damage with containment bypass. This concern relates to primary-to-secondary leakage from one or more tube ruptures or relatively large numbers of tubes that have not burst, such that the total leakage from all tubes is comparable to one or more tube ruptures.

The DPO estimate of core damage frequency and containment bypass frequency associated with SG tube leakage as a consequence of an MSLB was 1.0×10^{-4} per reactor year (RY).[1937] This estimate is based on assuming (1) an MSLB frequency of 1.0×10^{-4}/RY, (2) a conditional probability of 1.0 that primary-to-secondary leakage will be of tube rupture proportions under MSLB conditions, and (3) a conditional probability of 1.0 for failure to successfully mitigate the event before core damage occurs.

Staff PRAs considered by the ACRS ad hoc subcommittee assumed that the frequency of initiating secondary side depressurization events is dominated by stuck-open SG relief valves, with a frequency of 1×10^{-3}/RY estimated from operational event data. The frequencies of MSLB and main feed line break are estimated to be 6.8×10^{-4}/RY and 1.8×10^{-4}/RY, respectively, for a 4-loop plant. The DPO did not appear to have any concerns relative to these estimates, nor did the ACRS ad hoc subcommittee state any concern relative to these estimates.

Conditional Probability of SG Tube Rupture during MSLB

The DPO concern relates to plants with widespread stress-corrosion cracking, particularly those plants with ARC TS that allow many tubes with such cracks to remain in service, and that, because of eddy current limitations in reliably detecting such cracks, leakage of tube rupture proportions is the expected outcome. As discussed earlier, the ACRS ad hoc subcommittee acknowledged that ECT techniques are not capable of 100-percent accuracy in detecting flaws (though noting the technical advances that have led to improved detection performance). However, the subcommittee stated that this does not degrade the protection afforded to the public health and safety, provided the risk is properly managed.

Staff PRAs considered by the ACRS ad hoc subcommittee assumed the conditional probability of ruptures or leakage from multiple tubes of tube rupture proportions to be equal to or less than 0.05. The ACRS subcommittee did not make specific comments regarding the staff's assumption, but concluded that, if the risk can be managed properly, it is acceptable to operate plants with known, small flaws as well as undetected flaws in the SG tubes. As an example of managing risk, the ACRS ad hoc subcommittee cited the voltage-based ARC methodology that requires that the conditional probability of rupture be demonstrated periodically to be 0.01 or less (for tubes degraded by ODSCC at the tube-to-TSP intersections). Looking beyond voltage-based ARCs, the performance-based strategy for ensuring tube integrity in the new TS (i.e., ensuring and periodically demonstrating that all tubes satisfy the structural and accident leakage integrity performance criteria consistent with the design and licensing bases) is a risk management strategy. Meeting the performance criteria on a consistent basis ensures that the conditional probability of tube leakage of tube rupture proportions under MSLB is low relative to values assumed in PRAs. This conclusion is supported by operating experience, as discussed earlier.

Accident Mitigation/Human Factors Issues

The ACRS ad hoc subcommittee concluded that "analyses of human performance errors during design basis accidents appear consistent with current practices."[1898] The subcommittee reviewed the DPO concern that the staff's estimate of the probability that the operators will fail to perform tasks needed to establish the long-term cooling of the core (i.e., 10^{-3} or 1 in 1,000) is overly optimistic. The subcommittee concluded that the staff estimate appears consistent with the state of current understanding of human performance errors when only a single tube ruptures. The subcommittee stated that, in developing assessments of risk concerning these DBAs, the staff must consider the probabilities of multiple tube ruptures until adequate technical arguments have been developed to show that damage progression is improbable.[1898]

The DPO's and ACRS ad hoc subcommittee's concerns pertaining to damage progression were evaluated under item numbers 3.1 and 3.2 of the SGAP. As discussed above, the ACRS has concurred with the staff's conclusions drawn from the results of these studies and with the staff's closure of these item numbers. The staff concludes that the damage progression mechanisms cited in the DPO are unlikely to increase the probability of multiple tube ruptures beyond that which has already been considered in staff PRAs.

The ACRS ad hoc subcommittee also observed that the staff needs to develop defensible analyses of the uncertainties in its risk assessments, including uncertainties in its assessments of human error probabilities. The subcommittee noted that, as the staff develops a better understanding of the dynamic processes associated with depressurization during an MSLB, the staff may want to revisit estimates of operator error probability in light of the considerable distraction that might occur during such events. In response to the comments, the staff is developing improved methods for risk assessment under item number 3.5 of the SGAP.[1899, 1940] This item number is considered outside the scope of GSI-163 because it is focused on severe accidents and its completion is not expected (based on early results) to identify needed improvements to the current regulatory framework for ensuring SG tube integrity. With respect to operator distraction that may occur during such an event, the staff notes that the dynamic effects of the event will happen quickly. No mandatory operator actions are needed while the plant is experiencing these short-lived dynamic effects.

Severe Accident Risk Issue

The ACRS ad hoc subcommittee considered a DPO concern that severe accident sequences in which the primary system remains pressurized are more likely to evolve into steam generator tube rupture accidents than the staff predicts in NUREG-1570, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture," issued March 1998.[1954]

The ACRS ad hoc subcommittee concluded that "substantial uncertainties remain in the understanding of steam generator tube performance under severe accident conditions."[1898] The subcommittee stated the following:

> The staff has not developed persuasive arguments to show that the steam generator tubes will remain intact under conditions of risk-important accidents in which the reactor coolant system remains pressurized. The current analyses dealing with loop seals in the coolant system are not yet adequate for risk assessments. The treatment of mixing of flows in the inlet plenum to a steam generator under conditions of countercurrent natural convection flow are optimistic and are not substantiated by applicable data from experiments. Sensitivity studies have not explored the plausible ranges of parameter values or the space of uncertainties adequately. Finally, the Ad Hoc Subcommittee notes that analyses of failure of other locations in the coolant system subject to natural convection heating have not included a systematic examination of vulnerable locations in the system.

The ACRS ad hoc subcommittee's concerns relating to severe accidents were addressed under item number 3.4 of the SGAP.[1899, 1940] This item is outside the scope of GSI- 163 because, should any action be determined necessary to address severe accident risk concerns, these actions would likely be directed toward accident mitigation rather than modification of the current regulatory framework for ensuring SG tube integrity.

Iodine Spiking and Source Term Issues

As part of the voltage-based tube repair criteria,[1804] licensees must demonstrate that primary-to-secondary leakage that may potentially occur under MSLB conditions does not exceed values assumed in the licensing basis safety analyses to demonstrate that the associated dose consequences meet applicable regulations (i.e., 10 CFR 50.67 or 10 CFR 100, GDC 19). In accordance with the NRC's Standard Review Plan (NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition"), these dose calculations are based on an initial coolant equilibrium iodine concentration equal to the allowable limit in the technical specifications (typically 1.0 microcurie per gram) and an iodine spiking factor of 500. As part of their license amendment requests for voltage-based tube repair criteria, a number of licensees requested (and the staff approved) reduced limits in the TS on allowable equilibrium iodine concentrations in the primary coolant. This reduction in the allowable equilibrium iodine concentration means that a higher level of primary-to-secondary leakage can be tolerated, assuming the same iodine spiking factor of 500, consistent with the applicable regulatory dose limits, thus enabling additional degraded tubes to remain in service (provided all other requirements of the ARC are met). The ACRS ad hoc subcommittee reviewed a DPO concern that data (primarily from reactor trips, but including SGTR events) *indicate that spiking factor increases with decreasing steady-state iodine concentration*. Thus, there was a concern that the spiking factor used for the licensing basis accident analysis is too low when the TS limit on the iodine concentration in the primary coolant has been reduced.

The ACRS ad hoc subcommittee recommended the following: "The staff should develop a more technically defensible position on the treatment of radionuclide release to be used in safety analyses of design basis events."[1898] This recommendation was addressed under item number 3.9 of the SGAP and was discussed at the ACRS meeting on February 5–7, 2004. In a letter dated May 21, 2004, the ACRS stated: "The staff continues to treat iodine spiking in a conservative, empirical fashion. We recommend that the staff develop a mechanistic understanding of iodine spiking so that analyses reflect current plant operations and the capabilities of modern fuel rods."[1862] The ACRS continued with the following:

> The staff has not accepted our recommendation to develop a mechanistic understanding of the iodine spiking issue. The staff continues to use a conservative, empirical estimate of iodine spiking for accident consequence analyses. This estimate is based on historical data that may not reflect current practices in plant operations or the capabilities of modern fuels to prevent coolant contamination. We again encourage the staff to take advantage of iodine studies available in the literature and develop a mechanistic understanding of the phenomenon.

On the basis of these ACRS comments, the staff proposed a new generic issue, GSI-197, "Iodine Spiking Phenomena." This issue was screened[1867] by a review panel in accordance with NRC Management Directive 6.4.[1858] The review panel found the issue to be of low safety significance and concluded that it should not be continued as a safety issue. The review panel found that there is no evidence that the current regulatory approach is not bounding, even in event of a combined MSLB and SGTR, and that the current regulatory approach to iodine spiking, in spite of its empirical nature, is adequate. Generic Issue 197 and SGAP item number 3.9 are closed.[1867] The ACRS stated that it had considered the results of the staff's screening of GI-197 and had no objection to dropping this issue from further consideration.[1962]

Conclusion

To address the DPO concern, the staff evaluated the adequacy and effectiveness of industry practice and regulatory requirements relating to the management of SG tube integrity to ensure that all tubes will exhibit acceptable structural margins against burst or rupture under normal operating conditions and DBAs (including MSLB), and that leakage from one or multiple tubes under DBAs will be limited to very small amounts, consistent with the applicable regulations for offsite and control-room dose. As part of this effort, the staff considered the conclusions and recommendations of the ACRS ad hoc subcommittee, which served as the DPO review panel. The staff's followup actions taken in response to these findings served as part of its evaluation of the adequacy and effectiveness of regulatory requirements.

As of September 30, 2007, new performance-based TS requirements[1943] were in place and being implemented at all U.S. PWRs. These requirements are intended to ensure that all tubes exhibit adequate structural margins against burst or rupture for the spectrum of normal operating and DBA conditions, consistent with the original design basis. In addition, these requirements are intended to ensure that total leakage from tubes at a plant will not exceed values assumed in licensing bases accident analyses, even if no tubes actually rupture under these conditions. In addition, licensees are required to periodically demonstrate that these structural margin and accident leakage criteria are satisfied for all tubes or, if not satisfied, to report the occurrence in accordance with 10 CFR 50.72 and 50.73.

U.S. PWR licensees have used the basic elements of the required performance-based approach since 2000 as part of the industry's initiative under NEI 97-06.[1939] NEI 97-06 itself was an evolutionary development because tube inspection technologies, inspection practices, and tube integrity management practices had been undergoing significant improvement since the mid-1970s. These improvements contributed significantly to improved SG tube integrity performance during this period. Improved water chemistry practices and the increasing number of PWRs with SGs of improved design and more stress-corrosion crack-resistant tubing have also contributed to this trend. Since adoption of the NEI 97-06 performance-based strategy in licensee SG programs and the corresponding availability of more complete information about instances of failure to satisfy SG tube integrity performance criteria, actual incidences of failure to meet these criteria have been infrequent. This experience provides strong evidence that the potential for one or more tube ruptures, or leakage from multiple tubes totaling tube rupture proportions, under normal operating conditions or DBAs is well within that assumed in NRC risk studies to date.

The staff completed all SGAP[1899, 1940] tasks that were opened to address the ACRS ad hoc subcommittee's conclusions and recommendations stemming from its review of the DPO concerns relating to voltage-based ARCs, damage progression mechanisms, and iodine spiking. On the basis of the results of these tasks, the staff concluded that the DPO concerns relating to these issues were not substantiated and that no changes to existing requirements were needed to ensure public health and safety. The ACRS concurred with the closure of these issues. In response to ACRS ad hoc subcommittee conclusions and recommendations, the staff continued to evaluate risk issues associated with accident sequences involving ruptured or leaking SG tubes as part of SGAP[1899, 1940] item numbers 3.4 and 3.5. These studies are primarily focused on severe accidents and are not expected to identify needed changes to existing requirements for managing SG tube integrity; therefore, they are outside the scope of GSI-163.

On the basis of the above, the staff concluded that current TS requirements[1943] relating to SG tube integrity provide reasonable assurance that all tubes will exhibit acceptable structural margins against burst or rupture under normal operating conditions and DBAs, including MSLB, and that leakage from one or multiple tubes under DBAs will be limited to very small amounts, consistent with the applicable regulations for offsite and control-room dose. Thus, the staff concludes that the GSI-163 principal assertion and related concerns in the DPO are not substantiated, that no changes to existing regulations or guidance are needed, and that actions for the GSI are completed.

In accordance with Management Directive 6.4,[1858] the GSI closeout process includes an endorsement by the ACRS. The staff met with the ACRS on May 7, 2009, to discuss the staff's technical basis for resolution of GSI-163. In a letter dated May 20, 2009, to Gregory B. Jaczko, Chairman, NRC, the ACRS concluded that GSI-163 can be closed as proposed by the staff.[1963] On July 16, 2009, the staff issued a memorandum to the EDO to indicate the completion of actions for GSI-163.[1948]

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APPENDIX B

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

This appendix contains a list of those generic safety issues (GSIs) that are applicable to operating and future reactor plants, including issues that have been resolved with requirements (e.g., I, NOTE 3(a)) and issues that are in progress for resolution. The priority designations for all issues are consistent with those listed in Table II of the Introduction to NUREG-0933. In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 52.47(a)(21), applications for design certification must contain proposed technical resolutions of those unresolved safety issues, high- and medium-priority generic safety issues, which are identified in the version of NUREG-0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design. Similarly, in accordance with 10 CFR 52.79(a)(20), applications for combined licenses must contain proposed technical resolutions of those unresolved safety issues, high- and medium-priority generic safety issues, which are identified in the version of NUREG-0933 current on the date up to 6 months before the docket date of the application and which are technically relevant to the design.

In Management Directive (MD) 6.4, "Generic Issues Program," first issued July 21, 1999, the U.S. Nuclear Regulatory Commission replaced prioritization of generic issues (GIs) with the screening process, in which staff determines to either establish the proposed issue as a bone fide GI or reject the issue from the program. For the purposes of 10 CFR 52.47(a)(21) and 10 CFR 52.79(a)(20), any GI established by the MD 6.4 screening process is considered equivalent to a high-Priority GI.

<u>Legend</u>

ACTIVE	Work on the issue continues in accordance NRC Management Directive 6.4
B&W	Babcock & Wilcox Company
CE	Combustion Engineering Company
GE	General Electric Company
I	Resolved Three Mile Island (TMI) Action Plan item with implementation of resolution mandated by NUREG-0737
NOTE 3(a)	Resolution resulted in the establishment of new regulatory requirements (rule, regulatory guide, SRP change, or equivalent)
ROI	Regulatory office implementation: A formal GI for which Office of Nuclear Regulatory Research actions of safety/risk assessment or regulatory assessment are complete and remaining actions reside with program offices (e.g., regulatory compliance, Reactor Oversight Process, rulemaking, further research, coordination with industry initiatives)
MEDIUM	Medium safety priority (discontinued December 4, 2001)
MPA	Multiplant action
NA	Not applicable
TBD	To be determined
USI	Unresolved safety issue
<u>W</u>	Westinghouse Electric Corporation

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.1.1	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

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.G.2	Scope of Test Program	NOTE 3(a)	All	All	-	NA	07/81
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

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

Appendix B (continued)

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

Appendix B (continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor Trip for LOFW, TT, or Significant Decrease in SG Level	NOTE 3(a)	NA	B&W	-	03/31/80	03/31/80
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	NOTE 3(a)	All	NA	-	03/31/80	03/31/80
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	NOTE 3(a)	All	NA	-	03/31/80	03/31/80
II.K.1(24)	Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses between Reactor Trip and RCP Trip	NOTE 3(a)	NA	All	-	NA	
II.K.1(25)	Develop Operator Action Guidelines	NOTE 3(a)	NA	All	-	NA	
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	NOTE 3(a)	NA	All	-	NA	
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	NOTE 3(a)	NA	All	-	NA	
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	NOTE 3(a)	NA	All	-	01/01/81	01/01/82
<u>II.K.2</u>	<u>Commission Orders on B&W Plants</u>						
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

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		

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			
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	-	04/17/850	4/17/85
A-6	Mark I Short-Term Program (former USI)	NOTE 3(a)	GE	NA	-	12/77	NA
A-7	Mark I Long-Term Program (former USI)	NOTE 3(a)	GE	NA	D-01	08/82	08/82
A-8	Mark II Containment Pool Dynamic 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 BWR	Affected NSSS Vendor PWR	Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
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	MEDIUM	All	All	NA	NA	TBD
93	Steam Binding of Auxiliary Feedwater Pumps	NOTE 3(a)	NA	All	B-98	10/85	10/85
94	Additional Low Temperature Overpressure Protection for Light-Water Reactors	NOTE 3(a)	NA	CE, W	-	06/25/90	06/25/90
99	RCS/RHR Suction Line Valve Interlock on PWRs	NOTE 3(a)	NA	All	L-817	10/17/88	10/17/88
103	Design for Probable Maximum Precipitation	NOTE 3(a)	All	All	-	10/19/89	10/19/89
118	Tendon Anchorage Failure	NOTE 3(a)	All	All	NA	NA	07/90
124	Auxiliary Feedwater System Reliability	NOTE 3(a)	All	All	-	TBD	TBD

Appendix B (continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants—MPA No.	Operating Plants—Effective Date	Future Plants—Effective Date
			BWR	PWR			
128	Electrical Power Reliability	NOTE 3(a)	All	All	-	04/29/91	04/29/91
130	Essential Service Water Pump Failures at Multiplant Sites	NOTE 3(a)	NA	All	-	09/19/91	09/19/91
155	Generic Concerns Arising from TMI-2 Cleanup						
155.1	More Realistic Source Term Assumptions	NOTE 3(a)	All	All	NA	NA	02/95
177	Vehicle Intrusion at TMI	NOTE 3(a)	All	All	-	08/01/94	08/01/94
186	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants	ACTIVE	All	All	-	TBD	TBD
189	Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Combustion during A Severe Accident	ROI	All	All	-	TBD	TBD
191	Assessment of Debris Accumulation on PWR Sump Performance	ROI	NA	All	-	TBD	TBD
193	BWR ECCS Suction Concerns	ACTIVE	All	NA	-	TBD	TBD
199	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States	ACTIVE	All	All	-	TBD	TBD
	HUMAN FACTORS ISSUES						
HF1	STAFFING AND QUALIFICATIONS						
HF.1.1	Shift Staffing	NOTE 3(a)	All	All	-	01/84	01/84

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(See instructions on the reverse)

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

NUREG-0933, "Resolution of Generic Safety Issues," presents a description of the process and results of resolution of Generic Safety Issues (GSIs) prioritized, screened and resolved under the Generic Issues Program. GSIs are broken down into five groups: (1) TMI Action Plan items, documented in NUREG-0660 and NUREG-0737; (2) Task Action Plan items, documented in NUREG-0371 and NUREG-0471, as well as all Unresolved Safety Issues (USIs) not originally identified in these two documents; (3) new generic issues identified from various sources; (4) human factors issues, documented in NUREG-0985; and (5) Chernobyl issues, documented in NUREG-1251.

The Generic Issues Program process for resolving GSIs is described in Management Directive (MD) 6.4, "Generic Issues Program". This process includes five distinct stages that may be exercised: Identification, Acceptance Review, Screening, Safety / Risk Assessment, and Regulatory Assessment. Prior to this process, safety priority rankings of HIGH, MEDIUM, LOW, and DROP were assigned on the basis of risk significance estimates, the ratio of risk to cost and other impacts estimated to result if resolution of the safety issues were implemented, and the consideration of uncertainties and other quantitative or qualitative factors. With the issuance of MD 6.4 in 1999, the agency discontinued the use of the priority ranking model.

Supplement 33 to NUREG-0933 provides updates to the status of GSIs that completed a major milestone between July 1, 2008 and June 30, 2010. In addition, supplement 33 includes changes in the Introduction to the report and its associated Tables, changes in References and Appendix B to the report, "Applicability of NUREG-0933 Issues to Operating and Future Reactor Plants".

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