JULY 2008

SUPPLEMENT 32 TO NUREG-0933
"RESOLUTION OF GENERIC SAFETY ISSUES"

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ABSTRACT

This report presents the resolution of generic safety issues related to nuclear power plants. The purpose of these evaluations are to assist in the timely and efficient allocation of NRC resources for the resolution of those safety issues that have a significant potential for reducing risk.

Issues primarily concerned with the licensing process or environmental protection and not directly related to safety were excluded from prioritization/screening. The issues were broken down into five groups: (1) TMI Action Plan items, documented in NUREG-0660[48] and NUREG-0737[98]; (2) Task Action Plan items, documented in NUREG-0371[2] and NUREG-0471,[3] 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[651]; and (5) Chernobyl issues, documented in NUREG-1251.[1174] Future supplements to this report will include additional issues that completed major milestones as well as updated information on issues that have been resolved.

The agency's Generic Issues Program process for resolving GIs is described in MD 6.4, "Generic Issues Program", and SECY-07-0022. These documents provide recent program improvement initiatives. This new process includes five distinct stages that may be exercised: Identification, Acceptance Review, Screening, Safety / Risk Assessment, and Regulatory Assessment. Prior to implementation of MD 6.4 (1999), the safety priority rankings were HIGH, MEDIUM, LOW, and DROP and 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 priority ranking model described above.

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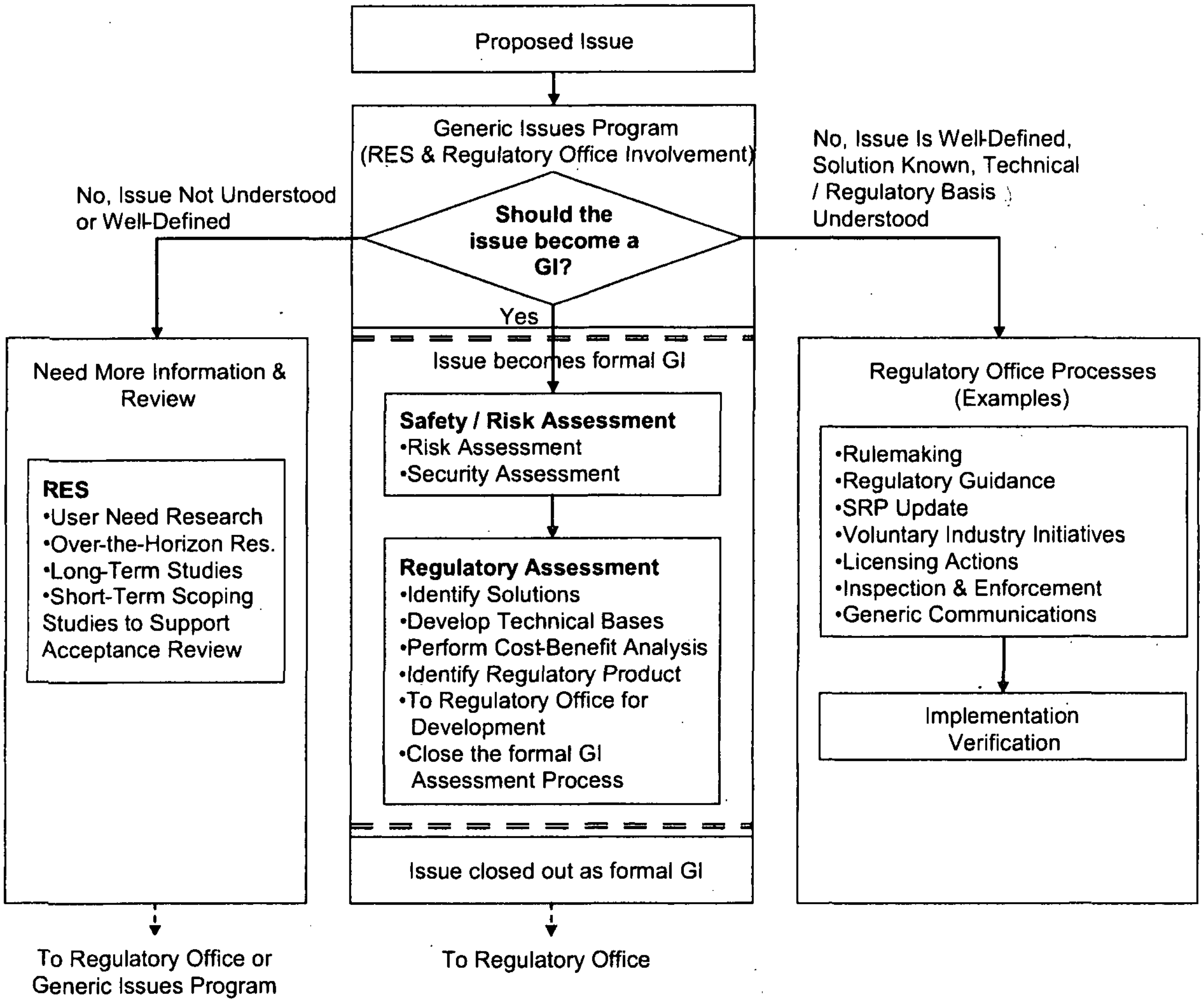
INTRODUCTION

NRC has identified by its assessment of plant operation 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 NRC maintain this program (see Section 210 of the 1974 Energy Reorganization Act (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 which 3) applies to two or more facilities and/or licensees/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 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 Generic Issues Program process for resolving GIs is described in MD 6.4, "Generic Issues Program", and SECY-07-0022. These documents provide recent program improvement initiatives. This process includes five distinct stages that may be exercised: Identification, Acceptance Review, Screening, Safety / Risk Assessment, and Regulatory Assessment. During each stage, staff determines if the issue needs more information, if the issue proceeds 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 to the person proposing the GI (requestor) and the appropriate Regulatory office of the outcome at each stage. Issues that proceed through all five stages result in regulatory solutions being provided to Regulatory offices for implementation and verification. The following figure presents an illustration of the GIP in perspective with other regulatory programs and processes. The GIP historical procedures are documented in Appendix G of this report.

Generic Issues Program in Perspective with Other Regulatory Programs and Processes



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

TABLE II

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

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

Legend

NOTES:
1 - Possible Resolution Identified for Evaluation
2 - Resolution Available (Documented in NUREG, NRC Memorandum, SER, or equivalent)
3 - Resolution Resulted in either: (a) The Establishment of New Regulatory Requirements (By Rule, SRP Change, or equivalent) or (b) No New Requirements
4 - Issue to be Prioritized in the Future
5 - Issue that is not a Generic Safety Issue but should be Assigned Resources for Completion

HIGH	- High Safety Priority
MEDIUM	- Medium Safety Priority
LOW	- Low Safety Priority
DROP	- Issue Dropped as a Generic Issue
EI	- Environmental Issue
I	- Resolved TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737
LI	- Licensing Issue
MPA	- Multiplant Action
NA	- Not Applicable
RI	- Regulatory Impact Issue
S	- Issue Covered in an NRC Program Outside the Scope of This Document
USI	- Unresolved Safety Issue
Continue	- As defined in NRC Management Directive 6.4[1858]

Table II

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
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
<u>I.E</u>	<u>ANALYSIS AND DISSEMINATION OF OPERATING EXPERIENCE</u>						
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
<u>I.F</u>	<u>QUALITY ASSURANCE</u>						
I.F.1	Expand QA List	J. Pittman	RES/DRA/ARGIB	NOTE 3(b)	4	12/31/98	NA
I.F.2	Develop More Detailed QA Criteria	-	-	-			
I.F.2(1)	Assure the Independence of the Organization Performing the Checking Function	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	J. Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	4	12/31/98	NA
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	J. Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	4	12/31/98	NA
I.F.2(4)	Establish Criteria for Determining QA Requirements for Specific Classes of Equipment	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA
I.F.2(5)	Establish Qualification Requirements for QA and QC Personnel	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA
I.F.2(6)	Increase the Size of Licensees' QA Staff	J. Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	4	12/31/98	NA
I.F.2(7)	Clarify that the QA Program Is a Condition of the Construction Permit and Operating License	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/31/98	NA
I.F.2(8)	Compare NRC QA Requirements with Those of Other	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	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Agencies						
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	J. Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	4	12/31/98	NA
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/30/98	NA
I.F.2(11)	Define Role of QA in Design and Analysis Activities	J. Pittman	OIE/DQASIP/QUAB	LOW	4	12/30/98	NA
<u>I.G</u>	<u>PREOPERATIONAL AND LOW-POWER TESTING</u>						
I.G.1	Training Requirements	-	NRR/DHFS/PSRB	I	3	12/31/97	
I.G.2	Scope of Test Program	H. Vandermolen	NRR/DHFS/PSRB	NOTE 3(a)	3	12/31/97	NA
<u>II.A</u>	<u>SITING</u>						
II.A.1	Siting Policy Reformulation	H. Vandermolen	NRR/DE/SAB	NOTE 3(b)	2	12/31/97	NA
II.A.2	Site Evaluation of Existing Facilities	H. Vandermolen	NRR/DE/SAB	V.A.1	2	12/31/97	NA
<u>II.B</u>	<u>CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW</u>						
II.B.1	Reactor Coolant System Vents	-	NRR/DL	I	4	12/31/97	F-10
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	-	NRR/DL	I	4	12/31/97	F-11
II.B.3	Post-Accident Sampling	-	NRR/DL	I	4	12/31/97	F-12
II.B.4	Training for Mitigating Core Damage	-	NRR/DL	I	4	12/31/97	F-13
II.B.5	Research on Phenomena Associated with Core Degradation and Fuel Melting	-	-	-			
II.B.5(1)	Behavior of Severely Damaged Fuel	H. Vandermolen	RES/DSR/AEB	LI (NOTE 5)	4	12/31/97	NA
II.B.5(2)	Behavior of Core-Melt	H. Vandermolen	RES/DSR/AEB	LI (NOTE 5)	4	12/31/97	NA
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structure	H. Vandermolen	RES/DSR/AEB	LI (NOTE 5)	4	12/31/97	NA
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	J. Pittman	NRR/DST/RRAB	NOTE 3(a)	4	12/31/97	
II.B.7	Analysis of Hydrogen Control	P. Matthews	NRR/DSI/CSB	II.B.8	4	12/31/97	
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	H. Vandermolen	RES/DRAO/RAMR	NOTE 3(a)	4	12/31/97	
<u>II.C</u>	<u>RELIABILITY ENGINEERING AND RISK ASSESSMENT</u>						
II.C.1	Interim Reliability Evaluation Program	J. Pittman	RES/DRAO/RRB	NOTE 3(b)	3	12/31/97	NA
II.C.2	Continuation of Interim Reliability Evaluation Program	J. Pittman	NRR/DST/RRAB	NOTE 3(b)	3	12/31/97	NA

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
II.J	GENERAL IMPLICATIONS OF TMI FOR DESIGN AND CONSTRUCTION ACTIVITIES						
II.J.1	Vendor Inspection Program						
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.1.2	Modify Existing Vendor Inspection Program	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.2	Construction Inspection Program						
II.J.2.1	Reorient Construction Inspection Program	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.2.3	Assign Resident Inspectors to All Construction Sites	L. Riani	OIE/DQASIP	LI (NOTE 3)	1	12/31/98	NA
II.J.3	Management for Design and Construction						
II.J.3.1	Organization and Staffing to Oversee Design and Construction	J. Pittman	NRR/DHFS/LQB	I.B.1.1	1	12/31/98	NA
II.J.3.2	Issue Regulatory Guide	J. Pittman	NRR/DHFS/LQB	I.B.1.1	1	12/31/98	NA
II.J.4	Revise Deficiency Reporting Requirements						
II.J.4.1	Revise Deficiency Reporting Requirements	L. Riani	AEOD/DSP/ROAB	NOTE 3(a)	3	12/31/98	NA
II.K	MEASURES TO MITIGATE SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS AND LOSS-OF-FEEDWATER ACCIDENTS						
II.K.1	IE Bulletins	-	-	-			
II.K.1(1)	Review TMI-2 PNs and Detailed Chronology of the TMI-2 Accident	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(2)	Review Transients Similar to TMI-2 That Have Occurred at Other Facilities and NRC Evaluation of Davis-Besse Event	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(4)	Review Operating Procedures and Training	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	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Instructions						
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 Reduce Likelihood of Automatic PZR PORV Actuation in Transients	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(20)	Provide Procedures and Training to Operators for Prompt Manual Reactor Trip for LOFW, TT, MSIV Closure, LOOP, LOSG Level, and LO PZR Level	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor	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	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Trip for LOFW, TT, or Significant Decrease in SG Level						
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(24)	Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and RCP Trip	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(25)	Develop Operator Action Guidelines	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2	Commission Orders on B&W Plants	-	-	-			
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System	R. Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(2)	Procedures and Training to Initiate and Control AFW Independent of Integrated Control System	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	R. Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(4)	Small-Break LOCA Analysis, Procedures and Operator Training	R. Emrit	NRR/DHFS/OLB	NOTE 3(a)		12/31/84	-
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	R. Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2(6)	Reevaluate Analysis for Dual-Level Setpoint Control	R. Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(7)	Reevaluate Transient of September 24, 1977	R. Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(8)	Continued Upgrading of AFW System	R. Emrit	NRR	II.E.1.1, II.E.1.2		12/31/84	NA
II.K.2(9)	Analysis and Upgrading of Integrated Control System	R. Emrit	NRR	I		12/31/84	F-27
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	R. Emrit	NRR	I		12/31/84	F-28
II.K.2(11)	Operator Training and Drilling	R. Emrit	NRR	I		12/31/84	F-29
II.K.2(12)	Transient Analysis and Procedures for Management of Small Breaks	R. Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.2(13)	Thermal-Mechanical Report on Effect of HPI on Vessel Integrity for Small-Break LOCA With No AFW	R. Emrit	NRR	I		12/31/84	F-30
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	R. Emrit	NRR	I		12/31/84	F-31
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes After Primary System Voiding	R. Emrit	NRR	I		12/31/84	-
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break	R. Emrit	NRR	I		12/31/84	F-32

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Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	LOCA With Loss of Offsite Power						
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	-
II.K.3	Final Recommendations of Bulletins and Orders Task Force	-	-	-			
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	R. Emrit	NRR	I		12/31/84	F-36
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	R. Emrit	NRR	I		12/31/84	F-37
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	R. Emrit	NRR	I		12/31/84	F-38
II.K.3(4)	Review and Upgrade Reliability and Redundancy of Non-Safety Equipment for Small-Break LOCA Mitigation	R. Emrit	NRR	II.C.1, II.C.2, II.C.3		12/31/84	NA
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	R. Emrit	NRR	I		12/31/84	F-39, G-01
II.K.3(6)	Instrumentation to Verify Natural Circulation	R. Emrit	NRR/DSI	I.C.1(3), II.F.2, II.F.3		12/31/84	NA
II.K.3(7)	Evaluation of PORV Opening Probability During Overpressure Transient	R. Emrit	NRR	I		12/31/84	-
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs	R. Emrit	NRR/DST/GIB	II.C.1, II.E.3.3		12/31/84	NA
II.K.3(9)	Proportional Integral Derivative Controller Modification	R. Emrit	NRR	I		12/31/84	F-40
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	R. Emrit	NRR	I		12/31/84	F-41
II.K.3(11)	Control Use of PORV Supplied by Control Components, Inc. Until Further Review Complete	R. Emrit	NRR	I		12/31/84	-
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine	R. Emrit	NRR	I		12/31/84	F-42

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Trip						
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
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

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
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
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

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
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
<u>III.A</u>	<u>EMERGENCY PREPAREDNESS AND RADIATION EFFECTS</u>						
<u>III.A.1</u>	<u>Improve Licensee Emergency Preparedness - Short-Term</u>						
III.A.1.1	Upgrade Emergency Preparedness	-		-			
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	-	OIE/DEPER/EPB I		2	06/30/91	
III.A.1.1(2)	Perform an Integrated Assessment of the Implementation	-	OIE/DEPER/EPB	NOTE 3(b)	2	06/30/91	NA
III.A.1.2	Upgrade Licensee Emergency Support Facilities	-	-	-	2	06/30/91	
III.A.1.2(1)	Technical Support Center	-	OIE/DEPER/EPB	I	2	06/30/91	F-63
III.A.1.2(2)	On-Site Operational Support Center	-	OIE/DEPER/EPB I		2	06/30/91	F-64
III.A.1.2(3)	Near-Site Emergency Operations Facility	-	OIE/DEPER/EPB I		2	06/30/91	F-65
III.A.1.3	Maintain Supplies of Thyroid-Blocking Agent	-	-	-	2	06/30/91	
III.A.1.3(1)	Workers	R. Riggs	OIE/DEPER/EPB	NOTE 3(b)	2	06/30/91	NA
III.A.1.3(2)	Public	R. Riggs	OIE/DEPER/EPB	NOTE 3(b)	2	06/30/91	NA
<u>III.A.2</u>	<u>Improving Licensee Emergency Preparedness - Long-Term</u>						
III.A.2.1	Amend 10 CFR 50 and 10 CFR 50, Appendix E	-	-	-			
III.A.2.1(1)	Publish Proposed Amendments to the Rules	-	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
<u>III.A.3</u>	<u>Improving NRC Emergency Preparedness</u>						
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	R. Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	06/30/85	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Regional Offices						
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 Requests	J. Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.2(2)	Provide Training for Members of the Technical Staff	J. Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.D	RADIATION PROTECTION						
III.D.1	Radiation Source Control						
III.D.1.1	Primary Coolant Sources Outside the Containment Structure	-	-	-			
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining	-	NRR	I	1	12/31/88	

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	to Reducing Leakage from Operating Systems						
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
<u>III.D.2</u>	<u>Public Radiation Protection Improvement</u>						
III.D.2.1	Radiological Monitoring of Effluents	-	-	-			
III.D.2.1(1)	Evaluate the Feasibility and Perform a Value-Impact Analysis of Modifying Effluent-Monitoring Design Criteria	R. Emrit	NRR/DSI/METB	LOW	3	12/31/98	NA
III.D.2.1(2)	Study the Feasibility of Requiring the Development of Effective Means for Monitoring and Sampling Noble Gases and Radioiodine Released to the Atmosphere	R. Emrit	NRR/DSI/METB	LOW	3	12/31/98	NA
III.D.2.1(3)	Revise Regulatory Guides	R. Emrit	NRR/DSI/METB	LOW	3	12/31/98	NA
III.D.2.2	Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis	-	-	-			
III.D.2.2(1)	Perform Study of Radioiodine, Carbon-14, and Tritium Behavior	R. Emrit	NRR/DSI/RAB	NOTE 3(b)	3	12/31/98	NA
III.D.2.2(2)	Evaluate Data Collected at Quad Cities	R. Emrit	NRR/DSI/RAB	III.D.2.5	3	12/31/98	NA
III.D.2.2(3)	Determine the Distribution of the Chemical Species of Radioiodine in Air-Water-Steam Mixtures	R. Emrit	NRR/DSI/RAB	III.D.2.5	3	12/31/98	NA
III.D.2.2(4)	Revise SRP and Regulatory Guides	R. Emrit	NRR/DSI/RAB	III.D.2.5	3	12/31/98	NA
III.D.2.3	Liquid Pathway Radiological Control	-	-	-			
III.D.2.3(1)	Develop Procedures to Discriminate Between Sites/Plants	R. Emrit	NRR/DE/EHEB	NOTE 3(b)	3	12/31/98	NA
III.D.2.3(2)	Discriminate Between Sites and Plants That Require Consideration of Liquid Pathway Interdiction Techniques	R. Emrit	NRR/DE/EHEB	NOTE 3(b)	3	12/31/98	NA
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction	R. Emrit	NRR/DE/EHEB	NOTE 3(b)	3	12/31/98	NA
III.D.2.3(4)	Prepare a Summary Assessment	R. Emrit	NRR/DE/EHEB	NOTE 3(b)	3	12/31/98	NA
III.D.2.4	Offsite Dose Measurements	-	-	-			
III.D.2.4(1)	Study Feasibility of Environmental Monitors	H. Vandermolen	NRR/DSI/RAB	NOTE 3(b)	3	12/31/98	NA
III.D.2.4(2)	Place 50 TLDs Around Each Site	H. Vandermolen	OIE/DRP/ORPB	LI (NOTE 3)	3	12/31/98	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
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
III.D.3	Worker Radiation Protection Improvement						
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
IV.A	STRENGTHEN ENFORCEMENT PROCESS						
IV.A.1	Seek Legislative Authority	R. Emrit	GC	LI (NOTE 3)		11/30/83	NA
IV.A.2	Revise Enforcement Policy	R. Emrit	OIE/ES	LI (NOTE 3)		11/30/83	NA
IV.B	ISSUANCE OF INSTRUCTIONS AND INFORMATION TO LICENSEES						
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	R. Emrit	OIE/DEPER	LI (NOTE 3)		11/30/83	NA
IV.C	EXTEND LESSONS LEARNED TO LICENSED ACTIVITIES OTHER THAN POWER REACTORS						

Table II (Continued)

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

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	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
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

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	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 Dyanmic Loads Long-Term Program (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-9	ATWS (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-10	BWR Feedwater Nozzle Cracking (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-25
A-11	Reactor Vessel Materials Toughness (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	NA
A-13	Snubber Operability Assurance	R. Emrit	NRR/DE/MEB	NOTE 3(a)	1	06/30/91	B-17, B-22
A-14	Flaw Detection	P. Matthews	NRR/DE/MTEB	DROP		11/30/83	NA
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning	J. Pittman	NRR/DE/CHEB	NOTE 3(b)		11/30/83	NA
A-16	Steam Effects on BWR Core Spray Distribution	R. Emrit	NRR/DSI/CPB	NOTE 3(a)		11/30/83	D-12
A-17	Systems Interactions in Nuclear Power Plants (former (USI)	R. Emrit	RES/DSIR/EIB	NOTE 3(b)	1	12/31/89	NA
A-18	Pipe Rupture Design Criteria	R. Emrit	NRR/DE/MEB	DROP		11/30/83	NA
A-19	Digital Computer Protection System	W. Milstead	RES/DSR/HFB	LI (NOTE 5)	1	06/30/91	NA
A-20	Impacts of the Coal Fuel Cycle	-	NRR/DE/EHEB	LI (NOTE 5)		11/30/83	NA
A-21	Main Steamline Break Inside Containment - Evaluation of Environmental Conditions for Equipment Qualification	H. Vandermolen	NRR/DSI/CSB	DROP	1	12/31/98	NA
A-22	PWR Main Steamline Break - Core, Reactor Vessel and Containment Building Response	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

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Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
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
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

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Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
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 Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems	R. Emrit	NRR/DSI/METB	NOTE 3(a)		11/30/83	
B-37	Chemical Discharges to Receiving Waters	-	NRR/DE/EHEB	EI (NOTE 5)		11/30/83	
B-38	Reconnaissance Level Investigations	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-39	Transmission Lines	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-40	Effects of Power Plant Entrainment on Plankton	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-41	Impacts on Fisheries	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-42	Socioeconomic Environmental Impacts	-	NRR/DE/SAB	EI (NOTE 3)		11/30/83	NA

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Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
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	
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	D. Thatcher	NRR/DE/MEB	C-12		11/30/83	NA

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Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Pressure Vessel						
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						
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

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Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
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 Retaining Rings	J. Pittman	NRR/DE/MEB	DROP	1	12/31/85	NA
31.	Natural Circulation Cooldown	R. Riggs	NRR/DSI/RSB	I.C.1		11/30/83	NA
32.	Flow Blockage in Essential Equipment Caused by Corbicula	R. Emrit	NRR/DSI/ASB	51		11/30/83	NA
33.	Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power	J. Pittman	NRR/DSI/ICSB	A-47		11/30/83	NA
34.	RCS Leak	R. Riggs	NRR/DHFS/PSRB	DROP	1	06/30/84	NA
35.	Degradation of Internal Appurtenances in LWRs	H. Vandermolen	NRR/DSI/CPB, RSB	DROP	2	12/31/98	NA
36.	Loss of Service Water	L. Riani	NRR/DSI/ASB, AEB, RSB	NOTE 3(b)	3	06/30/91	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
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
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	R. Emrit	NRR/DST/TSIP	RI (NOTE 5)	1	06/30/85	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Inoperable						
60.	Lamellar Tearing of Reactor Systems Structural Supports	L. Riani	NRR/DST/GIB	A-12		11/30/83	NA
61.	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments	W. Milstead	NRR/DSI/CSB	NOTE 3(b)	2	12/31/86	NA
62.	Reactor Systems Bolting Applications	R. Riggs	RES/DSIR/EIB	29	1	12/31/88	NA
63.	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis	J. Pittman	RES/DRA/ARGIB	DROP	1	06/30/90	NA
64.	Identification of Protection System Instrument Sensing Lines	D. Thatcher	NRR/DSI/ICSB	NOTE 3(b)		11/30/83	
65.	Probability of Core-Melt Due to Component Cooling Water System Failures	H. Vandermolen	NRR/DSI/ASB	23	1	12/31/86	NA
66.	Steam Generator Requirements	R. Riggs	NRR/DEST/EMTB	NOTE 3(b)	2	12/31/88	NA
67.	<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, NRR/DSI/RSB	A-45, I.C.1 (2,3)	4	06/30/94	NA
67.10.0	Supplemental Tube Inspections	R. Riggs	NRR/DL/ORAB	LI (NOTE 5)	4	06/30/94	NA
68.	Postulated Loss of Auxiliary Feedwater System Resulting from Turbine-Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	J. Pittman	NRR/DSI/ASB	124	3	06/30/91	NA
69.	Make-up Nozzle Cracking in B&W Plants	R. Colmar	NRR/DE/MEB, MTEB	NOTE 3(b)	1	12/31/84	B43
70.	PORV and Block Valve Reliability	R. Riggs	RES/DE/EIB	NOTE 3(a)	3	06/30/91	
71.	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety	J. Pittman	RES/DRA/ARGIB	DROP	3	06/30/01	NA
72.	Control Rod Drive Guide Tube Support Pin Failures	R. Riggs	RES	DROP	1	06/30/91	NA

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Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
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 Back-flow Through Floor Drains	L. Riani	RES/DE/EIB	A-17		12/31/87	NA
78.	Monitoring of Fatigue Transient Limits for Reactor Coolant System	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 Personnel Safety	C. Rourk	RES/DSIR/EIB	LOW	4	06/30/95	NA
82.	Beyond Design Basis Accidents in Spent Fuel Pools	H. Vandermolen	RES/DRPS/RPSI	NOTE 3(b)	3	06/30/04	NA
83.	Control Room Habitability	R. Emrit	RES/DST/AEB	NOTE 3(b)	3	06/30/03	NA
84.	CE PORVs	R. Riggs	RES/DSIR/RPSI	NOTE 3(b)	2	06/30/90	NA
85.	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	W. Milstead	NRR/DSI/CSB	DROP	2	06/30/91	NA
86.	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	R. Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	06/30/88	B-84
87.	Failure of HPCI Steam Line Without Isolation	J. Pittman	RES/DSIR/EIB	NOTE 3(a)	2	06/30/95	
88.	Earthquakes and Emergency Planning	R. Riggs	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
89.	Stiff Pipe Clamps	T.Y. Chang	RES/DSIR/EIB	LOW	2	06/30/95	NA
90.	Technical Specifications for Anticipatory Trips	H. Vandermolen	NRR/DSI/RSB,	DROP	2	12/31/98	NA

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Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
			ICSB				
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
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

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Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
118.	Tendon Anchorage Failure	S. Shaukat	RES/DSIR/EIB	NOTE 3(a)	1	06/30/95	NA
119.	Piping Review Committee Recommendations	-	-	-			
119.1	Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads	R. Riggs	NRR/DE	RI (NOTE 3)	3	12/31/97	NA
119.2	Piping Damping Values	R. Riggs	NRR/DE	RI (DROP)	3	12/31/97	NA
119.3	Decoupling the OBE from the SSE	R. Riggs	NRR/DE	RI (S)	3	12/31/97	NA
119.4	BWR Piping Materials	R. Riggs	NRR/DE	RI (NOTE 5)	3	12/31/97	NA
119.5	Leak Detection Requirements	R. Riggs	NRR/DE	RI (NOTE 5)	3	12/31/97	NA
120.	On-Line Testability of Protection Systems	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)	2	06/30/95	NA
121.	Hydrogen Control for Large, Dry PWR Containments	R. Emrit	RES/DSIR/SAIB	NOTE 3(b)	2	06/30/95	NA
122.	Davis-Besse Loss of All Feedwater Event of June 9, 1985: Short-Term Actions						
122.1	Potential Inability to Remove Reactor Decay Heat	-	-	-			
122.1.a	Failure of Isolation Valves in Closed Position	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	
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	-	-	-	7	12/31/98	
125.I.2.a	Need for a Test Program to Establish Reliability of the PORV	H. Vandermolen	NRR/DSRO/SPEB	70	7	12/31/98	NA
125.I.2.b	Need for PORV Surveillance Tests to Confirm Operational Readiness	H. Vandermolen	NRR/DSRO/SPEB	70	7	12/31/98	NA
125.I.2.c	Need for Additional Protection Against PORV Failure	H. Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.I.2.d	Capability of the PORV to Support Feed-and-Bleed	H. Vandermolen	NRR/DSRO/SPEB	A-45	7	12/31/98	NA
125.I.3	SPDS Availability	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)	7	12/31/98	NA
125.I.4	Plant-Specific Simulator	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.5	Safety Systems Tested in All Conditions Required by DBA	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.6	Valve Torque Limit and Bypass Switch Settings	H. Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.7	Operator Training Adequacy	-	-	-			
125.I.7.a	Recover Failed Equipment	J. Pittman	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.I.7.b	Realistic Hands-On Training	H. Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA

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Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
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
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

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Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
133.	Update Policy Statement on Nuclear Plant Staff Working Hours	J. Pittman	NRR/DLPQ/LHFB	LI (NOTE 3)	1	12/31/91	NA
134.	Rule on Degree and Experience Requirement	J. Pittman	RES/DRA/RDB	NOTE 3(b)		12/31/89	NA
135.	Steam Generator and Steam Line Overfill	R. Emrit	RES/DSIR/EIB	NOTE 3(b)	3	06/30/95	NA
136.	Storage and Use of Large Quantities of Cryogenic Combustibles On Site	W. Milstead	RES/DRA/ARGIB	LI (NOTE 3)		06/30/88	NA
137.	Refueling Cavity Seal Failure	W. Milstead	RES/DRA/ARGIB	DROP		06/30/90	NA
138.	Deinerting of BWR Mark I and II Containments During Power Operations Upon Discovery of RCS Leakage or a Train of a Safety System Inoperable	W. Milstead	RES/DSIR/SAIB	DROP	2	12/31/98	NA
139.	Thinning of Carbon Steel Piping in LWRs	R. Riggs	RES/DRA/ARGIB	RI (NOTE 3)	1	06/30/95	NA
140.	Fission Product Removal Systems	R. Riggs	RES/DRA/ARGIB	DROP		06/30/90	NA
141.	Large-Break LOCA With Consequential SGTR	R. Riggs	RES/DRA/ARGIB	DROP		06/30/90	NA
142.	Leakage Through Electrical Isolators in Instrumentation Circuits	W. Milstead	RES/DSIR/EIB	NOTE 3(b)	4	12/31/97	NA
143.	Availability of Chilled Water Systems and Room Cooling	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)	2	06/30/95	NA
144.	Scram Without a Turbine/Generator Trip	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
155.	<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
156.	<u>Systematic Evaluation Program</u>	-	-	-			

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
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
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	HIGH	1	06/30/08	
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

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
169.	BWR MSIV Common Mode Failure Due to Loss of Accumulator Pressure	R. Emrit	RES/DET/GSIB	DROP	1	06/30/00	NA
170.	Fuel Damage Criteria for High Burnup Fuel	R. Emrit	RES/DET/GSIB	NOTE 3(b)	2	06/30/01	NA
171.	ESF Failure from LOOP Subsequent to a LOCA	C. Rourk	RES/DET/GSIB	NOTE 3(b)	1	12/31/98	NA
172.	Multiple System Responses Program	R. Emrit	RES/DET/GSIB	NOTE 3(b)	2	06/30/02	NA
173.	Spent Fuel Storage Pool	-	-				
173.A	Operating Facilities	R. Emrit	RES/DET/GSIB	NOTE 3(b)	4	06/30/02	NA
173.B	Permanently Shutdown Facilities	R. Emrit	RES/DET/GSIB	NOTE 3(b)	4	06/30/02	NA
174.	Fastener Gaging Practices	-	-				
174.A	SONGS Employees' Concern	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
174.B	Johnson Gage Company Concern	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
175.	Nuclear Power Plant Shift Staffing	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
176.	Loss of Fill-Oil in Rosemount Transmitters	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
177.	Vehicle Intrusion at TMI	R. Emrit	RES/DET/GSIB	NOTE 3(a)	1	06/30/00	NA
178.	Effect of Hurricane Andrew on Turkey Point	R. Emrit	RES/DET/GSIB	LI (NOTE 3)	2	06/30/00	
179.	Core Performance	R. Emrit	RES/DET/GSIB	LI (NOTE 5)	1	06/30/00	
180.	Notice of Enforcement Discretion	R. Emrit	RES/DET/GSIB	LI (NOTE 3)	1	06/30/00	
181.	Fire Protection	R. Emrit	RES/DET/GSIB	LI (NOTE 5)	1	06/30/00	
182.	General Electric Extended Power Uprate	R. Emrit	RES/DET/GSIB	RI (NOTE 5)	1	06/30/00	
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	CONTINUE		06/30/04	
187.	The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump in Nuclear Power Plants	H. Vandermolen	RES/DSARE/REAHFB	DROP		06/30/01	NA
188.	Steam Generator Tube Leaks/Ruptures Concurrent with Containment Bypass	H. Vandermolen	RES/DSARE/REAHFB	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	CONTINUE	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/SSI	HIGH	2	06/30/08	
192.	Secondary Containment Drawdown Time	H. Vandermolen	RES/DSARE/REAHFB	DROP		06/30/03	NA
193.	BWR ECCS Suction Concerns	P. Kadambi	RES/DSA/NARB	CONTINUE		06/30/04	
194.	Implications of Updated Probabilistic Seismic Hazard	D. Harrison	NRR/DSSA/SPSB	DROP		06/30/04	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	Estimates						
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	L. Killian	RES/DRA/OEGIB	CONTINUE		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
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

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
HF4.5	Application of Automation and Artificial Intelligence	J. Pittman	NRR/DHFT/HFIB	HF5.2	6	06/30/95	NA
HF5	MAN-MACHINE INTERFACE						
HF5.1	Local Control Stations	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	4	06/30/95	NA
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	4	06/30/95	NA
HF5.3	Evaluation of Operational Aid Systems	J. Pittman	NRR/DHFT/HFIB	HF5.2	4	06/30/95	NA
HF5.4	Computers and Computer Displays	J. Pittman	NRR/DHFT/HFIB	HF5.2	4	06/30/95	NA
HF6	MANAGEMENT AND ORGANIZATION						
HF6.1	Develop Regulatory Position on Management and Organization	J. Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF6.2	Regulatory Position on Management and Organization at Operating Reactors	J. Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF7	HUMAN RELIABILITY						
HF7.1	Human Error Data Acquisition	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.2	Human Error Data Storage and Retrieval	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.3	Reliability Evaluation Specialist Aids	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.4	Safety Event Analysis Results Applications	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF8	Maintenance and Surveillance Program	J. Pittman	NRR/DLPQ/LPEB	NOTE 3(b)	2	06/30/88	NA
	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	-	-				

Table II (Continued)

Action Plan Item/ Issue No.	Title	Responsible Project Manager	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
CH1.4A	Engineered Safety Feature Availability	R. Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.4B	Technical Specifications Bases	R. Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.4C	Low Power and Shutdown	R. Emrit	RES/DSR/PRAB	LI (NOTE 5)		06/30/89	NA
CH1.5	Operating Staff Attitudes Toward Safety	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH1.6	Management Systems	-	-				
CH1.6A	Assessment of NRC Requirements on Management	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.7	Accident Management	-	-				
CH1.7A	Accident Management	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
<u>CH2</u>	<u>DESIGN</u>						
CH2.1	Reactivity Accidents	-	-				
CH2.1A	Reactivity Transients	R. Emrit	RES/DSR/RPSB	LI (NOTE 5)		06/30/89	NA
CH2.2	Accidents at Low Power and at Zero Power	R. Emrit	RES/DRA/ARGIB	CH1.4		06/30/89	NA
CH2.3	Miltiple-Unit Protection	-	-				
CH2.3A	Control Room Habitability	R. Emrit	RES/DRA/ARGIB	83		06/30/89	NA
CH2.3B	Contamination Outside Control Room	R. Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.3C	Smoke Control	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH2.3D	Shared Shutdown Systems	R. Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.4	Fire Protection	-	-				
CH2.4A	Firefighting With Radiation Present	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
<u>CH3</u>	<u>CONTAINMENT</u>						
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
<u>CH4</u>	<u>EMERGENCY PLANNING</u>						
CH4.1	Size of the Emergency Planning Zones	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH4.2	Medical Services	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH4.3	Ingestion Pathway Measures	-	-				
CH4.3A	Ingestion Pathway Protective Measures	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4.4	Decontamination and Relocation	-	-				
CH4.4A	Decontamination	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4.4B	Relocation	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
<u>CH5</u>	<u>SEVERE ACCIDENT PHENOMENA</u>						

Table II (Continued)

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

TABLE III

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

Legend

NOTES:	1 - Possible Resolution Identified for Evaluation
	2 - Resolution Available
	3 - Resolution Resulted in either the Establishment of New Requirements or No New Requirements
	4 - Issues to be Prioritized in the Future
	5 - Issues that are not GSIs but Should be Assigned Resources for Completion
DROP	- GSI Dropped from Further Pursuit
EI	- Environmental Issue
GSI	- Generic Safety Issue
HIGH	- High Safety Priority
I	- TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737
LI	- Licensing Issue
LOW	- Low Safety Priority
MEDIUM	- Medium Safety Priority
RI	- Regulatory Impact Issue
S	- Issue Covered in an NRC Program Outside the Scope of This Document
USI	- Unresolved Safety Issue
Continue	- As defined in NRC Management Directive 6.4[1858]

TABLE III (Continued)

ACTION ITEM/ISSUE GROUP	I	S	RESOLVED STAGES			USI	HIGH	MEDIUM	LOW	DROP	CONT.	NOTE 4	NOTE 5	TOTAL
			NOTE 1	NOTE 2	NOTE 3									
TMI ACTION PLAN ITEM (369)														
GSI	84	46	0	0	135	0	0	0	12	9	-	-	-	286
LI	-	0	-	-	75	-	-	-	-	-	-	-	8	83
TASK ACTION PLAN ITEMS (142)														
USI	-	-	-	-	27	0	-	-	-	-	-	-	-	27
GSI	-	20	0	0	36	-	0	0	0	14	-	-	-	70
RI	-	-	-	-	6	-	-	-	-	-	-	-	1	7
LI	-	-	-	-	11	-	-	-	-	-	-	-	12	23
EI	-	-	-	-	13	-	-	-	-	-	-	-	2	15
NEW GENERIC ISSUES (283)														
GSI	-	54	0	0	88	0	2	0	4	105	4	-	-	257
RI	-	1	-	-	5	-	-	-	-	1	-	-	5	12
LI	-	1	-	-	8	-	-	-	-	-	-	-	4	13
EI	-	-	-	-	-	-	-	-	-	-	-	-	1	1
HUMAN FACTORS ISSUES (27)														
GSI	-	8	0	0	8	0	0	0	0	0	-	-	-	16
LI	-	-	-	-	3	-	-	-	-	-	-	-	8	11
CHERNOBYL ISSUES (32)														
LI	-	2	-	-	7	-	-	-	-	-	-	-	23	32
TOTAL:	84	132	0	0	422	0	2	0	16	129	4	0	64	853

ISSUE 156: SYSTEMATIC EVALUATION PROGRAM

In 1977, the NRC initiated the Systematic Evaluation Program (SEP) to review the designs of 51 older, operating nuclear power plants. The SEP was divided into 2 phases. In Phase I, the staff defined 137 issues for which regulatory requirements had changed enough over time to warrant an evaluation of those plants licensed before the issuance of the SRP.[11] In Phase II, the staff compared the design of 10 of the 51 older plants to the SRP[11] issued in 1975. Based on these reviews, the staff identified 27 of the original 137 issues that required some corrective action at one or more of the 10 plants that were reviewed. The staff referred to the issues on this smaller list as the SEP "lessons learned" issues and concluded that they would generally apply to operating plants that received operating licenses before the SRP[11] was issued in 1975.

In SECY-84-133,[814] the staff presented the 27 SEP issues to the Commission as part of a proposal for an ISAP, the intent of which was to review safety issues for a specific plant in an integrated manner. Two SEP plants participated in the ISAP pilot efforts. Following the review of these two pilot plants, ISAP was discontinued.

In SECY-90-160,[1443] the staff forwarded for Commission approval a proposed license renewal rule and supporting regulatory documents. In this paper, the staff stated that certain unresolved safety issues could weaken the generic justification of the adequacy of the current licensing bases argument. These issues included SEP topics for 41 older plants that had not been explicitly reviewed under Phase II of the SEP. The Commission requested that the staff keep it informed of the status of the program to determine how the SEP "lessons learned" issues had been factored into the licensing bases of operating plants.

Resolution of the 27 SEP issues was deemed by the staff to be important to the development of the license renewal rulemaking. The key regulatory principle underlying the license renewal rule is that the current licensing bases (CLBs) at all operating nuclear power plants, with the exception of age-related degradation, provide adequate protection to the public health and safety. This principle is reflected in the provisions of the license renewal rule which limit the renewal decision to whether age-related degradation has been adequately addressed to assure continued compliance with a plant's CLB. In order to adopt this approach, the NRC must be able to provide a technical basis for the key principle of license renewal. Accordingly, the rulemaking included a technical discussion documenting the adequacy of the CLB for all nuclear power plants, in both the statement of considerations and in NUREG-1412.[1444] However, as discussed in SECY-90-160,[1443] the staff identified a potential weakness in the discussion of the adequacy of the CLB with regard to the 41 older, non-SEP plants. To address this potential weakness, the staff undertook an effort to determine whether or not each SEP issue either had been or was being addressed by other regulatory programs and activities.

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

ISSUE 156.1.1: SETTLEMENT OF FOUNDATIONS AND BURIED EQUIPMENT

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The objective of this issue was to ensure that safety-related structures, systems, and components were adequately protected against excessive settlement. The scope included the review of subsurface materials (soils or geologic) and foundations to assess the potential static and seismically-induced settlement of all safety-related structures and buried equipment.

Excessive settlement or collapse of foundations and buried equipment for structures, systems, and components under either static or seismic loading could result in failure of structures, interconnecting piping, control systems or cables, or other equipment (tanks, etc.) such that the capability to safely shut down a plant, or mitigate the consequences of an accident, could be compromised.

There were two specific concerns in this issue: (1) the potential impact of static soil settlements on foundations and buried equipment where the soil may not have been properly prepared; and (2) seismically-induced differential settlement and potential soil liquefaction following a postulated seismic event. These two concerns were limited only to plants that have soil-supported, safety-related structures (including vertical, field-erected tanks) and soil-buried piping and components (including tanks) that have the potential for excessive settlement but were not reviewed to the pertinent SRP[11] Sections 2.5.4 and 2.5.5.

For the 41 older, non-SEP plants with OLs issued before 1975, any impact of static settlement on structural foundations (including the foundations of buried components) should become noticeable in the first 5 to 10 years. Thus, any significant settlement would have been revealed already and warranted corrective action. In addition, the ongoing IPEEE program[1354] has elements in its seismic task which requires that, for plants on soil sites, potential seismically-induced settlement and soil liquefaction should be assessed during its implementation.

CONCLUSION

This issue is being addressed by the SRP[11] for future plants as well as for operating plants with OLs issued after 1975. For the 51 older, operating plants, this issue was considered resolved for the 10 SEP plants. For the remaining 41 non-SEP, operating plants, any significant static settlement would have been revealed already and warranted corrective action. The concern on the seismically-induced settlement and soil liquefaction for these 41 older, non-SEP operating plants will be addressed during the implementation of the IPEEE Program. Therefore, Issue 156.1.1 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.1.2: DAM INTEGRITY AND SITE FLOODING

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90 -343.[1351] The safety concern was the ability of a dam to prevent site flooding and ensure a cooling water supply. The safety features of a dam would normally include remaining stable under all conditions of reservoir operation, controlling seepage to prevent excessive uplifting water pressure or erosion of soil materials, and providing sufficient freeboard and outlet capacity to prevent overtopping. The objective of this issue was to ensure that adequate margins of safety are available under all loading conditions and uncontrolled releases of retained water are prevented. Plants must provide the basis for ensuring that all safety-related structures, systems, and components are adequately protected against flooding that might result from dam failures. Further, review of licensee procedures would determine whether an adequate supply of cooling water exists in the ultimate heat sink during normal and emergency operations. The 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

If a dam exists in the vicinity of a nuclear power plant, it will have to meet one of the following criteria:

(1) If the dam provides impoundment for an UHS at a plant or provides flood protection, the dam is an essential part of the plant and the safety of the dam needs to be ensured throughout the life of the plant. The dam has to be designed and remain stable under both static and seismic conditions.[688,916]

(2) If the dam provides impoundment only for plant operation, but not as a part of the UHS, there are no regulatory requirements for dam design. However, the flood conditions that could be caused by dam failures should be considered in establishing the design basis flood.[687] When upstream dams or other features that provide flood protection are present, in addition to the analyses of the most severe floods that may be induced by either hydrometeorological or seismic mechanisms, reasonable combinations of less severe flood conditions and seismic events should be considered in establishing the design basis flood.

The IPEEE Program will address the safety and the flooding effects of dams. Under this program, the safety of dams will be assessed by all licensees in the process of searching for severe accident vulnerabilities due to external events.[1222,1354] If the failure of these dams would have significant consequences, i.e., a breach of an UHS which might lead to a severe accident, they would have to be evaluated and inspected to assess their existing condition and vulnerability to earthquakes. If the failure of an upstream dam could lead to significant flooding at a site, i.e., the postulated flood exceeded the design basis flood and might lead to a severe accident, the effect of flooding will have to be addressed in the IPEEE.

CONCLUSION

The safety concerns of dam integrity and site flooding will be addressed in the implementation of the IPEEE Program at the 41 plants affected by this issue.[1575] Therefore, Issue 156.1.2 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.1.3: SITE HYDROLOGY AND ABILITY TO WITHSTAND FLOODS

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The concerns of this issue included identifying the site hydrologic characteristics, the capability of structures important to safety to withstand flooding, the determination of the adequacy of the cooling water supply, and the ISI of water control structures. Hydrologic considerations are the interface of the plant with the hydrosphere, the identification of hydrologic causal mechanisms that may require special plant design, or operating limitations with regard to floods, and water supply requirements. The specific items to be reviewed in this issue were:

(1) Hydrologic Description - To ensure that plant design reflects appropriate hydrologic conditions.

(2) Flooding Potential and Protection - To ensure that the plant is adequately protected against floods.

(3) Ultimate Heat Sink - To ensure an appropriate supply of cooling water is available during normal and emergency shutdowns.

(4) ISI of Water Control Structures - To ensure an adequate inspection program is in place to prevent water control structure deterioration or failure which could result in flooding or loss of the UHS.

The 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

At a nuclear plant, the safety-related structures, systems, and components, identified in accordance with Regulatory Guide 1.29,[916] must be designed to withstand the conditions resulting from the worst probable site-related flood and retain the capability for shutdown and maintenance.[687] Alternatively, NRC permits licensees not to design against the worst flood conditions for safety-related structures, systems, and components if sufficient warning time is shown to be available to shut down the plant and implement adequate emergency procedures. However, the safety-related structures, systems, and components must be designed to withstand the conditions resulting from a Standard Project Flood (with a flow-rate about 40% to 60% of the PMF).[687]

On June 28, 1991, the NRC requested all licensees to conduct an IPEEE to search for severe accident vulnerabilities due to external events[1222]; external flooding is one of the events that will be addressed in the IPEEE.[1354] All licensees will have to examine the flood designs and associated flood protection measures at their sites to determine if severe accident vulnerabilities due to external floods exist. Therefore, the above Items 1 and 2 have been addressed in the external flood portion of the IPEEE program.

Item 3 is related to maintaining the functioning of the SWS and the DHR system of a plant. The severe accident vulnerability resulting either from failure or unavailability of the UHS is one of the important items to be examined in the IPE and IPEEE programs.

The NRC will require the affected licensees to upgrade their ISI programs for water control structures where inspection findings and any subsequent analyses reveal inadequacies in meeting the intent of Item 4.

CONCLUSION

The safety concerns of site hydrologic characteristics and the capability of plants to withstand flooding will be addressed in the implementation of the IPE and IPEEE Programs at the 41 plants affected by this issue.[1575] Therefore, Issue 156.1.3 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.1.4: INDUSTRIAL HAZARDS

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The objective of this issue was to ensure that the integrity of safety-related structures, components, and systems will not be damaged by potential hazards from nearby transportation, storage, or industrial facilities. Such hazards include: (1) shock waves and thermal flux from nearby explosions of munitions or explosive gases or chemicals; (2) drifting toxic/explosive vapor clouds; (3) aircraft; and (4) missiles that can result from nearby explosions, such as a rocketing chemical tank car. In a few past licensing cases, reactor containment and intake structure hardening and pipeline relocation have been required to ensure safety of the plants. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

Regulatory Guide 4.7[1372] and SRP[11] Sections 2.2.1, 2.2.2, and 2.2.3 have been used since 1975 in the design of nuclear power plants for protection against industrial hazards. In addition, Regulatory Guides 1.78,[1373] 1.91,[1374] and 1.95[1375] were issued to provide further regulatory guidance in this area. Prior to the issuance of these criteria, offsite hazards had been an area of long-standing concern and were reviewed on a case-by-case basis.

Supplement 4 to Generic Letter No. 88-20[1222] required all licensees to conduct an IPEEE to search for severe accident vulnerabilities due to external events. Industrial hazards comprise one of the external events that will be addressed in the IPEEE.[1354]

CONCLUSION

Based on past staff reviews, existing review criteria and guidance, and the implementation of the IPEEE program for all plants, the concern for industrial hazards was adequately addressed. Therefore, Issue 156.1.4 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.1.5: TORNADO MISSILES

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] All plants licensed after 1972 were designed for protection against tornadoes. The concern existed, however, that plants constructed prior to 1972 may not be adequately protected, in particular, those reviewed before 1968 when criteria on tornado protection were first developed. The objective of this issue was to ensure that safety structures, systems, and components can withstand the impact of an appropriate postulated spectrum of tornado-generated missiles. The failure of safety-related structures, systems, or components due to a tornado-induced missile could compromise the ability of a plant to safely shut down. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

A plant must be designed to remain in a safe condition in the event that the most severe tornado that can be reasonably predicted occurs at the plant site as a result of severe meteorological conditions. All safety-related structures, systems, and components must be designed to withstand the effects of the design basis tornado, tornado-generated missiles, and other tornado-induced effects.[42,916]

Under the IPEEE program, all licensees are required to examine their plants to determine if severe accident vulnerabilities due to high winds/tornadoes exist.[1222,1354] The criteria used for plant design (such as the design basis wind speed, parameters of the design basis tornado along with missile spectrum, and the allowable stresses and load combinations) will be examined. The reporting criterion, 10^{-6}/year CDF, specified for the IPEEE, however, is considered to be less stringent compared to the CDF associated with tornado missiles design criteria (a product of combining the probability of exceedance associated with the design basis tornado and the conditional failure probability associated with engineering design and construction against tornado missiles). Therefore, meeting the objectives of the IPEEE does not mean, in this situation, that current NRC guidelines for tornado design have been met. Thus, the staff believes that any vulnerability associated with tornado missiles will be evaluated and reported in the IPEEE submittals.

CONCLUSION

The safety concern for tornado missiles will be addressed in the implementation of the IPEEE Program at the 41 plants affected by this issue. Therefore, Issue 156.1.5 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.1.6: TURBINE MISSILES

DESCRIPTION

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

As a result of turbine disc failures at two nuclear plants and a number of non-nuclear plants prior to 1973, the staff believed that high energy missiles could be generated from steam turbines with the potential for causing failures in safety-related systems. The two areas of concern were: (1) failures at design overspeed because of degraded disc material, poor ISI of flaws, or chemistry conditions leading to SCC; and (2) destructive overspeed failures that would bring into question the reliability of electrical overspeed protection systems, the reliability and testing programs for stop and control valves, and the ISI of valves. For plants licensed after 1973, the safety concerns of this issue were reviewed by the staff as part of its OL activities; turbine overspeed protection designs were found acceptable and the magnitude of the potential damage from turbine missiles was determined to be plant-specific.

CONCLUSION

The safety concerns of this issue were addressed in the evaluation of Issue A-37, which focused primarily on plants licensed prior to November 1976; SRP[11] requirements for turbine design were issued for use by CP applicants after this date. Based on the historical failure rate of turbines used in the evaluation, Issue A-37 was determined to have little safety significance. No new data were provided in SECY-90-343[1351] that changed this conclusion. Therefore, this issue was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.2.1: SEVERE WEATHER EFFECTS ON STRUCTURES

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] Safety-related structures, systems, and components should be designed to function under all severe weather conditions to which they may be exposed. Meteorological phenomena to be considered include straight winds, tornadoes, snow and ice loads, and other phenomena judged to be significant for a particular site. The objective of this issue was to identify those meteorological conditions which should be considered in the structural reviews to determine the ability of structures to withstand conditions such as flooding, wind, tornadoes, hurricanes, tsunamis, and seiches. The dynamic effects of waves, tornado pressure drop loading, and possible in-leakage due to floods were to be considered. The 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

A nuclear power plant must be designed to remain in a safe condition in the event that the most severe weather conditions that can reasonably be predicted at the site occurs. All the safety-related structures must be designed to withstand the effects of the design basis flood, wind, hurricane, tornado, wind/tornado-generated missiles, and other wind/tornado-induced effects.[916]

Under the IPEEE Program, all licensees were requested to examine their plants to determine if severe accident vulnerabilities due to floods or high winds/tornadoes exist.[1222,1354] Licensees were expected to examine their design criteria (such as the design flood level, the hydrostatic pressures against the structures, the design basis wind speed, parameters of the design basis tornado along with missile spectrum, and the allowable stresses and load combinations) used for plant structures to determine if the 1975 SRP[11] criteria are satisfied. If a plant conforms to these criteria, it will be judged that the contribution to CDF from the effects of severe weather is less than 10^{-6}/year and the IPEEE screening criterion would be met. Otherwise, additional evaluation will have to be made to establish severe accident vulnerabilities due to the effects of severe weather. The reporting criterion of 10^{-6}/year CDF specified for the IPEEE will provide a means by which the ability of a nuclear power plant to withstand severe weather conditions can be reviewed and examined for severe weather-induced vulnerabilities.

Snow and ice loads, when accompanied by strong winds, have caused several complete and partial losses of offsite power and the potential of causing severe accidents at a particular site will be evaluated in the IPE program. Snow and ice loads alone, are judged, based on limited PRA experience, to be unlikely to cause significant structural failure that might lead to severe accidents at nuclear power plants.

CONCLUSION

The safety concern of severe weather effects on structures will be addressed in the implementation of the IPEEE program. Therefore, Issue 155.2.1 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.2.2: DESIGN CODES, CRITERIA, AND LOAD COMBINATIONS

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] With the development of nuclear power, provisions addressing nuclear power plants were progressively introduced into codes and standards to which plant buildings and structures are constructed. Because of this evolutionary development, older nuclear power plants conform to a number of different versions of codes and standards, some of which have since undergone considerable revision. There has likewise been a corresponding development of other licensing criteria, resulting in similar non-uniformity in many of the requirements to which plants have been licensed.

Individual SEP plant reviews identified specific areas of structural design code changes for which the previous codes used in the SEP review required greater safety margins than earlier versions of the codes, or for which no original code provision existed. Most plants demonstrated that safety margins in building structures were not significantly lower than those required by the codes and standards used in the SEP review. A few SEP plants required certain modifications to plant structures.

The concern of this issue was to provide assurance that building structures that house systems and components important to safety are capable of withstanding the effects of natural phenomena such as earthquakes,[916] tornadoes (See Issue 156.1.5), hurricanes, and floods without loss of capability to perform their safety function. These events could cause walls or roofs to collapse damaging equipment that perform a safety function, thereby increasing the likelihood of a transient or LOCA.

CONCLUSION

On June 28, 1991, Supplement 4 to Generic Letter 88-20[1222] was issued requesting all licensees to perform an IPEEE to determine if vulnerabilities to severe accidents initiated by natural phenomena existed.[1354] The as-built structures, systems, and components in conjunction with operating plant conditions will be used to assess the adequacy of plant safety. Although this program does not directly address the effects of specific structural design code changes, it

does in part focus on evaluating the capability of building structures to withstand natural phenomena and to search for cost-effective improvements that can be made to either prevent or reduce the impact of severe accidents. Thus, the staff believed that any severe accident vulnerabilities associated with the effects of natura phenomena on building structures will be evaluated and reported in the IPEEE submittals.
The safety concern with respect to the capability of building structures to withstand the effects of natural phenomena will be sufficiently addressed in the implementation of the IPEEE Program at the 53 operating plants (34 PWRs and 19 BWRs) affected by this issue. Therefore, Issue 156.2.2 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.2.3: CONTAINMENT DESIGN AND INSPECTION

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The objective of this issue was to review the inspection program for tendons in prestressed concrete containment structures to determine whether the inspection programs included testing of prestressed tendons, checking for corrosion or relaxation and possible deterioration of prestressed containments, and whether the concrete in the containment dome or walls degraded due to shrinkage or creep. The 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

The concerns about the tendons were addressed in Issue 118 which was identified when a dented and leaking tendon grease cap was found during inspection at Farley Unit 2. The generic implications of tendon anchor head failures were studied under Issue 118 and tendon inspection and surveillance programs were developed that could be followed by licensees to mitigate or reduce such problems. The guidance for inspection and surveillance are contained in Regulatory Guides 1.35[481] and 1.35.1.[1360]

The containment dome or wall degradation due to shrinkage or creep is an age-related factor and is also addressed in Regulatory Guide 1.35.1.[1360] For license renewal applications, this concern was addressed in Draft Regulatory Guide DE-1009, "Standard Format and Content of Technical Information for Applications to Renew Nuclear Power Plant Operating Licenses," which will resolve the concern when issued in final form.

10 CFR 50 Appendix A (GDC 53), as implemented by Regulatory Guide 1.35,[481] requires that measured tendon forces (guidance provided in Regulatory Guide 1.35.1[1360]) be compared with acceptance criteria. This issue was reviewed by the staff for all SEP plants and accepted on a case-by-case basis, as documented in SERs; some of these plants also developed ISI programs.

CONCLUSION

The safety concerns of containment design and inspection at the 41 plants affected by this issue were addressed in the resolution of Issue 118. Beyond the normal life of the plants, the age-related concrete degradation concern will be addressed in the License Renewal Program. Therefore, 156.2.3 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

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

DESCRIPTION

This issue is of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The objective of this issue was to review and evaluate the original seismic design (seismic input, analysis methods, design criteria, seismic instrumentation, seismic classification) of safety-related plant structures, systems, and components to ensure the capability of plants to withstand the effects of an earthquake. Further, this issue would verify whether the free field ground motion specified for plant design adequately represents the vibratory ground motion associated with a postulated SSE at each plant. The free field ground motion will be utilized as the input to analyses to verify the design adequacy of structures, piping, and equipment. This review and evaluation will address the SSE only, since it represents the most severe event that must be considered in plant design. The scope of the review includes three major areas: (1) the integrity of the reactor coolant pressure boundary; (2) the integrity of fluid and electrical distribution systems related to safe shutdown; and (3) the integrity of mechanical and electrical equipment and engineered safety features systems (including containment). This issue did not call for a detailed review of all safety-related structures, systems, and components; rather, a sampling approach supported by a set of confirmatory analyses were to be performed. The sample size and confirmatory analyses were to be increased, if necessary. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

GDC 2 of Appendix A to 10 CFR 50 requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. An earthquake is one of the natural phenomena whose effects nuclear power plants must be designed to withstand and remain in a safe condition.

In Supplement 4 to Generic Letter No. 88-20,[1222] licensees were required to conduct an IPEEE to search for severe accident vulnerabilities due to external events. A seismic event is one of the external events that should be addressed in the IPEEE.[1371] All licensees will have to review and evaluate the seismic capabilities of their plants (the as-built, as-operated plants) to withstand the earthquake effects well beyond the design basis and to determine if severe accident vulnerabilities due to seismic events exist at their plants. The seismic input has been evaluated by the staff in the Eastern United States Probabilistic Seismic Hazard Program and the results have been factored into the process of determining the seismic review scope in the IPEEE.

The seismic qualification of mechanical and electrical equipment is being resolved by the implementation of the resolution of Issue A-46. A seismic IPEEE can be accomplished by performing either a seismic PRA with enhancements or a seismic evaluation using a seismic margins method with enhancements. The review scope may vary from plant to plant depending on the selected method and the prescribed seismic hazard condition at the site. Even with the minimum effort under the IPEEE seismic program, at least two success paths (a preferred and an alternative) to shut down and maintain a plant in a safe shutdown condition will be evaluated.[1371] This process, when using the seismic margins approach, might not provide a detailed review of all safety-related structures, systems, and components, but it will represent a sampling approach, thus fulfilling the objective of Issue 156.2.4. Furthermore, if warranted as a result of staff review, additional analyses on selected safety-related structures, systems, and components can be performed.

CONCLUSION

The safety concerns for the seismic design of structures, systems, and components will be addressed in the implementation of the IPEEE. Therefore, Issue 156.2.4 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.3.1.1: SHUTDOWN SYSTEMS

DESCRIPTION

Issues 156.3.1.1 and 156.3.1.2 were combined and evaluated together. These issues are two of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by these issues.

Issue 156.3.1.1 addressed the capability of plants to ensure reliable shutdown using safety-grade equipment. Systems and components important to safety should be designed, fabricated, installed, and tested to quality standards commensurate with the safety function to be performed. Also, systems and components that are required to withstand the effects of an SSE and remain functional should be classified as Seismic Category I. Due to the evolutionary nature of design codes and standards, the staff believed that operating plants may have been designed to requirements that are not as conservative as those currently required. Systems needed to remove decay heat and reach safe shutdown should have sufficient redundancy to ensure that their function can be accomplished with a loss of offsite power and a single failure. Systems needed to shut down must also remain functional following external events. In addition, the plant operating procedures which direct the use of these systems during normal and abnormal events were to be evaluated.

Issue 156.3.1.2 addressed the review of electrical instrumentation and control features of systems required for safe shutdown, including support systems, to determine whether they met existing licensing requirements. This review was to include the capability and methods of bringing the plant from a high pressure to a low pressure cooling condition, assuming the use of only safety equipment.

The intent of these issues have been met by a number of NRC requirements and initiatives that are already in place to secure reliable plant shutdown capability. These are as follows:

(1) The fire protection rule (10 CFR 50, Appendix R) requires that the capability for shutdown be maintained, in the event of a fire in any location;

(2) The station blackout rule (10 CFR 50.63) requires the capability to cope with a complete loss of AC power and maintain safe shutdown at the same time;

(3) A number of initiatives under the TMI Action Plan[48] enhance auxiliary feedwater capability, including emergency power provisions;

(4) Improved capability for natural circulation cooldown was required by Generic Letter No. 81-21[1355] and improved TS that enhance RHR operability in all modes were required by Generic Letter Nos. 80-42 and 80-53[1356];

(5) TMI Action Plan[48] Item I.C.I requires upgraded procedures for emergency conditions, including alternate means of providing a heat sink;

(6) The TMI Action Plan,[48] as clarified by NUREG-0737,[98] resulted in the issuance of requirements to licensees to implement Regulatory Guide 1.97[55] which specifies instrumentation for monitoring important parameters such as pressure, flow, and temperature (Continuing improvements in emergency procedures and training also address these issues);

(7) The resolution of Issue A-46 and the imposition of Generic Letter Nos. 87-02[1069] and 87-03[1387] required licensees to address the seismic adequacy of equipment needed to bring a plant to hot shutdown and maintain that condition for a minimum of 72 hours;

(8) The resolution of Issue 99 addressed corrective actions to reduce risk during shutdown with requirements issued in Generic Letter No. 88-17.[1145] The program described in this letter was included in a broader program described in SECY-91-283[1370] to evaluate the risk associated with shutdown and low power.

The resolution of Issue A-45 spanned the period from March 1981 to September 1988 during which time, extensive, PRA-based determinations of the risk resulting from shutdown cooling system failures at 6 representative operating plants were made. These studies included (but were not limited to) the concerns of Issues 156.3.1.1 and 156.3.1.2. The technical resolution of Issue A-45 was described in SECY-88-260[1143] in which the following conclusions were presented:

(1) The risk due to loss of DHR systems could be unduly high for some plants;

(2) DHR failure vulnerabilities and the optimum corrective actions for those vulnerabilities are strongly plant-specific;

(3) Detailed plant-specific analyses under the IPE program, including extension of the IPE program to require consideration of externally-initiated events (anticipated at the time of the resolution of Issue A-45 but since accomplished), will be needed to impose and implement the resolution of this issue.

The staff concluded from the PRA studies that the risk from DHR-related failures might be too high at some plants, but a generic corrective action or a set of actions could not be identified that would both reduce that risk to an acceptable level and be cost-effective at all plants. It was believed, however, that cost-effective plant-specific actions might be possible that would reduce DHR-failure-related risk and it was concluded that the most efficient method to identify any such actions would be through the IPE program.

Appendix 5 of Generic Letter No. 88-20[1222] provided a specific description of those topics addressed in Issue A-45 and related to internally-initiated events (including those raised in Issues 156.3.1.1 and 156.3.1.2) that are to be considered in the IPE program. The IPE process was extended to include externally-initiated events (IPEEE) upon issuance of Supplement 4 to Generic Letter No. 88-20.[1222] Section 5 of this supplement specifically described how the IPEEE program was to be used to implement the technical resolution of those topics in Issue A-45 that are related to externally-initiated events.

The studies performed in the resolution of Issue A-45 included the analysis of events that initiate at full power conditions. Although the final results (total risk resulting from DHR-related failures) were increased by 20% for PWRs and 30% for BWRs to account for risk from DHR-related failures, during events that initiate when a plant is not at full power (such as hot standby and cold shutdown), such events were not investigated in detail. The IPE process was consistent with the analyses completed for Issue A-45 in that it only required consideration of events that initiate at full power conditions.

However, detailed attention is currently being paid to DHR failure-related events that initiate at conditions other than full power by an extensive NRC program initiated with the issuance of Generic Letter No. 88-17[1145] which resulted from an Augmented Inspection Team (AIT) investigation of a 1987 loss-of-DHR event at Diablo Canyon.[1369] This letter required licensees to investigate and, if necessary, improve procedures involving containment isolation and cooling and DHR-related equipment operation methods and training during non-power operations, when the reactor primary coolant inventory is reduced. This work received additional impetus since the issuance of Generic Letter No. 88-17[1145] by a loss-of-DHR event at the Vogtle nuclear plant. The Vogtle event resulted in the issuance of SECY-91-283[1370] which described all aspects of the extensive program including, but not limited to, the program outlined in Generic Letter No. 88-17.[1145] Some aspects of the program described in SECY-91-283[1370] will contribute to the imposition and implementation of the resolution of Issue A-45. This program now includes the NRC-sponsored Low Power and Shutdown (LP&S) Program which was originally formulated as part of the NRC response to the Chernobyl event.[1195] The LP&S work is being performed by BNL and SNL with additional work regarding seismically-initiated events being performed by Future Resources Associates (FRA), Inc. The objectives of the LP&S program were to: (1) assess the frequency and risk of accidents initiated during LP&S modes of operation for two nuclear power plants; (2) compare the assessed frequency and risk with those of accidents initiated during full power operations; and (3) develop new methods for assessing LP&S accident frequency and risk, as necessary.

CONCLUSION

The safety concerns of Issues 156.3.1.1 and 156.3.1.2 were addressed in the resolution of Issue A-45 and in the IPE and IPEEE programs which were supplemented by the Evaluation of Shutdown and Low Power Risk Issues Program described in SECY-91-283.[1370] Therefore, Issues 156.3.1.1 and 156.3.1.2 were DROPPED from further consideration as new and separate issues. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issues.

ISSUE 156.3.1.2: ELECTRICAL INSTRUMENTATION AND CONTROLS

This issue was evaluated with Issue 156.3.1.1 above and DROPPED from further consideration as a new and separate issue.

ISSUE 156.3.2: SERVICE AND COOLING WATER SYSTEMS

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The safety concern was the capability of service and cooling water systems to meet their design objective with adequate margin. This issue was raised to provide assurance that service and cooling water systems are: (1) capable of transferring heat from structures, systems, and components important to safety to the ultimate heat sink; (2) provided with adequate physical separation such that there are no adverse interactions among the systems under any mode of operation; and (3) provided with sufficient cooling water inventory or that adequate provisions for makeup are available. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

Concerns for the potential unavailability of SWS were addressed in Issues 51, 130, and 153. Issue 51 was resolved and implemented at operating plants in accordance with Generic Letter No. 89-13.[1259] The resolution identified a recommended improvement in the reliability of open cycle SWS that could result from reducing the potential for flow blockage in safety-related components caused by bivalves, sediment, and corrosion products. This improvement was in the form of an integrated, baseline fouling surveillance and control program for all nuclear power plant open cycle SWS.

Issue 130 was resolved and is being implemented at certain specific plants in accordance with Generic Letter 91-13.[1368] This issue addressed the concerns regarding the SWS reliability of 14 PWRs at multi-unit sites with two SWS trains per unit and a crosstie capability. The resolution identified several cost-effective options that were considered for reducing the risk from loss of SWS (due to causes other than fouling), including a backup means of RCP seal cooling plus additional SWS TS and emergency procedures.

Issue 153 affected all LWRs except those that were addressed in Issue 130. All potential causes of SWS unavailability were to be considered, except those that were resolved and implemented in accordance with Generic Letter No. 89-13.[1259] The resolution plan for Issue 153 was divided into two phases: Phase I, a pilot study; and Phase II, a generic evaluation. The results of Phase I were to be used to determine if an interim resolution was viable and how to proceed with Phase II; Issue B-32 was also addressed in the resolution of Issue 153.

Concerns for the availability of cooling water systems were addressed in the resolution of Issue 143. This issue addressed the potential unavailability of chilled water systems which provide room cooling to maintain adequate environmental temperature for non-safety-related and safety-related equipment. The potential loss of room cooling could affect the operability of the safety-related systems including the SWS system.

CONCLUSION

All of the concerns regarding the performance capability and reliability of service and cooling water systems at the 41 affected plants either have been addressed or are being addressed in the issues discussed above. Additionally, a staff action plan was developed that established NRR as the focal point to ensure that all existing and future SWS issues are adequately addressed.[1367] Therefore, Issue 156.3.2 was DROPPED from further consideration as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.3.3: VENTILATION SYSTEMS

DESCRIPTION

This issue is one of nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] At issue was the adequacy of ventilation systems to provide a safe environment for plant personnel and ESF systems under normal, anticipated transient, and design basis operational conditions. A safe environment is one that is effectively controlled with respect to radiation, heat, humidity, smoke, and toxic gases. Five ventilation systems were identified in SRP[11] Section 9.4 to effect ESF equipment and plant personnel: the control room area, spent fuel area, auxiliary and radwaste area, turbine area, and ESF area.

With respect to plant personnel, the concerns about ventilation are grouped under radiation exposure as the first, and exposure to excessive levels of environmental pollutants such as smoke, toxic gases, heat, and humidity as the second. These concerns may be considered for both normal operating and abnormal conditions. For normal conditions, the first concern is addressed by existing regulations in 10 CFR 20 which is quite clear and comprehensive concerning monitoring of restricted and unrestricted areas and radiation limits in each. In particular, 10 CFR 20.106 applies to radioactivity in effluent between restricted and unrestricted areas. Coverage includes limits of concentrations of radioactive material in air as well as water. For applications filed after January 2, 1971, 10 CFR 50.34a requires ALARA programs which are elaborated upon in 10 CFR 50, Appendix I. In addition, 10 CFR 50.34a requires design and installation of equipment "to maintain control over radioactive materials in gaseous and liquid effluent" not only during normal operations but also during expected operational occurrences. 10 CFR 50.36a requires TS on effluent from nuclear power reactors.

For normal operating conditions, the second concern is the responsibility of OSHA whenever the safety of licensed radioactive materials is not involved. This responsibility was outlined in an MOU between OSHA and the NRC issued on October 25, 1988. For abnormal conditions, the second concern comprises potentially unpleasant plant nuisance factors with the exception of the control room and turbine area. One potentially serious atmospheric contaminant in the turbine building and the auxiliary building of PWRs is H_2 with its potential for deflagration or detonation. Issue 106 addressed the role of ventilation systems in the prevention of H_2 deflagration from leaks in the H_2 distribution piping.

Issue 136 addressed the issue of vapor clouds from liquified combustible gases drifting into safety-related air intakes.

Abnormal control room environmental conditions could exist that adversely affect operator performance to a degree sufficient to cause operator-initiated transients. These conditions are within the NRC scope as defined in the above MOU. Conditions affecting mitigation of accidents are also clearly NRC responsibility. The resolution of Issue 83 will address the limits of plant personnel functioning from radiation and toxic gas exposure. The scope of Issue 83 includes "provisions for personnel to remain in the control room as needed to manage accidents which have the potential for offsite and onsite radiological consequences, and protection of control room occupants to the degree necessary to prevent an accident occurring as a result of operator incapacitation." SRP[11] Section 6.4, Rev. 2, describes review of the control room ventilation system with the objective of assuring protection for plant operators from the effects of accidental releases of toxic and radioactive gases. A third revision draft is under consideration as part of the resolution of Issue 83. Thus, accident initiation and mitigation capabilities of control room personnel are being addressed with respect to radiation and toxic gas exposure. Control room concerns remaining are high temperature and humidity and smoke.

With respect to high temperature and humidity, the ACRS recommended that "[t]emperature limits should be revised taking into account low air exchange rate, operation of ESF filter system heaters and perspiration." The ACRS considers a temperature limit of 120□F for the control room as unacceptable; this is a TS limit derived for control room equipment.[678] Under accident conditions, no NRC requirement exists for temperature limits for reliable performance of control room personnel. However, documentation exists that supports a maximum effective temperature of 85□F for reliable human performance. (A defined effective temperature includes some combination of dry bulb temperature, relative humidity, and air velocity). Although no accident condition temperature limit has been formalized, SRP[11] Section 9.4.1, "Control Room Area Ventilation System," concerns itself in part with "...the comfort of control room personnel during normal operating, anticipated operational transient, and design basis accident conditions." The control room area ventilation system (CRAVS) is reviewed, among other things, with respect to ability to maintain a suitable ambient temperature for control room personnel. The single failure criterion is applied in the CRAVS review. In addition, the CRAVS must function unaffected by loss of equipment that is not seismic Category 1 and the integrated system design must satisfy GDC 2 with respect to earthquakes. The designs are reviewed for protection from floods, hurricanes, tornadoes, internally- or externally-generated missiles, fires, and loss of offsite power. At some plants, the CRAVS is capable of functioning in an internal-filtered recirculation mode of operation.

A survey of 12 plants reported some problems with adequacy and demonstration of adequacy of control room cooling for a postulated 30-day accident period.[1371] The plants surveyed were a mix of ages, ranging from some of the oldest to some of the newest. While the problems identified produced no added industry requirements, a recommendation was made for more [staff] attention to detail in evaluations of control room cooling systems design and operations that rely on two separate cooling systems, i.e., a non-safety-related system for normal operations and a safety-related system for emergency operations only. In sum, no additional regulatory requirements or guidance are warranted for investigation with respect to high temperature and humidity vis-a-vis control room personnel under accident conditions.

Issue 143 is to be resolved and will address the importance of ventilation systems on cooling for the operation of ESF equipment. Activities in support of the resolution of Issue 143 will identify the vulnerabilities of safety-related systems and their support systems to the effects of HVAC and chilled water system failures and adverse temperature fluctuations. An evaluation will be made of equipment environmental qualification, equipment room heat load and heat-up rate to identify areas in which a reduction in the dependence of equipment operability on HVAC and room cooling may be required. The control of smoke in plants is being addressed in Issue 148.

CONCLUSION

The safety concerns of Issue 156.3.3 were either being addressed in ongoing staff actions on Issues 83, 106, 136, 143, and 148, or were covered by existing regulations. Therefore, Issue 156.3.3 was DROPPED from further pursuit as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.3.4: ISOLATION OF HIGH AND LOW PRESSURE SYSTEMS

DESCRIPTION

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

Issue 105 addressed the possible breach of those interfacing boundaries that are created by a series of PIVs and the consequences of failure of a boundary by mechanical failure, human error, or external event. Thus, Issue 105 covered all interfacing systems, including those identified in Issue 156.3.4. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

CONCLUSION

The safety concern of Issue 156.3.4 was addressed in the resolution of Issue 105. Therefore, Issue 156.3.4 was DROPPED from further pursuit as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.3.5: AUTOMATIC ECCS SWITCHOVER

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] Most PWRs require operator action to realign the ECCS for the recirculation mode following a LOCA. Existing guidelines state that automatic transfer to the recirculation mode is preferable to manual transfer. However, a design that provides manual switchover is sufficient provided that adequate instrumentation and information displays are available for the operator to manually transfer from the injection mode to the recirculation mode at the correct time. Automatic in lieu of manual switchover could possibly provide an improvement of ECCS reliability at a cost that could result in a worthwhile safety enhancement. This issue addressed the procedures for manual switchover, the adequacy of available instrumentation, and the possible operator errors associated with the switchover process. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

CONCLUSION

All 41 plants affected by this issue were to be considered in the resolution of Issue 24 which was directed at studying the merits of manual, automatic, and semi-automatic ECCS switchover to recirculation. Thus, Issue 156.3.5 was covered in the resolution of Issue 24. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change this conclusion.

ISSUE 156.3.6.1: EMERGENCY AC POWER

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The electrical independence and redundancy of safety-related onsite power sources must meet the single failure criterion. Diesel generators, which provide emergency standby power for safe reactor shutdown in the event of total loss of offsite power, have experienced a significant number of failures over the years that have been attributed to a variety of causes, including failure of the air startup, fuel oil, and combustion air system. The objective of this issue was to review the reliability of protection interlocks and testing of diesel generators to assure that diesel generator systems meet the availability requirements for providing emergency standby power to the engineered safety features, as well as the independence of onsite power distribution systems and features, such as automatic bus transfers and breaker connections, that could affect the independence of redundant trains. The 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

CONCLUSION

The safety concern of this issue was addressed in the resolution of Issues A-44, 128, and B-56. The requirements that resulted from the resolution of these three issues will affect the 41 non-SEP plants. In addition, MPAs B-23, "Degraded Grid Voltage," and B-48, "Adequacy of Station Electric Distribution Voltage," have been implemented at several of the 41 plants affected by this issue and will not have to be repeated in the implementation of the resolution of Issue A-44.[1108] Based on the above considerations, Issue 156.3.6.1 was DROPPED from further pursuit as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.3.6.2: EMERGENCY DC POWER

DESCRIPTION

Historical Background

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343[1351] following its study of how the lessons learned from the SEP have been factored into the licensing bases of operating plants. The issue addresses the concern that safety-related DC power system bus voltage monitoring and annunciation may not adequately notify operators of DC bus status. Responses to Generic Letter 91-06[1399] indicated that a significant number of licensees could be affected by the concerns of this issue. Based upon a PRA analysis of the DC power system at six plants, it was concluded that additional DC power system bus voltage monitoring and annunciation for licensed facilities would not have a significant impact on safety and would not be a cost-effective means of increasing plant safety.

This issue addressed the criteria in 10 CFR 50.55a(h) and 10 CFR 50 (GDC 2, 4, 5, 17, 18, and 19) which require that the control room operator be given timely indication of the status of the safety-related DC power system batteries and their availability. The current staff position is that the following separate and independent control room indications and alarms for the Class 1E DC power system status are recommended in order to meet these criteria:

(1) battery disconnect or circuit breaker open alarm
(2) battery charger disconnect or circuit breaker open alarm (both input AC and output DC)
(3) DC system ground alarm
(4) DC bus undervoltage alarm
(5) DC bus overvoltage alarm
(6) battery charger failure alarm
(7) battery discharge alarm
(8) battery float charge current ammeter
(9) battery circuit output current ammeter
(10) battery discharge indicator
(11) bus voltage voltmeter

These annunciators and alarms are needed in order to ensure that the control room operators are alerted in the event of DC power system or battery failure. If a less extensive configuration of equipment is used, it is possible that a DC power system or battery failure mode could exist which would not result in the actuation of any alarms or annunciators. In this event, the DC power supply would remain in the degraded condition until a periodic surveillance test or maintenance was performed to identify the condition of the batteries.

Safety Significance

Based upon the SEP reviews, it was apparent that some licensees had received operating licenses without providing the above recommended alarms and annunciators. However, in most cases the licensees in the SEP reviews were able to demonstrate to the staff that modifications were unnecessary. The concern in this issue is that some licensees that were not reviewed in the SEP program might have insufficient annunciators and alarms in the control room to alert the operators to some safety-related DC power supply or battery failure modes, which would increase the likelihood that a DC power supply is unavailable when needed.

PRIORITY DETERMINATION

The issue of control room annunciation and alarms for the safety-related DC power supplies was also addressed in Issue A-30 which was combined with other generic issues involving safety-related power supplies to form Issue 128. Generic Letters 91-06[1399] and 91-11[1400] were issued in the resolution of Issue 128; Generic Letter 91-06 addressed the concerns of Issue A-30. Industry organizations such as NUMARC and INPO asserted that most licensees already had alarm and annunciator configurations that were equivalent to the existing staff recommendations which were based in part on industry standards. Therefore, the questions in Generic Letter 91-06[1399] which addressed available alarms and annunciators did not represent a minimum acceptable configuration, but were formulated to provide sufficient information to the staff to determine if licensees had met or adequately addressed the current recommendations.

An INEL review[1457] of the responses to Generic Letter 91-06[1399] showed that 42 licensees do not have any separate and independent alarms in the control room for their DC power system. However, these licensees typically had local alarms which were separate and independent, and a single battery condition monitor which alarms in the control room in the event that one or more of the local battery alarms actuate. In addition, the INEL review indicated that 15 licensees have not performed a human factors review of their testing and maintenance procedures, and 5 licensees do not have procedures that specifically prevent simultaneous testing or maintenance of redundant safety-related DC power sources. In most cases, the licensees supplied justification for the discrepancies between their licensed configuration and the current staff position. INEL did not evaluate licensee responses to determine what modifications would be required to adequately resolve the concerns of Issue A-30, and recommended that the staff perform a PRA study to determine the impact on plant safety of existing configurations of safety-related DC power supply annunciation and alarms.

Frequency Estimate

The concern in this issue was that the safety-related DC power supplies might be unavailable because of inadequate control room annunciators and alarms. This concern correlates with the results of NUREG-0666,[164] which included a FMEA and a PRA of a model DC power system. This model system consisted of two independent DC buses each of which were supplied by a single battery charger and had a single battery back-up. In addition, this system had the following alarms and annunciators in the control room: (1) battery charger ground alarm; (2) battery charger AC power supply failure alarm; (3) DC bus undervoltage alarm; (4) battery charger DC ammeter; and (5) battery charger DC voltmeter.

NUREG-0666[164] concluded that battery unavailability is dominated by inadequate maintenance practices and failure to detect battery unavailability due to bus connection faults. By improving battery surveillance, DC power system unreliability could be decreased by a factor of two, and improving maintenance and testing practices could decrease DC power system unavailability by a factor of 10. The report does not quantify a safety benefit which would result from additional alarms or annunciators in the control room, but additional alarms and annunciators would result in the enhancement of surveillance, maintenance and testing capabilities. Additional recommendations were made in NUREG-0666,[164] but these relate to aspects of the DC system which would not be enhanced by the addition of alarms or annunciators, such as the addition of a third DC power train.

In addition to the concerns relating to alarms and annunciators, the responses to Generic Letter 91-06[1399] also identified concerns with the probability of CCF of the DC power supplies. In order to evaluate these two concerns, the PRAs for 6 licensees were reviewed and found to include basic events which modeled the probability of battery unavailability and common cause battery failure. A study was performed to determine the effect on the CDF of decreasing battery unavailability and common cause battery failure probability. This study was performed by the staff using the SARA[1456] software. The results are described below.

The assumption was made that improved alarms and annunciators would result in continuous battery condition indication and would essentially result in an undetected battery failure probability of zero, since the operators would be notified of a DC power system failure immediately. However, this approximation would give a greater estimate of the effectiveness of modifications of alarms and annunciators than could actually be obtained. A better estimate of the effect on DC power system reliability resulting from an increase in the number of alarms and annunciators in the control room was obtained by decreasing the battery unavailability from the base case value to a test case value of 10^{-6}. For the plants considered in this analysis, the base case values ranged from 6.12×10^{-3} to 7.2×10^{-4}, which reflects an hourly failure rate of approximately 10^{-6}/hour, and an interval between tests which are capable of detecting a failed battery ranging from 6,120 to 720 hours.

This modification in battery unavailability will also account for any decrease in the battery charger unavailability resulting from the additional hardware. Because the battery must be instantaneously available to supply power if the battery charger fails, the battery unavailability terms in a PRA model are always multiplied by the battery charger unavailability terms. This analysis is conservative because it overestimates the effectiveness of additional alarms and annunciators, which will improve DC power system reliability by a much smaller factor. In addition, this approximation is made under the assumption that the DC power systems have been accurately modeled by PRA analysts for the existing PRAs and is only valid if the configuration of alarms and annunciators modelled by the existing PRAs is less effective than the currently recommended configuration.

CCF of the DC power system can be caused by maintenance activity, the most significant of which is inadvertent connection of redundant trains. Generic Letter 91-11[1400] addressed the use of interconnections between Class 1E vital instrument buses and LCOs for Class 1E vital instrument buses. The purpose of this generic letter was to decrease the probability and sources of CCF of redundant Class 1E AC and DC buses and inverters. It was assumed that CCF of the Class 1E buses and inverters has been adequately addressed and the scope of this issue was limited to the batteries and battery chargers.

The SARA[1456] software was used to model the effect of decreasing battery unavailability. There are currently nine operating plants which have PRA models which can be used with SARA. These are listed below, in addition to the configuration of the DC power system at the plant.

Plant	Number of 125V DC Batteries	Number of Battery Chargers
Grand Gulf 1[1318]	3	6
Brunswick 1 & 2*	4 (each)	4 (each)
Peach Bottom 2*	4	4
Surry 1[1318]	2 + diesel	2
Sequoyah 1[1318]	2 + diesel + 1 common	2 + 1 common
Oconee-3[889]	2	3
Zion[1318]	2 + 1 common	2 + 1 common
Indian Point-2	4	4

* Based on IPE Submittal

<u>Peach Bottom-2</u>: This unit has two independent divisions of safety-related 125V DC power, one of which is required to safely shut down the plant. Each division is comprised of two batteries, each with it's own charger. The control room has 3 of 7 recommended alarms and 1 of 4 recommended annunciators. The Peach Bottom PRA included probability terms for battery unavailability due to common mode failure and unavailability of the individual Unit 2B and 3C battery banks. The terms for the remaining battery banks (2A, 2C, 2D, and 3D) were not included in any significant minimal cutsets, and decreasing these basic event probabilities would have a negligible effect on the CDF. The probability of battery unavailability was estimated in the original PRA to be 0.001.

Peach Bottom-2: Common Mode Battery Failure

Probability	CDF/RY	Change/RY
0.001	3.6×10^{-6}	base case
0.000001	3.4×10^{-6}	-2.0×10^{-7}

Peach Bottom-2: Battery 2B and 3C Failure

Probability	CDF/RY	Change/RY
0.001	3.6×10^{-6}	base case
0.000001	3.6×10^{-6}	-

Decreasing the probability of common mode battery unavailability by three orders of magnitude would result in a decrease in CDF of 2.0×10^{-7}/year, whereas decreasing the probability of the unavailability of batteries 2B and 3C would result in less than a 10^{-7} decrease in CDF.

Grand Gulf-1: This unit has three independent divisions of safety-related 125V DC power, two of which are required to safely shut down the plant. The control room has 1 of 7 recommended alarms and 1 of 4 recommended annunciators. The Grand Gulf PRA included terms for the probability of battery common mode failure and failure of the individual Unit 1A3, 1B3, and 1C3 battery banks. All battery banks were included in significant minimal cutsets.

Grand Gulf-1: Common Mode Battery Failure

Probability	CDF/RY	Change/RY
0.001	2.1×10^{-6}	base case
0.000001	1.6×10^{-6}	-5.0×10^{-7}

Grand Gulf 1 - Loss of Power from Batteries 1A3, 1B3, 1C3

Probability	CDF/RY	Change/RY
0.001	2.1×10^{-6}	base case
0.000001	1.9×10^{-6}	-2.0×10^{-7}

Decreasing common mode battery unavailability by three orders of magnitude would result in a decrease in CDF of 5×10^{-7}/RY, whereas decreasing the unavailability of battery 1A3, 1B3 and 1C3 would result in a decrease of 2×10^{-7} in CDF.

Brunswick-1 and 2: These units each have two independent divisions of safety-related 125V DC power, one of which is required to safely shut down the plant. Each division is comprised of two independent batteries, each with its own charger. The control room has 5 of 7 recommended alarms and 2 of 4 recommended annunciators. The Brunswick Units 1 and 2 PRAs included terms for the probability of individual battery bank unavailability but not for common cause unavailability. The terms for failure of three of the four batteries were included in some minimal cutsets.

Brunswick-1: Battery Bank 1A1, 1A2, and 1B1 Fault

Probability	CDF/RY	Change/RY
0.00033	2.47×10^{-5}	base case
0.000001	2.46×10^{-5}	-1.0×10^{-7}

Brunswick-2: Battery Bank 2A1, 2A2, and 2B1 Fault

Probability	CDF/RY	Change/RY
0.00033	2.08×10^{-5}	base case
0.000001	2.06×10^{-5}	-2.0×10^{-7}

Units 1 and 2 differed slightly in their response to battery failure rate changes. However, decreasing the unavailability of battery 2A1, 2A2, and 2B1 would result in a decrease of 10^{-7}/RY and 2×10^{-7}/RY in CDF for Unit 1 and 2, respectively.

Surry-1: This unit has two independent divisions of safety-related 125V DC power, one of which is required to safely shut down the plant. The unit also has dedicated batteries for starting the diesel generators. The control room has 4 of 7 recommended alarms and 1 of 4 recommended annunciators. The Surry PRA included terms for the probability of battery common mode failure and failure of the individual I and II battery banks. Neither the common mode battery failure term or individual battery failure terms were included in any significant minimal cutsets. The assumed battery unavailability was 7.2×10^{-4}, which suggests a 2-month interval between tests that would detect battery problems for the typical failure rate. Because the CDF magnitude cutoff for exclusion of core damage sequences from the group of minimal cutsets is usually less than 10^{-8}, decreasing battery unavailability or common mode failure probability would result in a negligible decrease in CDF.

Sequoyah-1: This unit has two independent divisions of safety-related 125V DC power, one of which is required to safely shut down the plant. The unit also has dedicated batteries for starting the diesel generators. The control room has zero of 7 recommended alarms and 3 of 4 recommended annunciators. The Sequoyah PRA included probabilities for battery common mode unavailability and unavailability of the individual I and II battery banks. Battery unavailability was initially estimated to be 7.2×10^{-4}, which suggests a two-month surveillance test or maintenance interval for a failure rate of 10^{-6}/hour. The common mode unavailability was estimated to be 5.8×10^{-6}. Neither the common mode unavailability or individual battery unavailability were included in any significant minimal cutsets. The unavailabilities used in this analysis were slightly lower than those used in other analyses. However, the CDF magnitude cutoff for exclusion of core damage sequences from the group of minimal cutsets is usually less than 10^{-8} or less. Therefore, decreasing battery unavailability or common mode failure probability would result in a negligible decrease in CDF.

Oconee-3: This unit has two independent divisions of safety-related DC power, one of which is required to safely shut down the plant. The control room has 1 of 7 recommended alarms and none of 4 recommended annunciators. The Oconee PRA[889] included terms for unavailability of the individual 1CA, 1CB, 3CA, and 3CB battery banks. The probability of battery unavailability was estimated to be 6.12×10^{-3}, which is based on a one-year surveillance test or maintenance interval and a failure rate of 1.4×10^{-6}/hour. Common mode unavailability was not included in the PRA model. The individual battery unavailability terms were not included in any significant minimal cutsets. The probabilities used in this analysis were significantly greater than those used in other analyses. However, the CDF magnitude cutoff for exclusion of core damage sequences from the group of minimal cutsets is usually less than 10^{-8} or less. Therefore, decreasing battery unavailability or common mode failure probability would result in a negligible decrease in CDF.

The average decrease in CDF from the proposed modifications was estimated to be approximately 10^{-7}/RY.

Consequence Estimate

It was assumed that all affected operating plants had an average remaining life of 20 years, based on their original licenses. It was also assumed that each of these plants would be granted a life extension of 20 years. Thus, the average remaining life for all affected plants was 40 years.

The public risk associated with the event considered in this issue was estimated[64] to be 6.76 x 10^6 man-rem and 2.52 x 10^6 man-rem for BWRs and PWRs, respectively. For BWRs, the total potential risk reduction was estimated to be (6.76 x 10^6)(10^{-7})(40) man-rem/reactor or 27 man-rem/reactor. For PWRs, the total potential risk reduction was estimated to be (2.52 x 10^6)(10^{-7})(40) man-rem/reactor or 10 man-rem/reactor.

Cost Estimate

Improving the control room annunciators and alarms for all safety-related DC power systems at each plant would involve a different amount of effort for each licensee, depending upon the amount of instrumentation currently installed, available space for additional annunciators and alarms, and whether existing raceway could hold additional cables. In addition, new procedures and operator training would be required. This additional hardware would include the following:

(1)	Data transmitters at each battery room. Design, installation and testing assumed to be $100,000/battery room, with 3 battery rooms per facility	$300,000
(2)	Raceway and cable from each battery room to the control room. Design, installation and testing costs assumed to be $100 per linear foot, with 1000 linear feet of raceway per battery room and 3 battery rooms per facility	$300,000
(3)	Control room modifications to add annunciators and alarms. Design, installation and testing assumed to be $100,000/battery, 3 batteries per facility	$300,000
(4)	Procedure changes, drawing changes, training, and administrative costs	$100,000
	TOTAL:	$1,000,000

Value/Impact Assessment

Separate value/impact scores were calculated for PWRs and BWRs.

BWRs: Based on a potential public risk reduction of 27 man-rem/reactor and an estimated cost of $1M/reactor for a possible solution, the value/impact score was given by:

$$S = \frac{27 \text{ man-rem/reactor}}{\$1\text{M/reactor}}$$

$$= 27 \text{ man-rem/\$M}$$

PWRs: Based on a potential public risk reduction of 10 man-rem/reactor and an estimated cost of $1M/reactor for a possible solution, the value/impact score was given by;

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

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

Other Considerations

(1) It is important to monitor the condition of the safety-related DC power system, including the condition of batteries which may be needed in the event of a station blackout. In addition, it is also necessary to have procedures which minimize the probability of a common cause fault of the safety-related DC power systems. Operating experience so far does not indicate that significant problems exist in this area.

(2) Based upon the results of this study, it could be asserted that the control room alarms and annunciators recommended by the staff in current licensing guidelines do not result in a significant increase in plant safety beyond that realized by existing alarm and annunciator configurations and weekly or quarterly maintenance programs. It should be noted that the empirical battery failure rate of approximately 10^{-6}/hour, which is used to determine battery unavailability, is dependent upon the frequency of battery failures for systems with existing configurations of control room annunciators and alarms. Therefore, it might not be accurate to conclude that the existing recommendations for annunciators and alarms should be relaxed.

(3) Battery unavailability and CCF are recognized by some licensees to be sufficiently probable so as to require modeling in PRAs. Based upon these PRA models, decreasing the unavailability of the batteries and safety-related DC power supplies by several orders of magnitude over that used in the base case does not result in a significant decrease in CDF for these licensees. This observation must be tempered with the knowledge that licensees currently monitor important DC bus parameters, and that other DC power system design features, such as the number of batteries, have a greater impact on DC power system reliability than the number of alarms and annunciators.

CONCLUSION

Based on the potential public risk reduction, this issue had a low priority ranking for BWRs and was in the drop category for PWRs (see Appendix C). Overall, the issue was given a low priority ranking in March 1993. Consideration of a 20-year license renewal period did not change the priority of the issue.[1564] Further prioritization, using the conversion factor of $2,000/man-rem approved by the Commission in September 1995, resulted in an impact/value ratio (R) of $37,037/man-rem which placed the issue in the DROP category.

ISSUE 156.3.8: SHARED SYSTEMS

DESCRIPTION

This issue is one of the nineteen category 3 issues identified by NRR in SECY-90-343.[1351] The sharing of the ESFS for a multi-unit plant, including onsite emergency power systems and service systems, can result in a reduction of the number and capacity of onsite systems to below that which is needed to bring either unit to a safe shutdown condition, or to mitigate the consequences of an accident. Shared systems for multiple unit stations should include equipment powered from each of the units involved. There were 13 multi-unit sites that could be affected by this issue among the 41 non-SEP plants identified in SECY-90-343[1351] that received OLs before 1976.

CONCLUSION

The safety concerns associated with systems that are shared by two or more units at multi-unit sites have been previously identified by the staff. The most important contributors to core damage probability at these sites have been determined to be air, cooling water, and electric power systems. These systems have been adequately addressed in Issues 43, 130, 153, and A-44. Based on these considerations, this issue was DROPPED from further pursuit as a new and separate issue. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change the priority of the issue.

ISSUE 156.4.1: RPS AND ESFS ISOLATION

DESCRIPTION

This issue is one of the three Category 4 issues identified by NRR in SECY-90-343.[1351] The safety concern was that, in the event of non-safety system failures, the lack of isolation devices could result in the propagation of faults to safety systems and common cause failures may result. In its study, the staff found that approximately 39 plants at 28 sites were not required to meet IEEE 279-1971[397] and have not been reviewed for this safety concern since the time of their licensing.
Non-safety systems generally receive control signals from the RPS and ESF sensor current loops. The non-safety circuits are required to be isolated to ensure the independence of the RPS and ESF channels. Requirements for the design and qualification of isolation devices are quite specific. Evaluation of the quality of isolation devices is not the safety issue of concern; rather, the issue is the existence of isolation devices which will preclude the propagation of non-safety system faults to safety systems.

CONCLUSION

The safety concerns of leakage through electrical isolators in instrumentation circuits and electrical isolation in plants not required to meet IEEE 279-1971[397] were addressed in the resolution of Issue 142. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change this conclusion.

ISSUE 156.4.2: TESTING OF THE RPS AND ESFS

DESCRIPTION

This issue is one of the nineteen Category 3 issues identified by NRR in SECY-90-343.[1351] The objective of this issue was to review plant designs to ensure that: (1) all ECCS components, including the pumps and valves, are included in the component and system test; (2) the frequency and scope of periodic testing are identified; and (3) the test programs will provide adequate assurance that the systems will function when needed. The 41 plants identified in SECY-90-343[1351] that received OLs before 1976 were affected by this issue.

CONCLUSION

A portion of this issue was covered by existing requirements; specifically, ECCS pumps and valves are required to be tested quarterly by the ASME Code in accordance with 10 CFR 50.55(a), unless the NRC grants relief to defer testing until refueling outages. The remainder of this issue was covered in the resolution of Issue 120 which addressed the concern regarding on-line (at-power) testability af protection systems (both the RPS and the ESFS) and the possibility that some plants may not provide complete testing capability at power. In an RES evaluation,[1564] it was concluded that consideration of a 20-year license renewal period did not change this conclusion.

ISSUE 156.6.1: PIPE BREAK EFFECTS ON SYSTEMS AND COMPONENTS

DESCRIPTION

Historical Background

In 1967, the AEC published draft GDCs for comment and interim use and, until 1972, the staff's implementation of the GDCs required consideration of pipe break effects inside containment. However, due to the lack of documented review criteria, AEC staff positions continued to evolve. Review uniformity was finally developed in the early 1970s, initiated by a November 9, 1972, note from L. Rogers to R. Fraley, in which a Draft Safety Guide entitled "Protection Against Pipe Whip Inside Containment" was proposed. This Draft Guide contained some of the first documented deterministic criteria that the staff had used for several years (to varying degrees) as guidelines for selecting the locations and orientations of postulated pipe breaks inside containment, and for identifying the measures that should be taken to protect safety-related systems and equipment from the dynamic effects of such breaks. Prior to use of these deterministic criteria, the staff used non-deterministic guidelines on a plant-specific basis. The Draft Safety Guide was subsequently revised and issued in May 1973 as Regulatory Guide 1.46[18] for implementation on a forward-fit basis only.

The AEC issued two generic letters to all licensees and CP or OL applicants regarding pipe break effects outside containment in December 1972[139] and July 1973. These letters, known as the "Giambusso" and "O'Leary" letters, respectively, extended pipe break concerns to locations outside containment, and provided deterministic criteria for break postulation and evaluation of the dynamic effects of postulated breaks. The letters requested all recipients to submit a report to the staff summarizing each plant-specific analysis of the issue. All operating reactor licensees and license applicants submitted the requested analyses in separate correspondence or updated the SARs for their proposed plants to include the analysis. The staff reviewed the submitted analyses and prepared safety evaluations for all plants. In November 1975, the staff published SRP[11] Sections 3.6.1 and 3.6.2 that slightly revised the two generic letters discussed above. Thus, after 1975, the specific structural and environmental effects of pipe whip, jet impingement, flooding, etc., on systems and components relied on for safe reactor shutdown were considered.

As stated above, the AEC/NRC has provided requirements to the industry regarding pipe breaks outside of containment through the issuance of the Giambusso and O'Leary generic letters. Since these requirements are applicable to all the affected plants, pipe breaks outside of containment were judged to be a compliance issue and were not considered in this analysis. Compliance matters are dealt with promptly and do not await the generic issue resolution process. Therefore, the issue of pipe breaks outside of containment for the 41 affected plants was brought to the attention of NRR by separate correspondence.[1761] The remainder of this evaluation only addressed pipe breaks inside containment.

As a part of its plant-specific reviews between 1975 and 1981, the staff used the guidelines in Regulatory Guide 1.46[18] for postulated pipe breaks inside containment, and SRP[11] Sections 3.6.1 and 3.6.2 for outside containment. In July 1981, SRP[11] Sections 3.6.1 and 3.6.2 were revised to be applicable to both outside and inside containment, thus eliminating the need for further use of Regulatory Guide 1.46,[18] which was subsequently withdrawn.

Between the period 1983-1987, the general issue of pipe breaks inside and outside containment was revisited in the SEP. The objective of the SEP was to determine to what extent the earliest 10 plants (i.e., SEP-II) met the licensing criteria in existence at that time. This objective was later interpreted to ensure that the SEP also provided safety assessments adequate for conversion of provisional operating licenses (POLs) to full-term operating licenses (FTOLs). As a result of these reviews, plants were required to perform engineering evaluations, TS or procedural changes, and physical modifications both inside and outside containment. Regarding inside containment modifications: of the two SEP-II plants evaluated in this analysis (one BWR and one PWR), the BWR was required to modify four piping containment penetrations and the PWR was required to modify steam generator blowdown piping supports. This indicates there was a wide spectrum of implementation associated with the original reviews of these early plants for pipe breaks inside and outside containment.

As with the above-described evolution of uniform pipe break criteria, electrical systems design criteria were also in a state of development. Prior to 1974, electrical system designs were generally reviewed in accordance with the guidelines provided in IEEE-279; however, significant variations in interpretations of that document resulted in substantial design differences in plants. Specifically, true physical separation of wiring to redundant components was not necessarily accomplished. In 1974, Regulatory Guide 1.75 was published, clarifying the requirements.

An earlier evaluation of this issue resulted in a medium-priority ranking (see Appendix C) with the finding that the scope could be limited to pipe breaks inside containment, since the NRC had already provided requirements regarding outside containment pipe breaks to the industry through the issuance of the Giambusso and O'Leary generic letters. However, the uncertainty in the analysis was much wider than desired for a definitive priority ranking. Thus, the issue appeared to warrant additional analysis to enhance the prioritization. In July 1994, a contract was awarded to INEEL to:

(1) Review pipe failure rate data, pipe break methodologies, and related publications to determine recommended pipe failure rates (initiating events) applicable to the affected SEP-III plants.

(2) Review updated FSARs and related SERs for SEP-II, SEP-III, and for representative non-SEP plants to identify and prioritize potential safety concerns (i.e., accident sequences). Several plant visits and walkdowns were included as part of this review.

(3) Estimate changes to core damage frequencies for accident sequences that are determined to be of high or medium priority.

(4) Identify potential corrective actions and their estimated costs.

The evaluation that follows was based on the results of the INEEL research documented in Draft NUREG/CR-6395..

Safety Significance

GDC 4 is the primary regulatory requirement of concern. It requires, in part, that structures, systems and components important to safety be appropriately protected against the *environmental and dynamic effects that may result from equipment failures, including* the effects of pipe whipping and discharging fluids. Several possible scenarios for plants that do not have adequate protection against pipe whip were identified as a result of the research performed in support of the enhanced prioritization. Related regulatory criteria include common cause failures, protection system independence, and the single failure criterion.

Possible Solution

Issue generic letters to the affected plants requesting that they perform plant-specific reviews and walkdowns, identify vulnerable pipe break locations, and inform the NRC of proposed corrective actions.

PRIORITY DETERMINATION

Numerous scenarios of potential concern were evaluated. The following were considered important enough to be specifically identified for future consideration. All estimated frequencies and probabilities are mean values.

Frequency Estimate

BWRs

Case 1: Failure of Main Steam or Feedwater Piping Resulting in Pipe Whip and Containment Impact/Failure, with Resultant Failure of All Safety Injection Systems

This event (INEEL BWR Event 1) involved a BWR with a Mark I steel containment; 15 of the 16 affected BWRs were of this design. A DEGB of an unprotected (i.e., no pipe whip restraint or containment liner impact absorber) large reactor coolant recirculation pipe inside containment and near the containment liner might result in puncturing of the liner. The resulting unisolable LOCA steam environment would be introduced into the secondary containment building, possibly disabling the ECCS equipment located there. This scenario would greatly increase the probability of core damage and potential offsite doses.

All of the affected BWRs were more than 10 years old and most used Type 304SS in the primary system piping, a material that was susceptible to IGSCC degradation. It should be noted that piping of this material did not qualify for the extremely low rupture probability (leak-before-break) provision of GDC 4. From NUREG-1150,[1081] the recirculation loop DEGB frequency for this material was estimated to be 10^{-4}/RY. The fraction of BWR primary piping inside containment that was either main steam or feedwater was estimated to be 0.4. The fraction of main steam or feedwater piping that can impact the containment metal shell was estimated to be 0.25.

The research performed indicated that there was considerable variation among the affected plants regarding the amount of pipe whip protection provided and the proximity of high energy lines to potential targets of concern, including redundant trains (see Other Considerations). It was assumed that the probability of a main steam or feedwater broken pipe rupturing the containment metal shell was 0.25.

The postulated event may also cause a common mode failure of the ECCS system since much of this equipment was located within the secondary containment and will be exposed to a harsh environment beyond its design basis, or that the ECCS piping will fail due to overpressurization of the containment annulus. In most of the affected plants, the ECCS is located in four different quadrants outside the suppression pool (torus). On the other hand, as stated above, redundant electrical power systems and initiating circuitry may not be physically separated in the older plants. Also, if the ECCS operates initially, the ECCS equipment rooms may not be fully protected from internal flooding as the water from the suppression pool flows out the broken pipe into the secondary containment. Based on these considerations, the mean probability of loss of ECCS function was assumed to be 0.8. Based on the above assumptions, the mean value of change in CDF was 2 x 10^{-6}/RY.

From WASH-1400,[16] the nearest scenario to that described above was the large LOCA BWR-3 release category involving a large LOCA and subsequent containment failure. However, in the WASH-1400[16] case, the containment failure results from overpressurization, not from pipe whip. Three of the four specific BWR-3 large LOCA accident sequences have an incidence frequency of 10^{-7}/RY, and the remaining one is 10^{-6}/RY; 10^{-7}/RY was chosen as the base case for this analysis.

<u>Case 2:</u> <u>Failure of Recirculation Piping Resulting in Pipe Whip and Containment Impact/Failure, With Resultant Failure of All Emergency Core Cooling Systems</u>

This event (INEEL BWR Event 9) was similar to Case 1 but involved the recirculation system piping. From NUREG-1150,[1081] the recirculation loop DEGB mean frequency for this material was estimated to be 10^{-4}/RY. The fraction of BWR primary piping inside containment that is recirculation piping was estimated to be 0.2. The fraction of recirculation piping that can impact the containment metal shell was estimated to be 0.5. It was estimated that the mean probability of a recirculation system broken pipe rupturing the containment metal shell was 0.5. The mean probability of eventual failure of all ECCS by the same modes described for Case 1 was estimated to be 0.8. Based on the above assumptions, the mean value of change in CDF was 4 x 10^{-6}/RY.

<u>Case 3:</u> <u>Failure of RHR Piping Resulting in Pipe Whip and Containment Impact/Failure, With Resultant Failure of All Emergency Core Cooling Systems</u>

This event (INEEL BWR Event 12) was similar to Cases 1 and 2 but involved the RHR System piping. From NUREG-1150,[1081] the RHR DEGB frequency for this material was estimated to be 10^{-4}/RY. The fraction of BWR primary piping inside containment that is RHR piping was estimated to be 0.1. The fraction of RHR piping that can impact the containment metal shell was estimated to be 0.5. The mean probability of a recirculation system broken pipe rupturing the containment metal shell was 0.1. The mean probability of eventual failure of all ECCS by the same modes described for Cases 1 and 2 was estimated to be 0.8. Based on the above assumptions, the mean value of change in CDF/RY was 4 x 10^{-7}/RY.

<u>Case 4:</u> <u>Failure of Recirculation Piping Resulting in Pipe Whip or Jet Impingement on Control Rod Drive Bundles, Causing Failure by Crimping of Enough Insert/Withdraw Lines to Result in Failure to Scram the Reactor</u>

This case corresponded to INEEL BWR Event 5. From NUREG-1150,[1081] the recirculation loop DEGB frequency for this material was estimated to be 10^{-4}/RY. The fraction of BWR primary piping inside containment that is recirculation piping was estimated to be 0.2. The fraction of recirculation piping that can impact or impinge on the CRD lines was estimated to be 0.25. It was estimated that the mean probability of a broken RHR pipe crimping enough CRD lines to prevent a scram (about 5 to 10 adjacent lines) was 1. Based on the above assumptions, the mean value of change in CDF was estimated to be 5×10^{-6}/RY.

<u>Case 5:</u> <u>Failure of RHR Piping Resulting in Pipe Whip or Jet Impingement on Control Rod Drive Bundles, Causing Failure by Crimping of Enough Insert/Withdraw Lines to Result in Failure to Scram the Reactor</u>

This event (INEEL BWR Event 10) was similar to Case 3 but involved the RHR system piping. The research performed indicated that there was considerable variation among the affected plants regarding the amount of pipe whip protection provided and the proximity of high energy lines to potential targets of concern. Walkdowns showed that, in at least one case, a large "unisolable from the RCS" RHR line was routed directly between the two banks of CRD bundles. An RHR pipe break in this vicinity would impinge and/or impact on both banks simultaneously.

From NUREG-1150,[1081] the RHR DEGB frequency for this material was estimated to be 10^{-4}/RY. The fraction of BWR primary piping inside containment that consitutes RHR piping was estimated to be 0.1. The fraction of RHR piping that can impact or impinge on the CRD lines was estimated to be 0.25. It was estimated that the mean probability of a broken RHR pipe crimping enough CRD lines to prevent a scram (about 5 to 10 adjacent lines) was 1. Based on the above assumptions, the mean value of change in CDF was 2.5×10^{-6}/RY.

<u>Case 6:</u> <u>Failure of High Energy Piping Resulting in Pipe Whip or Jet Impingement on Reactor Protection or Instrumentation & Control Electrical, Hydraulic or Pneumatic Lines, or Components and Eventually Resulting in Failure of Mitigation Systems and Core Damage</u>

This case corresponded to INEEL BWR Event 14. From NUREG-1150,[1081] the large LOCA frequency is 10^{-4}/RY. All high energy piping inside containment was considered. The fraction of high energy piping that can impact or impinge on these lines or components was estimated to be 0.5. The mean probability of a broken high energy line failing some of these lines or components to the extent that core damage results was estimated to be 0.75. Based on the above assumptions, the mean value of change in CDF was 3.8×10^{-5}/RY.

<u>Case 7:</u> <u>Failure of High Energy Piping Resulting in Pipe Whip Impact on Reactor Building Component Cooling Water (RBCCW) System to the Extent That the RBCCW Pressure Boundary is Broken, Potentially Opening a Path to Outside Containment if Containment Isolation Fails to Occur; Also Possible Loss of RBCCW Outside Containment for Mitigation</u>

This case corresponded to INEEL BWR Event 16. From NUREG-1150,[1081] the large LOCA frequency was 10^{-4}/RY. All high energy piping inside containment was considered. The fraction of high energy piping that can impact the RBCCW system was estimated to be 0.1. The probability of an HELB broken pipe rupturing the RBCCW system was 0.5. The probability of failure to close of containment isolation check valve was 10^{-3}; the probability of failure to close of a containment isolation MOV was 3×10^{-3}. These scenarios had a combined total probability of 4×10^{-3}. Since the RBCCW surge tank in the secondary containment is vented to atmosphere and has a relatively small volume, it was assumed that its water inventory will drain quickly; for this reason, the mean probability of opening a path to atmosphere outside containment was 1. Once this scenario proceeds to this point, the RBCCW system in the secondary containment will become unavailable, including the RHR heat exchanger; therefore, the probability of losing the RBCCW function outside containment to the extent that core damage occurs was 1. Based on the above assumptions, the mean value of change in CDF was estimated to be 2×10^{-8}/RY.

The total change in CDF for the above 7 BWR cases was estimated to be 5.2×10^{-5}/RY. For all 16 affected BWRs, ΔCDF was 8.3×10^{-4}/RY.

PWRs

Case 1: Failure of Non-Leak-Before-Break Reactor Coolant System, Feedwater, or Main Steam Piping Resulting in Pipe Whip or Jet Impingement on Reactor Protection or Instrumentation & Control Electrical, Hydraulic or Pneumatic Lines or Components and Eventually Resulting in Failure of Mitigation Systems and Core Damage

This case corresponded to INEEL PWR Event 9. From NUREG-1150,[1081] the HELB frequency in the above-listed systems was 1.5×10^{-3}/RY. All of the listed high energy piping inside containment was considered. The fraction of high energy piping that can impact or impinge on these lines or components was estimated to be 0.1. The mean probability of a broken high energy line failing some of these lines or components to the extent that core damage results was estimated to be 0.5. Based on the above assumptions, the mean value of change in CDF was 7.5×10^{-5}/RY.

Case 2: Failure of Main Steam or Feedwater Piping Resulting in Pipe Whip and Containment Impact/Failure, with Resultant Failure of All Emergency Core Cooling Systems

This case corresponded to INEEL PWR Event 16. From NUREG-1150,[1081] the DEGB frequency in feedwater piping was estimated to be 4×10^{-4}/RY; for main steam piping, it was estimated to be 10^{-4} /RY. The fraction of feedwater piping that can impact the containment shell was estimated to be 0.1. The fraction of main steam piping was also estimated to be 0.1; this fraction remained 0.1. The mean probability of a feedwater or main steam system broken pipe rupturing the containment metal shell was 0.5. The mean probability of additional I&C or ECCS systems failures to the extent that core damage results was estimated to be 4.8×10^{-5} for the case involving feedwater piping breaks, and 9.8×10^{-5} for the case involving main steam piping breaks. Based on the above assumptions, the mean value of change in CDF was 1.4×10^{-9}/RY.

Case 3: Failure of Main Steam or Feedwater Piping Resulting in Pipe Whip Impact on CCW System to the Extent That the CCW Pressure Boundary is Broken, Potentially Opening a Path to Outside Containment if Containment Isolation Fails to Occur; Also Possible Loss of CCW Outside Containment for Mitigation

This case corresponded to INEEL PWR Event 17. From NUREG-1150,[1081] the DEGB frequency in feedwater piping was estimated to be 4×10^{-4}/RY; for main steam piping, it was estimated to be 10^{-4}/RY; this combined for a total frequency of 5 x 10^{-4}/RY. The fraction of feedwater piping that can impact the CCW system was estimated to be 0.1; the fraction of main steam piping was also estimated to be 0.1; this fraction remained 0.1. The probability of a feedwater or main steam system broken pipe rupturing the CCW system was 0.5. The probability of failure to close of containment isolation check valve was 10^{-3}; the probability of failure to close of a containment isolation MOV was 3 x 10^{-3}; this combined for a total probability of 4 x 10^{-3}. Since the CCW surge tank is in the auxiliary building near mitigation equipment, is vented to atmosphere, and has a relatively small volume, it was assumed that its water inventory will drain quickly. For this reason, the mean probability of opening a path to atmosphere outside containment was 1. Once this scenario proceeds to this point, the CCW system outside containment will become unavailable, including the RHR heat exchanger. Therefore, the probability of losing the CCW function outside containment, to the extent that core damage occurs, is 1. Based on the above assumptions, the mean value of change in CDF was 10^{-7} /RY.

The total change in CDF for the above three PWR cases was 7.5 x 10^{-5}/RY. For all 25 affected PWRs, the ΔCDF was estimated to be 1.9 x 10^{-3}/RY.

Consequence Estimate

TABLE 3.156-1
BWR Offsite Dose Table

NUREG/CR-6395 Event Number	ΔCDF (Event/RY)	WASH-1400[16] Release Category	WASH-1400[16] Offsite Dose (Man-rem/Event)	Offsite Dose (Man-rem/RY)
Event 1	2.0 x 10^{-6}	BWR-3	5.1 x 10^{6}	10.2
Event 5	5.0 x 10^{-6}	BWR-4	6.1 x 10^{5}	3.1
Event 9	4.0 x 10^{-6}	BWR-3	5.1 x 10^{6}	20.4
Event 10	2.5 x 10^{-6}	BWR-4	6.1 x 10^{5}	1.5
Event 12	4.0 x 10^{-7}	BWR-3	5.1 x 10^{6}	2.0
Event 14	3.8 x 10^{-5}	BWR-4	6.1 x 10^{5}	23.2
Event 16	2.0 x 10^{-8}	BWR-3	5.1 x 10^{6}	0.1
TOTAL:				**60.5**

For the 16 affected BWRs with an average remaining life of 17 years, the estimated change in offsite dose was (60.5 man-rem/RY)(16 reactors)(17years) or 16,464 man-rem.

TABLE 3.156-2
PWR Offsite Dose Table

NUREG/CR-6395 Event Number	ΔCDF (Event/RY)	WASH-1400[16] Release Category	WASH-1400[16] Offsite Dose (man-rem/event)	Offsite Dose (man-rem/RY)
Event 9	7.5×10^{-5}	PWR-6	1.5×10^{5}	11.3
Event 16	1.4×10^{-9}	PWR-4	2.7×10^{6}	0.004
Event 17	1.0×10^{-7}	PWR-4	2.7×10^{6}	0.3
TOTAL:				**11.6**

For the 25 affected PWRs with an average remaining life of 17 years, the estimated change in offsite dose was (11.6 man-rem/RY)(25 reactors)(17 years) or 4,925 man-rem. Thus, the estimated total offsite dose for the 41 affected plants was (16,464 + 4,925) man-rem or 21,389 man-rem.

Cost Estimate

Industry Cost: Implementation of the possible solution was assumed to require the performance of engineering analyses inside containment, perform system walkdowns, and provide a report to the NRC. Ultimately, it was expected that operating procedures and/or TS will be modified, inservice inspections will be enhanced, or physical modifications will be done either to piping (probably addition of pipe whip restraints or jet shields) or to the inside containment leakage detection system. It is expected that the cost to each plant will be $1M. Therefore, for the 41 affected plants (16 BWRs and 25 PWRs), the total implementation cost was estimated to be $41M. This estimate was based on the presumption that the level of effort at the affected plants would be similar to that which resulted for this issue during the SEP program review of the 10 earliest SEP plants.

NRC Cost: Development and implementation of a resolution was estimated to cost $1M, primarily involving review of industry submittals and possible proposed changes to hardware.

Total Cost: The total industry and NRC cost associated with the possible solution was estimated to be $42M.

Impact/Value Assessment

Based on a potential public risk reduction of 21,389 man-rem and an estimated cost of $42M for a possible solution, the impact/value ratio was given by:

$$R = \frac{\$42M}{21{,}389 \text{ man-rem}}$$

$$= \$1{,}960/\text{man-rem}$$

Other Considerations

(1) The updated SAR for an SEP-III BWR (i.e., one of the 41 plants potentially affected by this issue) stated that, in the event of a DEGB, the broken pipe would strike the Mark I Containment and deform it significantly. However, another BWR of about the same vintage is known to have been required to add energy absorbing structures to protect the Mark I Containment from pipe whip, prior to receipt of an operating license. Therefore, it appeared that there was considerable variation among the affected plants regarding the amount of pipe whip protection provided.

(2) Pipe breaks have actually occurred in the industry. Examples include a Surry feedwater line break, a WNP-2 Fire System valve structural pressure boundary failure, and a Ft. Calhoun 12" steam line break.

(3) Some suspect configurations were observed in the SEP-III walkdown plants, e.g., at one BWR a very close proximity exists between a large RHR (unisolable from RCS) pipe and both banks of the CRD piping, and at one PWR it appeared that a large volume of piping penetrated the containment near where a large amount of electrical wiring also penetrated the containment. This demonstrated that, even through modest efforts (i.e., sampling walkdowns of a sampling of plants), configurations of potential concern have been identified.

(4) Readily available plant documentation provides very little insights regarding actual proximity of high energy piping and potential targets or concern. The potential lack of adequate separation of redundant system targets (e.g., I&C electrical wiring) is also a concern.

(5) Uncertainty remains a significant factor because of the large scope of this issue. This is because of the large number and types of plants, and significant differences in the specific as-built details applicable to this issue.

(6) Many of the affected plants are either currently applying for life extension or are expected to in the near future. Most of the lead life extension applications will be from the affected plants for many years to come.

(7) Although there is a large apparent disparity between the BWR and PWR cases evaluated, it must be remembered that much of the background of this issue was based on sampling walkdowns, i.e., only selected portions of selected plants were available for these walkdowns. Therefore, it is important to treat the BWR and PWR evaluations equally during the next phase of the evaluation. Also, some of the listed scenarios seem to have low probabilities but potentially high consequences. They should be further evaluated.

(8) Assuming a life extension of 20 years for the 31 affected plants, the public risk reduction would be 35,824 man-rem and 10,725 man-rem for BWRs and PWRs, respectively. This would produce an impact/value ratio of $900/man-rem.

CONCLUSION

Several potential accident scenarios were identified; 7 for BWRs and 3 for PWRs. Mean values for core damage were estimated for each and the cumulative effect of each group was also estimated. The total change in CDF was 8.3×10^{-4}/year for the 16 affected BWRs and 7.5×10^{-5}/RY for the 3 PWR cases. This would give the issue a medium/high priority ranking. For all 25 affected PWRs, ΔCDF/Year was 1.9×10^{-3}, which would also give the issue a high/medium priority ranking. Further evaluations which included estimates of offsite doses and costs for potential solutions showed that the issue has a HIGH priority ranking.[399]

For BWRs, the accident sequences of interest all involve a pipe whip that penetrates the steel primary containment wall, thereby discharging steam into the gap between that wall and the secondary concrete shield wall. Steam that is discharged into this gap will find its way to the rooms containing the equipment associated with the ECCS, which may fail due to the resulting harsh environment. Consequently, a severe core damage event could result, with the integrity of the primary containment already lost.

To address these scenarios, the staff performed a series of calculations using a nonlinear finite element program to estimate the effect of a pipe whip on the containment wall. The results of these calculations indicated that the containment wall would be dented, but not penetrated. In the more extreme cases, the dented steel containment wall would touch the concrete shield, but this contact would arrest any further displacement. Based on these calculations, the staff concluded that penetration of the steel wall has an exceedingly low probability of occurrence, and the BWR scenarios can be eliminated from further consideration.

Of the PWR accident scenarios, only one has an estimated frequency high enough to warrant further investigation. That sequence is initiated by a high-energy secondary system pipe break within containment. In such an event, pipe whip or jet impingement would then cause failure of instrumentation or control cables within containment, leading to failure of accident-mitigating systems.

Because of the variation in containment designs for the early PWRs, a generic approach was not possible. Instead, staff from RES and the Office of Nuclear Reactor Regulation (NRR) examined the cabling and piping layout of each SEP PWR that is still operating. In so doing, the staff discovered that some plant designs anticipated the SRP requirements for channel separation and separate penetrations on opposite sides of the containment. In other plant designs, cables were separated from piping by walls, floors, or other structures, or were spatially separated by significant distances. No instance was found where a whipping pipe or fluid jet would directly disable both channels of any safety-significant system. Thus, the staff concluded that the PWR scenario could also be eliminated from further consideration.

A technical assessment report was prepared and transmitted to the Advisory Committee on Reactor Safeguards (ACRS) on July 18, 2007. In addition, the staff met with the ACRS on September 6, 2007, and presented the rationale for closing this issue with no further actions. On September 26, 2007 the ACRS reviewed and formally endorsed the staff's recommendation that GI-156.6.1 be closed and that no further actions by the NRC staff of licensees with respect to this issue are necessary. [1895] The issue was closed in December 21, 2007. [1896] The background information, basis for the closeout, and the staff's technical assessment was presented in Enclosure 1 to the closure memorandum. [1896]

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ISSUE 163: MULTIPLE STEAM GENERATOR TUBE LEAKAGE

DESCRIPTION

This issue was identified[1031] to address the safety concern associated with potential multiple steam generator tube leaks during a main steam line break that cannot be isolated. This sequence could lead to core damage that could result from the loss of all primary system coolant and safety injection fluid in the refueling water storage tank. The issue was based on a DPO filed in June 1992.

BACKGROUND HISTORY

The issue was given 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 to technical specification requirements existing at that time) that allows a degradation-specific management approach. The staff discontinued this effort in 1997 after a regulatory analysis indicated that rule making was unnecessary. With Commission approval, the staff undertook an effort to develop a generic letter requesting that all PWR licensees submit proposed changes to their plant technical specifications that would ensure SG tube integrity is maintained. This generic letter initiative included a draft regulatory guide and sample technical specifications incorporating a programmatic, performance based strategy for ensuring SG tube integrity.

On December 1, 1997, the industry informed the staff of an industry initiative, NEI 97-06, "Steam Generator Tube Integrity Guidelines," which paralleled the above draft regulatory guide and which all PWR licensees had committed (among themselves) to implement. NEI 97-06 provides a programmatic, performance based approach to ensuring SG tube integrity. With commission approval, the staff put the above generic letter initiative on hold and worked with the industry to identify revised technical specifications which would be aligned with the NEI 97-06 initiative and which would ensure that all PWR licensee's are implementing programs which ensure that SG tube integrity will be maintained. This effort was completed in May 2005 with the staff's approval of the TSTF-449, Rev. 4 [1897] which includes a new standard technical specification template governing SG tube integrity.

The nature of the DPO evolved considerably in the years subsequent to 1991, adding additional concerns relating to alternate tube repair criteria, iodine spiking assumptions for radiological analysis, severe accidents, and many other concerns. The staff prepared a DPO consideration document which was provided to the 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[1898] on February 1, 2001 documenting its conclusions and recommendations. By memorandum dated May 11, 2001[1899], NRR and RES 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 Steam Generator Action Plan, the status of which was presented to the Commission in SECY-03-0080

[1900] and discussed at a Commission meeting on May 19, 2003. A copy of the NRC SG Action Plan, milestones, schedule, and current status can be found on the NRC public web page at: http://www.nrc.gov/reactors/operating/ops-experience/steam-generator-tube.html

The scope of the DPO issues and followup SG Action Plan tasks relevant to GSI 163 are those which could potentially impact needed SG tube inspection, maintenance and repair activities. In contrast, any needed actions 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, are outside the scope of GSI-163. Similarly, iodine spiking and radiological assessment issues are outside the scope of GSI-163. DPO issues outside the scope of GSI-163 will continue to be managed under the SG Action Plan umbrella.

STATUS

In response to NRC Generic Letter 2006-01[1901], all PWR licensees have submitted license amendment applications to change their technical specifications in accordance with TSTF-449[1897].

SG Action Plan tasks relevant to resolution of GSI-163 have been completed with the exception of task 3.1.k. SG Action Plan task 3.1.k involves evaluation of the conditional probabilities of multiple tube failures for risk assessment pertaining to SG alternate repair criteria. To support the needs of the GSI, the staff is actually performing this task from the broad standpoint of the integrity of the overall tube rather than being narrowly focused on tube locations with alternate repair criteria.

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

DESCRIPTION

Historical Background

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

Safety Significance

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

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

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

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

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

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

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

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

Possible Solution

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

PRIORITY DETERMINATION

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

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

PWR Ice Condenser

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

Frequency Estimate

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

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

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

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

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

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

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

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

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

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

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

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

Large Early Release Frequency (LERF) Estimate

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

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

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

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

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

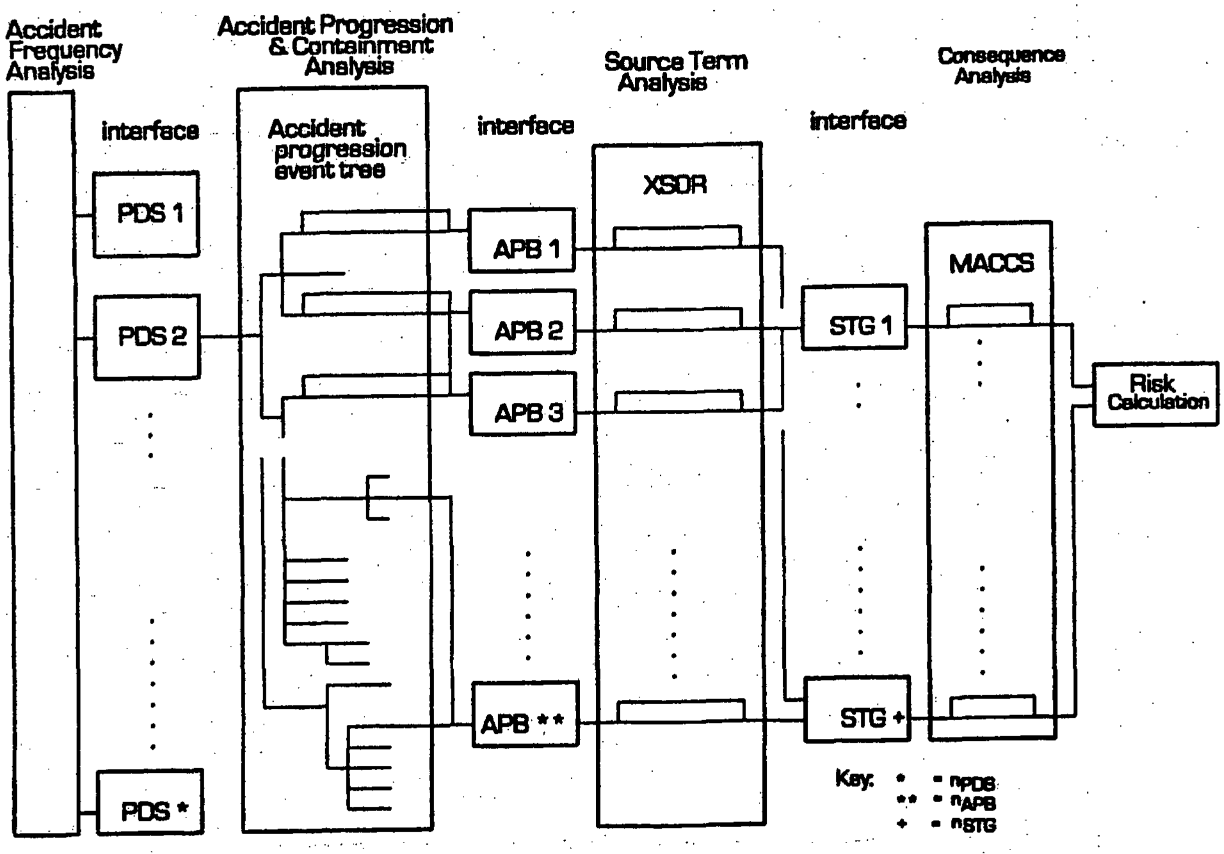
Hybridization: The various core damage frequencies and associated changes in LERF are based on a hybridization of several PRAs. Moreover, the estimates of the station blackout portion of the seismic and fire CDFs are, at best, educated guesses. Nevertheless, if the change in containment failure probability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

Consequence Estimate

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

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

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



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

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

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

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

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

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

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

Several other assumptions were necessary:

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

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

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

5th percentile -	3.86×10^{-3} man-rem
95th percentile -	20.3 man-rem

Median - 2.24 man-rem
Mean - 6.43 man-rem

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

Generic Population Distribution: The man-rem/RY figure above is based on the NUREG-1150[1081] model which is specific to the Sequoyah site. For generic issue calculations, such figures are generally based on a uniform population density of 340 persons/square-mile and a typical central Midwest plains meteorology. It is not practical to re-run the consequence analysis for the generic site but, as a first approximation, the risk figures can be re-normalized to the generic population. Interpolating between the 30- and 100-mile radius population figures given in NUREG/CR-4551[1795] (Volume 5, Rev. 1, Part 1, Page 4.2), the Sequoyah population density for a 50- mile radius is approximately 159 persons/square-mile. Thus, to get a generic risk figure, the 6.43 man-rem/RY (mean) figure should be multiplied by 340/159. This gives a generic estimate of 13.73 man-rem/RY.

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

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

Cost Estimate

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

Other Considerations

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

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

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

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

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

BWR MARK III Containments

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

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

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

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

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

Frequency Estimate

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

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

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

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

Site	NUREG-1150 Non-recoverable Fast SBO CDF	NUREG-1150 Non-recoverable Slow SBO CDF	IPE CDF	IPE SBO CDF	IPEEE Fire CDF	IPEEE Seismic CDF
Clinton			2.66E-5	9.80E-6	3.64E-6	SMA
Grand Gulf	4.3E-7 (11%)	6.6E-8 (2%)	1.72E-5	7.46E-6	8.89E-6	SMA
Perry			1.30E-5	2.25E-6	3.27E-5	SMA
River Bend			1.55E-5	1.35E-5	2.25E-5	SMA
LaSalle						7.6E-7
"Average"				8.25E-6	1.69E-5	7.6E-7
	From CRIC-ET database[1796]		From IPE database		From NUREG/CR-1742[1798] (Vol. 2, Table 3.2)	

Large Early Release Frequency (LERF) Estimate

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

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

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

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

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

Other Considerations

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

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

Consequence Estimate

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

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

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

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

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

All of the seismic sequences are slow, non-recoverable blackouts.

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

The other PDS frequencies were set to zero, so that the analysis would only include the non-recoverable station blackout plant damage states.

The sequences ending in no containment failure ("characteristic 6" in the Grand Gulf model - see NUREG/CR-4551[1795] (Volume 6, Rev. 1, Part 1, Table 2.4-1) were re-directed to the "rupture before vessel breach" accident progression bin, to account for the assumed high susceptibility of the containment to fail due to hydrogen combustion. This is a slight overestimate, since the model presumed that the igniters were not available in PDS 7 and 8 in any case.

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

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

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

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

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

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

Other Considerations

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

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

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

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

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

ASSESSMENT

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

The staff conducted studies to determine whether providing an independent power supply for the igniter systems provides a substantial increase in the overall protection of the public health and safety with implementation costs that are justified in view of the increased protection. The staff briefed the ACRS on June 6, 2002, and again on November 13, 2002. The ACRS recommended that the form of regulatory action should be through the plant-specific severe accident management guidelines. [1902] RES provided its technical assessment for resolving GI-189 to NRR in a memorandum dated December 17, 2002. [1903] RES concluded that further action to provide back-up to one train of igniters is warranted for both ice condenser and MARK III plants.

On January 30, 2003, NRR prepared a reply memorandum that outlined the next steps in the resolution of this GI. NRR prepared a Task Action Plan to complete MD 6.4, Stage 4, Regulation and Guidance Development, based on a preliminary decision to issue an Order. The staff reviewed the proposed regulatory actions and associated draft documents with senior management and OGC, and senior management decided to pursue Rulemaking rather than an Order. The staff held a public meeting on June 18, 2003, [1904] to receive feedback from licensees and other stakeholders regarding the need to provide a backup power supply to the hydrogen igniters and NRR's consideration of rulemaking for the resolution of GI-189. NRR staff briefed the ACRS on November 6, 2003, and recommended providing a backup power supply to the hydrogen igniters. On November 17, 2003, the ACRS Chairman wrote the NRC Chairman recommending the NRC proceed with rulemaking to require a backup power supply to the hydrogen igniters for PWR ice-condenser and BWR MARK III plants.[1905] The ACRS recommended that rulemaking include a small pre-staged generator with installed cables, conduit, panels, and breakers, or an equivalent diverse power supply. The ACRS also recommended that the rulemaking be accompanied by guidance that specifies the design requirements.

NRR developed design criteria for the backup power supply, and administered a contract to merge and enhance the existing technical assessment into a regulatory analysis.[1906] NRR held a public meeting with the public and industry on September 21, 2004,[1907, 1908] to get external stakeholders' input on the draft design criteria. In November 2004, the staff reached a consensus to evaluate the proposed voluntary initiatives from stakeholders and pursue that path as a preferential solution. The NRR staff met with representatives of RES, NSIR, and OEDO to develop an understanding of newly identified safety/security interface issues and actions initiated in the security arena that could impact the solution of the issue. On March 30, 2005, the staff met with senior representatives of the six affected utilities to present security-related insights.

On June 14, 2005, the EDO issued a memorandum to the Commissioners to inform the Commission of the regulatory analysis results and recent staff activities on GSI-189.[1909] The regulatory analysis indicated that the backup power modification may provide a substantial safety benefit at a justifiable cost for the PWRs with an ice-condenser containment, and the proposed voluntary actions provide the majority of the benefit. The costs exceed the benefits for all BWR regulatory options, and none of the options for the BWRs provides a substantial increase in the overall protection of public health and safety. However, external events and security insights were not fully evaluated in the regulatory analysis, and defense-in-depth considerations in improving the balance among accident prevention and mitigation provide an additional un-quantified benefit for both containment types.

STATUS:

Based on an understanding that many of the voluntary physical modifications had been completed, the staff elected to delay seeking specific commitments while security-related reviews of the facilities were ongoing. On March 1, 2006, the EDO issued a memo informing the Commission of the staffs' intent to delay the request for commitments until after the security-related reviews were completed in September 2006. Because this issue was not incorporated in the scope of security-related modifications, the staff has held closed meetings in December 2006 and January 2007 to further explore the proper consideration of security insights in the design of the modifications. The staff received industry proposals for modifications that incorporate security insights in late February and early March 2007. The staff reviewed the industry proposals and concluded that the proposed modifications would resolve GSI-189 and provide benefit for some security scenarios. On April 23, 2007, the EDO issued a memo informing the Commission of the staff's intent to accept the commitments and perform verification inspections at the affected sites. On June 15, 2007, the NRC staff issued letters to affected licensees accepting the commitments. The NRC staff also notified licensees of the intent to perform verification inspections at the affected sites and clarified the scope of the inspection relative to the commitments.

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ISSUE 191: ASSESSMENT OF DEBRIS ACCUMULATION ON PWR SUMP PERFORMANCE

DESCRIPTION

Results of research on BWR ECCS suction strainer blockage identified new phenomena and failure modes that were not considered in the resolution of Issue A-43. In addition, operating experience identified new contributors to debris and possible blockage of PWR sumps, such as degraded or failed containment paint coatings. Thus, this issue was identified[1691] by NRR and called for an expanded research effort to address these new safety concerns and to examine whether the events and new research being conducted for the BWR strainers warranted similar evaluation and/or changes for ensuring the adequacy of PWR recirculation performance.

BACKGROUND

A study was deemed to be required to determine whether PWR ECCS sumps are adequate to ensure proper ECCS operation. Based on the existence of an action plan[1692] to address the safety concerns, the issue was considered nearly-resolved in September 1996. It was later given a HIGH priority ranking in SECY-98-166.[1718]

Preliminary parametric calculations were completed in July 2001 indicating the potential for debris accumulation for 69 cases. These 69 cases were representative of, but not identical to, the operating PWR population. The staff's Technical Assessment concluded that GSI-191 was a credible concern for the population of domestic PWRs, and that detailed plant-specific evaluations were needed to determine the susceptibility of each U.S.-licensed PWR to ECCS sump blockage. Following the ACRS agreement with the staff's Technical Assessment of the issue in 09/2001, the issue was forwarded to NRR. [1910] NRR has evaluated the technical assessment, and prepared a Task Action Plan [1911] for developing appropriate regulatory guidance and resolution of GSI-191.

Following meetings with stakeholders, the NRC issued Bulletin 2003-01 [1912] to PWR licensees to: (1) confirm their compliance with 10 CFR 50.46 (b)(5) and other existing applicable regulatory requirements, or (2) describe any compensatory measures that have been implemented to reduce the potential risk due to post-accident debris blockage, as evaluations to determine compliance proceed. All PWR licensees provided a response to the Bulletin, indicating interim compensatory measures and candidate operator actions that would be implemented. Closure letters were issued to the PWR licensees as these reviews were completed and generic close-out of Bulletin 2003-01 [1912] was completed in December 2005. Responses to the Bulletin from licensees and Bulletin closure letters are available at: http://www.nrc.gov/reactors/operating/ops-experience/pwr-sump-performance/bulletin03-01-correspondence.html

The NRC staff concluded that plant-specific analyses should be conducted to determine whether debris accumulation in PWR containments could impede or prevent ECCS operation during recirculation, and that appropriate corrective actions should be taken if necessary. This expectation was communicated to licensee via Generic Letter 2004-02. [1913]

GL 2004-02 [1913] was issued in September 2004 requesting licensees to perform plant-specific mechanistic evaluations of sump performance following LOCA and HELB events, and to implement corrective actions as required to ensure compliance with regulatory requirements. NEI provided a guidance report (GR) to the staff in May 2004 containing the industry's proposed evaluation methodology for performing the plant specific evaluations. The staff reviewed the GR and issued a draft Safety Evaluation (SE), which supplemented the GR. The staff presented the SE to CRGR and to the ACRS Subcommittee and Full Committee in September and October 2004, respectively. The final SE was issued in December 2004, [1914] resulting in an NRC-approved evaluation methodology.

GL 2004-02 [1913] required licensees to respond within 90 days to document the actions planned by the licensee to perform the sump evaluation, and the proposed schedule for completion. All PWR licensees responded to the GL on schedule in September 2005. All PWR licensees committed to modify their containment sump strainer, except for three plants who had modified their containment sump strainers within the last five years. The staff evaluated all 90-day responses to GL 2004-02 [1913] and in January 2006 issued comments to licensees to be addressed in their final response submittals. (Licensees responses and NRC's comments are available at: http://www.nrc.gov/reactors/operating/ops-experience/pwr-sump-performance/generic04-02-correspondence.html)

To address concerns regarding the potential for chemical precipitates and corrosion products to significantly block a fiber bed and increase the head loss across an ECCS sump screen, a joint NRC/Industry Integrated Chemical Effects Testing program was started in 2004 and completed in August 2005. Chemical precipitation products were identified during the test program, and follow-up testing and analyses will be needed to address the effect on head loss. IN 2005-26, [1915] "Results of Chemical Effects Head Loss Tests in a Simulated PWR Sump Pool Environment," was issued on September 16, 2005.

The NRC conducted additional research in certain areas to support evaluation efforts and provide confirmatory information. These areas include research on chemical effects to determine if the pressurized-water reactor sump pool environment generates byproducts which contribute to sump clogging, research on pump head losses caused by accumulation of containment materials and chemical byproducts, and research to predict the chemical species that may form in these environments. The staff completed reports on the chemical effects on ice condenser containments, [1916] and on PWR containments. [1917] Supplement 1 to IN 2005-26 [1918] was issued on January 26, 2006 to specifically provide additional information regarding test results related to chemical effects in environments containing dissolved phosphate (e.g., from trisodium phosphate) and dissolved calcium.

Between July and September 2006, the staff completed research on: (1) the thermodynamic simulation of containment sump pool chemical constituents, to predict the chemical reactions/byproducts in the pools; (2) the pressure loss across containment sump screens due to fiber insulation, chemical precipitates, and coating debris; and (3) a literature survey to summarize the knowledge base to date on the potential contribution of material leached from containment coatings to the chemical products formed in the containment sump pool, after a loss-of-coolant accident.

STATUS

As part of the plan to confirm adequate implementation and resolution of GSI-191, the NRC has conducted detailed plant audits examining the analyses and design changes used to address the technical issues. Visits to strainer vendor test facilities have often been included as part of this audit process. The NRC staff is also systematically evaluating remaining technical questions related to GSI-191 to support a decision on whether additional confirmatory research is needed and if so, on what time frame. In early 2008 the NRC issued additional guidance [1919] on several important subjects related to GSI-191 for the use of licensees and NRC staff reviewers of licensee submittals. This guidance is intended to help ensure an adequate technical basis is provided for conclusions that corrective actions are complete and sufficient. In December 2007 the NRC approved topical reports intended to address chemical and ex-vessel downstream effects [1920, 1921]. An additional topical report on in-vessel downstream effects is under NRC review.

In addition to the plant audits identified above, the staff will use inputs from review of licensee responses to GL 2004-02 [1913] and items identified from Regional inspections using Temporary Instruction TI-2515/166 [1922] to support closure of GSI-191. Inspections by regional staff will verify proper implementation of planned modifications. All licensees submitted supplemental responses to GL 2004-02 in early 2008. However, the NRC had authorized many plants some additional time (generally on the order of a few months) to complete one or more specified corrective actions (e.g., a particular plant modification or a test of strainer function). Therefore, some of the supplemental responses received in early 2008 did not support a final conclusion that the licensee had fully addressed GL 2004-02. In such cases an additional response will be due within 90 days of completion of the last corrective action for a given plant. The NRC expects that only a very few corrective actions will occur later than the end of 2008.

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ISSUE 199: IMPLICATIONS OF UPDATED PROBABILISTIC SEISMIC HAZARD ESTIMATES IN CENTRAL AND EASTERN U.S. FOR EXISTING PLANTS

DESCRIPTION

Historical Background

On May 26, 2005, the Office of Nuclear Reactor Regulation's Division of Engineering (NRR/DE) recommended that issues related to a closed generic seismic issue (GI-194, "Implications of Updated Probabilistic Seismic Hazard Estimates," dated September 23, 2003), and the impact of higher seismic hazard on current nuclear power plants in the Central and Eastern United States (CEUS) region, be examined under the Generic Issue (GI) identification and resolution process.[1930] On June 9, 2005, GI-199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States," joined the list of GIs.[1931]

Safety Significance

Recent data and models indicate that estimates of the potential for earthquake hazards for some nuclear power plants in the Central and Eastern United States may be larger than previous estimates. While it has been determined that currently operating plants remain safe, the recent seismic data and models warrant further study and analysis. This further analysis will allow the NRC to better understand the current margins at plants for earthquakes.

Regulatory Guide 1.165,[1932] developed in the early 1990s, specifies a reference probability for exceedance of a safe shutdown earthquake (SSE) ground motion, i.e., seismic hazard, at a median annual value of 1E-5. This reference probability value is based on the annual probability of exceeding the SSEs for 29 CEUS nuclear power sites and is used to establish the SSEs for future nuclear facilities. Based on preliminary results from work performed by the United States Geological Survey (USGS) in 2004, it appears the reference probability for the 29 CEUS has increased to about 6 to 7E-5. The increase in the reference probability value is primarily due to recent developments in the modeling of earthquake ground motion in the CEUS. When the staff first identified this issue, no new plants had applied for a Construction Permit or Early Site Permit (ESP) since 10 CFR Part 100 was revised and Regulatory Guide 1.165 [1932] was issued in 1997. When the staff began review of the ESP applications, the staff realized the impact of the revised regulation and the regulatory guide as they relate to future plants and operating reactors.

From the staff's review of the ESP applications with support from the 2004 USGS draft report, it appeared that the perception of seismic hazard for operating plants in the CEUS region had increased. Based on the evaluations of the Individual Plant Examination of External Events (IPEEE) Program, the staff had determined that seismic designs of operating plants in the CEUS provided an adequate level of protection. However, in light of the preliminary results from the USGS work of 2004 and ESP applications, the staff also recognized that the probability of exceeding the SSE at some of the currently operating sites in the CEUS is higher than previously understood. Therefore, the staff initiated this GI to assess the impact of increased estimates of seismic hazards on selected current nuclear power plants in the CEUS region that might be impacted by the updated seismic research, information, and models.

SCREENING ANALYSIS

The staff completed the screening analysis using guidance contained in MD 6.4 [1858] and SECY-07-0022 [1888] in December 2007, and the screening panel reviewed the analysis in January 2008. On February 1, 2008, the Director of the Office of Nuclear Regulatory Research (RES) approved the screening panel recommendation [1933] to begin the Safety/Risk Assessment Stage of the Generic Issue Process.

The screening panel's recommendation was based on the screening analysis which showed that the current knowledge of this issue and its potential impact on Central and Eastern United States (CEUS) plants passed the seven GI screening criteria. The discussion under each criterion below provides the screening analysis for GI-199.

1. The issue affects public health and safety, the common defense and security, or the environment.

The estimated risk to public health and safety and the environment associated with the occurrence of seismic events at some nuclear power plant (NPP) sites might have increased from previous estimates. The issue stems from ongoing research being conducted by a number of scientists into the seismic history of the CEUS and the details of wave propagation and attenuation in this region. In particular, information submitted to the NRC by ESP applicants contained updated seismic information that included new models to estimate earthquake ground motion and updated models for earthquake sources in seismic regions such as eastern Tennessee, and around both Charleston, South Carolina, and New Madrid, Missouri. In addition, information summarized by the USGS as part of the National Seismic Hazard Mapping Program indicates that the estimated likelihood of seismic activity (i.e., seismic hazard) in some CEUS locations has increased from previous estimates. Some of these locations are near existing NPP sites. An increase in the seismic hazard at these sites has the potential to adversely impact public health and safety if the estimated increased seismic hazard were to significantly exceed plant design capabilities; substantially reduce perceived safety margins for plant structures, systems, and components important to safety; or appreciably increase the risk associated with the plant's response to a seismic event. From a qualitative perspective, if the increased hazard is significant at sites that have relatively small safety margins for seismic events, then the estimated risk for these sites could increase.

2. The issue applies to two or more facilities and/or licensees/certificate holders or holders of other regulatory approvals.

The updated information described above results in increased estimates of the seismic hazard that could occur at multiple, although not all, NPP sites in the CEUS. Specifically, updated models for earthquake sources in seismic regions such as eastern Tennessee, and around both Charleston, South Carolina, and New Madrid, Missouri indicate the rate of earthquake occurrence in these regions is greater than previously recognized. Since this change applies to several large regions, it has the potential to affect more than one NPP site. Further, new models used to estimate earthquake ground motion have been revised relative to those used in the 1980's. This change also has the potential to affect more than one NPP site. Updated estimates of seismic hazard values at some of the sites could potentially exceed the design basis as well as the review level earthquake spectrum used as part of the IPEEE Program.

3. The issue cannot be readily addressed through other regulatory programs and processes; existing regulations, policies, or guidance; or voluntary industry initiatives.

In a memorandum to RES, dated May 26, 2005, NRR identified this issue and recommended that it be examined under the Generic Issues Program (GIP).[1934] In this memorandum the staff concluded that seismic designs of operating plants in the CEUS still provide adequate safety margins while the staff continues to evaluate new seismic hazard data and models and their potential impact on plant risk estimates. At the same time, the staff also recognized that this new seismic data and models could reduce available safety margins due to increased estimates of the probability associated with seismic hazards at some of the currently operating sites in the CEUS. Therefore, to help assess potential reduction in available safety margins using a probabilistic approach, the staff of NRR recommended in a memorandum to RES dated May 26, 2005 [1934], that the new data and models on CEUS seismic hazards be examined under the GIP. Accordingly, at the time of the memorandum, the NRR staff determined that this issue was not sufficiently characterized to address it under existing licensing processes for licensees of plants that might be impacted. In a memorandum dated June 9, 2005, RES informed NRR that the issue would be accepted into the GIP for screening in accordance with MD 6.4.[1858]

Based on the limited evaluation of available information, this issue does not appear to be adequately characterized for complete treatment under existing regulatory programs and processes. Examples of regulatory programs and processes that might apply after obtaining additional information and performing further evaluations are listed below. Additional analysis will help determine whether this issue is amenable to these or other regulatory programs or industry initiatives.

- LIC-100, "Control of Licensing Bases for Operating Reactors"
- LIC-105, "Managing Regulatory Commitments Made By Licensees to the NRC"
- LIC-202, "Procedures for Managing Plant-Specific Backfits and 50.54(f) Information Requests"
- LIC-300, "Rulemaking Procedures"
- LIC-400, "Procedures for Controlling the Development of New and Revised Generic Requirements for Power Reactor Licensees"
- LIC-401, "NRR Reactor Operating Experience Program"
- LIC-501, "Program Coordination for Risk-Informed Activities"
- LIC-503, "Generic Communications Affecting Nuclear Reactor Licensees"
- LIC-504, "Integrated Risk-Informed Decision-Making Process for Emergent Issues"

4. The issue can be resolved by new or revised regulation, policy, or guidance.

Further analysis of the risk or safety impact would provide sufficient additional information to properly characterize the issue and its potential impact on CEUS plants and support consideration under other existing regulatory programs or industry initiatives. The regulatory office has authority to take appropriate regulatory action(s) as necessary to protect the public health and safety and the environment. Depending on the outcome of the additional analysis, as well as industry initiatives to address any safety issues, the regulatory office could address this issue through one or more actions involving regulation, policy, or guidance.

5. The issue's risk or safety significance can be adequately determined (i.e., it does not involve phenomena or other uncertainties that would require long-term studies and/or experimental research to establish the risk or safety significance).

The assessment performed thus far is based on the staff's review of updated seismic data and models submitted by ESP applicants and also updated seismic hazard data and models available from the USGS as part of the National Seismic Hazard Mapping Program. The seismic hazard at CEUS plant sites of interest can be evaluated using an approach like the detailed assessment performed by EPRI [1935] for 28 of the 29 sites included in Regulatory Guide 1.165.[1932] This study used updated attenuation models and incorporated updates to the EPRI seismic source model developed during the preparation of the Early Site Permits. The risk significance of the updated seismic hazard information can be evaluated for CEUS plant sites of interest by performing a comparison of uniform hazard spectra or other hazard results to the beyond-design-basis review level earthquake or hazard curve used as part of the IPEEE evaluation.[1798] The available IPEEE Program results will allow a general assessment of the potential safety impact of increases in seismic hazard at specific sites. This analysis could be performed as part of the Safety/Risk assessment under the GIP and could also include participation by industry stakeholders, if appropriate.

6. The issue is well defined, discrete, and technical.

The seismic hazard will be adequately defined upon detailed assessment of available updated seismic data and models submitted by ESP applicants and also updated seismic hazard data and models available for other CEUS plant sites of interest using an approach like that performed by EPRI [1935] for 28 of the 29 sites included in Regulatory Guide 1.165.[1932] This will allow the seismic hazard estimates for CEUS plant sites of interest to reflect the state of current knowledge. As new information and research becomes available, future updates might be warranted. The plants' response to seismic hazards involves technical analyses using established techniques.

7. Resolution of the issue may potentially involve review, analysis, or action by the affected licensees, certificate holders, or holders of other regulatory approvals.

After further characterization of site-specific seismic hazards and an analysis of the plant's response to the increased seismic hazard, some plants may be identified as having a vulnerability that must be addressed to maintain adequate safety margins. Determining a plant's margin and potential need for action to maintain an adequate margin could involve regulatory actions (e.g., requests for information from plant licensees, reviews, additional analysis, mitigation actions, physical enhancements, administrative controls) for some plant licensees or could involve actions by industry stakeholders.

CONCLUSION

The screening analysis shows that the current knowledge of this issue and its potential impact on CEUS plants passes the seven GI screening criteria and, therefore, warrants further analysis under the GIP.

The screening analysis shows that the estimated increase in spectral acceleration for some existing CEUS plant sites might exceed the design basis and values used for the NRC's review of IPEEE submittals. This translates to an equivalent increase in seismic demand on plant structures, systems, and components. As a result, this issue has the potential to result in increased seismic core damage frequency estimates for some plants. However, the screening analysis provided a limited evaluation that did not assess the safety response of the plants. The next phase of the analysis under the GIP assesses the risk impact at plants where the estimated increase in seismic hazard exceeds previous levels to an extent that might challenge available seismic margins.

The results from the limited scope screening analysis is that seismic designs of operating plants in the CEUS still provide adequate safety margins while the staff continues to evaluate new seismic hazard data and models and their potential impact on plant risk estimates. Specific reasons for this conclusion include:

- The estimated annual probability of exceedance of seismic hazard is small in an absolute sense.
- Earthquakes cause ground motion over a range of frequencies. Lower frequency motions are more damaging to buildings and equipment than higher frequency motions. Based on the NRC staff's reviews associated with ESPs, the staff is confident that the recent seismic data and models will show that increased estimates of the seismic hazards will occur primarily in the higher ground motion frequencies. Accordingly, the staff anticipates that these increased estimates of seismic hazards would primarily have little impact on previous estimates of the potential damage to buildings and equipment.
- The plants are designed to withstand anticipated earthquakes with substantial design margins. Plants may have seismic margins beyond those reflected in their IPEEE submittals and these could compensate for the increase in estimated seismic load. Such additional seismic margins at plants may be inherent in the design and construction, realized from improved data and analysis methods, or result from plant modifications or enhancements completed since the IPEEE submittals.

GI-199 is now in the Safety/Risk Assessment stage of the GIP, in accordance with MD 6.4 [1858] and SECY-07-0022.[1888]

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

This appendix contains a listing of those residual GSIs that are applicable to operating and future reactor plants and includes: issues that have been resolved with requirements [I, NOTE 3(a)]; USI, HIGH- and MEDIUM-priority issues scheduled for resolution; nearly-resolved issues scheduled for resolution (NOTES 1 and 2); and issues that are scheduled for prioritization (NOTE 4). The priority designations for all issues are consistent with those listed in Table II of the Introduction. In accordance with 10 CFR 52.47(a)(1)(iv), any future application for design certification must contain proposed technical resolutions for the issues in this listing that are designated USI, HIGH, MEDIUM, NOTE 1, and NOTE 2. (In July 1998, the priority categories NOTES 1 and 2 were eliminated and all GSIs in these categories were given a HIGH priority ranking.[1718]) Also included in this listing are those GSIs that were either prioritized or resolved with no impact on operating reactor plants but contain recommendations for future reactor plants (NOTE 6).

Legend

NOTES: 1	- Possible Resolution Identified for Evaluation (Discontinued 07-06-98)
2	- Resolution Available [Documented in NUREG, NRC Memorandum, SER or equivalent] (Discontinued 07-06-98)
3(a)	- Resolution Resulted in the Establishment of New Regulatory Requirements [Rule, Regulatory Guide, SRP Change, or equivalent]
4	- Issue to be Prioritized in the Future
6	- New Requirements for Future Plants Recommended
B&W	- Babcock & Wilcox Company
CE	- Combustion Engineering Company
GE	- General Electric Company
CONTINUE	- Work on the issue continues in accordance NRC Management Directive 6.4[1858]
HIGH	- High Safety Priority
I	- Resolved TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737
MEDIUM	- Medium Safety Priority
MPA	- Multiplant Action
NA	- Not Applicable
TBD	- To Be Determined
USI	- Unresolved Safety Issue
W	- Westinghouse Electric Corporation

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

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
<u>I.C</u>	<u>OPERATING PROCEDURES</u>						
I.C.1	Short-Term Accident Analysis and Procedures Revision	-	-	-	-	-	-
I.C.1(1)	Small Break LOCAs	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
<u>I.D</u>	<u>CONTROL ROOM DESIGN</u>						
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
<u>I.F</u>	<u>QUALITY ASSURANCE</u>						
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
<u>I.G</u>	<u>PREOPERATIONAL AND LOW-POWER TESTING</u>						
I.G.1	Training Requirements	I	All	All		NA	06/26/80
I.G.2	Scope of Test Program	NOTE 3(a)	All	All		NA	07/--/81

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

Appendix B (Continued)

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

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

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

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

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

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 Dyanmic Loads - Long Term Program (former USI)	NOTE 3(a)	GE	NA		08/--/81	08/--/81
A-9	ATWS (former USI)	NOTE 3(a)	All	All		06/26/84	06/26/84

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

Appendix B (Continued)

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

Action Plan *Item/Issue No.*	Title	Safety *Priority/Status*	Affected NSSS Vendor		Operating *Plants-* MPA No	Operating *Plants -* Effective Date	Future *Plants-* Effective Date
			BWR	PWR			
	Sites						
155	Generic Concerns Arising from TMI-2 Cleanup	-	-	-	-	-	-
155.1	More Realistic Source Term Assumptions	NOTE 3(a)	All	All	NA	NA	02/--/95
163.	Multiple Steam Generator Tube Leakage	HIGH	NA	All		TBD	TBD
177.	Vehicle Intrusion at TMI	NOTE 3(a)	All	All		08/01/94	08/01/94
186.	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants	CONTINUE	All	All		TBD	TBD
189.	Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Combustion During A Severe Accident	CONTINUE	All	All		TBD	TBD
191.	Assessment of Debris Accumulation on PWR Sump Performance	HIGH	NA	All		TBD	TBD
193.	BWR ECCS Suction Concerns	CONTINUE	All	NA		TBD	TBD
199.	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States	CONTINUE	All	All		TBD	TBD

HUMAN FACTORS ISSUES

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

APPENDIX F

NUCLEAR MATERIAL SAFETY AND SAFEGUARDS GSIs

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

TABLE F.1

LISTING OF NMSS GSIs

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

Legend

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

Issue No.	Title	Priority Engineer	LeadOffice/ Division/Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date
NMSS-0001	Door Interlock Failure Resulting from Faulty MicroSelectron-High Dose Rate Remote Afterloader	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0002	Significant Quantities of Fixed Contamination Remain in Krypton-85 Leak-Detection Devices After Venting	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0003	Corrosion of Sealed Sources Caused by Sensitization of Stainless Steel Source Capsules During Shipment	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0004	Overexposures Caused by Sources Stolen from Facility of Bankrupt Licensee	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0005	Potential for Erroneous Calibration, Dose Rate, or Radiation Exposure Measurements With Victoreen Electrometers	Ramsey	NMSS/IMNS/IMOB	NOTE 3(a)		12/31/1998
NMSS-0006	Criticality in Low-Level Waste	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)	1	06/30/2000
NMSS-0007	Criticality Benchmarks Greater Than 5% Enrichment	C. Hrabal	NMSS/FCSS	NOTE 3(b)	2	06/30/2008
NMSS-0008	Year 2000 Computer Problem - Non-Reactor Licensees	Ramsey	NMSS/IMNS	NOTE 3(b)	1	06/30/2000
NMSS-0009	Amersham Radiography Source Cable Failures	Ramsey	NMSS/IMNS	NOTE 3(b)		12/31/1998
NMSS-0010	Troxler Gauge Source Rod Weld Failures	Ramsey	NMSS/IMNS	NOTE 3(b	1	06/30/2002

Issue No.	Title	Priority Engineer	LeadOffice/ Division/Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date
NMSS-0011	Spent Fuel Dry Cask Weld Cracks	Ramsey	NMSS/SFPO	NOTE 3(b)		12/31/1998
NMSS-0012	Inadequate Transportation Packaging Puncture Tests	Ramsey	NMSS/SFPO	NOTE 3(b)	1	06/30/2000
NMSS-0013	Use of Different Dose Equivalent Models to Show Compliance	Ramsey	NMSS/IMNS	NOTE 3(b)	1	06/30/2000
NMSS-0014	Surety Estimates for Groundwater Restoration at In-Situ Leach Fields	Ramsey	NMSS/DWM	NOTE 3(b)	1	06/30/2007
NMSS-0015	Adequacy of 10 CFR 150 Criticality Requirements	Ramsey	NMSS/DWM	NOTE 3(b)	1	06/30/2000
NMSS-0016	Adequacy of 0.05 Weight Percent Limit in 10 CFR 40	Ramsey	NMSS/IMNS	NOTE 3(b)	1	06/30/2007
NMSS-0017	Misleading Marketing Information to General Licensees	C. Mattsen	NMSS/IMNS	NOTE 3(a)		06/30/2001
NMSS-0018	Problems Encountered When Manually Editing Treatment Planning Data on Nucletron MicroSelectron-HDR Model 105.999	B. Ayres	NMSS/IMNS	NOTE 3(b)		06/30/2001
NMSS-0019	Control Unit Failures of Classic Nucletron HDR Units	B. Ayres	NMSS/IMNS	NOTE 3(b)		06/30/2001
NMSS-0020	Leaking Pools	M. Sitek	NMSS/IMNS	DROP		06/30/2001
NMSS-0021	Unlikely Events	M. Sitek	NMSS/IMNS	DROP		06/30/2001
NMSS-0022	Gamma Stereotactic Radiosurgery	M. Sitek	NMSS/IMNS	DROP		06/30/2001

NMSS-0007: CRITICALITY BENCHMARKS GREATER THAN 5% ENRICHMENT

DESCRIPTION

The importance of computer software (methods and data) in establishing the criticality safety of systems with fissile material is increasing as licensees work to optimize facilities and storage/transport packages at the same time that access to experimental data is decreasing. Available experimental data are insufficient to validate nuclear criticality safety evaluations for all required configurations at U^{235} enrichments in the range of 5% to 20%. This issue was identified[1709] by NMSS to develop and confirm the adequacy of methods, analytical tools, and guidance for criticality safety software to be used in licensing nuclear facilities.

BACKGROUND

Computer codes used for criticality calculations must be benchmarked against critical experiments that represent the specific fissile materials, configurations, moderation, and neutron-poisoning conditions that represent the facility being licensed. However, it is well recognized that existing critical benchmark experiments will never precisely match these conditions. In addition, there are fewer benchmark experiments that are available at higher enrichment ranges [e.g., between 5 to 20 percent and lower-moderation (i.e., H/X, where H is hydrogen and X is fissile media)] ranges, that could be of future interest to potential applicants. Methods are needed to extend the range of applicability of current benchmark experiments via sensitivity/uncertainty (S/U) analysis techniques.

NMSS has performed extensive work with Oak Ridge National Laboratory (ORNL) to further develop criticality safety computer codes [e.g., Standardized Computer Analyses for Licensing Evaluation (SCALE)] to address these challenges. The final reports for the S/U methods were published in November 1999 as Volumes I [1923] and 2 [1924] of NUREG/CR-6655. The reports covered the following subjects: (1) methodology for defining range of applicability, including extensions of enrichments from 5 to 11 percent; (2) test applications and results of the method; (3) test application for higher enrichments using foreign experiments; and (4) feasibility study for extending the method to multidimensional analyses, such as transport casks and reactor fuel.

DISCUSSION

Results of the test applications of the ORNL methods showed that, for simple geometries with neutron spectra that are well-moderated (high H/X), benchmark experiments at 5 percent enrichment are applicable to calculations up to 11 percent enrichment. On the other hand, these test applications also show that benchmark experiments at intermediate and higher H/X values are not applicable to calculations at very low H/X. There are relatively few benchmarks at these very low H/X values for many compositions of interest to low-enriched uranium licensees.

Although licensees must apply the ORNL method to each individual process, to determine an acceptable subcritical margin, the results indicated that there may be situations where there are no applicable benchmarks. In these cases, the method provides sensitivity and uncertainty information, to help designers allow adequately large margins to cover the lack of benchmark validation. The computer codes [(i.e., Tools for Sensitivity and Uncertainty Analysis Methodology Implementation (TSUNAMI)] for S/U methods were incorporated into the release of SCALE 5.0 in 2004. The TSUNAMI code in SCALE 5.0 systematically quantifies the degree of similarity between a set of critical experiments and applications. For those applications where few benchmarks exist, as described previously for low H/X values, the TSUNAMI code can be used to apply adequately large margins to ensure the application is properly validated by SCALE 5.0.

In June 2006, staff issued Fuel Cycle Safety and Safeguards (FCSS) Interim Staff Guidance (ISG) -10 [1925], "Justification of Minimum Margin of Subcriticality for Safety". The ISG clarified guidance to the NRC staff when reviewing criticality safety analyses in integrated safety analysis, license applications, or amendment requests or other related licensing activities for fuel cycle facilities, under 10 CFR Part 70. The ISG communicates the acceptability of the TSUNAMI computer code in SCALE 5.0, as one method for determining minimum margins of subcriticality with limited benchmark experiments. For applications where few benchmarks exist, TSUNAMI can be used to apply larger margins to ensure validity of the SCALE criticality codes. Further benchmark experiments may be needed if future applicants request lower margins.

CONCLUSION

The issue was initially given a low priority ranking[1709] which was later changed to a high priority.[1787] This issue was closed out [1926] after the issuance of FCSS ISG-10, as the final milestone required to close out the GI.

REFERENCES

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

1787. Memorandum to F. Eltawila from D. Cool, "NMSS Input for Second Quarter FY-2001 Update of the Generic Issue Management Control System," April 12, 2001.

1923. NUREG/CR-6655, Vol. 1, "Sensitivity and Uncertainty Analyses Applied to Criticality Safety Validation, Methods Development," U.S. Nuclear Regulatory Commission, November 1999. [ML003726900]

1924. NUREG/CR-6655, Vol. 2, "Sensitivity and Uncertainty Analyses Applied to Criticality Safety Validation, Illustrative Applications and Initial Guidance," U.S. Nuclear Regulatory Commission, November 1999. [ML003726890]

1925. Fuel Cycle Safety and Safeguards (FCSS) Interim Staff Guidance (ISG) -10, "Justification of Minimum Margin of Subcriticality for Safety," June 15, 2006. [ML061650370]

1926. Memorandum to L. Reyes from M. Weber, "Closure of Generic Issue NMSS-0007, Criticality Benchmarks Greater than 5% Enrichment," August 28, 2007. [ML072340091]

APPENDIX G

GENERIC ISSUES PROGRAM CURRENT AND HISTORICAL PROCEDURES

I. BACKGROUND

History

On October 8, 1976, the Commission directed the staff to develop "a program plan for resolution of generic issues and completion of technical projects." The Commission further requested that "this plan should include: task schedules ... task priority and manpower requirements (with proportions of staff contract efforts explicitly identified)." On December 12, 1977, the Energy Reorganization Act of 1974 was amended by Congress through Public Law 95-209 to include, among other things, a new Section 210 as follows:

UNRESOLVED SAFETY ISSUES PLAN

> Sec. 210. The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report of the Commission thereafter.

In order to meet both Commission and Congressional directives, the staff developed a generic issues program that provided for the identification of generic issues, the assignment of priorities, the development of detailed action plans, projections of dollar and manpower costs, continuous high level management oversight of progress, and public dissemination of information related to the issues as they progressed. This program was published in NUREG-0410[387] in January 1978 and, shortly thereafter, the Commission issued a Policy Statement[1190] on the NRC "Program for Resolution of Generic Issues Related to Nuclear Power Plants."

The NRC generic issues program published in NUREG-0410[387] was considerably broader than the "Unresolved Safety Issues Plan" required by Section 210. It included plans for the resolution of generic environmental issues, for the development of improvements in the reactor licensing process, and for consideration of less conservative design criteria or operating limitations in areas where existing requirements might be unnecessarily restrictive or costly.

The first attempts by the staff to implement the generic issues program stated in NUREG-0410[387] were based largely on engineering judgments. This qualitative effort to rank unresolved generic issues continued through two phases:

(1) In 1977, all issues were classified into four categories according to importance, from "significant" to "little or no importance."

(2) In the early part of 1978, the issues were reclassified into Groups 1 through 8 by type rather than by order of importance.

Later in 1978, the staff began to take a quantitative approach by using risk assessment to place the issues into four categories ranging from I (potential high risk items) to IV (items not directly relevant to risk).[140] With increased confidence in this risk assessment approach, the staff introduced a more comprehensive quantitative system in early 1979. Points were assigned to each issue based on an assessment of safety significance, environmental significance, licensing effectiveness, deadline pressure, and retrofit versus forward-fit. Although the point system was still quite subjective, it was nevertheless a major improvement over the previous methods used.

In the aftermath of the Three Mile Island Unit 2 (TMI-2) accident, many new generic issues were raised and the staff came to the conclusion that the point system was too subjective to be used for ranking the issues. One of the TMI Action Plan[48] items, IV.E.2, called for the staff to develop a plan for the early resolution of safety issues. It was in resolving this issue that the staff developed a quantitative "prioritization" methodology whereby a numerical priority score could be assigned to each generic safety issue (GSI). With this approach, priorities were to be based on an evaluation of the estimated risk reduction associated with the potential change in requirements that could result from resolution of an issue, and the estimated costs to the NRC and the industry in implementing such a change. This methodology was submitted to the Commission for information in SECY-81-513.[1] In April 1983, this approach was refined and resubmitted to the Commission for approval in SECY-83-221.[1188] After Commission review, approval to use the methodology was given in December 1983.[1189]

In April 1993, after approximately ten years of experience with the methodology, adjustments were made in the numerical thresholds, while the basic features of the method were retained. These adjustments involved raising risk thresholds and simplifying the way in which costs entered the priority rankings. What motivated the raising of risk thresholds was the observation[1479] that, of the issues resolved, only 3 of the 27 MEDIUM-priority and about half of the HIGH-priority issues resulted in decisions to take regulatory action, i.e., in retrospect, it appeared that resources had been devoted to resolving a large number of issues with no resulting safety improvement. This outcome must be interpreted with the qualification that generic issue resolution efforts that have not led to regulatory action have, nevertheless, in many instances, produced safety benefits through licensee actions taken voluntarily, in consideration of the issues raised, or in response to interim guidance. However, the extent of these benefits, when they occurred, was generally in proportion to the priority rank and MEDIUM-priority issues usually resulted in marginal improvements. The proposed revisions were submitted to the Commission in SECY-93-108[1479]; in July 1993, Commission approval was obtained.[1505]

The threshold adjustments were intended to cause the prioritization process to model the resolution process without the earlier, apparently excessive margin for initial uncertainties, to reduce resolution efforts that do not produce safety improvements, while still ensuring attention to issues that require it. The raising of the numerical safety thresholds was accompanied by strengthened attention to uncertainties and special considerations, to help recognize instances when a priority rank higher than the indication from the numerical formula was warranted, the objective being to improve the efficiency of the prioritizations without impairing their prudence.

The priority ranking chart and risk thresholds used in prioritization analyses completed before July 24, 1993, are shown in Appendix C.

The simplification of the way in which costs were considered reflected the confirmation from experience that risk significance was indeed the primary factor in priority ranking, with a more bounded role for safety-cost trade-offs.

Operating Plan

The initial work in prioritizing issues was essentially done by various Staff Working Groups. Following a reorganization of the Office of Nuclear Reactor Regulation (NRR) in April 1980, the lead responsibility for prioritization was assigned to the Safety Program Evaluation Branch, Division of Safety Technology, Office of Nuclear Reactor Regulation (SPEB/DST/NRR).

The 1983 NRC Policy and Planning Guidance (NUREG-0885, Issue 2),[210] in addressing the area of Coordinating Regulatory Requirements (Planning Guidance, Item 5, p.6) called for "...a priority list of generic safety issues including TMI-related issues based on the potential safety significance and cost of implementation of each issue..." to be submitted to the Commission for approval. Using the prioritization methodology outlined below, this list was developed by SPEB in response to the Planning Guidance and forwarded to the Commission in SECY-83-221.[1188]

After another NRR reorganization in November 1985, the task of preparing and maintaining the list of GSIs and their priority was assigned to the Safety Program Evaluation Branch, Division of Safety Review and Oversight (SPEB/DSRO/NRR). Following an NRC reorganization in April 1987, this responsibility was assigned to the Advanced Reactors and Generic Issues Branch, Division of Regulatory Applications, Office of Nuclear Regulatory Research (ARGIB/DRA/RES). In July 1991, this responsibility was transferred to the Division of Safety Issue Resolution (DSIR) in RES. With the elimination of DSIR in December 1994, this function was transferred to the Generic Safety Issues Branch (GSIB), Division of Engineering Technology (DET), RES.

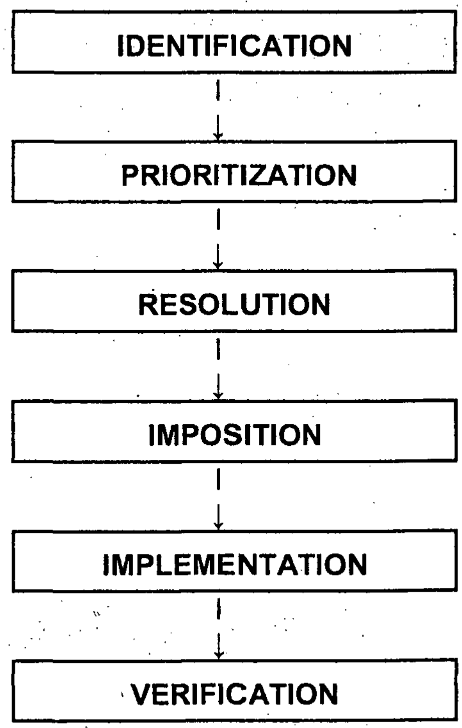
The prioritization of GSIs was an ongoing staff function that was reflected annually in the NRC Policy and Planning Guidance.[210] This document was superseded in 1987 by the NRC Five-Year Plan.

II. GENERIC ISSUES PROGRAM (1983-1999)

After issuance of the Policy Statement[1190] in 1978, the NRC program to resolve generic issues underwent many reviews and changes. As a result, the Commission concluded in April 1989 that the 1978 Policy Statement no longer reflected the NRC's generic issues program and withdrew it from the public record.[1191] From 1983 to 1999, the generic issues program consisted of six separate and distinct steps: identification, prioritization, resolution, imposition, implementation, and verification (See Exhibit A). An explanation of each of these six steps is given below. During this period, approximately 836 generic issues were processed in accordance with the steps outlined below. Beginning in 1999, all new generic issues identified were subjected to the process delineated in NRC Management Directive 6.4, "Generic Issues Program" (MD 6.4).[1858]

Exhibit A

GENERIC ISSUES PROGRAM (1983 - 1999)



Identification

Generic concerns may be identified by individuals or organizations within the NRC staff or by the Advisory Committee on Reactor Safeguards (ACRS), the nuclear power industry, or the public. MD 6.4[1858] and RES Office Letter No. 7 (OL #7)[1192] provide the procedures and suggested content for individuals or organizational units within the NRC to request consideration of a concern as a new generic issue. These procedures may also be used by parties outside the NRC to express their concerns to the staff for consideration as potential generic issues. Sources of potential generic issues are many and varied and include, but are not limited to, the following: evaluation of safety-related research, risk assessment analyses, and public and industry concerns. This step was retained as Stage 1 in MD 6.4.[1858]

Prioritization

This report focuses on the prioritization step of the generic issues program which is explained in detail in Paragraph III below. This step was replaced by Initial Screening (Stage 2) in MD 6.4.[1858]

Resolution

After an issue was prioritized and approved for resolution, the first task was the development of a plan to delineate the work to be done, assignment of major responsibilities, identification of project resource needs, and scheduling of milestone dates. These activities varied in scope and depth in accordance with issue priority and the depth of information on a given issue. The second task involved development of a technical solution. Typically, the information used to resolve an issue came from experience data, experiments, tests, analyses, and probabilistic risk assessments (PRAs). The results of such work or the technical findings may have been published in contractor and staff NUREG reports which were made available through the NRC Public Document Room (PDR), Washington, D.C., or the National Technical Information Service, Department of Commerce, Springfield, Virginia.

In the final stage of resolution, the technical findings were used as a basis to develop a proposed resolution for the issue involving a change to NRC requirements or guidance. Several alternatives were considered. A regulatory analysis, including a detailed cost/benefit analysis of each practical alternative, and consideration of the best methods of imposition, implementation, and verification were used in selecting a proposed resolution. If a backfit was proposed, first, a determination was made as to whether the backfit was required to provide adequate protection to the health and safety of the public, or simply provided for enhancement of public health and safety. If it was determined that the backfit was necessary to provide an adequate level of protection, the backfit was imposed, regardless of the costs to achieve it. If it was determined that the backfit provided for enhancement of public health and safety, a generic analysis was required that treated the nine factors specified in 10 CFR 50.109(c).

Once the cognizant NRC Office Directors agreed to a proposed resolution, it was then forwarded to the Committee for the Review of Generic Requirements (CRGR), the ACRS, the Executive Director for Operations (EDO), and the Commission for review and approval as appropriate. Changes to regulations, Policies, the Standard Review Plan (SRP), and Regulatory Guides were published in the Federal Register for public comment. Comments received were then incorporated, as appropriate, with the final product published in the Federal Register. Resolution of a generic issue took from several months to a few years, depending on the length of time required by the deliberations involved at each of the above steps.

OL #7[1338] described the procedure to be followed in the resolution of a generic issue, denoted the required elements of the resolution plan and resolution package, and identified review procedures and organizational responsibilities for the approval of the resolution of a generic issue. Prior to June 2, 1994, this procedure was issued separately in RES Office Letter No. 3 (OL #3)[1194]; however, OL #3 became obsolete[1339] when it was merged with OL #1.[1192] Milestone information and reporting requirements as well as organizational responsibilities for the tracking of generic issue resolution were also required by OL #7. Prior to June 16, 1996, these functions were outlined in RES OL #1.[1192]All issues scheduled for resolution were tracked by the Generic Issue Management Control System (GIMCS) which was updated quarterly and placed in the PDR. Guidance for the preparation, review, and required content of the regulatory analysis portion of the resolution packages was provided in RES Office Letter No. 3C.[1690] Prior to February 23, 1996, these procedures were outlined in RES Office Letter No. 2.[1193] This step was replaced by Technical Assessment (Stage 3) and Regulation and Guidance Development (Stage 4) in MD 6.4.[1858]

Imposition

Imposition was the step in the generic issues program where each affected licensee and/or applicant was required or guided to prepare a schedule for implementing the generic issue resolution consistent with a Rule, Policy, Regulatory Guide, generic letter, bulletin, and/or licensing guidance developed during the resolution stage. Normally, NRC requirements, policies, and/or guidance did not provide for NRC consideration of a licensee's modifications prior to their implementation at an affected plant. This facilitated completion of plant modifications to enhance safety within two refueling outages, not to exceed three years after issuance of NRC requirements, policies, and/or guidance. However, in a few exceptional cases, licensees were expected to submit (normally for NRC approval) their plans (including schedules) for plant modifications prior to their implementation. In all cases, licensees were expected to certify in writing to the NRC that plant modifications had been completed.

For the exceptional cases, the staff reviewed each applicant's and/or licensee's submittal with regard to proposed modifications to site, equipment, structures, procedures, technical specifications, operating instructions, etc., and schedules proposed for the accomplishment of the modifications. For backfits, imposition was complete when each affected licensee had committed to compliance actions and schedules for implementing these actions. For forward-fits, the imposition of a generic issue resolution was complete when the new requirement or guidance became effective as an integral part of NRC regulations, policies, and/or guidance.

During this stage, a resolved GSI was identified as a Multiplant Action (MPA) for licensee action. The imposition status of all MPAs was tracked in the Safety Issue Management System (SIMS). This step was replaced by Regulation and Guidance Issuance (Stage 5) in MD 6.4.[1858]

Implementation

Implementation is the step in the generic issues program where the affected licensees perform the actions on existing plants to satisfy the commitments made during the imposition stage. These may include modifications/additions to equipment, structures, procedures, technical specifications, operating instructions, etc. No later than 30 days after each affected licensee has completed all of the actions required for a particular generic issue resolution, and the modified/additional system is fully operational, the licensee is required to certify in writing to the NRC that plant modifications have been completed in accordance with NRC requirements, policies, and/or guidance. When all affected licensees have officially notified the NRC of completion of all required/committed actions, the implementation stage is complete, unless it is determined by the staff from subsequent verification inspection that additional licensee actions are needed for compliance. This step was retained as Implementation (Stage 6) in MD 6.4.[1858]

Verification

The verification step consists of three parts. First, the portions of a licensee's actions, if any, that warrant NRC inspection must be determined. This decision is made during the resolution stage based on the judgment of the safety significance of the issue relative to other matters in the inspection program, licensee performance, and the resources needed to accomplish a meaningful inspection. Next, as necessary, inspection instructions are prepared to ensure that the NRC inspection is performed in a consistent and appropriate manner at all affected plants; the inspection, by its very nature, is an audit. Therefore, carefully thought-out instructions must be provided to the NRC inspectors so that the maximum safety benefit is achieved for the limited resources devoted to this effort. The third part of the verification process is the actual verification and documentation of the results in an inspection report. Physical inspections are performed on an audit basis in a manner consistent with general inspection procedures which involve a sampling of changes made by licensees or applicants, as opposed to a 100% inspection of all actions. Verification of licensee implementation of generic issue resolution was required to be reported by the staff in SIMS. This step was retained as Verification (Stage 7) in MD 6.4.[1858]

III. PRIORITIZATION (1983-1999)

Purpose and Scope

The primary purpose of prioritization was to assist in the timely and efficient allocation of resources to those safety issues that had a high potential for reducing risk ,and in decisions to remove from further consideration issues that had little safety significance and held little promise of worthwhile safety enhancement. However, issues of such gravity that consideration of immediate action was called for were excluded from prioritization because of the compressed time scale in which decisions for such issues had to be made. Generally, immediate action took the form of a Bulletin or Order. Both operating and future plants were considered in the priority ranking process.

Prioritization focused on generic safety issues (GSIs) i.e., safety concerns that may affect the design, construction, or operation of all, several, or a class of nuclear power plants and may have the potential for safety improvements and promulgation of new or revised requirements or guidance. However, the method was used to identify changes in existing requirements that may have significantly reduced the impact (usually cost) on licensees without any substantial change in public risk. Issues of this type were classified as Regulatory Impact issues (RI) to clearly differentiate them as not improving the safety of nuclear power plants but, nevertheless, possibly worthwhile.

In order to identify GSIs, all issues originated in accordance with OL #1[1192] were reviewed to determine their safety significance. Issues that primarily concerned environmental protection or the licensing process and did not involve significant safety improvement elements were classified accordingly and noted for separate consideration outside the GSI priority ranking process. These issues were classified as either environmental issues or licensing issues. Environmental issues (EI) involved impacts on the human environment and the values sought to be protected by the National Environmental Policy Act (NEPA). Licensing issues (LI) were not directly related to protecting public health and safety or the environment, but related to: (1) increasing the staff's knowledge, certainty, and understanding of safety issues in order to increase its confidence in assessing levels of safety; (2) improving or maintaining the NRC capability to make independent assessments of safety; (3) establishing, revising, and carrying out programs to identify and resolve GSIs; (4) documenting, clarifying, or correcting existing requirements and guidance; and (5) improving the effectiveness or efficiency of the review of applications.

The list of issues subjected to prioritization contained the following groups:

(1) TMI Action Plan items identified for development in NUREG-0660[48]; these issues are covered in Section 1. The priority recommendations in this report excluded those issues that were designated for implementation in NUREG-0737.[98]

(2) Task Action Plan items identified in NUREG-0371[2] and NUREG-0471,[3] plus the subsequently added issues A-42 through A-49 that were designated as Unresolved Safety Issues (USIs); these issues are covered in Section 2. However, issues designated as USIs were excluded from prioritization because of the high-priority attention they were given based on priority decisions previously made.

(3) New Generic issues identified by the staff, ACRS, or others; these issues are covered in Section 3. All new issues identified are included in Section 3 and published in supplements to this report.

(4) Human Factors Program Plan (HFPP) items identified for development in NUREG-0985[603]; these items are covered in Section 4.

(5) Chernobyl Issues identified in NUREG-1251[1195]; these issues are covered in Section 5.

A comprehensive listing of all issues in the above five groups is given in Table II which includes the following information for each issue: (1) the NRC person responsible for the prioritization evaluation; (2) the lead NRC office, division, and branch responsible for reviewing the prioritization analysis and/or resolving the issue; (3) the priority ranking or status; (4) the latest version of the evaluation; (5) the issuance date of the latest version of the evaluation; and (6) the MPA number for those issues that have been resolved and require licensee actions. A summary of the number of issues in each category is shown in Table III. A cross-reference listing of reports prepared by the Office for Analysis and Evaluation of Operational Data (AEOD) and their corresponding generic issues is provided in Table IV.

How the Work Was Done

The work was done, in accordance with the criteria described below, by the responsible NRC Branch in consultation with others in the NRC with knowledge of the issues or expertise in the technical disciplines involved. In a number of instances, technical or cost information was obtained from industry and other outside sources. The Battelle Pacific Northwest Laboratories (PNL), under a technical assistance contract, developed detailed methods to quantify safety benefits and costs and provided safety-benefit analyses and cost information for many of the issues. The responsible NRC Branch, with internal consultations as necessary, reviewed and applied the PNL-supplied technical factors, in conjunction with additional factors, in developing the priority rankings and recommendations.

Systematic peer review of each prioritization evaluation within the NRC contributed to the assurance that the analysis was complete and accurate, and that the judgments were soundly based. This review was done in two stages. First, each analysis was reviewed by the NRC organizational unit or units whose area of responsibility or specialized knowledge was substantially involved. Second, any comments made were then resolved, where practical, and factored into the analysis, as appropriate. Upon completion of peer review, the analysis was then finalized and prepared for approval by the responsible Office Director. Once approved, it was placed in the PDR and published in a supplement to this report, after which, additional comments from the ACRS, the industry, and the public were considered in any further reassessment of the issue's priority.

Priority Categories: Their Meaning and Proposed Use

Four priority rankings were used: HIGH, MEDIUM, LOW, and DROP. They were intended for use in guiding allocation of NRC resources and scheduling of efforts to resolve the various issues, in conjunction with other pertinent factors such as: (1) the nature, extent, and availability of manpower and material resources estimated to be required; (2) length of time needed to resolve; (3) conflicts in resource allocation and scheduling among items of comparable priority; (4) status of affected reactors; and (5) budget constraints.

A HIGH priority ranking meant that strong efforts to achieve the earliest practical resolution were appropriate. This was because: (a) an important safety concern may have been involved (though generally the concern was not severe enough to require prompt plant shutdown); or (b) the uncertainty of the safety assessment was unusually large and an upper-bound risk assessment would have indicated an important safety concern. All unresolved HIGH priority issues were periodically reviewed in accordance with the criteria stated in NUREG-0705[44] for possible designation as USIs. A USI is defined as a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements for which a final resolution has not yet been developed and that involves conditions not likely to be acceptable over the lifetime of the plants affected.[186] In accordance with Section 210 of the Energy Reorganization Act of 1974, progress on the resolution of USIs was reported to Congress in each NRC Annual Report. However, with the passage of the Federal Reports Elimination and Sunset Act of 1995, the statutory requirement to send Congress an NRC Annual Report ended on December 21, 1999. In accordance with SECY-00-0038,[1859] the last annual report to the Congress that included unresolved GSIs was the 1999 edition.

A MEDIUM priority ranking meant that no safety concern demanding high-priority attention was involved, but there was believed to be potential for safety improvements or reductions in uncertainty of analysis that may have been substantial and worthwhile. Efforts at resolution were planned over the ensuing years, but on a basis that did not interfere with pursuit of HIGH-priority GSIs or other high-priority work.

A LOW priority ranking meant that no safety concerns demanding at least MEDIUM-priority attention were involved, and there was little or no prospect of safety improvements that were both substantial and worthwhile. When the prioritization process resulted in a LOW priority ranking for an issue, approval of this ranking by the responsible Office Director signified that the issue had been eliminated from further pursuit. However, in accordance with Staff Requirements Memorandum (SRM) 871021A,[1493] the staff conducted a periodic review of existing LOW-priority GSIs to determine whether there was any new information that would necessitate reassessment of the original prioritization evaluations.

The DROP category covered proposed issues that were without merit, or whose significance was clearly negligible. Issues were also DROPPED from further consideration if it was determined that their safety concerns had been addressed in previously prioritized or resolved issues. When the prioritization process resulted in a DROP priority ranking for an issue, approval of this ranking by the responsible Office Director signified that the issue had been eliminated from further pursuit.

An issue was considered RESOLVED, indicated by NOTE 3 in Table II, when its resolution resulted in either: (a) the establishment of regulatory requirements or guidance (by Rule, SRP[11] change, or equivalent); or (b) a documented authoritative decision that no change in requirements was warranted. Priority rankings were not assigned to issues that had been resolved. However, in those cases where issues were resolved after having been identified for further pursuit by the prioritization process, the related calculations were retained in the text of this document for future use.

Priority rankings were not assigned to issues that were nearly-resolved (denoted by NOTES 1 and 2 in Table II) because approval of changes to requirements, based on the resolution of an issue, required that a detailed value/impact evaluation of the safety benefit, implementation costs, and other relevant factors be made. Prioritization would have duplicated this value/impact analysis, but in a less comprehensive manner. Therefore, the effort that would have been needed to prioritize an issue was devoted to completing the final evaluation of the issue, rather than making a tentative judgment as to the importance and value of the issue. Possible resolution of an issue was considered to be identified, indicated by NOTE 1 in Table II, when a possible technical resolution was under evaluation and the evaluation was nearing completion. Further work may have been required as part of the review and approval process before a change in requirements or guidance was issued. Resolution of an issue was considered available, indicated by NOTE 2 in Table II, when proposed or recommended changes to requirements or guidance were documented in a NUREG report, NRC memorandum, Safety Evaluation Report (SER), or equivalent.

Priority rankings were also not assigned to those issues whose safety concerns were determined to be covered (at the time of prioritization) in other issues of broader scope that were being prioritized, or were being resolved. Issues in this category were integrated into the issues of broader scope. A detailed listing of all such issues is given in Table V.

Criteria For Assigning Priorities

1. Basic Approach

The method of assigning priority rank involved two primary elements: (i) the estimated safety importance of the issue; and (ii) the estimated cost of developing and implementing a resolution. Special considerations may have influenced the proper use of the estimates. These elements were applied as follows:

(a) The issue was identified and defined. Since issues are often complex and interrelated with other issues, careful definition of an issue's scope and bounds was essential in arriving at a sound and applicable assessment.

(b) A quantitative estimate was made of the safety importance of the issue, measured in terms of the risk (the product of accident probabilities and radiological consequences) attributable to the issue, and the decrease in that risk that may have been attainable by resolving the issue.

(c) A quantitative estimate was made of the cost of resolution.

(d) A numerical impact/value ratio was calculated by dividing the estimated cost entailed by the estimated potential risk reduction. The ratio measured the safety value received in return for the cost impact incurred.

(e) A priority rank (HIGH, MEDIUM, LOW, or DROP) was obtained by application of criteria in which both the safety significance of the issue and the impact/value ratio were taken into account. The ratio was not always directly applied to determine the priority rankings. In some cases, the safety significance of the issue was so great that it demanded a HIGH priority, or so minor that only a LOW priority (or a decision to DROP) was warranted irrespective of the impact/value assessment.

(f) The priority ranking was reviewed and modified, if appropriate, in light of any special factors (discussed below) that: (i) might bring into question the applicability of the necessarily simplified calculation technique; and (ii) call for special consideration of NRC management decisions or large uncertainties in the quantitative estimates.

In summary, while the method had a quantitative emphasis, the calculated numerical values were used as an aid to judgment and not as determinative of the ranking results. The nature of the specific issue, the quality of the data base, and the scope of the necessarily limited analysis determined in each case the dependability of the numerical indications as a judgment aid.

2. <u>Safety Significance</u>

The safety significance of an issue was represented by the reduction in risk that resolution could affect. Risk was ordinarily expressed here in terms of the product of the frequency of an accident occurrence and the public dose (in person-rem) that would result in the event of the accident. If more than one accident scenario was important within the necessarily rough risk estimates, the risks were summed.

The potential risk reduction calculated in this way was used in calculating the impact/value ratio as part of the simplified impact/value analysis, discussed in

Paragraph III.3 below. It was also used directly as a measure of safety significance, as discussed in Paragraph III.4 below, in arriving at a priority rank that was influenced by the safety significance of an issue, as well as by the estimated value/impact relation of a projected solution, or was determined on the basis of safety significance alone.

The person-rem-based risk reduction estimate may not have been the only appropriate measure of an issue's safety significance in all cases. For example, when a possible core damage was involved but release outside containment would be minor or highly improbable, contribution to the core-damage probability may well have been more indicative of safety significance. Provision was made, as described in Paragraph III.4 below, for use of alternative measures of safety significance in determining a priority ranking when such alternative measures were useful.

3. Impact/Value Relation

(a) The Impact/Value Ratio Formula

To the extent reasonably possible, quantitative estimates were made of the possible solutions to a GSI by calculating an Impact/Value Ratio that reflected the relation between the risk reduction value expected to be achieved and the associated cost impact. The formula for the impact/value ratio (R) was:

$$R = \frac{\text{Cost}}{\text{Safety Benefit}}$$

where the safety benefit was the estimated risk reduction (event frequency x public dose averted) that may have been achieved, and the cost was that thought necessary to develop and implement a resolution in the number of plants involved. The scoring computation for any issue was then:

$$R = \frac{C}{NFTD}$$

where,

where,	N	=	number of reactors involved
	T	=	average remaining life (years) of the affected plants, based on an original license period of 40 years
	F	=	the accident frequency reduction (event/reactor-year)
	D	=	public dose from the radioactive material released from containment (person-rem)
	C	=	total cost of developing and implementing the resolution of the issue for all plants affected (dollars).

The total cost (C) included both the cost of developing the generic solution, typically NRC cost, and the cost of implementing the possible solution at all affected plants, typically industry cost, including design, equipment, installation, test, operation, and maintenance. The priority ratio (R) had the units of dollars per person-rem.

Simplified calculations usually sufficed, since only an approximate impact/value ratio was required. Reference was made to the current version of the Value-Impact Handbook,[970] where necessary, to supplement the general guidelines provided below.

(b) Rationale for the Formula

The qualitative diversity of factors entering impact/value analyses in support of GSI prioritization, together with inevitable quantitative uncertainties, made any of various possible impact/value score formulas necessarily imperfect. Accordingly, provisions were made to compensate for those imperfections to the extent practical (as discussed in Paragraph III.5 below).

The formula selected measured a total-cost/total-safety-benefit relation. As discussed herein, it was applied within limits set by other possible considerations where a safety issue was either too important to depend on safety-cost tradeoffs, or too trivial to merit attention at all. Two principal arguments favored a formula of this type:

(1) The denominator was designed as a direct measure of the safety values that it is NRC's primary mission to protect. The numerator was designed to measure the overall cost impact, including industry as well as NRC costs, and should thus reflected the entire public interest in economy. The resulting impact/value ratio, subject to the stated caveats, should have reasonably approximated measuring the overall public interest in safety value received for total resources expended.

(2) The allocation of national resources, which in most cases were primarily industry resources, was optimized.

(c) <u>Risk Estimates</u>

The risk estimates developed for GSIs were useful as rough approximations for comparative purposes, but were not necessarily applicable to the assessment of absolute levels of risk attributable to particular issues. Similarly, the impact/value ratios provide, for the limited purpose of prioritization, tentative assessments of relative potential for cost-effective resolution. They were not intended to be applied as impact/value determinations for any regulatory proposal that may ultimately result from efforts to resolve an issue. In addition, the assumed resolutions were not intended to prejudge the final resolutions, but are only assumptions that are necessary to perform quantitative analyses.

The basis of frequency estimates generally involved the following:

(1) Identification of the specific events which were the basis for the concern, for which the consequences were to be established, and which were to be eliminated or ameliorated by a proposed technical solution

(2) Use of event sequence diagrams, fault trees, or decision trees, if possible

(3) Identified references and calculations, or stated assumptions for the numbers used

(4) Consideration of the probability of common mode as well as random independent failures.

Where possible, numerical estimates were based on operating experience, usually Licensee Event Reports (LERs). Other sources included prior PRAs and other risk and reliability studies. Some numbers were based on engineering judgment; in such cases, the basis for that judgment was stated.

For the identified end event(s), the expected radiological consequences were expressed in person-rem generally based on the radioactive release categories described in WASH-1400[16] (Appendix VI, pp. 2-1 to 2-5), reproduced as Appendix A to this report. Exhibit B gives estimated Curies released and approximate population doses for each release category. The computer program CRAC2, applied to a typical midwest site (Braidwood) meteorology, was used for the dose calculations. However, the calculated doses were adjusted to reflect the mean of the population density within a 50-mile radius of U.S. nuclear power plants.[64] Assumptions and parameters used for the calculations at this stage (Step (b) described under "Basic Approach") were as follows:

-Consequences were represented by the whole body population dose (person-rem) received within 50 miles of the site.

-An exclusion area of 1/2 mile was assumed with a uniform population density of 340 persons per square mile beyond 1/2 mile. This was the mean 50-mile radius population density projected for the year 2000 (NUREG-0348, p.T52).[70]

-Evacuation of people was not considered because of the possible large variations in evacuation capability for each plant site.

-All exposure pathways were included in the basis of the tabulated numbers except ingestion pathways, i.e., interdiction of contaminated foods was assumed. (Farmland usage parameters for the State of Illinois were used for separate ingestion pathway calculations where made.)

-Meteorological data was taken from the U.S. National Weather Service station at Moline, Illinois.

The person-rem factors for each release category are given in Exhibit B. Although generally used, consequence estimates were not solely based on these factors. Other factors were used in some cases when more appropriate.

An estimated occupational dose of 20,000 person-rem from postaccident cleanup, repair, and refurbishment was also considered.

Where significant occupational radiological exposure (ORE) was incurred or averted in implementing current requirements or the proposed resolution of a GSI, such exposure was taken into account but stated separately.

Exhibit B

Release Category	Release (Curies)	Estimated Public Dose** (Person-rem)
PWR-1	1.2×10^9	5,400,000
PWR-2	9.3×10^8	4,800,000
PWR-3	5.2×10^8	5,400,000
PWR-4	2.8×10^8	2,700,000
PWR-5	1.3×10^8	1,000,000
PWR-6	1.0×10^8	150,000
PWR-7	2.1×10^6	2,300
PWR-8*	7.7×10^5	75,000
PWR-9*	1.1×10^3	120
BWR-1	1.1×10^9	5,400,000
BWR-2	1.1×10^9	7,100,000
BWR-3	5.0×10^8	5,100,000
BWR-4	2.1×10^8	610,000
BWR-5*	1.7×10^5	20

* Non-core-melt (Other release categories involve core-melt).

** The Release value (Curies) and Estimated Public Dose (Person-rem) will be updated in the future to be consistent with the ongoing evaluation to revise the Source Term following a postulated severe accident.

Where more direct issue-specific ORE information was lacking, dose estimates were obtained by assuming an average dose rate of 2.5 millirem/hour (based on the PNL analysis[64] cited above) and multiplying by the estimated number of man-hours involved.

A second factor was that the risk associated with an issue was more likely to be overestimated than underestimated. Where risk estimates were widely uncertain, a reasonably conservative value of risk reduction was generally selected to help assure adequate priority to issues that may have warranted attention.

The sum of the estimated risks of all the separate issues were likely to exceed the existing estimate of the total risk of nuclear power plants because of two factors. First, individual accident sequences could have been affected by more than one issue. The resolution of one issue would have reduced the probability or consequences of a certain set of accident sequences. Some or even all of these sequences could have been the same as some or even all of the sequences affected by another issue.

However, issues were assessed independently, and this interaction of their risk significance was not ordinarily considered. This interaction was strongest for issues related to human factors, since human error affected almost all sequences. The sum of the reductions in core-melt frequency estimated for all of the human factors-related issues may have been as much as twice as great as the total human factors contribution to total risk. However, most of the issues not related to human factors were much less strongly interrelated.

(d) Cost Estimates

Because cost estimates were used here only in relation to risk estimates which were generally subject to more or less wide uncertainties, only approximate costs were needed.

No separate estimates were generally made for offsite property damage; reasonably conservative use of the public dose estimates was an adequate surrogate in this application. Furthermore, there was no readily-available data on offsite damage that was realistic and detailed enough to make estimates meaningful, reasonably accurate, and generically applicable. If unusual or special offsite effects were not adequately represented by the public dose in some issues, this fact was considered separately and explicitly in evaluating such issues.

The expected technical solution on which the cost estimate was based was identified. Estimated costs were established by collecting available data regarding engineering, procurement, installation, testing, and periodic inspection and maintenance. Where data were non-existent, estimates were based on judgments by the experts involved. Assumptions and estimated uncertainties were identified. Costs were estimated in 1982 dollars.

NRC costs included the following: (1) issue identification, analysis, resolution, and report issuance; (2) research to establish proposed specific changes to licensing requirements (or to determine that no change is required); (3) technical assistance contracts (including associated NRC effort); (4) discussions and correspondence with industry owners' groups; (5) plant reviews; and (6) preparation and review of SERs and requirement documents. The estimated cost of NRC professional time was based on $100,000 per person-year.

The costs to industry generally consisted of some combination of the following: (1) licensing; (2) design; (3) equipment procurement; (4) installation; (5) testing, inspection, monitoring, and periodic maintenance; and (6) plant downtime to effect a change, taken as the cost of replacement power at $300,000/day. Industry manpower costs were ordinarily taken as $100,000 per person-year.

Averted plant damage costs may have affected the priority of a GSI. Estimates for such averted costs were multiplied by the accident frequency and used as negative costs, i.e., subtracted from the (positive) costs of implementing the resolution of the issue.[1473] The averted costs may have included those of averted equipment failures, limited-time plant outage, or limited plant-contamination cleanup. In the extreme, they also included averted permanent loss of the plant, estimated at approximately $2 billion present worth. This estimate for a "generic" plant included the costs of both plant-wide cleanup and permanent loss of use of the plant, discounted to present worth based on a 7% real discount rate. This figure was multiplied in each case by the reduction in frequency of such events that would be brought about by resolution of the GSI. The plant loss estimate included allowance for typical plant age at the time of the accident, as well as replacement power costs together with apportioned cost of a replacement plant. The plant-wide cleanup estimate reflected cleanup to the point at which the plant was ready for decommissioning or refurbishing for restart.[393] Refurbishing costs, when restart was more economical than decommissioning, depended on the nature of the accident and ranged from a fraction of the total plant loss figure to a cost approaching that figure.

Some fixed costs were one-time, initial costs; others may have occurred at future times. Future costs were discounted to present worth at a 7% rate. Where costs were continuous or periodically recurring throughout a plant's remaining life, the periodic cost was taken into account using an approximation of the present worth of the continuing (or repetitive) costs for plants with remaining operating lives of 20 years or longer.

(e) <u>Uncertainty Bounds</u>

Major sources of uncertainty in the priority score were identified and judgments as to their quantitative significance were indicated as information warrants. Where data warranted, the method described in NUREG/CR-2800,[64] Section 5, for the general case of combining uncertainties for random variables with unknown distributions (as well as some special cases) were used. [See also Paragraph III.5(a)]. Most often, however, a rigorous uncertainty analysis was not warranted. In most cases, the uncertainty in the point estimates of risks and costs was known to be large. However, sufficient information was not usually available to make a meaningful quantitative analysis of the uncertainty bounds of these point estimates. Decisions were tempered by the knowledge that the uncertainty is generally large. This knowledge was also used in developing the chart of tentative priority rankings (Figure 1). The wide spread between a level of risk, for example, at which an issue would be ranked as having a high priority and the level at which an issue would be ranked as low priority (a factor of 100) was partially based on the recognition that the uncertainties are large. In cases where uncertainty had a special character or importance, this was discussed and considered in the conclusion of the analysis of the GSI.

4. <u>Priority Ranking</u>

(a) Priority Ranking Chart

A chart showing how the tentative priority rankings were derived from the safety significance of an issue and its impact/value ratio is presented in Figure 1. The thresholds on the chart are discussed in Paragraphs III.4(b) and III.4(c) below. A conversion factor of $1,000/person-rem was used until September 18, 1995, when an increase to $2,000/person-rem was approved by the Commission.[1689]

(b) Preliminary Screening for Safety Significance

The determination of a priority rank started with a triage based on safety significance, i.e., the incremental risk associated with the issue. For a reduction in core damage frequency (ΔCDF) greater than 10^{-4} per reactor-year (RY), a HIGH priority was assigned on the basis of safety importance alone, regardless of other considerations, such as an initially estimated high cost, which might result in a low priority score.

At the other extreme, an issue's safety significance could have been too minor to warrant diversion of attention from more important safety issues, even if it had a low impact/value ratio because an inexpensive solution was believed to be available. Below a minimal safety significance threshold, the priority was always DROP; where the potential risk reduction was trivial, there was no basis for regulatory action on safety grounds.

In between, there may have been issues of less extreme importance or unimportance, for which a HIGH, MEDIUM, LOW, or DROP priority may have been appropriate, based on consideration of the impact/value relation as well as safety significance. As indicated in Figure 1, a HIGH priority was assigned to an issue exclusively on the basis of a high safety significance; the threshold shown on the chart is $\Delta CDF=10^{-4}$/RY. For an issue with a safety significance lower than the threshold for an always-HIGH priority but at least 10% of that threshold ($\Delta CDF=10^{-5}$ /RY), the chart indicates a HIGH or MEDIUM priority based on cost trade-offs. At the low-risk end of the abscissa, the priority rank indicated was always DROP for $\Delta CDF<10^{-7}$/RY. Cost trade-offs entered in the 10^{-7} to 10^{-4}/RY ΔCDF range, as discussed in Section 4(c) below.

The abscissa in Figure 1 provides a measure of an issue's estimated safety significance in terms of the change (Δ) in CDF attributable to resolution of the issue. This was often the most useful safety significance measure in GSI prioritization, though for some issues other measures may have been required or appropriate. For example, a measure based on radiological consequences (probability-averaged over the remaining reactor life) was used when the issue under consideration involved containment bypass, or related to containment performance or other features or actions to mitigate the radiological consequences of a core damage. Also, the thresholds may have needed to accommodate the possible influence of the number of reactors affected on the appropriate priority ranking. Therefore, Figure 1 was repeated in Figure 2, with auxiliary abscissae providing additional measures of safety significance. These were used when the principal abscissa was inapplicable, or when an auxiliary abscissa led to a higher priority indication. Thus, the abscissae for total effect on all plants were considered when more than 30 plants were affected.

(c) Impact/Value Ratio Thresholds

When the safety significance was in the intermediate range discussed above, i.e., •CDF between 10^{-7} and 10^{-4}/RY, or between 0.1% and 100% of the threshold for an always-HIGH priority, the impact/value ratio (R) was taken into account in the ranking indicated by the chart (Figure 1). This was done as follows:

(1) In the range of 10% to 100% of the threshold for an always-HIGH priority, the indicated priority was HIGH if R was below $2,000/person-rem; otherwise, the indicated priority was MEDIUM.

(2) In the range of 1% to 10% of the threshold for an always-HIGH priority, the indicated priority was MEDIUM if R was below $2,000/person-rem; otherwise, the indicated priority was LOW.

(3) In the range of 0.1% to 1% of the always-HIGH threshold, the indicated priority was LOW or DROP, depending on whether R was below or above $2,000/person-rem.

5. Other Considerations

The formula-based rankings represented the primary concern of the NRC: public safety. The secondary concern was the impact on licensees, evaluated in terms of cost. However, the tentative priority rankings were subject to the limitations of an often incomplete and imprecise data base, and to possible distortions due to the nature of the necessarily highly simplified quantitative formula underlying them. Special situations with respect to some issues may have caused added difficulty in priority assignment. While the formula-based tentative rankings generally indicated that the safety significance was sufficient to justify NRC action, other considerations not adequately reflected, or not reflected at all, in the numerical formula were often needed to corroborate or adjust the results. Decision-making was helped by explicit identification of such other considerations and explanation of their bearing on the resulting final priority ranking, whether the effect was one of corroborating or of changing the estimates.

Listed below are some factors that may have been important in arriving at a sound priority ranking, and may have led to adjustment of a tentative, formula-derived ranking. Possible effects of occupational doses and uncertainty bounds [1(a)(1), (a)(2), and (b)(1) below] required particularly careful consideration for all issues. The factors listed were not considered all-inclusive. Others thought significant were discussed and, when practical, quantified appropriately in the overall risk significance measure and impact/value ratio along with their associated uncertainties. Sometimes, there were special considerations that were quite specific to an issue or some aspect of it. However, it should be noted that, in determining an issue's priority, those factors that related to safety were given the most consideration. The following is a partial list of other factors considered:

(a) Special risk and cost aspects not included in or potentially masked by the numerical formulas:

   (1) The additional risk associated with a license renewal period of 20 years for the affected plants. GSIs prioritized and resolved up to March 31, 1994, were evaluated for license renewal implications; these evaluations were documented in NUREG/CR-5382[1563] and an RES report.[1564] All other GSIs prioritized and resolved after March 31, 1994, were required to consider the impact of license renewal.

   (2) The net change in occupational doses entailed by implementing the current versus the proposed requirements.

   (3) Any significant non-radiation-related occupational risk affected by the proposed resolutions.

   (4) Loss or severe degradation of a layer in the defense-in-depth concept (e.g., one mode of core cooling or containment cooling)

   (5) Issues for which solutions of widely differing costs may be applicable to different classes of plants, or various plants are otherwise affected in vastly different ways.

(b) Factors related to uncertainties stemming from an incomplete or imprecise data base for the priority formula:

   (1) Uncertainty bounds, imbalance in uncertainty factors, certainty of cost to fix versus uncertainty that safety is really improved and the true extent of such improvement.

   (2) Situations where uncertainty is extraordinarily large (in accident probability, consequences, or cost, or any or all of these). If there are large uncertainties in either the numerator or the denominator, the mean of the impact/value ratio (mean ratio) should be used with caution in assigning a priority ranking. The ratio of the means is a good approximation to the mean ratio provided only that the uncertainty in the denominator is small. However, if the uncertainty in the denominator is large, then the ratio of the means is a poor estimate of the mean ratio.

(3) Problems which are ill-defined and problems for which solutions are not evident so that at least the resources necessary to understand the problem are assigned.

(4) The potential for a proposed change to affect more than one accident or transient sequence, thus affecting risk to a greater or lesser degree than assessed in the description of the issue; notably, the potential for a new safety decrement, or increase in risk, due to unidentified effects of a proposed change, or added complexity, or for other reasons.

(5) Circumstances imparting unusual significance to accident consequences (such as ingestion pathway effects) or mitigating measures (such as evacuation) that are not directly included in the public dose calculations.

(6) Potential for human intervention, using available equipment.

(7) Acute knowledgeable professional controversy concerning the importance of an issue or modes of dealing with it.

(c) Change with passage of time:

(1) The effect of license renewal should be considered in every prioritization. The effect, if any, on the priority rank of an additional 20 years of operation should be separately stated.

(2) Potential substantial deterioration of the impact/value ratio while awaiting regulatory resolution (e.g., a potential design fix that is inexpensive to apply before construction, much more expensive after the plant is largely built, and extremely expensive and problematical to apply to an operating plant).

(3) The amount of resources already spent on an issue, and how close to completion it may be; the value of continuity in efforts to resolve an issue.

(4) The span of time predicted to resolve an issue and implement the resolution.

(5) The clarity of an "issue" and the objectivity with which it is currently defined. (Perhaps additional research effort is necessary to identify and define a specific risk reduction of interest.)

(6) Change of perceptions (of safety importance or impact/value relation or some special issue-peculiar factor) in the course of time.

Generally, in situations of large doubt or conflicting indications, the highest priority rank reasonably consistent with the nature of an issue was assigned.

Thus, where no solution was evident, assignment of a priority consistent with the safety significance of the issue may have led to a search for resolution or mitigation at an acceptable cost. Generally, when uncertainties narrowed or perceptions changed in the course of time, the priority rankings were reexamined in the light of new developments and retained or changed. When different classes of plants were expected to be very differently affected by a potential resolution, the priority assignment was governed by the class of plants for which resolution was most worthwhile and urgent. (Resolution in such cases could have involved a new requirement for some class of plants and no action for others.) Where resolution differed for different classes of plants, differing priorities were assigned.

6. Concluding Remarks

The criteria and estimating process on which the priority rankings were based were neither rigorous nor precise. Considerable application of professional judgment, sometimes guided by good information but often tenuously based, occurred at a number of stages in the process when numerical values were selected for use in the formula calculations, and when other considerations were taken into account in corroborating or changing a priority ranking. What was important in the process was that it was systematic, that it was guided by analyses that were as quantitative as the situation reasonably permitted, and that the bases and rationale were explicitly stated, providing a "visible" information base for decision. The impact of imprecision was blunted by the fact that only approximate rankings (in only four broad priority categories) were necessary and sought. Beginning in June 1999, the above method of prioritizing generic issues was replaced with the screening process of MD 6.4.[1858]

Results of Prioritazation

The results of the prioritization and resolution of all issues contained in this report are summarized and tabulated by group in Table III. In addition, a listing of those issues that affect operating and future plants is given in Appendix B. This appendix reflects the results of prioritization and resolution and only includes: (1) issues that have been resolved with new requirements [NOTE 3(a); (2) USI, HIGH-, and MEDIUM-priority issues that are being resolved; (3) nearly-resolved issues (NOTES 1 and 2); (4) issues whose impact is not yet known (NOTE 4); and (5) issues that were resolved without requirements for operating plants but with staff requirements for future plants under development. Tables II and III, and Appendix B also incorporate the results of those generic issues processed in accordance with MD 6.4[1858] since 1999.

IV. GENERIC ISSUES PROGRAM (1999-2007)

Introduction

The Generic Issues Program (GIP) provides internal guidance for determining whether a candidate generic issue (GI) represents an adequate protection issue, a substantial safety enhancement issue, or a reduction in unnecessary regulatory burden issue. In addition, the GIP identifies cost-effective solutions to GIs, implements, and then verifies the adequacy of solutions for GIs, as appropriate. Thus, the GIP provides an opportunity for the NRC and Agreement State staff and other parties to recommend safety or security-related (or reduction in unnecessary regulatory burden) improvements to the agency's regulations and/or implementation of these regulations. Candidate generic issues may arise from many sources, including safety evaluations, operational events, or even suggestions on the part of individual staff members, outside organizations, or members of the general public. Additionally, new issues identified as Unresolved Safety Issues (USIs) or any staff concerns that are raised as part of the NRC's Differing Professional Opinion (DPO) program may also be addressed under the GIP. The staff periodically conducts reviews of the open GIs to identify USIs. Detailed staff guidance is provided in Appendix B, "Unresolved Safety Issue Assessment Criteria," of MD 6.4. [1858]

Because of the varying technical disciplines and levels of difficulty encompassed by GIs, the processing of GIs does not lend itself to a strict, proceduralized process. The guidance in MD 6.4 [1858] is intended to provide a useful, consistent framework for handling, tracking, and defining the minimum documentation associated with the processing of GIs.

- Only potential adequate protection, substantial safety enhancement, and reduction in unnecessary regulatory burden issues are subject to the GIP process.
- Resolution of a GI may involve developing and imposing new or revised rules, developing new or revised guidance, revising the interpretation of rules or guidance, or providing information for voluntary actions.
- Resolution of a GI may affect licensees, certificate holders, or other entities regulated by or subject to NRC's regulatory jurisdiction.
- The process stages in the GIP are identification, initial screening, technical assessment, regulation and guidance development, regulation and guidance issuance, implementation, and verification.

Appendices A through G of MD 6.4 [1858], give detailed information on the submittal and assessment of GIs.

<u>Overview of Generic Issues Program Stages</u>

Only generic issues (GIs) that potentially involve adequate protection, substantial safety enhancement, or reduction in unnecessary regulatory burden are included in the Generic Issues Program.

The GIP consists of the following stages:

- Identification: Stage 1
- Initial Screening: Stage 2
- Technical Assessment: Stage 3
- Regulation and Guidance Development: Stage 4
- Regulation and Guidance Issuance: Stage 5
- Implementation: Stage 6
- Verification: Stage 7
- Closure: Stage 8

Descriptions of each of the stages, including products, are given below.

1. Identification

Candidate GIs may be identified by organizations or individuals internal or external to NRC, including the NRC staff, the Agreement State staff, the ACRS, the Advisory Committee on Nuclear Waste (ACNW), the Advisory Committee on the Medical Uses of Isotopes (ACMUI), licensees, certificate holders, industry groups, or the general public.

If any identified candidate GI has the potential for involving an adequate protection issue, prompt actions is taken to evaluate the issue and to initiate any necessary compensatory measures.

Candidate GIs may be identified by NRC or Agreement State staff during routine activities. Sources of candidate GIs include, but are not limited to, NRC staff concerns; public concerns; licensee event reports; morning reports; inspection reports; investigation reports; accident sequence precursor program reports; major studies; allegation reports; component failure reports; 10 CFR Part 21, "Reporting of Defects and Noncompliance," reports; industry reports; and reports of operational events at foreign facilities..

Guidance for identifying GIs from operational safety data reviews is contained in Management Directive (MD) 8.5, "Operational Safety Data Review." [1927]

Candidate GIs are submitted to the GIP Manager in RES, who forwards them to either the Reactor Generic Issue Review Panel or the Materials Generic Issue Review Panel, as appropriate. For candidate GIs that involve both program areas, the GIP Manager consults with the program offices to establish a combined review panel including representatives of the Office of Nuclear Reactor Regulation (NRR), the Office of Nuclear Material Safety and Safeguards (NMSS), and RES. For security-related candidate GIs, NSIR participation is required.

The issues identified as Unresolved Safety Issues (USIs) or any staff concerns identified as part of the Differing Professional Opinion program may also be addressed under the GIP.

2. Initial Screening

During initial screening, the appropriate Generic Issue Review Panel assesses whether the candidate GI should be processed in the GIP, should be excluded from further analyses, or should be sent to another NRC program for review. Also, the scope of the candidate GI (and thus the GI) is defined at this stage.

Initial screening is complete after the appropriate Generic Issue Review Panel reviews the information submitted in accordance with Appendix A of MD 6.4 [1858], including any other supporting documentation, as well as any staff-generated screening analysis of the candidate issue, and submits its findings and recommendations to the Director of RES for reactor issues or to the Director of NMSS for materials issues.

For reactor candidate GIs, risk and cost benefit thresholds are provided in Appendix C of MD 6.4 [1858], "Criteria and Guidance for Technical Assessment of Candidate Reactor Generic Issues." During initial screening (Stage 2), the Reactor Generic Issue Review Panel uses, to the extent practicable, Appendix C of MD 6.4 [1858] in a comparative manner to determine whether the issue should be excluded from further analyses, or continue on to Stage 3, technical assessment, in which a quantitative analysis would be performed. For materials candidate GIs, the initial screening stage may be folded into the technical assessment stage. Appendix F of MD 6.4 [1858], "Criteria and Guidance for Technical Assessment of Candidate Materials Generic Issues," provides guidance on the conduct of panel meetings.

Figure A.G.1, "Candidate Generic Issue Screening Process," and Appendix G of MD 6.4 [1858] provide the questions that must be addressed during the GI classification process, primarily in Stages 2 and 3 of the GIP.

On the basis of established risk thresholds or engineering judgment, the Reactor or the Materials Generic Issue Review Panel assesses whether the candidate GI has the potential to be classified as either an adequate protection, a substantial safety enhancement, or a reduction in unnecessary regulatory burden issue. (The actual classification into one of these categories will be made at the technical assessment stage.) The Reactor or the Materials Generic Issue Review Panel makes its assessment on the basis of information readily available or easily obtained with reasonable resources.

For a candidate GI, either the Reactor or the Materials Generic Issue Review Panel, as appropriate, issues an initial screening memorandum consisting of a forwarding note with attached findings and recommended actions. In some instances, the appropriate Generic Issue Review Panel may recommend that the screening and assessment stages for reduction in unnecessary regulatory burden issues be modified, or performed at a lower level of effort. The panel documents its recommendation in its initial screening memorandum. As a minimum, the initial screening memorandum is to include a clear, concise description of the GI, its safety significance, and Appendix A of MD 6.4 [1858] information prepared by the submitter.

3. Technical Assessment

In the technical assessment stage, staff (a) perform additional review of those GIs that may represent an adequate protection issue, a substantial safety enhancement issue, or a reduction in unnecessary regulatory burden issue; (b) determine if these should be designated as unresolved safety issues (USIs); and (c) identify a cost-effective solution to the GI.

Technical assessment also provides technical justification for excluding from further analyses a GI that has little safety significance, would not result in a substantial safety enhancement, is not cost justifiable, or is a necessary regulatory burden.

Guidance for performing a technical assessment of a reactor GI is provided in Appendix B, "Unresolved Safety Issue Assessment Criteria," and Appendix C of MD 6.4 [1858]. Guidance for performing a technical assessment of a materials GI would use more qualitative methods, expert elicitation, and judgment as outlined in Appendix F MD 6.4 [1858].

Technical assessment is an "indepth" study of a GI and may involve contractor support. To form a technical basis for taking or not taking regulatory action, the technical assessment stage may include the following:

- expert elicitation
- a review of operational data and events
- a review of related generic communications and GIs
- model development, experiments, and tests
- system and computational analyses
- field studies and inspections
- probabilistic risk assessments
- integrated safety assessments
- a detailed regulatory analysis
- corrective action development, including recommendations

The extent of these activities varies in accordance with the scope, complexity, or significance of the GI and the depth of information available on a given GI.

The target completion date for the technical assessment stage will be determined by office management in the course of approving the Task Action Plan (TAP) for this stage (see Appendix D of MD 6.4 [1858], "Generic Issue Task Action Plan"). The implementation of this plan will be given a priority consistent with the generic issue's safety significance, other work efforts, and budget constraints of the implementing office. This priority assignment is the prerogative of the NRC office responsible for the technical assessment.

Either RES (for reactor issues) or NMSS (for materials issues), as appropriate, conducts or oversees the technical evaluation of the GI, verifies the legitimacy of the concern expressed, verifies that the benefits sought will be obtained, establishes the technical basis for new or revised regulations or guidance, and identifies solutions that are likely to result in substantial net facility safety improvements or reduction in regulatory burden without significant decrease in safety.

Technical assessment is complete when the RPM informs either the Director of NRR (for reactor issues), or the Director of NMSS (for materials issues) whether (1) the issue should be excluded from further consideration, (2) new or revised rules or guidance are needed, and/or (3) new or revised NRC programs are needed, or

4. Regulation and Guidance Development

Regulation and guidance development involves an indepth review of potential facility or program changes to address the GI and selection of needed regulatory actions. Technical findings obtained during the technical assessment stage are used, as necessary, as a basis for developing or revising rules, guidance, and programs. Products to be produced during the regulation and guidance development stage could include draft rules, regulatory guides, bulletins, generic letters, information notices, new or revised inspection procedures, and the CRGR briefing packages.

Typically, NRC rules and guidance are contained in Title 10 of the Code of Federal Regulations, standard review plans, regulatory guides, and, to some extent, bulletins, generic letters, information notices, and regulatory issue summaries, as well as pertinent office staff guidance.

The development of rules, guidance, or programs can take from several months to a few years, depending on the length of time required by the deliberations involved. If rulemaking is a potential option to address the GI, coordination between this directive and MD 6.3, [1928] "The Rulemaking Process," is required. The GI TAP, in accordance with this directive, and the rulemaking plan, in accordance with MD 6.3, [1928] is coordinated to reduce duplication of effort.

Regulation and guidance development is complete when the RPM informs either the Director of NRR (for reactor issues) or the Director of NMSS (for materials issues), whether (1) the GI should be excluded from further consideration or (2) new or revised regulations, guidance, or programs have been developed to address the GI.

5. Regulation and Guidance Issuance

The staff issue documents clearly describing the facility or program changes developed during the regulation and guidance development stage to address the GI in a timely and effective manner. New or revised regulations may require the review and approval of the Commission except in limited circumstances when the EDO is authorized to conduct rulemaking in accordance with MD 6.3, [1928] "The Rulemaking Process."

Regulation and guidance issuance is complete when the RPM informs either the Director of NRR (for reactor issues), or the Director of NMSS (for materials issues), whether (1) the issue should be excluded from further consideration or (2) new or revised regulations, guidance, or programs have been issued to address the GI.

Regulation and guidance issuance is complete when the RPM informs either the Director of NRR (for reactor issues) or the Director of NMSS (for materials issues) whether (1) the issue should be excluded from further consideration or (2) new or revised regulations, guidance, or programs have been issued to address the GI.

6. Implementation

The objective of the implementation stage is to determine whether the licensee, the certificate holder, or other entity regulated by or subject to the regulatory jurisdiction of NRC has established and is implementing a program to ensure that facility or program changes made to address a GI are effective and in accordance with commitments.

The implementation stage occurs when the affected licensee, certificate holder, or other entity performs the actions necessary to implement the regulatory action to close the GI. These may include modifications or additions to

- the systems, structures, components, or design of a facility;

• the design approval or manufacturing license for a facility;
• the technical specifications, procedures, programs, or organization required to design, construct, or operate a facility.

The implementation stage is complete for an affected licensee, certificate holder, or other entity once it has formally informed the appropriate NRC program office that facility or program changes have been implemented. The GIP Manager in RES monitors the implementation of GI facility or program changes by the licensee, the certificate holder, or other entity as reported by the RPM and includes this information in updates to the GIMCS.

7. Verification

In the verification stage, the appointed staff determines whether licensees, certificate holders, or other entities have adequately demonstrated the efficacy of facility or program changes in addressing the GI.

The verification stage involves auditing and inspection of individual licensees and certificate holders to verify that effective actions have been implemented. Depending on the number of affected licensees, certificate holders, or other entities, the risk significance of the GI, and the complexity of the corrective actions, it may not be necessary to perform a 100-percent inspection of facility or program changes made in order to declare a GI closed.

The verification stage is complete for an affected licensee or certificate holder once the final inspection report has been issued, and the appropriate NRC program office determines that facility or program changes are adequate. The RPM provides documentation giving the basis for declaring the verification stage complete for a specific licensee, certificate holder, or other entity to the GIP Manager in RES for review.

8. Closure

Closure can begin when the verification stage is complete for all affected licensees, certificate holders, or other entities once:

• All final verification inspection reports have been issued.
• The appropriate NRC program office has determined that actions have been implemented and are adequate to classify the GI as closed.
• The RPM has prepared a memorandum to the Executive Director for Operations, through the GIP Manager in RES and the Director of RES, indicating the basis for declaring the GI closed.

V. GENERIC ISSUES PROGRAM (2007-Current)

SECY-07-0022 [1888] describes improvements to the GIP, which the staff will implement to ensure comprehensive and timely resolution of future GIs. The staff will implement these conceptual GIP improvements through a revision to MD 6.4.

To improve the GIP, the following elements were indentified to be incorporated in MD 6.4:

1. With the appropriate regulatory office involvement, RES will have overall responsibility for GIP management, including routine tracking and documentation of GIP status as well as periodic reporting to Congress and the Commission.

2. The appropriate regulatory office will have well-defined roles, responsibilities, and accountability in all stages of GI assessment and resolution.

3. All offices will be involved with applying the screening criteria to identify issues that are suitable for the GIP. Issues for which the risk or safety significance cannot be adequately determined due to phenomena or other uncertainties, and would require long-term studies and/or experimental research to establish the risk or safety significance will be excluded from the GIP, consistent with current processes specified in MD 6.4.

4. Issues, particularly high-risk issues, that should be addressed by other NRC programs and processes or industry initiatives, will be appropriately directed to those programs and processes. The role of the GIP will be clarified with the roles of other programs that address the concerns of employees and stakeholders such as the Differing Professional Opinion (DPO) Program and the Allegation Program to ensure that GIP does not serve as an alternative to these programs.

5. To gain efficiency and effectiveness and improve timely assessment of GIs, the staff will employ enhanced risk-informed techniques, which have already been developed as part of other established programs (e.g., the Accident Sequence Precursor [ASP] Program).

6. RES will ensure necessary inter-office coordination throughout the process. After the issue is screened in as a formal GI, the GIP will consider participation by nuclear industry stakeholders, when feasible, to identify possible solutions (e.g., a regulatory product or industry initiative).

7. The GI process will be concluded when the regulatory product is identified. The regulatory office will proceed, under other established programs and processes, to develop and implement the identified regulatory solution, and perform appropriate verification.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report presents the resolution of generic safety issues related to nuclear power plants. The purpose of these evaluations are to assist in the timely and efficient allocation of NRC resources for the resolution of those safety issues that have a significant potential for reducing risk.

Issues primarily concerned with the licensing process or environmental protection and not directly related to safety were excluded from prioritization/screening. The issues were 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. Future supplements to this report will include additional issues that completed major milestones as well as updated information on issues that have been resolved.

The agency's Generic Issues Program process for resolving GIs is described in MD 6.4, "Generic Issues Program", and SECY-07-0022. These documents provide recent program improvement initiatives. This new process includes five distinct stages that may be exercised: Identification, Acceptance Review, Screening, Safety / Risk Assessment, and Regulatory Assessment. Prior to implementation of MD 6.4 (1999), the safety priority rankings were HIGH, MEDIUM, LOW, and DROP and 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 priority ranking model described above.

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