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

March 1988

SUPPLEMENT 7 TO NUREG-0933
"A PRIORITIZATION OF GENERIC SAFETY ISSUES"

REVISION INSERTION INSTRUCTIONS

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References	pp. R-1 to 59, Rev. 3	pp. R-1 to 69, Rev. 4

TABLE II

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

This table contains the priority designations for all issues listed in this report. For those issues found to be covered in other issues, 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 resolved issues that have resulted in new requirements for operating plants, the appropriate multiplant licensing action number is listed. The licensing action numbering system bears no relationship to the numbering systems used for identifying the prioritized issues. An explanation of the classification and status of the issues is provided in the legend below.

Legend

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

HIGH	- High Safety Priority
MEDIUM	- Medium Safety Priority
LOW	- Low Safety Priority
DROP	- Issue Dropped as a Generic Issue
E	- Environmental Issue
I	- TMI Action Plan Item With Implementation of Resolution Mandated by NUREG-0737[98]
LI	- Licensing Issue
MPA	- Multiplant Action
NA	- Not Applicable
RI	- Regulatory Impact Issue
USI	- Unresolved Safety Issue

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
	TMI ACTION PLAN ITEMS						
I.A	OPERATING PERSONNEL						
I.A.1	Operating Personnel and Staffing						
I.A.1.1	Shift Technical Advisor	-	NRR/DHFS/LQB	I	2	12/31/86	F-01
I.A.1.2	Shift Supervisor Administrative Duties	-	NRR/DHFS/LQB	I	2	12/31/86	
I.A.1.3	Shift Manning	-	NRR/DHFS/LQB	I	2	12/31/86	F-02
I.A.1.4	Long-Term Upgrading	Colmar	RES/DFO/HFBR	NOTE 3(a)	2	12/31/86	
I.A.2	Training and Qualifications of Operating Personnel						
I.A.2.1	Immediate Upgrading of Operator and Senior Operator Training and Qualifications	-	-	-			
I.A.2.1(1)	Qualifications - Experience	-	NRR/DHFS/LQB	I	5	12/31/87	F-03
I.A.2.1(2)	Training	-	NRR/DHFS/LQB	I	5	12/31/87	F-03
I.A.2.1(3)	Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	-	NRR/DHFS/LQB	I	5	12/31/87	F-03
I.A.2.2	Training and Qualifications of Operations Personnel	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.2.3	Administration of Training Programs	-	NRR/DHFS/LQB	I	5	12/31/87	
I.A.2.4	NRR Participation in Inspector Training	Colmar	NRR/DHFS/LQB	LI (NOTE 3)	5	12/31/87	NA
I.A.2.5	Plant Drills	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.2.6	Long-Term Upgrading of Training and Qualifications	-	-	-			
I.A.2.6(1)	Revise Regulatory Guide 1.8	Colmar	NRR/DHFT/HFIB	NOTE 3(a)	5	12/31/87	NA
I.A.2.6(2)	Staff Review of NRR 80-117	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.2.6(3)	Revise 10 CFR 55	Colmar	NRR/DHFS/LQB	I.A.2.2	5	12/31/87	NA
I.A.2.6(4)	Operator Workshops	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.2.6(5)	Develop Inspection Procedures for Training Program	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.2.6(6)	Nuclear Power Fundamentals	Colmar	NRR/DHFS/LQB	DROP	5	12/31/87	NA
I.A.2.7	Accreditation of Training Institutions	Colmar	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/87	NA
I.A.3	Licensing and Requalification of Operating Personnel						
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	Emrit	NRR/DHFS/LQB	I	5	12/31/86	
I.A.3.2	Operator Licensing Program Changes	Emrit	NRR/DHFS/OLB	NOTE 3(b)	5	12/31/86	NA
I.A.3.3	Requirements for Operator Fitness	Colmar	RES/DRAO/HFSB	NOTE 3(b)	5	12/31/86	NA
I.A.3.4	Licensing of Additional Operations Personnel	Thatcher	NRR/DHFS/LQB	NOTE 3(b)	5	12/31/86	NA
I.A.3.5	Establish Statement of Understanding with INPO and DOE	Thatcher	NRR/DHFS/HFEB	LI (NOTE 3)	5	12/31/86	NA
I.A.4	Simulator Use and Development						
I.A.4.1	Initial Simulator Improvement	-	-	-			
I.A.4.1(1)	Short-Term Study of Training Simulators	Thatcher	NRR/DHFS/OLB	NOTE 3(b)	4	12/31/87	NA
I.A.4.1(2)	Interim Changes in Training Simulators	Thatcher	NRR/DHFS/OLB	NOTE 3(a)	4	12/31/87	
I.A.4.2	Long-Term Training Simulator Upgrade	-	-	-			
I.A.4.2(1)	Research on Training Simulators	Colmar	NRR/DHFT/HFIB	NOTE 3(a)	4	12/31/87	

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
I.A.4.2(2)	Upgrade Training Simulator Standards	Colmar	RES/DFO/HFBR	NOTE 3(a)	4	12/31/87	
I.A.4.2(3)	Regulatory Guide on Training Simulators	Colmar	RES/DFO/HFBR	NOTE 3(a)	4	12/31/87	
I.A.4.2(4)	Review Simulators for Conformance to Criteria	Colmar	NRR/DLPQ/LOLB	HIGH	4	12/31/87	
I.A.4.3	Feasibility Study of Procurement of NRC Training Simulator	Colmar	RES/DAE/RSRB	LI (NOTE 3)	4	12/31/87	NA
I.A.4.4	Feasibility Study of NRC Engineering Computer	Colmar	RES/DAE/RSRB	LI (NOTE 3)	4	12/31/87	NA
I.B.	SUPPORT PERSONNEL						
I.B.1	Management for Operations						
I.B.1.1	Organization and Management Long-Term Improvements	-	-	-			
I.B.1.1(1)	Prepare Draft Criteria	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	3	12/31/86	NA
I.B.1.1(2)	Prepare Commission Paper	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	3	12/31/86	NA
I.B.1.1(3)	Issue Requirements for the Upgrading of Management and Technical Resources	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	3	12/31/86	NA
I.B.1.1(4)	Review Responses to Determine Acceptability	Colmar	NRR/DHFT/HFIB	NOTE 3(b)	3	12/31/86	NA
I.B.1.1(5)	Review Implementation of the Upgrading Activities	Colmar	OIE/DQASIP/ORPB	NOTE 3(b)	3	12/31/86	NA
I.B.1.1(6)	Prepare Revisions to Regulatory Guides 1.33 and 1.8	Colmar	NRR/DHFS/LQB	I.A.2.6(1), 75	3	12/31/86	NA
I.B.1.1(7)	Issue Regulatory Guides 1.33 and 1.8	Colmar	NRR/DHFS/LQB	I.A.2.6(1), 75	3	12/31/86	NA
I.B.1.2	Evaluation of Organization and Management Improvements of Near-Term Operating License Applicants	-	-	-			
I.B.1.2(1)	Prepare Draft Criteria	-	NRR/DHFS/LQB	NOTE 3(b)	3	12/31/86	NA
I.B.1.2(2)	Review Near-Term Operating License Facilities	-	NRR/DHFS/LQB	NOTE 3(b)	3	12/31/86	NA
I.B.1.2(3)	Include Findings in the SER for Each Near-Term Operating License Facility	-	NRR/DL/ORAB	NOTE 3(b)	3	12/13/86	NA
I.B.1.3	Loss of Safety Function	-	-	-			
I.B.1.3(1)	Require Licensees to Place Plant in Safest Shutdown Cooling Following a Loss of Safety Function Due to Personnel Error	Sege	RES	LI (NOTE 3)	3	12/31/86	NA
I.B.1.3(2)	Use Existing Enforcement Options to Accomplish Safest Shutdown Cooling	Sege	RES	LI (NOTE 3)	3	12/31/86	NA
I.B.1.3(3)	Use Non-Fiscal Approaches to Accomplish Safest Shutdown Cooling	Sege	RES	LI (NOTE 3)	3	12/31/86	NA
I.B.2	Inspection of Operating Reactors						
I.B.2.1	Revise OIE Inspection Program	-	-	-			
I.B.2.1(1)	Verify the Adequacy of Management and Procedural Controls and Staff Discipline	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(2)	Verify that Systems Required to Be Operable Are Properly Aligned	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Lead SPEB Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
I.B.2.1(3)	Follow-up on Completed Maintenance Work Orders to Assure Proper Testing and Return to Service	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(4)	Observe Surveillance Tests to Determine Whether Test Instruments Are Properly Calibrated	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(5)	Verify that Licensees Are Complying with Technical Specifications	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(6)	Observe Routine Maintenance	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.1(7)	Inspect Terminal Boards, Panels, and Instrument Racks for Unauthorized Jumpers and Bypasses	Sege	OIE/DQASIP/RCPB	LI (NOTE 3)		11/30/83	NA
I.B.2.2	Resident Inspector at Operating Reactors	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)		11/30/83	NA
I.B.2.3	Regional Evaluations	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)		11/30/83	NA
I.B.2.4	Overview of Licensee Performance	Sege	OIE/DQASIP/ORPB	LI (NOTE 3)		11/30/83	NA
I.C	OPERATING PROCEDURES						
I.C.1	Short-Term Accident Analysis and Procedures Revision	-	-	-			
I.C.1(1)	Small Break LOCAs	-	NRR	I	3	12/31/86	
I.C.1(2)	Inadequate Core Cooling	-	NRR	I	3	12/31/86	F-04
I.C.1(3)	Transients and Accidents	-	NRR	I	3	12/31/86	F-05
I.C.1(4)	Confirmatory Analyses of Selected Transients	Riggs	NRR/DSI/RSB	NOTE 3(b)	3	12/31/86	NA
I.C.2	Shift and Relief Turnover Procedures	-	NRR	I	3	12/31/86	
I.C.3	Shift Supervisor Responsibilities	-	NRR	I	3	12/31/86	
I.C.4	Control Room Access	-	NRR	I	3	12/31/86	
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	-	NRR/DL	I	3	12/31/86	F-06
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	-	NRR/DL	I	3	12/31/86	F-07
I.C.7	NSSS Vendor Review of Procedures	-	NRR/DHFS/PSRB	I	3	12/31/86	
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	-	NRR/DHFS/PSRB	I	3	12/31/86	
I.C.9	Long-Term Program Plan for Upgrading of Procedures	Riggs	NRR/DHFS/PSRB	NOTE 3(b)	3	12/31/86	NA
I.D	CONTROL ROOM DESIGN						
I.D.1	Control Room Design Reviews	-	NRR/DL	I	3	12/31/86	F-08
I.D.2	Plant Safety Parameter Display Console	-	NRR/DL	I	3	12/31/86	F-09
I.D.3	Safety System Status Monitoring	Thatcher	RES/DE/MEB	MEDIUM	3	12/31/86	
I.D.4	Control Room Design Standard	Thatcher	RES/DRPS/RHFB	MEDIUM	3	12/31/86	NA
I.D.5	Improved Control Room Instrumentation Research	-	-	-			
I.D.5(1)	Operator-Process Communication	Thatcher	RES/DFO/HFBR	NOTE 3(b)	3	12/31/86	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
I.D.5(2)	Plant Status and Post-Accident Monitoring	Thatcher	RES/DFO/HFBR	NOTE 3(a)	3	12/31/86	
I.D.5(3)	On-Line Reactor Surveillance System	Thatcher	RES/DE/MEB	NOTE 1	3	12/31/86	
I.D.5(4)	Process Monitoring Instrumentation	Thatcher	RES/DFO/ICBR	NOTE 3(b)	3	12/31/86	NA
I.D.5(5)	Disturbance Analysis Systems	Thatcher	RES/DRPS/RHFB	MEDIUM	3	12/31/86	NA
I.D.6	Technology Transfer Conference	Thatcher	RES/DFO/HFBR	LI (NOTE 3)	3	12/31/86	NA
I.E	ANALYSIS AND DISSEMINATION OF OPERATING EXPERIENCE						
I.E.1	Office for Analysis and Evaluation of Operational Data	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.2	Program Office Operational Data Evaluation	Matthews	NRR/DL/ORAB	LI (NOTE 3)	1	6/30/84	NA
I.E.3	Operational Safety Data Analysis	Matthews	RES/DRA/RRBR	LI (NOTE 3)	1	6/30/84	NA
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.5	Nuclear Plant Reliability Data System	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.6	Reporting Requirements	Matthews	AEOD/PTB	LI (NOTE 3)	1	6/30/84	NA
I.E.7	Foreign Sources	Matthews	IP	LI (NOTE 3)	1	6/30/84	NA
I.E.8	Human Error Rate Analysis	Matthews	RES/DFO/HFBR	LI (NOTE 3)	1	6/30/84	NA
I.F	QUALITY ASSURANCE						
I.F.1	Expand QA List	Pittman	RES/DRA/ARGIB	HIGH	1	12/31/85	
I.F.2	Develop More Detailed QA Criteria	-	-	-			
I.F.2(1)	Assure the Independence of the Organization Performing the Checking Function	Pittman	OIE/DQASIP/QUAB	LOW	1	12/31/85	NA
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	1	12/31/85	NA
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	1	12/31/85	NA
I.F.2(4)	Establish Criteria for Determining QA Requirements for Specific Classes of Equipment	Pittman	OIE/DQASIP/QUAB	LOW	1	12/31/85	NA
I.F.2(5)	Establish Qualification Requirements for QA and QC Personnel	Pittman	OIE/DQASIP/QUAB	LOW	1	12/31/85	NA
I.F.2(6)	Increase the Size of Licensees' QA Staff	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	1	12/31/85	NA
I.F.2(7)	Clarify that the QA Program Is a Condition of the Construction Permit and Operating License	Pittman	OIE/DQASIP/QUAB	LOW	1	12/31/85	NA
I.F.2(8)	Compare NRC QA Requirements with Those of Other Agencies	Pittman	OIE/DQASIP/QUAB	LOW	1	12/31/85	NA
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	Pittman	OIE/DQASIP/QUAB	NOTE 3(a)	1	12/31/85	NA
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation	Pittman	OIE/DQASIP/QUAB	LOW	1	12/31/85	NA
I.F.2(11)	Define Role of QA in Design and Analysis Activities	Pittman	OIE/DQASIP/QUAB	LOW	1	12/31/85	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
I.G	PREOPERATIONAL AND LOW-POWER TESTING						
I.G.1	Training Requirements	-	NRR/DHFS/PSRB	I			
I.G.2	Scope of Test Program	V'Molen	NRR/DHFS/PSRB	NOTE 3(a)	1	12/31/84	NA
II.A	SITING						
II.A.1	Siting Policy Reformulation	V'Molen	NRR/DE/SAB	NOTE 3(b)	1	12/31/84	NA
II.A.2	Site Evaluation of Existing Facilities	V'Molen	NRR/DE/SAB	V.A.1	1	12/31/84	NA
II.B	CONSIDERATION OF DEGRADED OR MELTED CORES IN SAFETY REVIEW						
II.B.1	Reactor Coolant System Vents	-	NRR/DL	I			F-10
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	-	NRR/DL	I			F-11
II.B.3	Post-Accident Sampling	-	NRR/DL	I			F-12
II.B.4	Training for Mitigating Core Damage	-	NRR/DL	I			F-13
II.B.5	Research on Phenomena Associated with Core Degradation and Fuel Melting	-	-	-			
II.B.5(1)	Behavior of Severely Damaged Fuel	V'Molen	RES/DRAA/AEB	HIGH	1	12/31/85	
II.B.5(2)	Behavior of Core Melt	V'Molen	RES/DRAA/PRAB	HIGH	1	12/31/85	
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structure	V'Molen	RES/DRAA/AEB	MEDIUM	1	12/31/85	
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	Pittman	NRR/DST/RRAB	NOTE 3(a)	1	12/31/85	
II.B.7	Analysis of Hydrogen Control	Matthews	NRR/DSI/CSB	II.B.8	1	12/31/85	
II.B.8	Rulemaking Proceeding on Degraded Core Accidents	V'Molen	RES/DRAO/RAMR	NOTE 3(a)	1	12/31/85	
II.C	RELIABILITY ENGINEERING AND RISK ASSESSMENT						
II.C.1	Interim Reliability Evaluation Program	Pittman	RES/DRAO/RRB	NOTE 3(b)	1	12/31/85	NA
II.C.2	Continuation of Interim Reliability Evaluation Program	Pittman	NRR/DST/RRAB	NOTE 3(b)	1	12/31/85	NA
II.C.3	Systems Interaction	Pittman	NRR/DST/GIB	A-17	1	12/31/85	NA
II.C.4	Reliability Engineering	Pittman	RES/DRPS/RHFB	HIGH	1	12/31/85	
II.D	REACTOR COOLANT SYSTEM RELIEF AND SAFETY VALVES						
II.D.1	Testing Requirements	-	NRR/DL	I			F-14
II.D.2	Research on Relief and Safety Valve Test Requirements	Riggs	RES	LOW		11/30/83	NA
II.D.3	Relief and Safety Valve Position Indication	-	NRR	I			

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.E	SYSTEM DESIGN						
II.E.1	Auxiliary Feedwater System						
II.E.1.1	Auxiliary Feedwater System Evaluation	-	NRR/DL	I	1	12/31/86	F-15
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	-	NRR/DL	I	1	12/31/86	F-16, F-17
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	Riggs	RES/DRA/RRBR	NOTE 3(a)	1	12/31/86	
II.E.2	Emergency Core Cooling System						
II.E.2.1	Reliance on ECCS	Riggs	NRR/DSI/RSB	II.K.3(17)	1	12/31/85	NA
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	Riggs	RES/DAE/RSRB	NOTE 3(b)	1	12/31/85	NA
II.E.2.3	Uncertainties in Performance Predictions	V'Molen	NRR/DSI/RSB	LOW	1	12/31/85	NA
II.E.3	Decay Heat Removal						
II.E.3.1	Reliability of Power Supplies for Natural Circulation	-	NRR	I			
II.E.3.2	Systems Reliability	V'Molen	NRR/DST/GIB	A-45		11/30/83	NA
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements	V'Molen	NRR/DST/GIB	A-45		11/30/83	NA
II.E.3.4	Alternate Concepts Research	Riggs	RES/DAE/FBRB	NOTE 3(b)		11/30/83	NA
II.E.3.5	Regulatory Guide	Riggs	NRR/DST/GIB	A-45		11/30/83	NA
II.E.4	Containment Design						
II.E.4.1	Dedicated Penetrations	-	NRR/DL	I			F-18
II.E.4.2	Isolation Dependability	-	NRR/DL	I			F-19
II.E.4.3	Integrity Check	Milstead	RES/DRPS/RPSI	HIGH		11/30/83	
II.E.4.4	Purging	-	-	-			
II.E.4.4(1)	Issue Letter to Licensees Requesting Limited Purging	Milstead	NRR/DSI/CSB	NOTE 3(a)		11/30/83	
II.E.4.4(2)	Issue Letter to Licensees Requesting Information on Isolation Letter	Milstead	NRR/DSI/CSB	NOTE 3(a)		11/30/83	
II.E.4.4(3)	Issue Letter to Licensees on Valve Operability	Milstead	NRR/DSI/CSB	NOTE 3(a)		11/30/83	
II.E.4.4(4)	Evaluate Purging and Venting During Normal Operation	Milstead	NRR/DSI/CSB	NOTE 3(b)		11/30/83	NA
II.E.4.4(5)	Issue Modified Purging and Venting Requirement	Milstead	NRR/DSI/CSB	NOTE 3(b)		11/30/83	NA
II.E.5	Design Sensitivity of B&W Reactors						
II.E.5.1	Design Evaluation	Thatcher	NRR/DSI/RSB	NOTE 3(a)	1	12/31/84	
II.E.5.2	B&W Reactor Transient Response Task Force	Thatcher	NRR/DL/ORAB	NOTE 3(a)	1	12/31/84	
II.E.6	In Situ Testing of Valves						
II.E.6.1	Test Adequacy Study	Thatcher	RES/DE/EIB	MEDIUM		11/30/83	

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.F	INSTRUMENTATION AND CONTROLS						
II.F.1	Additional Accident Monitoring Instrumentation	-	NRR/DL	I			F-20, F-21, F-22, F-23, F-24, F-25 F-26
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	-	NRR/DL	I			
II.F.3	Instruments for Monitoring Accident Conditions	V'Molen	RES/DFO/ICBR	NOTE 3(a)		11/30/83	
II.F.4	Study of Control and Protective Action Design Requirements	Thatcher	NRR/DSI/ICSB	DROP		11/30/83	NA
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	Thatcher	RES/DE	MEDIUM		11/30/83	
II.G	ELECTRICAL POWER						
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	-	NRR	I			
II.H	TMI-2 CLEANUP AND EXAMINATION						
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact	Matthews	NRR/TMIPO	NOTE 3(b)		11/30/83	NA
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	Milstead	RES/DRAA/AEB	HIGH		11/30/83	
II.H.3	Evaluate and Feed Back Information Obtained from TMI	Milstead	NRR/TMIPO	[illegible]		11/30/83	NA
II.H.4	Determine Impact of TMI on Socioeconomic and Real Property Values	Milstead	RES/DHSWM/SE[illegible]	[illegible]TE 3)		11/30/83	NA
II.J	GENERAL IMPLICATIONS OF TMI FOR DESIGN AND CONSTRUCTION ACTIVITIES						
II.J.1	Vendor Inspection Program						
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.1.2	Modify Existing Vendor Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.J.2	Construction Inspection Program						
II.J.2.1	Reorient Construction Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.2.3	Assign Resident Inspectors to All Construction Sites	Riani	OIE/DQASIP	LI (NOTE 3)		11/30/83	NA
II.J.3	Management for Design and Construction						
II.J.3.1	Organization and Staffing to Oversee Design and Construction	Pittman	NRR/DHFS/LQB	I.B.1.1		11/30/83	NA
II.J.3.2	Issue Regulatory Guide	Pittman	NRR/DHFS/LQB	I.B.1.1		11/30/83	NA
II.J.4	Revise Deficiency Reporting Requirements						
II.J.4.1	Revise Deficiency Reporting Requirements	Riani	AEOD/DSP/ROAB	NOTE 2		11/30/83	
II.K	MEASURES TO MITIGATE SMALL-BREAK LOSS-OF-COOLANT ACCIDENTS AND LOSS-OF-FEEDWATER ACCIDENTS						
II.K.1	IE Bulletins	-	-	-			
II.K.1(1)	Review TMI-2 PNs and Detailed Chronology of the TMI-2 Accident	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(2)	Review Transients Similar to TMI-2 That Have Occurred at Other Facilities and NRC Evaluation of Davis-Besse Event	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(4)	Review Operating Procedures and Training Instructions	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(5)	Safety-Related Valve Position Description	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred out of Containment Inadvertently	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(11)	Make All Operating and Maintenance Personnel Aware of the Seriousness and Consequences of the Erroneous Actions Leading up to, and in Early Phases of, the TMI-2 Accident	Emrit	NRR	NOTE 3(a)		12/31/84	-

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.K.1(12)	One Hour Notification Requirement and Continuous Communications Channels	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(15)	For Facilities with Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(16)	Implement Procedures That Identify PRZ PORV "Open" Indications and That Direct Operator to Close Manually at "Reset" Setpoint	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(17)	Trip PZR Level Bistable so That PZR Low Pressure Will Initiate Safety Injection	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(18)	Develop Procedures and Train Operators on Methods of Establishing and Maintaining Natural Circulation	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(19)	Describe Design and Procedure Modifications to Reduce Likelihood of Automatic PZR PORV Actuation in Transients	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(20)	Provide Procedures and Training to Operators for Prompt Manual Reactor Trip for LOFW, TT, MSIV Closure, LOOP, LOSG Level, and LO PZR Level	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor Trip for LOFW, TT, or Significant Decrease in SG Level	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(24)	Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and RCP Trip	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(25)	Develop Operator Action Guidelines	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2	Commission Orders on B&W Plants	-	-	-			
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(2)	Procedures and Training to Initiate and Control AFW Independent of Integrated Control System	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(4)	Small-Break LOCA Analysis, Procedures and Operator Training	Emrit	NRR/DHFS/OLB	NOTE 3(a)		12/31/84	-

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	Emrit	NRR	NOTE 3(a)		12/31/84	-
II.K.2(6)	Reevaluate Analysis for Dual-Level Setpoint Control	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(7)	Reevaluate Transient of September 24, 1977	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.2(8)	Continued Upgrading of AFW System	Emrit	NRR	II.E.1.1, II.E.1.2		12/31/84	NA
II.K.2(9)	Analysis and Upgrading of Integrated Control System	Emrit	NRR	I		12/31/84	F-27
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	Emrit	NRR	I		12/31/84	F-28
II.K.2(11)	Operator Training and Drilling	Emrit	NRR	I		12/31/84	F-29
II.K.2(12)	Transient Analysis and Procedures for Management of Small Breaks	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.2(13)	Thermal-Mechanical Report on Effect of HPI on Vessel Integrity for Small-Break LOCA With No AFW	Emrit	NRR	I		12/31/84	F-30
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	Emrit	NRR	I		12/31/84	F-31
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes After Primary System Voiding	Emrit	NRR	I		12/31/84	-
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA With Loss of Offsite Power	Emrit	NRR	I		12/31/84	F-32
II.K.2(17)	Analysis of Potential Voiding in RCS During Anticipated Transients	Emrit	NRR	I		12/31/84	F-33
II.K.2(18)	Analysis of Loss of Feedwater and Other Anticipated Transients	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.2(19)	Benchmark Analysis of Sequential AFW Flow to Once-Through Steam Generator	Emrit	NRR	I		12/31/84	F-34
II.K.2(20)	Analysis of Steam Response to Small-Break LOCA That Causes System Pressure to Exceed PORV Setpoint	Emrit	NRR	I		12/31/84	F-35
II.K.2(21)	LOFT L3-1 Predictions	Emrit	NRR/DSI	NOTE 3(a)		12/31/84	-
II.K.3	Final Recommendations of Bulletins and Orders Task Force	-	-	-			
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	Emrit	NRR	I		12/31/84	F-36
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	Emrit	NRR	I		12/31/84	F-37
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	Emrit	NRR	I		12/31/84	F-38
II.K.3(4)	Review and Upgrade Reliability and Redundancy of Non-Safety Equipment for Small-Break LOCA Mitigation	Emrit	NRR	II.C.1, II.C.2, II.C.3		12/31/84	NA
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	Emrit	NRR	I		12/31/84	F-39, G-01
II.K.3(6)	Instrumentation to Verify Natural Circulation	Emrit	NRR/DSI	I.C.1(3), II.F.2, II.F.3		12/31/84	NA
II.K.3(7)	Evaluation of PORV Opening Probability During Overpressure Transient	Emrit	NRR	I		12/31/84	-

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs	Emrit	NRR/DST/GIB	II.C.1, II.E.3.3		12/31/84	NA
II.K.3(9)	Proportional Integral Derivative Controller Modification	Emrit	NRR	I		12/31/84	F-40
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	Emrit	NRR	I		12/31/84	F-41
II.K.3(11)	Control Use of PORV Supplied by Control Components, Inc. Until Further Review Complete	Emrit	NRR	I		12/31/84	-
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	Emrit	NRR	I		12/31/84	F-42
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	Emrit	NRR	I		12/31/84	F-43
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	Emrit	NRR	I		12/31/84	F-44
II.K.3(15)	Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	Emrit	NRR	I		12/31/84	F-45
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification	Emrit	NRR	I		12/31/84	F-46
II.K.3(17)	Report on Outage of ECC Systems - Licensee Report and Technical Specification Changes	Emrit	NRR	I		12/31/84	F-47
II.K.3(18)	Modification of ADS Logic - Feasibility Study and Modification for Increased Diversity for Some Event Sequences	Emrit	NRR	I		12/31/84	F-48
II.K.3(19)	Interlock on Recirculation Pump Loops	Emrit	NRR	I		12/31/84	F-49
II.K.3(20)	Loss of Service Water for Big Rock Point	Emrit	NRR	I		12/31/84	-
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level - Design and Modification	Emrit	NRR	I		12/31/84	F-50
II.K.3(22)	Automatic Switchover of RCIC System Suction - Verify Procedures and Modify Design	Emrit	NRR	I		12/31/84	F-51
II.K.3(23)	Central Water Level Recording	Emrit	NRR	I.D.2, III.A.1.2(1), III.A.3.4		12/31/84	NA
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	Emrit	NRR	I		12/31/84	F-52
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	Emrit	NRR	I		12/31/84	F-53
II.K.3(26)	Study Effect on RHR Reliability of Its Use for Fuel Pool Cooling	Emrit	NRR/DSI	II.E.2.1		12/31/84	NA
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	Emrit	NRR	I		12/31/84	F-54
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	Emrit	NRR	I		12/31/84	F-55
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	Emrit	NRR	I		12/31/84	F-56
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	Emrit	NRR	I		12/31/84	F-57
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	Emrit	NRR	I		12/31/84	F-58

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
II.K.3(32)	Provide Experimental Verification of Two-Phase Natural Circulation Models	Emrit	NRR/DSI	II.E.2.2		12/31/84	NA
II.K.3(33)	Evaluate Elimination of PORV Function	Emrit	NRR	II.C.1		12/31/84	NA
II.K.3(34)	Relap-4 Model Development	Emrit	NRR/DSI	II.E.2.2		12/31/84	NA
II.K.3(35)	Evaluation of Effects of Core Flood Tank Injection on Small-Break LOCAs	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(36)	Additional Staff Audit Calculations of B&W Small-Break LOCA Analyses	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(37)	Analysis of B&W Response to Isolated Small-Break LOCA	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(38)	Analysis of Plant Response to a Small-Break LOCA in the Pressurizer Spray Line	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(39)	Evaluation of Effects of Water Slugs in Piping Caused by HPI and CFT Flows	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(40)	Evaluation of RCP Seal Damage and Leakage During a Small-Break LOCA	Emrit	NRR	II.K.2(16)		12/31/84	NA
II.K.3(41)	Submit Predictions for LOFT Test L3-6 with RCPs Running	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(42)	Submit Requested Information on the Effects of Non-Condensible Gases	Emrit	NRR	I.C.1(3)		12/31/84	NA
II.K.3(43)	Evaluation of Mechanical Effects of Slug Flow on Steam Generator Tubes	Emrit	NRR	II.K.2(15)		12/31/84	NA
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	Emrit	NRR	I		12/31/84	F-59
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	Emrit	NRR	I		12/31/84	F-60
II.K.3(46)	Response to List of Concerns from ACRS Consultant	Emrit	NRR	I		12/31/84	F-61
II.K.3(47)	Test Program for Small-Break LOCA Model Verification Pretest Prediction, Test Program, and Model Verification	Emrit	NRR	I.C.1(3), II.E.2.2		12/31/84	NA
II.K.3(48)	Assess Change in Safety Reliability as a Result of Implementing B&OTF Recommendations	Emrit	NRR	II.C.1, II.C.2		12/31/84	NA
II.K.3(49)	Review of Procedures (NRC)	Emrit	NRR/DHFS/PSRB	I.C.8, I.C.9		12/31/84	NA
II.K.3(50)	Review of Procedures (NSSS Vendors)	Emrit	NRR/DHFS/PSRB	I.C.7, I.C.9		12/31/84	NA
II.K.3(51)	Symptom-Based Emergency Procedures	Emrit	NRR/DHFS/PSRB	I.C.9		12/31/84	NA
II.K.3(52)	Operator Awareness of Revised Emergency Procedures	Emrit	NRR	I.B.1.1, I.C.2, I.C.5		12/31/84	NA
II.K.3(53)	Two Operators in Control Room	Emrit	NRR	I.A.1.3		12/31/84	NA
II.K.3(54)	Simulator Upgrade for Small-Break LOCAs	Emrit	NRR	I.A.4.1(2)		12/31/84	NA
II.K.3(55)	Operator Monitoring of Control Board	Emrit	NRR	I.C.1(3), I.D.2, I.D.3		12/31/84	NA
II.K.3(56)	Simulator Training Requirements	Emrit	NRR/DHFS/OLB	I.A.2.6(3), I.A.3.1		12/31/84	NA
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	Emrit	NRR	I		12/31/84	F-62

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Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
III.A	EMERGENCY PREPAREDNESS AND RADIATION EFFECTS						
III.A.1	Improve Licensee Emergency Preparedness - Short Term						
III.A.1.1	Upgrade Emergency Preparedness	-	-	-			
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	-	OIE/DEPER/EPB	I			
III.A.1.1(2)	Perform an Integrated Assessment of the Implementation	-	OIE/DEPER/EPB	I			
III.A.1.2	Upgrade Licensee Emergency Support Facilities	-	-	-			
III.A.1.2(1)	Technical Support Center	-	OIE/DEPER/EPB	I			F-63
III.A.1.2(2)	On-Site Operational Support Center	-	OIE/DEPER/EPB	I			F-64
III.A.1.2(3)	Near-Site Emergency Operations Facility	-	OIE/DEPER/EPB	I			F-65
III.A.1.3	Maintain Supplies of Thyroid-Blocking Agent	-	-	-			
III.A.1.3(1)	Workers	Riggs	OIE/DEPER/EPB	NOTE 3(b)	1	12/31/85	NA
III.A.1.3(2)	Public	Riggs	OIE/DEPER/EPB	NOTE 3(b)	1	12/31/85	NA
III.A.2	Improving Licensee Emergency Preparedness-Long Term						
III.A.2.1	Amend 10 CFR 50 and 10 CFR 50, Appendix E	-	-	-			
III.A.2.1(1)	Publish Proposed Amendments to the Rules	-	RES	I			
III.A.2.1(2)	Conduct Public Regional Meetings	-	RES	I			
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules	-	RES	I			
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	-	OIE	I			F-67
III.A.2.2	Development of Guidance and Criteria	-	NRR/DL	I			F-68
III.A.3	Improving NRC Emergency Preparedness						
III.A.3.1	NRC Role in Responding to Nuclear Emergencies	-	-	-			
III.A.3.1(1)	Define NRC Role in Emergency Situations	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	6/30/85	NA
III.A.3.1(2)	Revise and Upgrade Plans and Procedures for the NRC Emergency Operations Center	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	6/30/85	NA
III.A.3.1(3)	Revise Manual Chapter 0502, Other Agency Procedures, and NUREG-0610	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	6/30/85	NA
III.A.3.1(4)	Prepare Commission Paper	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	6/30/85	NA
III.A.3.1(5)	Revise Implementing Procedures and Instructions for Regional Offices	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	6/30/85	NA
III.A.3.2	Improve Operations Centers	Riggs	OIE/DEPER/IRDB	NOTE 3(b)	1	6/30/85	NA
III.A.3.3	Communications	-	-	-			
III.A.3.3(1)	Install Direct Dedicated Telephone Lines	Pittman	OIE/DEPER/IRDB	NOTE 3(a)	1	6/30/85	NA
III.A.3.3(2)	Obtain Dedicated, Short-Range Radio Communication Systems	Pittman	OIE/DEPER/IRDB	NOTE 3(a)	1	6/30/85	NA
III.A.3.4	Nuclear Data Link	Thatcher	OIE/DEPER/IRDB	NOTE 3(b)	1	6/30/85	
III.A.3.5	Training, Drills, and Tests	Pittman	OIE/DEPER/IRDB	NOTE 3(b)	1	6/30/85	NA
III.A.3.6	Interaction of NRC and Other Agencies	-	-	-			
III.A.3.6(1)	International	Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	6/30/85	NA
III.A.3.6(2)	Federal	Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	6/30/85	NA
III.A.3.6(3)	State and Local	Pittman	OIE/DEPER/EPLB	NOTE 3(b)	1	6/30/85	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
III.B	EMERGENCY PREPAREDNESS OF STATE AND LOCAL GOVERNMENTS						
III.B.1	Transfer of Responsibilities to FEMA	Milstead	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.B.2	Implementation of NRC and FEMA Responsibilities	-	-	-			
III.B.2(1)	The Licensing Process	Milstead	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.B.2(2)	Federal Guidance	Milstead	OIE/DEPER/IRDB	NOTE 3(b)		11/30/83	NA
III.C	PUBLIC INFORMATION						
III.C.1	Have Information Available for the News Media and the Public	-	-	-			
III.C.1(1)	Review Publicly Available Documents	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.1(2)	Recommend Publication of Additional Information	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.1(3)	Program of Seminars for News Media Personnel	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.2	Develop Policy and Provide Training for Interfacing With the News Media	-	-	-			
III.C.2(1)	Develop Policy and Procedures for Dealing With Briefing Requests	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.C.2(2)	Provide Training for Members of the Technical Staff	Pittman	PA	LI (NOTE 3)		11/30/83	NA
III.D	RADIATION PROTECTION						
III.D.1	Radiation Source Control						
III.D.1.1	Primary Coolant Sources Outside the Containment Structure	-	-	-			
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining to Reducing Leakage from Operating Systems	-	NRR	I			
III.D.1.1(2)	Review Information on Provisions for Leak Detection	Emrit	NRR/DSI/METB	NOTE 4			
III.D.1.1(3)	Develop Proposed System Acceptance Criteria	Emrit	NRR/DSI/METB	NOTE 4			
III.D.1.2	Radioactive Gas Management	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
III.D.1.3	Ventilation System and Radioiodine Adsorber Criteria	-	-	-			
III.D.1.3(1)	Decide Whether Licensees Should Perform Studies and Make Modifications	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
III.D.1.3(2)	Review and Revise SRP	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Adsorber	Emrit	NRR/DSI/METB	NOTE 3(b)		11/30/83	NA
III.D.1.4	Radwaste System Design Features to Aid in Accident Recovery and Decontamination	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
III.D.2	Public Radiation Protection Improvement						
III.D.2.1	Radiological Monitoring of Effluents	-	-	-			
III.D.2.1(1)	Evaluate the Feasibility and Perform a Value-Impact Analysis of Modifying Effluent-Monitoring Design Criteria	Emrit	NRR/DSI/METB	LOW	2	12/31/85	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
III.D.2.1(2)	Study the Feasibility of Requiring the Development of Effective Means for Monitoring and Sampling Noble Gases and Radioiodine Released to the Atmosphere	Emrit	NRR/DSI/METB	LOW	2	12/31/85	NA
III.D.2.1(3)	Revise Regulatory Guides	Emrit	NRR/DSI/METB	LOW	2	12/31/85	NA
III.D.2.2	Radioiodine, Carbon-14, and Tritium Pathway Dose Analysis	-	-	-			
III.D.2.2(1)	Perform Study of Radioiodine, Carbon-14, and Tritium Behavior	Emrit	NRR/DSI/RAB	NOTE 3(b)	2	12/31/85	NA
III.D.2.2(2)	Evaluate Data Collected at Quad Cities	Emrit	NRR/DSI/RAB	III.D.2.5	2	12/31/85	NA
III.D.2.2(3)	Determine the Distribution of the Chemical Species of Radioiodine in Air-Water-Steam Mixtures	Emrit	NRR/DSI/RAB	III.D.2.5	2	12/31/85	NA
III.D.2.2(4)	Revise SRP and Regulatory Guides	Emrit	NRR/DSI/RAB	III.D.2.5	2	12/31/85	NA
III.D.2.3	Liquid Pathway Radiological Control	-	-	-			
III.D.2.3(1)	Develop Procedures to Discriminate Between Sites/Plants	Emrit	NRR/DE/EHEB	NOTE 3(b)	2	12/31/85	NA
III.D.2.3(2)	Discriminate Between Sites and Plants That Require Consideration of Liquid Pathway Interdiction Techniques	Emrit	NRR/DE/EHEB	NOTE 3(b)	2	12/31/85	NA
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction	Emrit	NRR/DE/EHEB	NOTE 3(b)	2	12/31/85	NA
III.D.2.3(4)	Prepare a Summary Assessment	Emrit	NRR/DE/EHEB	NOTE 3(b)	2	12/31/85	NA
III.D.2.4	Offsite Dose Measurements	-	-	-			
III.D.2.4(1)	Study Feasibility of Environmental Monitors	V'Molen	NRR/DSI/RAB	NOTE 3(b)	2	12/31/85	NA
III.D.2.4(2)	Place 50 TLDs Around Each Site	V'Molen	OIE/DRP/ORPB	LI (NOTE 3)	2	12/31/85	NA
III.D.2.5	Offsite Dose Calculation Manual	V'Molen	NRR/DSI/RAB	NOTE 3(b)	2	12/31/85	NA
III.D.2.6	Independent Radiological Measurements	V'Molen	OIE/DRP/ORPB	LI (NOTE 3)	2	12/31/85	NA
III.D.3	Worker Radiation Protection Improvement						
III.D.3.1	Radiation Protection Plans	V'Molen	NRR/DSI/RAB	NOTE 3(b)	3	12/31/87	NA
III.D.3.2	Health Physics Improvements	-	-	-			
III.D.3.2(1)	Amend 10 CFR 20	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(2)	Issue a Regulatory Guide	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(3)	Develop Standard Performance Criteria	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.2(4)	Develop Method for Testing and Certifying Air-Purifying Respirators	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	3	12/31/87	NA
III.D.3.3	In-plant Radiation Monitoring	-	-	-			
III.D.3.3(1)	Issue Letter Requiring Improved Radiation Sampling Instrumentation	-	NRR/DL	I	2		F-69
III.D.3.3(2)	Set Criteria Requiring Licensees to Evaluate Need for Additional Survey Equipment	-	NRR	NOTE 3(a)	2	12/31/86	NA
III.D.3.3(3)	Issue a Rule Change Providing Acceptable Methods for Calibration of Radiation-Monitoring Instruments	-	RES	NOTE 3(a)	2	12/31/86	NA
III.D.3.3(4)	Issue a Regulatory Guide	-	RES	NOTE 3(a)	2	12/31/86	NA
III.D.3.4	Control Room Habitability	-	NRR/DL	I			F-70
III.D.3.5	Radiation Worker Exposure	-	-	-			
III.D.3.5(1)	Develop Format for Data To Be Collected by Utilities Regarding Total Radiation Exposure to Workers	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
III.D.3.5(2)	Investigative Methods of Obtaining Employee Health Data by Nonlegislative Means	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA
III.D.3.5(3)	Revise 10 CFR 20	V'Molen	RES/DFO/ORPBR	LI (NOTE 3)	2	12/31/86	NA

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Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
IV.A	STRENGTHEN ENFORCEMENT PROCESS						
IV.A.1	Seek Legislative Authority	Emrit	GC	LI (NOTE 3)		11/30/83	NA
IV.A.2	Revise Enforcement Policy	Emrit	OIE/ES	LI (NOTE 3)		11/30/83	NA
IV.B	ISSUANCE OF INSTRUCTIONS AND INFORMATION TO LICENSEES						
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	Emrit	OIE/DEPER	LI (NOTE 3)		11/30/83	NA
IV.C	EXTEND LESSONS LEARNED TO LICENSED ACTIVITIES OTHER THAN POWER REACTORS						
IV.C.1	Extend Lessons Learned from TMI to Other NRC Programs	Emrit	NMSS/WM	NOTE 3(b)		11/30/83	NA
IV.D	NRC STAFF TRAINING						
IV.D.1	NRC Staff Training	Emrit	ADM/MDTS	LI (NOTE 3)		11/30/83	NA
IV.E	SAFETY DECISION-MAKING						
IV.E.1	Expand Research on Quantification of Safety Decision-Making	Colmar	RES/DRA/RABR	LI (NOTE 3)	2	12/31/86	NA
IV.E.2	Plan for Early Resolution of Safety Issues	Emrit	NRR/DST/SPEB	LI (NOTE 3)	2	12/31/86	NA
IV.E.3	Plan for Resolving Issues at the CP Stage	Colmar	RES/DRA/RABR	LI (NOTE 5)	2	12/31/86	NA
IV.E.4	Resolve Generic Issues by Rulemaking	Colmar	RES/DRA/RABR	LI (NOTE 3)	2	12/31/86	NA
IV.E.5	Assess Currently Operating Reactors	Matthews	NRR/DL/SEPB	NOTE 3(b)	2	12/31/86	NA
IV.F	FINANCIAL DISINCENTIVES TO SAFETY						
IV.F.1	Increased OIE Scrutiny of the Power-Ascension Test Program	Thatcher	OIE/DQASIP	NOTE 3(b)	1	12/31/86	NA
IV.F.2	Evaluate the Impacts of Financial Disincentives to the Safety of Nuclear Power Plants	Matthews	SP	NOTE 3(b)	1	12/31/86	NA

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Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
IV.G	IMPROVE SAFETY RULEMAKING PROCEDURES						
IV.G.1	Develop a Public Agenda for Rulemaking	Emrit	ADM/RPB	LI (NOTE 3)	1	12/31/86	NA
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules	Milstead	RES/DRA/RABR	LI (NOTE 3)	1	12/31/86	NA
IV.G.3	Improve Rulemaking Procedures	Milstead	RES/DRA/RABR	LI (NOTE 3)	1	12/31/86	NA
IV.G.4	Study Alternatives for Improved Rulemaking Process	Milstead	RES/DRA/RABR	LI (NOTE 3)	1	12/31/86	NA
IV.H	NRC PARTICIPATION IN THE RADIATION POLICY COUNCIL						
IV.H.1	NRC Participation in the Radiation Policy Council	Sege	RES/DHSWM/HEBR	LI (NOTE 3)		11/30/83	NA
V.A	DEVELOPMENT OF SAFETY POLICY						
V.A.1	Develop NRC Policy Statement on Safety	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.B	POSSIBLE ELIMINATION OF NONSAFETY RESPONSIBILITIES						
V.B.1	Study and Recommend, as Appropriate, Elimination of Nonsafety Responsibilities	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.C	ADVISORY COMMITTEES						
V.C.1	Strengthen the Role of Advisory Committee on Reactor Safeguards	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.C.2	Study Need for Additional Advisory Committees	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.C.3	Study the Need to Establish an Independent Nuclear Safety Board	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.D	LICENSING PROCESS						
V.D.1	Improve Public and Intervenor Participation in the Hearing Process	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.D.2	Study Construction-During-Adjudication Rules	Emrit	GC	LI (NOTE 5)		12/31/86	NA
V.D.3	Reexamine Commission Role in Adjudication	Emrit	GC	LI (NOTE 5)		12/31/86	NA
V.D.4	Study the Reform of the Licensing Process	Emrit	GC	LI (NOTE 5)		12/31/86	NA
V.E	LEGISLATIVE NEEDS						
V.E.1	Study the Need for TMI-Related Legislation	Emrit	GC	LI (NOTE 5)		12/31/86	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
V.F	ORGANIZATION AND MANAGEMENT						
V.F.1	Study NRC Top Management Structure and Process	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.F.2	Reexamine Organization and Functions of the NRC Offices	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.F.3	Revise Delegations of Authority to Staff	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.F.4	Clarify and Strengthen the Respective Roles of Chairman, Commission, and Executive Director for Operations	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.F.5	Authority to Delegate Emergency Response Functions to a Single Commissioner	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.G	CONSOLIDATION OF NRC LOCATIONS						
V.G.1	Achieve Single Location, Long-Term	Emrit	GC	LI (NOTE 3)		12/31/86	NA
V.G.2	Achieve Single Location, Interim	Emrit	GC	LI (NOTE 3)		12/31/86	NA
	TASK ACTION PLAN ITEMS						
A-1	Water Hammer	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]		1	6/30/85 NA
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]		1	6/30/85 D-10
A-3	Westinghouse Steam Generator Tube Integrity	-	NRR/DEST/EMTB	USI		11/30/83	
A-4	CE Steam Generator Tube Integrity	-	NRR/DEST/EMTB	USI		11/30/83	
A-5	B&W Steam Generator Tube Integrity	-	NRR/DEST/EMTB	USI		11/30/83	
A-6	Mark I Short-Term Program	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]		1	6/30/85
A-7	Mark I Long-Term Program	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]		1	6/30/85 D-01
A-8	Mark II Containment Pool Dyanmic Loads Long-Term Program	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]		1	6/30/85 NA
A-9	ATWS	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]		1	6/30/85
A-10	BWR Feedwater Nozzle Cracking	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]		1	6/30/85 B-25
A-11	Reactor Vessel Materials Toughness	Emrit	NRR/DST/GIB	USI [NOTE 3(a)]		1	6/30/85
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	Emrit	NRR/DST/GIB	USI [NOTE 2]	1	6/30/85	NA
A-13	Snubber Operability Assurance	Emrit	NRR/DE/MEB	NOTE 3(a)		11/30/83	
A-14	Flaw Detection	Matthews	NRR/DE/MTEB	DROP		11/30/83	NA
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning	Pittman	NRR/DE/CHEB	NOTE 3(b)		11/30/83	NA
A-16	Steam Effects on BWR Core Spray Distribution	Emrit	NRR/DSI/CPB	NOTE 3(a)		11/30/83	D-12
A-17	Systems Interaction	-	RES/DE/EIB	USI		11/30/83	
A-18	Pipe Rupture Design Criteria	Emrit	NRR/DE/MEB	DROP		11/30/83	NA
A-19	Digital Computer Protection System	Thatcher	NRR/DSI/ICSB	NOTE 4		11/30/83	
A-20	Impacts of the Coal Fuel Cycle	-	NRR/DE/EHEB	LI (NOTE 5)		11/30/83	NA
A-21	Main Steamline Break Inside Containment - Evaluation of Environmental Conditions for Equipment Qualification	V'Molen	NRR/DSI/CSB	LOW		11/30/83	NA

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Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
A-22	PWR Main Steamline Break - Core, Reactor Vessel and Containment Building Response	V'Molen	NRR/DSI/CSB	DROP		11/30/83	NA
A-23	Containment Leak Testing	Matthews	NRR/DSI/CSB	RI (NOTE 5)		11/30/83	
A-24	Qualification of Class 1E Safety-Related Equipment	-	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	B-60
A-25	Non-Safety Loads on Class 1E Power Sources	Thatcher	NRR/DSI/PSB	NOTE 3(a)		11/30/83	
A-26	Reactor Vessel Pressure Transient Protection	-	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	B-04
A-27	Reload Applications	-	NRR/DSI/CPB	LI (NOTE 5)		11/30/83	NA
A-28	Increase in Spent Fuel Pool Storage Capacity	Colmar	NRR/DE/SGEB	NOTE 3(a)		11/30/83	
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	Colmar	RES/DRPS/RPSI	MEDIUM		11/30/83	
A-30	Adequacy of Safety-Related DC Power Supplies	Sege	NRR/DSI/PSB	128	1	12/31/86	NA
A-31	RHR Shutdown Requirements	-	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	
A-32	Missile Effects	Pittman	NRR/DE/MTEB	A-37, A-38, B-68		11/30/83	NA
A-33	NEPA Review of Accident Risks	-	NRR/DSI/AEB	E(NOTE 3)		11/30/83	NA
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents	V'Molen	NRR/DSI/ICSB	II.F.3		11/30/83	NA
A-35	Adequacy of Offsite Power Systems	Emrit	NRR/DSI/PSB	NOTE 3(a)		11/30/83	
A-36	Control of Heavy Loads Near Spent Fuel	-	NRR/DSI/GIB	USI [NOTE 3(a)]	1	6/30/85	C-10, C-15
A-37	Turbine Missiles	Pittman	NRR/DE/MTEB	DROP		11/30/83	NA
A-38	Tornado Missiles	Sege	NRR/DSI/ASB	LOW		11/30/83	NA
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits	-	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	
A-40	Seismic Design Criteria - Short Term Program	-	RES/DE/EIB	USI		11/30/83	
A-41	Long Term Seismic Program	Colmar	NRR/DE/MEB	NOTE 3(b)	1	12/31/84	NA
A-42	Pipe Cracks in Boiling Water Reactors	-	NRR/DST/GIB	USI [NOTE 3(a)]	1	6/30/85	B-05
A-43	Containment Emergency Sump Performance	-	NRR/DST/GIB	USI [NOTE 3(a)]	1	12/31/87	
A-44	Station Blackout	-	RES/DRPS/RPSI	USI		11/30/83	
A-45	Shutdown Decay Heat Removal Requirements	-	RES/DRPS/RPSI	USI		11/30/83	
A-46	Seismic Qualification of Equipment in Operating Plants	-	NRR/DSRO/EIB	USI [NOTE 3(a)]	1	12/31/87	
A-47	Safety Implications of Control Systems	-	RES/DE/EIB	USI		11/30/83	
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	-	NRR/DRAA/SAIB	USI		11/30/83	
A-49	Pressurized Thermal Shock	-	NRR/DSRO/RSIB	USI [NOTE 3(a)]	1	12/31/87	
B-1	Environmental Technical Specifications	-	NRR/DE/EHEB	E (NOTE 3)		11/30/83	NA
B-2	Forecasting Electricity Demand	-	NRR	E (NOTE 3)		11/30/83	NA
B-3	Event Categorization	-	NRR/DSI/RSB	LI (DROP)		11/30/83	NA
B-4	ECCS Reliability	Emrit	NRR/DSI/RSB	II.E.3.2		11/30/83	NA
B-5	Ductility of Two Way Slabs and Shells and Buckling Behavior of Steel Containments	Thatcher	RES/DE/EIB	MEDIUM		11/30/83	
B-6	Loads, Load Combinations, Stress Limits	Pittman	NRR/DSRO/EIB	119.1		12/31/87	NA
B-7	Secondary Accident Consequence Modeling	-	NRR/DSI/AEB	LI (DROP)		11/30/83	NA
B-8	Locking Out of ECCS Power Operated Valves	Riggs	NRR/DSI/RSB	DROP		11/30/83	NA
B-9	Electrical Cable Penetrations of Containment	Emrit	NRR/DSI/PSB	NOTE 3(b)		11/30/83	NA
B-10	Behavior of BWR Mark III Containments	V'Molen	NRR/DSI/CSB	NOTE 3(a)	1	12/31/84	NA
B-11	Subcompartment Standard Problems	-	NRR/DSI/CSB	LI (NOTE 5)		11/30/83	NA

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Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
B-12	Containment Cooling Requirements (Non-LOCA)	Emrit	NRR/DSI/CSB	NOTE 3(b)	1	12/31/86	NA
B-13	Marviken Test Data Evaluation	-	NRR/DSI/CSB	LI (NOTE 5)		11/30/83	NA
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA	Emrit	NRR/DST/GIB	A-48		11/30/83	NA
B-15	CONTEMPT Computer Code Maintenance	-	NRR/DSI/CSB	LI (DROP)		11/30/83	NA
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	Emrit	NRR/DE/MEB	A-18		11/30/83	NA
B-17	Criteria for Safety-Related Operator Actions	Milstead	RES/DRPS/RHFB	MEDIUM	2	12/31/86	
B-18	Vortex Suppression Requirements for Containment Sumps	Emrit	NRR/DST/GIB	A-43		11/30/83	NA
B-19	Thermal-Hydraulic Stability	Colmar	NRR/DSI/CPB	NOTE 3(b)		6/30/85	NA
B-20	Standard Problem Analysis	-	RES/DAE/AMBR	LI (NOTE 5)		11/30/83	
B-21	Core Physics	-	NRR/DSI/CPB	LI (DROP)		11/30/83	NA
B-22	LWR Fuel	V'Molen	NRR/DSI/CPB	NOTE 4		11/30/83	
B-23	LMFBR Fuel	-	NRR/DSI/CPB	LI (DROP)		11/30/83	NA
B-24	Seismic Qualification of Electrical and Mechanical Components	Emrit	NRR	A-46		11/30/83	NA
B-25	Piping Benchmark Problems	-	NRR/DE/MEB	LI (NOTE 5)		11/30/83	
B-26	Structural Integrity of Containment Penetrations	Riggs	NRR/DE/MTEB	NOTE 3(b)	1	12/31/84	NA
B-27	Implementation and Use of Subsection NF	-	NRR/DE/MEB	LI (NOTE 5)		11/30/83	
B-28	Radionuclide/Sediment Transport Program	-	NRR/DE/EHEB	E (NOTE 3)		11/30/83	NA
B-29	Effectiveness of Ultimate Heat Sinks	Pittman	NRR/DE/EHEB	NOTE 4		11/30/83	
B-30	Design Basis Floods and Probability	-	NRR/DE/EHEB	LI (NOTE 5)		11/30/83	
B-31	Dam Failure Model	Milstead	NRR/DE/SGEB	NOTE 4		11/30/83	
B-32	Ice Effects on Safety Related Water Supplies	Milstead	NRR/DE/EHEB	NOTE 4		11/30/83	
B-33	Dose Assessment Methodology	-	NRR/DSI/RAB	LI (NOTE 3)		11/30/83	NA
B-34	Occupational Radiation Exposure Reduction	Emrit	NRR/DSI/RAB	III.D.3.1		11/30/83	NA
B-35	Confirmation of Appendix I Models for Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light Water Cooled Power Reactors	-	NRR/DSI/METB	LI (NOTE 5)		11/30/83	
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems	Emrit	NRR/DSI/METB	NOTE 3(a)		11/30/83	
B-37	Chemical Discharges to Receiving Waters	-	NRR/DE/EHEB	E (NOTE 5)		11/30/83	
B-38	Reconnaissance Level Investigations	-	NRR/DE/EHEB	E (DROP)		11/30/83	NA
B-39	Transmission Lines	-	NRR/DE/EHEB	E (DROP)		11/30/83	NA
B-40	Effects of Power Plant Entrainment on Plankton	-	NRR/DE/EHEB	E (DROP)		11/30/83	NA
B-41	Impacts on Fisheries	-	NRR/DE/EHEB	E (DROP)		11/30/83	NA
B-42	Socioeconomic Environmental Impacts	-	NRR/DE/SAB	E (NOTE 3)		11/30/83	NA
B-43	Value of Aerial Photographs for Site Evaluation	-	NRR/DE/EHEB	E (NOTE 5)		11/30/83	
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants	-	NRR/DE/SAB	E (NOTE 3)		11/30/83	NA
B-45	Need for Power - Energy Conservation	-	NRR/DE/SAB	E (B-2)		11/30/83	NA
B-46	Cost of Alternatives in Environmental Design	-	NRR/DE/SAB	E (DROP)		11/30/83	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
B-47	Inservice Inspection of Supports-Classes 1, 2, 3, and MC Components	Colmar	NRR/DE/MTEB	DROP		11/30/83	NA
B-48	BWR CRD Mechanical Failure (Collet Housing)	Emrit	NRR/DE/MTEB	NOTE 3(b)		11/30/83	
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments	-	NRR	LI (NOTE 5)		11/30/83	
B-50	Post-Operating Basis Earthquake Inspection	Colmar	NRR/DE/SGEB	RI (LOW)	1	06/30/85	NA
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components	Emrit	NRR/DE/MEB	A-40		11/30/83	NA
B-52	Fuel Assembly Seismic and LOCA Responses	Emrit	NRR/DST/GIB	A-2		11/30/83	NA
B-53	Load Break Switch	Sege	NRR/DSI/PSB	RI (NOTE 3)		11/30/83	
B-54	Ice Condenser Containments	Milstead	NRR/DSI/CSB	NOTE 3(b)	1	12/31/84	NA
B-55	Improved Reliability of Target Rock Safety Relief Valves	V'Molen	RES/DE/EIB	MEDIUM		11/30/83	
B-56	Diesel Reliability	Milstead	RES/DRPS/RPSI	HIGH		11/30/83	D-19
B-57	Station Blackout	Emrit	NRR/DST/GIB	A-44		11/30/83	
B-58	Passive Mechanical Failures	Colmar	NRR/DE/EQB	NOTE 3(b)	1	12/31/85	NA
B-59	(N-1) Loop Operation in BWRs and PWRs	Colmar	NRR/DSI/RSB	RI (NOTE 3)	1	6/30/85	E-04,E-05
B-60	Loose Parts Monitoring System	Emrit	NRR/DSI/CPB	NOTE 3(b)	1	12/31/84	NA
B-61	Allowable ECCS Equipment Outage Periods	Pittman	RES/DRAA/PRAB	MEDIUM		11/30/83	
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions	-	NRR/DSI/CPB	LI (DROP)		11/30/83	NA
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	Emrit	NRR/DE/MEB	NOTE 3(a)		11/30/83	
B-64	Decommissioning of Reactors	Colmar	RES/DE/MEB	NOTE 2		11/30/83	
B-65	Iodine Spiking	Milstead	NRR/DSI/AEB	DROP	2	12/31/84	NA
B-66	Control Room Infiltration Measurements	Matthews	NRR/DSI/AEB	NOTE 3(a)		11/30/83	
B-67	Effluent and Process Monitoring Instrumentation	Colmar	NRR/DSI/METB	III.D.2.1		11/30/83	NA
B-68	Pump Overspeed During LOCA	Riani	NRR/DSI/ASB	DROP		11/30/83	NA
B-69	ECCS Leakage Ex-Containment	Riani	NRR/DSI/METB	III.D.1.1(1)		11/30/83	NA
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps	Emrit	NRR/DSI/PSB	NOTE 3(a)		11/30/83	
B-71	Incident Response	Riani	NRR	III.A.3.1		11/30/83	NA
B-72	Health Effects and Life Shortening from Uranium and Coal Fuel Cycles	-	NRR/DSI/RAB	LI (NOTE 5)		11/30/83	NA
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel	Thatcher	NRR/DE/MEB	C-12		11/30/83	NA
C-1	Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	Milstead	NRR/DE/EQB	NOTE 3(a)		11/30/83	
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure	Emrit	NRR/DSI/CSB	NOTE 3(b)		11/30/83	NA
C-3	Insulation Usage Within Containment	Emrit	NRR/DST/GIB	A-43		11/30/83	NA
C-4	Statistical Methods for ECCS Analysis	Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-5	Decay Heat Update	Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
C-6	LOCA Heat Sources	Riggs	NRR/DSRO/SPEB	RI (NOTE 3)	1	06/30/86	NA
C-7	PWR System Piping	Emrit	NRR/DE/MTEB	NOTE 3(b)		11/30/83	NA
C-8	Main Steam Line Leakage Control Systems	Milstead	RES/DRPS/RPSI	HIGH		11/30/83	
C-9	RHR Heat Exchanger Tube Failures	V'Molen	NRR/DSI/RSB	DROP		11/30/83	NA
C-10	Effective Operation of Containment Sprays in a LOCA	Emrit	NRR/DSI/AEB	NOTE 3(a)		11/30/83	NA
C-11	Assessment of Failure and Reliability of Pumps and Valves	Emrit	NRR/DE/MEB	NOTE 3(b)		12/31/85	NA
C-12	Primary System Vibration Assessment	Thatcher	NRR/DE/MEB	NOTE 3(b)		11/30/83	NA
C-13	Non-Random Failures	Emrit	NRR/DST/GIB	A-17		11/30/83	NA
C-14	Storm Surge Model for Coastal Sites	Emrit	NRR/DE/EHEB	NOTE 4		11/30/83	
C-15	NUREG Report for Liquids Tank Failure Analysis	-	NRR/DE/EHEB	LI (DROP)		11/30/83	NA
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	-	NRR/DE/EHEB	E (DROP)		11/30/83	NA
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	Emrit	NRR/DSI/METB	NOTE 3(a)		11/30/83	NA
D-1	Advisability of a Seismic Scram	Thatcher	RES/DET/MSEB	LOW		11/30/83	NA
D-2	Emergency Core Cooling System Capability for Future Plants	Emrit	NRR/DSI/RSB	NOTE 4		11/30/83	
D-3	Control Rod Drop Accident	Emrit	NRR/DSI/CPB	NOTE 3(b)		11/30/83	NA

NEW GENERIC ISSUES

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
1.	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems	Emrit	NRR/DSI/METB	DROP		11/30/83	NA
2.	Failure of Protective Devices on Essential Equipment	Colmar	NRR/DSI/ICSB	NOTE 4		11/30/83	NA
3.	Set Point Drift in Instrumentation	Emrit	NRR/DSRO/RSIB	NOTE 3(b)	1	06/30/86	NA
4.	End-of-Life and Maintenance Criteria	Thatcher	NRR/DE/EQB	NOTE 3(b)		11/30/83	NA
5.	Design Check and Audit of Balance-of-Plant Equipment	Pittman	NRR/DSI/ASB	I.F.1		11/30/83	NA
6.	Separation of Control Rod from Its Drive and BWR High Rod Worth Events	V'Molen	NRR/DSI/CPB	NOTE 3(b)		11/30/83	NA
7.	Failures Due to Flow-Induced Vibrations	V'Molen	NRR/DSI/RSB	DROP		11/30/83	NA
8.	Inadvertent Actuation of Safety Injection in PWRs	Colmar	NRR/DSI/RSB	I.C.1		11/30/83	NA
9.	Reevaluation of Reactor Coolant Pump Trip Criteria	Emrit	NRR/DSI/RSB	II.K.3(5)		11/30/83	NA
10.	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges	Riggs	NRR/DSI/ICSB	DROP		11/30/83	NA
11.	Turbine Disc Cracking	Pittman	NRR/DE/MTEB	A-37		11/30/83	NA
12.	BWR Jet Pump Integrity	Sege	NRR/DE/MTEB, MEB	NOTE 3(b)	1	12/31/84	NA
13.	Small Break LOCA from Extended Overheating of Pressurizer Heaters	Riani	NRR/DSI/RSB	DROP		11/30/83	NA
14.	PWR Pipe Cracks	Emrit	NRR/DE/MTEB	NOTE 3(b)	1	12/31/85	NA
15.	Radiation Effects on Reactor Vessel Supports	Emrit	NRR/DE/MTEB	LOW		11/30/83	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
16.	BWR Main Steam Isolation Valve Leakage Control Systems	Milstead	NRR/DSI/ASB	C-8		11/30/83	NA
17.	Loss of Offsite Power Subsequent to LOCA	Colmar	NRR/DSI/PSB, ICSB	DROP		11/30/83	NA
18.	Steam Line Break with Consequential Small LOCA	Riggs	NRR/DSI/RSB	I.C.1		11/30/83	NA
19.	Safety Implications of Nonsafety Instrument and Control Power Supply Bus	Sege	NRR/DST/GIB	A-47		11/30/83	NA
20.	Effects of Electromagnetic Pulse on Nuclear Power Plants	Thatcher	NRR/DSI/ICSB	NOTE 3(b)	1	06/30/84	NA
21.	Vibration Qualification of Equipment	Riggs	NRR/DE/EIB	DROP	1	06/30/86	NA
22.	Inadvertent Boron Dilution Events	V'Molen	NRR/DSI/RSB	NOTE 3(b)	1	12/31/84	NA
23.	Reactor Coolant Pump Seal Failures	Riggs	RES/DE/EIB	HIGH		11/30/83	
24.	Automatic Emergency Core Cooling System Switch to Recirculation	V'Molen	NRR/DSI/RSB	NOTE 4		11/30/83	
25.	Automatic Air Header Dump on BWR Scram System	Milstead	NRR/DSI/RSB	NOTE 3(a)		11/30/83	
26.	Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power	Emrit	NRR/DSI/ASB	17		11/30/83	NA
27.	Manual vs. Automated Actions	Pittman	NRR/DSI/RSB	B-17		11/30/83	NA
28.	Pressurized Thermal Shock	Emrit	NRR/DST/GIB	A-49		11/30/83	NA
29.	Bolting Degradation or Failure in Nuclear Power Plants	V'Molen	RES/DE/EIB	HIGH		11/30/83	
30.	Potential Generator Missiles - Generator Rotor Retaining Rings	Pittman	NRR/DE/MEB	DROP	1	12/31/85	NA
31.	Natural Circulation Cooldown	Riggs	NRR/DSI/RSB	I.C.1		11/30/83	NA
32.	Flow Blockage in Essential Equipment Caused by Corbicula	Emrit	NRR/DSI/ASB	51		11/30/83	NA
33.	Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power	Pittman	NRR/DSI/ICSB	A-47		11/30/83	NA
34.	RCS Leak	Riggs	NRR/DHFS/PSRB	DROP	1	06/30/84	NA
35.	Degradation of Internal Appurtenances in LWRs	V'Molen	NRR/DSI/CPB, RSB	LOW	1	06/30/85	NA
36.	Loss of Service Water	Colmar	NRR/DSI/ASB, AEB, RSB	NOTE 3(b)	2	06/30/86	NA
37.	Steam Generator Overfill and Combined Primary and Secondary Blowdown	Colmar	NRR/DST/GIB, NRR/DSI/RSB	A-47, I.C.1(2)	1	06/30/85	NA
38.	Potential Recirculation System Failure as a Consequence of Injection of Containment Paint Flakes or Other Fine Debris	Milstead	RES/DRA/ARGIB	NOTE 4		11/30/83	
39.	Potential for Unacceptable Interaction Between the CRD System and Non-Essential Control Air System	Pittman	NRR/DSI/ASB	25		11/30/83	NA
40.	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	Colmar	NRR/DSI/ASB	NOTE 3(a)	1	06/30/84	B-65
41.	BWR Scram Discharge Volume Systems	V'Molen	NRR/DSI/RSB	NOTE 3(a)		11/30/83	B-58
42.	Combination Primary/Secondary System LOCA	Riggs	NRR/DSI/RSB	I.C.1	1	06/30/85	NA
43.	Contamination of Instrument Air Lines	Milstead	NRR/DSI/ASB	DROP		11/30/83	NA
44.	Failure of Saltwater Cooling System	Milstead	NRR/DSI/ASB	43		11/30/83	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
45.	Inoperability of Instrumentation Due to Extreme Cold Weather	Milstead	NRR/DSI/ICSB	NOTE 3(a)	1	06/30/84	
46.	Loss of 125 Volt DC Bus	Sege	NRR/DSI/PSB	76		11/30/83	NA
47.	Loss of Off-Site Power	Thatcher	NRR/DSI/RSB, ASB	NOTE 3(b)		11/30/83	
48.	LCO for Class 1E Vital Instrument Buses in Operating Reactors	Sege	NRR/DSI/PSB	128	1	12/31/86	NA
49.	Interlocks and LCOs for Redundant Class 1E Tie Breakers	Sege	NRR/DSI/PSB	128	2	12/31/86	NA
50.	Reactor Vessel Level Instrumentation in BWRs	Thatcher	NRR/DSI/RSB, ICSB	NOTE 3(b)	1	12/31/84	NA
51.	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	Emrit	RES/DE/EIB	MEDIUM		11/30/83	
52.	SSW Flow Blockage by Blue Mussels	Emrit	NRR/DSI/ASB	51		11/30/83	NA
53.	Consequences of a Postulated Flow Blockage Incident in a BWR	V'Molen	NRR/DSI/CPB, RSB	DROP	1	12/31/84	NA
54.	Valve Operator-Related Events Occurring During 1978, 1979, and 1980	Colmar	NRR/DE/MEB	II.E.6.1	1	06/30/85	NA
55.	Failure of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	Emrit	NRR/DSI/PSB	DROP	1	12/31/85	NA
56.	Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event	Colmar	NRR/DHFS/HFEB	A-47, I.D.1		11/30/83	NA
57.	Effects of Fire Protection System Actuation on Safety-Related Equipment	Milstead	RES/DRA/ARGIB	NOTE 4		11/30/83	
58.	Inadvertent Containment Flooding	Sege	NRR/DSI/ASB, CSB	DROP		11/30/83	
59.	Technical Specification Requirements for Plant Shutdown when Equipment for Safe Shutdown is Degraded or Inoperable	Emrit	NRR/DST/TSIP	RI (NOTE 5)	1	06/30/85	NA
60.	Lamellar Tearing of Reactor Systems Structural Supports	Colmar	NRR/DST/GIB	A-12		11/30/83	NA
61.	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments	Milstead	NRR/DSI/ASB	NOTE 3(b)	2	12/31/86	NA
62.	Reactor Systems Bolting Applications	Riggs	RES/DRA/ARGIB	NOTE 4		11/30/83	
63.	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis	Pittman	RES/DRA/ARGIB	NOTE 4		11/30/83	
64.	Identification of Protection System Instrument Sensing Lines	Thatcher	NRR/DSI/ICSB	NOTE 3(b)		11/30/83	
65.	Probability of Core-Melt Due to Component Cooling Water System Failures	V'Molen	NRR/DSI/ASB	23	1	12/31/86	NA
66.	Steam Generator Requirements	Riggs	NRR/DEST/EMTB	NOTE 2	1	06/30/85	
67.	Steam Generator Staff Actions	-	-	-		-	
67.2.1	Integrity of Steam Generator Tube Sleeves	Riggs	NRR/DE/MEB	RI (135)	2	12/31/87	NA
67.3.1	Steam Generator Overfill	Riggs	NRR/DST/GIB NRR/DSI/RSB	A-47, I.C.1	2	12/31/87	NA
67.3.2	Pressurized Thermal Shock	Riggs	NRR/DST/GIB	A-49	2	12/31/87	NA
67.3.3	Improved Accident Monitoring	Riggs	NRR/DSI/ICSB	NOTE 3(a)	2	12/31/87	A-17

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
67.3.4	Reactor Vessel Inventory Measurement	Riggs	NRR/DSI/CPB	II.F.2	2	12/31/87	NA
67.4.1	RCP Trip	Riggs	NRR/DSI/RSB	II.K.3(5)	2	12/31/87	NA
67.4.2	Control Room Design Review	Riggs	NRR/DHFS/HFEB	I.D.1	2	12/31/87	NA
67.4.3	Emergency Operating Procedures	Riggs	NRC/DHFS/PSRB	I.C.1	2	12/31/87	NA
67.5.1	Reassessment of SGTR Design Basis	Riggs	RES/DRPS/RPSI	LI (NOTE 5)	2	12/31/87	NA
67.5.2	Reevaluation of SGTR Design Basis	Riggs	RES/DRPS/RPSI	LI (NOTE 5)	2	12/31/87	NA
67.5.3	Secondary System Isolation	Riggs	NRR/DSI/RSB	DROP	2	12/31/87	NA
67.6.0	Organizational Responses	Riggs	OIE/DEPER/IRDB	III.A.3	2	12/31/87	NA
67.7.0	Improved Eddy Current Tests	Riggs	RES/DE/EIB	135	2	12/31/87	NA
67.8.0	Denting Criteria	Riggs	RES/DE/EIB	RI (135)	2	12/31/87	NA
67.9.0	Reactor Coolant System Pressure Control	Riggs	NRR/DSI/GIB NRR/DSI/RSB	A-45, I.C.1 (2,3)	2	12/31/87	NA
67.10.0	Supplement Tube Inspections	Riggs	NRR/DL/ORAB	LI (NOTE 5)	2	12/31/87	NA
68.	Postulated Loss of Auxiliary Feedwater System Resulting from Turbine-Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	Pittman	NRR/DSI/ASB	124	2	12/31/86	NA
69.	Make-up Nozzle Cracking in B&W Plants	Colmar	NRR/DE/MEB, MTEB	NOTE 3(b)	1	12/31/84	(later)
70.	PORV and Block Valve Reliability	Riggs	RES/DE/EIB	MEDIUM	1	6/30/84	
71.	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety	Pittman	RES/DRA/ARGIB	NOTE 4		11/30/83	
72.	Control Rod Drive Guide Tube Support Pin Failures	Riggs	RES/DRA/ARGIB	NOTE 4		11/30/83	
73.	Detached Thermal Sleeves	Riggs	RES/DRA/ARGIB	NOTE 4		11/30/83	
74.	Reactor Coolant Activity Limits for Operating Reactors	Milstead	NRR/DSI/AEB	DROP	1	06/30/86	NA
75.	Generic Implications of ATWS Events at the Salem Nuclear Plant	Thatcher	RES/DRA/ARGIB	NOTE 1		11/30/83	B-76,B-77 B-78,B-79 B-80,B-81 B-82,B-85 B-86,B-87 B-88,B-89 B-90,B-91 B-92,B-93
76.	Instrumentation and Control Power Interactions	Pittman	RES/DRA/ARGIB	NOTE 4		11/30/83	
77.	Flooding of Safety Equipment Compartments by Back-flow Through Floor Drains	Colmar	RES/DE/EIB	A-17		12/31/87	NA
78.	Monitoring of Fatigue Transient Limits for Reactor Coolant System	Riggs	RES/DRA/ARGIB	NOTE 4		11/30/83	
79.	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	Colmar	RES/DE/EIB	MEDIUM	1	12/31/84	
80.	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments	V'Molen	NRR/DSI/RSB, ASB, CPB	LOW		11/30/83	NA
81.	Impact of Locked Doors and Barriers on Plant Personnel and Safety	Colmar	NRR/DHFS/PSRB	DROP	1	12/31/84	NA
82.	Beyond Design Basis Accidents in Spent Fuel Pools	V'Molen	RES/DRPS/RPSI	MEDIUM		11/30/83	
83.	Control Room Habitability	Emrit	RES/DRAA/SAIB	NOTE 1	1	12/31/86	
84.	CE PORVs	Riggs	NRR/DEST/SRXB	NOTE 1	1	06/30/85	
85.	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	Milstead	NRR/DSI/CSB	DROP	1	12/31/85	NA

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
86.	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	Emrit	NRR/DEST/EMTB	NOTE 2		12/31/84	B-84
87.	Failure of HPCI Steam Line Without Isolation	Pittman	RES/DRPS/RPSI	HIGH		12/31/85	
88.	Earthquakes and Emergency Planning	Riggs	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
89	Stiff Pipe Clamps	Riggs	RES	NOTE 4		(later)	
90.	Technical Specifications for Anticipatory Trips	V'Molen	NRR/DSI/RSB, ICSB	LOW		12/31/84	NA
91.	Main Crankshaft Failures in Transamerica DeLaval Emergency Diesel Generators	Emrit	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
92.	Fuel Crumbling During LOCA	V'Molen	NRR/DSI/RSB, CPB	LOW		12/31/84	NA
93.	Steam Binding of Auxiliary Feedwater Pumps	Pittman	RES/DRPS/RPSI	HIGH		12/31/84	
94.	Additional Low Temperature Overpressure Protection Issues for Light Water Reactors	Pittman	RES/DRPS/RPSI	HIGH		13/31/85	
95.	Loss of Effective Volume for Containment Recirculation Spray	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
96.	RHR Suction Valve Testing	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
97.	PWR Reactor Cavity Uncontrolled Exposures	V'Molen	NRR/DSI/RAB	III.D.3.1		06/30/85	NA
98.	CRD Accumulator Check Valve Leakage	Pittman	NRR/DSI/ASB	DROP		06/30/85	NA
99.	RCS/RHR Suction Line Valve Interlock on PWRs	Pittman	RES/DRPS/RPSI	HIGH	1	06/30/86	
100.	OTSG Level	Riggs	RES/DRA/ARGIB	NOTE 4		(later)	
101.	Break Plus Single Failure in BWR Water Level Instrumentation	V'Molen	RES/DE/EIB	HIGH		06/30/85	
102.	Human Error in Events Involving Wrong Unit or Wrong Train	Emrit	NRR/DLPQ/LPEB	NOTE 1		12/31/86	
103.	Design for Probable Maximum Precipitation	Emrit	RES/DE/EIB	NOTE 1		12/31/85	
104.	Reduction of Boron Dilution Requirements	Pittman	RES	NOTE 4		(later)	
105.	Interfacing Systems LOCA at BWRs	Milstead	RES/DE/EIB	HIGH		06/30/85	
106.	Piping and Use of Highly Combustible Gases in Vital Areas	Milstead	RES/DRPS	MEDIUM		12/31/87	
107.	Generic Implications of Main Transformer Failures	Milstead	RES	NOTE 4		(later)	
108.	BWR Suppression Pool Temperature Limits	Colmar	NRR/DSI/CSB	RI (LOW)		06/30/85	NA
109.	Reactor Vessel Closure Failure	Riggs	RES/DRA/ARGIB	NOTE 4		(later)	
110.	Equipment Protective Devices on Engineered Safety Features	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
111.	Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments	Riggs	NRR/DE/MTEB	LI (NOTE 5)		12/31/85	NA
112.	Westinghouse RPS Surveillance Frequencies and Out-of-Service Times	Pittman	NRR/DSI/ICSB	RI (NOTE 3)		12/31/85	NA
113.	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	Riggs	RES/DE/EIB	HIGH		12/31/87	
114.	Seismic-Induced Relay Chatter	Riggs	NRR/DSRO/SPEB	A-46		06/30/86	NA
115.	Enhancement of the Reliability of Westinghouse Solid State Protection System	Milstead	RES/DRPS/RPSI	HIGH		12/31/86	
116.	Accident Management	Pittman	RES/DRA/ARGIB	NOTE 4		(later)	

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
117.	Allowable Outage Times for Diverse Simultaneous Equipment Outages	Pittman	RES/DRA/ARGIB	NOTE 4		(later)	
118.	Tendon Anchorage Failure	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
119.	Piping Review Committee Recommendations	-	-	-			
119.1	Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads	Riggs	NRR/DE	RI (NOTE 5)		12/31/85	NA
119.2	Piping Damping Values	Riggs	NRR/DE	RI (NOTE 5)		12/31/85	NA
119.3	Decoupling the OBE from the SSE	Riggs	NRR/DE	RI (NOTE 5)		12/31/85	NA
119.4	BWR Piping Materials	Riggs	NRR/DE	RI (NOTE 5)		12/31/85	NA
119.5	Leak Detection Requirements	Riggs	NRR/DE	RI (NOTE 5)		12/31/85	NA
120.	On-Line Testability of Protection Systems	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
121.	Hydrogen Control for Large, Dry PWR Containments	Emrit	RES/DRA/RDB	HIGH		12/31/85	
122.	Davis-Besse Loss of All Feedwater Event of June 9, 1985 - Short-Term Actions	-	-	-			
122.1	Potential Inability to Remove Reactor Decay Heat	-	-	-			
122.1.a	Failure of Isolation Valves in Closed Position	V'Molen	NRR/DSRO/RSIB	124	1	12/31/86	NA
122.1.b	Recovery of Auxiliary Feedwater	V'Molen	NRR/DSRO/RSIB	124	1	12/31/86	NA
122.1.c.	Interruption of Auxiliary Feedwater Flow	V'Molen	NRR/DSRO/RSIB	124	1	12/31/86	NA
122.2	Initiating Feed-and-Bleed	V'Molen	NRR/DEST/SRXB	HIGH	1	12/31/86	
122.3	Physical Security System Constraints	V'Molen	NRR/DSRO/SPEB	LOW	1	12/31/86	NA
123.	Deficiencies in the Regulations Governing DBA and Single-Failure Criteria Suggested by the Davis-Besse Event of June 9, 1985	Riggs	RES/DRA/ARGIB	NOTE 4		(later)	
124.	Auxiliary Feedwater System Reliability	Emrit	NRR/DEST/SRXB	NOTE 1	1	12/31/86	
125.	Davis-Besse Loss of All Feedwater Event of June 9, 1985 - Long-Term Actions	-	-	-			
125.I.1	Availability of the STA	V'Molen	RES/DRA/ARGIB	DROP	2	12/31/87	
125.I.2	PORV Reliability	-	-	-		-	
125.I.2.a	Need for a Test Program to Establish Reliability of the PORV	V'Molen	NRR/DSRO/SPEB	70	2	12/31/87	NA
125.I.2.b	Need for PORV Surveillance Tests to Confirm Operational Readiness	V'Molen	NRR/DSRO/SPEB	70	2	12/31/87	NA
125.I.2.c	Need for Additional Protection Against PORV Failure	V'Molen	NRR/DSRO/SPEB	DROP	2	12/31/87	NA
125.I.2.d	Capability of the PORV to Support Feed-and-Bleed	V'Molen	NRR/DSRO/SPEB	A-45	2	12/31/87	NA
125.I.3	SPDS Availability	Milstead	RES/DRA/ARGIB	NOTE 4	2	12/31/87	
125.I.4	Plant-Specific Simulator	Riggs	RES/DRA/ARGIB	DROP	2	12/31/87	
125.I.5	Safety Systems Tested in All Conditions Required by Design Basis Analysis	Riggs	RES/DRA/ARGIB	NOTE 4	2	12/31/87	
125.I.6	Valve Torque Limit and Bypass Switch Settings	Emrit	RES/DRA/ARGIB	DROP	2	12/31/87	
125.I.7	Operator Training Adequacy	-	-	-			

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
125.I.7.a	Recover Failed Equipment	Pittman	RES/DRA/ARGIB	DROP	2	12/31/87	
125.I.7.b	Realistic Hands-On Training	V'Molen	RES/DRA/ARGIB	DROP	2	12/31/87	
125.I.8	Procedures and Staffing for Reporting to NRC Emergency Response Center	V'Molen	RES/DRA/ARGIB	DROP	2	12/31/87	
125.II.1	AFW System Evaluation	-	-	-			
125.II.1.a	Two-Train AFW Unavailability	V'Molen	NRR/DSRO/SPEB	DROP	2	12/31/87	NA
125.II.1.b	Review Existing AFW Systems for Single Failure	V'Molen	NRR/DSRO/SPEB	124	2	12/31/87	NA
125.II.1.c	NUREG-0737 Reliability Improvements	V'Molen	NRR/DSRO/SPEB	DROP	2	12/31/87	NA
125.II.1.d	AFW/Steam and Feedwater Rupture Control System/ICS Interactions in B&W Plants	V'Molen	NRR/DSRO/SPEB	DROP	2	12/31/87	NA
125.II.2	Adequacy of Existing Maintenance Requirements for Safety-Related Systems	Riggs	RES/DRA/ARGIB	DROP	2	12/31/87	NA
125.II.3	Review Steam/Feedline Break Mitigation Systems for Single Failure	V'Molen	NRR/DSRO/SPEB	DROP	2	12/31/87	NA
125.II.4	Thermal Stress of OTSG Components	Riggs	NRR/DSRO/SPEB	DROP	2	12/31/87	NA
125.II.5	Thermal-Hydraulic Effects of Loss and Restoration of Feedwater on Primary System Components	Riggs	RES/DRA/ARGIB	DROP	2	12/31/87	
125.II.6	Reexamine PRA-Based Estimates of the Likelihood of a Severe Core Damage Accident Based on Loss of All Feedwater	V'Molen	RES/DRA/ARGIB	DROP	2	12/31/87	
125.II.7	Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break	V'Molen	RES/DRPS/RPSI	HIGH	2	12/31/87	
125.II.8	Reassess Criteria for Feed-and-Bleed Initiation	V'Molen	RES/DRA/ARGIB	DROP	2	12/31/87	
125.II.9	Enhanced Feed-and-Bleed Capability	V'Molen	NRR/DSRO/SPEB	DROP	2	12/31/87	NA
125.II.10	Hierarchy of Impromptu Operator Actions	Riggs	RES/DRA/ARGIB	DROP	2	12/31/87	
125.II.11	Recovery of Main Feedwater as Alternative to AFW	Riggs	RES/DRA/ARGIB	NOTE 4	2	12/31/87	
125.II.12	Adequacy of Training Regarding PORV Operation	Riggs	RES/DRA/ARGIB	DROP	2	12/31/87	
125.II.13	Operator Job Aids	Pittman	NRR/DRA/ARGIB	NOTE 4	2	12/31/87	
125.II.14	Remote Operation of Equipment Which Must Now Be Operated Locally	V'Molen	NRR/DSRO/SPEB	LOW	2	12/31/87	NA
126.	Reliability of PWR Main Steam Safety Valves	Riggs	RES/DRA/ARGIB	NOTE 4		(later)	
127.	Testing and Maintenance of Manual Valves in Safety-Related Systems	Pittman	RES/DRA/ARGIB	LOW		12/31/87	NA
128.	Electrical Power Reliability	Emrit	RES/DE/EIB	HIGH		12/31/86	
129.	Valve Interlocks to Prevent Vessel Drainage During Shutdown Cooling	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
130.	Essential Service Water Pump Failures at Multiplant Sites	Riggs	RES/DRPS/RPSI	HIGH		12/31/87	
131.	Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System in Westinghouse Plants	Riggs	RES/DRA/ARGIB	NOTE 4		(later)	
132.	RHR Pumps Inside Containment	Riggs	RES/DRA/ARGIB	NOTE 4		(later)	
133.	Update Policy Statement on Nuclear Plant Staff Working Hours	Pittman	NRR/DLPQ/LHFB	LI (NOTE 5)		12/31/87	NA
134.	Rule on Degree and Experience Requirement	Pittman	RES/DRA/RDB	HIGH		12/31/87	
135.	Integrated Steam Generator Issues	Emrit	RES/DE/EIB	MEDIUM		12/31/87	
136.	Storage and Use of Large Quantities of Cryogenic Combustibles	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
137.	Refueling Cavity Seal Failure	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
138.	Deinerting Upon Discovery of RCS Leakage	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
139.	Thinning of Carbon Steel Piping in LWRs	Riggs	RES/DRA/ARGIB	NOTE 4		(later)	
140.	Fission Product Removal by Containment Sprays	Riggs	RES/DRA/ARGIB	NOTE 4		(later)	
141.	LBLOCA with Consequential SGTR	Riggs	RES/DRA/ARGIB	NOTE 4		(later)	
142.	Leakage Through Electrical Isolators	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
143.	Availability of Chilled Water Systems	Milstead	RES/DRA/ARGIB	NOTE 4		(later)	
	HUMAN FACTORS ISSUES						
HF1	STAFFING AND QUALIFICATIONS						
HF1.1	Shift Staffing	Pittman	RES/DRPS/RHFB	HIGH	1	12/31/86	
HF1.2	Engineering Expertise on Shift	Pittman	NRR/DHFT/HFIB	NOTE 3(b)	1	12/31/86	
HF1.3	Guidance on Limits and Conditions of Shift Work	Pittman	NRR/DHFT/HFIB	NOTE 3(b)	1	12/31/86	
HF2	TRAINING						
HF2.1	Evaluate Industry Training	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF2.2	Evaluate INPO Accreditation	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF2.3	Revise SRP Section 13.2	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF3	OPERATOR LICENSING EXAMINATIONS						
HF3.1	Develop Job Knowledge Catalog	Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF3.2	Develop License Examination Handbook	Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF3.3	Develop Criteria for Nuclear Power Plant Simulators	Pittman	NRR/DHFT/HFIB	I.A.4.2(4)	2	12/31/87	NA
HF3.4	Examination Requirements	Pittman	NRR/DHFT/HFIB	I.A.2.6(1)	2	12/31/87	NA
HF3.5	Develop Computerized Exam System	Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF4	PROCEDURES						
HF4.1	Inspection Procedure for Upgraded Emergency Operating Procedures	Pittman	NRR/DLPQ/LHFB	HIGH	1	12/31/86	
HF4.2	Procedures Generation Package Effectiveness Evaluation	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF4.3	Criteria for Safety-Related Operator Actions	Pittman	NRR/DHFT/HFIB	B-17	1	12/31/86	NA
HF4.4	Guidelines for Upgrading Other Procedures	Pittman	RES/DRPS/RHFB	HIGH	1	12/31/86	
HF4.5	Application of Automation and Artificial Intelligence	Pittman	NRR/DHFT/HFIB	HF5.2	1	12/31/86	NA
HF5	MAN-MACHINE INTERFACE						
HF5.1	Local Control Stations	Pittman	RES/DRPS/RHFB	HIGH	1	12/31/86	
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	Pittman	RES/DRPS/RHFB	HIGH	1	12/31/86	

TABLE II (Continued)

Action Plan Item/ Issue No.	Title	Priority Evaluation Engineer	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date	MPA No.
HF5.3	Evaluation of Operational Aid Systems	Pittman	NRR/DHFT/HFIB	HF5.2	1	12/31/86	NA
HF5.4	Computers and Computer Displays	Pittman	NRR/DHFT/HFIB	HF5.2	1	12/31/86	NA
HF6	MANAGEMENT AND ORGANIZATION						
HF6.1	Develop Regulatory Position on Management and Organization	Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF6.2	Regulatory Position on Management and Organization at Operating Reactors	Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF7	HUMAN RELIABILITY						
HF7.1	Human Error Data Acquisition	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.2	Human Error Data Storage and Retrieval	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.3	Reliability Evaluation Specialist Aids	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.4	Safety Event Analysis Results Applications	Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF8	Maintenance and Surveillance Program	Pittman	NRR/DLPQ/LPEB	HIGH	1	12/31/86	NA

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TABLE III

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

Legend

NOTES:
1 - Possible Resolution Identified for Evaluation
2 - Resolution Available
3 - Resolution Resulted in either the Establishment of New Requirements or No New Requirements
4 - Issues to be Prioritized in the Future
5 - Issue that is not a Generic Safety Issue but should be Assigned Resources for Completion

HIGH	- High Safety Priority
MEDIUM	- Medium Safety Priority
LOW	- Low Safety Priority
DROP	- Issue Dropped as a Generic Issue
USI	- Unresolved Safety Issue
I	- TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737

TABLE III (Continued)

ACTION ITEM/ISSUE GROUP	I	COVERED IN OTHER ISSUES	RESOLVED STAGES NOTE 1	NOTE 2	NOTE 3	USI	HIGH	MEDIUM	LOW	DROP	NOTE 4	NOTE 5	TOTAL
1. TMI ACTION PLAN ITEMS (369)													
(a) Safety													
(i) Generic Safety	88	46	1	1	121	0	7	6	12	7	2	-	291
(b) Non-Safety													
(i) Licensing	-	0	-	-	73	-	-	-	-	0	0	5	78
2. TASK ACTION PLAN ITEMS (142)													
(a) Safety													
(i) USI	-	-	0	1	17	9	-	-	-	-	-	-	27
(ii) Generic Safety	-	19	0	1	28	-	2	5	3	9	7	-	74
(iii) Regulatory Impact	-	0	0	0	5	-	-	-	1	0	0	1	7
(b) Non-Safety													
(i) Licensing	-	0	0	0	1	-	-	-	-	7	0	11	19
(ii) Environmental	-	1	0	0	6	-	-	-	-	6	0	2	15
3. NEW GENERIC ISSUES (194)													
(a) Safety													
(i) Generic Safety	-	45	6	2	20	0	16	6	8	36	40	-	179
(ii) Regulatory Impact	-	2	0	0	1	-	-	-	1	0	0	6	10
(b) Non-Safety													
(i) Licensing	-	0	0	0	0	-	-	-	-	0	0	5	5
4. HUMAN FACTORS ISSUES (27)													
(a) Safety													
(i) Generic Safety	-	8	0	0	2	0	6	0	0	0	0	-	16
(b) Non-Safety													
(i) Licensing	-	0	0	0	3	-	-	-	-	-	0	8	11
TOTAL	88	121	7	5	277	9	31	17	25	65	49	38	732

TABLE IV

LISTING OF AEOD REPORTS AND RELATED GENERIC ISSUES

This listing shows all AEOD reports that have been addressed either as completely new safety issues or as part of new or existing safety issues. It should be noted that, in some cases, more than one AEOD report has been generated on a single topic. However, all AEOD reports related to the identified safety issues are listed alphanumerically including those that have been superseded by other AEOD reports. The following is a description of the types of AEOD reports:

C - Reactor Case Study
E - Reactor Engineering Evaluation
S - Special Study Report
T - Technical Review Report

AEOD Report No.	AEOD Report Title	Related Safety Issue No.	Related AEOD Report
C001	Report on the Browns Ferry 3 Partial Failure to Scram Event on June 28, 1980	41	-
C003	Report on Loss of Offsite Power Event at Arkansas Nuclear One, Units 1 and 2	47	-
C004	AEOD Actions Concerning the Crystal River 3 Loss of Non-Nuclear Instrumentation and Integrated Control System Power on February 26, 1980	33	E122
C005	AEOD Observations and Recommendations Concerning the Problem of Steam Generator Overfill and Combined Primary and Secondary Side Blowdown	37, 42	-
C101	Report on the Saint Lucie 1 Natural Circulation Cooldown on June 11, 1980	31	-
C102	H. B. Robinson Reactor Coolant System Leak on January 29, 1981	34	-
C103	AEOD Safety Concerns Associated with Pipe Breaks in the BWR Scram System	40	-
C104	Millstone Unit 2 Loss of 125 V DC Bus Event on January 2, 1981	46	-
C105	Report on the Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980	36	-
C201	Safety Concern Associated with Reactor Vessel Level Instrumentation in Boiling Water Reactors	50, 101	-

TABLE IV (Continued)

AEOD Report No.	AEOD Report Title	Related Safety Issue No.	Related AEOD Report
C202	Report on Service Water System Flow Blockages by Bivalve Mollusks at Arkansas Nuclear One and Brunswick	32	E016
C203	Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980	54	E305
C204	San Onofre Unit 1 Loss of Salt Water Cooling Event of March 10, 1980	44	-
C205	Abnormal Transient Operating Guidelines (ATOG) as Applied to the April 1981 Overfill Event at Arkansas Nuclear One, Unit 1	56	-
C301	Failures of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand	55	-
C401	Low Temperature Overpressure Events at Turkey Point Unit 4	94	E426
.403	Edwin I. Hatch Unit No. 2 Plant Systems Interaction Event on August 25, 1982	85	E322
C404	Steam Binding of Auxiliary Feedwater Pumps	93	E325
C501	Safety Implications Associated With In-Plant Pressurized Gas Storage and Distribution Systems in Nuclear Power Plants	106	-
C503	Decay Heat Removal Problems at U.S. Pressurized Water Reactors	99	-
C701	Air Systems Reliability	43	E123
E002	BWR Jet Pump Integrity	12	-
E005	Operational Restrictions for Class 1E 120 VAC Vital Instrument Buses	48	-
E007	Potential for Unacceptable Interaction Between the Control Rod Drive System and Non-Essential Control Air System at the Browns Ferry Plant	39	-
E010	Tie Breaker Between Redundant Class 1E Buses - Point Beach Nuclear Plant, Units 1 and 2	49	-
E011	Concerns Relating to the Integrity of a Polymer Coating for Surfaces Inside Containment	38	-
E016	Flow Blockage in Essential Equipment at ANO Caused by Corbicula sp. (Asiatic Clams)	32	C202
E101	Degradation of Internal Appurtenances in LWR Piping	35	-
E112	Inoperability of Instrumentation Due to Extreme Cold Weather	45	E226
E122	AEOD Concern Regarding Inadvertent Opening of Atmospheric Dump Valves on B&W Plants During Loss of ICS/NNI Power	33	C004
E123	Common Cause Failure Potential at Rancho Seco - Desiccant Contamination of Air Lines	43	C701
E204	Effects of Fire Protection System Actuation on Safety-Related Equipment	57	-

TABLE IV (Continued)

AEOD Report No.	AEOD Report Title	Related Safety Issue No.	Related AEOD Report
E209	Generator Rotor Retaining Ring as a Potential Missile (Incident at Barseback 1 on 4/13/79)	30	-
E215	Engineering Evaluation of the Salt Service Water System Flow Blockage at the Pilgrim Nuclear Power Station by Blue Mussels	52	-
E226	Inoperability of Instrumentation Due to Extreme Cold Weather	45	E112
E304	Investigation of Backflow Protection in Common Equipment and Floor Drain Systems to Prevent Flooding of Vital Equipment in Safety-Related Compartments	77	-
E305	Inoperable Motor-Operated Valve Assemblies Due to Premature Degradation of Motors and/or Improper Limit Switch/Torque Switch Adjustment	54	C203
E322	Damage to Vacuum Breaker Valves as a Result of Relief Valve Lifting	85	C403
E325	Vapor Binding of Auxiliary Feedwater Pumps at Robinson 2	93	C404
E414	Stuck Open Isolation Check Valve on the Residual Heat Removal System at Hatch Unit 2	105	-
E417	Loosening of Flange Bolts on RHR Heat Exchanger Leading to Primary to Secondary Side Leakage	C-9	-
E426	Single Failure Vulnerability of Power Operated Relief Valve (PORV) Actuation Circuitry for Low Temperature Overpressure Protection (LTOP)	94	C401
S401	Human Error in Events Involving Wrong Unit or Wrong Train	102	-
T302	Postulated Loss of Auxiliary Feedwater System Resulting from a Turbine Driven Auxiliary Feedwater Pump Steam Supply Line Rupture	68	-
T305	Flow Blockage in Essential Raw Cooling Water System Due to Asiatic Clam Instrusion at Sequoyah 1	51	-
T420	Failure of an Isolation Valve of the Reactor Core Isolation Cooling System to Open Against Operating Reactor Pressure	87	-

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TABLE V

SUMMARY OF CONSOLIDATED GENERIC ISSUES

This table shows the consolidation of those issues whose technical concerns were found to be addressed either partially or completely in other (major) issues. The table reflects the findings of the prioritization process that are summarized in Table II.

Major Item/Issue No.	Priority	Item(s)/Issue(s) Covered in Major Issues				
TMI ACTION PLAN ITEMS						
I.A.1.3	I	II.K.3(53)				
I.A.2.2.	NOTE 3(b)	I.A.2.6(3) [II.K.3(56)]				
I.A.2.6(1)	HIGH	I.B.1.1(6),	I.B.1.1(7),	HF3.4		
I.A.3.1	I	II.K.3(56)				
I.A.4.1(2)	NOTE 3(a)	II.K.3(54)				
I.A.4.2(4)	HIGH	HF3.3				
I.B.1.1 (1,2,3,4)	NOTE 3(b)	II.J.3.1,	II.J.3.2,	II.K.3(52),	HF6.1,	HF6.2
I.C.1		8, 67.4.3,	18, 67.9.0	31,	42,	67.3.1,
I.C.1(2)	I	37				
I.C.1(3)	I	II.K.2(12), II.K.3(37), II.K.3(43),	II.K.2(18), II.K.3(38), II.K.3(55),	II.K.3(6), II.K.3(39), 37	II.K.3(35), II.K.3(41),	II.K.3(36), II.K.3(42),
I.C.2	I	II.K.3(52)				
I.C.5	I	II.K.3(52)				
I.C.7	I	II.K.3(50)				
I.C.8	I	II.K.3(49)				
I.C.9	I	II.K.3(49),	II.K.3(50)	II.K.3(51)		
I.D.1	I	56,	67.4.2			
I.D.2	I	II.K.3(23),	II.K.3(55)			

TABLE V (cont)

Major Item/Issue No.	Priority	Item(s)/Issue(s) Covered in Major Issues			
I.D.3	MEDIUM	II.K.3(55)			
I.F.1	HIGH	5			
II.B.8	NOTE 3(a)	II.B.7			
II.C.1	NOTE 3(b)	II.K.3(4),	II.K.3(8),	II.K.3(33),	II.K.3(48)
II.C.2	NOTE 3(b)	II.K.3(4),	II.K.3(48)		
II.E.1.1	I	II.K.2(8)			
II.E.1.2	I	II.K.2(8)			
II.E.2.2	NOTE 3(b)	II.K.3(32),	II.K.3(34),	II.K.3(47)	
II.E.6.1	MEDIUM	54			
II.F.2	I	II.K.3(6),	67.3.4		
II.F.3	NOTE 3(a)	II.K.3(6),	A-34		
II.H.2	HIGH	II.H.3			
II.K.2(15)	I	II.K.3(43)			
II.K.2(16)	I	II.K.3(40)			
II.K.3(5)	I	9,	67.4.1		
II.K.3(17)	I	II.E.2.1[II.K.3(26)]			
III.A.1.2(1)	I	II.K.3(23)			
III.A.3.1	NOTE 3(b)	B-71			
III.A.3	NOTE 3	67.6.0			
III.A.3.4	NOTE 3(b)	II.K.3(23)			
III.D.1.1(1)	I	B-69			
III.D.2.1	LOW	B-67			
III.D.2.5	NOTE 3(b)	III.D.2.2(2),	III.D.2.2(3),	III.D.2.2(4)	
III.D.3.1	NOTE 3(b)	B-34,	97		

TABLE V (cont)

Major Item/Issue No.	Priority	Item(s)/Issue(s) Covered in Major Issues
V.A.1	LI (NOTE 3)	II.A.2
TASK ACTION PLAN ITEMS		
A-2	USI [NOTE 3(a)]	B-52
A-12	USI [NOTE 2]	60
A-17	USI	II.C.3[II.K.3(4)], C-13, 77
A-18	DROP	B-16
A-37	DROP	A-32, 11
A-38	LOW	A-32
A-40	USI	B-51
A-43	USI [NOTE 3(b)]	B-18, C-3
A-44	USI	B-57
A-45	USI	II.E.3.2[B-4], II.E.3.3[II.K.3(8)], II.E.3.5, 67.9.0, 125.I.2.d
A-46	USI	B-24 114
A-47	USI	19, 33, 37, 56, 67.3.1
A-48	USI	B-14
A-49	USI	28, 67.3.2
B-2	E (NOTE 3)	B-45
B-17	MEDIUM	27, HF4.3
B-68	DROP	A-32
C-8	HIGH	16
C-12	NOTE 3(b)	B-73
NEW GENERIC ISSUES		
17	DROP	26

TABLE V (cont)

Major Item/Issue No.	Priority	Item(s)/Issue(s) Covered in Major Issues				
23	HIGH	65				
25	NOTE 3(a)	39				
43	DROP	44				
51	MEDIUM	32,	52			
67.5.2	LI	36				
70	MEDIUM	125.I.2.a,	125.I.2.b			
75	NOTE 1	I.B.1.1(6),	I.B.1.1(7)			
76	NOTE 4	46				
119.1	RI	B-6				
124	NOTE 1	68,	122.1.a,	122.1.b,	122.1.c,	125.II.1.b
128	HIGH	48,	49,	A-30		
135	MEDIUM	67.7				
HUMAN FACTORS ISSUES						
HF5.2	HIGH	HF4.5,	HF5.3,	HF5.4		

TASK I.A.2: TRAINING AND QUALIFICATIONS OF OPERATING PERSONNEL

The objectives of this task are as follows: (1) to improve the capability of operators and supervisors to understand and control complex reactor transients and accidents, (2) to improve the general capability of an operations organization to respond rapidly and effectively to upset conditions, and (3) to increase the education, experience, and training requirements for operators, senior operators, supervisors, and other personnel in the operations organization to substantially improve their capability to perform their duties.

ITEM I.A.2.1: IMMEDIATE UPGRADING OF OPERATOR AND SENIOR OPERATOR TRAINING AND QUALIFICATIONS

This item required all operating plant licensees and all licensee applicants to provide specific improvements in training and qualifications of senior operators and control room operators. The three parts of this item are listed below.

ITEM I.A.2.1(1): QUALIFICATIONS - EXPERIENCE

DESCRIPTION

This NUREG-0660[48] item set specific experience requirements that were to be met by applicants for senior operator licenses by May 1, 1980. Applicants for senior operator licenses were required to have been a licensed operator for one year effective December 1, 1980.

CONCLUSION

This item was clarified in NUREG-0737,[98] new requirements were established and MPA F-03 was established by DL for implementation purposes.

ITEM I.A.2.1(2): TRAINING

DESCRIPTION

This NUREG-0660[48] item set the following specific requirements:

(1) Effective August 1, 1980, senior operator applicants were required to have 3 months of continuous on-the-job training as an extra person on shift.
(2) Effective August 1, 1980, control room operator applicants were required to have 3 months training on shift as an extra person in the control room.
(3) Training programs were to be modified to provide: (a) training in heat transfer, fluid flow, and thermodynamics; (2) training in the

use of installed plant systems to control or mitigate an accident in which the core is severely damaged; and (3) increased emphasis on reactor and plant transients.

CONCLUSION

This item was clarified in NUREG-0737,[98] new requirements were established, and MPA F-03 was established by DL for implementation purposes.

ITEM I.A.2.1(3): FACILITY CERTIFICATION OF COMPETENCE AND FITNESS OF APPLICANTS FOR OPERATOR AND SENIOR OPERATOR LICENSES

DESCRIPTION

This NUREG-0660[48] item required all applicants for operator and senior operator licenses, pursuant to Sections 55.10(a)(6), 55.33(a)(4), and 55.33(a)(5) of 10 CFR 55, to be certified by the highest level of the corporate management of their respective plants. This requirement was effective May 1, 1980.

CONCLUSION

This item was clarified in NUREG-0737,[98] new requirements were established, and MPA F-03 was established by DL for implementation purposes.

ITEM I.A.2.2: TRAINING AND QUALIFICATIONS OF OPERATIONS PERSONNEL

DESCRIPTION

Under the TMI Action Plan,[48] the NRC may require reactor licensees to review their training and qualification programs for all operations personnel. This is interpreted to include licensed and auxiliary operators, technicians, maintenance personnel and supervisors. The review is to examine current practices in light of the safety significance of the duties of the operations staff. If the review determines that the current practices adequately assure proper safety-related staff conduct, then documentation of the justification for this determination is required. The documentation need not be submitted to the NRC but must be maintained on site. If the review uncovers inadequacies, the licensee is required to upgrade the training and qualification practices to ensure adequate performance of operations personnel. The evaluation of this issue includes the consideration of Item I.A.2.6(3).

PRIORITY DETERMINATION

The first step in estimating the effect of training reviews on operator-error contributions to plant risk was to assemble a panel of experts from the PNL staff. This panel represented considerable experience in reactor operations, utility training programs, and reactor plant systems. The panel included members with utility field experience and reactor operator licensing examiners.

The judgments of the panel, as detailed below, are based on the two following considerations:[64]

(1) The potential effect of this issue is limited by its semi-voluntary nature, i.e., the judgment of adequacy is in the hands of the individual utilities. Furthermore, the current INPO and NRC research work in task analysis deals with generic routine operations. Plant-specific operation and operation under upset conditions are left to the individual utilities. This dilutes the effectiveness of the task analysis efforts in providing the basis for the training and qualification review.

Related issues which are supported by and in turn support this issue are the conducting of plant drills and accreditation of training programs. While neither of these is directly required by the training and qualifications review, both could be a part of the response and both would have a positive effect on personnel performance.

(2) There is a wide variation among utilities in both the training programs and the performance of operations staff. In many facilities there is much room for improvement. Therefore, while the potential effect of the training and qualifications review effort is limited, a significant overall reduction in safety-related human error for operations personnel is expected because of the wide margin available for improvement.

Assumptions

In estimating the benefit and costs, the PNL panel divided licensees into three groups:

(1) Minimally-affected group: These utilities currently have a good effective training and qualification program and good operations personnel performance. They should be minimally affected by this safety issue. The fractional population of this group is estimated to be 15% of the reactor licensees.

(2) Intermediately-affected group: These utilities' training and qualification programs and/or operations performance have room for improvement. This group, estimated to be 60% of the population, would undergo improvements and therefore be affected by the issue.

(3) Maximally-affected group: These utilities have deficiencies in their training and qualification programs and in operations personnel performance. They would be significantly affected by this safety issue and major restructuring of programs would be expected. This group is estimated to contain 25% of reactor licensees.

From the estimates for these groups, weighted composite estimates can be derived. NUREG/CR-2800[64] shows the safety benefit estimates from the panel for each of the groups and also gives the weighted averages.

The values given in NUREG/CR-2800[64] are in terms of percent changes. For inclusion into the value/impact score formula, they must be converted to other measures. The reduction in human error must be transformed into the resulting reduction in risk as measured by change in probabilistic exposure (man-rem/RY). The change in annual ORE must be transformed from percent improvement into man-rem/RY.

The reduction in risk will be developed by examining the quantitative impact on accident event frequencies of human error rates in key scenarios. The reduction in human error will thereby be translated into a reduction in accident frequency. No additional reduction due to accident mitigation will be assumed. The values given in NUREG/CR-2800[64] will be used for the best estimate of improvement: 17% for operator error and 28% for maintenance.

Frequency/Consequence Estimate

This issue centers around operator and maintenance training programs to improve personnel performance. This issue relates generically to both BWRs and PWRs, and ideally the risk reduction attributable to its resolution would be estimated by selecting a representative plant of each type. However, maintenance and operator performance impact essentially accident sequences in the risk equations. To save time, the calculations were performed for one representative PWR and inferences drawn for all reactors. The Oconee 3 (a RSSMAP PWR) plant risk equations developed in NUREG/CR-1659,[54] Vol. 4 (Hatch 1981) were used for this analysis.

It will be assumed that the 17% reduction in operator error can be applied directly to elements containing an operator error frequency and the 28% reduction can be applied directly to maintenance variables. This assumption introduces some error in the maintenance contribution. This is because some maintenance operations on nuclear systems have fixed times associated with cooldown and preparation, etc., in addition to the actual hands-on time for maintenance that would be subject to improvement through training. Maintenance done properly the first time also reduces the frequency of maintenance outage and downtime for proper repairs at some future date. Thus, fixed time periods in maintenance outages are indirectly reduced over the long run with improved maintenance performance simply because the need for maintenance may be reduced except for systems that undergo preventive maintenance at set intervals.

To calculate the total public risk reduction it was assumed that issue resolution would apply to all plants existing and planned as given in NUREG/CR-2800, Appendix C.[64] This would represent a grand total of 4,000 RY of operation (143 plants with an average life expectancy of 28 years). Implementation of the solution would provide a reduction of 9 man-rem/RY. For all plants, assuming a typical midwest-type meteorology and an average population density of U.S. reactor sites of 340 people per square mile, the total public risk reduction totals 122,400 man-rem.

Cost Estimate

Industry Cost: In estimating the costs to industry of implementing and operating under the resolution of this issue the PNL panel divided the industry once again into three categories. These groups and their estimates are shown in NUREG/CR-2800.[64] The total costs to industry for implementation is the product of the number of plants and the per-plant cost, (143)($0.335M) = $48M. The total operation cost is the product of the number of plants, the average remaining life, and the plant annual cost, (143)(28)($0.16M) = $640M. The overall cost to industry is the sum of the total implementation and operational cost, $[640 + 48]M = $688M.

NRC Cost: The cost for the NRC to implement the safety issue resolution was taken from NUREG-0660.[48] This called for 1.1 man-years of NRC effort which is equivalent to $110,000. The annual NRC effort through OIE to review the justification documentation and new training programs is estimated to be one person-year. This is $100,000/year. Over the lifetime of the completed and planned reactors this is $2.8M. Therefore, the overall cost to the NRC is the sum of the implementation and operation costs, $[0.11 + 2.8]M or $2.9M.

According to PNL estimates and calculations, the total cost for the implementation and operation of this safety issue is then $[688 + 2.9]M or approximately $691M.

Value/Impact Assessment

Based on an estimated public risk reduction of 122,400 man-rem, the value/impact is given by:

$$S = \frac{122{,}400 \text{ man-rem}}{\$691\text{M}}$$

$$= 177 \text{ man-rem/\$M}$$

Other Considerations

Including the occupational dose reduction (2.4×10^5 man-rem) in the value/impact equation gives a score of 524 man-rem/$M. PNL calculated[64] the occupational risk reduction for accident-related ORE to be 880 man-rem. However, it was estimated that with improved training the operational doses could be reduced by 2.4×10^5 man-rem for 143 plants over the average remaining plant lifetime.

CONCLUSION

Because of the extensive number of sequences considered to be affected by this issue, the base-case risk is high at a calculated range of from 60 to 73 man-rem/RY. Based on the potential reduction in public risk and ORE, this issue was determined to be high priority. However, in June 1985, the Commission recognized that the industry had made progress in developing programs to improve nuclear utility training and personnel qualification. As a result, the Commission adopted a Policy Statement on Training and Qualifications which made the training accreditation program managed by INPO the focus of training improvement in the industry.[777] Thus, this item was RESOLVED and no new requirements were established.

ITEM I.A.2.3: ADMINISTRATION OF TRAINING PROGRAMS

DESCRIPTION

This NUREG-0660[48] item required the staff to develop criteria and procedures to be used in auditing training programs, including those provided by reactor vendors, and to increase the amount of auditing. Specifically, NRR was to: (1) audit training programs to assure training is formalized and eventually in conformance with accreditation; (2) conduct cold operator licensing certification at simulators; and (3) pending accreditation, require certain instructors to be SRO-certified.

Elements (2) and (3) were implemented and have been incorporated into the Examiner Standards and Inspection Procedures. The issue of training audits was addressed by the Commission's Policy Statement on Training and Qualification of Nuclear Power Plant Personnel (50 FR 11147)[966] which endorsed the INPO-managed accreditation program.[956]

CONCLUSION

This item was clarified in NUREG-0737[98] and new requirements were established.

ITEM I.A.2.4: NRR PARTICIPATION IN INSPECTOR TRAINING

DESCRIPTION

Historical Background

Based on NUREG-0660,[48] NRR was required to provide supplemental instruction to the OIE inspectors by the licensing and human factors staff as an addition to the already established OIE inspector training program. The purpose of such instruction would be to focus the inspector's attention on problems associated with human factors. With such training it is expected that the inspectors would become more sensitive to such problems and hence more apt to instigate corrective action and thereby improve plant safety in this area. This would provide a means of responding to the TMI-related concern on human factors problems for plant operations staff.

Safety Significance

The principal safety benefit to be derived from NRR participation in OIE inspector training is in the improvements those inspectors will bring about because of that enhanced training. The training will increase inspector awareness in human factors and personnel-related problems. In areas such as emergency procedures reviews, routine operational practices and hardware-to-human interface deficiencies may be found by inspectors and corrected. A panel of PNL experts explored the potential significance of this issue.[64] This panel included three reactor operator license examiners, members with utility field experience, experience in training as well as general reactor safety experience.

The panel envisioned that the solution of this issue would be the addition of one week of instruction in human factors to the OIE inspector training course. The staff from NRR would participate in the instruction but would probably rely on a qualified consultant to conduct the majority of the instruction. It was assumed that the principal target of the training would be the resident inspectors. The potential effect of the training upon the OIE review of emergency procedures, plant hardware and routine practices could be significant, but the overall effect is thought to be limited because of two factors: the short exposure of the inspector to human factors training, and the indirect nature of the safety benefit. That is, a marginal improvement in inspector awareness will result in some corrective actions which would result in some safety improvement. The separation between initial action and the safety benefit complicates assessment of the effectiveness of the proposed resolution of the issue.

PNL estimated[64] a human-error rate reduction of 2% for operators and maintenance personnel (operations staff assumed most likely to affect plant safety). It is important to note that this is an overall industry-wide estimate. Some isolated actions could be highly significant. The PNL estimated cost for this additional training is about $1,000.

Capabilities of inspectors could clearly be improved through the proposed training. There would be an indirect effect on risk, since better trained inspectors would identify more cost-effective improvements in plant operations. However, there is no reasonable way that the magnitude of the safety significance and cost of these improvements can be estimated quantitatively. This additional training would enhance the capabilities and thus contribute to the effectiveness and efficiency of the NRC in performing its regulatory safety mission. Thus, this training proposal was determined to be a Licensing Issue.

CONCLUSION

This Licensing Issue was resolved in September 1983 with the regionalization of the operator licensing function which provided for training and guidance of the regional operator licensing personnel.[956]

ITEM I.A.2.5: PLANT DRILLS

DESCRIPTION

The intent of this TMI Action Plan item is to upgrade operator training by requiring operating personnel to conduct plant drills during shifts. Normal and off-normal operating maneuvers would be simulated for walk-through drills on a plant-wide basis. Drills would also be required to test the adequacy of reactor and plant operating procedures.

This is an effort to reduce the risk of off-normal operating conditions by improving the capability of operators and supervisors to understand and control complex reactor transients and accidents, and also to improve the general capability of an operations organization to respond rapidly and effectively to upset conditions.

PRIORITY DETERMINATION

Assumptions

Assume that the frequency of a core-melt incident is 5×10^{-5}/RY based on WASH-1400.[16] Also, assume that operator error accounts for 50% of these events, but that the plant drills will improve operator performance by 2%. In addition, assume that the release associated with core-melt is the value averaged over the probabilities of the WASH-1400[16] accident categories for PWRs and BWRs and weighted by the number of PWRs (95) and BWRs (48). This results in a total of 2.4×10^6 man-rem/accident. The remaining average plant lifetime is assumed to be 28 years.

Frequency/Consequence Estimate

Based on the assumptions above, the reduction in the core-melt frequency resulting from the plant drills is calculated to be $(0.02)(0.50)(5 \times 10^{-5})$/RY or 5×10^{-7}/RY.

$$\text{Risk Reduction} = (5 \times 10^{-7}/\text{RY})(2.4 \times 10^{6} \text{ man-rem})(28 \text{ years})(143 \text{ reactors})$$
$$= 4{,}805 \text{ man-rem}$$

Cost Estimate

Industry Cost: The industry resources required for implementation are estimated to be one person-month per plant. This is the estimated personnel requirement associated with the utility staff time for attendance at the drill, preparation by staff and management, and staff time dedicated to the dissemination of insights gained from the drills. At a cost of $100,000/man-year and with 4.33 weeks/month, this yields a cost of $8,333/plant. Across the industry, i.e., 143 plants, this would be $1.2M.

The industry resources required annually to participate in the plant drills are estimated to be 2 man-months/plant, which includes drill attendance, preparation before the drill, and dissemination of information afterward. This would be equivalent to $16,660/RY. For the total industry (143 plants), this works out to an estimated 143 man-months/year or $2.38M/year. Given the average remaining lifetime for the plants (28 years), this gives a total operational cost of $67M.

The total cost to industry is then the sum of the implementation and operational costs, $(1.19 + 67)M or approximately $68.2M.

NRC Cost: The total costs to the NRC to implement the resolution of this issue includes NRC staff labor and services of a contractor. Since the activities of the NRC staff and the contractor are to some degree interchangeable, no attempt was made to provide separate estimates so that the total implementation cost is estimated to be $300,000. The annual cost to the NRC was also estimated to be $300,000. Again, this was assumed to contain some mixture of staff and contractor expenses. Over the average remaining life (28 years), the operational cost comes to $8.4M. Therefore, the total cost to the NRC is the sum of implementation and operation costs, $(8.4 + 0.3)M or $8.7M.

Hence, the total costs associated with this issue are $(68.2 + 8.7)M or $76.9M.

Value/Impact Assessment

Based on a public risk reduction of 4,805 man-rem, the value/impact score is given by:

$$S = \frac{4{,}805 \text{ man-rem}}{\$76.9\text{M}}$$

$$= 62 \text{ man-rem/\$M}$$

CONCLUSION

Based on the above value/impact score, the ranking of this issue would be low to medium. Because the risk may have been estimated to be well on the conservative side, the issue was given a low priority ranking. However, ongoing work by DHFS on the subject was completed in July 1985 and published for information only as NUREG/CR-4258.[800] Thus, this item was RESOLVED and no new requirements were established.[801]

ITEM I.A.2.6: LONG-TERM UPGRADING OF TRAINING AND QUALIFICATIONS

ITEM I.A.2.6(1): REVISE REGULATORY GUIDE 1.8

Items I.A.2.6(1), I.A.2.6(2), I.A.2.6(3), and I.A.2.6.(5) have been combined and evaluated together.

DESCRIPTION

Historical Background

Item I.A.2.6 of the TMI Action Plan[48] calls for the long-term upgrading of training and qualifications for operations personnel. The specific paragraphs of this item in NUREG-0660[48] called for a revision of Regulatory Guide 1.8,[226] (ANSI/ANS 3.1),[253] in order to incorporate short-term requirements into this issue and any other changes resulting from a national standards effort. Also, it is stated that more explicit guidance regarding exercises in simulator requalification programs will be included in the regulatory guide (Recommendation 8 of SECY-79-330E[251]) as will qualifications of shift supervisors and senior reactor operators [NUREG-0585,[174] Recommendations 1.6(1) and (2)]. In addition, based on the NRC staff review of NRR 80-117,[252] recommendations will be made to the Commission and Commission decisions will be factored into the regulatory guide or regulation changes. Moreover, appropriate revisions to 10 CFR 55, Operator Licenses, are to be recommended for action by the Commission in order to incorporate the applicable short-term changes plus requirements based on Commission action on SECY-79-330E[251] for mandatory simulator training for applicants for licenses (Recommendation 4); mandatory simulator training in requalification programs (Recommendation 7); NRC administration of requalification examinations (Recommendation 9 as modified by the Commission); and mandatory operating tests at simulators (Recommendation 11). Finally, it is noted that the Nuclear Waste Policy Act of 1982, Public Law 97-425, Section 306 authorized and directed NRC to promulgate regulations or guidance for the training and qualifications of civilian nuclear power plant personnel. A task force has been formed within NRC as a result of this bill. As part of the task force objectives, Items I.A.2.6 (1, 2, and 3) are to be addressed.

The numerical assessment of this safety issue was conducted by the PNL staff[64] with experience in reactor operator licensing, reactor operation, and general reactor safety in consultation with General Physics Corporation. General Physics Corporation provides utility training services and has significant experience in reactor simulators, providing procurement and startup assistance, operation and maintenance services, and simulator modifications.

Safety Significance

A public risk reduction is anticipated as a result of a reduction in core-melt frequency which follows from a reduction in operator error rates. Reduction in operator errors is expected to result from the upgraded training and qualifications which form the assumed resolution of this safety issue.

Possible Solutions

The upgrades are assumed to include an increase in time spent in simulator operation both in training and in requalification. The simulator time is assumed to improve in quality as well as quantity. Emphasis on improvements on the operators' diagnostic capability is felt to be especially important in contributing to a reduction in core-melt frequency. Furthermore, the enforcement activities in term of NRC-administered examinations and OIE inspection of training programs is likely to emphasize the value of this long-term training and qualification of reactor operators.

PRIORITY DETERMINATION

Assumptions

It is assumed that the resolution of this safety issue will take the form of upgrading utility training and qualification programs that will represent a major enhancement of the training and qualification programs.

It is noted that many of the TMI Action Plan Items associated with operator training are interrelated and it is, therefore, difficult to assess them independently. For example, this issue is related to I.A.4.1, Initial Simulator Improvement, which deals with the improvement of simulators and provides for more realistic modeling of the plant whereas this issue, [I.A.2.6(1,2,3,5)], deals with training improvements, including the enhanced use of existing simulators. Either issue, by itself, would improve operator performance. However, there may be significant overlaps in improving operator performance if both items were implemented. Even though it is recognized that the total improvement would be less than the sum of the individual contributions when each is assessed separately, the extent of any overlap is not identified here.

Based on engineering judgment, it was estimated by the PNL panel that the resolution of this safety issue would result in a 30% reduction in operator error rates. The number of plants to which this issue is applicable is assumed to be 95 PWRs and 49 BWRs with average lifetimes of 28.5 years and 27 years respectively.

For the analysis performed by PNL,[64] Oconee-3 is taken as the representative PWR plant. It is assumed that the fractional risk and core-melt frequency reductions for the representative BWR (Grand Gulf) will be equivalent to those for the representative PWR. Therefore, the analysis is conducted only for the PWR but the fractional risk and core-melt frequency reductions are also applied to the BWR. The dose calculations are based on a reactor site population density of 340 people per square mile and a typical mid-west meteorology is assumed.

Frequency/Consequence Estimate

Based on the affected accident sequences and the parameters affected by this safety issue resolution (SIR), the original core-melt frequencies of 8.2×10^{-5}/RY for PWRs and 3.71×10^{-5}/RY for BWRs are calculated to be reduced by about 16%. The associated reduction in public risk is 31 man-rem/RY for PWRs and 37.4 man-rem/RY for BWRs resulting in a total public risk reduction of 132,600 man-rem.

Cost Estimate

Industry Cost: The resolution of this safety issue was assumed to be a major enhancement of the training and qualification programs. The programs would have to be upgraded in order to meet the requirements of INPO accreditation. These requirements are assumed to be far-reaching and require significant effort on the part of utility training staffs. The amount of effort will vary among the utilities, depending on the present state of their programs. The effort required to implement the program is estimated by the PNL panel to require 10 to 20 man-years of effort for each plant. The mean value is expected to be shifted toward the lower end since many utilities are currently improving their training programs. A 12 man-year effort is taken as the central estimate.

Operation under the upgraded programs would require enhanced training activities and more operator time in training. The training staff is estimated to require three additional people. It is assumed the major cost of additional operator time can be estimated from increased time at simulators. It is estimated that 40 hours of simulator time will be added to operator training and requalification. For 20 operators per year passing through these programs, this is equivalent to 800 additional hours. It is further assumed that operators can be trained three at a time on the simulator and that simulator time can be acquired for \$600/hour. This gives an additional simulator cost of \$160,000/year. The industry costs are estimated as follows:

(1) Implementation of the SIR

(12 man-yrs/plant) (49 + 95) plants (\$100,000/man-yr) = \$173M

(2) Operation and Maintenance of the SIR

(a) Labor

Training Staff = (3 man-yr/RY) (52 man-weeks/man-year)
= 156 man-weeks/RY

Operators = (800 man-hr/RY/(40 man-hours/man-week)
= 20 man-wk/RY

Thus, the total labor is 176 man-wk/RY.

(b) Simulator Time (Operators)

(800 man-hours/RY)/(3 man-hours/simulator-hr) = 267 simulator-hr/RY

Therefore, the industry cost per plant-year for operation and maintenance is given by:

$$\left[\frac{176\text{ man-wk}}{RY}\right]\left[\frac{\$100{,}000/\text{man-yr}}{52\text{ man-wk/man-yr}}\right] + \left[\frac{267\text{ simulator-hr}}{RY}\right]\left[\frac{\$600}{\text{simulator-hr}}\right]$$

$$= 500{,}000/RY$$

Therefore, for all affected plants, the total industry cost for operation and maintenance is given by:

$$(\$500{,}000/RY)\ [(49)(27) + (95)(28.5)]\ RY = \$2{,}000M$$

The total industry cost for implementation, operation, and maintenance of the solution is then $(173 + 2,000)M or $2,173M.

<u>NRC Cost</u>: The NRC effort to implement the resolution of this issue would be significant. It is estimated in NUREG-0660[48] that 5.4 man-years plus $259,000 would be required. Some of these development activities have been completed. However, much work remains to be done. The remaining effort is estimated to be 4.5 man-years and $100,000.

The operational activities of the NRC would include reviews of training programs, increase inspection and additional examination. The annual labor for reviews and inspections is estimated to be equivalent to 3 person-years. The principal addition in examinations is assumed to be NRC conduct of a portion of requalification examinations. It is assumed the NRC will conduct 25% of the requalification examinations and the 20 operators are requalified at each plant every year. It is estimated that one person-month is required for each plant. This assumes the five (25% of 20) operators selected for NRC examination at each plant are tested at the same time. NRC costs are estimated as follows:

(1) <u>Implementation of the SIR</u>

Staff Labor + Other Costs
= (1.4 man-wk/plant)($1,600/man-wk) + ($100,000/144 plants)
= $3,386/plant

Total cost for all affected plants is ($3,386/plant)(144 plants) or $488,000.

(2) <u>Review of Maintenance and Operation of SIR</u>

(a) Review and Inspection = (3 man-yr/yr)(52 man-wk/man-yr)/144 plants
= 1.08 man-wk/RY

(b) Examination = (1 man-month/RY)(3.7 man-wk/man-month)
= 3.7 man-wk/RY

Thus, the total time spent is 4.78 man-wk/RY.

The NRC cost per plant-yr due to review of operation and maintenance is (4.78 man-wk/RY)($1,900/man-wk) = $9,088/RY.

The total NRC cost for operation and maintenance of the SIR is then
($9,088)[(49)(27) + (95)(28.5)] = ($9,088)(4,030) = $36.6M

Therefore, the total industry and NRC costs are estimated to be $[2,173 + 0.488 + 36.6]M = $2,210M.

Value/Impact Assessment

Based on the estimated public risk reduction of 132,600 man-rem, the value/impact score is given by:

$$S = \frac{132{,}600 \text{ man-rem}}{\$2{,}210M}$$

$$= 60 \text{ man-rem/\$M}$$

Other Considerations

The total occupational risk reduction is associated only with accident avoidance inasmuch as there is no dose associated with implementation or maintenance of this SIR. With a dose of 20,000 man-rem associated with accident cleanup and with the calculated reductions in core-melt frequencies of 1.3×10^{-5}/RY and 5.9×10^{-5}/RY for PWRs and BWRs, respectively, the total occupational dose reduction is calculated to be 860 man-rem.

CONCLUSION

Although the value/impact score was low, this issue was determined to be high priority because of the large potential public risk reduction. Resolution of this issue included the consideration of Items I.B.1.1(6,7) regarding changes to Regulatory Guide 1.8.[226]

In November 1986, SECY-86-348[1043] was submitted to the Commission with recommended revisions to Regulatory Guide 1.8[226] to endorse ANSI/ANS 3.1-1981 for the positions of shift supervisor, senior operator, licensed operator, shift technical advisor, and radiation protection manager. These revisions to Regulatory Guide 1.8[226] were subsequently approved by the Commission and published in May 1987.[1044] Thus, this issue was RESOLVED and new requirements were established.[1045]

ITEM I.A.2.6(2): STAFF REVIEW OF NRR 80-117

This item was evaluated in Item I.A.2.6(1) above and, in accordance with an RES memorandum,[437] was RESOLVED. No new requirements were established.

ITEM I.A.2.6(3): REVISE 10 CFR 55

This item was evaluated in Item I.A.2.6(1) above and, as a result of the Nuclear Waste Policy Act of 1982 (Public Law 97-425), was determined to be covered in Item I.A.2.2.[438]

ITEM I.A.2.6(4): OPERATOR WORKSHOPS

DESCRIPTION

Historical Background

On the basis of NUREG-0660,[48] NRR was required to develop a Commission paper on training workshops for licensed personnel. NUREG-0585,[174] the source of this safety issue, states that the intent of the issue is to conduct seminar-type workshops to exchange information on operations experience between the NRC and licensees and among licensees. This would assist in the improvement of operator performance and in improvements to reactor regulation, both resulting in improved safety. The proposed requirements would have one representative for each shift at each unit attend such a workshop annually.

Safety Significance

It is expected that there are two potential pathways to improved safety benefit emerging from this issue: (1) improved operator performance through the sharing of safety-related experiences and (2) the effect of improved regulation arising out of interaction between the operators and the NRC attending the workshops. The second pathway is considered to be a second-order effect and very difficult to quantify. Therefore, it was assumed that all the benefit would be derived through the reduction in operator-error rates.

PRIORITY DETERMINATION

Assumptions

PNL has conducted and is conducting a series of these workshops for NRR. In the assessment of this issue, PNL staff responsible for these workshops were consulted. Their judgments form the basis of our analysis.

This analysis assumes the major gains in reactor safety will come through the improvement in operator performance; that is, a reduction in their error rates. There is also a pathway to improve safety by means other than human performance through improved regulations developed from operator input at the workshops. The latter would be extremely difficult to quantify so that only the human error rate-reduction pathway to improved safety will be treated.

A panel of PNL experts was assembled and included staff that conduct operator licensing examinations, staff with experience in reactor operations, reactor safety and risk assessment, and the staff responsible for the conduct of the current operator feedback workshops. This panel produced the estimates that form the basis of this analysis.

The analysis is based on the following additional assumptions:

1. Applicable Plants: 95 PWRs and 48 BWRs

2. Selected Analysis Plant: Oconee 3 - representative PWR. It is assumed that the fractional risk and core-melt frequency reductions for

the representative BWR (Grand Gulf) will be equivalent to those for the representative PWR. Therefore, the analysis is conducted only for the PWR, but the fractional risk and core-melt frequency reductions are also applied to the BWR.

3. Affected Accident Sequences and Base-Case Frequencies: Most sequences are affected. The affected sequences and the base-case frequencies are shown in NUREG/CR-2800.[64]

4. Affected Release Categories and Base-Case Frequencies: All release categories are affected by issue resolution. The original base-case frequencies are used as given below.

Oconee	Grand Gulf
PWR-1 = 1.10×10^{-7}/RY	BWR-1 = 1.09×10^{-7}/RY
PWR-2 = 1.0×10^{-5}/RY	BWR-2 = 3.35×10^{-5}/RY
PWR-3 = 2.86×10^{-5}/RY	BWR-3 = 1.44×10^{-6}/RY

Frequency/Consequence Estimate

The PNL panel estimated[64] the most likely reduction in human error rates for operators due to the conduct of the proposed workshops would be 3%. This is assuming the workshops are conducted in the manner now perceived. That is, to focus on data gathering for the NRC. This reduces the amount of time that could be devoted to inter-licensee sharing of operational experiences which would have a more direct effect on safety-related operational performance in the plants. The possible range of reduction stretched from 1% to 10%. If the focus could be shifted toward the inter-licensee exchange of operational experiences, the most likely reduction in error rate would shift upward. However, it is not expected to exceed 10%.

Based on the PNL estimates and calculations,[64] and assuming a typical midwest-type meteorology and an average population density of U.S. reactor sites of 340 people per square mile, the public risk reduction is 7,140 man-rem for 143 plants with an average existing lifespan of 28 years. The occupational dose reduction is minor at a calculated value of 46 man-rem.

Cost Estimate

Industry Cost: The industry resources required for implementation are estimated to be one person-month per plant. This is the estimated personnel requirement associated with the trial workshops currently being conducted. It includes utility staff time for attendance of the workshop, preparation by staff and management, and staff time dedicated to the dissemination of insights gained at the workshop. At a cost of $100,000/man-year and with 4.33 weeks/month, this yields a cost of $8,333/plant. Across the industry, i.e., 143 plants, this would be $1.19M.

The industry resources required annually to participate in the training workshops are estimated to be the same as those for implementation. That is, one person-month per plant, which includes workshop attendance, preparation before the workshop, and dissemination of information afterward, would be needed. This would be equivalent to \$8,333/RY. For the total industry (143 plants), this works out to an estimated 143 man-months/year or \$1.19M/year. Given the average remaining lifetime for the plants, this gives a total operational cost of \$33.3M. Therefore, the total industry cost associated with this issue is \$34.5M.

NRC Cost: The total cost to the NRC to implement the resolution of this issue was estimated to be \$0.3M. This includes NRC staff labor and services of a contractor. Since the activities of the NRC staff and the contractor are to some degree interchangeable, no attempt was made to provide separate estimates. The annual cost to the NRC was also estimated to be \$0.3M. Again, this was assumed to contain some mixture of staff and contractor expenses. Over the average remaining life, the operational cost comes to \$8.4M. While not specific, these estimates for implementation and operation are firmly based on the experience of conducting the present trial workshops. Therefore, the total cost to the NRC is the sum of implementation and operation costs which amounts to \$8.7M.

Value/Impact Assessment

Based on the estimated public risk reduction of 7,140 man-rem, the value/impact score is given by:

$$S = \frac{7{,}140 \text{ man-rem}}{\$(34.5 + 8.7)\text{M}}$$

$$= 165 \text{ man-rem/\$M}$$

Other Considerations

The accident avoidance cost is the product of the change in accident frequency (ΔF) and the estimated cost to the utility of a major accident (A). This latter term is estimated[64] to be \$1.65 Billion. The cost per plant-year is then estimated to be:

PWRs: $(\Delta F)(A) = (7 \times 10^{-7})(\$1{,}650\text{M})/\text{RY} = \$1{,}200/\text{RY}$
BWRs: $(\Delta F)(A) = (3.2 \times 10^{-7})(\$1{,}650\text{M})/\text{RY} = \$530/\text{RY}$

The total cost for all plants is the per-plant-year cost multiplied by the number of plants (N) and the average remaining lifetime (T) for each type of plant:

$$\Sigma(NT)(\Delta F)(A) = \$(95)(28.5)(1{,}200)\text{M} + \$(48)(27.0)(530)\text{M} = \$3.9\text{M}$$

CONCLUSION

Because of the extensive number of sequences considered by PNL to be affected by this issue, the base-case risk is high at a calculated range of from 60 to 73 man-rem/RY. With a value/impact score of 165 man-rem/\$M and an estimated risk reduction of 7,140 man-rem, this issue was given a medium priority ranking.

The staff conducted three workshops and a mail survey in order to evaluate the effectiveness of both mechanisms for obtaining feedback to the NRC from utility operating staffs. The results of these two approaches were documented in NUREG/CR-3739[802] and NUREG/CR-4139,[803] respectively. The staff concluded that both feedback mechanisms have proved to be effective methods of gathering data from operations personnel and did not recommend conducting workshops or surveys on an annual basis; it would be preferable to use such mechanisms judiciously when a real need existed.[804] Thus, this item was RESOLVED and no new requirements were established.

ITEM I.A.2.6(5): DEVELOP INSPECTION PROCEDURES FOR TRAINING PROGRAM

This item was evaluated in Item I.A.2.6(1) above and, in accordance with an OIE memorandum,[379] was RESOLVED. No new requirements were established.

ITEM I.A.2.6(6): NUCLEAR POWER FUNDAMENTALS

DESCRIPTION

This NUREG-0660[48] item called for NRR to develop requirements for the inclusion of nuclear power fundamentals within the instruction given to reactor operators. This arose out of a concern[174] that the 12 weeks of fundamentals training given to operators at that time was insufficient.

PRIORITY DETERMINATION

In order to assess this safety issue, a panel of experts was assembled from the PNL staff. This panel was comprised of members experienced in reactor operator licensing, reactor operations, utility field work, and general reactor safety areas. The results of the PNL assessment are contained in NUREG/CR-2800.[64]

Assumptions

The panel felt there had been significant progress across the industry in the area of instruction in nuclear power fundamentals since the issuance of NUREG-0585[174] in 1979. Further increase in emphasis on fundamentals was felt to be unlikely to improve operator performance. The current trend in operator licensing examinations is to stress operational knowledge and de-emphasize fundamentals. This supports the view that further fundamental training would not add to plant safety.

It was assumed that, if implemented, the additional nuclear power fundamentals training would add 4 weeks to the training period. Also, it was assumed that 20 operators complete the training course each year at every plant. In addition, one full-time instructor was assumed to be required. This yields 80-man-weeks for the operators and 44 man-weeks for the instructors, or 124 man-weeks/plant overall each year. To implement this practice, an effort equivalent to one year of operation (124 man-weeks) was estimated to be required.

Frequency/Consequence Estimate

Safety issues which deal with operator training can affect the public risk by improvements in the operator safety-related performance. This can lead to a reduction in core-melt frequency and a reduced probabilistic risk. For this safety issue the PNL panel felt that the current level of instruction in nuclear power fundamentals was adequate. Further emphasis of fundamentals was viewed as not likely to improve operator safety performance. Therefore, there would be no measurable public risk reduction associated with the implementation of this issue. The PNL panel also saw no reduction in occupational dose associated with the implementation of the solution.

Cost Estimate

NRC effort to implement the solution is estimated[48] to be 0.4 man-year or approximately 18 man-weeks. No added costs are estimated for operation for the NRC. The review of the additional instruction could be contained in the current routine function thereby causing no added expense.

Value/Impact Assessment

Based on the judgment that there would be no risk reduction resulting from this issue, the value/impact score is zero.

CONCLUSION

In view of the fact that it is believed that the current level of instruction in nuclear power fundamentals is adequate for reactor operators, further emphasis of fundamentals as required by this issue is viewed as not likely to improve operator safety performance. The resulting value/impact score of zero indicates that this issue should be DROPPED from further consideration.

ITEM I.A.2.7: ACCREDITATION OF TRAINING INSTITUTIONS

DESCRIPTION

Historical Background

Based on the requirements of NUREG-0660,[48] this item required NRR to complete a study to establish the procedures and requirements for NRC accreditation of reactor operator training programs. The resulting study would be developed into a Commission paper describing the various options for accreditation.

Safety Significance

There are two aspects to the safety benefit for this issue. One is the reduction of public risk through the improvement of operator performance, which is expected from the improved training accreditation. The second is a reduction in occupational exposure. This will primarily be for operators who often supervise maintenance or perform other duties in radiation zones. However, some reduction in routine occupational exposure can also be expected for other operations personnel as a result of the increased awareness by the operators.

Possible Solution

In order to assess this safety issue, a panel of experts was assembled from the PNL staff. This panel was comprised of members experienced in reactor operator licensing, reactor operations, utility field work, and general reactor safety areas.

The panel envisioned the resolution of this safety issue as the formation of an accreditation board consisting of representatives from the NRC, industry, and academia. This board would develop and apply criteria for accreditation. This would include training programs of utilities, university-related programs, and independent training institutions. While theoretically applying to training for all operations staff, the PNL panel felt the current thrust was focused on reactor operators. Therefore, the assessment was made assuming only operators would be affected.[64]

PRIORITY DETERMINATION

Assumptions

The views of the panel include an awareness of the fact that some training programs are very near to accreditation already. Either through association with the universities or through other means of providing high quality instruction, these programs would be likely to acquire accreditation from the board easily. Other training programs are not so well prepared for accreditation and may require significant effort and expense to upgrade them. Some savings may be gained for multi-unit sites in sharing costs.

Therefore, the resolution of this safety issue was assumed to be an improvement in operator performance. For some utilities, approximately 10% of the total, this issue will have essentially no effect. This is because: (1) their current training programs would be accredited with little effort and (2) the quality of their programs is sufficiently high that accreditation would result in no discernible improvement in their operators' performance. Other utilities will see varying degrees of improvement. Those with training programs that are below the accreditation standards will be brought up nearer to the high quality enjoyed by the outstanding utilities. Overall, the effect on operator human error is estimated to be a reduction of 10% across the affected portion of the industry The detailed assumptions for this analysis are as follows:

1. Applicable Plants: BWRs and PWRs - 90% of total plants; 43 BWRs, 86 PWRs, or 129 plants in all.

2. Selected Analysis Plant: Oconee 3 - representative PWR. It is assumed that the fractional risk and core-melt frequency reductions for the representative BWR (Grand Gulf) will be equivalent to those for the representative PWR. Therefore, the analysis is conducted only for the PWR, but the fractional risk and core-melt frequency reductions are also applied to the BWR.

Frequency/Consequence Estimate

Based on the PNL analysis,[64] and assuming a typical midwest-type meteorology and an average population density of U.S. reactor sites of 340 people per square mile, the anticipated public risk reduction is calculated to be 26,180 man-rem.

Cost Estimate

The PNL panel estimated[64] the costs associated with implementation and operation of the resolution to this safety issue. The one-time costs to industry to implement the change initially was estimated to be in the range of \$0.1M to \$1M per reactor. Those with training programs closer to accreditable status would enjoy the smaller costs. The best estimate for the average plant was taken to be \$0.3M. Operation under the accreditation program was estimated to cost between \$0.05M and \$0.25M per plant annually for additional funding to maintain an accredited training program. The best estimate was \$0.1M/plant annually.

The cost to the NRC to implement the accreditation was estimated to be \$0.635M which is equivalent to 330 person-weeks. The annual operational cost to the NRC is estimated[64] to be \$100,000 or one man-year.

The detailed breakdown of these costs are as follows:

\$300,000/Plant Industry Implementation (approximately 3 man-yrs):

- to review accreditation standards
- to compare the present utility practices with the developed standards
- plan the necessary upgrades
- implement the program upgrades to fulfill the accreditation requirements.

\$100,000/Plant-yr Industry Operation and Maintenance:

- time invested by the staff in upgraded training (increased course time, quality, etc.)
- instruction upgrade (time, quality, etc.)

\$500,000 NRC Implementation (approximately 5 man-yrs)

- predicated on the possibility that INPO accreditation will not be forthcoming; NRC may have to do
- NRC to develop accreditation standards, regulations, and implement to adoption by the industry.

\$100,000 NRC Operation and Maintenance (approximately 1 man-yr/yr)

- additional OIE efforts to assure industry maintenance of standards (all plants).

The total costs for this safety issue are, therefore, estimated[64] by PNL as follows:

1.	Implementation of the SIR by Industry		$ 39,000,000
2.	Operation and Maintenance of the SIR by Industry		360,000,000
3.	NRC Implementation of the SIR		635,000
4.	NRC Operation and Maintenance of the SIR		2,800,000
		Total:	$402,435,000

Value/Impact Assessment

Based on the estimated public risk reduction of 26,180 man-rem, the value/impact score is given by:

$$S = \frac{26,180 \text{ man-rem}}{\$402.4M}$$

$$= 65 \text{ man-rem/\$M}$$

Other Considerations

The industry accident avoidance cost was estimated by PNL[64] to be $14M. The occupational risk reduction is estimated to be 22,170 man-rem resulting from accident avoidance (170 man-rem) and from operation and maintenance of the SIR (22,000 man-rem).

CONCLUSION

Although the value/impact score was low, this issue was determined to be medium priority because of the magnitude of the potential public risk reduction. However, in June 1985, the Commission recognized that the industry had made progress in developing programs to improve nuclear utility training and personnel qualification. As a result, the Commission adopted a Policy Statement on Training and Qualifications which made the training accreditation program managed by INPO the focus of training improvement in the industry.[777] Thus, this item was RESOLVED and no new requirements were established.

REFERENCES

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

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

54. NUREG/CR-1659, "Reactor Safety Study Methodology Application Program," U.S. Nuclear Regulatory Commission, 1981.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

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

174. NUREG-0585, "TMI Lessons Learned Task Force Final Report," U.S. Nuclear Regulatory Commission, October 1979.

226. Regulatory Guide 1.8, "Personnel Selection and Training," U.S. Nuclear Regulatory Commission, May 1977.

251. SECY-79-330E, "Qualifications of Reactor Operators," July 30, 1979.

252. NRR-80-117, "Study of Requirements for Operator Licensing," February 4, 1980.

253. ANSI/ANS 3.1, "Selection, Qualification, and Training of Personnel for Nuclear Power Plants," American National Standards Institute, 1981.

379. Memorandum for H. Denton from R. DeYoung, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," January 24, 1983.

437. Memorandum for H. Denton from R. Minogue "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," March 29, 1983.

438. Memorandum to Office Directors from W. Dircks, "NRC Actions Required by Enactment of the Nuclear Waste Policy Act of 1982," January 19, 1983.

651. NUREG-0985, Revision 1, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," U.S. Nuclear Regulatory Commission, September 1984.

777. Memorandum for W. Dircks from H. Denton, "Closeout of TMI Action Plan Items I.A.2.2 and I.A.2.7 Training and Qualifications of Operating Personnel," June 24, 1985.

800. NUREG/4258, "An Approach to Team Skills Training of Nuclear Power Plant Control Room Crews," U.S. Nuclear Regulatory Commission, July 1985.

801. Memorandum for W. Dircks from H. Denton, "Team Training for Nuclear Power Plant Control Room Crews," July 10, 1985.

802. NUREG/CR-3739, "The Operator Feedback Workshop: A Technique for Obtaining Feedback from Operations Personnel," U.S. Nuclear Regulatory Commission, September 1984.

803. NUREG/CR-4139," The Mailed Survey: A Technique for Obtaining Feedback from Operations Personnel," U.S. Nuclear Regulatory Commission, May 1985.

804. Memorandum for W. Dircks from H. Denton, "TMI Action Plan Item I.A.2.6(4)," September 25, 1985.

956. Memorandum for V. Stello from H. Denton, "Close-out of the Division of Human Factors Technology TMI Action Plan Items," January 6, 1987.

966. Federal Register Notice 50 FR 11147, "10 CFR Ch. 1, Commission Policy Statement on Training and Qualification of Nuclear Power Plant Personnel," March 20, 1985.

1043. SECY-86-348, "Final Rulemaking for Revisions to Operator Licensing - 10 CFR 55 and Conforming Amendments," November 21, 1986.

1044. Federal Register Notice 52 FR 16007, "Regulatory Guides; Issuance and Availability," May 1, 1987.

1045. Memorandum for V. Stello from E. Beckjord, "Resolution of TMI Action Plan Items and Human Factors Issues," May 18, 1987.

TASK I.A.4: SIMULATOR USE AND DEVELOPMENT

The objectives of this task were as follows: (1) to establish and sustain a high level of realism in the training and retraining of operators, including dealing with complex transients involving multiple permutations and combinations of failures and errors, and (2) to improve operators' diagnostic capability and general knowledge of nuclear power plant systems.

ITEM I.A.4.1: INITIAL SIMULATOR IMPROVEMENT

ITEM I.A.4.1(1): SHORT-TERM STUDY OF TRAINING SIMULATORS

DESCRIPTION

The TMI Action Plan[48] called for a short-term study of training simulators. The purpose was to collect and develop corrections for presently identified weaknesses. A study of training simulators was undertaken and a report, NUREG/CR-1482,[299] was issued in June 1980.

CONCLUSION

This item has been RESOLVED and no new requirements were established.

ITEM I.A.4.1(2): INTERIM CHANGES IN TRAINING SIMULATORS

DESCRIPTION

The TMI Action Plan[48] stated that requirements to correct specific training simulator weaknesses should be developed based on the short-term study resulting from Item I.A.4.1(1). This item was completed with the issuance of Regulatory Guide 1.149,[439] "Nuclear Power Plant Simulators for Use in Operator Training," in April 1981.

CONCLUSION

This item has been RESOLVED and new requirements were established.

ITEM I.A.4.2: LONG-TERM TRAINING SIMULATOR UPGRADE

The four parts of this item have been combined and evaluated together.

DESCRIPTION

Historical Background

Nuclear power plant simulators are recognized as an important part of reactor operator training. The TMI Action Plan[48] called for a number of actions to improve simulators and their use. There is significant interaction among the simulator-related action items and clear separation is difficult.

Item I.A.4.2 has a number of components dealing with long-term upgrades. The NUREG-0660[48] description calls for research to improve the use of simulators in training operators, develop guidance on the need for and nature of operator action during accidents, and gather data on operator performance. Specific research items mentioned include simulator capabilities, safety-related operator action, and simulator experiments. The item also calls for the upgrading of training simulator standards, specifically updating of ANSI/ANS 3.5-1979. A regulatory guide endorsing that standard and giving the criteria for acceptability is also mentioned. The final portion of Item I.A.4.2 calls for a review of simulators to assure their conformance to the criteria.

A significant portion of the activities to be conducted under this action plan item has been completed. For example, ANSI/ANS 3.5 was revised and issued in 1981. The regulatory guide endorsing this standard, Regulatory Guide 1.149,[439] "Nuclear Power Plant Simulators for Use in Operator Training," as well as numerous research reports have been published.

It is clear that the regulations, the ANS standard, and the regulatory guide do not require a site-specific simulator. 10 CFR 55 states that, if a simulator is used in training, it "... shall accurately reproduce the operating characteristics of the facility involved and the arrangement of the instrumentation and controls of the simulator shall closely parallel that of the facility involved." ANSI/ANS 3.5-1981 calls for a high degree of fidelity between the simulator and the "reference plant." However, there is no requirement that the reference plant be the same facility that the personnel in training will in fact operate. Regulatory Guide 1.149,[439] explicitly makes the distinction stating "... the similarity that must exist between a simulator and the facility that the operators are being trained to operate is not addressed in the guide and should not be confused with the guidance provided that specifies the similarity that should exist between a simulator and its reference plant."

The work that has been completed for Item I.A.4.2(1) includes the issuance of NUREG/CR-2353[300] (Volumes I and II), NUREG/CR-1908,[416] NUREG/CR-2598,[417] NUREG/CR-2534,[418] NUREG/CR-3092,[419] and NUREG/CR-3123.[653] This item, however, has long-range requirements calling for: (1) the review of operating experience to provide data on operator responses, and (2) the design and conduct of experiments to determine operator error rates under controlled conditions. Therefore, this item is not completed at this time. However, Items I.A.4.2(2) and I.A.4.2(3) have been completed with the issuance of Regulatory Guide 1.149.[439] Item I.A.4.2(4) concerns the long-term training simulator improvement criteria which were also established in Regulatory Guide 1.149,[439] issued in April 1981, and the criteria were initiated in FY 1982. However, the review of submittals from simulator owners for conformance with the criteria is an on-going task which is still not complete. Therefore, the outstanding portions of this issue that have yet to be completed are the continuation of simulator research and the review for conformance to acceptability criteria.

The assessment of this safety issue was conducted by PNL staff[64] with experience in reactor operator licensing, reactor operation, and general reactor safety, in consultation with General Physics Corporation. General Physics Corporation provides utility training services and is greatly experienced in reactor simulators, providing procurement and startup assistance, operation and maintenance services, and simulator modifications.

In the assessment of this issue it is necessary to acknowledge that many of the TMI action items associated with operator training are interrelated and that ranking problems become involved when an attempt is made to assess these independently. For example, the present issue relates to Items I.A.2.6(1,2,3, and 5), which deal with training improvements including the enhanced use of existing simulators, and I.A.4.1, which deals with initial simulator improvement, including short-term and interim changes in training simulators. However, it is useful to note that the final safety ranking of this issue is relatively insensitive to changes in the basic assumptions used to distinguish these inter-related issues, by the very nature of the ranking matrix. Therefore, it is possible to establish a priority ranking for this issue, despite the possible overlapping of potential benefits and costs with the other inter-related issues.

Safety Significance

Use of simulators with high fidelity to the reference plant would significantly improve operator training in dealing with abnormal conditions thereby reducing operator error. The operators' performance under accident conditions is expected to be enhanced. Thus, potential core melts would be avoided and overall core-melt frequency reduced.

Possible Solution

A possible solution would be to establish a high level of realism in the training and retraining of plant operators by developing simulators with a high degree of fidelity to the reference plant.

PRIORITY DETERMINATION

Assumptions

It was assumed that the major effect of these issues, both in terms of safety benefit and cost incurred, would be in the enhancement of the level of realism imparted by simulators. The specific modeling capabilities given under Item I.A.4.1(2) and in the specification of ANSI/ANS 3.5-1981 specify this feature.

It was assumed for the resolution to this safety issue, that in order to provide the intended level of realism, site-specific simulators would be acquired. Such simulators would be significantly more realistic when compared to the specific facilities, both in layout and operation, than existing generic simulators. In addition, they are assumed to have enchanced transient and accident modeling capabilities.

In our assessment, it was clear that provision of site-specific simulators, while not explicitly required, would meet the requirements of Item I.A.4.1(2), the fidelity requirements of ANSI/ANS 3.5-1981, and the accurate reproduction requirements of 10 CFR 55. Less sweeping simulator enhancements might also fulfill these requirements but would have to be decided on a case-by-case basis. Therefore, for risk, dose, and cost estimates we assumed the enhancement would be effected by the introduction of site-specific simulators.

The public risk reduction (and occupational dose reduction due to accident avoidance) are associated with the reduction in operator error expected to result from the training and requalification of operators on improved simulators. Inasmuch as any studies relating human error rates to the realism of simulator training are not available, this assessment will be based primarily on PNL engineering judgment. Therefore, it is estimated that a reduction in operator error rates of 30% will result from the resolution of this safety issue. This sole-value estimate implies that for specific instances the improvement could be much greater but the 30% reduction is used as an estimate of the average improvement for the purposes of calculation.

The number of plants and the average remaining lifetimes are taken as 90 plants and 28.8 yrs for PWRs and 44 plants and 27.4 years for BWRs. The plants selected for analysis are the Oconee 3 as representative of the PWRs and Grand Gulf as representative of the BWRs. (It is assumed that the fractional risk and core-melt frequency reductions for Grand Gulf will be equivalent to those for the PWR which is calculated directly.)

The dose calculations are based on a reactor site population density of 340 people per square mile and a typical midwest meteorology is assumed.

Frequency Estimate

All release categories are affected by the resolution of this issue. The calculated core-melt frequencies are 8.2×10^{-5}/RY for PWRs and 3.7×10^{-5}/RY for BWRs. The reduction in these frequencies, based on the 30% reduction estimated for operator error, is 1.3×10^{-5}/RY for PWRs and 5.9×10^{-6}/RY for BWRs.

Consequence Estimate

The resulting total reduction in public risk is 150,000 man-rem. The estimated reduction in occupational dose is 820 man-rem based on accident avoidance only since there are no implementation or maintenance dose reductions associated with resolution of this issue.

Cost Estimate

Industry Cost: The major effect of the resolution of these safety issues was assumed to be the acquisition and use of site-specific simulators. The costs to industry of such an undertaking would be substantial. It is important to recognize that if improved modelling changes were possible on existing simulators, the cost to industry would be substantially smaller. However, this is not clear at this time and it is assumed that new simulators would be required. (The impact of this assumption can be weighed subsequently in the final safety priority ranking. The assumption can be reevaluated at that time for any appropriate modifications.)

Assuming that new simulators would be required, the principal industry costs for implementation of this safety issue would be the purchase of the simulators and provision of the new training materials. The capital cost of a simulator is estimated to be \$7M. The provision of training materials is estimated to be equivalent to a 7 man-year effort.

It was assumed that all reactors, both operating and planned, would be affected. However, not every reactor would require a simulator. Many reactor sites have two or more reactors located together. If these reactors are sufficiently similar, a single simulator could serve them. Examining the list of 134 operating and planned power reactors, it was estimated that 62 additional site-specific simulators would be adequate. This assumed that 20% of the potential simulators are not required because either a site-specific simulator already exists or the plant in question is an older facility with limited lifetime remaining.

The costs for the 62 new simulators spread over 134 reactors yields \$3.2M/reactor in capital cost and 3.2 man-year/reactor to provide new training materials. The operation and maintenance of the new simulators is estimated to require 3 man-years of effort per simulator. Again, sharing the expense for 62 simulators over 134 reactors yields 1.4 man-years/reactor. Industry may also experience costs stemming from participation in simulator experiments and research. However, in comparison to the costs related to new simulators, these costs would be small.

Based on these assumptions the total industry costs are obtained as follows:

(1) Safety Issue Resolution (SIR) Implementation

(a) Labor: $\left(\frac{7 \text{ man-yr}}{\text{simulator}}\right) \left(\frac{62 \text{ simulators}}{134 \text{ plants}}\right) \left(\frac{\$100{,}000}{\text{man-year}}\right) = \$320{,}000/\text{plant}$

(b) Equipment: $\left(\frac{62 \text{ simulators}}{134 \text{ plants}}\right) \left(\frac{\$7\text{M}}{\text{simulator}}\right) = \$3.2\text{M per plant}$

Thus, the total industry cost for implementation is (134 plants) (\$320,000/plant + \$3,200,000/plant) or \$470M.

(2) Operation and Maintenance of the SIR

$\left(1.4 \frac{\text{man-yr}}{\text{reactor}}\right) \left(\frac{\$100{,}000}{\text{man-yr}}\right) [(90 \text{ PWRs})(28.8 \text{ yrs}) + (44 \text{ BWRs})(27.4 \text{ yrs})]$

$= \$530\text{M}$

Therefore, the total combined industry cost is \$(470 + 530)M or \$1,000M.

NRC Cost: The principal costs to the NRC are the continuation of research and the conduct of the confirmatory reviews. No additional development costs are foreseen as ANSI/ANS 3.5 is currently being revised and will necessitate a revision to Regulatory Guide 1.149.[439]

The continuing research is treated as an implementation cost. It is estimated to require one NRC man-year and \$1M in contractor support. (This includes the remaining costs associated with Item I.E.8.) The confirmatory reviews are also treated as an implementation cost and are estimated to require 4 man-weeks/simulator, or 248 man-weeks in all for the assumed 62 new simulators.

The operational review cost to the NRC is minimal. It is assumed that annually each simulator will be audited to assure that reference plant updates have been adequately represented on the simulator. Such an annual review is estimated to require 2 man-weeks/simulator or 124 man-weeks/year for all 62 new simulators assumed.

NRC costs are estimated as follows:

(1) SIR Development

There is no cost for SIR development since all work is essentially complete and a solution has been identified.

(2) SIR Implementation

(a) Continuing Research: $\frac{1 \text{ man-yr}}{134 \text{ plants}} = 0.33 \ \frac{\text{man-wk}}{\text{plant}}$

(b) Initial Simulator Reviews: $\frac{248 \text{ man-wk}}{134 \text{ plants}} = 1.9 \ \frac{\text{man-wk}}{\text{plant}}$

Based on a total NRC manpower of 2.23 man-wk/plant, the NRC manpower cost for implementation is

$$\left(\frac{2.23 \text{ man-wk}}{\text{plant}}\right) \left(\frac{\$2,270}{\text{man-wk}}\right) (134 \text{ plants}) = \$678,300$$

(c) NRC Contractor Support = $1M

Therefore, total NRC Cost for SIR Implementation is ($678,300 + $1M) or $1.7M.

(3) Review of SIR Operation and Maintenance

$$\left(\frac{2 \text{ man-wk}}{\text{simulator-yr}}\right) \left(\frac{67 \text{ simulators}}{134 \text{ plants}}\right) \left(\frac{\$2,270}{\text{man-wk}}\right) = \$2,100/\text{RY}$$

The total NRC cost for review of SIR operation and maintenance for all affected plants is [(90 PWRs)(28.8 yr) + (44 BWRs)(27.4 yrs)]($2,100/RY) = $8M. Thus, the total NRC cost is $(1.7 + 8)M or $9.7M.

Therefore, total industry and NRC cost for the SIR is $(1,000 + 9.7)M or $1,010M.

Value/Impact Assessment

Based on a public risk reduction of 150,000 man-rem, the value/impact score is given by:

$$S = \frac{150,000 \text{ man-rem}}{\$1,010\text{M}}$$

$$= 148.7 \text{ man-rem/\$M}$$

CONCLUSION

Based on the estimated risk reduction of 150,000 man-rem and the value/impact score of approximately 150 man-rem/$M, the safety priority ranking of this issue would be HIGH. In view of the large estimated risk reduction, this safety priority ranking is essentially unaffected by any reasonable uncertainties in the cost estimates.

ITEM I.A.4.2(1): RESEARCH ON TRAINING SIMULATORS

This item was evaluated in Item I.A.4.2 above and was determined to be high priority. In April 1987, the issue was RESOLVED with the publication of Revision 1 to Regulatory Guide 1.149.[439] New requirements were established.[1045]

ITEM I.A.4.2(2): UPGRADE TRAINING SIMULATOR STANDARDS

This item was evaluated in Item I.A.4.2 above and was determined to be RESOLVED with the issuance of Regulatory Guide 1.149[439] in April 1981. New requirements were established.

ITEM I.A.4.2(3): REGULATORY GUIDE ON TRAINING SIMULATORS

This item was evaluated in Item I.A.4.2 above and was determined to be RESOLVED with the issuance of Regulatory Guide 1.149[439] in April 1981. New requirements were established.

ITEM I.A.4.2(4): REVIEW SIMULATORS FOR CONFORMANCE TO CRITERIA

This item was evaluated in Item I.A.4.2 above and was determined to be a HIGH priority issue.

ITEM I.A.4.3: FEASIBILITY STUDY OF PROCUREMENT OF NRC TRAINING SIMULATOR

DESCRIPTION

The description of this safety issue in NUREG-0660[48] is as follows:

> "In addition to the increased use of industry simulators for training of NRC staff (notably, the work by OIE with the TVA training center simulators), a feasibility study of the lease or procurement of one or more simulators to be located in the NRC headquarters area will be performed. These simulators would be used in familiarizing the NRC staff with reactor operations, in assessing the effectiveness of operating and emergency procedures and in gathering data on operator performance. The study will include development of specifications, development of procurement and commissioning schedules, estimation of costs, and comparison with other methods of providing such training for NRC personnel."

Technical studies[262,263,264] that have been performed by BNL on this issue have indicated that existing simulators have significant modelling limitations. It was established that the capability of existing simulators was not acceptable at any but near-normal operating conditions, and that the lack of technical capability during two-phase conditions was significant. These results have an adverse effect on the feasibility of a training simulator for the NRC staff.

The intent of this issue is to improve the NRC staff's familiarization with reactor operations. The study is an effort to establish the feasibility of procuring an NRC training simulator. The resolution of this issue has no direct bearing on any public risk reduction and, therefore, it is concluded that this issue is a licensing issue.

CONCLUSION

This Licensing Issue has been resolved.

ITEM I.A.4.4: FEASIBILITY STUDY OF NRC ENGINEERING COMPUTER

DESCRIPTION

The purpose[48] of this study is to fully evaluate the potential value of and, if warranted, propose development of an engineering computer that realistically models PWR and BWR plant behavior for small-break LOCA and other non-LOCA accidents and transients that may call for operator actions. Final development of the proposed engineering computer will depend on a number of research efforts. Risk assessment tasks (interim reliability evaluation program, or IREP, for example) to define accident sequences covering severe core damage will also provide the guidelines for the experimental and analytical research programs needed to improve the diagnostics and general knowledge of nuclear power plant systems. The programs will assist the development and testing of fast running computer codes used to predict realistic system behavior for these multiple accident studies. These codes will provide the basic models for use in the improved engineering computer as well as the capability for NRC audit of NSSS analyses.

A report on the review of PWR simulators was completed and issued by BNL.[262] A final report on BWR simulators was also completed by BNL.[263] Work on Plant Analyzers continued at BNL, INEL, and LASL. The RES staff believed that Plant Analyzers (Engineering Computer) would be helpful in uncovering potential operational safety problems in LWRs, caused by operator errors or equipment malfunctions, which will lead to risk reductions through increased operator awareness, improved procedures, and equipment redundancy.

The Plant Analyzer is not a design tool but rather an aid to the NRC staff in performing an audit function in the licensing process. Thus, this issue will not result in a direct reduction in public risk and, therefore, is considered a licensing issue.

CONCLUSION

After the second year of research on the Engineering Computer (Nuclear Plant Analyzer), it was concluded that it was not feasible to develop a device that would be sufficiently accurate and function with sufficient speed (i.e., faster than real accident progression time) to give a plant operator information adequate to guide action he or she should take during an accident. It was found, however, that a Nuclear Plant Analyzer, which takes output from an NRC safety analysis code such as TRAC or RELAP and displays plant accident conditions in schematic form on a video screen, will considerably ease the burden of understanding the results of complex safety analysis calculations. The Plant Analyzer also allows the safety analyst to interpose simulated operator actions into an accident calculation underway. Based on these findings, the objectives of the development program were reoriented toward assistance for plant safety analysis and away from operator accident assistance.

A Management Plan[968] for the Nuclear Plant Analyzer was prepared by the staff and included a listing of products expected to enter the regulatory arena in fiscal years 1985 through 1989. The staff concluded that it was not feasible to develop an Engineering Computer to provide input for operator actions during plant accidents; it was feasible to develop a device to give NRC an improved capability to audit NSSS analyses and this is being done in accordance with the Management Plan. Thus, this Licensing Issue has been resolved.

REFERENCES

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

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

262. BNL/NUREG-28955, "PWR Training Simulator and Evaluation of the Thermal-Hydraulic Models for Its Main Steam Supply System," Brookhaven National Laboratory, 1981.

263. BNL/NUREG-29815, "BWR Training Simulator and Evaluation of the Thermal-Hydraulic Models for Its Main Steam Supply System," Brookhaven National Laboratory, 1981.

264. BNL/NUREG-30602, "A PWR Training Simulator Comparison with RETRAN for a Reactor Trip from Full Power," Brookhaven National Laboratory, 1981.

299. NUREG/CR-1482, "Nuclear Power Plant Simulators: Their Use in Operator Training and Requalification," U.S. Nuclear Regulatory Commission, August 1980.

300. NUREG/CR-2353, "Specification and Verification of Nuclear Power Plant Training Simulator Response Characteristics," U.S. Nuclear Regulatory Commission, 1982.

416. NUREG/CR-1908, "Criteria for Safety-Related Nuclear Power Plant Operator Actions: Initial Pressurized Water Reactor (PWR) Simulator Exercises," U.S. Nuclear Regulatory Commission, September 1981.

417. NUREG/CR-2598, "Nuclear Power Plant Control Room Task Analysis: Pilot Study for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, July 1982.

418. NUREG/CR-2534, "Criteria for Safety-Related Nuclear Power Plant Operator Actions: Initial Boiling Water Reactor (BWR) Simulated Exercises," U.S. Nuclear Regulatory Commission, November 1982.

419. NUREG/CR-3092, "Criteria for Safety-Related Nuclear Power Plant Operator Actions: Initial Simulator to Field Data Calibration," U.S. Nuclear Regulatory Commission, February 1983.

439. Regulatory Guide 1.149, "Nuclear Power Plant Simulators for Use in Operator Training," U.S. Nuclear Regulatory Commission, April 1981, (Revision 1) April 1987.

651. NUREG-0985, Revision 1, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," U.S. Nuclear Regulatory Commission, September 1984.

653. NUREG/CR-3123, "Criteria for Safety-Related Nuclear Power Plant Operator Actions: 1982 Pressurized Water Reactor (PWR) Simulator Exercises," U.S. Nuclear Regulatory Commission, June 1983.

954. Memorandum for V. Stello from E. Beckjord, "Closeout of TMI Action Plan Items," November 13, 1986.

968. Memorandum for J. Roe from R. Minogue, "Nuclear Plant Analyzer (NPA) Management Plan," December 12, 1985.

1045. Memorandum for V. Stello from E. Beckjord, "Resolution of TMI Action Plan Items and Human Factors Issues," May 18, 1987.

TASK III.D.3: WORKER RADIATION PROTECTION IMPROVEMENT

The objective of this task is to improve nuclear power plant worker radiation protection to allow workers to take effective action to control the course and consequences of an accident, as well as to keep exposures as low as reasonably achievable (ALARA) during normal operation and accidents, by improving radiation protection plans, health physics, inplant radiation monitoring, control room habitability, and radiation worker exposure data base.

ITEM III.D.3.1: RADIATION PROTECTION PLANS

DESCRIPTION

Historical Background

The purpose of this TMI Action Plan[48] item is to improve nuclear power plant worker radiation protection programs by better defining the criteria and responsibility for such programs. Detailed appraisals of health physics programs at all operating nuclear power plants were performed in 1980 and 1981. These appraisals, summarized in NUREG-0855,[204] indicated that certain generic deficiencies existed at many plants due in part to lack of specific performance criteria and/or assigned responsibility for programs. The establishment of a radiation protection plan as a guiding document for implementing procedures has been proposed as a method for formalizing commitment to specific performance criteria contained in Regulatory Guides and SRP Section 12.[11] Proposed guidance and acceptance criteria for radiation protection plans have been published in draft form as NUREG-0761.[205] A proposed amendment to 10 CFR 50 has been drafted.[206]

Safety Significance

The development of radiation protection plans has no impact on public safety. Instead, the safety significance lies in the reduction of occupational exposure.

Possible Solutions

As currently envisioned, radiation protection plans would tie together specific implementing procedures, many of which currently exist at licensed plants. Additional procedures may be required at many plants to fully implement the plan; however, extensive revision of procedures should not generally be required. Administrative and technical manpower would be required to develop the plan, revise and write procedures as necessary, and some additional equipment (such as additional survey equipment) may be required. Installation of such equipment should not require any significant work in radiation areas. The benefit of radiation protection plans would be in two primary areas: (1) reduction of individual and collective dose due to improved planning and controls for work in radiation areas, and (2) improved confidence in results of radiation protection programs.

PRIORITY DETERMINATION

The assessment of this issue and its resolution was first performed[64] by consensus opinions of four PNL health physicists who were extensively involved in the Health Physics Appraisal Program. These personnel included expertise from both industry and regulatory sides of the issue. Estimates of routine cost and probable man-rem reductions were discussed and agreed upon. For core-melt accident recovery and refurbishing, the panel assumed man-rem savings comparable on a percentage basis to those for routine operations. The cost impact of these man-rem savings was then estimated by a PNL expert involved in estimating accident recovery costs.

Frequency/Consequence Estimate

There are three terms in the estimation of occupational dose change due to this safety issue. These are the change due to accidents, the change due to issue resolution implementation, and the change due to resolution operation.

The estimated change due to accidents (the first term) is the change in the product of accident frequency and occupational dose associated with the recovery from an accident. As previously stated, no change in accident frequency is expected to occur due to this issue. However, a small change in occupational accident recovery dose is expected. Radiation protection plans are primarily oriented toward routine plant operation. In the event of a major core-melt accident, specialized procedures would have to be developed. Having the upgraded radiation protection plan for normal operation in place, however, is expected to result in improved specialized procedures if required. The resulting reduction in occupational dose for plant recovery is estimated to be slightly less than 5%. Using the estimates of total occupational dose resulting from recovery from an accident, as listed in Appendix D of NUREG/CR-2800,[64] this works out to 3.3×10^{-2} man-rem/RY for BWRs and 7.4×10^{-2} man-rem/RY for PWRs.

The implementation of radiation protection plans (the second term) would be an administrative effort. Therefore, there is zero exposure associated with implementation.

The establishment of radiation protection plans is estimated to result in a reduction of occupational risk during operation (the third term). This reduction would be due to improved controls on personnel dose and an improved ALARA Program. PNL's experts estimated the occupational dose reduction to be on the order of 5%.[64] However, the Occupational Radiation Protection Branch (ORPBR) of RES has been investigating the costs and benefits associated with radiation protection plans. Based on a comparison of plants with and without major radiation protection plans, it was estimated that occupational doses could be reduced by at least 10%. Savings of 25% or more appear achievable.[207] The 1980 average occupational dose was about 800 man-rem. Therefore, we will assume that radiation protection plans could avert 200 man-rem/RY.

Cost Estimate

PNL estimated that 35 man-weeks at a cost of $35,000 and equipment worth $50,000 would be required per plant to implement the radiation protection plans.[64] In order to operate under the new radiation protection plans, it was felt that most plants would have to add personnel. It was estimated that one professional and one technical staff member would be needed. At 52 weeks per year, this gives an additional 104 man-weeks per year for each plant, or \$104,000 plant cost per year.

However, ORPBR has noted that the licensees' cost will vary widely depending on the adequacy of the present program.[206] In addition, since radiation protection plans have the effect of reducing the time workers are exposed, individual tasks are often speeded up. Some licensees have found that the savings resulting from reduced downtime have compensated for the cost of the program.

Currently, there are 43 operating PWRs with a cumulative experience of 350 RY and 27 BWRs with a cumulative experience of 260 RY. If we add to these the 36 PWRs and 21 BWRs under construction and assume a plant lifetime of 30 years, there are 3,200 RY remaining: 1,180 RY for BWRs and 2,020 RY for PWRs.

ORPBR has estimated that 5 NRC staff-years will be required.[206] Thus, NRC costs are estimated to be \$500,000.

The total cost associated with the solution to this issue is \$340.5M.

Value/Impact Assessment

The total risk reduction associated with this issue is 6.4×10^5 man-rem. Therefore, the value/impact score is given by:

$$S = \frac{6.4 \times 10^5 \text{ man-rem}}{\$340.5\text{M}}$$

$$= 1{,}880 \text{ man-rem/\$M}$$

Uncertainties

The dominant parameters in the evaluation of this issue are the percent saving in occupational dose during normal operation, which is unlikely to be incorrect by more than a factor of ten, and the cost to the licensee, which is expected to be within a factor of 5. This implies a range in S from 100 to 30,000 man-rem/\$M and a range in total man-rem saved of 6×10^4 to 6×10^6.

CONCLUSION

Based on the value/impact score and potential reduction in occupational dose, this issue was give a high priority ranking. In resolving this issue, the staff agreed to support alternative regulatory concepts which recognize the contributions of industry self-policing programs to the extent that such programs are effective and consistent with NRC regulatory responsibilities. As a result, the staff entered into a "Coordination Plan for Radiological Protection Activities" with INPO under a "Memorandum of Agreement Between INPO and the USNRC." Under this agreement, over the two-year period outlined in the Coordination Plan, NRR staff developed a method for evaluating industry performance in radiation protection programs incorporating ALARA concepts at power reactors and observed the INPO evaluation and assistance process at a number of operating facilities.

The staff performed analyses of a number of radiological data trends as part of the effort to determine if the power reactor industry has achieved successful ALARA-integrated radiation protection programs. An analysis of these trends and portions of the supporting data bases were documented in the report, "Summary Analysis of Selected Radiological Trends at Power Reactors."[912]

Following the staff's compilation of data and evaluation of a number of trends in radiological protection at power reactors, the staff concluded that most power reactor radiation protection programs are adequately incorporating ALARA concepts and can satisfactorily perform at a level which meets the objectives of Item III.D.3.1 Thus, this issue was RESOLVED and no new requirements were established.[913]

ITEM III.D.3.2: HEALTH PHYSICS IMPROVEMENTS

The four parts of this item have been combined and evaluated together.

DESCRIPTION

Historical Background

In this TMI Action Plan[48] item, four specific items were identified for resolution: (1) Requirement for Use of Certified Personnel Dosimeter Processors; (2) Audible Alarm Dosimeter Regulatory Guide; (3) Develop Standard Performance Criteria for Radiation Survey and Monitoring Instruments; and (4) Develop Air Purifying Respirator Radioiodine Cartridge Testing and Certification Criteria. Item (2) will not be considered further since Regulatory Guide 8.28 was issued in final form prior to this evaluation. Thus, Item (2) is considered resolved.

Safety Significance

(1) Requirement for Use of Certified Personnel Dosimetry Processors

The proposed resolution would amend 10 CFR 20 to require that only nationally certified dosimetry processors be used by NRC licensees for personnel radiation dosimetry. Processors would be required to meet ANSI N13.11 (or its replacement standard) criteria for testing. Certification of processors would be performed by the National Voluntary Laboratory Accreditation Program (NVLAP) administered under the auspices of the U.S. Department of Commerce (DOC).

This issue does not specifically address core-melt accidents nor the public risk, occupational dose, or accident avoidance costs associated with such accidents. It is related to the worker's right to accurate measurements of occupational dose. The proposed resolution would require accurate and precise determinations of individual worker doses using dosimeters, readout systems, and processing procedures certified to be capable of meeting minimum criteria defined in a national standard. The administrative and regulatory limits for occupational dose would be unaffected by this work.

A draft ANSI standard (ANSI N13.11) for dosimeter testing was issued for trial use in 1978. This standard has undergone substantial testing and remains only to be finalized for issuance as a new ANSI standard. Once issued, it will form the basis for amending 10 CFR 20. Testing and certification of dosimeter processors for criteria contained in this standard will be performed by NVLAP under DOC.

(2) This item has been resolved as discussed before.

(3) Develop Standard Performance Criteria for Radiation Survey and Monitoring Instruments

Testing of radiation survey and monitoring instruments will provide a high degree of quality assurance that instruments are capable of performing intended functions under specified conditions. This will allow consistent utilization of workers without impacting current individual or collective occupational dose. A draft standard for health physics instrumentation testing (ANSI N42.17-D2) has been developed.

This standard will undergo a field trial period, using off-the-shelf instruments, to determine its adequacy. This trial period is presently estimated to continue through FY-1984 and is jointly funded by NRC and the Department of Energy (DOE) at $400,000 each. Following the trial period, a final standard will be adopted by NRC and only those instruments meeting this standard would be acceptable for use in NRC licensed facilities.

At this time, a plan for implementing the testing program has not been developed. It is anticipated, however, that independent testing laboratories would, for a fee, test instruments submitted by vendors or reactor licensees. The testing laboratories would be certified by NVLAP under DOC. Costs associated with NVLAP certification and instrument testing fees would be passed on to industry in the form of higher instrument prices.

(4) Develop Air Purifying Respirator Radioiodine Cartridge Testing and Certification Criteria

Air purifying respirators are not currently acceptable for radioiodine protection due to the lack of accepted test procedures for certifying cartridge filtering efficiency. The result is that bulky self-contained breathing apparatus (SCBA) must be worn by workers in radioiodine environments. Such environments are expected during and after core-melt accidents. The results of wearing SCBA is to substantially reduce worker efficiency due to physical stress and the relatively short working time limited by air tank capacity. Use of air purifying respirators would reduce worker stress and improve worker efficiency.

It is expected that operator dose would be unaffected by the availability of respirators. Immediately after an accident, SCBA would still be used due to immediate hazards. During long-term recovery activities respirators could be used. However, reduced external dose due to efficient use of time in radiation zones is expected to be offset by the reduced effectiveness of the respirators, compared to SCBA, in avoiding internal exposures. Criteria and test procedures for radioiodine cartridges have been under development by LASL using NRC funds. The technology has been developed and is in the process of being transferred to NIOSH. When transfer is complete, it is anticipated that NIOSH will amend 30 CFR 11 to incorporate the testing methods and criteria into respirator test and certification schedules. Respirator and cartridge manufacturers would submit products for certification testing and periodic quality control checks would be performed.

Following establishment of certification programs, NRC evaluation is anticipated regarding the need to specify the quantity and types of respirators necessary for normal and emergency use at a typical power reactor.

This issue will have no impact on public risk associated with core-melt accidents. The occupational dose impact is also considered to be zero, the benefit to workers being reduced stress, improved comfort and, consequently, better worker performance.

CONCLUSION

The above issues and their proposed resolutions do not impact public risk nor are they expected to increase or decrease occupational dose. They relate to the rights of workers to be assured of adequate radiation protection and would reduce stress during the performance of work in radiation zones. Therefore, this item is considered to be a Licensing Issue. The disposition of the four parts of this item is listed below.

ITEM III.D.3.2(1): AMEND 10 CFR 20

This Licensing Issue was evaluated in Item III.D.3.2 and was later resolved in February 1987 with the publication of a final rule on the requirement for the use of NBS-accredited personnel dosimetry processors.[1046]

ITEM III.D.3.2(2): ISSUE A REGULATORY GUIDE

This Licensing Issue was evaluated in Item III.D.3.2 above and was determined to be resolved.

ITEM III.D.3.2(3): DEVELOP STANDARD PERFORMANCE CRITERIA

The NRC/DOE project has produced several procedure manuals for future performance testing of radiation survey instruments and airborne radioactivity monitoring systems, after a certification program is established. These manuals are based on laboratory tests of sample instruments and monitoring systems using a draft of ANSI 42.17, "Performance Specifications for Health Physics Instrumentation." The IEEE Standard development working group is now using the results of the NRC/DOE project to finalize the standard for use in the accreditation program.

No further NRC action will be taken unless the instrument manufacturing industry fails to establish a satisfactory certification program within a reasonable period of time following final publication of ANSI 42.17. The final draft of this standard is under review by ANSI participants; some manufacturers' concerns still need to be resolved.

The NRC staff has taken the position that the industry should establish its own certification program and that the NRC would intervene only if the industry failed to do so, or if its program proved to be unsatisfactory. Thus, this Licensing Issue has been resolved.[954]

ITEM III.D.3.2(4): DEVELOP METHOD FOR TESTING AND CERTIFYING AIR-PURIFYING RESPIRATORS

A research project has been completed that provides experimental data and recommendations for establishing a standard test procedure and acceptance criteria for air purifying respirator cartridges and canisters used to protect workers, and simultaneously measure penetrations of radioiodine and normal iodine vapor species through beds of various charcoals. The effects of various conditions of use (bed depth, contact time, concentration, relative humidity, temperature, flowrate, and flow cycling) were studied to identify testing requirements. Recommendations for testing and approval were based on consideration of the effects of these parameters. An apparatus designed and built for testing has been delivered to NIOSH, the responsible institute for testing and certifying respiratory protection equipment. Such certification is required in 10 CFR Part 20. In 1983, the staff published NUREG/CR-3403.[969]

NIOSH certification is now available. Licensees who wish to take credit for such equipment may do so after obtaining individual authorization from NRC. Thus, this Licensing Issue has been resolved.[954]

ITEM III.D.3.3: IN-PLANT RADIATION MONITORING

The four parts of this item are listed separately below.

ITEM III.D.3.3(1): ISSUE LETTER REQUIRING IMPROVED RADIATION SAMPLING INSTRUMENTATION

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-69 was established by DL for implementation purposes.

ITEM III.D.3.3(2): SET CRITERIA REQUIRING LICENSEES TO EVALUATE NEED FOR ADDITIONAL SURVEY EQUIPMENT

DESCRIPTION

This NUREG-0660[48] item required NRR to set criteria requiring licensees to evaluate in their plants the need for additional survey equipment and radiation monitors in vital areas and requiring, as necessary, installation of area monitors with remote readout. NRR was to evaluate the need to specify the minimum types and quantities of portable monitoring instrumentation, including very high dose rate survey instruments. Operating reactors were to be reviewed for conformance with SRP[11] Section 12.3.4, "Area Radiation and Airborne Radioactivity Monitoring Instrumentation." NRR was to revise SRP Sections 12.5 and 12.3.4 to incorporate additional monitor requirement criteria.

CONCLUSION

In December 1980, the requirements for high range area and portable monitors were incorporated into Regulatory Guide 1.97, Revision 2. In July 1981, SRP[11] Sections 12.3 and 12.5 were revised to incorporate the requirements for in-plant radiation monitoring. Thus, this item was RESOLVED and new requirements were established.

ITEM III.D.3.3(3): ISSUE A RULE CHANGE PROVIDING ACCEPTABLE METHODS FOR CALIBRATION OF RADIATION-MONITORING INSTRUMENTS

DESCRIPTION

This NUREG-0660[48] item required RES to issue a rule change providing acceptable methods for calibration of radiation-monitoring instruments.

CONCLUSION

The required change was covered in the overall revision to 10 CFR 20, Paragraph 20.501(c). Thus, this item was RESOLVED and new requirements were established.

ITEM III.D.3.3(4): ISSUE A REGULATORY GUIDE

DESCRIPTION

This NUREG-0660[48] item required RES to issue a Regulatory Guide providing acceptable methods for calibration of air-sampling instruments.

CONCLUSION

Regulatory Guide 8.25 was issued in August 1980. Thus, this item was RESOLVED and new requirements were established.

ITEM III.D.3.4: CONTROL ROOM HABITABILITY

This item was clarified in NUREG-0737,[98] requirements were issued, and MPA F-70 was established by DL for implementation purposes.

ITEM III.D.3.5: RADIATION WORKER EXPOSURE

The three parts of this item have been combined and evaluated together.

DESCRIPTION

This TMI Action Plan[48] item called for the NRC to continue its efforts to improve and expand the data base on industry employees in order to facilitate possible future epidemiological studies on worker health. The three parts of this item are as follows:

(1) "Improve and expand the data base on industry employees." This item is considered important in improving a data base used by the NRC in judging the adequacy of its radiation protection standards. Meetings have been held with DOE, ORM, NCI, AIF, and officials of Canadian and British national dose registries and health statistics organizations to discuss issues related to this item. Although these meetings have resolved certain generic issues, this item is a long-term goal requiring on-going cooperation between nuclear regulators, industries, and workers.[409]

(2) "Investigate non-legislative means of obtaining employee health data." This item was completed in September 1982 following discussions about worker health data with DOE, AIF, EPRI, and officials of British and Canadian national dose registries and health statistics organizations.[409]

(3) "Include as part of the overall rewrite of 10 CFR Part 20 consideration of a requirement for licensees to collect worker medical data." This item was completed in February 1981 following a decision by the Part 20 task force not to require the collection of worker medical data.[409]

The value of this item does not lie in the reduction of public or occupational risk. Instead, it will provide data on which future regulatory decisions will be based. Therefore, this item is not directly related to public safety and is considered a licensing issue.

CONCLUSION

The disposition of the three parts of this Licensing Issue is listed below.

ITEM III.D.3.5(1): DEVELOP FORMAT FOR DATA TO BE COLLECTED BY UTILITIES REGARDING TOTAL RADIATION EXPOSURE TO WORKERS

10 CFR 20.408 requires utilities that operate nuclear power plants to submit to the NRC a report that provides identification and exposure information for each monitored individual at the time of completion of the individual's assignment or employment at a particular plant. In order to improve the processing of this worker dose data, the NRC staff developed NRC Form 439, "Report of Terminating Individual's Occupational Exposure." This new form improved and expanded the dose data base that would be needed to support possible future epidemiological studies. The NRC staff, in cooperation with HHS, plans to recommend that the Committee for Interagency Radiation Research Policy Coordination (CIRRPC) review the issue of a worker registry and epidemiologic studies and formulate recommendations. The staff concluded[954] that the NRC does not have the authority or the resources to support a worker registry or epidemiological health effects studies. Thus, this Licensing Issue has been resolved.

ITEM III.D.3.5(2): INVESTIGATIVE METHODS OF OBTAINING EMPLOYEE HEALTH DATA BY NON-LEGISLATIVE MEANS

This Licensing Issue was evaluated in Item III.D.3.5 above and was determined to be resolved.

ITEM III.D.3.5(3): REVISE 10 CFR 20

This Licensing Issue was evaluated in Item III.D.3.5 above and was determined to be resolved.

REFERENCES

11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.

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

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

202. Memorandum for G. Cunningham, et al., from K. Goller, "Proposed Amendment to Part 50 on Radiation Programs, Including ALARA," September 10, 1982.

204. NUREG-0855, "Health Physics Appraisal Program," U.S. Nuclear Regulatory Commission, March 1982.

205. NUREG-0761 "Radiation Protection Plans for Nuclear Power Reactor Licensees," U.S. Nuclear Regulatory Commission, March 1981.

206. Memorandum for L. Rubinstein from M. Ernst, "Proposed Position Regarding Containment Purge/Vent Systems," April 17, 1981.

409. Memorandum for W. Minners from W. Mills, "Prioritization of Generic Issue III.D.3.5, Radiation Worker Data Base," February 22, 1983.

912. Memorandum to T. Murley, et al., from H. Denton, "Evaluation of Industry Success in Achieving ALARA-Integrated Radiation Protection Plans - Data Trend Assessments," May 19, 1986.

913. Memorandum for V. Stello from H. Denton, "Resolution of Generic Issue III.D.3.1, 'Radiation Protection Plans,'" May 19, 1986.

954. Memorandum for V. Stello from E. Beckjord, "Closeout of TMI Action Plan Items," November 13, 1986.

969. NUREG/CR-3403, "Criteria and Test Method for Certifying Air-Purifying Respirator Cartridges and Canisters Against Radioiodine," U.S. Nuclear Regulatory Commission, September 1982.

1046. Memorandum for V. Stello from E. Beckjord, "Closeout of TMI Action Plan Item," February 27, 1987.

ITEM A-43: CONTAINMENT EMERGENCY SUMP PERFORMANCE

DESCRIPTION

Historical Background

This issue deals with a concern for the availability of adequate recirculation cooling water following a LOCA when long-term recirculation of cooling water from the PWR containment sump, or the BWR RHR system suction intake, must be initiated and maintained to prevent core-melt. This water must be sufficiently free of LOCA-generated debris and potential air ingestion so that pump performance is not impaired thereby seriously degrading long-term recirculation flow capability. The concern applies to both PWRs and BWRs. The RHR suction strainers in a BWR are analogous to the PWR sump debris screen and adequate recirculation cooling capacity is necessary to prevent core-melt following a postulated LOCA. The issue was declared a USI in January 1979 and published in NUREG-0510.[186]

The technical concerns evaluated under USI A-43 are as follows:

(1) PWR sump (or BWR RHR suction intake) hydraulic performance under post-LOCA adverse conditions resulting from potential vortex formation and air ingestion and subsequent pump failure.

(2) The possible transport of large quantities of LOCA-generated insulation debris resulting from a pipe break to the sump debris screen(s), and the potential for sump screen (or suction strainer) blockage to reduce net positive suction head (NPSH) margin below that required for the recirculation pumps to maintain long-term cooling.

(3) The capability of RHR and containment spray system (CSS) pumps to continue pumping when subjected to possible air, debris, or other effects such as particulate ingestion on pump seal and bearing systems.

The staff's proposed resolution for USI A-43 was issued for public comment on May 10, 1983. The public comment package included NUREG-0869,[1056] the staff's technical findings report NUREG-0897,[1057] proposed Regulatory Guide 1.82, Revision 1, and proposed SRP[11] Section 6.2.2, Revision 4, "Containment Heat Removal Systems." A summary of the public comments received and the staff's response are contained in Appendix A of NUREG-0869,[1056] Revision 1.

CONCLUSION

In October 1985, the resolution of USI A-43 was presented to the Commission in SECY-85-349.[1060] The staff is implementing the resolution of USI A-43 through the following actions:

(1) The staff's technical findings (NUREG-0897, Revision 1)[1057] were published for use as an information source by applicants, licensees, and the staff.

(2) SRP[11] Section 6.2.2 and Regulatory Guide 1.82[1058] were revised to reflect the staff's technical findings reported in NUREG-0897, Revision 1. This revised licensing guidance applies only to reviews of: (a) future construction permit applications and preliminary design approvals (PDAs); (b) final design approvals (FDAs) for standardized designs which are intended for referencing in future construction permit applications that have not received approval; and (c) applications for licenses to manufacture. This revised guidance became effective 6 months after issuance of Regulatory Guide 1.82, Revision 1.

(3) Generic Letter 85-22[1059] (for information only) was sent to all holders of an operating license or construction permit outlining the safety concerns regarding potential debris blockage and recirculation failure due to inadequate NPSH. It was recommended (but not required) that licensees utilize Regulatory Guide 1.82,[1058] Revision 1, as guidance for conduct of the 10 CFR 50.59 analysis for future plant modifications involving replacement of insulation on primary system piping and/or equipment. If, as a result of NRC staff review of licensee actions associated with replacement or modification to insulation, the staff decides that SRP[11] 6.2.2, Rev. 4 and/or Regulatory Guide 1.82,[1058] Rev. 1, criteria should be (or should have been) applied by the licensees, and the staff seeks to impose these criteria, then NRC will treat such actions as plant-specific backfits pursuant to 10 CFR 50.109.

Thus, this issue was RESOLVED and new requirements were established.

REFERENCES

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

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

1056. NUREG-0869, "USI A-43 Regulatory Analysis," U.S. Nuclear Regulatory Commission, (Revision 1) October 1985.

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

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

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

1060. SECY-85-349, "Resolution of Unresolved Safety Issue A-43, 'Containment Emergency Sump Performance,'" October 31, 1985.

ITEM A-46: SEISMIC QUALIFICATION OF EQUIPMENT IN OPERATING PLANTS

DESCRIPTION

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically-induced loads and perform the intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this issue was to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants. This guidance will concern equipment required to safely shut down the plant as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions. Also, explicit guidelines will be established for use in requalifying equipment whose qualification was found to be inadequate. This issue was declared a USI in February 1981 and published in NUREG-0705.[44] A detailed action plan for resolving this issue was published in NUREG-0649,[1061] Rev. 1.

CONCLUSION

Work completed on this USI resulted in the publication of NUREG/CR-3017,[1063] NUREG/CR-3875,[1064] NUREG/CR-3357,[1065] NUREG/CR-3266,[1066] NUREG-1030,[919] and NUREG-1211.[1067] The resolution of USI A-46 was mainly based on work completed by the Seismic Qualification Utility Group (SQUG) and EPRI using the seismic and test experience data approach and reviewed and endorsed by the Senior Seismic Review and Advisory Panel (SSRAP) and the NRC staff. The scope of the review was narrowed down to equipment required to bring each affected plant to hot shutdown and maintain it there for a minimum of 72 hours. A walk-through of each plant is required to inspect equipment in the scope. Evaluation of equipment will include: (a) adequacy of equipment anchorage; (b) functional capability of essential relays; (c) outliers and deficiencies (i.e., equipment with non-standard configurations); and (d) seismic systems interaction. This issue was RESOLVED and requirements were issued in Generic Letter 87-02[1069] in February 1987.

REFERENCES

44. NUREG-0705, "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plant Stations," U.S. Nuclear Regulatory Commission, February 1981.

919. NUREG-1030, "Seismic Qualification of Equipment in Operating Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1987.

1061. NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Revision 1) September 1984.

1063. NUREG/CR-3017, "Correlation of Seismic Experience Data in Non-Nuclear Facilities With Seismic Equipment Qualification in Nuclear Plants (A-46)," U.S. Nuclear Regulatory Commission, August 1983.

1064. NUREG/CR-3875, "The Use of In-Situ Procedures for Seismic Qualification of Equipment in Currently Operating Plants," U.S. Nuclear Regulatory Commission, June 1984.

1065. NUREG/CR-3357, "Identification of Seismically Risk Sensitive Systems and Components in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, June 1983.

1066. NUREG/CR-3266, "Seismic and Dynamic Qualification of Safety-Related Electrical and Mechanical Equipment in Operating Nuclear Power Plants," U.S. Nuclear Regulatory Commission, September 1983.

1067. NUREG-1211, "Regulatory Analysis for Resolution of Unresolved Safety Issue A-46, 'Seismic Qualification of Equipment in Operating Plants,'" U.S. Nuclear Regulatory Commission, February 1987.

1069. NRC Letter to All Holders of Operating Licenses Not Reviewed to Current Licensing Criteria on Seismic Qualification of Equipment, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46 (Generic Letter 87-02)," February 19, 1987.

ITEM A-49: PRESSURIZED THERMAL SHOCK

DESCRIPTION

Neutron irradiation of reactor pressure vessel weld and plate materials decreases the fracture toughness of the materials. The fracture toughness sensitivity to radiation-induced change is increased by the presence of certain materials such as copper. Decreased fracture toughness makes it more likely that, if a severe overcooling event occurs followed by or concurrent with high vessel pressure, and if a small crack is present on the vessel's inner surface, that crack could grow to a size that might threaten vessel integrity.

Severe pressurized overcooling events are improbable since they require multiple failures and improper operator performance. However, certain precursor events have happened that could have potentially threatened vessel integrity if additional failures had occurred and/or if the vessel had been more highly irradiated. Therefore, the possibility of vessel failure due to a severe pressurized overcooling event cannot be ruled out. In December 1981, this issue was declared a USI in SECY-81-687 and was later published in the NRC 1982 Annual Report. A detailed action plan for resolving this issue was published in NUREG-0649,[1061] Rev. 1.

CONCLUSION

An Advance Notice of Proposed Rulemaking was submitted to the Commission in SECY-83-288[1073] and was later approved in January 1984.[1074] After public comments were addressed, the final rule (10 CRR 50.61) on pressurized thermal shock was approved by the Commission in July 1985. Regulatory Guide 1.154[1068] was later published in February 1987. Thus, this issue was RESOLVED and new requirements were established.

REFERENCES

1061. NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Revision 1) September 1984.

1068. Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, January 1987.

1073. SECY-83-288, "Pressurized Thermal Shock (PTS) Rule," July 15, 1983.

1074. Memorandum for W. Dircks from S. Chilk, "SECY-83-288, 'Proposed Pressurized Thermal Shock (PTS) Rule,'" January 13, 1984.

ITEM B-6: LOADS, LOAD COMBINATIONS, STRESS LIMITS

DESCRIPTION

Historical Background

This issue was identified as a generic problem in NUREG-0471[3] and concerns the design of pressure vessels and piping systems components which must be designed to accommodate individual and combined loads due to normal operating conditions, system transients, and postulated low probability events (accidents and natural phenomena). This issue became more controversial in recent years because postulated large LOCA and SSE loads were each increased by a factor of 2 or more to account for such phenomena as asymmetric blowdown and because better techniques for defining loading have been developed. The work efforts to investigate and establish a position on dynamic response combination methodology was completed and reported in NUREG-0484,[135] Revision 1. NUREG-0800,[11] Section 3.9.3, was revised to reflect the new position on load combinations and stress limits.[136] SEB concluded from studies completed (NUREG/CR-2039[549] and NUREG/CR-1890[550]) that seismic loads and LOCA and SRV loads on containment structures should continue to be combined using the absolute sum method.[137] Hence, the only work remaining is research on decoupling LOCA and SSE events. It is on this aspect that this prioritization focuses. Reports on two investigations addressing this issue have been released: NUREG/CR-2136[62] and NUREG/CR-2189.[63]

The Code of Federal Regulations requires that structures, systems, and components that affect the safe operation of nuclear power plants be designed to withstand combinations of loads that can be expected to result from natural phenomena, normal operating conditions, and postulated accidents. An example load combination requirement mandated for nuclear power plants includes coupling the effects of SSE with a LOCA. In a recent evaluation, these combined loads were increased to further account for phenomena such as asymmetric blowdowns in PWRs and because improved techniques for defining loading have been developed.

These changes have raised questions concerning implementation of new regulations, increased construction costs, increased radiation exposure of maintenance crews performing increased inspection and maintenance actions, and reduced reliability of stiffer systems under normal operating transients.

Possible Solutions

Research Information Letter No. 117,[65] in addressing the probability of large LOCA-induced earthquakes, identifies the following results:

(1) Through-wall cracks are about a million times more likely to occur than double-ended guillotine breaks. This supports the leak before break hypothesis.

(2) Fatigue crack growth due to all transients, including earthquakes, is an extremely unlikely mechanism for inducing a large LOCA. The contribution

of earthquakes to the occurrence of this unlikely event is a small percentage of the total probability.

(3) An upper bound estimate of the probability of asymmetric blowdown loads (resulting from rupture of in-cavity piping) due to direct and indirect mechanisms is 10^{-4} over the 40-year plant life, the primary contribution to this estimate being indirect seismically induced asymmetric blowdown. It is felt that the best estimate of the probability is several orders of magnitude lower.

While the described research was performed on PWRs, it is assumed that BWRs are similar for this analysis. This assumption may need revision if additional studies for BWRs are completed.

The proposed resolution for this issue is to decouple the SSE-LOCA load requirements. This would permit: (1) the removal of some snubbers, (2) the removal of pipe whip restraints, and (3) the deletion of the requirements for asymmetric blowdown analyses for forward-fit plants which would eliminate the additional stiffening of the reactor pressure vessel.

PRIORITY DETERMINATION

Assumptions

In the quantitative analysis of this issue by PNL,[64] it was assumed that there will be a small reduction in risk to the public due to the removal of appropriate snubbers in systems designed to withstand SSE + LOCA-induced loads. This reduction in system stiffeners should help preclude potential lockup of snubbers during normal operating transients, thus reducing large stresses on piping under normal operating conditions. The actual removing of equipment (snubbers and pipe restraints) will introduce an added (one-time) occupational dose for those plants having the devices installed. However, the deleted snubbers will result in a reduction in occupational exposure because inspection and maintenance will no longer be neccessary on these deleted items. The removal of the pipe restraints will improve the access to many equipment items and, as a result, will reduce plant personnel time in high radiation areas for maintenance and inspection, providing a further reduction in occupational exposure.

The risk reduction and cost estimates are based on all reactors built since 1972 or yet to be constructed. Reactors constructed prior to 1972 did not have design requirements which included SSE, LOCA, and pipe cooling considerations.

Frequency/Consequence Estimate

It has been suggested that removing the snubbers required for the combined LOCA and SSE loads would reduce the stiffness and potential lockup of the snubbers during normal operation. This would result in a reduction in the probability of pipe rupture during normal operating transients (e.g., startup, thermal transients, etc.). The best estimate, by engineering judgment, is that the probability of pipe rupture would be reduced by 25% across the board. This estimate reduces S_1, S_2, and S_3, (the initiating event probabilities for the PWR) and the S value for the BWR by 25%. These changes, applied to the dominant cut sets, produced a change in core-melt frequency which in turn reduced the frequency of each release category (e.g., PWR-1). The computed reduction in core-melt frequency is 6.4×10^{-6}/RY for PWRs and 1.2×10^{-6}/RY for BWRs. The reduction in frequency of the various release categories results in a public risk reduction for PWRs of 13 man-rem/RY and a risk reduction for BWRs of 8.2 man-rem/RY. This public risk reduction when applied to the total reactor population lifetime results in a risk reduction of 31,700 man-rem for PWRs and 8,500 man-rem for BWRs, for a total of 40,200 man-rem.

Cost Estimate

Industry Cost: The total plant user cost estimate is based on the cost of engineering efforts to design the change, the labor costs to incorporate the change, and the increase in test and maintenance cost to maintain the equipment for the remaining plant life.

The implementation manpower requirements were based upon confirmatory analyses performed at PNL in conjunction with reviews prior to the granting of operating licenses.

The labor requirements per reactor for analyses and craft work is computed to be 360 man-weeks for PWRs and 391 man-weeks for BWRs, of which 250 man-weeks are utilized for the analysis of each type of reactor. This results in backfit implementation costs of $39.6M ($27.6M for PWRs and $12M for BWRs) and forward-fit costs of $17.3M ($11.5M for PWRs and $5.8M for BWRs). Therefore, the total industry implementation cost is $56.9M.

It is assumed for the maintenance and operating costs that approximately 50% of the pipe snubbers associated with LOCA and SSE as well as many unnecessary pipe restraints can be removed following leak before break concept. As reported in NUREG/CR-2136,[62] there are approximately 800 snubbers in a typical PWR and 950 snubbers in a typical BWR. If we assume that 50% are removed, then the number of snubbers removed is 400 in a PWR and 475 in a BWR.

Using labor hour estimates from NUREG/CR-2800,[64] it is calculated that a reduction in labor costs will be attained due to the decrease in the number of snubbers to be inspected and maintained. In addition, there will be improved access to pumps, valves, etc., due to the removal of pipe whip restraints.

The total estimated saving in labor time (inspection, testing, and maintenance) resulting from the deletion of snubbers and pipe restraints is calculated to be 1,120 man-hrs/RY or 28 man-weeks/RY for PWRs and 1,140 man-hrs/RY or 36 man-weeks/RY for BWRs.

This results in an industry cost for operation and maintenance of -$53,800/reactor for PWRs and -$69,000/reactor for BWRs. For all reactors built since 1972, it results in operation and maintenance costs of -$57.9M for backfit PWRs, -$77.5M for forward-fit PWRs, -$26.7M for backfit BWRs, and -$47.6M for all forward-fit BWRs. Thus, the total maintenance and operating costs are -$209.7M. Therefore, the total industry cost for this issue is as follows:

Best Estimate	-$152.3M
Upper Bound	$ 5.6M
Lower Bound	-$300.0M.

NRC Cost: The NRC costs for this issue are based upon the time used to review the proposed changes prior to the implementation of equipment modifications. It is estimated that the following support will be required:

Generic issue resolution	20 man-weeks
Backfit plant implementation	15 man-weeks/plant
Forward-fit plant implementation	10 man-weeks/plant.

These manpower expenditures result in an NRC cost for development and implementation of \$3.1M. There will be no change in NRC cost due to the review of operation and maintenance resulting from this change. Therefore, the total NRC cost for this issue is as follows:

Best Estimate	\$3.5M
Upper Bound	\$5.2M
Lower Bound	\$1.8M

Value/Impact Assessment

Based upon the best estimates of total risk reduction and industry NRC cost, the value/impact score is given by:

$$S = \frac{40{,}200 \text{ man-rem}}{\$3.5M + (-\$152.3M)}$$

$$= -270 \text{ man-rem/\$M.}$$

The negative value results from the reduced costs of operation and maintenance because of the deleted snubbers and pipe restraints.

Other Considerations

Of further importance to this issue is the reduction in ORE brought about by the reduction of work time to perform ISI in a radiation environment. An accumulated exposure of 1,100 man-rem/plant for PWRs and 1,410 man-rem/plant for BWRs is expected in the removal of snubbers and pipe restraints.[64] For all backfit plants, this results in an exposure of 4.5×10^4 man-rem for all PWRs and 2.26×10^4 man-rem for BWRs. The removal of snubbers and the elimination of pipe restraint removal to accomplish pipe inspections is estimated to save 1,120 man-hours/year/plant for PWRs and 1,440 man-hours/year/plant for BWRs in maintenance and operation time in radiation environments. For all applicable reactors' lifetimes, this accumulated exposure reduction is calculated to be 6.77×10^5 man-rem for PWRs and 3.68×10^5 man-rem for BWRs. This results in a total reduction in ORE of 9.8×10^5 man-rem.

CONCLUSION

This issue was given a high priority issue because both risk and cost had a large potential for reduction. Since no pipe failure due to excessive restraint had been reported up to the time of prioritization, the estimated 25% reduction in pipe break frequency and public risk may be overstated. It was concluded that, even if the risk reduction were less and some costs were incurred rather than saved, the priority would still be high.

Research completed for W and CE plants showed that the yearly probability of having a large LOCA induced directly by seismic loads was no greater than 10^{-10}; the yearly probability of having a LOCA induced indirectly by structure or support failures under seismic loads was found to be 10^{-6}. Based on the research results, the staff proposed a position that would decouple SSE and LOCA for all PWR primary loops. Research work for decoupling of LOCA and SSE loads for GE plants was also performed; indications were that pipe rupture probabilities in GE reactor coolant loops are substantially greater than in any of the PWR loops. In addition, a limited application of the leak-before-break hypothesis for PWR main coolant loops in 16 W Owners' Group plants, based on deterministic fracture mechanics analysis, was approved in Generic Letter 84-04. For decoupling of SSE and LOCA loads, the probability of a LOCA occurrence due to an earthquake was to be addressed in the resolution of Issue 119.1, "Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads," with a possible revision to SRP[11] Section 3.9.3. In January 1987, all staff work on the resolution of Item B-6 was terminated because of the parallel effort in addressing decoupling of SSE and LOCA loads in the resolution of Issue 119.1.[1041] Thus, the resolution of Item B-6 is covered in Issue 119.1.

REFERENCES

3. NUREG-0471, "Generic Task Problem Descriptions (Categories B, C, and D)," U.S. Nuclear Regulatory Commission, June 1978.

11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.

62. NUREG/CR-2136, "Effects of Postulated Event Devices on Normal Operation of Piping Systems in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, May 1981.

63. NUREG/CR-2189, "Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant," U.S. Nuclear Regulatory Commission, September 1981.

64. NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," U.S. Nuclear Regulatory Commission, February 1983.

65. Memorandum for H. Denton from R. Minogue, "Research Information Letter No. 117 - Probability of Large LOCA Induced by Earthquakes," April 10, 1981.

135. NUREG-0484, "Methodology for Combining Dynamic Responses," U.S. Nuclear Regulatory Commission, May 1980.

136. Memorandum for W. Minners from R. Bosnak, "Comments on Generic Issue B-6," August 26, 1982.

137. Memorandum for W. Minners from F. Schauer, "Generic Issue B-6," September 2, 1982.

549. NUREG/CR-2039, "Dynamic Combinations for Mark II Containment Structures," U.S. Nuclear Regulatory Commission, June 1982.

550. NUREG/CR-1890, "ABS, SRSS and CDF Response Combination Evaluation for Mark III Containment and Drywell Structures," U.S. Nuclear Regulatory Commission, March 1982.

1041. Memorandum for K. Kniel from R. Bosnak, "Request for Subsumption of Generic Issue B-6 (GI B-6) Into Generic Issue 119.1 (GI 119.1)," January 8, 1987.

ISSUE 67: STEAM GENERATOR STAFF ACTIONS

DESCRIPTION

Following the SGTR event at Ginna on January 25, 1982, increased staff effort was placed on developing means to mitigate and reduce steam generator tube degradations and ruptures. To meet these objectives, a dual approach was taken. The first approach was to develop staff requirements to be implemented by the licensees. The proposed staff requirements are evaluated in Issue 66. In addition to these proposed requirements, the staff identified and recommended certain staff actions. The status of these staff's actions as determined in this evaluation are listed in Table 3.67-1. For reference proposes, the sub-item numbers are consistent with the staff action numbers provided in a DL memorandum.[752] These items are also included in the CRGR review package[753] and EDO recommendations to the Commission.[753,757,758] The following is a summary of the evaluation of the 16 parts of this issue.

(a) Three of the proposed staff actions should be considered as Licensing Issues:

5.1 Reassessment of Radiological Consequences
5.2 Reevaluation of SGTR Design Basis
10.0 Supplemental Tube Inspections

(b) Two of the proposed staff actions are Regulatory Impact issues that could provide cost-benefits to the NRC and industry:

2.1 Integrity of Steam Generator Tube Sleeves
8.0 Denting Criteria

(c) Nine of the proposed staff actions are considered part of ongoing staff activities and no new staff efforts need be initiated:

3.1 Steam Generator Overfill
3.2 Pressurized Thermal Shock
3.3 Improved Accident Monitoring
3.4 Reactor Vessel Inventory Measurement
4.1 RCP Trip
4.2 Control Room Design Review
4.3 Emergency Operating Procedures
6.0 Organizational Responses
9.0 Reactor Coolant System Pressure Control

(d) The improved Eddy Current Tests (Item 67.7.0) recommendation is ranked as a MEDIUM priority issue principally because of potential reductions in ORE. The remaining proposed staff action (Item 67.5.3) is in the DROP category and is not recommended for further consideration.

The basis for each of the recommended staff actions is provided in separate evaluations below.

TABLE 3.67-1

Sub-Item	Staff Action	Priority*	MPA No.
67.2.1	Integrity of Steam Generator Tube Sleeves	RI(135)	N/A
67.3.1	Steam Generator Overfill	USI A-47, 1.C.1	N/A
67.3.2	Pressurized Thermal Shock	USI A-49	N/A
67.3.3	Improved Accident Monitoring	NOTE 3(a)	A-17
67.3.4	Reactor Vessel Inventory Measurement	II.F.2	F-26
67.4.1	RCP Trip	II.K.3(5)	G-01
67.4.2	Control Room Design Review	1.D.1	F-08
67.4.3	Emergency Operating Procedures	1.C.1	F-05
67.5.1	Reassessment of Radiological Consequences	LI	N/A
67.5.2	Reevaluation of SGTR Design Basis	LI	N/A
67.5.3	Secondary System Isolation	DROP	N/A
67.6.0	Organizational Responses	III.A.3	N/A
67.7.0	Improved Eddy Current Tests	135	N/A
67.8.0	Denting Criteria	RI(135)	N/A
67.9.0	Reactor Coolant System Pressure Control	USI A-45,1.C.1(2, 3)	F-04, F-05
67.10.0	Supplemental Tube Inspections	LI	N/A

*For a description of the terms used for priority, see Table II in the Introduction.

ITEM 67.2.1: INTEGRITY OF STEAM GENERATOR TUBE SLEEVES

DESCRIPTION

Historical Background

This item is Recommendation 2.1 of the DL memorandum[752] and called for the staff to develop a SRP[11] to clarify staff positions on the materials design, fabrication, installation, examination, and inspection of steam generator tube sleeves.

Safety Significance

At the present time, there is no specific SRP[11] to direct the staff/industry reviews related to the design, installation, and inspection of tube sleeves. The SRP[11] would provide an acceptable means to meet GDC 14 and GDC 32 of 10 CFR 50, Appendix A.

PRIORITY DETERMINATION

Consequence Estimate

The public risk reduction that can be attributed to this recommendation is not quantifiable. Some small improvement in the effectiveness of the sleeves to perform their intended function (i.e., assure retention of structural integrity of degraded tubes) would result from improved guidance.

Cost Estimate

The major reason for improved guidance is reduced cost. The estimated cost to develop the SRP[11] is 3 man-months of NRC staff time ($25,000). We estimate that 25% of the operating and planned PWRs (22 plants) will require tube sleeve modifications. The SRP[11] may reduce plant-specific reviews from 2 man-months to 1 man-month and is expected to also reduce industry man-power requirements by approximately the same amount. The SRP[11] would, therefore, result in a NRC cost saving of $158,000 and an industry cost saving of $183,000. The combined NRC and industry cost saving is estimated to be $341,000.

CONCLUSION

A small public risk reduction is perceived from development of an SRP[11] on steam generator tube sleeves. However, the SRP would be cost-effective in that it would reduce NRC review cost and industry costs associated with the design, installation, and inspection requirements for tube sleeves. The earlier the SRP[11] is developed, the greater the cost saving. Therefore, this issue is classified as a Regulatory Impact issue that will be addressed in the resolution of Issue 135.[1075]

ITEM 67.3.1: STEAM GENERATOR OVERFILL

DESCRIPTION

Historical Background

This item is Recommendation 3.1 of the DL memorandum[752] and called for the the NRC to select a small number of PWRs representing the PWR spectrum of designs and determine the potential for, and consequences of, steam generator overfill as a result of a SGTR. This recommendation is closely related to Items 67.5.1, 67.5.2, and 67.9. Further NRC or licensee actions should be determined based on the results of these studies. The recommendation as addressed herein does not consider potential steam generator overfill resulting from control system failures. Steam generator overfill via control systems failures are being evaluated simultaneously under USI A-47. Issue 37 (Steam Generator Overfill and Combined Primary and Secondary Blowdown) and Issue 56 (Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event) are also related issues.

Safety Significance

Following an SGTR, the affected steam generator could fill up to the steamline safety valve due to primary-to-secondary leakage from continued operation of the safety injection pumps. The safety valve may lift at successively lower pressures and fail to fully reseat. The failure to completely reseat could contribute to steam generator overfill by lowering the damaged steam generator pressure, thus raising the differential pressure across the broken tube and sustaining the leakage despite reduced primary system pressure. Failure of the valve to reseat would also provide a direct pathway for release of radioactive primary water to the environment. This sequence of events is beyond the design basis for SGTR events in SRP[11] Section 15.6.3 to establish that the radiological consequences meet 10 CFR 100.

For the B&W OTSG design in particular, it may not be possible to stop the primary-to-secondary leakage in an SGTR while maintaining the RCS in a subcooled state. The increased tendency for the OTSG leakage to continue throughout the event is a result of the tubes being directly exposed to the OTSG steam space. Generally, the emergency procedures instruct the operator to discharge steam to the atmosphere or, if available, to the condenser to control level in the damaged SG as necessary. In at least one B&W plant, however, if the water supply for safety injection pumps is approaching a minimum level or if the offsite radiological consequences are becoming excessive, the OTSG is allowed to completely fill, thus terminating the leakage. The number of B&W plants that permit filling of the OTSG is not known at present. We do not believe the potential for prolonged leakage and the associated offsite radiological consequences have been factored into OR or NTOL FSAR SGTR accident analyses. (See Item 67.5.2).

Possible Solutions

Solutions could involve improved RCS pressure control to reduce the differential pressure and leakage across the broken SG tube (primary to secondary), and/or improved EOPs to preclude overfill. The above measures are discussed in response to staff recommendations concerning RCS pressure control and EOPs. (See Items 67.9.1 and 67.4.3). With regard to the concern that the steam lines cannot support the dead-weight load if the lines are filled with water, additional supports or stronger steam lines could resolve this aspect of the concern.

PRIORITY DETERMINATION

Cost Estimate

The NRC cost would be dependent on the number of PWRs selected for this study and the design variations within this selected group.

Other Considerations

Following the Ginna event, concerns were raised relative to the potential for failing the steam lines under the additional dead-weight load, if the steam lines are filled with water as a result of SG overfill. (The Point Beach SGTR, which was a relatively low leak rate, resulted in a near overfill condition.)[755] Should the steam lines fail, the SGTR could become a LOCA outside containment. However, analyses[753] conducted for 4 plants indicate that the steam lines are unlikely to fail under the additional dead-weight load.

Accordingly, the staff's risk analyses[753] assume a conditional probability of steam line break, given an SG overfill, of 10^{-3} which is believed to be reasonably conservative. If the steam lines were redesigned to withstand an overfill condition, the analysis[753] would indicate a reduction in core-melt frequency of 1.2×10^{-7}/RY.

The consequences resulting from failure of the steam lines by overfilling the steam generators is assumed to involve releases typical of a PWR Category 4 release. Exposure is calculated assuming a typical mid-West meteorology and a population density of 340 persons/square-mile within a 50-mile radius of the plant. The potential public risk reduction is therefore $[(1.2 \times 10^{-7})(2.7 \times 10^{6})]$ man-rem/RY or 3.2×10^{-1} man-rem/RY. Considering an average remaining plant life of 24 years, the public risk reduction is 8 man-rem/reactor.

CONCLUSION

This item encompasses several considerations related to steam generator overfills and is closely related to staff studies identified in Items 67.5.1, 67.5.2, and 67.9. The primary concern (mitigation of a steam generator overfill) is part of the following ongoing staff programs: (1) USI A-47 and (2) NUREG-0737,[98] Item 1.C.1, "Emergency Operating Procedures." (See Item 67.4.3). Therefore, the SG overfill issue is covered by the above-stated ongoing staff programs.

Rupture of steam lines as a result of a steam generator overfill is a secondary concern predicated on the condition that an overfill occurs. The public risk associated with rupture of the steam lines is low and strengthening of the steam lines is considered a LOW priority.

ITEM 67.3.2: PRESSURIZED THERMAL SHOCK

DESCRIPTION

Historical Background

This item is Recommendation 3.2 of the DL memorandum[752] and called for the staff to address the effects of RCS flow stagnation associated with isolation of a steam generator in the Pressurized Thermal Shock program (USI A-49).

Safety Significance

During the Ginna SGTR event, the affected steam generator was isolated and the RCPs were tripped. As a result, the flow in the 'B' Reactor Coolant Loop was reduced to a few hundred gallons per minute while cold high pressure injection water was being injected into the loop. The cold leg piping apparently experienced a cooldown of approximately 260°F in 30 minutes. The reactor vessel apparently did not experience this rapid cooldown since the flow in the cold leg was in the reverse direction, that is, from the reactor vessel towards the steam generator. Other events, as discussed in NUREG-0916,[754] resulting in a steam generator isolation and continued safety injection could result in adding cold water to the reactor vessel.

CONCLUSION

The probability, consequences, and resolution of the above events were addressed in USI A-49.

ITEM 67.3.3: IMPROVED ACCIDENT MONITORING

DESCRIPTION

Historical Background

This item is Recommendation 3.3 of the DL memorandum[752] and called for the staff to address the accident monitoring weaknesses of the type observed at Ginna by implementation of Regulatory Guide 1.97[55] and the Safety Parameter Display System.

Safety Significance

During the event at Ginna, several weaknesses in accident monitoring were apparent. These include: (1) non-redundant monitoring of RCS pressure; (2) failure of the position indication for the steam generator relief and safety valves; and (3) the limited range of the charging pump flow indicator for monitoring charging flow during accidents. These conditions make it more difficult for correct operator action in response to such events.

Possible Solution

Had Regulatory Guide 1.97[55] been implemented at Ginna before the January 1982 event, the monitoring of the event would have been substantially improved and there would have been more assurance of correct operator actions. Improved accident monitoring would also have improved the NRC's ability to assess the plant status and the appropriateness of the licensee's actions and recommendations.

CONCLUSION

The recommendation was resolved by MPA A-17 and the resolution was issued in Supplement 1 to NUREG-0737 (Generic letter No. 82-33).[376]

ITEM 67.3.4: REACTOR VESSEL INVENTORY MEASUREMENT

DESCRIPTION

Historical Background

This item is Recommendation 3.4 of the DL memorandum[752] and called for implementation of TMI Action Plan Item II.F.2 because it would have substantially improved the Ginna situation by ensuring that steam bubble formation in the reactor vessel upper head could be more accurately monitored.

Safety Significance

During the Ginna SGTR event, the formation of a steam bubble in the reactor vessel upper head significantly complicated the course of the event. The uncertainty about the bubble size was a significant factor in the operator's decisions to continue safety injection beyond the point when termination is called for in the emergency procedures.

Possible Solution

Implementation of NUREG-0737,[98] Item II.F.2.

CONCLUSION

Following Commission approval for implementation of Item II.F.2, letters to individual licensees and orders to B&W licensees and ANO-2 were issued on December 10, 1982.[491] This issue is part of Item II.F.2 which is being implemented as MPA F-26.

ITEM 67.4.1: REACTOR COOLANT PUMP TRIP

DESCRIPTION

Historical Background

This item is Recommendation 4.1 of the DL memorandum[752] and called for the NRC staff to develop requirements for licensees to provide RCP trip criteria that will ensure continued forced RCS flow during steam generator tube breaks, up to and including the design basis tube rupture.

Safety Significance

Analyses indicate that continued operation of the RCPs following a range of small LOCAs could lead to excessive inventory loss for which the high pressure injection system would be unable to compensate. Generally, the range of break size of concern is from 0.02 to 0.2 ft^2 (2 to 5 inches equivalent diameter). The interim position (documented in NUREG-0623)[97] requires manual tripping of the RCPs on the symptoms of a small LOCA (i.e., a safety injection signal and low RCS pressure).

CONCLUSION

This recommendation is being developed under NUREG-0737, Item II.K.3(5)[98] and is being implemented as MPA G-1.

ITEM 67.4.2: CONTROL ROOM DESIGN REVIEW

DESCRIPTION

This item is Recommendation 4.2 of the DL memorandum.[752] As a result of a review of the Ginna control room following the tube rupture, several items related to the event were identified that are contrary to good human factors engineering principles. These items should be reviewed by HFEB as part of the detailed control room design review required by NUREG-0737.[98] This information should be used in the basis for a study to determine what changes can be made to improve control room designs.

CONCLUSION

It has been determined that items identified at Ginna have been covered in the work to be done for the TMI Action Plan Item I.D.1 control room reviews, thus assuring that these items will be factored into all Item I.D.1 control room design reviews. This recommendation will be resolved as part of NUREG-0737,[98] Item I.D.1 and is being implemented as MPA F-08.

ITEM 67.4.3: EMERGENCY OPERATING PROCEDURES

DESCRIPTION

Historical Background

This item is Recommendation 4.3 of the DL memorandum.[752] The purpose is to ensure that newly-developed EOPs consider the experiences from the Ginna SGTR event. PSRB should review the items listed below prior to emergency procedure implementation for inclusion in its review plan. This staff effort should be considered in conjunction with ongoing work under NUREG-0737,[98] Item I.C.1.

- RCP Restart
- Availability of Faulted SG Safety and Relief Valve

- Multiple and Second Order Failures
- Bubble Formation
- Cooling Faulted SG
- Cooling Intact SG
- Safety Injection Pump Termination and Restart Criteria
- Procedure Format and Clutter
- Criteria for Natural Circulation Determination
- Accommodation of Plant Differences from Reference Plant in Emergency Procedure Development
- Rapid Determination and Isolation of Faulted SG and Timely Depressurization of RCS to Minimize RCS Inventory Loss and Releases
- MSIV Closure During Plant Cooldown
- Use of Charging and Letdown Systems
- Operation of the RCP in the Damaged Loop
- Operation of Loop Isolation Valves
- Use of Pressurizer PORV
- Potential Complicating Events
- Site-Specific Operator Training
- SG Level Control for CE Plants

Safety Significance

The above list includes transients and plant conditions that form the basis of many of the emergency procedures, reliability analyses, human factors engineering, crisis management, and operator training. Plant conditions may exist, in addition to those pertinent to design bases, which could prevent proper operator actions during such events/conditions and possibly pose a serious threat to reactor safety.

Possible Solution

The solution to this recommendation is to consider the Ginna event in the development of EOPs.

PRIORITY DETERMINATION

Guidance for the evaluation and development of procedures for transients and accidents is covered by Item I.C.1 of NUREG-0737.[98] Some of the items in the above list are explicitly included in the review requirements of Item I.C.1. Other items in the list are believed to be implicitly within the intent of Item I.C.1 in that the availability of systems under expected conditions (like Ginna) should be used in developing diagnostic guidance for operator and procedural development.

CONCLUSION

This recommendation is covered in Item I.C.1 of NUREG-0737[98] and is being implemented as MPA F-05.

ITEM 67.5.1: REASSESSMENT OF RADIOLOGICAL CONSEQUENCES

DESCRIPTION

Historical Background

This item is Recommendation 5.1 of the DL memorandum[752] and called for the staff to reassess SGTR accidents to determine the effects of releases made for periods substantially longer and via other release points than those previously analyzed. These analyses should specifically address the applicability of the assumptions in SRP[11] Section 15.6.3 and address the costs and benefits of requiring revised analyses by licensees. This issue is closely related to Items 67.5.2 and 67.3.1.

Safety Significance

Public risk from an SGTR, even considering steam generator overfill, is considered low for typical PWRs. This low risk is expected to remain valid even if new source term results are applied. However, the safety significance of this issue is derived from concern over the number of SGTR events and potential for exceeding the bounds of the analyses that are currently required in SRP[11] Section 15.6.3 to demonstrate that doses from SGTR events will not exceed 10 CFR 100.

PRIORITY DETERMINATION

SRP[11] Section 15.6.3 does not address a steam generator overfill in the SGTR scenario. In addition, termination of the leak from an SGTR within 30 minutes, as assumed in typical PWR FSARs, may be non-conservative and not consistent with operating experiences. Therefore, implementation of this recommendation will allow the staff to upgrade SRP Section 15.6.3 and provide a better understanding and means to assess future SGTR events in operating plants relative to the consequence limits in 10 CFR 100.

Information generated from implementation of this recommendation will also assist licensees in their understanding of similar events and help determine the course of action needed to mitigate the consequences of SGTRs and overfilling of the steam generators.

CONCLUSION

Implementation of this recommendation is not expected to result in significant overall risk reduction for the public. Therefore, with regard to potential risk reduction to the public, this recommendation is considered low priority. However, AEB considers this recommendation a Licensing Improvement issue and recommends the reassessment. DST agrees that a "best estimate" analysis modeled after plant experiences, like Ginna, could be beneficial in more realistically determining the risk and conservatisms inherent in the current SRP requirements. If this limited scope comparison of the SRP model with a best estimate analysis is followed, this issue could be considered as an improvement to current licensing positions (a licensing issue.)

ITEM 67.5.2: REEVALUATION OF SGTR DESIGN BASIS

DESCRIPTION

Historical Background

This item is Recommendation 5.2 of the DL memorandum[752] and called for the NRC to reevaluate and consider reclassifying or redefining the design basis SGTR event. This issue is closely related to issues being addressed under Items 67.3.1 and 67.5.1.

A SGTR accident is one of the events for which the NRC requires a safety analysis to show that the reactor will respond in an acceptable manner and that the health and safety of the public are adequately protected. The SGTR accident is the loss of integrity (development of a leak) in a steam generator tube (or tubes) so that reactor coolant water from the primary system flows into the secondary water in the steam generator. This provides a potential path for the release of radioactivity to the environment.

As analyzed in SARs, the event is a break of a single steam generator tube with flow out of the full flow area of both ends of the steam generator tube at the break. The reactor is assumed to be at full power at the time of the accident.

The SGTR accident serves as the design basis for allowable reactor coolant activity since the amount of radioactivity released to the environment is directly proportional to the amount of activity in the coolant. The analysis of this event in SARs is intended to bound the potential release of radioactivity, should a SGTR occur. The behavior of reactor systems during this event has not traditionally received much emphasis, either in the analyses reported by the licensees or during review by the NRC.

Safety Significance

The safety significance of this recomme .ion is derived from the concern over the number of SGTR events and the potential for exceeding the bounds of the analyses that are currently required in SRP[11] Section 15.6.3 to demonstrate that doses from SGTR events will not exceed 10 CFR 100.

PRIORITY DETERMINATION

The analysis of an SGTR is performed to bound potential offsite doses using many conservative assumptions (i.e., accident terminated within 30 minutes) to maximize the predicted doses (SRP Section 15.6.3).[11]

The probability of the simultaneous occurrence of the SRP conditions is extremely low. SGTR events have occurred at a frequency of approximately 2×10^{-2}/RY. This event might therefore be classified as an incident which may occur during the lifetime of a particular plant.

SGTR events which have actually occurred were not as severe as the SRP design basis event. Had the frequencies of the conservative assumptions been included in a calculation of a design basis frequency, a much lower frequency would result. A change in classification would necessarily require changes to the conservative analysis assumptions (listed in the SRP). Changes to the design basis assumptions may include more conservative limits on the reactor coolant activity for those plants that do not have STS limits on coolant iodine concentrations, SGTR overfill conditions, multiple ruptures of the steam generator tubes, and other conditional failure scenarios.

CONCLUSION

The general basis for Item 67.5.2 is derived from the number of SGTR events that have occurred and the potential existing for SGTR doses exceeding 10 CFR 100 guidelines. However, these doses would occur only if there were an unlikely (but not impossible) set of circumstances as discussed in detail in Section 8.1 of NUREG-0916.[754]

For the 4 SGTRs that have occurred in domestic operating reactors, no significant consequences (doses) to the public have occurred and the existing design basis SGTR has proven to be adequate.

At the present time, and in regard to the safety significance of this issue, we believe it is premature to establish a priority for reclassification of the design basis SGTR event, prior to obtaining the results from other Staff Actions (See Item 67.5.1). Until the results from Item 67.5.1 are obtained, this issue should be considered a Licensing Issue.

ITEM 67.5.3: SECONDARY SYSTEM ISOLATION

DESCRIPTION

Historical Background

This item is Recommendation 5.3 of the DL memorandum[752] and called for the NRC to reevaluate the provisions for isolating the steam generators in conjunction with Items 67.3.1 and 67.5.1. The evaluation should consider whether the current provisions for isolating the main steam and feedwater lines are adequate with particular emphasis on isolation of the steam generator with RCS loop isolation valves, utilizing closed bonnet secondary safety valves or containing the discharge from the steam generator safety and relief (atmospheric dump) valves.

Safety Significance

The primary safety significance of SGTR events is the potential for a direct path for a loss of radioactive coolant from the RCS through the steam generator to outside the containment. This event could also increase the probability of a core-melt because the reactor coolant leaking from a steam generator tube cannot be recirculated. Other systems that penetrate the containment and interface either with the RCS or the containment have two containment isolation valves that close automatically or are locked closed. The steam generator safety and atmospheric valves open automatically and, as required by the ASME Code, cannot be isolated.

Possible Solution

Some of the older PWRs have block valves in the reactor coolant loops that could be used to isolate the steam generators and prevent the loss of coolant and radioactivity from the RCS. Alternatively, the discharge from the steam generator safety and relief valves could be routed to return to the containment or a quench tank. GDC 57 currently requires each line that penetrates containment (and is neither part of the RCS nor connected to the containment atmosphere) to have at least one isolation valve that is locked closed, automatic, or capable of remote operation. GDC 57 is not currently interpreted to apply to the valves on the steam generator. However, some improved means of isolating the steam generator, possibly either by requiring loop isolation valves in the RCS or containment of the safety valve discharge, could be considered.

PRIORITY DETERMINATION

Recommendation 8 of NUREG-0651[755] states: "For those plants provided with loop isolation valves, the use of these valves following an SGTR should be investigated. Isolating the affected loop would provide an almost immediate abatement of SG tube leakage, but would prohibit cooldown of the damaged SG. Licensees should, therefore, examine the advantages and disadvantages in their plant of loop isolation."

As pointed out in NUREG-0651,[755] the determination and isolation of the damaged SG appears to be taking longer than the assumed 30 minutes in the FSAR analysis. In this regard, Item 67.5.1 could address this aspect of SG isolation.

The EOPs involved with isolation of the secondary system following an SGTR have already been identified in Item 67.4.3 as selected events for staff review. In isolating the SG, the operator's worst error could be isolating the wrong steam generator. If this were to occur, overfill of the broken steam generator could still result. In addition, the intact steam generator which is isolated could boil dry. Saturated conditions in this hot leg could result. When the operator recognizes the error, isolates the faulted steam generator, and opens the intact steam generator, he might have no steam generator cooling since natural circulation might have become inhibited through the intact steam generator due to void formation. The faulted steam generator is now isolated, resulting in minimal transfer of heat. He could unisolate the faulted steam generator and steam either to the condenser (if available) or to the atmosphere, but this would result in increased offsite doses.

The W SGTR guidelines contain a note which advises the operator not to use these loop isolation valves in the event of an SGTR. It goes on to state that "any use of LSIVs (Loop Stop Isolation Valves) must be justified on a plant-specific basis." W reasons for not using these valves are: (1) their use has not been included in any accident analyses; (2) they are not meant to be safety components; (3) their use has not been recommended, since steam generator isolation has not been shown necessary to limit releases to an acceptable value; (4) the valves are very slow acting, taking on the order of minutes to close; and (5) their subsequent reopening required a rather careful procedure.

CONCLUSION

Many PWRs do not have these valves for use in an SGTR accident. For those plants that have LSIVs, modifications would likely be required.

However, based on the above discussions, the valves do not appear to be necessary. In each of the SGTR events that have occurred, the operator took correct action and in none of the events did incorrect action result in any significant adverse effect to the public. In each event, the SGTR was isolated to the faulted steam generator. Therefore, this issue was placed in the DROP category.

ITEM 67.6.0: ORGANIZATIONAL RESPONSES

DESCRIPTION

Historical Background

This item is Recommendation 6.0 of the DL memorandum[752] and called for the staff to establish, as soon as possible, improved NRC emergency preparedness to handle nuclear accidents at licensed reactor facilities.

Safety Significance

In the event of a nuclear accident, improved NRC emergency preparedness procedures will enable NRC to monitor and evaluate the situation and its potential hazards, advise the licensee's operating staff as needed, and, in an extreme case, issue orders governing such operations.

Possible Solution

Resolution of this item centers around implementation of TMI Action Plan Item III.A.3.

CONCLUSION

This item is part of the TMI Action Plan Item III.A.3.

ITEM 67.7.0: IMPROVED EDDY CURRENT TESTS

DESCRIPTION

Historical Background

Improved Eddy Current Tests (ECT) were originally proposed by the staff as requirements to be implemented by the licensees. Improved ECT could enhance earlier detection of degradations and thereby minimize, or mitigate, steam generator tube degradations and ruptures. The evaluation of improved ECT as a requirement (Item 66.3) showed that use of current state-of-the-art improvements provided only small reductions in public risk. Likewise, since ECT is an evolving technology, it was determined to be premature to impose a requirement at this time. However, it was also recognized that significant potential reductions in ORE could result from use of improved ECT. Therefore, this item was believed to warrant a medium priority ranking. The Item 66.3 conclusion is consistent with the position that improved ECT should be handled as a Staff Action item and developed in accordance with the possible solution described below.

Safety Significance

The SG tube that ruptured at Ginna exhibited no ECT indication during earlier testing. Improved ECT techniques would most likely have given ECT indications and avoided the SGTR event at Ginna.

Possible Solution

This effort, conducted in parallel with ongoing ASME Code Committee activities, would incorporate updated eddy current inspection procedures in the ASME Boiler and Pressure Vessel Code, Sections V and XI for NDE and ISI, respectively. The improved test procedures would be considered part of the in-service eddy current testing of PWR steam generator tubing.

CONCLUSION

In a previous evaluation[756] by the staff, it was determined that improved ECT techniques would provide small reductions in public risk and was therefore ranked as low priority relative to public risk reduction. It was also concluded that significant reductions in ORE could result from use of improved ECT techniques. The priority ranking based on the ORE reduction potential was medium. Improved ECT would also enhance the certainty that defective or degraded tubes would be identified and removed from service to assure meeting 10 CFR 100 release limits. The latter condition could be argued to classify improved ECT as a licensing improvement issue. In either classification, an economic incentive for use of improved ECT of up to $5M/plant, based on avoided cost of forced outages, could be obtainable. Based on a combination of the above potential benefits, development of improved ECT procedures was recommended as a medium priority principally because of the potential reductions in ORE. However, this item was later integrated into the resolution of Issue 135.[1075]

ITEM 67.8.0: DENTING

DESCRIPTION

Historical Background

This item concerns a staff recommendation to develop generic inspection criteria and methods to quantify steam generator tube denting. Operating experience has shown that surveillance of steam generator tubes is necessary to identify denting and to take corrective action to mitigate the stress corrosion cracking induced by denting.

Safety Significance

Denting can enhance stress corrosion cracking leading to through-wall cracks and leaks in steam generator tubes. Denting, combined with flow slot hourglassing, caused the U-bend stress corrosion cracking that led to the SGTR at Surry Unit 2 in September 1976.

Possible Solution

Development of a generic inspection requirement and criteria for steam generator tube denting will provide assurance that minimum standards for denting are applied uniformly.

PRIORITY DETERMINATION

Frequency Estimate

Only one SGTR event has been attributed to the denting phenomena in approximately 300 years of reactor operation. This corresponds to a SGTR frequency of 3×10^{-3}/RY. The SGTR contribution to a core-melt frequency of 4.7×10^{-6}/RY therefore contains a contribution of approximately 15% (7×10^{-7}/RY) due to denting.

Consequence Estimate

The PWR Category 4 release of 2.7×10^{6} man-rem is used to estimate the consequences of a core-melt associated with an SGTR. Using the above frequencies, the public risk, annualized over a remaining plant life of 24 years, yields a public risk of $[(7 \times 10^{-7})(2.7 \times 10^{6})(24)] = 45$ man-rem/plant. If we assume that approximately 40 of the operational and planned PWRs (~90 plants) have or will experience denting problems, the total public risk is approximately 1,800 man-rem. Assuming a 30% reduction due to improved denting surveillance criteria results in a total public risk reduction of 13.5 man-rem/plant and 540 man-rem for 40 plants.

Cost Estimate

Industry Cost: It is estimated that, as a minimum, with the use of generic denting criteria from the STS, the industry cost benefit will parallel the NRC cost benefit.

NRC Cost: The estimated NRC cost to develop the denting criteria is based on 3 man-months of effort. At $100,000/man-year, this cost is $25,000. The implementation mechanism is assumed to be a revision to the STS. It is assumed that the denting criteria in the STS will apply to NTOL and CP plants and those operating plants that experience denting problems. Using the same ratio (40/90) as used in the above risk determination, 40 of the total of 90 plants will require implementation of the STS denting criteria. It is also estimated that development of generic denting criteria will reduce NRC plant-specific review time by 2 man-weeks/plant. The result is a cost savings of (40)(2)($1,920) or $153,600. The net cost benefit to the NRC is therefore approximately $128,600.

Based on the above assumptions, the total cost derived from development of generic denting criteria is a total net cost benefit of approximately $250,000.

Value/Impact Assessment

The public risk reduction associated with implementation of generic denting criteria is not significant. The major value in development of the generic denting criteria is that it may provide a net cost benefit to the NRC and industry. No negative impacts (adverse changes to existing plant-specific criteria) are assumed in this evaluation.

CONCLUSION

In consideration of low potential public risk reduction, development of generic denting criteria is considered low priority. However, the generic denting criteria provide a small public risk reduction potential and should result in a net cost reduction for the NRC and industry. Therefore, subject to the above implementation assumptions, development of the generic denting criteria is a Regulatory Impact issue that will be addressed in the resolution of Issue 135.[1075]

ITEM 67.9.0: REACTOR COOLANT SYSTEM PRESSURE CONTROL

DESCRIPTION

Historical Background

This item addresses Recommendation 9 of the DL memorandum[752] and calls for a study to determine the need for controlling and reducing RCS pressure during and following an SGTR with emphasis on existing plant systems and equipment. The spectrum of possible initial conditions, RCS thermal-hydraulic conditions, and break sizes should be considered. The use of the pressurizer auxiliary system should be explicitly examined since its use may eliminate the necessity to use the pressurizer PORV in cases where forced RCS flow has been lost. The study should address the following objectives: (1) minimizing the primary to secondary leakage through the broken steam generator tube; (2) maximizing control over system pressure; and (3) minimizing the chances of producing voids in the RCS and other complicating effects.

Safety Significance

RCS depressurization following an SGTR is more difficult because of the loss of normal pressurizer spray. RCS fluid contraction, caused by the cooldown from the dumping of secondary-side steam to either the main condenser or to the atmosphere, will result in some reduction in RCS pressure, but other measures must be taken to expeditiously reduce the RCS pressure to the point where primary coolant flow into the damaged steam generator stops. The pressurizer PORV was used during the Ginna and Prairie Island SGTR events to reduce RCS pressure. However, control of RCS pressure is difficult with the PORV since its use creates an additional loss of coolant. The decrease in RCS pressure can be so rapid that steam voids may be formed in the reactor vessel upper head and at the top of the steam generator U-tubes and may further complicate the RCS depressurization. Void formation can lead to concerns regarding core cooling. The Ginna operators were sufficiently concerned that they left the safety injection pumps operating, thereby overfilling the steam generator via primary-to-secondary leakage through the ruptured tube. The resulting secondary-side pressure transient caused the main steam safety valves to lift, releasing radioactive material directly to the atmosphere. It is not apparent that the auxiliary spray from the charging system could have successfully lowered RCS pressure to the point where primary coolant flow into the steam generators could have been stopped. It may have been that, by spraying cold charging fluid into the pressurizer, the decrease in pressure would have resulted in void formation thus expanding the RCS fluid volume, filling the pressurizer, and rendering further spray flow ineffective. This phenomenon should be examined as well as the thermal stresses on the spray nozzle.

Possible Solution

With optimized RCS pressure control, risk associated with an SGTR may be reduced by reducing the potential radiological consequences.

PRIORITY DETERMINATION

Frequency Estimate

Independent analyses by the staff considered three categories of SGTR events: (1) SGTR and loss of DHR; (2) SGTR resulting from LOCA; and (3) SGTR with loss of secondary system integrity. For Categories 1 and 2 above, the core-melt probability was not dominated by SGTRs. The core-melt probabilities calculated for Categories (1) and (2) were 5.5×10^{-7}/RY and 3×10^{-8}/RY, respectively.

Category 3 included single and multiple tube ruptures followed by stuck open SG safety valves, MSLB, failure of the MSIVs, SG overfill, and failure to depressurize the RCS before the RWST was exhausted. The latter was considered since recirculation water from the sump might not be available following a SGTR event should a loss of secondary system integrity (e.g., stuck open safety valve, MSLB) occur outside containment.

We assumed that RCS pressure control would enhance depressurization of the RCS by a factor of 10 for the Category 3 sequences involving less than 10 SGTRs. For greater than 10 SGTRs, the depressurization is assumed to be too rapid for the RCS pressure control to be effective. The result is a reduction in core-melt frequency of 1.8×10^{-6}/RY for enhanced RCS pressure control.

Consequence Estimate

The consequences (doses) resulting from an SGTR would involve releases typical of a PWR Category 4 release as used in WASH-1400[16] and modified to a typical meteorology with a population density of 340 persons/square-mile within a 50-mile radius. The public risk reduction is $(1.8 \times 10^{-6})(2.7 \times 10^{6})$ man-rem/RY or 4.9 man rem/RY. Considering an average remaining plant life of 24 years, the annualized public risk reduction is 117 man-rem/reactor.

Cost Estimate

NRC Cost: The cost of the recommended separate staff study depends on the present capability for RCS pressure control following an SGTR and the incremental improvement required. As a minimum, the study may require a review and documentation of how existing systems and procedures already provide the requisite capability. In some plants, the study may require thermal-hydraulic modeling of the primary and secondary coolant systems as well as detailed stress analysis of selected components such as the pressurizer auxiliary spray nozzle.

A study of this depth and the development of an optimized approach for RCS pressure control could cost on the order of one man-year ($100,000) or more.

TMI Action Plan Item I.C.1, clarified in NUREG-0737,[98] has within its scope the development of EOPs for accidents and transients including multiple SGTRs. Likewise, the USI A-45 study is also developing the adequacy of current and alternate means of satisfying LWR shutdown decay heat removal requirements. The USI A-45 study will also be looking into shutdown requirements in effect during SGTRs in PWRs. Therefore, ongoing NRC studies, if properly coordinated, would negate the need for a separate study on RCS pressure control.

Industry Cost: The major cost of the study, as recommended, would be borne by the NRC and its contractors; however, input by and consultation with specific plants, plant types, or perhaps separate PWR owners' groups would be involved. In the latter case, NSSS owners' groups are currently evaluating means of controlling reactor coolant pressure during an SGTR. The depth and scope of the steam generator owners' group (SGOG) study can be expected to at least parallel the above NRC (study) cost.

The cost of implementing an optimized approach for RCS pressure control is likely to be highly variable, depending on the adequacy of the present RCS pressure control capability and the differences between the present and the optimized approach. The cost associated with implementing an optimized approach for RCS pressure control is not presently quantifiable, but may include some or all of the following items of cost: (1) developing, validating, and implementing new emergency procedures; (2) training plant operators; or (3) replacing equipment or upgrading equipment qualification if existing equipment must be operated outside of the conditions for which it was originally designed and qualified. In the present scope of the recommended study, the implementation cost is moot. However, in an overall value/impact, the implementation cost could be significant.

Value/Impact Assessment

The value of the recommended NRC staff study on Reactor Coolant System Pressure Control is that it may uncover, or result in development of, optimized means (procedures, equipment, instrumentation) to control reactor coolant pressure to minimize primary to secondary leakage following an SGTR. Thus, the potential for overfilling a steam generator and the quantity of radioactive material released directly to the atmosphere following an SGTR should be reduced.

Based on the above frequency and consequence estimates, the value is a potential public risk reduction of 117 man-rem/reactor over an average remaining plant life of 24 years. The major initial impact is the cost of performing the study. Subsequent impacts will depend on the results of the study and cannot be quantified at the present time.

CONCLUSION

Based on the above, the potential public risk reduction of 117 man-rem/reactor that may be derived by a separate (new) NRC study on RCS pressure control is not highly significant. The potential value which would result from such a study would most likely be improved RCS pressure control for both accidents and transients. In this regard, current staff actions being developed under TMI

Action Plan Items I.C.1(2,3) and USI A-45 would also resolve the objective of this issue. In addition, the ongoing work by the SGOG on RCS pressure control could be factored into the ongoing Items I.C.1(2,3) and USI A-45 reviews.

In summary, in view of the above findings, RCS pressure control is considered part of ongoing studies of Items I.C.1(2,3) of NUREG-0737[98] (being implemented under MPAs F-04 and F-05) and USI A-45.

ITEM 67.10.0: SUPPLEMENTAL TUBE INSPECTIONS

DESCRIPTION

Supplemental Tube Inspection (STI) was originally proposed by the staff as a recommended licensee action.[752] The value/impact analysis[756] ranked the proposed staff recommendation as a licensing issue. This ranking inferred that the staff-proposed STI would provide only small potential public risk reductions and a low value/impact ratio. However, as a minimum, the statistical sample size of the proposed STI would ensure that no more than the limiting number of defective tubes would go undetected. The limiting number of sample tubes to be inspected would be based on meeting 10 CFR 100 release limits from, and concurrent with, a MSLB. Thus, STI would provide additional assurance that existing regulatory requirements on radiological releases would be maintained and further reduce SGTRs. Subsequent information[753] from industry indicated that the staff-proposed STI would result in higher costs and greater ORE than that previously estimated by the staff. The staff reevaluated[753] their proposed STI and agreed in part with the industry assessment. However, it is also the current staff position that some form of STI can be formulated that would provide added assurance of tube integrity with less ORE and an improved value/impact relationship.

In view of the above, the STI was dropped as an issue for licensee implementation and categorized as a licensee issue for further staff action and reevaluation.

CONCLUSION

Based on the above discussion, the STI is recommended as a Licensing Issue staff action to investigate more practical alternatives for STI.

REFERENCES

11. NUREG-0800, "Standard Review Plan," U.S. Nuclear Regulatory Commission.

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

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

55. Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," U.S. Nuclear Regulatory Commission.

97. NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small-Break Loss-of-Coolant Accidents in Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, December 1979.

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

376. NRC Letter to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, "Supplement 1 to NUREG-0737 Requirements for Emergency Response Capability (Generic Letter No. 82-33)," December 17, 1982.

491. NRC letter to All Licensees of Operating Westinghouse and CE PWRs (Except Arkansas Nuclear One - Unit 2 and San Onofre Units 2 and 3), "Inadequate Core Cooling Instrumentation System (Generic Letter No. 82-28)," December 10, 1982.

752. Memorandum for S. Hanauer from D. Eisenhut, "Value/Impact Assessment of Proposed Steam Generator Generic Requirements," October 12, 1982.

753. SECY-84-13, "NRC Integrated Program for the Resolution of Steam Generator USI's," January 11, 1984.

754. NUREG-0916, "Safety Evaluation Report Related to Restart of R.E. Ginna Nuclear Power Plant," U.S. Nuclear Regulatory Commission, May 1982.

755. NUREG-0651, "Evaluation of Steam Generator Tube Rupture Events," U.S. Nuclear Regulatory Commission, March 1980.

756. Memorandum for D. Eisenhut from T. Speis, "Prioritization of Staff Actions Concerning S.G. Tube Degradation and Rupture Events," February 23, 1983.

757. SECY-84-13A, "NRC Integrated Program for the Resolution of Steam Generator USIs," September 7, 1984.

758. SECY-84-13B, "NRC Integrated Program for the Resolution of Steam Generator USI's - Response to Commission Comments (Memo from Chilk to Dircks dated September 13, 1984)," November 5, 1984.

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

ISSUE 77: FLOODING OF SAFETY EQUIPMENT COMPARTMENTS BY BACKFLOW THROUGH FLOOR DRAINS

DESCRIPTION

Historical Background

On November 11, 1981, the DAILY REPORT-REGION I carried a "prompt report" from Calvert Cliffs Units 1 and 2 indicating the licensee had been notified that the water tight integrity of the service water pump rooms in both units could be impaired because check valves had not been installed in the floor drain system which drains by gravity to the turbine condenser pit in the turbine building. Without these check valves, the operability of the service water pumps for both units could not be assured in the event of a circulating water conduit break in the turbine building of one unit. This event was subsequently reported as LERs 81-79 and 81-47 for Units 1 and 2, respectively.

This matter was presented in an AEOD report[525] in which an evaluation was performed on the generic implications of these events. It was noted that the Systematic Evaluation Program, begun in 1978, did not specifically review the matter of backflow flooding protection through drain lines in safety-related equipment compartments. In addition, AEOD reviewed other programs to establish whether this issue had been treated elsewhere. It was established that a generic review entitled, "Flood of Equipment Important to Safety," was tracked as Topic 3-18 in the Regulatory Licensing-Status Summary (NUREG-0328) and was applicable to all operating plants as of March 1974. Topic 3-18 was not concluded successfully, however, and the problem was assigned to NRR Generic Technical Issue B-11, "SubCompartment Standard Problems." A review by AEOD led to the conclusion that the drain line problems and the matter of backflow flooding protection had not been addressed adequately. The most relevant ongoing work tnat had been identified by AEOD was USI A-17, "Systems Interactions in Nuclear Power Plants," and an adjunct TMI Action Plan Item, II.C.3, "Systems Interaction." However, it was concluded that these activities did not explicitly address the issue of improperly-designed floor drains system and Issue 77 was prioritized separately. An IE Information Notice[527] concerning the potential generic implications of this issue was published on July 1, 1983.

Safety Significance

The service water systems at Calvert Cliffs Units 1 and 2 each have three pumps and serve both safety and nonsafety equipment. The three service water pumps for each unit are located in a single room and Units 1 and 2 service water systems can be cross-connected by spool pieces to allow the Unit 1 system to backup Unit 2 and vice-versa. However, Units 1 and 2 share a common turbine building, so both of the service water pump rooms would be simultaneously affected by a circulating water conduit break in the turbine building if backflow flooding protection was not provided. Additional specific details concerning the Calvert Cliffs plants are presented in AEOD/E304.[525]

The safety significance of the loss of the service water pumps lies in the fact that the service water system serves as the ultimate heat sink in nuclear plants. In addition to being the AFW pump emergency suction supply, the service water provides cooling, either directly or indirectly, for the following plant components: component cooling water heat exchangers, containment fan coolers, diesel-generator coolers, control-room air-conditioning system condensers, computer room air-conditioning system condensers, auxiliary building ventilation system cooling coils, containment spray pump diesel engine coolers, and auxiliary building room coolers. The component cooling water, in turn, is required for the proper operation of essential pumps and heat exchangers required for the safe shutdown of a nuclear plant. Without these essential systems, the probability of core-melt becomes unacceptable.

This issue does not apply to plants reviewed and licensed in accordance with the SRP because SRP[11] Sections 9.3.3, "Equipment and Floor Drainage Systems," and 10.4.5, "Circulating Water System," adequately deal with the concern. The safety significance is limited to older plants that were licensed some time prior to the formalization of the SRP, but the extent of possible design deficiencies in these older plants is unknown at present.

In addition, it is noted that the fundamental problem of backflow flooding of safety systems through drains is a potential problem with implications that are much broader than those related to the specific situation at Calvert Cliffs, used for the purposes of analysis herein. Safety components other than service water pumps may be affected in either BWR or PWR systems and the flooding may be from sources other than circulating water conduits and the turbine condenser pit. An example illustrating this point is the flooding incident which occurred at the Oconee Nuclear Station resulting from the inadvertent opening of a main condenser isolation valve.

Possible Solution

A temporary preventive measure is the installation of inflatable drain plugs, but this is of limited value as the drains are prevented from functioning by these plugs. A permanent solution is the installation of check valves in the drain lines to prevent backflow flooding and permit proper drain operation. Both of these solutions have been employed at the Calvert Cliffs Nuclear Plant, Units 1 and 2.

PRIORITY DETERMINATION

Assumptions

Inasmuch as the possible design defects which could lead to backflow flooding through floor drains are plant-specific and the details are not known at this time, the prioritization will be based on the circumstances and events as noted for the Calvert Cliffs plants and generalized as needed.

Frequency Estimate

Based on a review of the LERs performed by AEOD,[525] it was noted that Quad Cities Unit 1 had experienced a rupture of an expansion bellows in the circulating water system in 1972. The resultant flooding caused some degradation of engineered safety feature equipment. No other similar event has been noted in the operating experience of nuclear plants. Therefore, based on this one event in (72 plants)(12 years) = 864 plant-years, the internal flooding frequency is estimated to be approximately 10^{-3} event/plant-year. This is an overestimate because plants have been previously reviewed to assess the potential for internal flooding and corrective actions have been taken as a result of this incident.

Consequence Estimate

The consequences for this event are assumed to result from the following scenario. As a result of the flooding of the turbine condenser pit and the service water compartment, it is assumed that the reactor would be tripped.

Inasmuch as the component cooling water (CCW) system would fail following the failure of the service water pumps, essentially all of the ESFs would be unavailable because of their dependency on CCW for cooling. In addition, primary pump seal failure would follow within a short time after loss of CCW[526] causing a small break in the primary system. Moreover, the containment spray and containment fan coolers would be inoperative following the loss of service water. The primary system would be depressurizing through the small break associated with the pump seal failure without the capability of make-up available because of the failure of ECCS. Natural convection cooling would be available for a short period of time inasmuch as the auxiliary feedwater pumps would still be operative. However, as the primary system depressurizes without the availability of the charging pumps, a void will form in the vessel head which will eventually interface with continued natural convection flow. Simultaneously, the containment continues to be pressurized because of the unavailability of containment sprays and heat removal capacity. Eventually the core will uncover and melt. The molten core will slump into the lower vessel head presenting a distinct possibility of a steam explosion on contact of the molten core with coolant that may still be contained in the lower vessel head. Containment failure will occur as a result of overpressurization and/or the steam explosion. This sequence of events is closely approximated by PWR Release Category 3.[16] For this category, the release is estimated to result in an exposure of 5.4×10^6 man-rem.

This estimate is also an overestimation of the conditional probability and consequences of a core-melt resulting from an internal flooding incident. The location of ESFs relative to the location of the flooding can greatly reduce or eliminate the probability of core-melt. For example, most plants have the service water pumps outside the plant at a crib house. The interaction of systems can also change the probability.

Based on the frequency of flooding resulting from a rupture of the circulating water pipe bellows of 10^{-3}, the probability of failure of the containment due to overpressurization of 0.6, the PWR release of Category 3 estimated to be 5.4×10^6 man-rem/event,[16] and 25 years of remaining average reactor life, the risk reduction is estimated to be $(10^{-3})(5.4 \times 10^6$ man-rem/event)(25 yr) or 81,000 man-rem/reactor.

Cost Estimate

The costs associated with the resolution of this issue are difficult to assess in general because the deficiencies that may exist will be plant-specific. However, on the basis of informal contact with representatives of the Calvert Cliffs nuclear plant, it was established that the purchase and installation of ball-type check valves (13 in all) as well as expandable plugs in some of the additional drain lines and maintenance of these valves will not exceed a total cost of $10,000. This cost reflects easy access to the drain lines for the installation of the valves in the case of the Calvert Cliffs plant. Assuming that a typical plant may have greater difficulties installing similar valves, the cost for a typical plant is estimated to be approximately $100,000.

Value/Impact Assessment

Based on an estimated risk reduction of 81,000 man-rem/reactor, the value/impact score is given by:

$$S = \frac{81,000 \text{ man-rem/reactor}}{\$0.1M/\text{reactor}}$$

$$= 8.1 \times 10^5 \text{ man-rem/\$M.}$$

CONCLUSION

Based on the estimated core-melt frequency of 10^{-3} as well as the calculated risk reduction of 81,000 man-rem/reactor, this issue would have a high priority ranking. Even if the cost of the resolution of the issue is substantially greater, the risk alone justifies a high priority ranking. In addition, it is concluded that this issue has broader potential safety implications than the Calvert Cliffs situation and flooding can affect many safety systems in BWRs or PWRs and may occur from many sources. These risks estimates are conservative and, as noted, specifics of each plant design can affect the risk greatly. Without further detailed information, the degree of conservatism in these estimates cannot be known. Thus, a high priority was assigned to this issue to more accurately determine the risks involved and to develop a solution. However, in May 1986, this issue was integrated[1075] into the resolution of USI A-17.

REFERENCES

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

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

525. Memorandum for H. Denton from C. Heltemes, "Engineering Evaluation Report, Investigation of Backflow Protection in Common Equipment and Floor Drain Systems to Prevent Flooding of Vital Equipment in Safety-Related Compartments," March 11, 1983.

526. NUREG/CR-3226, "Station Blackout Accident Analyses (Part of NRC Task Action Plan A-44)," U.S.Nuclear Regulatory Commission, May 1983.

527. IE Information Notice No. 83-44, "Potential Damage to Redundant Safety Equipment as a Result of Backflow Through the Equipment and Floor Drain System," U.S Nuclear Regulatory Commission, July 1, 1983.

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

ISSUE 88: EARTHQUAKES AND EMERGENCY PLANNING

DESCRIPTION

Historical Background

This issue was initiated to address concerns raised by the Union of Concerned Scientists. (References 1032, 1033, and 1034.) The purposes for including this issue as a generic issue are to: (1) provide brief background information that summarizes the history of the issue; (2) reduce the probability of resurrecting the issue and duplicating effort; and (3) identify the final disposition of the issue.

Safety Significance

Recent PRAs have indicated that earthquakes (and other external events) can cause severe reactor accidents which are comparable with internally initiated accident event sequences. The results argued for a reexamination of the emergency response measures to ascertain whether they are adequate to protect the health and safety of the public.[1034] This issue effects all operating and planned nuclear power plants.

Proposed Solution

In June 1979, the Commission began a formal reconsideration of the role of emergency planning in ensuring the continued protection of the public health and safety in areas around nuclear power plants. On August 19, 1980, the Commission published its rule on emergency planning establishing 16 planning standards [see 10 CFR 50.47(b)][197] which must be generally met by both onsite and offsite emergency response plans for nuclear power plants. The planning standards are addressed by specific evaluation criteria in NUREG-0654,[224] Revision 1. Thus, the NRC emergency planning requirements and guidance reflect coordinated efforts with the Federal Emergency Management Agency (FEMA). Both the NRC and FEMA shared the view that the required emergency response plans have considerable flexibility to respond to a wide variety of adverse conditions, including those resulting from an earthquake.

However, on December 21, 1984, the Commission published proposed amendments to its emergency planning requirements [10 CFR 50.47(b) and 10 CFR 50, Appendix E]. These amendments proposed to explicitly adopt by rule the Commission's interpretations of its existing rules. Final amendments to 10 CFR 50, Appendix E, were prepared by the staff and forwarded to the Commission in SECY-85-283.[1035]

In October 1986, the Commission determined that the proposed amendments were not necessary. Based on examinations and reviews of public comments concerning the proposed amendment, it was stated:[1062] "The Commission is satisfied that none of the information submitted by commenters indicates that its interpretations of emergency planning rules in the San Onofre and Diablo Canyon proceeding was mistaken or that the potential for seismic impacts on emergency planning is a significant enough concern for large portions of the nation to warrant amendment of the regulations. Nor did the comments suggest any additional cost-effective measures which might be taken to provide further assurance of protection in the event of an earthquake occurring simultaneously with a radiological release. Moreover, the en banc decision of the United States Court Of Appeals for the District Of Columbia Circuit, affirming the Commission's interpretation of its emergency planning rules, has removed regulatory uncertainty in this area. ... If the need to consider earthquakes in emergency planning is raised in an adjudication, the Commission expects to adhere to the Diablo Canyon and San Onofre precedents unless a convincing case is made that application of these precedents to the facts of the case would cause a significant safety problem."

In view of the above, the Commission decided that a rulemaking which would simply make explicit the Commission's interpretation of its rules is unnecessary and the proposed amendment was withdrawn.[1062] The withdrawal of the proposed amendment will therefore not have a significant effect on the emergency preparedness requirements established in August 1980.

CONCLUSION

This issue was RESOLVED and no new requirements were established.

REFERENCES

197. Code of Federal Regulations, Title 10, Energy.

224. NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1980, (Revision 1) November 1980.

1032. Memorandum for H. Denton from T. Speis, "Earthquakes And Emergency Planning," January 18, 1984.

1033. Letter to W. Dircks (NRC) from S. Sholly (Union of Concerned Scientists), December 22, 1983.

1034. Letter to J. Asselstine (NRC) from S. Sholly (Union of Concerned Scientists), December 22, 1983.

1035. SECY-85-283, "Final Amendments to 10 CFR Part 50, Appendix E; Consideration of Earthquakes in Emergency Planning," August 21, 1985.

1062. Federal Register Notice 51 FR 39390, "10 CFR Part 50, Emergency Planning and Preparedness; Withdrawal," October 28, 1986.

ISSUE 91: MAIN CRANKSHAFT FAILURES IN TRANSAMERICA DELAVAL EMERGENCY DIESEL GENERATORS

DESCRIPTION

Historical Background

On August 12, 1983, one of the three emergency diesel generators (EDG) at the Shoreham Plant failed during overload testing as a result of a fractured crankshaft. The failure occurred in EDG-102 and similar crankshaft cracks were discovered in EDG-103 and EDG-101 on August 22 and 23, 1983, respectively. In addition to the crankshaft cracks, 4 of 24 connecting rod bearings were found to contain cracks in the bearing shells. All 3 EDGs were supplied by Transamerica Delaval, Inc. (TDI) and were Model DSR-48 diesels.

On August 30, 1983, IE Information Notice No. 83-58[780] was issued to inform licensees of the Shoreham event. Prior to this, IE Information Notice No. 83-51[781] had been issued to inform licensees of various diesel-generator problems. The staff reviewed the operating status of the 3 plants with TDI engines and sent letters to all TDI diesel owners requesting specific information about their respective engines. A letter was also sent to TDI on December 1, 1983 requesting information on the design development history of various parts of TDI machines. A response from TDI was sent on December 16, 1983 and, on December 23, 1983, the staff was informed that a TDI Diesel Generator Owners' Group had been formed to address the problem.

As a result of the EDG failure at Shoreham, a TDI Project Group was established by NRR on January 16, 1984.[782] On January 25, 1984, the staff provided the Commission with a status report in SECY-84-34.[783] In order to more clearly define the issue and to determine remedial action, the staff issued a letter to TDI on February 14, 1984 requesting more information.[784] In March 1984, the TDI Diesel Generators Owners' Group submitted to the NRC its program for addressing the issue.[785] In April 1984, the staff recommended to the Commission in SECY-84-155[786] that the question of reliability of TDI diesels had generic implications and should be reported to Congress as an abnormal occurrence. An SER on the TDI Diesel Generator Owners Group Program Plan (OGPP) was issued by the staff on August 13, 1984.[787]

In its SER, the staff's overall finding was that the OGPP incorporates the essential elements needed to resolve the outstanding concerns relating to the reliability of the TDI diesel generators for nuclear service, and to ensure that the TDI diesel engines comply with GDC 1 and GDC 17. These corrective actions include: (1) resolution of known generic problems (Phase I), (2) systematic DR/QR of all components important to reliability and operability of the engines (Phase II), (3) appropriate engine inspections and testing as identified by the results of Phases I and II, and (4) appropriate maintenance and surveillance programs as indicated by the results of Phases I and II.

After licensees complete Phases I and II of the OGPP, the licensing basis will be reviewed by the staff to determine what modifications to the license conditions will be required. A final SER will be issued for each of the plants that are being licensed or restarted on an interim basis. These are expected to include: Shoreham, Grand Gulf, San Onofre, Catawba, and Comanche Peak. For plants where Phases I and II are scheduled to be completed sufficiently ahead of licensing or restart, a final TDI Diesel SER will be developed that encompasses the results of Phases I and II and the operational history of an engine.

Safety Significance

In the event of loss of offsite power, the power to operate the equipment necessary to maintain core cooling is provided in most plants by EDGs. Although to varying degrees, plants can withstand the loss of both offsite and onsite AC power (and further requirements are being proposed in USI A-44), EDG unreliability is a significant contributor to the estimated frequency of core damage events. The question of diesel-generator reliability in general is addressed in Item B-56, "Diesel Reliability." Issue 91 applies to the design and operation of the 16 plants which have or have not ordered TDI diesel-generators.

Possible Solutions

The possible solutions to this issue are considered to be the three elements of the TDI OGPP:

(a) Phase I: Resolution of 16 identified generic problem areas intended (by the Owners' Group) to serve as a basis for the licensing of plants during the period prior to completion and implementation of the OGPP.

(b) Phase II: A design review/quality revalidation of a larger set of important engine components to assure that their design and manufacture (including specifications, quality control, quality assurance, operational surveillance, and maintenance) are adequate.

(c) Identification of any needed additional engine testing or inspections based on findings from Phases I and II.

CONCLUSION

In response to the problems raised in this issue, the Owners' Group performed extensive design reviews of all key engine components and developed recommendations to be implemented by the individual owners concerning needed component replacements and modifications, component inspections to validate the "as-manufactured" and "as-assembled" quality of key engine components, engine testing, and an enhanced engine maintenance and surveillance program.

The staff's evaluation of the Owners Group program, documented in NUREG-1216,[1070] concluded that implementation of the Owners Group recommendations, plus additional actions identified, will establish the adequacy of the TDI diesel generators for nuclear standby service as required by GDC 17 of 10 CFR 50, Appendix A. The staff further concluded that these actions will ensure that the design and manufacturing quality of the TDI engines is within the range normally assumed for diesel engines designed and manufactured in accordance with 10 CFR 50, Appendix B. Continued reliability and operability of the TDI engines for the life of the facilities will be ensured by implementation of the maintenance/surveillance program described in NUREG-1216.[1070] Thus, this issue was RESOLVED and no new requirements were established.

REFERENCES

780. IE Information Notice No. 83-58, "Transamerica Delaval Diesel Generator Crankshaft Failure," U. S. Nuclear Regulatory Commission, August 30, 1983.

781. IE Information Notice No. 83-51, "Diesel Generator Events", U.S. Nuclear Regulatory Commission, August 5, 1983.

782. Memorandum for C. Berlinger from H. Denton, "Detail Assignment to DOL, Transamerica Delaval Emergency Diesel Generator Project Group (TDI Project Group)," January 25, 1984.

783. SECY-84-34, "Emergency Diesel Generators Manufactured by Transamerica Delaval, Inc.," January 25, 1984.

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ISSUE 106: PIPING AND THE USE OF HIGHLY COMBUSTIBLE GASES IN VITAL AREAS

DESCRIPTION

Historical Background

Combustible gases such as H_2, propane, acetylene, and other fuel gases are used during normal operation of nuclear power plants, as well as in plant laboratories. Most combustible gases are used in limited quantities and for relatively short periods of time at a nuclear plant. H_2, the most prevalent combustible gas used in nuclear power plants, is used as a coolant for electric generators in both BWRs and PWRs and is also used in PWRs in association with the reactor water chemistry as well as in the waste gas disposal functions. H_2 is used in the volume control tank (VCT) which is usually located in the auxiliary systems building of PWRs. It is stored as high pressure gas in storage vessels and is supplied as process to the various systems in the auxiliary systems building through standard piping, usually 3/4-inch in diameter. As such, the piping is field-run and its location is plant-specific. Leaks or breaks in the H_2 piping and supply system could result in the accumulation of a combustible or explosive mixture of air and H_2 within the auxiliary systems building. Inasmuch as the auxiliary systems building is a safety-related structure which houses most of the components of the safety-related systems of the plant, the accumulation of combustible or explosive mixtures of gas represents a threat to the safety of the plant by virtue of the potential disablement of safety-related equipment in the event that the combustible gases are inadvertently ignited. H_2 detectors can signal the presence and accumulation of gas, but these are not qualified as safety-grade equipment and do not have an emergency power source. Thus, they are not regarded as sufficient protection against the development of H_2 leakage and subsequent uncontrolled combustion or explosion.

SRP[11] Section 9.5-1, "Fire Protection," currently addresses the safe use of combustible gases on site so that this matter is a concern primarily for operating reactors licensed prior to the issuance of SRP[11] Section 9.5-1.

This issue was identified in NUREG-0705[44] and is related to Issue 136, "Storage and Use of Large Quantities of Cryogenic Combustibles on Site." Whereas Issue 106 is concerned with the normal process system use of relatively small amounts of combustible gases on site, Issue 136 deals with the considerably greater hazards of much greater amounts of combustible materials introduced by new needs at the site (i.e., solid waste processing and BWR hydrogen water chemistry control and the unique hazards associated with the transport and storage of large quantities of combustibles on site in a cryogenic liquid state).

Safety Significance

The auxiliary systems building is a safety-related structure housing safety-related system components. Inasmuch as the most frequently used combustible gas, H_2 is piped into this building for use in the VCT, there is a potential for leakage and the inadvertent ignition of the gas. The ensuing combustion or explosion can cause damage or failure of safety-related equipment, thereby contributing to a possibly significant increase in the core-melt probability of the plant.

Possible Solutions

Large releases of combustible gas and the accumulation of combustible or explosive mixtures in air, in the event of a piping system break or large leak, can be prevented by the installation of excess flow check valves located close to the source of the combustible gas. SRP[11] Section 9.5-1, "Fire Protection," recommends the use of excess flow check valves. Other measures are needed to reduce the frequency of, or cause of, combustible gas accumulation accidents from such events as valve malfunctions or leaks, connection or fitting leaks, operations errors, material failures, etc. Plants licensed in accordance with the guidelines of SRP Section 9.5-1 are assumed to be not affected by this issue. For the purpose of this analysis, the backfitting of excess flow check valves at all plants not licensed in accordance with SRP 9.5-1 plants is assumed. Excess flow check valves are an effective "fix" for piping system breaks, but other fixes, such as installation or upgrading of H_2 detection systems, design changes, procedural changes, etc., will be required for other types of accidental releases. The risk and cost analyses performed for the installation of excess flow check valves as a fix are extrapolated to develop a proper perspective for the prioritization of this issue.

PRIORITY DETERMINATION

Assumptions

It is assumed that, of all the combustible gases routinely used in a nuclear power plant, the most significant safety concern is associated with the use of H_2 because, unlike most other gases used in small quantities at nuclear power plants, H_2 is used almost continuously while most other gases are used intermittently and most likely in the presence of trained personnel, such as during welding operations. Hydrogen leaks could continue unnoticed as a result of leaks or pipe breaks that go undetected for a sufficient time to accumulate a combustible mixture; it is assumed that H_2 detectors are either not provided (as was the case in a recent event at the Vogtle Plant)[1031] or are inoperative. In addition, it is assumed that the operating plants licensed prior to the SRP Section 9.5-1 do not have excess flow check valves in place. This latter assumption is a conservative element in this analysis because it is likely that some of the plants licensed prior to SRP Section 9.5-1 may already have excess flow check valves in place.

It is also noted that the auxiliary systems building is a safety-related structure that contains most of the components of the safety systems of the plant. However, the design of this structure and the location of safety-related components within the structure are plant-specific. In addition, location of the hydrogen source and, in particular, the hydrogen field-run piping layout are also plant-specific. In view of this, it is not possible to identify a particular damage scenario that represents a bounding sequence for the purposes of a generic analysis. Therefore, a reasonable but not necessarily bounding damage scenario will be assumed in order to formulate a prioritization of this issue. This scenario entails the assumption of a H_2 piping system leak or break, the accumulation of a combustible mixture within a room or space containing safety-related equipment, an ignition source, and damage contained within that room or space. The PNL analysis[64] based on pipe break is extrapolated to estimate the frequency of all events which might result in the release and accumulation of combustible gases in the Auxiliary Building. It is assumed that the pipe break frequency (for 3/4" pipe) may be obtained from WASH-1400,[16] but that the probability of the accumulation of a combustible mixture, the probability of the availability of an ignition source, and the probability of total demolition of the safety-related redundant equipment are 1 in each instance. This latter assumption is conservative.

The scenario that is selected as a reasonable one for this prioritization analysis is the loss of both RHR heat exchangers (complete loss of heat sink). Resolution of this issue would affect operating plants using H_2 and not already in compliance with SRP Section 9.5-1 with respect to H_2 gas piping. Specifically, resolution is anticipated to include all operating PWRs. Therefore, the number of affected plants is 47 PWRs with an average remaining lifetime of about 28 years.

Frequency Estimate

The H_2 piping is standard piping, generally thought to be 3/4 inch in diameter. Based on the results of WASH-1400[16] (Tables III 2-1, 2-2), the pipe break frequency for piping less than 3 inches in diameter is 10^{-9}/hr per section. In general, it is assumed that the H_2 piping in nuclear plants is comprised of about 25 sections. With 8760 hrs/yr and an assumed plant utilization factor of 70%, the frequency of H_2 release due to pipe break (f_p) is estimated to be:

$$(10^{-9})(25 \text{ sections})(8760 \text{ hr/yr})(0.7) = (1.5)(10^{-4}) \text{ pipe breaks/RY}$$

A review of 96 hydrogen accidents by the National Aeronautics and Space Administration[1030] indicates that about 52% of the accidents could be attributed to causes which relate to use of H_2 in a gaseous state and about 48% could be attributed to causes which relate to the use of H_2 in its liquid (cryogenic) state. Only about 2% of the accidents were attributed to piping breaks. We therefore assumed that H_2 accidents from all gaseous state causes are 26 times as likely to occur (52%/2%) as a H_2 accident due to a pipe break. The probability (P) of a H_2 release (leak) is given by:

$$\begin{aligned} P &= (26)(\text{Probability of Pipe Break}) \\ &= (26)(1.5 \times 10^{-4}) \\ &= 3.9 \times 10^{-3} \end{aligned}$$

The probability of failing both RHR heat exchangers, f(RHR), is the product of the probability of H_2 leak, the probability of obtaining a combustible mixture, the probability of ignition, and the probability of being in the blast zone. Thus, $f(RHR) = (3.9 \times 10^{-3})(1)(1)(1) = 3.9 \times 10^{-3}$

Inasmuch as both trains of the RHR system are inoperable, the plant technical specifications require the plants to proceeed to the hot shutdown condition within 12 hours. This requirement to achieve hot shutdown within 12 hours is modeled in this analysis as a T3 (PWR) transient. Therefore, an H_2 explosion is modeled as an additional initiating transient (with a frequency of 3.9×10^{-3}/yr as calculated above). All other initiating transients and LOCA parameters are scaled by (12/8760 hr/RY) in order to model the occurrence of other random initiators during the 12 hours that the reactor is proceeding to hot shutdown. Finally, using the Oconee 3 PRA as representative of all PWRs, the RHR heat exchangers were modeled as inoperative by setting their representative system unavailabilities to 1. The Oconee PRA was then altered to incorporate the modified initiating event frequencies and the RHR systems unavailability in the affected minimal cut sets for each affected accident sequence. All affected Boolean equations were solved to calculate new core-melt frequencies for all containment failure modes and the affected core-melt frequencies were summed for each of the 7 distinct PWR core-melt categories. Public risk was then determined by summing the products of core-melt frequency and their respective release category dose factor for each release category.

The analysis was repeated for a time window of 96 hours (4 days) as an approximation of the time period necessary to achieve cold shutdown by alternate means such as feed-and-bleed. Based on these details, the following core-melt frequency results were calculated:

PWR Base Case: 5.46×10^{-6}/RY
Adjusted Case: 2.26×10^{-7}/RY
Reduction in core-melt frequency: 5.2×10^{-6}/RY

Consequence Estimate

Consequences estimated below were based on an analysis[64] performed by PNL. For the time required to come to hot shutdown (12 hours), the results of the PNL analysis indicate that resolution of this issue would result in a risk reduction of 8.8 man-rem/RY for PWRs. The total public risk reduction is estimated to be approximately 11,500 man-rem. The estimated occupational risk reduction due to accident avoidance is approximately 135 man-rem.

Cost Estimate

Industry Cost: PNL calculated[64] the costs to install excess-flow check valves in the H_2 line(s) outside of the safety-related area(s). It is assumed that these valves will be installed during scheduled reactor shutdown periods so that there is no additional power replacement cost incurred. Based on two vendor quotations, the average cost of one excess flow valve is approximately \$870. The costs for the implementation, maintenance, and operation of the excess flow check valve "fix" is detailed as follows:

(a) Implementation: Labor for design, procurement, and installation per plant is estimated as follows:

Hardware design and review:	2 days
Procurement:	1 day
Pre-installation check:	0.5 hr/valve
Installation:	2 day/valve [1 man-day/valve (welder), 1 man-day/valve (fitter)]

Post-Installation check: 1.5 hr/valve
Documentation: 0.5 day

Total labor time = 3.5 days + 2.25 days/valve
= 8 days (PWR)

Labor Cost = (8 days)($2270/man-wk)/(5 days/man-wk)
= $3632/plant (PWR)

(b) Equipment:

Valve Cost = 2($870) = $1740/plant (PWR)

The total implementation cost per plant is $(3,632 + 1,740) = $5,372 and the total implementation cost for all affected plants is approximately $255,000.

Operation and maintenance includes a semi-annual check of the installation to insure that the valve shaft is not "frozen" and replacement of the valve diaphragm as needed. The frequency of this replacement depends upon the valve environment. For the purposes of this analysis, the diaphragm is assumed to require replacement every 7 years with an associated labor requirement of 0.5 man-day.

(a) Labor for maintenance and operation:

Semi-annual check requires 2hrs/valves

Diaphragm replacement requires (average plant life/7-year replacements over the past lifetime at 4 hours per replacement.

Labor (PWR) = (2hr/valve)(. valves)(2 checks/yr)
+ (27.7yr/7yr)(4hr/valve)(2 valves)/(27.7yr)
= 8 hr/yr(checks)+1.14 hr/yr (avg. diaphragm repl.)
= 9.14 hr/RY

Labor cost is (9.14 hr/RY)($2270/man-wk)/(40hr/man-wk) = $519/RY.

The total industry labor cost for maintenance and operation is ($519/RY)(47 plants)(27.7 yr) = $675,686.

Therefore, the total industry cost for the resolution of this issue is approximately $930,000.

NRC Cost: NRC cost for the development of the safety issue implementation, including the formulation of guidelines and documentation requirements, review and inspection of final installation is given by (4 man-wks) ($2270/man-wk) = $9,080. NRC costs for implementation are (0.6 man-wk/plant)($2270/man-wk) = $1,362/plant. NRC costs for the review and inspection of plant operation and maintenance activities are (0.5 day/plant-test)(2 tests/yr) = 1 day/RY.

Thus, total NRC Costs = ($9,080) + ($1362/plant)(47 plants)
+ (1 day/plant-yr)(47 plants x 27.7 yr)
x ($2270/man-wk)/(5 days/man-wk)
= $656,000 approximately.

Installation of excess flow check valves is a satisfactory "fix" for the possibility of sudden accumulation of combustible or explosive mixtures of H_2 resulting from a piping system break, but it is not a solution for H_2 accidents arising from slow leaks in valves or fittings, purging errors, material degradation problems, contamination, etc. Other "fixes" are required to reduce or preclude H_2 accidents from scenarios other than pipe break. These other "fixes" would include the installation or upgrading of existing H_2 detection and alarm systems, complete combustible gas system design reviews and modifications to plant design and hardware, operating procedure reviews and modifications, improved preventive maintenance programs, and major modifications to the auxiliary building ventilation system. For an assumed population of 47 plants, it is apparent that the total industry cost of these other "fixes" would be very much more than the costs estimated for the very restrictive "fix" (excess flow check valves) estimated by PNL. We will therefore assume the costs of the other "fixes" to be an order of magnitude greater than those calculated for the installation of excess flow check valves alone. This leads us to a total estimate for complete resolution of the issue of at least \$15M. As will be shown in the conclusion, a rough estimate of cost is all that is required to arrive at an appropriate priority recommendation for this issue.

Value/Impact Assessment

Based on the estimated public risk reduction of 11,500 man-rem for the safety issue resolution and the estimated total cost of \$15M for this resolution, the priority score is given by:

$$S = \frac{11{,}500 \text{ man-rem}}{\$15M}$$

$$= 767 \text{ man-rem/\$M}$$

Other Considerations

Based on the reduction in core-melt frequency calculated by PNL for this issue of 5.2×10^{-6}/RY for PWRs, the cost savings resulting from accident avoidance is (\$1.65 billion)(5.2×10^{-6})(47)(27.7) = \$11.4M approximately.

CONCLUSION

The value/impact score arrived at above, the calculated potential public risk reduction, and potential reduction in core-melt frequency indicate that this issue should be assigned a medium priority for resolution. As acknowledged above, the cost estimate used in the value/impact score is very rough. However, examination of the matrix chart for priority assignment (Figure 1) indicates that, for the potential reductions in both public risk and core-melt frequency calculated, a medium priority is appropriate regardless of potential cost, unless the value/impact score is greater than 3000 man-rem/\$M, in which case, high priority would be appropriate. For the value/impact score to be \$3000 man-rem/\$M, the total industry cost for resolution of the issue could not exceed about \$3.8M. Based on the relatively detailed estimate of the cost of excess-flow check valve installation (\$1.6M), which is only a small portion of the total "fix", and extrapolation, it appears very unlikely that this issue could be completely resolved for less than \$4M total cost to the NRC and the 47 affected plants. We therefore conclude that this issue should be assigned a MEDIUM priority.

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

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ISSUE 113: DYNAMIC QUALIFICATION TESTING OF LARGE BORE HYDRAULIC SNUBBERS

DESCRIPTION

Historical Background

This issue was raised[1014] in March 1985 to address the staff's concern that there are no NRC requirements for dynamic qualification testing or dynamic surveillance testing of large bore hydraulic snubbers (> 50 kips load rating). The resolution of Issue A-13, "Snubber Operability Assurance," is the development of a Regulatory Guide (SC-708-4) pertaining to "Qualification and Acceptance Test for Snubbers Used in Systems Important to Safety". However, the Regulatory Guide may only be applied on a forward-fit basis and the need for dynamic testing requirements for large bore hydraulic snubbers (LBHS) in operating plants would remain unresolved.

The issue was raised because of the concern for the integrity of the steam generator lower support structures when subject to a seismic event. However, the issue is applicable to all LWRs with components, structures, and supports that rely on LBHS for seismic restraint and other dynamic loads such as high energy line breaks and water hammers.

Safety Significance

The safety concern identified[1014,1015] involves the integrity of the steam generator lower support structures (SGLS) when subject to a seismic event. In the absence of the restraint to the steam generators provided by the LBHS, the steam generator support structures (SGS) might fail. Failure of the SGS might subsequently result in rupture of the primary system piping (large break LOCA), the main steam lines (MSLB), and the feedwater (normal and auxiliary) piping lines. Such failures could result in a core-melt from the loss of all means of core cooling and could pose a significant risk to the public. Other dynamic load events could further increase the safety significance of this issue but, for prioritization purposes, this limited analysis will focus primarily on the seismic concern raised.

Possible Solution

The staff suggested[1014] a number of tests or alternative tests to provide adequate assurance of the operability of the LBHS when subject to a seismic event. The test options primarily focus on dynamic cyclic testing, to assure operability of the snubber control valves when subject to cyclic loads, and the determination (or correlation) of the snubber system spring rate when subject to cyclic loads. As previously stated, the resolution may effect all operating plants (BWRs and PWRs) that use LBHS as seismic restraint devices.

PRIORITY DETERMINATION

Frequency Estimate

Initiating Frequency: As stated above, the initiating event evaluated in this analysis focuses on the potential seismic-induced movement of steam generators in PWRs. The probability of failure of the SGS is 0.05[1016] for a peak ground acceleration of 0.5g, which is approximately three times that of a safe shutdown earthquake (SSE). The SGS failure probability corresponds to the failure probability of 0.05 for hydraulic snubbers from all design causes.[1017] Thus, given a failure of the LBHS in the SGS, the SGS under a 3SSE loading is assumed to have a failure probability of 1. Assuming the failure probability is proportional to the load, the failure probability of SGS and LBHS subject to a SSE loading is therefore 0.017.

The function of a LBHS during an earthquake is to lock-up and to resist motion of the steam generator. Failure to lock-up in either the compression or tension stroke will result in loss of snubber restraint (soft snubber strain rate under repetitive input loading). For purposes of this analysis, the inertial time lag (rocking) of the steam generator during its inertia-induced motion in one direction (snubber compression mode on one side and tension on the other side of the steam generator) combined with failure of the snubber to lock-up and resist the inertia-induced movement of the steam generator is considered. Thus, only one-half of the seismic input frequency is involved in the relative motion (rocking) between the steam generator and the snubber rigid attached wall.

Failure of the snubber from sticking of the control valve accounts for approximately 1% of the tested snubber failures.[1017] Assuming that the control valves are as likely to stick open (failure to lock-up) as to stick closed (failure to unlock), the LBHS failure to lock-up is $(0.017)(0.01)(0.5) = 8.5 \times 10^{-5}$/demand.

The strong ground motion of the SSE is assumed to contain an input frequency of 33 cycles/second over a duration of 10 seconds. Considering only one-half the input frequency as discussed above, the snubbers could experience (0.5)(33)(10) = 165 demands. The probability of the LBHS and SGS failure, given a SSE, would therefore be $(8 \times 10^{-5})(165) = 0.014$. A possible conservatism in this assumption is that all the LBHS in the SGS ganged (grouped LBHS arrangements) sets of snubbers are assumed to fail as one composite LBHS failure. Such a common mode failure is assumed representative of a generic design defect that results from the absence of adequate dynamic testing programs.

Core-Melt Frequency: Given a SSE event with a return period of 2×10^{-4}/RY (References 1018 and 1019), and the conditional failure probability of a SGS/SSE as (1.4×10^{-2}), the core-melt frequency is estimated at $(2 \times 10^{-4}/RY)(1.4 \times 10^{-2}) = (2.8 \times 10^{-6}/RY)$.

Containment Failure Frequency: For purposes of this analysis, the containment failure probability is assumed to be 1 due to overpressurization from the high energy released into the containment from the piping failures or containment bypass by way of the ruptured steam lines.

Consequence Estimate

Based on the above frequency estimates, the probability of a large release from a core-melt caused by dynamic (cyclic) failure of the LBHS during a SSE is $(2 \times 10^{-4}/RY)(1.4 \times 10^{-2})(1) = 2.8 \times 10^{-6}/RY$. The public dose within a 50-mile radius of the plant, with a surrounding uniform population density of 340 persons per square mile, no evacuation, and meteorology typical of the Braidwood site is $(2.8 \times 10^{-6}/RY)(5.1 \times 10^{6}$ man-rem$)$ = 14.3 man-rem/RY. Assuming an average remaining plant life of 30 years, the potential public risk is approximately 430 man-rem/reactor.

Cost Estimate

The cost of the proposed solution(s) will be highly dependent on the option selected to verify the dynamic capability of the LBHS in operating plants, i.e., a snubber vendor qualification of snubber types and/or in-plant tests that augment the current Technical Specifications (TS) functional test requirements. For operating plants, the cost will be highly dependent on the state-of-art of test equipment, the number of snubbers tested per plant, the surveillance frequency of the tests, the existence of or lack of prior qualification tests (snubber vendor-specific), the vintage and distribution of various vintage LBHS in the plants, and replacement power costs (should the LBHS tests result in extended plant outage time).

The expected large variations in all the above elements necessary to arrive at a realistic cost estimate for this issue clearly indicates that the costs used in this analysis must be regarded as very rough estimates.

Vendor Qualification Tests: The average cost for snubber qualification tests (including dynamic testing) is estimated to be \$100,000 per snubber type.[1017] This cost may be significantly higher per snubber type for the smaller population LBHS, but insignificant on an average per-plant basis when compared to other industry costs. Further functional (in-plant) tests of the LBHS that might augment the current TS requirements, given an adequate vendor qualification testing program (including dynamic testing), may be lower than the in-plant tests cost estimated in this analysis.

In-Plant Testing: The annual testing cost for hydraulic snubbers is estimated to be approximately \$1000/snubber. If we assume the snubber population ranges from 500 to 1000 snubbers/plant, and 15% of the snubbers are LBHS,[1017] each plant may have approximately 75 to 150 LBHS.[1020] Based on the current TS functional testing criteria,[1020] we assumed that 20% to 25% of the LBHS will be tested per refueling outage (approximately every 1.5 years). This amounts to 11 to 23 LBHS on an annual basis.

The current test requirements for LBHS are estimated to cost approximately (\$1000)(0.15)(75 to 150) = \$11,000 to \$23,000 per RY or, on an average, approximately \$17,000/RY. Estimating that a dynamic testing requirement (including setup, tests, and equipment leasing) would increase the current LBHS test cost by 50% to 100% yields an increased cost of approximately \$8,500 to \$17,000 per RY.

The present worth costs, at a 5% discount rate over 30 years, ranges from \$131,000 to \$262,000 per plant. These costs would be attributed to an in-plant dynamic testing requirement for LBHS only.

Replacement Power: Cost factors related to a hydraulic snubber test program according to the TS are cited in NUREG/CR-4279.[1017] The TS snubbers testing phase resulted in extending the plant outage time by approximately 3 days. Assuming the existing TS functional test surveillance requirements are linear with respect to the LBHS population (15%), the outage extensions due to the current LBHS testing may be extended an additional 0.45 day. If we estimate the outage extensions at only one additional hour per LBHS tested (11 to 23 LBHS), the outage extension ranges from one-half day to one day per year, which is consistent with the above estimate. Therefore, the estimated replacement power cost of \$500,000/day yields an annual replacement power cost of \$250,000 to \$500,000/plant year. Based on a 5% real discount rate, the present worth replacement power cost over 30 years may be \$3.85M to \$7.7M per plant.

NRC Cost: The estimated NRC cost for this issue ranges from \$50,000 to \$100,000, including technical assistance contractor costs. The effort would likely involve a review of the LBHS used in industry, determination of the need (risk reduction) for additional LBHS test requirements, discussions with snubber vendors, development of acceptable testing requirements, and possible preparation of additional TS requirements.

The NRC cost (\$50,000 to \$100,000) for this issue would be insignificant when compared to the industry costs. A per plant cost for vendor qualification of each snubber type would likely be distributed over the total population of the tested snubber type and not significantly affect total industry costs. In addition, some LBHS types may have existing and adequate testing programs. The present worth of surveillance testing costs (\$131,000 to \$262,000) and replacement power cost (\$3.85M to \$7.7M) yields a cost that ranges from approximately \$4M to \$8M per plant. Therefore, the average cost to implement a dynamic testing requirement in operating plants for the LBHS is estimated at approximately \$6M/plant, if plant outage time is extended because of the additional tests. If the LBHS test can be done within normal plant outages (refuelings), the total cost would be approximately \$200,000/plant.

Value/Impact Assessment

Based on a public risk reduction of 430 man-rem/plant and an estimated average cost of \$6M/plant (including replacement power costs), the value/impact score is given by:

$$S = \frac{430 \text{ man-rem}}{\$6\text{M}}$$

$$= 72 \text{ man-rem/\$M}$$

If replacement power costs are not involved, the value/impact score would be:

$$S = \frac{430 \text{ man-rem}}{\$0.2\text{M}}$$

$$= 2150 \text{ man-rem/\$M}$$

Other Considerations

The uncertainties in this analysis are large for both the risk and cost estimates. The risk for this issue is estimated to result from the absence of a LBHS dynamic qualification test requirement, or a LBHS dynamic functional surveillance test program in the TS. The risk estimates center on failure of the SGS due to a common mode failure of the LBHS in the event of a SSE seismic excitation. In estimating the probability of failure of the LBHS, it was assumed that the LBHS control valves may fail open during seismic excitation, e.g., due to cyclic (frequency) loading, and thereby result in free motion (lack of snubber restraint) and lateral movement of the steam generators. Lateral movement of the steam generators (absent LBHS restraint) is assumed to fail all the SGS and result in massive piping failures in the primary and secondary piping runs.

The current TS functional test requirements provide some unknown amount of assurance relative to cyclic operability to the LBHS, but the strain rate (k) during repetitive loadings is not assured and therefore may be even more uncertain. The analysis does not treat the above uncertainties in the estimates because of the lack of supportive data. Previous qualification testing (if dynamic tests were performed) might negate the need for further strain rate (dynamic) testing since it could be inferred that the operability and repetitive loading capacity for a given design was confirmed by the qualification tests and only operability need be verified in subsequent tests.[1021]

In the absence of any correlation between the LBHS operability and strain rate capacity when subject to dynamic loading, the assumption that control valves may stick open during the cyclic loading, and remain open, assumes a generic design defect might exist. This inherent assumption in the analysis may overestimate the probability of LBHS failure by one to two orders of magnitude. Thus, the risk estimates may be high for the analyzed SGS structure failure scenario.

The LBHS are also used to support other components in nuclear power plants. The consequences (plant damage states) from other dynamic loads or other accident scenarios that might be similar to the seismic-induced LBHS failures are assumed to be dominated by the seismic-induced SGS structure failure scenario. A more detailed analysis that examines all the potential plant damage states that may result from LBHS failures could be considered in the final resolution.

The resolution and implementation of the NRC Piping Review Committee recommendations (see Issue 119) may result in removal of some of the piping snubbers. Thus, more flexible piping systems may result in higher nozzle loads to the steam generators and other support structures during seismic events.[1018] In such cases, the reliability of the remaining smaller population of snubbers (including LBHS) may become more critical. As a further example, the GDC-4 limited scope rule would allow removal of many snubbers (including LBHS) from PWR reactor coolant system piping and large components, such as steam generators, subject to NRC approval of demonstrated acceptability of the licensees' requests. Additionally, the proposed GDC-4 broad scope rule change, if approved, will expand the scope of the limited scope rule to all high energy piping systems in all nuclear power plants. Thus, the determination of the need for dynamic qualification testing of the LBHS should also consider the potential impacts of Issue 119, and specifically the GDC-4 rule changes.

Some of the uncertainties in the cost estimates are due to the variations in snubber vendors' qualification programs, from plant to plant design differences, whether or not plant refueling outage times would be extended due to the additional testing, and other elements discussed above. Therefore, a wide uncertainty band for costs exists and no precise generic cost estimate seems appropriate for this issue, i.e., the costs may vary significantly from plant to plant. As previously stated, the cost estimates should be regarded as very rough estimates. The value/impact ratio of this issue will be strongly influenced by any outage extension that may or may not result from additional LBHS testing.

CONCLUSION

The purpose of this issue is to assess the need for an NRC requirement for dynamic qualification testing of LBHS in operating plants. The resolution of Issue A-13, "Snubber Operability Assurance," is expected to resolve the qualification testing requirements for future plants and snubber replacements on operating plants. Should NRC requirements on qualification testing of the existing LBHS in operating plants be determined necessary, the recommendations[1014] should be reviewed as possible candidate qualification test options.

The limited assessments provided in this analysis should only be considered as rough baseline risk, cost, and value/impact estimates. Further and more detailed analyses may show either higher or lower values. However, this analysis identifies that a broader and more complete evaluation is needed to resolve the issue. Based on the estimates determined in this analysis, the potential need for higher reliability LBHS (pipe snubber removal and optimization programs are currently being pursued), and the observed failures of LBHS in operating plants,[1015] we recommend that this issue be given a HIGH priority. Work being done by RES in the Nuclear Plant Aging Research,[1022] the resolution of Issue 119, and the effects of the GDC-4 rule changes should be considered in the resolution of this issue.

REFERENCES

1014. Memorandum for F. Schroeder from D. Crutchfield, "Dynamic Qualification Testing of Large Bore Hydraulic Snubbers," March 6, 1985.

1015. Memorandum for R. DeYoung, et al., from C. Heltemes, "Failure of Large Hydraulic Snubbers to Lock-up," September 21, 1984.

1016. NUREG/CR-4334, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, August 1985.

1017. NUREG/CR-4279, "Aging and Service Wear of Hydraulic and Mechanical Snubbers Used on Safety-Related Piping and Components of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1986.

1018. NUREG/CR-4263, "Reliability Analysis of Stiff Versus Flexible Piping Final Project Report," U.S. Nuclear Regulatory Commission, May 1985.

1019. NUREG/CR-3756, "Seismic Hazard Characterization of the Eastern United States," U.S. Nuclear Regulatory Commission, April 1984.

1020. NRC Letter to All Power Reactor Licensees (Except SEP Licensees) and All Applicants for Licensees to Operate Power Reactors, "Technical Specification for Snubbers (Generic Letter 84-13)," May 3, 1984.

1021. EPRI NP-2297, "Snubber Reliability Improvement Study," Electric Power Research Institute, March 1982.

1022. NUREG-1144, "Nuclear Plant Aging Research (NPAR) Program Plan," U.S. Nuclear Regulatory Commission, July 1985.

ISSUE 125: DAVIS-BESSE LOSS OF ALL FEEDWATER EVENT OF JUNE 9, 1985 - LONG TERM ACTIONS

On June 9, 1985, Davis-Besse had a partial loss of feedwater while operating at 90% power. Following a reactor trip, the loss of all feedwater occurred. The two OTSGs became dry and were ineffective as a heat sink. Consequently, the RCS pressure increased indicating a lack of heat transfer from the primary to secondary coolant systems. The PORV automatically opened and closed twice during the event upon reaching the approximate pressure setpoints; it opened a third time, but did not close for some unknown amount of time. The delayed response to close the third time aggravated the recovery of the event and allowed a rapid depressurization of the RCS.

In addition to the short-term actions identified and addressed in Issue 122, a staff report on the event was published in NUREG-1154[886] and an EDO memorandum[895] identifying 29 NRR action items was issued on August 5, 1985. These items became known as long-term generic actions and, in November 1985, were forwarded by DL to DST for prioritization.[940] The items were broken down into two groups: (I) Issues raised in NUREG-1154 and the EDO memorandum and (II) Other Issues. These 29 items are prioritized separately below and are identified by the numbering system established in the DL memorandum.[940]

ITEM 125.I.1: AVAILABILITY OF THE SHIFT TECHNICAL ADVISOR

DESCRIPTION

Historical Background

This issue is one of a list of long-term generic issues which arose during the investigation of the loss of all feedwater event which took place at the Davis-Besse plant on June 9, 1985.[940] During the event, neither the shift supervisor nor any of the other licensed operators requested the assistance of the shift technical advisor (STA). One reason for not doing so was the fact that the STA was not in the control room or immediately available when the event occurred, but rather was on an on-call status. (Note: he is allowed 10 minutes to reach the control room after being called.) Moreover, the event occurred so rapidly that it was essentially over when he did arrive.

STAs were first required as part of the TMI Action Plan Item I.A.1.1, "Shift Technical Advisor." The purpose of the STA was to provide readily available technical support to the plant operators. The STA's expertise was intended to aid in the mitigation of those transients and accidents which involve complex thermal-hydraulic behavior in the primary and secondary coolant systems. In summary, having the STA available was a post-TMI improvement to provide the shift supervisor with additional technical expertise, but his potential assistance and guidance was not available nor required during this event.[886]

Safety Significance

The safety question posed by this issue is whether the STA should be in the control room, or immediately available, to support the shift supervisor rather than being on an on-call status.

CONCLUSION

One year after the Davis-Besse incident, the staff conducted a survey to fulfill a Staff Requirements Memorandum to provide the Commissioners with the implementation results of the Commission Policy Statement on engineering expertise on shift and reported their findings in SECY-86-231.[1023] This survey found that there were only three plants that did not have "on-shift" STAs. On-shift STA means that there is an STA, or an STA-qualified SRO, in or near the control room on a shift basis during operations. The STA shift may or may not correspond to the same shift times and length as the licensed operators' shift. It further means that the STA does not work on an extended assignment period, e.g., 24 hours, during which time the STA is provided quarters to rest during a portion of his extended duty and is available on an on-call basis.

Based on the staff's findings, [1023] STAs are in the control room or immediately available at the majority of operating plants. For the three plants identified with a deficiency, licensee action is being reviewed by the staff on a plant-specific basis. Thus, this item should be DROPPED as a generic issue.

ITEM 125.I.2: PORV RELIABILITY

The PORV common to most PWRs (with the exception of CE 3410 and 3800 Mwt plants and ANO-2) is designed to limit system pressure if a transient recovery exceeds the capability of the pressurizer spray system. Davis-Besse has a solenoid-controlled PORV. However, many other PWRs have PORVs that are operated pneumatically (instrument air or nitrogen). Both designs have the same purpose. The PORV is designed to receive an actuation signal to open from the pressurizer pressure instrumentation at a design setpoint (typically 2425 psig) in order to prevent reactor pressure from rising and activating the code safety valves.

If a PORV is used for feed-and-bleed, it can either be: (1) set to stay open by the operator dropping the setpoint low enough such that the valve will remain open until reaching the lower setpoint for LPIS or RHR initiation, or (2) cycled open and closed many times, should there be a need for feed-and-bleed. Option 1 appears to be the more common practice. PORVs are also used in other functions such as mitigating SGTR accidents, LTOP, or RCS venting. Its performance is required for plant protection and accident mitigation.

The following is the evaluation of the four parts of this issue.

ITEM 125.I.2.A: NEED FOR A TEST PROGRAM TO ESTABLISH RELIABILITY OF THE PORV

DESCRIPTION

Historical Background

This issue was identified as Item 9c in the EDO memorandum[895] and is based on Finding 13 and Section 5.2.8 of NUREG-1154.[886]

Safety Significance

Although the PORV can be used successfully in recovering from certain plant transients, there has been no suitable test program established to verify its reliability.[886] This issue affects all PWRs that can use PORVs.

CONCLUSION

The need for improving the reliability of PORVs and block valves, in light of plant protection and accident mitigation requirements, is being addressed in the resolution of Issue 70, "PORV and Block Valve Reliability." Revised licensing criteria may be developed, if needed, to include testing requirements.[896] Therefore, this issue is covered in Issue 70.

ITEM 125.I.2.B: NEED FOR PORV SURVEILLANCE TESTS TO CONFIRM OPERATIONAL READINESS

DESCRIPTION

Historical Background

This issue was identified as Item 9d in the EDO memorandum[895] and is based on Finding 13 and Section 5.2.8 of NUREG-1154.[886]

Safety Significance

The review of the PORV maintenance and operating history reveals that the mechanical operation of the valve had not been tested and that the valve had not otherwise been operated for over 2 years and 9 months prior to the June 9, 1985 event. Therefore, it seems that there exists a need for surveillance tests to confirm operational readiness. This issue affects all PWRs that can use PORVs.

CONCLUSION

The number of times that PORV/Block Valves are used during a typical fuel cycle will be reviewed in the resolution of Issue 70, "PORV and Block Valve Reliability," in order to determine if a surveillance program should be initiated to confirm operational readiness.[896] Therefore, this issue is covered in Issue 70.

ITEM 125.I.2.C: NEED FOR ADDITIONAL PROTECTION AGAINST PORV FAILURE

DESCRIPTION

Historical Background

This issue was identified as Item 9e in the EDO memorandum[895] and is based on Sections 5.2.8 and 6.2.1 of NUREG-1154.[886]

Safety Significance

The PORV will receive an actuation signal from pressurizer pressure instrumentation at a design setpoint (typically 2425 psig) to open in order to prevent reactor pressure from activating the code safety valves. After the opened PORV has reduced the pressure sufficiently to reach its closure setpoint (typically 2375 psig), it is sent a signal to close. A simultaneous signal is also sent to the control room indicating to the operator that a close signal was sent to the PORV. PORV closure can be verified by an acoustic monitor installed on the tailpipe downstream of the PORV on all PWRs after the TMI-2 accident. At Davis-Besse, the PORV closure is indicated by a light located on a wall several feet from the operator's control panel. This was available to the operator at Davis-Besse to verify whether the PORV was closed, but was not looked at. Additionally, there is the SPDS, also a post-TMI improvement, that displays a summary of the most safety significant plant status information on a TV screen. Both channels were inoperable prior to the event.[886] This left the operators with only the pressurizer pressure indicator as a source of determining if the PORV was open or closed. Since the indicator appeared steady, the operator assumed that the PORV had closed, but closed the block valve as a precautionary measure. In actuality, however, the PORV had not closed until some time later into the event.

There have been several stuck open PORVs documented due to a variety of malfunctions some of which were identified to be mechanical failure, broken solenoid linkage, inoperability due to corrosion buildup, and sticking caused by foreign material.[886] As a precaution, the PORV block valve can be closed to insure no LOCA, but this can only be achieved if the operator closes the block valve by remote-manual operation from the control room. In the Davis-Besse event, the operator did close the block valve to prevent a further decrease in pressure and loss of primary coolant through the PORV when it did not reseat.

Possible Solution

Knowing that a stuck-open PORV may result in a potentially dangerous scenario (i.e., LOCA), this issue addresses the concern of whether there is a need for an automatic block valve closure in plants that have PORVs.

Considering available control room indicators such as an acoustic monitor, a reliable SPDS and the operator's acute sensitivity to the PORV's status because of historical events such as TMI-2 and Davis-Besse, another redundant feature (i.e., automating the block valve) would not necessarily result in a significant decrease in core-melt frequency. The acoustic monitor was available to the operator at Davis-Besse; the SPDS was not. However, there is an NRC requirement for the installation of "a concise display of critical plant variables to the control room operators to aid them in rapidly and reliably determining the safety status of the plant."[376]

Additionally, there is a DHFT program underway "to determine the need for and, if necessary, the scope of the NRC's SPDS post-implementation reviews."[900] The information obtained will "allow an assessment of how well the SPDS objectives are being met and provide the basis for an NRC regulatory position on SPDS post-implementation reviews. Following completion of this program DHFT will, if necessary, work with industry to develop appropriate standards for SPDS availability."[900]

The staff performed SARs on the three vendor group responses (CE, B&W, W) to TMI Action Plan Item II.K.3(2), "Report on Overall Safety Effect of Power-Operated Relief Valve (PORV) Isolation System." (References 897, 898, and 899). The SARs included an estimate of core-melt frequency due to a stuck open PORV-induced SBLOCA. The calculations were based on PORV operating data from April 1, 1980 to March 31, 1983 and concluded that post-TMI actions such as lowering the setpoint of the high pressure reactor trip and raising the setpoint of the PORV opening, eliminating the turbine runback feature, and improving operator capability decreased the challenge to the PORV and the probability of a SBLOCA-PORV sufficiently so as not to warrant a requirement for automatic block valve closure.

The Davis-Besse event may be viewed as another "data point" that should be considered in this determination. However, upon consideration of the occurrence of a PORV actuation and the conservative estimates made in the staff's SARs (References 897, 898, and 899), we conclude that the SBLOCA-PORV frequency would still remain within the range of the SBLOCA frequencies given in WASH-1400[16] (10^{-2} to 10^{-4}/RY). The opening of the PORV resulted from a loss of all feedwater to the steam generators and is regarded as a legitimate response and fulfillment of the real purpose for incorporating a PORV into the design. Therefore, the Davis-Besse event does not change the statistics for necessary challenge to the PORV. Consequently, the staff's SARs (References 897, 898 and 899) which concluded that block valve automation is unnecessary are unaffected.

Also it is clear that the automation of the block valve might reduce the initiator (SBLOCA-PORV) frequency, but not necessarily the net core-melt frequency. Since it has the potential for spurious actuation (e.g., spurious electrical signal sensed by the block valve could force it closed during a transient requiring use of the PORV) which would increase core-melt frequency.

The occurrence at Davis-Besse was the result of an initiator already considered in the SARs, i.e., the failure of the AFW system. It was an occurrence that would have resulted in no other outcome should an automatic block valve have been available because the operator closed the block valve himself as a result of his sensitivity to the PORV from post-TMI training.

CONCLUSION

In light of the control room indications available to the operators and the results of the staff SARs (References 897, 898 and 899) that concluded that an automatic PORV isolation system is not necessary, the safety concerns of this issue have been resolved. Thus, this issue should be DROPPED as a new issue.

ITEM 125.I.2.D: CAPABILITY OF THE PORV TO SUPPORT FEED-AND-BLEED

DESCRIPTION

Historical Background

This issue was identified in the EDO memorandum[895] and was also raised at an ACRS Subcommittee meeting on Emergency Core Cooling Systems held on July 31, 1985.

Safety Significance

Upon loss of the main and auxiliary feedwater systems, the feedwater flow to the steam generators is insufficient to maintain level. As the level of water in the steam generators decreases, the average temperature of the RCS increases because of the reduced heat transfer from the primary to the secondary coolant systems. When all steam generators are "dry," the plant emergency procedure requires the initiation of makeup/high pressure injection (MU/HPI) cooling of the primary system.[886] This method of decay heat removal is known as "feed-and-bleed" or "bleed-and-feed" depending on the HPI capability of the injection pumps and system design. When this method is initiated, the PORV and high point vents on the RCS, specifically the pressurizer, are locked open breaching one of the plant's radiological barriers and releasing radioactive coolant inside the containment building.[886] MU/HPI is often considered a drastic action because of the radioactive contamination of the containment. Nevertheless, MU/HPI cooling provides a diverse method of core cooling if the main and auxiliary feedwater systems should fail.

This issue is based on an ACRS concern that the PORVs are not qualified for the "hostile" environment in which they are placed when used for feed-and-bleed operation. There are several reasons for this concern. PORVs are usually called upon to respond when all other methods of removing decay heat are not available. The temperature, pressure, and moisture conditions of the containment environment can create a differential thermal expansion of the valve disc and body and may cause the PORV to stick,[886] failing open or closed, or the PORV can close shortly after beginning feed-and-bleed because of short circuits.

CONCLUSION

Under USI A-45, "Shutdown Decay Heat Removal Requirements," the NRC staff is investigating alternative means of decay heat removal in PWR plants using existing equipment or devising new methods. The use of the "feed-and-bleed" procedure is included in this program as well as the need for environmental qualification of the PORV for this method of emergency decay heat removal. Therefore, this issue is covered in USI A-45.[896]

ITEM 125.I.3: SPDS AVAILABILITY

This item is currently being prioritized.

ITEM 125.I.4: PLANT-SPECIFIC SIMULATOR

DESCRIPTION

Historical Background

This issue was identified as Item 10c in an EDO memorandum[895] which contained a list of NRR action items resulting from the Davis-Besse event on June 9, 1985. Item 10C was based on Findings 10 and 17 and Sections 6.1.1 and 6.1.2 of NUREG-1154.[886] Following the Davis-Besse reactor trip, the operator manually initiated actuation of the Steam and Feedwater Rupture Control System (SFRCS) in anticipation of the automatic initiation of the SFRCS; however, the operator pushed the wrong buttons. This was the first time he had manually actuated the SFRCS and had not received specialized classroom or simulator training on correctly initiating the SFRCS. The buttons pushed by the operator activated the SFRCS on low pressure for each steam generator instead of low level. By manually actuating the SFRCS on low pressure, the SFRCS was signalled that both steam generators had experienced a steamline break or leak and the system responded, as designed, to isolate both steam generators. Thus, the operator's anticipatory action defeated the safety function of the AFW system. The error was corrected within approximately one minute by resetting the SFRCS and, therefore, had no significant bearing on the outcome of the event. However, the lack of plant-specific simulator training was noted by the investigating team.

This event, however, was not the first event that indicated the need for plant-specific simulator training. The TMI-2 event on March 28, 1979, clearly focused industry and NRC attention on the need for better human engineering in control room design and for plant-specific simulator training. TMI Action Plan Task I.A[48] contained a series of requirements related to simulator uses and developments addressing short-term and long-term actions centered on simulator training. Some of the Task I.A items[48] were subsequently integrated into the Human Factors Program Plan (HFPP)[651] which was developed in response to NUREG-0885[210] and Section 306 of the Nuclear Waste Policy Act of 1982 (PL 97-425). In this regard, PL 97-425 required NRC to establish simulator training requirements for plant-licensed operators and operator requalification examinations. Item I.A.4.1, "Initial Simulator Improvement," has been completed; the "Long-Term Training Simulator Upgrade" [Item I.A.4.2(4)] will be completed upon publication of 10 CFR 55 and related NRC guidance on the evaluation of simulation facilities.

Safety Significance

A plant-specific simulator would improve operator actions and timing in response to plant transients and accidents. Thus, plant damage and possible core-melt accidents could be significantly reduced. This issue affects all licensed nuclear power plants.

Possible Solution

The use of plant-specific simulators is being addressed in the proposed rulemaking[957] amendments to 10 CFR 55 [TMI Action Plan Item I.A.4.2(4)]. This action will codify requirements that include the use of nuclear power plant simulators in initial and requalification examinations. In brief, the proposed rulemaking includes three choices for plants that are not the reference plant for a simulator: (1) acquire a plant-referenced simulator that meets the intent of Regulatory Guide 1.149[439]; (2) use a simulator that conforms to Regulatory Guide 1.149[439] and has been demonstrated to be suitable; or (3) substitute any device or combination of devices that meets the requirements of 10 CFR 55.45(b) and would be approved by the NRC.

CONCLUSION

Based on the above, the resolution of the need and use of plant-specific simulators is being addressed as part of the proposed rulemaking amending 10 CFR 55 under Item I.A.4.2(4). Thus, Issue 125.I.4 should be DROPPED as a separate issue.

ITEM 125.I.5: SAFETY SYSTEMS TESTED IN ALL CONDITIONS REQUIRED BY DBA

This item is currently being prioritized.

ITEM 125.I.6: VALVE TORQUE, LIMIT AND BYPASS SWITCH SETTINGS

DESCRIPTION

Historical Background

This issue is one of a list of longer-term generic issues which arose during the investigation of the loss of all feedwater event which took place at the Davis-Besse plant on June 6, 1985.[940]

One of the primary sources of failure of the Davis-Besse AFW isolation valves to reopen (see Issue 122.1) was ultimately traced to the torque, limit, and bypass switches which control the motor operators of the valves. During the event, these valves were closed due to an operator error, shutting off all AFW flow. Once closed, the resulting high differential pressure across the closed valves necessitated a relatively large force to start valve motion. The valve motor-operator torque bypass switches were not adjusted to accommodate such a force and manual operation was needed to reopen the valves.

Issue 122.1.a, "Failure of Isolation Valves in Closed Position," deals specifically with the case of AFW isolation valves. However, at least some of the other motor-operated valves in the plant are designed by the same people that designed the AFW system and virtually all the valves in the plant are maintained by the same crews. Therefore, the problems with torque, limit, and bypass switch settings are not limited to AFW systems, but may affect any motor-operated valve in the plant. Moreover, such problems have a high potential for causing common mode failures since redundant trains are probably maintained by the same maintenance personnel.

Safety Significance

The safety concern of this issue is exactly that of IE Bulletin No. 85-03,[1036] "Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings." This Bulletin required all licensees to develop and implement a program to ensure that valve operator switches are selected, set, and maintained properly for all valves in the high pressure injection, core spray and emergency feedwater systems (including BWR RCIC), that are required to be tested for operational readiness in accordance with 10 CFR 50.55a(g).

Possible Solution

IE Bulletin 85-03[1036] should resolve the safety concern of this issue for switch settings on valve operators in these specific safety systems. The extension of this issue to other valves and/or extension of the issue to more general testing adequacy also needs to be considered. However, the general question of test adequacy for all safety-related valves is the subject of Issue II.E.6.1, "Test Adequacy Study." Given the existence of II.E.6.1, there is no need to extend or generalize this new issue.

CONCLUSION

The safety concern of this issue is being addressed by IE Bulletin 85-03[1036] and in the resolution of Issue II.E.6.1. Thus, Item 125.I.6 should be DROPPED as a separate issue.

ITEM 125.I.7: OPERATOR TRAINING ADEQUACY

This item was broken down into two parts that were evaluated separately as shown below.

ITEM 125.I.7A: RECOVER FAILED EQUIPMENT

DESCRIPTION

Historical Background

This issue is one of a list of longer-term generic issues which arose during the investigation of the loss of all feedwater event which took place at the Davis-Besse plant on June 6, 1985.[940]

Safety Significance

The issue is based upon Finding 8 of the Incident Investigation Team's report[886] which states:

> "The operators' understanding of procedures, plant system designs, and specific equipment operation, and operator training all played a crucial role in their success in mitigating the consequences of the event. However, if the equipment operators had been more familiar with the operation of the auxiliary feedwater pump turbine trip-throttle valve, auxiliary feedwater could have been restored several minutes sooner."

During the Davis-Besse event, both AFW turbines tripped on overspeed. These trips are not remotely resettable from the control room, but instead must be reset manually at the turbines. Two equipment operators were dispatched to the AFW turbines, but were unable to get the turbines running because they had never performed this operation before. (Hands-on practice of this task is not now a part of operator training.) The turbines were not started until after the arrival of a more experienced operator.

The safety significance of this issue lies in the probability of non-recoverability of safety systems. In many cases, a given train of a given system may trip or otherwise fail to start on first demand, but may still successfully be placed in operation by prompt, knowledgeable human intervention.

Possible Solutions

TMI Action Plan Items I.A.2.2 and I.A.2.6 have addressed the issue of training and resulted in a policy statement[966] that endorsed the Institute of Nuclear Power Operations-managed training accreditation program which includes an element to ensure that feedback from operating events is included in all utility training programs. NRC monitors and evaluates industry implementation of the INPO accreditation program to ensure that: (1) plant personnel are able to meet job performance requirements; (2) training properly accounts for pertinent safety issues; and (3) mechanisms exist for upgrading and assuring the quality of training programs. Criteria to evaluate the industry training programs have been developed in NUREG-1220[993] in the resolution of Human Factors Issue HF2.1.

CONCLUSION

This issue has been resolved by the issuance of the Commission Policy Statement[966] on Training and Qualifications and by Issue HF2.1. Therefore, a new and separate issue for this concern is not warranted and the issue should be DROPPED from further consideration.

ITEM 125.I.7.B: REALISTIC HANDS-ON TRAINING

DESCRIPTION

Historical Background

The issue calls for an assessment of the adequacy of hands-on training with respect to conditions that may be encountered in realistic situations, such as the loss of feedwater event that occurred at the Davis-Besse plant on June 9, 1985.[940] The assessment may involve the operator's understanding of procedures, plant systems designs, specific equipment operations, and hands-on training in handling plant transient and upset conditions.

The issue stems from Findings 8 and 16 of the NRC investigation[886] of the Davis-Besse event in which the NRC staff noted that the post-TMI improvements that focused on EOPs and training played a crucial role in mitigating the Davis-Besse event. However, if the equipment operators had been more familiar with the operations of the AFW pump turbine trip throttle valve, AFW could have been restored several minutes sooner. Also, for events such as the Davis-Besse event involving conditions outside the plant design basis (multiple equipment failures), operator training and operator understanding of systems and equipment are crucial to the likelihood that plant operators can successfully handle similar events.

Safety Significance

Assessments of the hands-on experience, referred to as performance-based training or Systems Approach to Training (SAT), are considered essential to providing assurance that nuclear power plants are operated in a safe state under all operating conditions. This issue effects all operating nuclear power plants.

Possible Solution

TMI Action Plan[48] items I.A.2.2 and I.A.2.6 included development of procedures to provide assurance that: (1) plant personnel are able to meet job performance requirements; (2) training properly account for pertinent safety issues; and (3) mechanisms exist for upgrading and assuring the quality of training programs.

To help meet these objectives, NUREG-1220[993] was developed for use by NRC personnel to review the INPO-managed performance-based training programs in nuclear power plants. NRC will continue to closely monitor the process (INPO Accreditation) and its results to independently evaluate implementation of these programs. The NRC review procedures developed in NUREG-1220[993] considered the following five elements as essential to these training programs: (1) systematic analysis of the jobs to be performed; (2) learning objectives that are derived from the analysis and that describe desired performance after training; (3) training design and implementation based on the learning objectives; (4) evaluation of trainee mastery of the objectives during training; and (5) evaluation and revisions of the training based on the performance of trained personnel in job settings (hands-on experience).

In accordance with NUREG-0985,[651] the training issues included the closeout of the following TMI Action Plan[48] items: I.A.2.2, "Training and Qualifications of Operations Personnel"; I.A.2.7, "Training Accreditation"; I.A.2.5, "Plant Drills"; and I.A.2.3, "Administration of Training Programs." The specific issue of realistic hands-on training on equipment such as AFW pumps is a performance-based element of on-the-job training (OJT). As such, mastery is determined by completion of a job qualification card to the satisfaction of a qualified OJT instructor using approved evaluation criteria. The INPO Accreditation Program is intended to provide assurance that such training is included in industry programs. NRC evaluates industry implementation of the Accreditation Program in accordance with the Policy Statement on Training and Qualification.[966]

CONCLUSION

Based on the above discussion, this issue is covered by the Policy Statement[966] on Training and Qualifications and by the Human Factors Issue HF3.1. Therefore, a new and separate issue for this concern is not warranted and the issue should be DROPPED from futher consideration.

ITEM 125.I.8: PROCEDURES AND STAFFING FOR REPORTING TO NRC EMERGENCY RESPONSE CENTER

DESCRIPTION

Historical Background

This issue arose out of the Davis-Besse incident of June 9, 1985[940] and is based upon Finding 12 of the Incident Investigation Team's report[886] which states:

> "The event was not reported to the NRC Operations Center in a manner reflecting the safety significance of the event. The more serious the event, the more operator involvement required to maintain plant safety. For example, if the June 9 event had been protracted, knowledgeable personnel would not have been available to maintain an open telephone line with the NRC."

Safety Significance

It is evident from the Incident Investigation Team's report[886] of the event that there were two problems: one associated with staffing and one associated with procedures. The staffing problem was that all knowledgeable personnel were kept busy in dealing with the event. No one could be spared to keep the NRC Operations Center informed. Moreover, even if more plant staff had been available, it is likely that these additional persons would have been pressed into service for plant operations. Of course, bringing the plant to a safe condition does and should have priority. But this also calls into question the usefulness of the dedicated phone lines to the NRC Operations Center.

The procedural problem was evident in the fact that there was confusion because the emergency plan was silent on how to determine the emergency action level if the emergency classification changed during the event. Obviously, the emergency procedures contained some ambiguity.

For both problems, the result is a delay in notification of the NRC Operations Center. Although it can be argued that notification of the NRC can have little or no effect on plant events in the short term, the NRC can provide technical support and assistance over a period of several hours. Moreover, the NRC can assist in coordinating evacuations, etc., if such should ever prove necessary. Finally, the NRC has other responsibilities not directly related to plant safety but nevertheless of importance, such as providing accurate and timely information to the public, other government agencies, and the governments of other nations.

PRIORITY DETERMINATION

The staffing problem is a duplication[1003] of the concern of TMI Action Plan[48] Item III.A.3.4, "Nuclear Data Link." In addition, the procedural problem has already been addressed in existing regulatory requirements (10 CFR 50.72) and IE Information Notice No. 85-80. Furthermore, the IE Manual addresses the NRC regional responsibility for assuring that these reporting requirements are met.[1003]

CONCLUSION

This issue consists of two problems: the first is a duplication of TMI Action Plan[48] Item III.A.3.4 (which has been resolved) and the second has been resolved independently.[1003] Therefore, this issue should be DROPPED from further consideration as a new and separate issue.

ITEM 125.II.1: NEED FOR ADDITIONAL ACTIONS ON AFW SYSTEMS

During the event, the main feedwater system was lost and the reactor scrammed. The AFW system should have activated and supplied feedwater to the steam generators to enable them to remove decay heat. However, during the course of the event, several failures occurred (see Issue 122) that precluded using the steam generators to remove decay heat from the primary system. The event highlighted the importance of the AFW system and also demonstrated that the AFW system might not have a reliability commensurate with its importance.[940]

If the main feedwater system shuts down for any reason, the AFW system will supply sufficient feedwater to the steam generators to remove reactor decay heat. If the AFW system were to fail also, there would be no feedwater supply at all. The steam generators would boil off their remaining liquid water inventory and then dry out. Depending on specific plant design, core uncovery will take place roughly 30 to 90 minutes after the transient begins. After steam generator dryout, there would be no decay heat removal and the continuing thermal energy production in the core would result in primary system heatup.

In most cases, the only means of decay heat removal involve use of the AFW system, recovery of the main feedwater system, or the use of feed-and-bleed techniques. Of the three means, the use of the AFW system is subject to the highest availability. The failure of the main feedwater system has roughly a 20% probability of not being recoverable in time. Moreover, use of feed-and-bleed techniques will release primary coolant to the containment necessitating extensive (and expensive) cleanup. The use of feed-and-bleed techniques, which remove decay heat by venting hot primary coolant to the containment and replacing the lost inventory in the primary system by means of the high pressure ECCS, could still prevent core uncovery. If feed-and-bleed fails, the primary system will increase in temperature and pressure to the point where the primary system safety valves open. The pressure increase will then terminate, but the primary coolant will boil off until the core is uncovered and melts.

AFW systems are safety-grade systems. In addition, the availability of feed-and-bleed techniques provides a diverse backup. Nevertheless, AFW reliability is very important for two reasons. First, loss of main feedwater is a relatively common event, occurring roughly three orders of magnitude more often than (for example) small break LOCAs. Thus, the AFW system is challenged far more often than the high pressure ECCS and therefore has a commensurately greater need for high reliability. Second, although feed-and-bleed techniques provide a backup to AFW for removing reactor decay heat, feed-and-bleed is a means of core cooling for which the plant was not designed and may have a relatively high failure probability (see Item 125.II.9). Because of these two reasons (frequent challenges and poor backup capability), it is very important that the AFW system have very high reliability.

Because loss of feedwater events are relatively frequent, the AFW system is subject to frequent challenges. Therefore, the AFW system must be characterized by very high availability. This issue consists of four parts, each of which seeks to ensure adequate AFW reliability:

(a) Two-Train AFW Unavailability
This issue is concerned that AFW systems consisting of only two-trains may not have adequate reliability.

(b) Review Existing AFW Systems for Single Failures
This issue seeks confirmatory deterministic reviews of AFW systems at operating plants to ensure that they meet the single failure criterion.

(c) NUREG-0737 Reliability Improvements
This issue proposes that PRA analyses (i.e. fault trees) be performed on AFW systems at operating plants to ensure adequate reliability.

(d) AFW Steam and Feedwater Rupture Control System/ICS Interactions in B&W Plants
This issue is concerned explicitly with a possible design problem at B&W plants.

These four parts of the issue are prioritized separately below.

ITEM 125.II.1.A: TWO-TRAIN AFW UNAVAILABILITY

DESCRIPTION

There are seven older PWRs that have two-train AFW systems. (Originally, there were more but some plants have since added a third train or made other equivalent upgrades). These AFW systems generally consist of one motor-driven train and one turbine-driven train and thus possess some diversity as well as redundancy. However, the turbine-driven trains have not proven to be as reliable as the motor-driven trains (except, of course, for the case where all AC power is lost). The more modern practice has been to use a three-train system where two trains are motor-driven and one is driven by a steam turbine. Such a system will, in principle, be more reliable than the two-train systems described above, both because of the greater redundancy of the three vs. two trains and because of the lower reliance on the steam turbine.

CONCLUSION

This issue is the same as Issue 124, "AFW System Reliability." Issue 124 will consider whether AFW system unavailability needs to be improved for plants with two-train designs.[947] Therefore, this issue should be DROPPED as a separate issue.

ITEM 125.II.1.B: REVIEW EXISTING AFW SYSTEMS FOR SINGLE FAILURE

DESCRIPTION

Historical Background

The AFW system is considered an engineered safety feature and thus is required to meet the single failure criterion which can be considered a very primitive reliability requirement. An unsuspected single failure susceptibility could increase the AFW system failure probability by two orders of magnitude or more.

Safety Significance

The issue addresses the concern that there may be some unsuspected single failures which were not detected during the licensing process. Therefore, this issue proposes to re-review the AFW systems of all operating PWRs to make doubly sure that no single failures exist which by themselves could cause all AFW trains to fail.

Proposed Solution

The systems to be examined have already been subjected to licensing review. Therefore, any single failures are not going to be obvious, but instead are likely to be quite subtle. Very thorough reviews will be required. It must also be remembered that AFW trains are intentionally designed to be independent. Any single failure found is most likely to be a subtle design anomaly which the designer (as well as all subsequent reviewers) failed to notice.

Several AFW systems have been examined by OIE in the course of the Safety System Functional Inspection (SSFI) program. Conversations with the SSFI team have indicated that some single failure problems as well as other potential common mode failures have been found by this program. However, these problems were not discovered by examining system design, but instead arose in the course of very thorough investigations involving extended site visits, equipment inspection, and interviews as well as design reviews. Therefore, the proposed solution is not a simple design review, but instead is a more thorough investigation along the lines of the SSFI program.

Frequency Estimate

The sequence of interest is straightforward. It is initiated by a non-recoverable loss of main feedwater. If the AFW system fails, the SUFP is not re-enabled in time, and feed-and-bleed techniques fail, core-melt will ensue. For the initiating event frequency (non-recoverable loss of main feedwater), we will use 0.64 event/RY, based upon the Oconee PRA done by Duke Power Co.[947] This figure is based upon fault tree analysis and should be reasonably representative of most main feedwater system designs.

For a three-train AFW system, a "typical" unavailability is 1.8×10^{-5}/demand.[894] The presence of a single failure susceptibility will greatly increase this figure to perhaps the square root of the original figures because half the redundancy would be removed. The change in AFW unavailability would then be about 4.2×10^{-3} failure/demand. We will assume a typical value of 0.20 for the failure probability of feed-and-bleed cooling, based upon the calculations presented under Issue 125.II.9, "Enhanced Feed-and-Bleed Capability." Multiplying these figures out, the change in core-melt frequency is:

$$(0.64/\text{year})(4.2 \times 10^{-3})(0.20) = 5.4 \times 10^{-4}/\text{year}$$

Consequence Estimate

The core-melt sequence under consideration here involves a core-melt with no large breaks initially in the reactor coolant pressure boundary. The reactor is likely to be at high pressure (until the core melts through the lower vessel head) with a steady discharge of steam and gases through the PORV(s). These are conditions likely to produce significant hydrogen generation and combustion.

The Zion and Indian Point PRA studies used a 3% probability of containment failure due to hydrogen burn (the "gamma" failure). We will follow this example and use 3%, bearing in mind that specific containment designs may differ significantly from this figure. In addition, the containment can fail to isolate (the "beta" failure). Here, the Oconee PRA figure of 0.0053 will be used. If the containment does not fail by isolation failure or hydrogen burn, it will be assumed to fail by basemat melt-through (the "epsilon" failure).

Using the usual prioritization assumptions of a central midwest plains meteorology, a uniform population density of 340 persons per square mile, a 50-mile radius, and no ingestion pathways, the consequences are:

Failure Mode	Percent Probability	Release Category	Consequences (man-rem)
gamma	3.0%	PWR-2	4.8×10^6
beta	0.5%	PWR-5	1.0×10^6
epsilon	96.5%	PWR-7	2.3×10^3

The "weighted-average" core-melt will have consequences of 1.5×10^5 man-rem.

There are 80 PWRs operating or under construction. As of March 1988 (the earliest that any hardware changes are likely to be made), these 80 plants will have a combined remaining license lifetime of 2508.4 calendar-years. At a 75% capacity factor, this is about 23.5 years of operation per plant. Thus, the estimated risk reduction associated with the possible solution to this issue is $(5.4 \times 10^{-4})(23.5)(1.5 \times 10^{-5})$ man-rem/reactor or 1904 man-rem/reactor.

Cost Estimate

The SSFI program has required about 1000 staff-hours per plant and system. This is about $50,000 of salary and overhead. In addition, hardware changes are likely to cost on the order of $100,000 per plant (i.e. more than $10,000 but less than $1,000,000) plus another $50,000 in paperwork. Thus, we will assume a cost on the order of $200,000/plant.

Value/Impact Assessment

Based on a potential risk reduction of 1,904 man-rem/reactor and a cost of $0.2M/reactor, the value/impact score is given by:

$$S = \frac{1{,}904 \text{ man-rem/reactor}}{\$0.2\text{M/reactor}}$$

$$= 9{,}520 \text{ man-rem/\$M}$$

Other Considerations

(1) The AFW system and its support systems do not contain contaminated fluids and are located outside of containment. Thus, there is no ORE associated with the fix for this issue.

(2) Averted accident costs and averted cleanup exposure are considerations, but will only drive the priority figures still higher. Thus, they will change no conclusions and will not be treated here.

(3) The high values of the parameters are predicated on finding at least one plant that needs upgrading. The SSFI personnel emphasized that this is not likely to happen without an approach similar to that of the SSFI, but such an approach is likely to bear fruit. It may be feasible to incorporate this issue into the SSFI program.

CONCLUSION

Based upon the figures generated above, this issue was given a high priority, but was later integrated into the Phase II activities scheduled for the resolution of Issue 124.[973] Thus, this issue is now covered in Issue 124.

ITEM 125.II.1.C: NUREG-0737 RELIABILITY IMPROVEMENTS

DESCRIPTION

Historical Background

After the TMI-2 accident, all PWR licensees were asked to perform an unavailability analysis of their AFW systems. This information is now somewhat out of date partly because the AFW systems were subject to some (NUREG-0737)[98] modifications after the analyses were made[946] and partly because the analyses themselves are rather primitive by modern standards.

Safety Significance

This item seeks to upgrade the AFW unavailability analyses to reflect the NUREG-0737[98] modifications and improvements and to ensure that the AFW system reliability is commensurate with the system's safety importance.

Proposed Solution

The proposed solution for this issue is to perform a PRA of all AFW systems and require modification of any systems which have an unacceptably high failure probability.

PRIORITY DETERMINATION

Issue 124, "AFW System Reliability," will consider whether seven PWRs with two-train AFW systems have AFW system unavailabilities that need to be improved. Therefore, this issue need cover only the three-train AFW systems.

To prioritize this issue, several questions need to be answered. First, how reliable must the AFW system be to have reliability commensurate with its safety importance? Generic Issue 124 has selected an unavailability of 10^{-4} failure/demand as the upper limit of acceptability.[947] We will use this same figure. The second question is, how many plants are likely to be found which cannot meet the 10^{-4} failure/demand cutoff? Analyses of ten three-train AFW designs are summarized in an RRAB memorandum[894] as follows:

Design	Failure/Demand	log(failure/demand)
Summer 1	1.2×10^{-5}	-4.92
McGuire	2.0×10^{-5}	-4.70
Comanche Peak	2.0×10^{-5}	-4.70
Diablo Canyon	3.7×10^{-5}	-4.43
San Onofre 2&3	2.2×10^{-5}	-4.66
SNUPPS	2.0×10^{-5}	-4.70
Waterford	1.4×10^{-5}	-4.85
Midland	1.0×10^{-5}	-5.00
Seabrook	2.0×10^{-5}	-4.70
Catawba	0.7×10^{-5}	-5.15

Arithmetic Mean:	1.8×10^{-5}
Arithmetic Standard Deviation:	8.4×10^{-6}
Logarithmic Mean:	-4.78
Logarithmic Standard Deviation:	0.22

These 10 analyses can be considered a statistical sample. The cutoff of 10^{-4} failure/demand is 9.76 standard deviations above the mean on a linear scale and 3.55 standard deviations above the mean on a logarithmic scale. The shape of the distribution is unknown, of course, but we will examine both a normal and a log normal distribution and use the worst case. Based upon these distributions and in the absence of any other information, if another three-train AFW design were evaluated, the probability of this new design being above the cutoff is:

Normal Distribution:	essentially zero
Log Normal Distribution:	2×10^{-4}

What this means is that 10 sample designs are all well below the cutoff. Had the sample average been close to just below 10^{-4}, one would be confident of finding a plant or two over the limit. However, the mean is far below the limit (where "far" is defined in terms of the width of the distribution) and the per-plant probability of being over the limit is small.

There are 80 PWRs operating or under construction. Seven of these have two-train AFW systems and are covered by Issue 124; this leaves 73 plants. The probability of detecting one or more of these plants with an AFW unavailability greater than 10^{-4}/demand is:

$$1 - (1 - 2 \times 10^{-4})^{73} \approx (73)(2 \times 10^{-4}) \cong 0.014$$

That is, based upon the available knowledge regarding three-train AFW designs and in the absence of other information, a PRA of all three-train AFW systems has only a few percent chance of finding a system that needs upgrading. (This does not mean that these AFW systems are problem free. It does mean that the problems probably will not be found by means of PRA, unless considerably more information is available.)

Frequency Estimate

The sequence of interest is straightforward. It is initiated by a non-recoverable loss of main feedwater. If the AFW system fails and feed-and-bleed techniques fail, core-melt will ensue.

For the initiating event frequency (non-recoverable loss of main feedwater), we will use 0.64 event/RY, based upon the Oconee PRA done by Duke Power Co.[947] This figure is based upon fault tree analysis and should be reasonably representative of most main feedwater system designs.

Next, the change in AFW failure probability must be estimated. We will assume that the AFW system "as is" has an unavailability equal to that of a "typical" two-train AFW system which would be about 6.7×10^{-4}/demand, the average of the seven plants.[948] The AFW system failure probability after upgrading would be at most 10^{-4}. Therefore, the change in probability would be about 5.7×10^{-4}.

We will assume a typical value of 0.20 for the failure probability of feed-and-bleed cooling, based upon the calculations presented under Issue 125.II.9, "Enhanced Feed-and-Bleed Capability." Multiplying these figures, the change in core-melt frequency is:

$$(0.64/\text{year})(5.7 \times 10^{-4})(0.20) = 7.3 \times 10^{-5}/\text{year}$$

The number of hypothetical plants needing modification (expectation value) is 0.014. Thus, the change in core-melt frequency for all reactors is 10^{-6}/year.

Consequence Estimate

The core-melt sequence under consideration here involves a core-melt with no large breaks initially in the reactor coolant pressure boundary. The reactor is likely to be at high pressure (until the core melts through the lower vessel head) with a steady discharge of steam and gases through the PORV(s). These are conditions likely to produce significant hydrogen generation and combustion. The Zion and Indian Point PRA studies used a 3% probability of containment failure due to hydrogen burn (the "gamma" failure). We will follow this example and use 3%, bearing in mind that specific containment designs may differ significantly from this figure. In addition, the containment can fail to isolate (the "beta" failure). Here, the Oconee PRA figure of 0.0053 will be used. If the containment does not fail by isolation failure or hydrogen burn, it will be assumed to fail by basemat melt-through (the "epsilon" failure).

Using the usual prioritization assumptions of a central midwest plains meteorology, a uniform population density of 340 persons per square mile, a 50-mile radius, and no ingestion pathways, the consequences are:

Failure Mode	Percent Probability	Release Category	Consequences (man-rem)
gamma	0.3%	PWR-2	4.8×10^6
beta	0.5%	PWR-5	1.0×10^6
epsilon	96.5%	PWR-7	2.3×10^3

The "weighted-average" core-melt will have consequences of 1.5×10^5 man-rem.

Because this issue deals with only an expectation value for the number of plants, but does not necessarily expect to affect any specific plant, the per-plant parameters (core-melt/RY and man-rem/reactor) are not meaningful. Instead, the "aggregate" parameters (core-melt/year and total man-rem) are appropriate.

As of March 1988 (the earliest that any changes are likely to be made), the 73 subject plants will have a combined remaining life of 2317.8 calendar-years. At a 75% capacity factor, this works out to an average of 23.8 years of operation remaining per plant.

Therefore, the change in risk for the hypothetical plant is 11 man-rem/year and the total risk reduction for all reactors is 3.7 man-rem.

Cost Estimate

The costs involved would include administrative charges, the costs of the PRAs, and possibly costs of hardware changes, should they be required. It is not clear at this point whether the PRAs would be done by the licensees or the NRC. In any case, the cost of the PRA of one AFW system is likely to be on the order of $50,000 or more (half a staff-year). For 73 plants, this is $3.65M. We will not calculate the administrative and hardware costs, but instead will use the $3.65M as a minimum figure.

Value/Impact Assessment

Based on an estimated risk reduction of 3.7 man-rem and a minimum cost of $3.65M associated with the possible solution, the value/impact score is given by:

$$S \leq \frac{3.7 \text{ man-rem}}{\$3.65\text{M}}$$

$$\leq 1 \text{ man-rem/\$M}$$

Other Considerations

(1) The statistical logic presented above does not rule out specific systems needing attention. The proper conclusion is that, unless more information is forthcoming (for example, specific design or performance problems), a non-specific general search such as this is difficult to justify because

there is no specific reason to believe a problem will be found this way, based on past experience. Also, the continuous distribution assumption implies that design anomalies, such as the single failures of Item 125.II.1.B, have been fixed. This item must not be viewed in isolation.

(2) Issue 124, "AFW System Reliability," in addition to its attention to plants with two-train AFW systems, also is considering whether to require confirmation that the remaining PWRs have AFW system reliabilities that are less than 10^{-4}/demand. However, Issue 124 has not produced a decision at this time, nor does a decision appear to be forthcoming in the near future. Therefore, this issue cannot be subsumed within Issue 124.

(3) In most cases, the fix will not involve work within radiation fields and thus will not involve ORE.

(4) The ORE averted due to post-feed-and-bleed cleanup and post-core-melt cleanup is a minor consideration. ORE associated with cleanup is estimated to be 1800 man-rem after a primary coolant spill and 20,000 man-rem after a core-melt accident.[64] If the frequency of feed-and-bleed events is 5 x 10^{-6}/year, the actuarial cleanup ORE averted is only 0.2 man-rem. Similarly, a total core-melt frequency of 10^{-6}/year corresponds to an actuarial averted cleanup ORE of only 0.5 man-rem. If averted ORE were added to the man-rem/reactor and man-rem/$M figures above, no conclusions would change.

(5) The proposed fix would reduce core-melt frequency and the frequency of feed-and-bleed events and, therefore, would avert cleanup costs and replacement power costs. The cost of a feed-and-bleed usage is dominated by roughly six months of replacement power while the cleanup is in progress. If the average frequency of such events is 5 x 10^{-6}/year and the average remaining lifetime is 31.7 calendar-years at 75% utilization, then making the usual assumptions of a 5% annual discount rate and a replacement power cost of $300,000/day, the actuarial savings for feed-and-bleed cleanup are $3,300. Similarly, the actuarial savings of averted core-melt cleanup (which is assumed to cost one billion dollars if it happens) are about $12,000. The actuarial savings from replacement power after a core-melt up to the end of the plant life are also about $12,000. (This last figure represents the lost capital investment in the plant.) If these theoretical cost savings were subtracted from the expense of the fix, the man-rem/$M would not change significantly.

CONCLUSION

Based upon the figures above, this issue should be DROPPED from further consideration.

ITEM 125.II.1.D: AFW STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM/ICS INTERACTIONS IN B&W PLANTS

DESCRIPTION

This issue is centered upon the subject of the reliability of the AFW system which is safety-grade. This item is targeted specifically at B&W plants[940] and would require a reexamination of the AFW system reliability.[948] The reasons given are two-fold. First, assessments made shortly after the TMI accident indicated that the AFW system in B&W plants had (at that time) an unavailability approximately an order of magnitude higher than those in most other PWRs.[948] (This does not account for the subsequent modifications to these AFW systems.) Second, this item calls for explicit attention to the interactions between the AFW system and the Steam and Feedwater Rupture Control System (SFRCS) and between the AFW system and the Integrated Control System (ICS). Such interactions are important because the initiating transient may well be caused by a problem with the ICS and any possible interactions between the ICS and AFW or SFRCS would be a potential source of a common mode failure, defeating the system needed to mitigate the transient.

PRIORITY DETERMINATION

On the general question of AFW unavailability, the B&W plants have already updated their reliability analyses to reflect the post-TMI modifications.[946] These updates have satisfied the original concern.[949]

The specific issue of the ICS-SFRCS-AFW interactions deserves more discussion. The function of an SFRCS is to control the AFW system. The name (Steam and Feedwater Rupture Control System) is somewhat misleading in that the SFRCS also initiates AFW for loss of main feedwater events. Those plants with an SFRCS should have no interactions between the ICS and the SFRCS or AFW systems.

There are some B&W plants that have used the ICS to control the AFW system. Of these, two plants (Crystal River and ANO-1) have installed an "Emergency Feedwater Initiation and Control (EFIC) System" to replace the ICS as the control system for AFW. (The EFIC system is an improvement over SFRCS in that the EFIC system will not allow both steam generators to be isolated simultaneously. The SFRCS at Davis-Besse has also been modified such that it will no longer allow both steam generators to be isolated simultaneously.) Of the two remaining plants, Rancho Seco will install an EFIC system at its next refueling outage and TMI-1 will install a system similar to EFIC, but designed by the licensee, at its next refueling outage.

Under these circumstances, the concern is not with SFRCS-AFW interactions, but instead reduces to ensuring that there is no interaction between the ICS and the AFW or its control system that can cause a common mode failure. For plants with two-train AFW systems, this will be covered by the analyses of Issue 124.[947,949] The remaining plants will be examined under the B&W Reassessment Program which places considerable emphasis on the ICS.[950]

CONCLUSION

This item is covered in Issue 124 and the B&W Reassessment Program and should be DROPPED as a separate issue.

ITEM 125.II.2: ADEQUACY OF EXISTING MAINTENANCE REQUIREMENTS FOR SAFETY-RELATED SYSTEMS

DESCRIPTION

Historical Background

The objective of this issue is to assess the adequacy of existing maintenance requirements and their impact on the reliability of safety-related systems. The issue was identified[940] as a long-term generic action following the loss of main and auxiliary feedwater of the Davis-Besse plant on June 9, 1985. The NRC Incident Investigation Team (IIT) concluded that the underlying cause of the Davis-Besse event was the licensee's lack of attention to detail in the care of plant equipment.[886]

Safety Significance

Inadequate and/or improper maintenance of equipment, components, and systems relied on for safe operations of the plants can lead to loss of safety functions. The loss of safety functions of the safety-related systems can increase the severity of transients and lead to severe core damage and possibly a core-melt. Given a core-melt and loss of containment integrity, public radiation exposure would result from the release of fission product materials. The issue is applicable to all operating nuclear power plants.

Possible Solutions

For the Davis-Besse plant, the staff conducted a maintenance survey consistent with the NRC Maintenance and Surveillance Program Plan (MSPP) as a result of the IIT conclusions.[886] As a result of the survey, the staff identified a number of weaknesses impeding the conduct of maintenance activities at the Davis-Besse plant.[1011] A subsequent NRC follow-up survey of the Davis-Besse maintenance activities in March 1986 indicated that the licensee had made considerable progress in all maintenance areas except maintenance backlog since the previous survey. Particular strengths noted were in the areas of maintenance training, spare parts, and material readiness. Based on the results of the March 1986 survey, the NRC concluded that the Davis-Besse new maintenance organization was functioning as planned, and no major identifiable weaknesses were evident. The few remaining problem areas noted by the staff were not considered programmatic weaknesses that would adversely affect the functioning of the maintenance organization.[1011]

In response to Issue 3 of the Commission Policy and Planning Guidance,[210] the staff developed the MSPP that consisted of two phases: Phase I and Phase II. The findings of the Phase I activities are reported in NUREG-1212.[1013] Essentially, the Phase I objectives (which are complete) have addressed the objectives of this issue. In brief, Phase I of the MSPP was designed to survey current maintenance practices in the nuclear utility industry, evaluate their effectiveness, and address the technical and regulatory issues of nuclear power plant maintenance.

Thirty-one measures of maintenance were developed for Phase I of the MSPP. These measures were then organized into the following five categories: (1) overall system/component reliability; (2) overall safety system reliability; (3) challenges to safety systems; (4) radiological exposure; and (5) regulatory assessment. An analysis of the overall trends and patterns across the above five categories of maintenance revealed several important trends. In general, although plant maintenance performance showed some improvement from 1980 to 1985, the safety systems reliability for all plants did not significantly change since 1981. Thus, the contribution of maintenance to reliability problems indicated that some maintenance programs and practices are not effective. The Phase I findings confirmed that there are wide variations in maintenance practices among utilities and the industry has established a variety of programs aimed at self-improvement that do not appear to be well-integrated or effectively implemented in some cases. The resolution of the issues identified in Phase I of the MSPP will be addressed in Phase II of the MSPP.

The Phase II activities of the MSPP are being addressed under Issue HF8. In brief, Phase II of the MSPP requires the staff to: (1) gather data to support a definition of the role of maintenance in safety; (2) develop goals for plant reliability in ensuring effective maintenance; (3) assess data to determine performance-oriented maintenance criteria; (4) make recommendations for endorsement of good maintenance practices; (5) recommend improvements to the maintenance/operations interface; (6) provide input to draft industry standards for maintenance; and (7) assess industry programs in self-improvement of maintenance programs.

CONCLUSION

The maintenance-related problems identified by the NRC IIT for the Davis-Besse plant were resolved.[1011] For all operating plants, the objectives of this issue were essentially completed by Phase I of the existing MSPP. Phase II of the MSPP (Issue HF8) will follow up and address problem issues identified in Phase I of the MSPP that warrant further NRC and industry actions.[1013] Therefore, this issue should be DROPPED as a separate issue.

ITEM 125.II.3: REVIEW STEAM/FEEDLINE BREAK MITIGATION SYSTEMS FOR SINGLE FAILURE

DESCRIPTION

Historical Background

During the investigation of the Davis-Besse event, the importance of the SFRCS became evident. Although the name of this system implies that its purpose is to mitigate steam and feedwater line breaks, in actual practice this is the AFW control system. Thus, the functions of this control system are more general than the name implies.

Safety Significance

Steam/feed line break mitigation systems vary in title and in detailed design from plant to plant and from vendor to vendor. However, they are generally composed of two logic trains in order to meet the single failure criterion. The presence of an unsuspected single failure would have the potential to greatly increase the probability of system failure. This has safety significance for several accident scenarios.

First, the reliability of mitigation of a steam or feedwater line break would be adversely affected. During such an event, the mitigation system isolates both the steam line and the feedwater (main and auxiliary) lines associated with the depressurizing steam generator. For most breaks outside containment, this stops the blowdown. For a break inside containment, the secondary side of the affected steam generator will blow down to the containment atmosphere, but isolation of feedwater to the affected steam generator will prevent continued long-term steaming due to decay heat from the reactor core. This is necessary to ensure that the containment design pressure is not exceeded.

This scenario is also the concern of Issue 125.II.7, "Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break." The safety concern expressed here is not a duplication of Issue 125.II.7; rather, Issue 125.II.7 questions the necessity of having this automatic isolation provision and thus is opposite in its thrust. Nevertheless, a detailed examination of the significance of this scenario is presented in the prioritization of Issue 125.II.7 and will not be treated further here.

The second scenario is the loss of feedwater transient. If main feedwater is lost and not readily recoverable and a single failure in the AFW control system defeats AFW, most plants will have to use feed-and-bleed core cooling techniques to prevent core-melt. Because the viability of feed-and-bleed cooling is often questionable, and because non-recoverable loss of main feedwater events have in fact occurred many times, the reliability of the AFW system and its control system is of considerable importance. This is exactly the safety concern of Issue 125.II.1.b, "Review Existing AFW Systems for Single Failure." Thus, this safety concern is a duplicate of Issue 125.II.1.b.

The third scenario is specific to B&W plants. These plants provide AFW to the steam generators by means of a special AFW sparger. This sparger is located high in the steam generator and sprays water onto the steam generator tubes. The advantage of this arrangement is that it enhances natural convection through the primary system when forced circulation is lost. If a loss of forced circulation (i.e. trip of all four reactor coolant pumps) transient were to occur and AFW were to fail, natural circulation might not provide sufficient core cooling to prevent cladding failure, even if some feedwater were being supplied to the secondary side of the steam generators. This is somewhat different from the safety concern of Issue 125.II.1.b which is concerned with AFW reliability during loss of feedwater transients. Nevertheless, any upgrades brought about by the resolution of Issue 125.II.1.b should address the loss of forced circulation concern as well. Therefore, this concern is also covered by Issue 125.II.1.b.

CONCLUSION

This issue has three aspects: (1) line break mitigation, which is covered in Issue 125.II.7; (2) loss of feedwater, which is covered in Issue 125.II.1.b; and (3) loss of forced circulation, which is also covered in Issue 125.11.1.b. Therefore, this item should be DROPPED as a new and separate issue.

ITEM 125.II.4: THERMAL STRESS OF OTSG COMPONENTS

DESCRIPTION

Historical Background

This issue addresses the effects of thermal stresses induced on the OTSG from a loss of feedwater transient and was based on RES concerns.[941,942]

Safety Significance

The safety concern raised was that the introduction of the recovered feedwater to the dry OTSG, following the Davis-Besse transient, may have degraded the structural integrity of the OTSG and the steam generator tubes. The resulting transient-induced thermal stresses might lead to increased rupture frequencies for the steam generator components which, in turn, would increase the plant's core-melt frequency and the potential radiological risks to the public.

PRIORITY DETERMINATION

Following the Davis-Besse transient, the staff reviewed[943] the B&W analysis regarding the possible effects of the transient to the structural integrity of the Davis-Besse OTSG. Comparisons were made between the Davis-Besse event and the B&W design basis analyses. Therefore, the conclusions reached herein are considered applicable to similar transients of similar OTSGs (B&W) plants. This issue is not applicable to CE or W PWR plants that have U-Tube heat exchanger designs and AFW injection that does not spray directly on the steam generator tubes.

The following components were considered to be the most highly stressed during transients involving boiled-dry OTSGs and subsequent recovery of auxiliary and main feedwater: (1) AFW Nozzle, (2) Main Feedwater Nozzle, (3) AFW Jet Impingement on Steam Generator Tubes, (4) Stresses on Steam Generator Tubes Due to Steam Generator Shell/Tube Thermal Stress, (5) Degraded Steam Generator Tubes, and (6) Thermal Shock of Lower Tube Sheet.

AFW Nozzle: The stress and fatigue analyses of the AFW nozzle resulting from the Davis-Besse transient were compared to the original design basis temperature difference of 530°F between the hot steam generator shell and the AFW injection temperature. During the transient, the temperature difference was 501°F which is within the design basis analyses. The fatigue usage factor that was predicated on 875 AFW initiations, was also considered acceptable.[943]

Similar design basis analyses are conducted for all B&W OTSG designs except that the numbers of transients and nozzle designs are plant-specific.[945] Therefore, the thermal stresses and fatigue component resulting from similar events are bounded by the original B&W design basis analyses.

Main Feedwater Nozzle: The original design basis stress analysis for the Davis-Besse OTSG was based on a temperature difference of 445°F between the main feedwater nozzle and the feedwater. During the Davis-Besse transient, the temperature difference was approximately 162°F.[943] Therefore, the thermal stresses and fatigue factor resulting from the transient were considered bounded by the original B&W design basis. Similar design analyses are conducted for all B&W OTSG designs with the same exceptions as noted for the AFW nozzles.[945]

AFW Jet Impingement on Steam Generator Tubes: The original design basis assumed a temperature difference of 586°F between the AFW coolant and the steam generator tube surfaces. Based on thermocouple data, the temperature difference between the steam generator tubes and the AFW was determined to be approximately 523°F.[943] Therefore, the thermal stresses and the fatigue factor (based on 29,400 cycles in the original Davis-Besse OTSG design basis) resulting from the transient were considered bounded by the original B&W design basis. Similar analyses (with the exception of the number of transients) have been conducted for all B&W OTSGs.[945]

Steam Generator Shell/Tube Thermal Stress: Temperature differences between both steam generator shells and their tubes and the pressure differences across the tube sheets were analyzed based on thermocouple readings. The maximum temperature difference in one of the two steam generators was estimated to be approximately 72°F. The resulting stresses and fatigue component were determined to be acceptable by the staff.[943]

Degraded Steam Generator Tubes: In NUREG-0565,[96] the staff discussed its evaluation of B&W's analyses of potential defective steam generator tubes with up to 70% through-wall defects. The B&W thermal stress conditions included ten transients with maximum flaw orientations following a SBLOCA. The secondary side was postulated to have boiled dry and the primary system was significantly voided. The cold AFW impinging on the steam generator tubes and the pressure loads resulting from the tube-to-shell temperature differences, in combination with the potential effects of slug flow in the steam generator tubes from the voiding primary system, was evaluated. The staff concluded that the combination of conservative analyses and the test results provided assurance that structural integrity of the primary coolant pressure boundary (steam generator tubes) would be maintained.

Thermal Shock of Lower Tube Sheet: The stress and fatigue analyses relative to thermal shock of the lower tube sheet from the Davis-Besse transient were reviewed by the staff. The stresses and fatigue usage factor resulting from the transient were determined to be negligible. Therefore, it was concluded that the tube sheet was essentially unaffected by the Davis-Besse transient.[943]

CONCLUSION

The staff has raised concerns relative to potential beyond design basis conditions that may increase the primary system temperatures above those previously analyzed. The higher superheat temperatures will lower the steam generator tube strength or, in combination with injected cold AFW temperature, might increase the thermal stresses. These conditions might then further degrade or fail the primary pressure boundary. This potential phenomenon is being studied by the staff.[944]

The staff concluded that transients similar to the Davis-Besse transient are bounded by the original B&W design basis analyses. [illegible]fore, the B&W OT[illegible] design basis adequately accounts for such anticipa[illegible]ional occurrences. Based on the staff findings, this issue involves [illegible] risk to the public and should be DROPPED from further consid[illegible]

The potential superheat phenomena being studied by the staff is beyond the current design basis. Should the results of the superheat studies indicate a need for changes in the design basis of the primary and secondary pressure boundaries, it is recommended that any follow-up effort be prioritized as a new and separate issue.

ITEM 125.II.5: THERMAL-HYDRAULIC EFFECTS OF LOSS AND RESTORATION OF FEEDWATER ON PRIMARY SYSTEM COMPONENTS

DESCRIPTION

Historical Background

The Davis-Besse plant recovered feedwater flow following the loss of feedwater transient on June 9, 1985. With the loss of feedwater to the steam generators, heatup of the reactor coolant system peaked at about 592°F and then, following recovery of the feedwater, decreased to 540°F in approximately six minutes (normal post-trip average temperature is 550°F). Thus, the reactor coolant system experienced an overcooling transient rate of 520°F/hr for the 6-minute time interval.

Due to concerns identified,[941,942] the staff was requested[940] to review and evaluate the safety significance of the thermal-hydraulic effects (potential pressurized thermal shock) to reactor pressure vessels, nozzles, and downcomer surface areas from such overcooling transients.

Safety Significance

The potential for pressurized thermal shock (PTS) to the reactor pressure vessel (RPV) and components from overcooling transients is more critical to PWRs by virtue of their designs. Therefore, this issue is applicable to all PWRs. With increased neutron radiation exposure, the temperature at which the RPV materials fracture toughness decreases to unacceptable limits increases. Thus, with time (neutron radiation exposure), the magnitude of the thermal stresses which are also compounded by pressure-induced stresses during overcooling transients, could approach reduced fracture toughness capabilities of the RPV materials.

Structural failure (fracture) of the RPV, to an extent that would make the RPV unable to contain sufficient water to cover the reactor core, would result in a core-melt. Given a core-melt and subsequent loss of containment integrity, public radiation exposure would result from the release of fission product materials.

Possible Solutions

For the Davis-Besse plant, the staff reviewed and evaluated the licensee's PTS calculations and results related to the June 9, 1985 event. Based on the staff's findings,[1011] the temperature of the limiting weld in the Davis-Besse RPV would have had to drop an additional 377°F to cause crack-initiation to become a significant PTS event.

To ensure that nuclear power plants do not operate with unacceptable PTS risks, the NRC promulgated a final rule[1012] in July 1985 that amended its regulations to: (1) establish a screening criterion related to the fracture-resistance of PWR vessels; (2) require analyses and a schedule for implementation of neutron flux reduction programs to avoid exceeding the screening criterion; and (3) require detailed safety evaluations to be performed before plants commence operations beyond the screening criterion. The final PTS rule was a result of extensive analyses performed by the NRC staff (USI A-49, "Pressurized Thermal Shock") and several industry groups. The analyses covered all conceivable PTS events, including RPV overcooling transients, that were more severe than the Davis-Besse event.

CONCLUSION

The PTS concern from the Davis-Besse event was resolved in NUREG-1177.[1011] All other conceivable PTS concerns were addressed in the resolution of USI A-49 and the final PTS rule.[1012] Therefore, this issue should be DROPPED as a separate issue.

ITEMS 125.II.6: REEXAMINE PRA ESTIMATES OF CORE DAMAGE RISK FROM LOSS OF ALL FEEDWATER

DESCRIPTION

This issue is one of a list of longer-term generic issues which arose during the investigation of the loss of all feedwater event which took place at the Davis-Besse plant on June 9, 1985.[940] The memorandum which initiated this action recommends that plant-specific reliability data be solicited from Toledo Edison Company (the licensee for Davis-Besse).[1004] This information would then be used by the NRC staff to formulate a new and revised model for estimating the frequency of severe accidents involving loss of main feedwater at the Davis-Besse plant. The purpose of this effort was to provide information, in addition to the results of deterministic reviews, to aid in decision-making concerning the restart of the Davis-Besse plant.

PRIORITY DETERMINATION

This task is a legitimate action on the Davis-Besse unit, but is not intended to address other plants since they are not in need of a restart decision. Therefore, the issue is not generic but is specific to one unit. However, before dismissing the issue, its generic potential should be explored: What benefits would be reaped if other plants were investigated and modeled with plant-specific data? Evaluations of plants with two-train AFW systems are being made in the resolution of Issue 124, "AFW System Reliability," and investigations along this line for all plants are also being considered. In addition, Issue 125.II.1.b, "Review Existing AFW Systems for Single Failure," deals with gathering of plant-specific information and Issue 125.II.1.c, "NUREG-0737 Reliability Improvements," deals with specific AFW system reliabilities. Finally, USI A-45, "Shutdown Decay Heat Removal Requirements," deals with the question of plant safety for events (such as loss of all feedwater) where the plant's heat sink is lost. In view of the existence of all these issues, there is little to be gained by generalizing this new proposed action to form an additional generic task.

CONCLUSION

Based upon the above, this issue should be placed in the DROP category.

ITEM 125.II.7: REEVALUATE PROVISION TO AUTOMATICALLY ISOLATE FEEDWATER FROM STEAM GENERATOR DURING A LINE BREAK

DESCRIPTION

Historical Background

During the course of the investigation of the event, it was pointed out that the benefits of AFW isolation are probably more than outweighed by the negative aspects of this feature.[940,951]

Safety Significance

The automatic isolation of AFW from a steam generator is provided to mitigate the consequences of a steam or feedwater line break. The isolation logic, usually triggered by a low steam generator pressure signal, closes all main steam isolation valves and also isolates AFW from the depressurizing steam generator. (The AFW flow is diverted to an intact steam generator.) The purposes of the AFW isolation are three-fold:

(1) The break blowdown is minimized. Shutting off AFW will not prevent the initial secondary side inventory from blowing down. However, the isolation will prevent continued steaming out of the break as decay heat continues to produce thermal energy.

(2) Overcooling of the primary system is reduced. As the depressurizing steam generator blows down to atmospheric pressure, the primary system is cooled down, causing primary coolant shrinkage and (if the event occurs near the end of the fuel cycle) a return to criticality, which adds a modest amount of thermal energy to the transient. Shutting off feedwater to the faulted steam generator will reduce this effect, although once again the initial blowdown will be the dominant factor.

    The significance of these first two considerations is in containment pressure. The containment is designed to accommodate a primary system blowdown followed by decay heat boiloff (the large break LOCA). A steam or feedwater line break within containment might cause the containment design pressure to be exceeded if the AFW isolation were not present.

(3) The AFW isolation is needed to divert AFW flow to the intact steam generator(s). For the case of a two-loop plant with a two-train AFW system, this is needed to meet the single failure criterion in supplying feedwater to the intact steam generator. (The situation becomes more complex for other cases, e.g. a four-loop plant with a three-train AFW system.) Note that, unless the line break is in the AFW line, core cooling would still meet the single failure criterion even without the isolation, since the faulted steam generator would still be capable of heat transfer.

In summary, the automatic isolation is needed only to help mitigate a relatively rare event (steam or feedwater line break) and even then is only remotely connected with sequences leading to core-melt.

In contrast, this isolation has definite disadvantages. If both channels of the controlling system were to spontaneously actuate during normal operation, all AFW would be lost and the MSIVs would close. Most newer plants use turbine-driven main feedwater pumps. Thus, main feedwater would be lost also. If the plant operators fail to correctly diagnose and correct the problem, only feed-and-bleed cooling would be available to prevent core-melt. Similarly, if spurious AFW isolation were to occur during the course of another transient, once again only feed-and-bleed cooling would be available to prevent core-melt.

The long-term success of AFW for main feedwater transients, steam generator tube ruptures, and small LOCAs may also be compromised.[951] During controlled cooldown, the thresholds for automatic AFW isolation are crossed. Procedures call for operators to lock out the isolation logic as the steam generator pressure approaches the isolation setpoint. Under the circumstances, the accompanying distractions make it possible that the operators will forget to override the AFW isolation logic in the permissive window. Thus, AFW reliability in these scenarios may be significantly degraded.

The safety significance of this issue arises from the fact that the negative aspects involve accident sequences which have more frequent initiators, and more significant consequences, than those of the positive aspects.

Possible Solutions

A very straightforward solution has been proposed: simply disconnect the AFW isolation valve actuators from the automatic logic and depend on plant procedures, i.e., have the operators close the AFW isolation valves (by remote manual operation from the control room) in the event of a line break.[951] These procedures would require careful verification of the existence of a line break before isolating a steam generator from AFW.

PRIORITY DETERMINATION

Frequency Estimate

It is necessary to calculate estimates of both the positive and negative aspects of disabling the automatic AFW isolation. The positive aspects are due to a decrease in the frequency of loss of all feedwater events. There are three accident sequences of interest.

(1) The first sequence is initiated by a spontaneous actuation of both channels of the isolation logic. (We will assume a two-loop plant design for prioritization purposes.) There is no data readily available for such actuations. However, it is possible to make an educated guess. EPRI NP-2230[307] provides some perspective, based upon actual experience with other systems:

Inadvertent Safety Injection Signal, PWR	0.06/RY
MSIV Closure, PWR	0.03/RY
Steam Relief Valve Open, PWR	0.04/RY
Inadvertent Startup of BWR HPCI	0.01/RY

Based upon these figures, it is expected that spontaneous actuations will occur with a frequency on the order of 0.03/RY. Of course, this would isolate only one steam generator. However, such systems generally have a common mode failure probability on the order of 5%. (In addition, the second train of AFW has an unavailability due to other causes of roughly 1%. However, the main feedwater system would still be available in this case.) Thus, the frequency of both steam generators isolating is (0.03/RY) (0.05), or 1.5×10^{-3}/RY. Of course, the plant operators are likely to reset the logic and turn the transient around. We will assume a 1% (minimum) failure probability for recovery by operator action. This leaves feed-and-bleed cooling for which we will assign a typical failure probability value of 0.20 and a maximum failure probability of 0.60, based on the calculations presented under Item 125.II.9, "Enhanced Feed-and-Bleed Capability." Multiplying these figures gives a core-melt frequency of 3×10^{-6}/RY typical, 9×10^{-6}/RY maximum.

(2) The second sequence is initiated by another, independent transient. During the course of this transient, and the consequent perturbation of a great many plant systems, the AFW isolation logic is triggered. The MSIVs close, causing a loss of main feedwater (if main feedwater has not previously been lost), and the AFW isolates. Again, unless the AFW isolation valves are reopened, only feed-and-bleed is available as a means of core cooling.

The AFW isolation logic can be triggered during a transient in two ways. The first is by some type of inadvertent systems interaction, e.g. electromagnetic coupling. The proper fix for this problem is to eliminate the systems interaction which may well have other consequences in addition to AFW isolation. Therefore, this effect will not be considered here.

The second way to trigger AFW isolation is by the actual existence of low pressure in the secondary system, caused by the initiating transient. In this case, the isolation is working as designed (but not as intended). Low pressure transients are relatively rare, since the steam space in question is usually right on top of a significant quantity of water at saturation temperature. Low pressure will occur only if steam is vented at a rapid rate in sufficient quantity to cool the water inventory via boiloff to the point where saturation pressure drops below the AFW isolation setpoint. The other possibility is a dryout of the steam generator.

This is possible for B&W plants because of the relatively low water inventory in the steam generators. However, such an event in a Westinghouse or CE plant would probably imply that the main feedwater and AFW had already failed.

There is no readily available way of estimating the probability of a pressure drop, given a transient. However, EPRI NP-2230[307] gives a frequency of 0.04/RY for events where PWR steam relief valves open. Thus, we can assume that depressurization events occur with at least this frequency. If we further assume that perhaps 10% of these pressure drops are deep enough to trigger AFW isolation, and again assume a 1% probability of failure of the operators to recover AFW, the resulting core-melt frequencies are 8×10^{-6}/RY typical, 2.4×10^{-5}/RY maximum.

(3) The third sequence involves the long term success of AFW for main feedwater transients. During controlled cooldown, the thresholds for automatic AFW isolation are crossed. Procedures call for the operators to lock out the isolation logic as the steam generator pressure approaches the setpoint. If the operators fail to do so, both trains of AFW will isolate. Main feedwater is also unavailable, since its loss initiated the transient. Again, only feed-and-bleed would be available for core cooling.

Non-recoverable loss of main feedwater events are estimated to occur with a frequency of 0.64/RY.[952] We will assume a 1% minimum probability of operator failure to bypass the isolation logic and another 1% minimum probability of failure of the operators to recover the AFW system. In addition, there is still feed-and-bleed cooling which, because the plant is already partially cooled down, should have a better than usual chance of succeeding. We will therefore assume 10% instead of 20% or 60% for feed-and-bleed failure probability. The result is a core-melt frequency of 6.4×10^{-6}/RY.

The three sequences above add up to a "typical" core-melt frequency of 1.7×10^{-5}/RY and as much as 3.9×10^{-5}/RY for a plant with marginal feed-and-bleed capability. Now we must estimate the negative aspects of the proposed fix.

The first negative scenario is the feedwater line break. Here, a break in the feedwater line to one steam generator initiates the sequence. With the proposed fix, the line is not isolated and one train of AFW simply pumps water out of the break. If the operator fails to manually isolate the break, the remaining AFW train fails, and feed-and-bleed techniques fail, core-melt will result.

Steam and feedwater line breaks are estimated to occur at a combined rate of 10^{-3}/RY (see Issue A-22). Because steam lines are larger and not as subject to water hammer phenomena, the feedwater lines are expected to be more likely to break than the steam lines. We will therefore assume that feedwater lines will break with a frequency of 9×10^{-4}/RY, i.e. 90% of the total line break frequency.

The unaffected single train of AFW should have a failure probability on the order of 0.01 or less. Consistent with the positive scenario calculations, we will assume a 1% probability of operator failure to manually isolate the affected steam generator and a 20% typical, 60% maximum feed-and-bleed failure probability. The product is a core-melt frequency of 1.8×10^{-8}/RY typical and 5.4×10^{-8}/RY maximum.

The remaining scenario is a steam line break. This scenario may involve the theoretical possibility of containment failure by overpressure, but does not lead to core-melt. We will assume a 10^{-3}/RY frequency of line break as before and a 10% probability that the line break is in the steam lines as opposed to the feedwater line breaks of the previous scenario. Once again, the probability of the operator to fail to manually isolate is assumed to be 1%. The frequency of higher than expected containment pressure due to long term steaming in the faulted steam generator is then 10^{-6}/RY.

The change in core-melt frequency is the algebraic sum of the various scenarios:

	Core-melt Averted/RY	
	Typical	Maximum
Spontaneous Actuation	3.0×10^{-6}	9.0×10^{-6}
Transient Initiated	8.0×10^{-6}	2.4×10^{-5}
Cooldown Initiated	6.4×10^{-6}	6.4×10^{-6}
Feedwater Line Break	-1.8×10^{-8}	-5.4×10^{-8}
Net change in core-melt frequency	1.7×10^{-5}	3.9×10^{-5}

The estimated reduction in core-melt frequency for all reactors is 3.5×10^{-4}/year.

Consequence Estimate

The core-melt sequences under consideration here involve a core-melt with no large breaks initially in the reactor coolant pressure boundary. The reactor is likely to be at high pressure (until the core melts through the lower vessel head) with a steady discharge of steam and gases through the PORV(s). These are conditions likely to produce significant hydrogen generation and combustion.

The Zion and Indian Point PRA studies used a 3% probability of containment failure due to hydrogen burn (the "gamma" failure). We will follow this example and use 3%, bearing in mind that specific containment designs may differ significantly from this figure. In addition, the containment can fail to isolate (the "beta" failure). Here, the Oconee PRA figure of 0.0053 will be used. If the containment does not fail by isolation failure or hydrogen burn, it will be assumed to fail by basemat melt-through (the "epsilon" failure).

Using the usual prioritization assumptions of a central midwest plains meteorology, a uniform population density of 340 persons per square mile, a 50-mile radius, and no ingestion pathways, the consequences are:

Failure Mode	Percent Probability	Release Category	Consequences (man-rem)
gamma	3.0%	PWR-2	4.8×10^{6}
beta	0.5%	PWR-5	1.0×10^{6}
epsilon	96.5%	PWR-7	2.3×10^{3}

The "weighted-average" core-melt will have consequences of 1.5×10^{5} man-rem/event.

These figures should cover all PWRs with large dry containments. They do not apply to ice condenser containments. Because of the low free volume in such a containment, failures due to overpressure are more likely and the averaged consequences may be significantly greater. However, we are not aware of any ice condenser plant which has an automatic AFW isolation affected by this issue.

The steam-line-break/containment-rupture scenario is different. The containment pressure is unlikely to exceed the design pressure by more than a few percent, if at all. In most cases, the containment is calculated to fail at 2 to 2.5 times its design pressure. Therefore, containment failure by overpressure is at most a very remote theoretical possibility. We will assume that the overpressure failure probability cannot be greater than 3%, the hydrogen burn figure (a highly conservative assumption). The only radioactive release comes from the containment atmosphere and any primary coolant leakage or discharge from the PORV(s). We have no consequence estimates for such an event. However, the consequences can be conservatively bounded by those of a PWR-8 event, which is a successfully mitigated LOCA with failure of the containment to isolate. The PWR-8 consequences are 7.5×10^4 man-rem. Thus, the steam line break event will have "average" consequences of at most $(0.03)(7.5 \times 10^4)$ or 2250 man-rem, and probably much less.

It is not known how many plants are affected by this issue. In many plants, the AFW isolation logic has provisions to prevent isolation of feedwater to more than one steam generator. Others may not even have this isolation logic. We will assume that about 25% of the PWRs will be affected by this issue. There are 83 PWRs and, as of spring 1987 (the earliest that this issue is likely to result in changes), the remaining collective calendar life will be 2571 RY. At a 75% utilization factor, this is 1928 RY or about 23 operational years per reactor.

The net change in man-rem/RY is obtained by multiplying the change in core-melt frequency by 1.5×10^5 man-rem (average) per core-melt. Then, the steam line break scenario must be subtracted. The consequences of the steam line break scenario (upper bound) are simply (10^{-6} overpressure/RY) [2250 (average) man-rem/overpressure], or 2.3×10^{-3} man-rem/RY.

	Change in man-rem/RY	
	Typical	Maximum
Core-melt Scenarios	2.6	5.9
Steam Line Break	≤0.0023	≤0.0023
Net change:	2.6	5.9

The estimated risk reduction is 140 man-rem/reactor (maximum) and 1,300 man-rem for all reactors.

Cost Estimate

The proposed fix for this issue is simply to remove some leads from some equipment, an action which is likely to be more than paid for by decreased maintenance and testing. Nevertheless, even a relaxation of requirements as this will require review of each affected plant's isolation logic, to be certain that the net effect is an increase in plant safety. In addition, technical specification and procedural changes, with their associated paperwork, will be necessary. We will assume per plant costs of $32,000 to the industry and $25,000 to the NRC, which are typical for a complicated and controversial technical specification change. Thus, the estimated total cost associated with the resolution of this issue is (0.25)(83)($0.057M) or $1.18M.

Value/Impact Assessment

Based on an estimated risk reduction of 1,300 man-rem and a cost of \$1.18M, the value/impact score is given by:

$$S = \frac{1300 \text{ man-rem}}{\$1.18\text{M}}$$

$$= 1102 \text{ man-rem/\$M}$$

Other Considerations

(1) It should be noted that the maximum values are based upon a plant with marginal feed-and-bleed capability. The subset of PWRs which are affected by this issue may not include such a plant. Thus, the "maximum" plant may not exist.

(2) The proposed fix does not involve work within radiation fields and thus does not involve ORE. However, the ORE averted due to post feed-and-bleed cleanup and post-core-melt cleanup is a consideration. NUREG/CR-2800[64] estimates the ORE associated with cleanup to be about 1800 man-rem after a primary coolant spill and about 20,000 man-rem after a core-melt accident. The "typical" frequency of feed-and-bleed events is simply the "typical" core-melt frequency (1.8×10^{-5}/RY) divided by the feed-and-bleed failure probability (0.20). The actuarial figures are:

Averted Feed-and-Bleed Cleanup ORE/plant	3.6 man-rem
Averted Core-melt Cleanup ORE/plant	7.9 man-rem
Total:	11.5 man-rem

The total averted ORE for all plants is 240 man-rem. Thus, the averted ORE is not dominant, but is still a significant fraction of the averted public risk.

(3) The proposed fix reduces core-melt frequency and the frequency of feed-and-bleed events and therefore averts cleanup costs and replacement power costs. The cost of a feed-and-bleed usage is dominated by roughly six months of replacement power while the cleanup is in progress. If the average frequency of such events is $1.7 \times 10^{-5}/0.20$ or 8.5×10^{-5}/RY and the average remaining lifetime is 23 operational years at 75% utilization, and making the usual assumptions of a 5% annual discount rate and a replacement power cost of \$300,000/day, the actuarial savings for feed-and-bleed cleanup works out to be \$55,000. Similarly, the actuarial savings of averted core-melt cleanup (which is assumed to cost \$1 billion if it happens) are about \$200,000. The actuarial savings from replacement power after a core-melt up to the end of the plant life are about \$260,000. (This last figure represents the lost capital investment in the plant.) Obviously, these savings would more than offset the cost of the fix if they were included.

(4) The analysis of the first negative scenario, the feedwater line break, assumed that non-isolation of the ruptured line would cause one AFW train to fail. A special situation can arise for plants with a limited AFW

water supply (e.g. saltwater plants). In such a case, the continued loss of clean water out of the feedwater line break can in theory cause failure of the second AFW train by exhausting the water supply, provided that the loss is not terminated either by the operator or by protective trips (for runout protection) on the first AFW train. In such a case, the scenario's negative contribution (typical) to the averted core-melt frequency of the proposed fix rises from (-1.8×10^{-8}) to (-1.8×10^{-6}). The net change in core-melt frequency would then drop from 1.7×10^{-5} to 1.6×10^{-5}, which would not change the conclusion.

CONCLUSION

Based upon the figures above, particularly the core-melt frequencies, this issue should be placed in the HIGH priority category.

ITEM 125.II.8: REASSESS CRITERIA FOR FEED-AND BLEED INITIATION

DESCRIPTION

Historical Background

This issue is one of a number of longer-term generic actions which arose from the loss of all feedwater event at the Davis-Besse plant on June 9, 1985.[940] During the course of the investigation of this event, it was discovered that the Davis-Besse emergency procedures (EOPs) criteria for initiation of feed-and-bleed cooling were inadequate. The procedures directed the plant operators to initiate feed-and-bleed either if steam generator levels were below 8 inches on the startup range or if the steam generator secondary pressures were less than 960 psig and decreasing. The difficulties with these criteria were: (1) the control room instrumentation was inadequate for the operators to determine that levels were below 8 inches, and (2) there is calculational evidence that steam generator secondary pressures are unlikely to fall below 960 psig before the opportunity for successful feed-and-bleed cooling is past.[1002]

Licensees have been supplied with feed-and-bleed procedures by NSSS vendors.

Safety Significance

Feed-and-bleed capabilities are not currently required by the NRC although the techniques, benefits, and costs are being evaluated in the resolution of USI A-45. Basically, feed-and-bleed cooling is a method of last resort which can avert core damage if main and auxiliary feedwater are lost and other methods of decay heat removal are unavailable. PRAs give considerable credit for feed-and-bleed cooling. A failure rate of one or two percent is a typical assumption. However, the Davis-Besse event chronology leaves an impression that this failure probability may be overly optimistic.

Possible Solution

The Davis-Besse EOPs have been changed; there is now a single criterion for initiating feed-and-bleed which states that feed-and-bleed will be initiated if the primary coolant hot leg temperature rises above 610°F. This parameter is much easier to monitor with existing control room instrumentation and therefore the new criterion is much clearer and unambiguous. The purpose of this proposed generic action is to confirm that all of the remaining B&W plants are using the new criterion rather than the two old criteria.[1002]

CONCLUSION

The safety concern and possible solution of this issue are covered in Issue 122.2, "Initiating Feed-and-Bleed." Issue 122.2 is one of the short-term Davis-Besse issues and is somewhat more general in that it is also concerned with the reluctance of the operators to initiate feed-and-bleed (because of the economic consequences) in addition to being concerned with inadequacy of the criteria. (See References 885, 887, and 940). The two are related; less ambiguity in the written procedures implies less opportunity for reluctance to affect operator actions. Thus, this issue should be DROPPED as a new and separate issue.

ITEM 125.II.9: ENHANCED FEED-AND-BLEED CAPABILITY

DESCRIPTION

Historical Background

This particular issue arose because of the very limited capability of the Davis-Besse plant to remove decay heat using feed-and-bleed techniques.[940] The Davis-Besse plant had a relatively low capacity PORV on the pressurizer and thus limited "bleed" capability. In addition, the HPI pumps (a part of the ECCS) did not develop sufficient discharge pressure to provide injection at operating pressure. To supply coolant at elevated pressure, the plant operators would have to "piggyback" the makeup pumps on the HPI discharge, a complex procedure which will supply only rather limited flow. Thus, the "feed" capability was also limited. The issue is divided into two parts: Part A deals with pressure relief capacity (i.e., enhanced "bleed" capability), and Part B deals with makeup capacity and pressure (i.e., enhanced "feed" capability).

Safety Significance

Feed-and-bleed cooling is normally considered a method of last resort which can avert core damage if main and auxiliary feedwater are lost and not recovered. Nevertheless, main and auxiliary feedwater did both fail (but were recovered) at Davis-Besse and so this need for feed-and-bleed, although remote, is a possibility.

Feed-and-bleed cooling has the advantage of being a redundant and diverse method of core cooling. Its disadvantage (in addition to the economic consequences of releasing primary coolant to the containment) is that the plants were not designed for this mode of core cooling and thus their capabilities are uncertain.

An upgrading of the feed-and-bleed capability would benefit the viability of feed and bleed cooling in several ways: (1) the probability of failure due to component failure would be reduced. (Feed-and-bleed cooling can fail due to a single failure at most plants); (2) the thermal hydraulic uncertainty would be reduced. (Feed-and-bleed cooling is often only marginally viable. A slight change in the thermal hydraulic initial or dynamic conditions may well prevent adequate core cooling): (3) the "window" or time interval during which feed-and-bleed is viable would be lengthened, giving more time to (and less stress upon) the operating crew; and (4) the procedures for initiating feed-and-bleed would be simpler, thus reducing the probability of operator error.

Possible Solutions

The possible solutions for this issue are implicit in the definitions of the two parts: (1) increased pressure relief capacity and (2) increased makeup capacity and pressure. Increased relief capacity could be accomplished by installing larger PORVs, installing more PORVs, or installing a special valve intended for bleed operations. Increased makeup capacity would involve upgrading or replacing the pumps (and their motors) with ones of higher discharge pressure.

PRIORITY DETERMINATION

Frequency Estimate

To estimate changes in core-melt frequency due to the upgrades in pressure relief and makeup capacities, it is first necessary to calculate the change in failure probability of feed-and-bleed cooling. In the past, the usual assumptions have been either that the feed-and-bleed failure probability was dominated by the human failure mode (in NRC-generated PRAs) or that it was governed only by a few hardware failure probabilities (in industry-generated PRAs). Obviously, there is an inconsistency. Moreover, the issue to be addressed here affects both hardware and human failure rates. It is necessary to introduce a (somewhat) more sophisticated treatment of the problem. To do this, we will define four classes of plants.

Class 1: In this class, the plant's HPI pumps develop sufficient discharge pressure to lift the pressurizer safety valves. For such plants, feed-and-bleed cooling does not need the PORVs. Moreover, the HPI pumps are capable of raising the coolant level at any time right up to the point of core uncovery. There is no time interval "window" phenomenon.

Class 2: In this class, the plant's HPI pumps and/or charging pumps can force sufficient coolant in at operating pressure, but cannot lift the safety valves. Here, both PORVs must open for feed-and-bleed cooling to work. In addition, the viability of feed-and-bleed techniques is limited in time. Once the steam generators dry out, primary system pressure rises as the primary coolant heats up and expands. The PORVs will open and help keep pressure down, but eventually the pressure will rise up to the safety valve setpoint, by which time the HPI can no longer force coolant into the primary system. Thus, there is a definite "window" of time, pressure, and temperature during which feed-and-bleed cooling will work.

Class 3: In this class, the HPI pumps and/or charging pumps cannot force sufficient coolant into the primary system at operating pressure. Such plants must open the PORVs and reduce pressure to below normal in order to force sufficient coolant in. Of course, the timing is still more critical for such plants. Once the steam generators dry out, the PORV capacity will soon be overcome by primary coolant expansion and heating.

Class 4: This class is similar to Class 3 except that the PORV or PORVs are small. Such plants cannot sufficiently depressurize using PORVs after the steam generators dry out, but instead must open the PORVs and depressurize while the steam generators are still removing decay heat. In some cases, calculations have shown that the PORVs must be opened within 5 to 10 minutes after the beginning of the transient for core cooling to be successful.

It must be emphasized that real plants may not be easily classified into four neat classes. Nevertheless, these four classes will enable the benefits of enhanced feed-and-bleed to be scoped out. The benefit of enhanced pressure relief capacity can be seen by comparing Class 4 with Class 3 and the benefit of enhanced makeup by comparing Classes 2, 3 and 4 with Class 1.

Given the four classes of plants, it is now necessary to discuss the sources of failure for feed-and-bleed. These may be grouped into equipment, thermal-hydraulic, and human failure probabilities.

For feed-and-bleed to work, there must be both feed and bleed capabilities. Thus, a source of coolant at sufficient flow and pressure is necessary. This can be supplied either by the "charging" or "makeup" system (if of sufficient flow capacity) or by the HPI system (if of sufficient discharge pressure). In either case, the supply will generally be from a two-train system. Such systems generally have a failure probability on the order of 1%.

Class 1 plants will discharge through the safety valves which have a failure probability of essentially zero for our purposes. The other three classes must use (usually two) PORVs for coolant discharge. Each PORV has a probability of failure to open of about 1%.[54] When used for feed-and-bleed, these valves are not redundant; both must open.

Thermal-hydraulic effects are reasonably straightforward. For Class 1 plants, the thermal-hydraulic failure probability is essentially zero, since the high head HPI pumps will raise coolant level at any time. For Class 2 and Class 3, we will define two time intervals. The first is T1, which runs from the beginning of the transient up to the point of steam generator dryout. The second is T2, which starts at steam generator dryout and ends at the point of no return, when feed-and-bleed will no longer work. During interval T1, the initial conditions for feed-and-bleed onset are reasonably stable and there is high confidence that feed-and-bleed will work as planned. Thus, the probability of failure due to thermal-hydraulic effects is assumed to be zero during T1. During the second interval T2, the dynamic behavior of the reactor coolant system is much more complicated. In addition, the course of the transient may be significantly affected by a number of factors such as reactor coolant pump operations, PORV cycling, pressurizer sprays, etc. We estimate, based primarily on judgment, that the probability of failure is 50% during this interval.

For Class 4 plants, the point of no return comes well before steam generator dryout. Thus, it will be assumed that the probability of failure due to thermal-hydraulic effects is essentially zero for the first 10 minutes and unity thereafter.

Finally, we must account for human error. This will be divided into three parts:

(1) Simple Procedural Error: Assuming a decision has been made to go ahead with feed-and-bleed, and assuming also that all equipment is operable, there is still a finite probability that the operator will make a mistake in initiating, monitoring, and controlling the process. This failure probability is lowest for Class 1 plants since the operator need only initiate HPI and watch. We will assume 1% failure probability for this class. For Class 2, the initiation and control of feed-and-bleed are more complicated and we will assume 5% for interval T1. For Class 2 interval T2 and for Classes 3 and 4, the operator must depressurize first and then feed, being careful to keep pressure low enough to get adequate injection flow but high enough to avoid bulk boiling in the core (if possible). For this situation, we will assume a 10% failure rate.

(2) Time Stress: For this, we will use Swain's screening model.[339] The Class 2 and Class 3 interval T1 ends roughly 25 minutes into the transient, for which the screening model estimates a stress failure rate of about 3%. For the case of Class 4, where the point of no return is 10 minutes after the start of the transient, the screening model predicts a 50% failure probability. All the other classes and intervals are well over half an hour and the time stress failure rate is essentially zero.

(3) Simple Reluctance: The use of feed-and-bleed will release primary coolant to the containment atmosphere, contaminating the containment and necessitating a long expensive shutdown for purposes of cleanup. Moreover, feed-and-bleed techniques cause a small LOCA and thus have safety implications. Quite naturally, the plant operators will delay the use of feed-and-bleed as long as possible in the hope of recovering either main or auxiliary feedwater. Thus, there is a finite probability that initiation of feed-and-bleed will be delayed into interval T2 (for Classes 2 and 3) or even past the point of no return. Once again, it is necessary to use judgment. We will assume a 5% probability that the operators will wait until after the point of no return. For Classes 1 and 4, this translates directly into a 5% failure probability. For Classes 2 and 3, we will further assume that there is a 5% chance that feed-and-bleed will be started before the point of no return but after the point of steam generator dryout. This can perhaps best be understood in terms of success probabilities: there is a 90% chance of initiation during interval T1, a 5% chance of initiation during interval T2, and a 5% chance of either no initiation or initiation after interval T2.

For feed-and-bleed to succeed, all the potential pitfalls discussed above must be successfully overcome. Thus, the probability of successful feed-and-bleed is obtained by multiplying the success probabilities (not the failure probabilities) of the various contributors listed above. This is summarized in the following Table 3.125-1.

Table 3.125-1

Class	1	2		3		4
Interval		T1	T2	T1	T2	
Success Probabilities:						
HPI	0.99	0.99	0.99	0.99	0.99	0.99
PORV	---	0.99	0.99	0.99	0.99	0.99
PORV	---	0.99	0.99	0.99	0 99	0.99
Thermal-Hydraulic	1.00	1.00	0 50	1.00	0.50	1.00
Operator:						
Procedural	0.99	0.95	0.90	0.90	0.90	0.90
Time Stress	1.00	0.97	1.00	0.97	1.00	0.50
Reluctance	0.95	0.90	0.05	0.90	0.05	0.95
Interval Success Probability	0.9311	0.8047	0.0218	0.7624	0.0218	0.4148
Interval Failure Probability	0.0689	0.1953	0.9782	0.2376	0.9782	0.5852
Class Failure Probability	0.0689	0.1910		0.2324		0.5852

For Classes 1 and 4, the failure probability is calculated by first multiplying the equipment, thermal-hydraulic, and operator success probabilities together to obtain a net success probability. This success probability is then subtracted from unity to get a failure probability.

Classes 2 and 3 are more complicated. Within each time interval, the various success probabilities are multiplied together to get a net success probability for the interval. The interval success probabilities are then subtracted from unity to get an interval failure probability (i.e., the probability of no feed-and-bleed during that interval). Both intervals must fail to feed and bleed for feed-and-bleed to not take place at all. Therefore, the failure probability for the plant class is the product of the two interval failure probabilities.

With feed-and-bleed failure probabilities available, the next step is to calculate the changes in core-melt frequencies from these numbers. This is relatively straightforward in that the dominant sequence is almost always a transient involving a non-recoverable loss of main feedwater coupled with a failure of the AFW system and (of course) a failure to cool the core by means of feed-and-bleed techniques.

For the initiating event frequency (non-recoverable loss of main feedwater), we will use 0.64 event/RY, based upon the Oconee PRA done by Duke Power Co.[889] This figure is based upon fault tree analysis and should be reasonably representative of most main feedwater system designs.

For a three-train AFW system, a "typical" unavailability is 1.8×10^{-5}/demand.[894] The analogous figure for a two-train system is significantly higher. However, an existing program is attempting to upgrade all AFW systems to a point where the maximum unavailability would be 10^{-4}/demand.[947] Thus, we will consider 1.8×10^{-5} to be an average unavailability and 10^{-4} to be the maximum.

With the figures in hand, core-melt frequencies (F) can be estimated by taking the product of the transient frequency, the AFW unavailability, and the change in the feed-and-bleed failure probability.

From Class	To Class	Change in Core-Melt Frequency* Typical	Maximum	Reason
2	1	1.4×10^{-6}	7.8×10^{-6}	Enhanced makeup capacity
3	1	1.9×10^{-6}	1.1×10^{-5}	Enhanced makeup capacity
4	3	4.1×10^{-6}	2.3×10^{-5}	Enhanced relief capacity
4	1	6.0×10^{-6}	3.3×10^{-5}	Enhanced makeup and relief capacity

*in units of core-melt/RY

Consequence Estimate

The accident sequence under consideration here involves a core-melt with no large breaks initially in the reactor coolant pressure boundary. The reactor is likely to be at high pressure (until the core melts through the lower vessel head) with a steady discharge of steam and gases through the PORV(s). These are conditions likely to produce significant hydrogen generation and combustion. The Zion and Indian Point PRA studies used a 3% probability of containment failure due to hydrogen burn (the "gamma" failure). We will follow this example and use 3%, bearing in mind that specific containment designs may differ significantly from this figure.

In addition, the containment can fail to isolate (the "beta" failure). Here, the Oconee PRA[889] figure of 0.0053 will be used. If the containment does not fail by isolation failure or hydrogen burn, it will be assumed to fail by base mat melt-through (the "epsilon" failure).

Using the usual prioritization assumptions of a central midwest plains meteorology, a uniform population density of 340 persons per square mile, a 50-mile radius, and no ingestion pathways, the consequences are:

Failure Mode	Percent Probability	Release Category	Consequences (man-rem)
gamma	3.0%	PWR-2	4.8×10^6
beta	0.5%	PWR-5	1.0×10^6
epsilon	96.5%	PWR-7	2.3×10^3

The "weighted-average" core-melt will have consequences of 1.5×10^5 man-rem. These figures should cover all PWRs with large dry containments. However, they do not apply to ice condenser containments. There is no modern PRA currently available for such a plant. However, because of the low free volume in such a containment, failure due to overpressure is more likely and the average consequences may be significantly greater.

Cost Estimate

The core-melt figures for this issue are such that cost considerations will not affect the priority. Consequently, a quantitative cost analysis has not been attempted. However, it should be noted that these are not inexpensive fixes. A new or upgraded high pressure pump is likely to cost between $2M and $5M per train installed. Replacement PORVs or an additional, dedicated depressurization valve will not be as expensive, but will probably require replacement discharge piping with stronger bracing. The quench tank might also require extensive modification.

Value/Impact Assessment

To make the value/impact assessment, it is necessary to estimate the number of plants in each of the four classes. The first statement to be made is that all B&W plants except Davis-Besse have injection pumps capable of lifting the pressurizer safety valves. Thus, these plants are already in Class 1 and are outside the scope of this issue. This leaves 71 PWR plants. The earliest implementation of fixes for this issue is not likely to be before the spring refueling outages in 1988, at which time these plants will have a collective remaining lifetime of about 2240 RY. At a 75% utilization figure, this is about 23.7 years of operational life per plant. It is not clear how these 71 plants are distributed among Classes 2, 3 and 4. A plant-by-plant investigation is beyond the scope of a prioritization. Therefore, it will be assumed that roughly one-third fall in each class: 24 in Class 2, 24 in Class 3, and 23 in Class 4. With this data, priority parameters can be estimated.

	Part (a), Enhanced Relief	Part (b), Enhanced Makeup		
Plant Class	4-3	2-1	3-1	4-1
Number of Plants	23	24	24	23
ΔF (average)	4.1×10^{-6}	1.4×10^{-6}	1.9×10^{-6}	6.0×10^{-6}
ΔF (max)	2.3×10^{-5}	7.8×10^{-6}	1.1×10^{-5}	3.3×10^{-5}
Core-Melt/RY (max)	2.3×10^{-5}		3.3×10^{-5}	
Man-rem/reactor (max)	80		120	
Core-Melt/year (Total, all plants)	9.4×10^{-5}		2.2×10^{-4}	
Man-rem (Total, all plants)	330		770	

Other Considerations

(1) Upgrading the makeup capability would involve work on pumps which are located outside of containment. This should not result in a significant amount of ORE. However, upgrading the relief capacity involves work adjacent to the pressurizer which would have implications for occupational exposure. There is no readily available data upon which a direct estimate of this exposure can be based. However, it should be noted that pressurizer inservice inspection involves roughly 20 man-rem and pressurizer spray valve repair involves roughly 10 man-rem. Thus, because the average (not maximum) plant would avert a public risk of about 15 man-rem, the ORE involved in the fix may well be equal to or greater than the public exposure averted.

(2) In addition to ORE associated with the fix, there is averted ORE associated with cleanup of a core-melt. For prioritization purposes, core-melt cleanup exposure is assumed to be 20,000 man-rem. Using this and the core-melt frequencies calculated previously, the actuarial values (total, all plants) of averted core-melt cleanup ORE are about 45 man-rem for Part (a) and 100 man-rem for Part (b). On a per-plant basis, this is 2 man-rem/plant for both Parts (a) and (b). Thus, this is not a significant consideration.

(3) There are also averted costs associated with this issue. There are no averted precursor events that involve major cleanup, but there are averted cleanup costs associated with the reduction in core-melt frequency. In addition, averted core-melt implies averted replacement power costs for the remaining life of the plant. (Because the plant was built for the purpose of avoiding replacement power costs, this latter item represents the depreciated capital loss of the plant). Using the maximum core-melt frequencies above, a 31.5 calendar-year average remaining plant life, and the usual prioritization assumptions of $1 billion for core-melt cleanup, $300,000 per day for replacement power, and a discount rate of 5%, the actuarial cost credits are:

	Part (a)	Part (b)
Core-melt Cleanup	$270,000	$390,000
Averted Replacement Power Costs	$350,000	$510,000
Total:	$620,000	$900,000

This is probably not sufficient to offset more than a fraction of the cost of the proposed figures.

(4) The estimates of feed-and-bleed failure probability are based upon a time window assumption. That is, after continuing decay heat production in the reactor core has caused primary system pressure to rise to a certain point, the HPI pumps can no longer force coolant into the primary system. In addition, the PORVs are then venting at capacity and thus the primary system cannot be depressurized. Therefore, feed-and-bleed is assumed to fail if initiated after such conditions are reached.

However, a second opportunity for successful feed-and-bleed may exist. This would occur after the primary coolant boils away to the point where the core is starting to uncover. The steaming rate then begins to diminish and the PORVs may be able to depressurize the primary system to the point where the HPI pumps can reflood the core.

Of course, this depressurization is only possible because the decay heat is causing the uncovered fuel's temperature to rise instead of going into steam production. The pressure may not drop fast enough for core melt to be averted. Also, if the uncovered fuel slumps or crumbles and falls into the remaining liquid coolant, pressure will rise again. It is beyond the scope of a prioritization to address this (theoretical) second window possibility. However, any subsequent value/impact analyses should address the possibility of a second window.

(5) The analysis assumes a 1% failure probability for the PORV(s). Some plants have operated for extensive periods with the PORV block valves closed and electrically disabled. Restoration of power to the block valve operators, and subsequent opening of the block valves and PORVs to permit feed-and-bleed cooling, would take a significant amount of time as well as opening new possibilities for equipment malfunction and operator error. Thus, such plants might have feed-and-bleed failure probabilities significantly greater than those calculated in the analysis above.

CONCLUSION

Based upon the above analysis, particularly the maximum core-melt frequencies, this issue would normally be placed in the high priority category. However, feed-and-bleed techniques are being evaluated[738] and will be considered as one option in the resolution of USI A-45.[953] Therefore, this issue should be DROPPED as a separate issue.

ITEM 125.II.10: HIERARCHY OF IMPROMPTU OPERATOR ACTIONS

DESCRIPTION

Historical Background

This issue was identified as part of a list of long-term generic actions that resulted from the Davis-Besse event of June 9, 1985.[940] During the event, the operators did not initiate feed-and-bleed cooling immediately upon reaching plant conditions where feed-and-bleed operations were required by the emergency procedures. The feed-and-bleed method of cooling was delayed because of the operators' belief that recovery of feedwater was imminent and their reluctance to release reactor coolant to the containment structure. Even though feedwater flow was recovered before serious damage resulted, the event highlighted the need for establishing a hierarchy of actions in the procedures and/or training which would focus impromptu actions during an event to assure that decisions will be in the direction of safety, and not based on potential plant operational difficulties and financial impacts.

Safety Significance

Delays in implementing emergency operating procedures (EOPs) in a timely manner could defeat the design safety function of equipment and increase the severity of a transient or accident.

Possible Solution

Issue HF4.4 is to provide assurance that plant procedures are adequate and can be used effectively; the objective is to provide procedures that will guide the operators in maintaining the plant in a safe state under all operating conditions, including the ability to control upset conditions without first having to diagnose the specific initiating event. This objective is to be met by: (1) developing guidelines for preparing, and criteria for evaluating, EOPs, normal operating procedures, and other procedures that affect plant safety; and (2) upgrading procedures, training the operators in their use, and implementing the upgraded procedures.

In accordance with Appendix A of NUREG-0985, Revision 2,[651] comparative studies have been completed which examined the impact on operator performance in making the transition from procedure to procedure, using either event-based or function-oriented EOPs. The results of these studies are being incorporated into a larger, ongoing project to develop guidance for achieving successful transitions with nuclear power plant operating procedures. DHFT concluded that, while the procedural guidance package may develop the correct guidance to place the reactor in a safe state, it may not prevent reluctance on the part of supervision or an operator to take action which will invariably result in a financial penalty. The TMI Action Plan Item I.B.1.3 (Loss of Safety Function) resolution to use existing enforcement options (citations, fines, and shutdowns) provides a deterrent to such actions, including willful violations, that could effect the health and safety of the public (10 CFR 2, Appendix C).[197] The Commission noted[234] that, while the procedures for enforcement actions may not ensure compliance, civil penalties and possibly criminal prosecution for willful violations are strong incentives to comply. NRC policy is that noncompliance should be more expensive than compliance. In cases involving individual operators licensed under 10 CFR Part 55, the Commission policy statement[234] states that generally licensees are held responsible for the acts of their employees. Accordingly, the NRC policy should not be construed as excusing personnel errors. Thus, enforcement actions involving individuals, including licensed operators, will be determined on a case-by-case basis. The NRC policy is directed toward encouraging licensee initiatives for self-improvements and identification and correction of such problems.

CONCLUSION

The concern raised relative to reluctance of the licensee (or plant operators) to proceed with appropriate actions to place the plant in a safe state of operation, based on potential plant operational difficulties and financial impacts, is addressed by existing NRC policies.[197,234] Based on the above discussion, the issue involving development of the hierarchy of impromptu operator actions is to be addressed in Issue HF4.4. Therefore, Issue 125.II.10 should be DROPPED as a separate issue.

ITEM 125.II.11: RECOVERY OF MAIN FEEDWATER AS ALTERNATIVE TO AFW

This item is currently being prioritized.

ITEM 125.II.12: ADEQUACY OF TRAINING REGARDING PORV OPERATION

DESCRIPTION

Historical Background

This issue affects all operating PWRs with PORVs in the primary coolant loop and calls for an assessment of the adequacy of training regarding PORV operations.[940] The issue stems from Findings 8 and 14 of the NRC investigation of the Davis-Besse event[886] of June 9, 1985 in which the NRC staff noted that the post-TMI improvements that focused on EOPs and training played a crucial role in mitigating the event. Following actuation of the PORV during the event, the operator observed that the PORV open/close indicator showed that the PORV had closed. In fact, the PORV had not completely closed and, as a result, the reactor pressure decreased at a rapid rate for about 30 seconds. The operator however did not verify closure of the PORV by looking at the acoustical monitor installed after the TMI accident; instead, he looked at the indicated pressure level which appeared steady. As a precautionary measure, the operator closed the PORV block valve. Fortunately, when the block valve was subsequently opened to assure PORV availability, the PORV had closed during the time the block valve was closed. Had the operator looked at the acoustical monitor, the need to close the block valve may have been factually confirmed and may have precluded the need for relying on the precautionary action taken. However, it should be noted that the operators have not generally placed high reliance on the acoustical monitors because of PORV leakage problems.

Safety Significance

Assessments of the adequacy of training and hands-on experience, referred to as performance-based training or Systems Approach to Training (SAT), is considered essential for providing assurance that nuclear power plants are operated in a safe state under all operating conditions. The adequacy of training regarding the PORV operation is part of the assessments of the performance-based training evaluations described in Issue 125.I.7.b, "Realistic Hands-on Training."

Possible Solution

A possible solution to this issue is to include an assessment of the adequacy of training regarding PORV operations in the job catalog of necessary tasks and functions required to safely operate and control nuclear power plant operations.

PRIORITY DETERMINATION

Frequency Estimate

PORV Challenge Frequency: The PORV challenge frequency was determined to be approximately 1/RY in Issue 70, "PORV and Block Valve Reliability."

PORV/Block Valve Failure Frequency: The frequency of failure of the PORV to close, given that it has opened, is estimated to be 0.01/demand (See Issue 70). The frequency of failure of the block valve to function is estimated to be 0.003/demand (See Issue 70).

Operator Error Frequency: Based on the information in Issue 70, the human error probability (HEP) to close the PORV after the TMI Action Plan[48] improvements and increased emphasis on operator training is estimated to be 0.05.

PORV-SBLOCA Frequency: The estimated base-case PORV/block-valve SBLOCA frequency (5.3×10^{-4}/RY) is the product of the PORV challenge frequency (1.0), the probability that the PORV sticks open (0.01), and the probability that the operator will not close the PORV or the block valve fails to close (0.05 + 0.003).

To assess the potential improvement in HEP for PORV operations that may result from adequate hands-on training in upgraded simulators, a 30% reduction in HEP is assumed. (See Issue I.A.4.2, "Long-Term Training Simulator Upgrade.") Adjusting the above HEP = 0.05 to account for the potential reduction in HEP, the adjusted HEP = (0.7)(0.05) = 0.035. The resulting potential reduction in PORV-SBLOCA frequency derived by requiring the PORV training in the job catalog (Issue HF3.1) is therefore estimated to be $[(5.3 \times 10^{-4})/\text{RY} - (1.0)(0.01)(0.035 + 0.003] = 2.5 \times 10^{-4}$/RY. Given the visibility of PORV training since the TMI-2 accident, the above 30% reduction in HEP may over-estimate the potential HEP benefit. However, the assumed 30% reduction is expected to bound the safety significance of this issue.

Consequence Estimate

Ratioing the above reduction in PORV-SBLOCA frequency (2.5×10^{-4}/RY) to the PORV-SBLOCA frequency from Issue 70 (1.05×10^{-3}/RY) and multiplying by the core-melt frequency from Issue 70 (4.2×10^{-6}/RY) yields the potential reduction in core-melt frequency for this issue of $(0.24)(4.2 \times 10^{-4}/RY) = 10^{-6}/RY$. The public risk reduction is therefore (0.24)(31 man-rem/reactor) = 7.4 man-rem/reactor (See Issue 70).

CONCLUSION

Issue HF3.1 evaluated the task selection process for training program content based on the relative importance of operator tasks and requirements. Tasks involving the use of PORVs for both feed-and-bleed cooling and for identification of potential LOCAs are included in the generic INPO task analysis listings for PWRs and in NUREG-1122,[974] Item EK3.03, "Actions Contained in EOP for PZR Vapor Space Accident/LOCA." This event has one of the highest importance ratings (4.6 of 5.0) for PWRs and is included in both training and NRC exams. The high frequency of PORV challenges is to be addressed in Issue HF3.1. Therefore, Issue 125.II.12 should be DROPPED as a separate issue.

ITEM 125.II.13: OPERATOR JOB AIDS

This item is currently being prioritized.

ITEM 125.II.14: REMOTE OPERATION OF EQUIPMENT WHICH MUST NOW BE OPERATED LOCALLY

DESCRIPTION

Historical Background

During the course of the investigation of the event, it was noted that a startup feedwater pump (SUFP), a part of the main feedwater system that would have been very helpful in the mitigation of the transient, had been intentionally disabled because of an NRC concern with high energy line breaks in the area of essential safety equipment and the ability of ECCS equipment to meet single failure criteria. Although the Davis-Besse event specifically involved a SUFP, it is intended that this issue cover all equipment that has been disabled such that it is no longer remotely operable from the control room.

Safety Significance

The significance of purposely disabled equipment lies primarily in timing. Generally, it is possible to restore such equipment to an operable status. However, plant personnel must be dispatched to the equipment to perform local, manual operations such as unlocking and manipulating manual valves, restoring and closing breakers, etc. This can require considerable time and restoration to operability may well come too late to aid in accident mitigation. Moreover, the relatively complex procedures involved, done under emergency conditions, are prone to error. Finally, the nature of the incident may well be such that the disabled equipment is rendered inaccessible.

Possible Solution

The solution proposed[900] is straightforward: "Review each piece of motor-operated equipment originally designed to be operated from the control room or other panel areas which has been disabled physically such that it can only be operated locally to determine whether such disabling truly is in the interest of overall plant safety."

PRIORITY DETERMINATION

Over the years, there have been many instances where equipment has been intentionally disabled. In the case of the Davis-Besse SUFP, the reason was to ensure that the discharge lines, which are not seismically qualified and which also are routed near essential safety equipment, could not rupture and disable this equipment. Other reasons also exist. For example, equipment has in the past been disabled by removal of breakers to permit older ECCS designs to meet the single failure criterion.

This issue is non-specific in the sense that it addresses any of this disabled equipment. Thus, re-enabling of this equipment may affect LOCA sequences, transient-initiated sequences, etc. Because of this very general nature, it is impossible to quantify all aspects explicitly. The approach we will use is to evaluate a SUFP similar to that of Davis-Besse, but (unlike the case of Davis-Besse) capable of providing sufficient flow by itself to permit decay heat removal by means of the steam generators. Because such a pump would help mitigate transient-initiated sequences, which are relatively frequent compared to (for example) LOCA-initiated sequences, this scenario should provide an upper bound to the priority parameters.

Frequency Estimate

The sequence of interest is straightforward. It is initiated by a non-recoverable loss of main feedwater. If the auxiliary feedwater system fails, the SUFP is not re-enabled in time, and feed-and-bleed techniques fail, core melt will ensue.

For the initiating event frequency (non-recoverable loss of main feedwater), we will use 0.64 event/RY, based upon the Oconee PRA done by Duke Power Co.[889] This figure is based upon fault tree analysis and should be reasonably representative of most main feedwater system designs.

For a three-train AFW system, a "typical" unavailability is 1.8×10^{-5}/demand.[894] The analogous figure for a two-train system is significantly higher. However, an existing program (Issue 124) is considering whether to upgrade all AFW systems to a point where the maximum unavailability would be 10^{-4}/demand. These plants would almost certainly upgrade their SUFPs (if present) to help meet this criterion, which makes this issue moot for these plants; thus, we will use 1.8×10^{-5}/demand.

We will assume a typical value of 0.20 for the failure probability of feed-and-bleed cooling, based upon the calculations presented under Issue 125.II.9, "Enhanced Feed-and-Bleed Capability."

The SUFP non-recovery probability remains to be calculated. According to the Investigation Team's report on the Davis-Besse event,[886] restoration of the SUFP normally takes 15 to 20 minutes. Nevertheless, the assistant shift supervisor managed to do it in roughly 4 minutes during the June 9, 1985 event. Obviously, not all plant personnel are going to go through the procedure as rapidly as the assistant shift supervisor at Davis-Besse even given the extra motivation of a real event. We will assume that the time needed to restore the SUFP to operability can be described by a normal distribution, centered at 17.5 minutes and with a width such that the assistant shift supervisor's performance of 4 minutes is at the first 95 percentile point.

The time intervals above are measured from the start of the restoration procedure. It is desirable for calculational purposes to measure time from the initiation of the transient. Noting from NUREG-1154[886] that the SUFP was restored at t = 16.38 minutes (measured from the start of the transient) after four minutes of rapid work on the part of the assistant shift supervisor, the significant times are:

$t = 0$,	start of transient
$t = 12.38$ minutes,	start work on SUFP
$t_{95} = 16.38$ minutes,	95 percentile point
$t_0 = 29.88$ minutes,	mean time for restoration

Thus, the probability of the SUFP being restored within the interval from t to (t + dt) is given by:

$$P(t)dt = (\sqrt{2\pi}\,\sigma)^{-1} \exp\{-\tfrac{1}{2}[(t-t_0)/\sigma]^2\}dt$$

where $\sigma = 8.93$ minutes (based on $t_0 - t_{95} = 13.5$ minutes)

If one is willing to wait long enough, the integrated probability of restoration approaches unity. However, there is a point in time after which restoration of the SUFP will no longer save the core. Although it is not clear just when this time is, it is safe to assume that it occurs after steam generator dryout which is typically at least 25 minutes into the transient. The probability of no restoration is given by:

$$P_F\,(T) = \int_T^{\infty} P(t)\,dt, \text{ where } T \geq 25 \text{ minutes}$$

There is no closed form solution to this integral. However, standard statistical tables readily give an answer of $P_F(T) \leq 0.29$.

One last effect needs to be considered. Consistent with Issue 122.3, "Physical Security System Constraints," an additional 1% probability of the plant personnel being unable to reach the equipment location because of locked doors, etc., must be considered. The core-melt frequency then becomes:

Core-melt/RY $\leq$ (0.64 loss of main feedwater events/RY) $\times$ (1.8×10^{-5} AFW failure probability) $\times$ (0.20 feed-and-bleed failure probability) $\times$ (0.29 + 0.01 SUFP non-restoration probability)

$\leq 6.9 \times 10^{-7}$

Consequence Estimate

The core-melt sequence under consideration here involves a core-melt with no large breaks initially in the reactor coolant pressure boundary. The reactor is likely to be at high pressure (until the core melts through the lower vessel head) with a steady discharge of steam and gases through the PORV(s). These are conditions likely to produce significant hydrogen generation and combustion.

The Zion and Indian Point PRA studies used a 3% probability of containment failure due to hydrogen burn (the "gamma" failure). We will follow this example and use 3%, bearing in mind that specific containment designs may differ significantly from this figure.

In addition, the containment can fail tc isolate (the "beta" failure). Here, the Oconee PRA figure of 0.0053 will be used. If the containment does not fail by isolation failure or hydrogen burn, it will be assumed to fail by basemat melt-through (the "epsilon" failure).

Using the usual prioritization assumptions of a central midwest plains meteorology, a uniform population density of 340 persons per square mile, a 50-mile radius, and no ingestion pathways, the consequences are:

Failure Mode	Percent Probability	Release Category	Consequences (man-rem)
gamma	3.0%	PWR-2	4.8×10^6
beta	0.5%	PWR-5	1.0×10^6
epsilon	96.5%	PWR-7	2.3×10^3

The "weighted-average" core-melt will have consequences of 1.5×10^5 man-rem.

The plants to be examined include all operating plants (presently 94). As of the fall of 1987 (the earliest that changes are likely to be made), these plants will have an aggregate remaining license lifetime of 2718 RY. This corresponds to an average lifetime of 29 calendar-years per plant. At a 75% utilization factor, this is 22 operational years per plant.

It is not known how many plants would be affected by this issue. We will assume that at least a few plants will be found and will calculate priority parameters on a per-plant basis. Thus, the estimated risk reduction per plant is $(6.9 \times 10^{-7})(22)(1.5 \times 10^5)$ man-rem or 2.3 man-rem.

Cost Estimate

The fix for this issue, once equipment is identified, is to do a detailed analysis to see if the disabling of the subject equipment is truly in the interest of plant safety. If the analysis indicates that the equipment should not be disabled, the original reason for disabling must still be addressed. (Alternatives to disabling may be necessary to address the original concern.)

The minimum cost would correspond to a case where the equipment is process equipment, which is fully maintained and needs only to have valves opened and breakers re-installed, which would take (we assume) roughly 17.5 minutes of labor. If it also turns out that no other alternatives are necessary, the cost would be dominated by analysis and paperwork. We estimate that probabilistic analyses would require approximately 10 weeks of staff time (NRC and industry combined) per plant, at \$100,000/staff-year. In addition, per-plant costs of \$13,000 for NRC and \$16,000 for the licensee would be incurred for a typical straightforward technical specification change. The minimum cost is then about \$50,000/plant.

Value/Impact Assessment

Based on a potential risk reduction of 2.3 man-rem/reactor and a cost of \$50,000/reactor, the value/impact score is given by:

$$S = \frac{2.3 \text{ man-rem/reactor}}{\$0.05\text{M/reactor}}$$

$$= 46 \text{ man-rem/\$M}$$

Other Considerations

The aggregate parameters (total man-rem, all reactors, and total core-melt/year, all reactors) are not calculated here. An examination of the scale factors for these parameters readily shows that at least 50 plants must be affected before it is possible for these parameters to be limiting.

In most cases, the fix will not involve work within radiation fields and thus will not involve ORE. The ORE averted due to post-feed-and-bleed-cleanup and post-core-melt cleanup is a minor consideration. The ORE associated with cleanup is estimated to be 1800 man-rem, after a primary coolant spill, and 20,000 man-rem, after a core-melt accident.[64] If the frequency of feed-and-bleed events is 3.46×10^{-6}/RY, the actuarial cleanup ORE averted is only 0.14 man-rem/reactor. Similarly, a core-melt frequency of 6.9×10^{-7}/RY corresponds to an actuarial averted cleanup ORE of only 0.20 man-rem/reactor. If averted ORE were added to the man-rem/reactor and man-rem/\$M figures above, no conclusions would change.

The proposed fix would reduce core-melt frequency and the frequency of feed-and-bleed events and therefore would avert cleanup costs and replacement power costs. The cost of a feed-and-bleed usage is dominated by roughly six months of replacement power while the cleanup is in progress. If the average frequency of such events is 3.46×10^{-6}/RY and the average remaining lifetime is 29 calendar-years at 75% utilization, then making the usual assumptions of a 5% annual discount rate and a replacement power cost of \$300,000 per day, the actuarial savings for feed-and-bleed cleanup is estimated to be \$2,200. Similarly, the actuarial savings of averted core-melt cleanup (which is assumed to cost one billion dollars if it happens) are about \$7,900. The actuarial savings from replacement power after a core-melt up to the end of the plant life are about \$9,600. (This last figure represents the lost capital investment in the plant.)

If these theoretical cost savings were subtracted from the expense of the fix, the man-rem/$M would rise to 76 and would not change any conclusions.

Some caution is needed in the use of the numbers calculated above. It must be remembered that these are maximum numbers, calculated for a worst case scenario. It must also be remembered that equipment has often been disabled for good reasons. Re-enabling such equipment will generally have drawbacks as well as benefits and the net effect on plant safety is not necessarily positive.

CONCLUSION

Based upon the figures presented above, this issue should be placed in the LOW priority category.

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ISSUE 127: MAINTENANCE AND TESTING OF MANUAL VALVES IN SAFETY-RELATED SYSTEMS

DESCRIPTION

Historical Background

This issue was identified in the NRC Incident Investigation Team (IIT) report on the loss of integrated control system (ICS) power event at Rancho Seco on December 26, 1985 (NUREG-1195).[1006] Following the event, it was requested that the adequacy of the maintenance program for manual valves be prioritized as a generic issue.[1007] In addition, an Information Notice[1008] was drafted by the staff and was later issued as IE Information No. 86-61[1010] on July 28, 1986.

Safety Significance

In the Rancho Seco event, when power was lost to the ICS, the plant responded as designed; the AFW ICS flow control valves as well as other valves went to the 50% open position. However, AFW flow was excessive. After an unsuccessful attempt to manually close the flow control valve to the "A" OTSG, the operator attempted to close the manual isolation valve. This isolation valve was "frozen" in the open position and could not be moved even when a valve wrench was used. Therefore, the inability to reduce AFW flow resulted in an overcooling event. The IIT found that the failure of the AFW manual isolation valve was the result of a lack of preventive maintenance (including lubrication) on this valve during the entire operational life of the plant (about 10 to 12 years).

The manual isolation valve is a locked-open valve located in the AFW discharge header to the "A" OTSG. During the IIT investigation, a Sacramento Municipal Utility District (SMUD) representative stated that the entire AFW system which would include this manual isolation valve is safety-related. However, from other discussions with SMUD personnel, it appears that this valve was only intended to be used to isolate the AFW (ICS) flow control valve for maintenance. The valve is categorized as an ASME Category E valve (i.e., it is normally locked-open to fulfill its function). ASME Section XI (1974 edition) requires no regular testing of Category E valves. The position of the valves is merely recorded to verify that each valve is locked or sealed in its correct position. The current edition of ASME Section XI no longer includes a Category E for valves.

Following the incident, it was found that licensees do not have a regular maintenance program that applies to every manual valve. The NRC does not have a requirement for maintenance and testing of convenience valves such as the locked-open manual valve involved in the Rancho Seco incident. ASME Section XI specifies ISI, testing, repair, and replacement of valves that are components in systems classified ASME Classes 1, 2 and 3 and are required to perform a specific function in shutting down a reactor to a cold shutdown condition, or in mitigating the consequences of an accident. Manual valves in safety-related systems that are classified Quality Group A, B, or C in conformance with Regulatory Guide 1.26 are constructed to ASME Section III, Classes 1, 2, or 3 or to earlier codes and standards, as appropriate. These manual valves may be fill, vent, drain, or convenience valves and are constructed to the same code class as the system, or part of a system, of which they are a part. Such valves are not included in the IST program for valves that are in conformance with ASME Section XI as noted above because they are not required to change position to perform a safety function. In the event a manual valve is required to change position to perform a safety function, it is included in the ASME Section XI IST program and classified as a safety-related valve.

The NRC requirements for valve testing are contained in 10 CFR 50.55 (a)(g) which incorporates ASME Section XI. Therefore, regulatory requirements for valve testing extend only to valves that are within the IST program. The Quality Group (Safety Class) and construction code of each valve are verified and the valve category is also verified for conformance with Section XI, IWV-2000. In addition, the NRR staff performs a completeness review to assure that all appropriate valves that are within the scope of ASME Section XI are included in the IST program. It is the licensees' responsibility to perform the testing, repair, and maintenance on the valves that are within their IST and maintenance programs.

Possible Solutions

The two possible solutions are: (1) develop or revise regulatory requirements relating to the inspection, testing, and maintenance of those fill, vent, drain and convenience valves in safety-related systems that do not change position for the systems to perform their safety function; or (2) identify this as an item for which the NRC has concern, notify the licensees by an information notice, and let them determine the maintenance practices they wish to implement.

PRIORITY DETERMINATION

Frequency/Consequence Estimate

To determine the reduction in core-melt frequency which could result from improving the maintenance of manual valves, the ANO 1-Unit One IREP analysis was used.[366] This plant risk study provides a very detailed list of the cut sets and component failures which could result in system unavailability. After a thorough review, no manual valve faults where found for which the mode of failure was the inability to close the valve.

In retrospect, the absence of any identified failure modes concerning the inability to close a manual valve is not surprising; manual valves are, for the most part, installed to permit the isolation of other components, i.e., pumps and motor-operated valves, to permit testing or maintenance without the necessity of shutting the plant down. Hence, they are generally not used for normal or planned emergency operations to control fluid flow. The principal modes of failure associated with manual valves that are identified in risk analyses are either the blockage of a valve or the failure to restore a valve to the open position after having closed it for test or maintenance. In general, most manual valves of the category being considered in this issue are locked in the open position to minimize the chances for inadvertent closure.

Another reason for not finding the failure mode for manual valves in the IREP study[366] is that credit was not given for unplanned recovery actions. Planned operations as used herein include both normal and emergency operations which are directed by procedures. Hence, valve utilization as was attempted at Rancho Seco would be considered an unplanned recovery event.

Lastly, the expected frequency of any identified cut sets in which the failure mode included the failure to close a manual valve may have been less than the selected cutoff or truncation value. Considering the failure combinations necessary to involve a manual valve, such may be the case.

It should not be concluded that there is no contribution to core-melt and risk by failures which prevent the closure of manual valves (as was the case in the Rancho Seco event) because of their absence from available risk studies or PRAs. As is evident from the Rancho Seco event, the inability to close a manual isolation valve contributed in part to an overcooling event. However, it is probably justifiable to conclude that the inability to close a manual valve contributes only a small amount, i.e., less than 10^{-6}, to core-melt and hence to risk. Due to the lack of any identifiable failure or fault combinations in the PRAs, there is no practical basis on which to quantify in this limited analysis the contribution to core-melt and risk resulting from these valve failures.

Cost Estimate

Industry Cost: Approximately 100 manual isolation valves of the ASME Class of the AFW manual isolation valves were identified by SMUD that did not receive periodic preventive maintenance. One valve manufacturer recommends lubrication checks at six-month intervals and actuations (if only partial) on a monthly basis. It is estimated that 4 man-hours will be expended annually per valve performing preventive maintenance and actuation. Assuming that 100 valves are involved, 400 man-hours will be expended each year at each reactor maintaining this class of manual valves. At $35/hour for maintenance personnel,[1009] the direct maintenance cost amounts to $14,000/RY. In addition, assuming 20 hours/RY of additional supervisory time at $45/hour will be directed towards added valve maintenance results in $900 of increased costs. Further, assuming an added $100 for additional administrative costs, the total cost for added valve maintenance will be $15,000/RY. Assuming a 30-year plant life and a 5% discount rate, the lifetime plant costs associated with the added maintenance of manual valves would be approximately $230,000.

NRC Cost: The NRC cost is estimated to be similar to that incurred in processing a NUREG-0737[99] multiplant action ($6,000).[1009]

Value/Impact Assessment

Due to the inability to ascertain the expected reduction in public risk, a value/impact score was not calculated; however, the risk from this issue was judged to be very low.

Other Considerations

Due to the low costs associated with maintaining the manual isolation valves, it would appear to be cost effective for plant operators to maintain them as a good practice and not require a regulatory requirement. The power replacement cost for one day of plant outage which may result from the inability to isolate would pay the plant life costs for isolation valve maintenance. In view of this cost saving potential, the release of the Information Notice may resolve this issue.

CONCLUSION

Due to the minimal estimated reduction in public risk resulting from the resolution of this issue, a priority classification category of LOW is assigned.

REFERENCES

366. NUREG/CR-2787, "Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One - Unit One Nuclear Power Plant," U.S. Nuclear Regulatory Commission, June 1982.

1006. NUREG-1195, "Loss of Integrated Control System Power and Overcooling Transient at Rancho Seco on December 26, 1985," U.S. Nuclear Regulatory Commission, February 1986.

1007. Memorandum for T. Speis from F. Miraglia, "Generic Action as a Result of the Rancho Seco Event of December 26, 1985," May 14, 1986.

1008. Memorandum for E. Jordan from G. Holahan, "Proposed IE Information Notice," June 6, 1986.

1009. NUREG/CR-4568, "A Handbook for Quick Cost Estimates," U.S. Nuclear Regulatory Commission, April 1986.

1010. IE Information Notice No. 86-61, "Failure of Auxiliary Feedwater Manual Isolation Valve," U.S. Nuclear Regulatory Commission, July 28, 1986.

ISSUE 130: ESSENTIAL SERVICE WATER PUMP FAILURES AT MULTIPLANT SITES

DESCRIPTION

Historical Background

This issue was identified[958] as a result of the Byron Unit 1 vulnerability to core-melt sequences in the absence of the availability (not yet operational) of Byron Unit 2. Because of the licensing status of the multiplant configuration of Byron Units 1 and 2, the more immediate need to make a third service water pump available to Byron Unit 1, via a crosstie with one of the two Byron Unit 2 essential service water (ESW) pumps, the Byron Unit 1 concern is classified as a plant-specific (not generic) issue. However, the Byron plant-specific issue raised concerns relative to multiplant units that have only two ESW pumps/plant with crosstie capabilities. When Byron Unit 2 becomes operational, the Byron Units will be similar to this limited group of multiplant configurations.

The report[958] also contained a limited survey of W plants to help identify the generic applicability of multiplant configuration vulnerabilities with only 2 ESW pumps/plant. In the multiplant configurations identified (approximately 16 plants), all plants can share ESW pumps via crosstie between plants. It was stated[958] that B&W and CE plants would be surveyed to identify if similar multiplant configurations with 2 ESW pumps/plant and crosstie capabilities exist in the other NSSS vendors' designs.

Based on the above limited survey, this issue may affect at least 16 PWR plants. Single unit plants should also be surveyed to identify if similar ESW vulnerabilities exist.

Safety Significance

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

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

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

Possible Solutions

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

PRIORITY DETERMINATION

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

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

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

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

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

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

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

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

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

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

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

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

The cutsets (systemic event sequences) for the above loss of service water transient in Plant A (T_a), with Plant B operating (U_{bo}) are:

$$T_aU_{bo} \begin{array}{l} X(L+P+Q) \\ XS \\ (W_1+W_2) \end{array} = (5 \times 10^{-4}) \begin{array}{l} 3 \times 10^{-4} \\ 3 \times 10^{-4} \\ 2 \times 10^{-2} \end{array} = \begin{array}{l} 1.5 \times 10^{-7} \\ 1.5 \times 10^{-7} \\ 1.0 \times 10^{-5} \end{array}$$

and with Plant B in shutdown or refueling (U_{br})

$$T_aU_{br} \begin{array}{l} X(L+P+Q) \\ XS \\ (W_3+W_4) \end{array} = (5 \times 10^{-4}) \begin{array}{l} 3 \times 10^{-4} \\ 3 \times 10^{-4} \\ 2.5 \times 10^{-1} \end{array} = \begin{array}{l} 1.5 \times 10^{-7} \\ 1.5 \times 10^{-7} \\ 1.3 \times 10^{-4} \end{array}$$

The base-case frequencies for the cutsets shown above are:

$T_a = 10^{-3}$/RY	$W_3 = 0.25$
$U_{bo} = 5 \times 10^{-1}$	$W_4 = 2 \times 10^{-3}$
$U_{br} = 5 \times 10^{-1}$	$X = 3 \times 10^{-2}$
$W_1 = 10^{-2}$	$S = 10^{-2}$
$W_2 = 7 \times 10^{-3}$	

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

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

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

Frequency Estimate

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

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

Based on the above, TS requirements on the ESW pumps while plants are in Modes 5 and 6 may provide a reduction in core-melt frequency of approximately 10^{-4}/RY for the operational plant at a two-unit multiplant site.

When both plants are operating, the dominant core-melt frequency from a ESW transient (T) is estimated at 10^{-5}/RY. Improvements in valve realignments (crosstie) procedures are not estimated herein to contribute significantly to core-melt frequency, but the resolution of this issue should reexamine the need for TS or procedures for these crosstie operations. It also appears that changes to the ESW TS in Modes 1, 2, 3, and 4 would not provide significant reductions in the plants' core-melt frequency.

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

Consequence Estimate

As shown above, the two-unit multiplant configurations with only 2 ESW pumps/unit may have a core-melt frequency reduction potential (CM) on the order of 10^{-5}/RY when both units are running, or 10^{-4}/RY when one unit is running and the other is shut down. Because the indicated remedies for each dominant core-melt frequency are significantly different in scope and costs to implement, the risks are calculated separately.

In each case however, the estimated core-melt frequency is predicated on the potential unavailability of the ESW pumps in the companion unit of the multiplant configuration. The crosstie configurations and capability of the plant operators to realign the valves in the crosstie configurations are not estimated to be as significant an impediment to success in reducing core-melt frequency.

It is also estimated that recovery of the out-of-service ESW pumps cannot be assured in time to preclude a core-melt. We recognize that equipment such as the Screen Wash Pumps (non-safety grade) might provide alternate means of service water cooling. However, alternate equipment and its use in these situations will be highly plant-specfic.

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

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

Public Risk Parameters

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

Cost Estimate

Three cost estimates are provided for this issue. The first cost estimate considers the industry and NRC costs associated with the addition of a third pump per plant in a multiplant configuration.

The estimated cost of the third pump/plant is also considered applicable to the costs of a swing pump between the 2 plants. In this second option, the costs of the swing pump can be shared between the 2 plants. This significantly lowers the per plant costs in a multiplant configuration.

The third option involves modified TS on the LCOs for the ESW pumps. The analysis presented herein pertains to TS LCOs on the ESW pumps in Modes 5 and 6. However, the TS for all modes of operation should be reviewed for adequacy and updated accordingly. It is also expected that Options 1 and 2 stated above might require additional TS.

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

The industry costs to prepare the TS is estimated at $16,000/plant.[961] This estimate includes 8 man-weeks of utility technical, legal, management, and committee input.

The total industry cost per option per plant are:

Additional ESW pump/plant plus TS	= $15M
Additional Swing pump/plant	= $7.5M
TS/plant	= $0.016M

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

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

The NRC cost per plant is based on cost estimates given in NUREG/CR-4627.[961] This cost includes 6 staff-weeks of NRC technical staff, and three weeks for management and legal reviews and concurrences. Based on a rate of $50.00/staff-hour, the NRC costs are estimated at $18,000 per TS change/plant. Considering that possibly two Federal Register notices will be required ($800), the total NRC cost is estimated at approximately $19,000/plant. The total NRC cost, including the generic review costs distributed over the affected plants and the plant-specific TS costs, amounts to a total NRC cost of $32,000/plant.

The above NRC costs are applicable to each of the three options discussed in this analysis.

Value/Impact Assessment

Three value/impact assessments are calculated for this issue. The estimated risk reduction that may result from installing a third ESW pump/plant, or an

ESW swing pump per 2-unit multiplant configuration, is 755 man-rem/plant when both plants are in operation.

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

The estimated total industry and NRC cost for the above three conditions and options are \$15M, \$7.5M, and \$0.05M respectively. The value/impact scores are therefore:

ESW/plant: S = 50 man-rem/\$M

ESW/2-plants: S = 100 man-rem/\$M

TS/Modes 5, 6: $S = 2 \times 10^5$ man-rem/\$M

Other Considerations

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

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

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

Consideration and coordinaton with other ongoing staff actions such as USI A-45 (Shutdown Decay Heat Removal Requirements) is recommended. In this regard, USI A-45 is considering an independent and add-on dedicated decay heat removal system (ADHRS). Inclusive in the ADHRS is an additional service water pump. Therefore, if the proposed USI A-45 ADHRS is approved by the Commission, the ADHRS may remedy potential problems that may exist because of ESW vulnerabilities.

The need for requirements on cross-tie operations and ESW Technical Specifications in Modes 5 and 6 is identified herein as potentially significant in reducing public risk and is determined to be potentially cost effective. In this regard, we recommend that resolution of this issue be coordinated with the Technical Specifications Branch in NRR.

CONCLUSION

Based on this evaluation and other considerations described above, we recommend that resolution of this issue be ranked as HIGH priority.

REFERENCES

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

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

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

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

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

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

DESCRIPTION

Historical Background

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

In February 1982, NRC issued a policy statement on "Nuclear Power Plant Staff Working Hours" (47 FR 7352).[977] Based on public comments, the policy statement was revised and reissued in June 1982 (47 FR 23836).[978] Generic Letter 82-12[979] transmitted this version of the Policy Statement to OLs and CPs along with instructions to revise administrative procedures in Technical Specifications to conform to the policy statement. Guidance on incorporating limits on overtime into Technical Specifications were later issued in Generic Letters 82-16[980] and 83-02[981] to PWRs and BWRs, respectively. In March 1983, Generic Letter 83-14[982] was issued to clarify the definition of "Key Maintenance Personnel" stated in Generic Letter 82-12. In September 1985, the NRC staff was directed[983] to update the policy statement on "Nuclear Power Plant Staff Working Hours."

Since Commission policy is stated in several documents, revision of NRC's present policy guidance on limits on overtime and shift scheduling is needed to consolidate current guidance into a single document.

Safety Significance

The current Policy Statement and implementing documents are adequate from a safety perspective in that the amount of overtime worked by nuclear power plant personnel has not been identified as an actual contributor to reportable events nor has it degraded the safety of plant operations. However, one specific area of guidance relating to the use of 12-hour shifts is absent from the current policy statement. The staff has reviewed and approved on a case-by-case basis licensee programs for routine 12-hour shifts (e.g., Duke Power's Oconee Station and Union Electric's Callaway Station).

Possible Solution

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

The proposed Policy Statement is unchanged from current practice with respect to administrative procedures to prevent personnel who perform safety-related functions from working in a fatigued condition: (1) during normal operations, no more than 16 hours in a 24-hour period, 24 hours in a 48-hour period, or 72 hours in any 7-day period; and (2) control overtime on an individual basis with management approval of deviations from recommended limits on working hours.

CONCLUSION

A revised Policy Statement on shift and scheduling and hours of work would eliminate licensee confusion resulting from multiple policy and requirement documents and would clearly identify licensee management's responsibility to assure that nuclear power plant staff fatigue resulting from excessive working hours does not adversely affect public health or safety. The revised Policy Statement will aid the NRC staff in conducting reviews of licensee programs and in monitoring licensee implementation. Hence, this issue is categorized as a Licensing Issue.

REFERENCES

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

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

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

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

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

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

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

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

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

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

ISSUE 134: RULE ON DEGREE AND EXPERIENCE REQUIREMENT

DESCRIPTION

Historical Background

The Policy Statement on Engineering Expertise on Shift[996] is satisfied by providing engineering and accident management expertise on shift through the separate Shift Technical Advisor (STA) or through the combined STA/Senior Operator roles. The contemplated rule would further upgrade the levels of engineering and accident management expertise on shift by requiring Senior Operators (SOs) hold baccalaureate degrees in engineering or physical science. The contemplated rule is the result of the Commission's Staff Requirements Memorandum dated January 23, 1986 (COMLZ-85-6), directing the staff to prepare an Advance Notice of Proposed Rulemaking (ANPRM). In April 1987, issues and proposed options concerning the degree requirements for SOs were presented to the Commission in SECY-87-101.[1042]

Safety Significance

This rulemaking is being contemplated to further ensure the protection of the health and safety of the public by having personnel on shift with enhanced qualifications. A regulatory analysis will be completed after the public comments from the ANPRM.

Possible Solution

The rule under consideration would require after January 1, 1991, that applicants for licenses as SOs of a nuclear power plant hold a baccalaureate degree in engineering or physical science from an accredited institution. For candidates with a baccalaureate degree, the current requirement of two years of nuclear power plant experience would be amended to require at least one of the two years of operating experience be with a similar commercial nuclear reactor operating at greater than 20% power. SOs licensed prior to January 1, 1991, who do not hold degrees in engineering or physical science would be "grandfathered." Only one reexamination would be allowed for applicants who apply before January 1, 1991. No degree equivalency would be acceptable after January 1, 1991.

CONCLUSION

This issue is a HIGH priority safety issue based upon Commission direction to prepare an ANPRM on degreed SOs.

REFERENCES

996. Federal Register Notice 50 FR 43621, "Commission Policy Statement on Engineering Expertise on Shift," October 28, 1985.

1042. SECY-87-101, "Issues and Proposed Options Concerning Degree Requirement for Senior Operators," April 16, 1987.

ISSUE 135: STEAM GENERATOR AND STEAM LINE OVERFILL

DESCRIPTION

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

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

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

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

TASK 3: Reassess the following pending issues of Issue 67 for potential inclusion in an integrated resolution: reassessment of radiological consequences, reevaluation of design basis SGTR, supplemental tube inspections, integrity of steam generator tube sleeves, denting criteria, improved accident monitoring, reactor vessel inventory measurement, reactor coolant pump (RCP) trip, control room design review, emergency operating procedures, organizational responses, and reactor coolant system pressure control.

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

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

CONCLUSION

This issue has a MEDIUM priority ranking.

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TASK-HF3: OPERATOR LICENSING EXAMINATIONS

One purpose of this task is to ensure that the licensing examination for reactor operators and senior operators is a valid measure of the operator's knowledge and ability to perform the necessary tasks and functions required to safely operate and control commercial nuclear power plants. The second purpose is to ensure that examinations are administered in a consistent manner by the various NRC examiners to enhance reliability and efficiency. The intent is to perform these modifications to the examination process without unnecessary impact on current license candidates and training programs. This task was identified as five distinct items in Table 7 of the NRC 1985 Annual Report (Items 3.1, 3.2, 3.3, 3.4, and 3.5). The following is a discussion of these five items.

ITEM HF3.1: DEVELOP JOB KNOWLEDGE CATALOG

DESCRIPTION

A catalog of the reactor operator and senior operator tasks and duties and the required knowledge, skills, and abilities necessary for safe performance will be formulated using available generic job and task analyses. A computerized bank of examination questions for use in test construction and examination validation will be developed and updated using this catalog. Additionally, test specifications will be developed for licensing examinations to provide examination plans which outline the necessary types of knowledge required to be assessed during examinations. An evaluation of the feasibility of identifying or developing on-the-job performance measures which can be used in assessing the ability of the examination process to predict operator performance will be conducted. Long-term examination development/validation strategies will be developed based upon the results of current examination modifications and content validation.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

This Licensing Issue was resolved with the issuance of a Supplement 1 to the Knowledge and Abilities Catalog (NUREG-1122)[974] in April 1987.

ITEM HF3.2: DEVELOP LICENSE EXAMINATION HANDBOOK

DESCRIPTION

To increase the efficiency, reliability, and validity of the licensing examination process, DHFT will evaluate new examination procedures. These new procedures will take into consideration the problems and issues associated with the current examination process from the examiner, candidate, and utility perspectives. The examination process and practices of similar applicable agencies and organizations will be reviewed. The input from industry training staff and reactor operators regarding problems or issues underlying the current licensing examinations will be solicited. The results will be the identification of improvements to optimize the format and procedures relating to written, oral, and simulator examinations. From this identification activity, standardized examination practices and guidelines will be developed. The test examiners will also be trained on test development, administration, and grading techniques to assure consistency and reliability.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

This Licensing Issue was resolved with the issuance of Revision 4 to the Examiners Handbook (NUREG-1021)[962] in May 1987.

ITEM HF3.3: DEVELOP CRITERIA FOR NUCLEAR POWER PLANT SIMULATORS

DESCRIPTION

The TMI Action Plan identified a need for upgrading of training simulator standards, and for a regulatory guide giving criteria for acceptability. It also called for a review of simulators to assure their conformance to criteria. Work underway under this item will provide methods and criteria for evaluating the adequacy of nuclear power plant simulators for use in conducting operating examinations under proposed 10 CFR 55.45. The qualifications required of personnel who perform these evaluations will be identified.

CONCLUSION

This item is covered in TMI Action Plan Item I.A.4.2(4).

ITEM HF3.4: EXAMINATION REQUIREMENTS

DESCRIPTION

This item called for a revision to 10 CFR 55 to reflect changes in operator licensing examinations and the examination process. A revision to Regulatory Guide 1.149[439] is also to be accomplished.

CONCLUSION

This item is covered in TMI Action Plan Item I.A.2.6(1).

ITEM HF3.5: DEVELOP COMPUTERIZED EXAM SYSTEM

DESCRIPTION

As part of the examination process effort relative to Items HF3.1 and HF3.2, this item calls for the staff to develop a computerized exam system.

This item is related to increasing knowledge, certainty, and understanding of safety issues in order to increase confidence in assessing levels of safety and is, therefore, considered a licensing issue.

CONCLUSION

In December 1984, DHFT reported that the Examination Question Bank was fully operational and was available for routine access by all examiners.[997] Thus, this Licensing Issue has been resolved.

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The report presents the priority rankings for generic safety issues related to nuclear power plants. The purpose of these rankings is to assist in the timely and efficient allocation of NRC resources for the resolution of those safety issues that have a significant potential for reducing risk. The safety priority rankings are HIGH, MEDIUM, LOW, and DROP and have been assigned on the basis of risk significance estimates, the ratio of risk to costs and other impacts estimated to result if resolutions of the safety issues were implemented, and the consideration of uncertainties and other quantitative or qualitative factors. To the extent practical, estimates are quantitative.

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