UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

JULY 2007

SUPPLEMENT 31 TO NUREG-0933
"A PRIORITIZATION OF GENERIC SAFETY ISSUES"

REVISION INSERTION INSTRUCTIONS

	Remove	Insert
Introduction:	pp. 27 to 68, Rev. 30	pp. 27 to 68, Rev. 31
Section 3:	pp. 3.196-1 to 6	pp. 3.196-1 to 6, Rev. 1
	-	pp. 3.198-1 to 12
	-	pp. 3.200-1 to 3
	-	pp. 3.201-1 to 7
	-	pp. 3.202-1 to 3
	-	pp. 3.203-1 to 2
References:	pp. R-1 to R-126, Rev. 20	pp. R-1 to R-129, Rev. 21
Appendix B	pp. A.B-1 to 13, Rev. 21	pp. A.B-1 to 13, Rev. 22
Appendix F	pp. A.F.0-1 to 3, Rev. 4	pp. A.F.0-1 to 3, Rev. 5
	p. A.F.14-1	p. A.F.14-1, Rev. 1
	p. A.F.16-1	p. A.F.16-1, Rev. 1

Paperwork Reduction Act Statement
This NUREG contains information collection requirements that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget, approval number 3150-0011 and 3150-0132.

Public Protection Notification
The NRC may not conduct or sponsor, and a person is not required to respond to, a request for information or an information collection requirement unless the requesting document displays a currently valid OMB control number.

AVAILABILITY OF REFERENCE MATERIALS IN NRC PUBLICATIONS

NRC Reference Material

As of November 1999, you may electronically access NUREG-series publications and other NRC records at NRC's Public Electronic Reading Room at http://www.nrc.gov/reading-rm.html. Publicly released records include, to name a few, NUREG-series publications; *Federal Register* notices; applicant, licensee, and vendor documents and correspondence; NRC correspondence and internal memoranda; bulletins and information notices; inspection and investigative reports; licensee event reports; and Commission papers and their attachments.

NRC publications in the NUREG series, NRC regulations, and *Title 10, Energy*, in the Code of *Federal Regulations* may also be purchased from one of these two sources.

1. The Superintendent of Documents
   U.S. Government Printing Office
   Mail Stop SSOP
   Washington, DC 20402–0001
   Internet: bookstore.gpo.gov
   Telephone: 202-512-1800
   Fax: 202-512-2250
2. The National Technical Information Service
   Springfield, VA 22161–0002
   www.ntis.gov
   1–800–553–6847 or, locally, 703–605–6000

A single copy of each NRC draft report for comment is available free, to the extent of supply, upon written request as follows:

Address: U.S. Nuclear Regulatory Commission
Office of Administration
Mail, Distribution and Messenger Team
Washington, DC 20555-0001
E-mail: DISTRIBUTION@nrc.gov
Facsimile: 301–415–2289

Some publications in the NUREG series that are posted at NRC's Web site address http://www.nrc.gov/reading-rm/doc-collections/nuregs are updated periodically and may differ from the last printed version. Although references to material found on a Web site bear the date the material was accessed, the material available on the date cited may subsequently be removed from the site.

Non-NRC Reference Material

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions, *Federal Register* notices, Federal and State legislation, and congressional reports. Such documents as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings may be purchased from their sponsoring organization.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at—

The NRC Technical Library
Two White Flint North
11545 Rockville Pike
Rockville, MD 20852–2738

These standards are available in the library for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from—

American National Standards Institute
11 West 42nd Street
New York, NY 10036–8002
www.ansi.org
212–642–4900

Legally binding regulatory requirements are stated only in laws; NRC regulations; licenses, including technical specifications; or orders, not in NUREG-series publications. The views expressed in contractor-prepared publications in this series are not necessarily those of the NRC.

The NUREG series comprises (1) technical and administrative reports and books prepared by the staff (NUREG–XXXX) or agency contractors (NUREG/CR–XXXX), (2) proceedings of conferences (NUREG/CP–XXXX), (3) reports resulting from international agreements (NUREG/IA–XXXX), (4) brochures (NUREG/BR–XXXX), and (5) compilations of legal decisions and orders of the Commission and Atomic and Safety Licensing Boards and of Directors' decisions under Section 2.206 of NRC's regulations (NUREG–0750).

TABLE II

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

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

Legend

NOTES:	1 - Possible Resolution Identified for Evaluation
	2 - Resolution Available (Documented in NUREG, NRC Memorandum, SER, or equivalent)
	3 - Resolution Resulted in either: (a) The Establishment of New Regulatory Requirements (By Rule, SRP Change, or equivalent) or (b) No New Requirements
	4 - Issue to be Prioritized in the Future
	5 - Issue that is not a Generic Safety Issue but should be Assigned Resources for Completion
HIGH	- High Safety Priority
MEDIUM	- Medium Safety Priority
LOW	- Low Safety Priority
DROP	- Issue Dropped as a Generic Issue
EI	- Environmental Issue
I	- Resolved TMI Action Plan Item with Implementation of Resolution Mandated by NUREG-0737
LI	- Licensing Issue
MPA	- Multiplant Action
NA	- Not Applicable
RI	- Regulatory Impact Issue
S	- Issue Covered in an NRC Program Outside the Scope of This Document
USI	- Unresolved Safety Issue
Continue	- As defined in NRC Management Directive 6.4[1858]

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
A-33	NEPA Review of Accident Risks	-	NRR/DSI/AEB	EI(NOTE 3)		11/30/83	NA
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents	V'Molen	NRR/DSI/ICSB	II.F.3		11/30/83	NA
A-35	Adequacy of Offsite Power Systems	R. Emrit	NRR/DSI/PSB	NOTE 3(a)	1	12/31/94	B-23
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	R. Emrit	NRR/DSI/GIB	NOTE 3(a)	2	06/30/04	C-10, C-15
A-37	Turbine Missiles	J. Pittman	NRR/DE/MTEB	DROP		11/30/83	NA
A-38	Tornado Missiles	G. Sege	NRR/DSI/ASB	DROP	3	06/30/00	NA
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	
A-40	Seismic Design Criteria (former USI)	R. Emrit	RES/DSIR/EIB	NOTE 3(a)	1	12/31/89	NA
A-41	Long-Term Seismic Program	L. Riani	NRR/DE/MEB	NOTE 3(b)	1	12/31/84	NA
A-42	Pipe Cracks in Boiling Water Reactors (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	06/30/85	B-05
A-43	Containment Emergency Sump Performance (former USI)	R. Emrit	NRR/DST/GIB	NOTE 3(a)	1	12/31/87	
A-44	Station Blackout (former USI)	R. Emrit	RES/DRPS/RPSI	NOTE 3(a)	1	06/30/88	
A-45	Shutdown Decay Heat Removal Requirements (former USI)	R. Emrit	RES/DRPS/RPSI	NOTE 3(b)	1	12/31/88	NA
A-46	Seismic Qualification of Equipment in Operating Plants (former USI)	R. Emrit	NRR/DSRO/EIB	NOTE 3(a)	2	06/30/00	
A-47	Safety Implications of Control Systems (former USI)	R. Emrit	RES/DSIR/EIB	NOTE 3(a)	1	12/31/89	
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	R. Emrit	NRR/DSIR/SAIB	NOTE 3(a)	1	06/30/89	
A-49	Pressurized Thermal Shock (former USI)	R. Emrit	NRR/DSRO/RSIB	NOTE 3(a)	1	12/31/87	A-21
B-1	Environmental Technical Specifications	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
B-2	Forecasting Electricity Demand	-	NRR	EI (NOTE 3)		11/30/83	NA
B-3	Event Categorization	-	NRR/DSI/RSB	LI (NOTE 3)		11/30/83	NA
B-4	ECCS Reliability	R. Emrit	NRR/DSI/RSB	II.E.3.2		11/30/83	NA
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	D. Thatcher	RES/DE/EIB	NOTE 3(b)	1	06/30/88	NA
B-6	Loads, Load Combinations, Stress Limits	J. Pittman	NRR/DSRO/EIB	119.1		12/31/87	NA
B-7	Secondary Accident Consequence Modeling	-	NRR/DSI/AEB	LI (NOTE 3)		11/30/83	NA
B-8	Locking Out of ECCS Power Operated Valves	R. Riggs	NRR/DSI/RSB	DROP	1	12/31/94	NA
B-9	Electrical Cable Penetrations of Containment	R. Emrit	NRR/DSI/PSB	NOTE 3(b)		11/30/83	NA
B-10	Behavior of BWR Mark III Containments	H. Vandermolen	NRR/DSI/CSB	NOTE 3(a)	1	12/31/84	NA
B-11	Subcompartment Standard Problems	-	NRR/DSI/CSB	LI (NOTE 5)		11/30/83	NA
B-12	Containment Cooling Requirements (Non-LOCA)	R. Emrit	NRR/DSI/CSB	NOTE 3(b)	1	12/31/86	NA
B-13	Marviken Test Data Evaluation	-	NRR/DSI/CSB	LI (NOTE 5)		11/30/83	NA
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA	R. Emrit	NRR/DST/GIB	A-48		11/30/83	NA
B-15	CONTEMPT Computer Code Maintenance	-	NRR/DSI/CSB	LI (NOTE 3)		11/30/83	NA
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	R. Emrit	NRR/DE/MEB	A-18		11/30/83	NA

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
C-8	Main Steam Line Leakage Control Systems	W. Milstead	RES/DRPS/RPSI	NOTE 3(b)	1	06/30/90	NA
C-9	RHR Heat Exchanger Tube Failures	V'Molen	NRR/DSI/RSB	DROP		11/30/83	NA
C-10	Effective Operation of Containment Sprays in a LOCA	R. Emrit	NRR/DSI/AEB	NOTE 3(a)		11/30/83	NA
C-11	Assessment of Failure and Reliability of Pumps and Valves	R. Emrit	NRR/DE/MEB	NOTE 3(b)		12/31/85	NA
C-12	Primary System Vibration Assessment	D. Thatcher	NRR/DE/MEB	NOTE 3(b)		11/30/83	NA
C-13	Non-Random Failures	R. Emrit	NRR/DST/GIB	A-17	1	06/30/91	NA
C-14	Storm Surge Model for Coastal Sites	R. Emrit	NRR/DE/EHEB	LI (NOTE 3)		06/30/88	NA
C-15	NUREG Report for Liquid Tank Failure Analysis	-	NRR/DE/EHEB	LI (NOTE 3)		11/30/83	NA
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	-	NRR/DE/EHEB	EI (NOTE 3)		11/30/83	NA
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	R. Emrit	NRR/DSI/METB	NOTE 3(a)		11/30/83	NA
D-1	Advisability of a Seismic Scram	D. Thatcher	RES/DET/MSEB	DROP	1	12/31/98	NA
D-2	Emergency Core Cooling System Capability for Future Plants	R. Emrit	RES/DRA/ARGIB	DROP		12/31/88	NA
D-3	Control Rod Drop Accident	R. Emrit	NRR/DSI/CPB	NOTE 3(b)		11/30/83	NA
		NEW GENERIC ISSUES					
1.	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems	R. Emrit	NRR/DSI/METB	DROP		11/30/83	NA
2.	Failure of Protective Devices on Essential Equipment	Diab	RES/DSIR/EIB	DROP	2	06/30/95	NA
3.	Set Point Drift in Instrumentation	R. Emrit	NRR/DSIR/RPSIB	NOTE 3(b)	1	06/30/86	NA
4.	End-of-Life and Maintenance Criteria	D. Thatcher	NRR/DE/EQB	NOTE 3(b)		11/30/83	NA
5.	Design Check and Audit of Balance-of-Plant Equipment	J. Pittman	NRR/DSI/ASB	I.F.1		11/30/83	NA
6	Separation of Control Rod from Its Drive and BWR High Rod Worth Events	H. Vandermolen	NRR/DSI/CPB	NOTE 3(b)	1	12/31/94	NA
7.	Failures Due to Flow-Induced Vibrations	H. Vandermolen	NRR/DSI/RSB	DROP	1	06/30/91	NA
8.	Inadvertent Actuation of Safety Injection in PWRs	L. Riani	NRR/DSI/RSB	I.C.1		11/30/83	NA
9.	Reevaluation of Reactor Coolant Pump Trip Criteria	R. Emrit	NRR/DSI/RSB	II.K.3(5)		11/30/83	NA
10.	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges	R. Riggs	NRR/DSI/ICSB	DROP		11/30/83	NA
11.	Turbine Disc Cracking	J. Pittman	NRR/DE/MTEB	A-37		11/30/83	NA
12.	BWR Jet Pump Integrity	G. Sege	NRR/DE/MTEB, MEB	NOTE 3(b)	1	12/31/84	NA
13.	Small Break LOCA from Extended Overheating of Pressurizer Heaters	L. Riani	NRR/DSI/RSB	DROP		11/30/83	NA
14.	PWR Pipe Cracks	R. Emrit	NRR/DE/MTEB	NOTE 3(b)	2	12/31/94	NA
15.	Radiation Effects on Reactor Vessel Supports	R. Emrit	RES/DET/EMMEB	NOTE 3(b)	3	06/30/96	NA

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
75.	(Cont.)						B-90, B-91, B-92, B-93
76.	Instrumentation and Control Power Interactions	Zimmerman	RES/DSIR/EIB	DROP	3	06/30/95	NA
77.	Flooding of Safety Equipment Compartments by Back-flow Through Floor Drains	L. Riani	RES/DE/EIB	A-17		12/31/87	NA
78.	Monitoring of Fatigue Transient Limits for Reactor Coolant System	Rourk	RES/DET/GSIB	NOTE 3(b)	3	12/31/97	
79.	Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown	L. Riani	RES/DSIR/EIB	NOTE 3(b)	3	06/30/95	NA
80.	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments	H. Vandermolen	RES/DSARE/REAHFB	NOTE 3(b)	4	06/30/06	NA
81.	Impact of Locked Doors and Barriers on Plant and Personnel Safety	Rourk	RES/DSIR/EIB	LOW	4	06/30/95	NA
82.	Beyond Design Basis Accidents in Spent Fuel Pools	H. Vandermolen	RES/DRPS/RPSI	NOTE 3(b)	3	06/30/04	NA
83.	Control Room Habitability	R. Emrit	RES/DST/AEB	NOTE 3(b)	3	06/30/03	NA
84.	CE PORVs	R. Riggs	RES/DSIR/RPSI	NOTE 3(b)	2	06/30/90	NA
85.	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	W. Milstead	NRR/DSI/CSB	DROP	2	06/30/91	NA
86.	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	R. Emrit	NRR/DEST/EMTB	NOTE 3(a)	1	06/30/88	B-84
87.	Failure of HPCI Steam Line Without Isolation	J. Pittman	RES/DSIR/EIB	NOTE 3(a)	2	06/30/95	
88.	Earthquakes and Emergency Planning	R. Riggs	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
89.	Stiff Pipe Clamps	T.Y. Chang	RES/DSIR/EIB	LOW	2	06/30/95	NA
90.	Technical Specifications for Anticipatory Trips	H. Vandermolen	NRR/DSI/RSB, ICSB	DROP	2	12/31/98	NA
91.	Main Crankshaft Failures in Transamerica DeLaval Emergency Diesel Generators	R. Emrit	RES/DRA/ARGIB	NOTE 3(b)		12/31/87	NA
92.	Fuel Crumbling During LOCA	H. Vandermolen	NRR/DSI/RSB, CPB	DROP	1	12/31/98	NA
93.	Steam Binding of Auxiliary Feedwater Pumps	J. Pittman	RES/DRPS/RPSI	NOTE 3(a)		06/30/88	B-98
94.	Additional Low Temperature Overpressure Protection for Light Water Reactors	J. Pittman	RES/DSIR/RPSI	NOTE 3(a)		06/30/90	
95.	Loss of Effective Volume for Containment Recirculation Spray	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)		06/30/90	NA
96.	RHR Suction Valve Testing	W. Milstead	RES/DRA/ARGIB	105		06/30/90	NA
97.	PWR Reactor Cavity Uncontrolled Exposures	H. Vandermolen	NRR/DSI/RAB	III.D.3.1		06/30/85	NA
98.	CRD Accumulator Check Valve Leakage	J. Pittman	NRR/DSI/ASB	DROP		06/30/85	NA
99.	RCS/RHR Suction Line Valve Interlock on PWRs	J. Pittman	RES/DRPS/RPSI	NOTE 3(a)	3	06/30/91	L-817
100.	Once-Through Steam Generator Level	J. Jackson	RES/DSIR/EIB	DROP	1	06/30/95	NA

Table II (Continued)

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

Table II (Continued)

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

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
	of All Feedwater						
125.II.7	Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break	H. Vandermolen	RES/DRPS/RPSI	NOTE 3(b)	7	12/31/98	NA
125.II.8	Reassess Criteria for Feed-and-Bleed Initiation	H. Vandermolen	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.9	Enhanced Feed-and-Bleed Capability	H. Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
125.II.10	Hierarchy of Impromptu Operator Actions	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.11	Recovery of Main Feedwater as Alternative to Auxiliary Feedwater	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.12	Adequacy of Training Regarding PORV Operation	R. Riggs	RES/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.13	Operator Job Aids	J. Pittman	NRR/DRA/ARGIB	DROP	7	12/31/98	NA
125.II.14	Remote Operation of Equipment Which Must Now Be Operated Locally	H. Vandermolen	NRR/DSRO/SPEB	DROP	7	12/31/98	NA
126.	Reliability of PWR Main Steam Safety Valves	R. Riggs	RES/DRA/ARGIB	LI (NOTE 3)		06/30/88	NA
127.	Maintenance and Testing of Manual Valves in Safety-Related Systems	J. Pittman	RES/DRA/ARGIB	LOW		12/31/87	NA
128.	Electrical Power Reliability	R. Emrit	RES/DSIR/EIB	NOTE 3(a)	2	06/30/95	
129.	Valve Interlocks to Prevent Vessel Drainage During Shutdown Cooling	W. Milstead	RES/DRA/ARGIB	DROP		06/30/90	NA
130.	Essential Service Water Pump Failures at Multiplant Sites	R. Riggs	RES/DSIR/RPSIB	NOTE 3(a)	2	12/31/95	
131.	Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse-Designed Plants	R. Riggs	RES/DRA/ARGIB	S	1	06/30/91	NA
132.	RHR System Inside Containment	Su	RES/DSIR/SAIB	DROP	1	12/31/95	NA
133.	Update Policy Statement on Nuclear Plant Staff Working Hours	J. Pittman	NRR/DLPQ/LHFB	LI (NOTE 3)	1	12/31/91	NA
134.	Rule on Degree and Experience Requirement	J. Pittman	RES/DRA/RDB	NOTE 3(b)		12/31/89	NA
135.	Steam Generator and Steam Line Overfill	R. Emrit	RES/DSIR/EIB	NOTE 3(b)	3	06/30/95	NA
136.	Storage and Use of Large Quantities of Cryogenic Combustibles On Site	W. Milstead	RES/DRA/ARGIB	LI (NOTE 3)		06/30/88	NA
137.	Refueling Cavity Seal Failure	W. Milstead	RES/DRA/ARGIB	DROP		06/30/90	NA
138.	Deinerting of BWR Mark I and II Containments During Power Operations Upon Discovery of RCS Leakage or a Train of a Safety System Inoperable	W. Milstead	RES/DSIR/SAIB	DROP	2	12/31/98	NA
139.	Thinning of Carbon Steel Piping in LWRs	R. Riggs	RES/DRA/ARGIB	RI (NOTE 3)	1	06/30/95	NA
140.	Fission Product Removal Systems	R. Riggs	RES/DRA/ARGIB	DROP		06/30/90	NA
141.	Large-Break LOCA With Consequential SGTR	R. Riggs	RES/DRA/ARGIB	DROP		06/30/90	NA
142.	Leakage Through Electrical Isolators in Instrumentation Circuits	W. Milstead	RES/DSIR/EIB	NOTE 3(b)	4	12/31/97	NA
143.	Availability of Chilled Water Systems and Room Cooling	W. Milstead	RES/DRA/ARGIB	NOTE 3(b)	2	06/30/95	NA
144.	Scram Without a Turbine/Generator Trip	Hrabal	RES/DSIR/EIB	DROP	2	12/31/98	NA

Table II (Continued)

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

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
156.3.8	Shared Systems	R. Emrit	RES/DSIR/EIB	DROP	7	06/30/01	NA
156.4.1	RPS and ESFS Isolation	R. Emrit	RES/DSIR/EIB	142	7	06/30/01	NA
156.4.2	Testing of the RPS and ESFS	T.Y. Chang	RES/DSIR/SAIB	120	7	06/30/01	NA
156.6.1	Pipe Break Effects on Systems and Components.	J. Page	RES/DET/GSIB	HIGH	7	06/30/01	
157.	Containment Performance	J. Shaperow	RES/DSIR/SAIB	NOTE 3(b)		06/30/95	NA
158.	Performance of Power-Operated Valves Under Design Basis Conditions	C. Hrabal	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
159.	Qualification of Safety-Related Pumps While Running on Minimum Flow	N. Su	RES/DSIR/SAIB	DROP	1	06/30/95	NA
160.	Spurious Actions of Instrumentation Upon Restoration of Power	C. Rourk	RES/DSIR/EIB	DROP	1	06/30/95	NA
161.	Use of Non-Safety-Related Power Supplies in Safety-Related Circuits	C. Rourk	RES/DSIR/EIB	DROP	1	06/30/95	NA
162.	Inadequate Technical Specifications for Shared Systems at Multiplant Sites When One Unit Is Shut Down	U. Cheh	RES/DSIR/SAIB	DROP	1	06/30/95	NA
163.	Multiple Steam Generator Tube Leakage	Coffman	RES/DET/GSIB	HIGH		12/31/97	
164.	Neutron Fluence in Reactor Vessel	R. Emrit	RES/DSIR/EIB	DROP	1	06/30/95	NA
165.	Safety and Safety/Relief Valve Reliability	C. Hrabal	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
166.	Adequacy of Fatigue Life of Metal Components	R. Emrit	NRR/DE/EMEB	NOTE 3(b)	2	12/31/97	NA
167.	Hydrogen Storage Facility Separation	G. Burdick	RES/DSIR/SAIB	LOW	1	06/30/95	NA
168.	Environmental Qualification of Electrical Equipment	R. Emrit	NRR/DSSA/SPLB	NOTE 3(b)	3	06/30/04	NA
169.	BWR MSIV Common Mode Failure Due to Loss of Accumulator Pressure	R. Emrit	RES/DET/GSIB	DROP	1	06/30/00	NA
170.	Fuel Damage Criteria for High Burnup Fuel	R. Emrit	RES/DET/GSIB	NOTE 3(b)	2	06/30/01	NA
171.	ESF Failure from LOOP Subsequent to a LOCA	C. Rourk	RES/DET/GSIB	NOTE 3(b)	1	12/31/98	NA
172.	Multiple System Responses Program	R. Emrit	RES/DET/GSIB	NOTE 3(b)	2	06/30/02	NA
173.	Spent Fuel Storage Pool	-	-				
173.A	Operating Facilities	R. Emrit	RES/DET/GSIB	NOTE 3(b)	4	06/30/02	NA
173.B	Permanently Shutdown Facilities	R. Emrit	RES/DET/GSIB	NOTE 3(b)	4	06/30/02	NA
174.	Fastener Gaging Practices	-	-				
174.A	SONGS Employees' Concern	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
174.B	Johnson Gage Company Concern	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
175.	Nuclear Power Plant Shift Staffing	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
176.	Loss of Fill-Oil in Rosemount Transmitters	R. Emrit	RES/DET/GSIB	NOTE 3(b)	1	06/30/00	NA
177.	Vehicle Intrusion at TMI	R. Emrit	RES/DET/GSIB	NOTE 3(a)	1	06/30/00	NA
178.	Effect of Hurricane Andrew on Turkey Point	R. Emrit	RES/DET/GSIB	LI (NOTE 3)	2	06/30/00	
179.	Core Performance	R. Emrit	RES/DET/GSIB	LI (NOTE 5)	1	06/30/00	
180.	Notice of Enforcement Discretion	R. Emrit	RES/DET/GSIB	LI (NOTE 3)	1	06/30/00	
181.	Fire Protection	R. Emrit	RES/DET/GSIB	LI (NOTE 5)	1	06/30/00	
182.	General Electric Extended Power Uprate	R. Emrit	RES/DET/GSIB	RI (NOTE 5)	1	06/30/00	

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
183.	Cycle-Specific Parameter Limits in Technical Specifications	R. Emrit	RES/DET/GSIB	RI (NOTE 3)	2	06/30/00	
184.	Endangered Species	R. Emrit	RES/DET/GSIB	EI (NOTE 5)	1	06/30/00	
185.	Control of Recriticality Following Small-Break LOCA In PWRs	H. Vandermolen	RES/DSARE/REAHFB	NOTE 3(b)	1	06/30/06	NA
186.	Potential Risk and Consequences of Heavy Load Drops	R. Lloyd	RES/DSARE/REAHFB	CONTINUE		06/30/04	
187.	The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump in Nuclear Power Plants	H. Vandermolen	RES/DSARE/REAHFB	DROP		06/30/01	NA
188.	Steam Generator Tube Leaks/Ruptures Concurrent with Containment Bypass	H. Vandermolen	RES/DSARE/REAHFB	NOTE 3(b)	1	06/30/06	NA
189.	Susceptibility of Ice Condenser Containments to Early Failure from Hydogen Combustion During A Severe Accident	H. Vandermolen	RES/DSARE/REAHFB	CONTINUE		06/30/02	
190.	Fatigue Evaluation of Metal Components for 60-Year Plant Life	S. Shaukat	RES/DET/GSIB	NOTE 3(b)	2	06/30/00	NA
191.	Assessment of Debris Accumulation on PWR Sump Performance	M. Marshall	RES/DET/GSIB	HIGH	1	12/31/98	
192.	Secondary Containment Drawdown Time	H. Vandermolen	RES/DSARE/REAHFB	DROP		06/30/03	NA
193.	BWR ECCS Suction Concerns	H. Vandermolen	RES/DSARE/REAHFB	CONTINUE		06/30/04	
194.	Implications of Updated Probabilistic Seismic Hazard Estimates	D. Harrison	NRR/DSSA/SPSB	DROP		06/30/04	NA
195.	Hydrogen Combustion in Foreign BWR Piping	H. Vandermolen	RES/DSARE/REAHFB	DROP		06/30/04	NA
196.	Boral Degradation	H. Vandermolen	RES/DSARE/ARREB	NOTE 3(b)	1	06/30/07	NA
197.	Iodine Spiking Phenomena	H. Vandermolen	RES/DSARE/ARREB	DROP		06/30/06	NA
198.	Hydrogen Combustion in PWR Piping	H. Vandermolen	RES/DRASP/OERA	DROP		06/30/07	NA
199.	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States	T. Mitts	RES/DRASP/OERA	NOTE 4		(Later)	
200.	Tin Whiskers	C. Antonescu	RES/DRASP/OERA	DROP		06/30/07	NA
201.	Small-Break LOCA and Loss of Offsite Power Scenario	A. Salomon	RES/DRASP/OERA	DROP		06/30/07	NA
202.	Spent Fuel Pool Leakage Limits	T. Mitts	RES/DRASP/OERA	DROP		06/30/07	NA
203.	Potential Safety Issues with Cranes that Lift Spent Fuel Casks	T. Mitts	RES/DRASP/OERA	DROP		06/30/07	NA
			HUMAN FACTORS ISSUES				
HF1	STAFFING AND QUALIFICATIONS						
HF1.1	Shift Staffing	J. Pittman	RES/DRPS/RHFB	NOTE 3(a)	2	06/30/89	
HF1.2	Engineering Expertise on Shift	J. Pittman	NRR/DHFT/HFIB	NOTE 3(b)	2	06/30/89	NA
HF1.3	Guidance on Limits and Conditions of Shift Work	J. Pittman	NRR/DHFT/HFIB	NOTE 3(b)	2	06/30/89	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
HF2	TRAINING						
HF2.1	Evaluate Industry Training	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF2.2	Evaluate INPO Accreditation	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF2.3	Revise SRP Section 13.2	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF3	OPERATOR LICENSING EXAMINATIONS						
HF3.1	Develop Job Knowledge Catalog	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF3.2	Develop License Examination Handbook	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF3.3	Develop Criteria for Nuclear Power Plant Simulators	J. Pittman	NRR/DHFT/HFIB	I.A.4.2(4)	2	12/31/87	NA
HF3.4	Examination Requirements	J. Pittman	NRR/DHFT/HFIB	I.A.2.6(1)	2	12/31/87	NA
HF3.5	Develop Computerized Exam System	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 3)	2	12/31/87	NA
HF4	PROCEDURES						
HF4.1	Inspection Procedure for Upgraded Emergency Operating Procedures	J. Pittman	NRR/DLPQ/LHFB	NOTE 3(b)	6	06/30/95	NA
HF4.2	Procedures Generation Package Effectiveness Evaluation	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	6	06/30/95	NA
HF4.3	Criteria for Safety-Related Operator Actions	J. Pittman	NRR/DHFT/HFIB	B-17	6	06/30/95	NA
HF4.4	Guidelines for Upgrading Other Procedures	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	6	06/30/95	NA
HF4.5	Application of Automation and Artificial Intelligence	J. Pittman	NRR/DHFT/HFIB	HF5.2	6	06/30/95	NA
HF5	MAN-MACHINE INTERFACE						
HF5.1	Local Control Stations	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	4	06/30/95	NA
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	J. Pittman	RES/DRPS/RHFB	NOTE 3(b)	4	06/30/95	NA
HF5.3	Evaluation of Operational Aid Systems	J. Pittman	NRR/DHFT/HFIB	HF5.2	4	06/30/95	NA
HF5.4	Computers and Computer Displays	J. Pittman	NRR/DHFT/HFIB	HF5.2	4	06/30/95	NA
HF6	MANAGEMENT AND ORGANIZATION						
HF6.1	Develop Regulatory Position on Management and Organization	J. Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF6.2	Regulatory Position on Management and Organization at Operating Reactors	J. Pittman	NRR/DHFT/HFIB	I.B.1.1 (1,2,3,4)	1	12/31/86	NA
HF7	HUMAN RELIABILITY						

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
HF7.1	Human Error Data Acquisition	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.2	Human Error Data Storage and Retrieval	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.3	Reliability Evaluation Specialist Aids	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF7.4	Safety Event Analysis Results Applications	J. Pittman	NRR/DHFT/HFIB	LI (NOTE 5)	1	12/31/86	NA
HF8	Maintenance and Surveillance Program	J. Pittman	NRR/DLPQ/LPEB	NOTE 3(b)	2	06/30/88	NA
		CHERNOBYL ISSUES					
CH1	ADMINISTRATIVE CONTROLS AND OPERATIONAL PRACTICES						
CH1.1	Administrative Controls to Ensure That Procedures Are Followed and That Procedures Are Adequate	-	-				
CH1.1A	Symptom-Based EOPs	R. Emrit	NRR/DLPQ/LHFB	LI (NOTE 5)		06/30/89	NA
CH1.1B	Procedure Violations	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.2	Approval of Tests and Other Unusual Operations	-	-				
CH1.2A	Test, Change, and Experiment Review Guidelines	R. Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.2B	NRC Testing Requirements	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.3	Bypassing Safety Systems	-	-				
CH1.3A	Revise Regulatory Guide 1.47	R. Emrit	RES/DE/EMEB	LI (NOTE 5)		06/30/89	NA
CH1.4	Availability of Engineered Safety Features	-	-				
CH1.4A	Engineered Safety Feature Availability	R. Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.4B	Technical Specifications Bases	R. Emrit	NRR/DOEA/OTSB	LI (NOTE 5)		06/30/89	NA
CH1.4C	Low Power and Shutdown	R. Emrit	RES/DSR/PRAB	LI (NOTE 5)		06/30/89	NA
CH1.5	Operating Staff Attitudes Toward Safety	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH1.6	Management Systems	-	-				
CH1.6A	Assessment of NRC Requirements on Management	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH1.7	Accident Management	-	-				
CH1.7A	Accident Management	R. Emrit	RES/DSR/HFRB	LI (NOTE 5)		06/30/89	NA
CH2	DESIGN						
CH2.1	Reactivity Accidents	-	-				
CH2.1A	Reactivity Transients	R. Emrit	RES/DSR/RPSB	LI (NOTE 5)		06/30/89	NA
CH2.2	Accidents at Low Power and at Zero Power	R. Emrit	RES/DRA/ARGIB	CH1.4		06/30/89	NA
CH2.3	Miltiple-Unit Protection	-	-				
CH2.3A	Control Room Habitability	R. Emrit	RES/DRA/ARGIB	83		06/30/89	NA
CH2.3B	Contamination Outside Control Room	R. Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.3C	Smoke Control	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH2.3D	Shared Shutdown Systems	R. Emrit	RES/DRA/ARGIB	LI (NOTE 5)		06/30/89	NA
CH2.4	Fire Protection	-	-				
CH2.4A	Firefighting With Radiation Present	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA

Table II (Continued)

Action Plan Item/ Issue No.	Title	Priority Analyst	Lead Office/ Division/ Branch	Safety Priority Ranking	Latest Rev.	Latest Issuance Date	MPA No.
CH3	CONTAINMENT						
CH3.1	Containment Performance During Severe Accidents	-	-				
CH3.1A	Containment Performance	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH3.2	Filtered Venting	-	-				
CH3.2A	Filtered Venting	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4	EMERGENCY PLANNING						
CH4.1	Size of the Emergency Planning Zones	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH4.2	Medical Services	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH4.3	Ingestion Pathway Measures	-	-				
CH4.3A	Ingestion Pathway Protective Measures	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4.4	Decontamination and Relocation	-	-				
CH4.4A	Decontamination	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH4.4B	Relocation	R. Emrit	RES/DSIR/SAIB	LI (NOTE 5)		06/30/89	NA
CH5	SEVERE ACCIDENT PHENOMENA						
CH5.1	Source Term	-	-				
CH5.1A	Mechanical Dispersal in Fission Product Release	R. Emrit	RES/DSR/AEB	LI (NOTE 5)		06/30/89	NA
CH5.1B	Stripping in Fission Product Release	R. Emrit	RES/DSR/AEB	LI (NOTE 5)		06/30/89	NA
CH5.2	Steam Explosions	-	-				
CH5.2A	Steam Explosions	R. Emrit	RES/DSR/AEB	LI (NOTE 5)		06/30/89	NA
CH5.3	Combustible Gas	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH6	GRAPHITE-MODERATED REACTORS						
CH6.1	Graphite-Moderated Reactors	-	-				
CH6.1A	The Fort St. Vrain Reactor and the Modular HTGR	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH6.1B	Structural Graphite Experiments	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA
CH6.2	Assessment	R. Emrit	RES/DRA/ARGIB	LI (NOTE 3)		06/30/89	NA

TABLE III

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

Legend

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

TABLE III (Continued)

ACTION ITEM/ISSUE GROUP	I	S	RESOLVED STAGES			USI	HIGH	MEDIUM	LOW	DROP	CONT.	NOTE 4	NOTE 5	TOTAL
			NOTE 1	NOTE 2	NOTE 3									
TMI ACTION PLAN ITEM (369)														
GSI	84	46	0	0	135	0	0	0	12	9	-	-	-	286
LI	-	0	-	-	75	-	-	-	-	-	-	-	8	83
TASK ACTION PLAN ITEMS (142)														
USI	-	-	-	-	27	0	-	-	-	-	-	-	-	27
GSI	-	20	0	0	36	-	0	0	0	14	-	-	-	70
RI	-	-	-	-	6	-	-	-	-	-	-	-	1	7
LI	-	-	-	-	11	-	-	-	-	-	-	-	12	23
EI	-	-	-	-	13	-	-	-	-	-	-	-	2	15
NEW GENERIC ISSUES (283)														
GSI	-	54	0	0	87	0	4	0	4	105	2	1	-	257
RI	-	1	-	-	5	-	-	-	-	1	-	-	5	12
LI	-	1	-	-	8	-	-	-	-	-	-	-	4	13
EI	-	-	-	-	-	-	-	-	-	-	-	-	1	1
HUMAN FACTORS ISSUES (27)														
GSI	-	8	0	0	8	0	0	0	0	0	-	-	-	16
LI	-	-	-	-	3	-	-	-	-	-	-	-	8	11
CHERNOBYL ISSUES (32)														
LI	-	2	-	-	7	-	-	-	-	-	-	-	23	32
TOTAL:	84	132	0	0	421	0	4	0	16	129	2	1	64	853

ISSUE 196: BORAL DEGRADATION

DESCRIPTION

Historical Background

This issue was raised[1854, 1855] by an NMSS staff member whose concern arose from the observation that spent fuel pool racks using Boral for neutron absorption had experienced some problems with swelling and degradation of the Boral plates over long periods of time. Because the Boral material is commonly used for neutron absorption and shielding in a wide spectrum of applications, this degradation may have a number of implications for safety purposes. However, this issue specifically addresses the use of Boral in long-term dry storage casks for spent reactor fuel.

Safety Significance

Composition of Boral: "Boral" is the trade name for a product of AAR Advanced Structures of Livonia, Michigan and is a neutron absorber plate material which uses boron carbide for neutron absorption. One isotope of boron, specifically boron-10 (or ^{10}B), has a thermal neutron absorption cross section of over 3800 barns. Moreover, the nuclear reaction is unusual in that the resulting compound nucleus emits an alpha particle rather than de-exciting by gamma emission. Thus, unlike many other high cross section nuclides, boron-10 does not emit high energy secondary gammas, which (along with the high cross section) makes it an excellent shielding material. Consequently, Boral plate is widely used in many industrial, medical, and laboratory applications.

Boral is made by mixing boron carbide granules and aluminum powder inside an aluminum box, heating the box and its content to form an ingot, and then hot-rolling the ingot to form a plate consisting of a coarse core of B_4C-Al composite material bonded between two thin sheets of aluminum cladding.

Experience with Boral: One of the uses of Boral is in spent fuel pool racks. When the first generation of nuclear reactors was built, spent fuel racks ensured subcriticality by using rack designs which kept the spent fuel assemblies widely separated. (The thermal neutron absorption cross-section of the hydrogen in the water was sufficient to keep the array subcritical. In PWR pools, boric acid is dissolved in the water as well.) However, as more and more spent fuel was discharged into these pools, it was necessary to install new racks which held the spent fuel assemblies in a much more compact array. To ensure subcriticality, the new spent fuel racks incorporated Boral sheets between the fuel assemblies.

The Boral sheets were sandwiched (clad) within seal-welded stainless steel cover plates, apparently to keep water from contacting the Boral. Nevertheless, there were several instances (dating back to 1983) where the stainless steel cover plates experienced bulging, to the point where mechanical interference with the fuel assemblies became a problem. It was discovered upon investigation that there had been water ingress into the stainless steel sandwich, and the aluminum in the Boral had reacted chemically with the water to produce hydrogen gas and aluminum oxide. The hydrogen gas pressure had built up to the point where the stainless steel cladding bulged.

One fix was to clip the corners of the stainless steel cladding, allowing the hydrogen gas to escape. However, in follow-up investigations, it was found in some cases that the corrosion reactions resulted in a partial debonding of the Boral's aluminum cladding from the composite core absorber material, with some limited losses of B_4C granules and aluminum binder from the edges of the Boral plates.

A similar occurrence was discovered at an early generation BWR, where the spent fuel had been stored in the spent fuel pool since 1985. This plant had installed Boral cans around each fuel assembly, but without stainless steel cladding. It was discovered that there were blisters on about 5% of the Boral cans. The blisters were generally one inch in diameter or less, and tended to occur near the edge of the Boral sheet, where the internal composite core was in contact with the pool water.

Long-Term Dry Spent Fuel Storage Casks: Calculations showed that the observed B_4C losses did not result in a significant loss of shutdown margin in spent fuel pools, and that is not the focus of this generic issue. Instead, the question involves the situation with long term dry spent fuel storage casks. To understand the safety significance, it is necessary to first review the cask design and intended use. For the purposes of this screening analysis, the Holtec[1856] design was used. (Other designs exist, with somewhat different capacities, etc., but the designs are rather similar, and the differences should not affect any conclusions.)

These casks are intended for spent fuel which has been out of the reactor and in the spent fuel pool for a long time. As time goes on, the inventories of the various radioactive species in a spent fuel assembly will decrease in accordance with the half-life of each nuclide. Decay heat production will slowly diminish, and after several years, will be low enough that liquid coolant is no longer necessary to keep the spent fuel from overheating. (It is still necessary for shielding, however.) The Holtec HI-STORM design uses a multipurpose cannister (MPC) which can hold many fuel assemblies. (The MPC-24 will hold 24 PWR fuel assemblies; the MPC-68 will hold 68 BWR fuel assemblies.) Each MPC design consists of a sealed metallic canister, and the external dimensions are the same regardless of the intended contents. Once loaded, the MPC is backfilled with helium and sealed. The MPC can be placed into either a HI-TRAC transfer cask or a HI-STORM storage "overpack." The transfer cask uses lead and a water jacket for shielding, whereas the storage overpack is intended for long-term storage, and uses plain concrete for shielding. Cooling is passive; the large storage overpack incorporates air ducts for natural convection cooling.

The MPC designs include "baskets" to hold fuel assemblies, heat conduction elements that help transfer heat to the MPC shell, and Boral sheets between the baskets to provide reactivity control. Criticality is a concern; these MPCs hold about one-third of a full core for a small reactor such as Yankee Rowe.

The Boral sheets are necessary when the MPC is being loaded with spent fuel. Once the MPC is removed from the spent fuel pool, drained, and seal-welded, the lack of water as a moderator makes criticality unlikely. The MPC can reside within the HI-STORM storage overpack for many years.

Safety Concern: The HI-STORM dry storage system is designed for long term storage, but is not intended to be a permanent repository. Eventually, these MPC units will be transferred to transfer casks and shipped to a permanent repository. The criticality concern affects any MPC units that, years later, must be reopened for repairs of any kind. One scheme for doing so is to re-immerse the MPC in water for shielding, and perform the repair operations under water. Water immersion

has several advantages, including shielding, a lower working temperature, a transparent medium, and some limiting of the spread of any contamination.

However, if the MPC is re-flooded with water, the Boral sheets again become necessary to ensure a subcritical configuration. When the MPC was first loaded with fuel, these Boral sheets will have been soaked in water, with some water ingress into the coarse B_4C-Al composite material within the aluminum cladding. (The edges of the sheet are not sealed; the composite material, which is porous, will be exposed to the water.) The vacuum drying is likely to leave some residual water within the composite core. During long-term storage, these sheets will then be subjected to temperatures on the order of 500°F for many years. This is a more severe environment than that experienced by Boral sheets immersed in the spent fuel pool, where blistering has been observed after several years in warm water. It is likely that steam blisters will form in the short term, and possibly hydrogen blisters in the long term.

Thus, if there is any problem with the integrity of the Boral sheet, it is possible that, under such conditions, the material may crumble or otherwise relocate in storage, or may be physically damaged when "quenched" by reflooding of the MPC. Moreover, the blistering will displace some of the water, which will affect reactivity somewhat even if the B_4C-Al composite material does not relocate.

It is possible to form a critical array with sixteen to twenty fresh BWR fuel assemblies in cold clean water (NUREG-75/110, pp. 4-14).[1857] PWR assemblies, which are generally equivalent to four BWR assemblies each, would be expected to approach criticality with a commensurately smaller number. Of course, the fuel stored within the MPC will be, except in a few cases, fuel with significant burnup. Nevertheless, this spent fuel was still capable of producing some power (with equilibrium xenon and at reactor temperatures) before it was discharged. (Equilibrium xenon is typically worth 2.5% to 3% $\Delta K/K$, and the moderator and Doppler defects are generally worth 3% to 4% in addition.)

Thus, although this spent fuel is not likely to achieve high power levels, it is quite credible that reflooding an MPC unit will result in an inadvertent criticality if the Boral neutron absorber is not present. Such an event might not damage the fuel cladding, but it would certainly produce high neutron and fission gamma radiation fields, which can be quite hazardous to personnel unless adequate shielding is in place.

There are two other aspects to such an inadvertent criticality event. First, there will not be any "scram" system or similar safety system available to rapidly insert negative reactivity, and it may not be immediately obvious to the personnel what should be done to terminate the event.

Second, the existing neutron flux from transuranics in the spent fuel may not be high enough to ensure a controlled startup. This can lead to a classic criticality accident, where a critical configuration is achieved, but nothing happens because there are not enough neutrons to start the chain reaction. Then, as the evolution continues, the configuration might be significantly supercritical before the reaction starts, and when it does start, neutron flux will escalate with a very fast period, leading to a very hazardous situation.

Possible Solution

The proposed solution for this generic issue is in two steps. The first step would be to test samples of Boral under conditions duplicating the environmental conditions that would be experienced in these MPC units. This experiment can be done quite readily, and at a modest cost. If there is no

evidence for crumbling or relocation of the B_4C-Al composite material, the issue would be considered resolved.

However, if the experimental evidence indicates that relocation of the B_4C-Al composite material is credible, the second step would be to ensure that these MPC units either are repaired under dry conditions, or that the water used in submerged operations contain a soluble neutron absorber such as boric acid (or some other means be used for reactivity control).

Alternatively, it is the staff's understanding that the manufacturer has been conducting research to find ways to improve the performance of Boral. This also could resolve the issue.

SCREENING ANALYSIS

Criteria: The usual criteria for screening generic issues, specifically core damage frequency, large early release frequency, and person-rem per reactor-year, are not applicable to this issue. However, there are some statements in the regulations that address accidental criticalities.

10CFR72, Part 72.124, "Criteria for nuclear criticality safety," states, in part:

> (a) *Design for criticality safety.* Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety.

10CFR72, Part 72.236, "Specific requirements for spent fuel storage cask approval and fabrication," goes on to say:

> (c) The spent fuel storage cask must be designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions.

It also states:

> (h) The spent fuel storage cask must be compatible with wet or dry spent fuel loading and unloading facilities.

The essence of these statements is that an accidental criticality is not to occur, i.e., the probability of such an event should be low. For the purpose of this screening analysis, and in the absence of any formal guidance, a probability of 0.1 of an inadvertent criticality event will be used as a screening threshold.

Frequency Estimate

It is not possible to perform a probabilistic analysis in the usual sense, since there is no data upon which to base the analysis. Nevertheless, it is possible to have some qualitative probabilistic insights into the likelihood of such an event.

The central question of this generic issue is, will there be extensive blistering and degradation of the Boral sheets? If the blistering is extensive, it is quite likely that, in at least some of the MPCs, there will be an axial location where there is insufficient boron absorption to maintain a subcritical

condition. It will be assumed that at least one MPC in 100 will be damaged to the extent that an inadvertent criticality is possible if the MPC is refilled with water. This is, of course, an educated guess. However, if blistering is this extensive, it should be detectable by a few experiments.

At the time of writing of this analysis, there were approximately 117 reactors with spent fuel pools, twelve of which were no longer operating. A simple tabulation was used to estimate the number of MPC units needed for this entire population, assuming a 40-year lifetime (or the actual lifetime for the shutdown units), a 1.5 year fuel cycle, a third core replacement for each fuel cycle, and the Holtec design of 68 BWR fuel assemblies per MPC, or 24 PWR assemblies per MPC. The result was an estimate of 8,783 MPC units. Assuming a 20-year license extension for the units that were not already shut down, this number increased to 12,850 MPC units.

Based on this, it was reasonable to assume that the total number of MPCs needed to accommodate the present generation of power reactors was on the order of 10^4 units.

For an inadvertent criticality event to occur, two events must happen. First, the Boral sheets in an MPC unit must be damaged to the point where criticality becomes possible. Second, this same MPC unit must be flooded with water for repair.

Based purely on engineering judgment, it was assumed that less than 1% of the MPCs will need to be opened and reflooded for any reason. However, as a practical matter, it was doubtful that this number of MPCs which must be opened would be less that one per thousand, based on general engineering experience. Therefore, it was assumed that the number opened would be at least one per thousand, the lower limit.

Putting these figures together, with a population of 10,000 MPC units, and assumptions that at least one unit per thousand would be opened underwater for repair (or any other reason), and at least one per hundred will be damaged to the point where criticality is possible, the total number of criticality incidents will be at least 0.1. Thus, the probability of at least one criticality was given by the Poisson formula:

$$P(n \geq 1) = 1 - e^{-x}$$

where x is the expected number of criticalities. If x, the expectation value, is 0.1 or greater, the probability of at least one event is 0.095 (essentially equal to the expectation value) or greater. Thus, under these assumptions, this generic issue meets the screening criterion described earlier.

Other Considerations

The semi-quantitative estimate developed above assumes that the likelihood of opening the MPC underwater, and the likelihood of damage such that criticality is possible, are independent. This may not be completely true. For example, if an MPC unit were involved in a transportation accident of any kind, the robustness of the MPC and its transfer cask would preclude any release of radioactive material to the surroundings. However, the physical assault might cause relocation of the B_4C-Al composite material, and also increase the likelihood of the MPC being opened for inspection. This would tend to increase the probability of an accidental criticality above that estimated by the assumption of randomness above.

CONCLUSION

Based on the likelihood of an accidental criticality described above, and on the relatively modest resources needed for resolution, technical assessment of the issue was pursued.[1875] The issue was later RESOLVED with no new requirements or guidance for licensees.[1876]

REFERENCES

1854. Memorandum to F. Eltawila from D. Carlson, "Proposed Generic Safety Issue: Boral Degradation," November 4, 2003. [ML033090600]

1855. Memorandum to D. Carlson from F. Eltawila, "Generic Issue 196: Boral Degradation," November 10, 2003, [ML033160580]

1856. "Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM Cask System)," Docket 72-1014, July 19, 2000. [ML003777866]

1857. NUREG-75/110, "Safety Evaluation Report for Preliminary Design Approval for GESSAR-238 - Nuclear Island Standard Design," U.S. Nuclear Regulatory Commission, December 1975.

1875. Memorandum to C. Paperiello from J. Rosenthal, "Results of Initial Screening of Generic Issue 196, 'Boral Degradation,'" November 19, 2004. [ML042670379]

1876. Memorandum to L. Reyes from B. Sheron, "Closure of Generic Issue 196, 'Boral Degradation,'" February 22, 2007. [ML070100403]

ISSUE 198: HYDROGEN COMBUSTION IN PWR PIPING

DESCRIPTION

Historical Background

This issue was initiated[1833] in response to a recommendation in the screening analysis of Generic Issue (G)I-195, "Potential Hydrogen Combustion in BWR Piping." GI-195 investigated the potential safety significance of hydrogen combustion events that had actually occurred in BWR primary system piping. Although no similar events had been observed in PWRs, it was felt that the potential safety significance of such events in PWRs should also be investigated.

Safety Significance

In any water-cooled reactor, radiolysis of the water in the reactor core is always producing some elemental hydrogen and oxygen. In the BWR experience described in GI-195, these gases, because of their buoyancy in a steam atmosphere, tended to build up in high, stagnant points in the primary system over a period of time. This hydrogen-oxygen mixture, although somewhat diluted by steam, is also at reactor operating conditions, i.e., at a pressure of about 1000 psi and a temperature near 500°F. At these conditions, relatively little energy is required to ignite the mixture. There were three events where a combustible mixture built up in piping connected to the primary system and detonated. Although the gas buildup in all three cases was in a pipe volume isolated from the primary system by a check valve, the force of the detonations was sufficient to rupture the piping. In three other events, a combustible mixture accumulated in the top works of a safety/relief valve. When the mixture ignited, mechanical damage caused the valves to fail open and blow down the primary system.

The screening analysis of GI-195 concluded that the frequency of such events was sufficiently low that, given the many diverse systems available to mitigate such an event in BWRs, there was insufficient safety significance to justify regulatory action on BWRs. However, the review panel noted that, although a search of PWR operational history found no such events, PWRs also do not have as many ways of mitigating a loss of coolant accident. Because of this, the panel recommended that the hydrogen combustion phenomenon also be investigated for PWRs.

SCREENING ANALYSIS

Although there have been hydrogen fires at PWR plants, particularly during maintenance activities, no events which occurred in the primary system at power have been reported. Therefore, some deterministic methods must be used to estimate where such events are likely, and how frequently they might occur.

The PWR primary system is a closed system. In order to limit corrosion, it is common practice to operate with an excess of dissolved hydrogen in the primary coolant, which has the effect of scavenging the oxygen produced by radiolysis. This is accomplished by providing a hydrogen cover gas in the volume control tank in the Chemical and Volume Control System (CVCS), which supplies charging fluid to the primary system. Normally, dissolved oxygen is maintained below 0.1 ppm, and dissolved hydrogen will be in the range of 25 to 35 cc (at STP) per kilogram of water, which is equivalent to a weight percentage of 2.2 to 3.1 ppm.

Deflagration or detonation of a hydrogen-oxygen mixture could occur if a gas bubble formed in the primary system and was trapped in a stagnant volume in the upper portion of the system (e.g., the control rod travel housings, the reactor vessel head vent, or any other upward-leading pipe that normally carries no flow). Alternatively, the gases could accumulate in the pressurizer steam space. At pressures around 2000 psi and temperatures approaching 600°F, relatively little energy would be needed for ignition.

Phenomenology

A review of the physical phenomenology of gas solubility and bubble formation is helpful here. A more complete description can be found in NUREG/CR-2726.[1872] Consider a volume containing liquid water and hydrogen gas, in equilibrium with pressure P and temperature T. There will be some hydrogen gas dissolved in the liquid phase in the lower portion of the volume, and some water vapor in the gas phase above.

According to Henry's Law, the solubility of hydrogen is proportional to the partial pressure $P(H_2)$ in the cover gas.

$$P(H_2) = H(T)X(H_2)$$

Where $P(H_2)$ = partial pressure of hydrogen
$H(T)$ = Henry's Law constant
$X(H_2)$ = mole fraction of hydrogen in the liquid phase.

The total pressure is the sum of the partial pressure of hydrogen and the partial pressure of the water vapor, which is just the saturation pressure for water at temperature T.

$$P = P(H_2) + P_{Sat}(T)$$

For example, consider the conditions at the top of a PWR reactor vessel, above the core. From the PWR training manual,

P = 2235 psig = 2250 psia
T = 610.7°F = 321.5°C = 594.66°K
H(T) = 1.25E5 psia/mole fraction[1872]

At this temperature, the saturation pressure of water is

P(sat) = 1667.8 psia

The partial pressure of the hydrogen is the total pressure minus this saturation pressure:

$P(H_2)$ = 2250 - 1667.8 = 582.2 psia

Then, using Henry's Law,

$$X(H_2) = \frac{P(H_2)}{H(T)}$$

Working this out, $X(H_2)$ = 582.2 psia/1.25E5 psia/mole fraction = 4.66E-4 mole fraction, which works out to about 515 ppm by weight.

This is the equilibrium concentration. If more hydrogen is added to form a higher concentration, the solution would be supersaturated, and the excess hydrogen would bubble out in the gas phase. Conversely, if the hydrogen concentration in the liquid water were below 515 ppm, the gas bubble would dissolve into the liquid. Since the normal reactor chemistry maintains the hydrogen concentration below 3.1 ppm (about 0.6% of the saturation value), gas bubbles would not be expected to form in this area.

Similarly, the equilibrium concentration for oxygen is about 3560 ppm, and the normal concentration in the coolant is about 0.1 ppm *(D. M. Himmelblau, "Solubilities of Inert Gases in Water," Journal of Chemical & Engineering Data, 5:10-15, 1960)*. Clearly, oxygen is even less likely to come out of solution and form a bubble, since its equilibrium solubility is higher, and its normal concentration is lower. Similar calculations for various locations in the primary system result in the following equilibrium concentrations:

Location	Pressure (psia)	Temperature, (F)	Equilibrium hydrogen concentration (ppm)	Equilibrium oxygen concentration (ppm)
Reactor vessel head, nominal conditions	2250	610.7	515	3560
RCS cold legs & vessel downcomer	2250	545.7	323	2150
Control rod drive travel housings	2250	248	340	1980
Pressurizer	2250	653	essentially zero	essentially zero
Volume Control Tank (one atm. H_2 overpressure)	30	115	3.5	—

Again, given that the normal hydrogen concentration is 2.2 to 3.1 ppm, and normal oxygen concentration is 0.1 ppm, gas bubbles are not expected to form in the normally liquid-filled portions of the primary system. If such bubbles were somehow introduced, they would be expected to dissolve into the liquid coolant over a period of time. Therefore, collection and detonation of these gases in the liquid-filled portion of the primary system do not appear to be credible.

The exception is the pressurizer steam space. In theory, if the pressure were exactly equal to the saturation pressure for water at that temperature, the partial pressure of hydrogen (and oxygen) would be zero and any dissolved gas would be driven out of solution until some partial pressure of hydrogen and oxygen existed in the steam space. For 3.1 ppm, this translates to a partial pressure of 2.27 psia of hydrogen in the pressurizer steam space, if the gas were evenly dispersed. For a nominal steam volume of 720 cubic feet, this is equivalent to 62 moles of hydrogen gas, which, at 2250 psia and 653°F, would occupy a bubble approximately 14 inches in diameter if collected into one volume.

Similarly, if the oxygen concentration in the liquid coolant is 0.1 ppm, the partial pressure at equilibrium conditions would be 0.00647 psia, implying the presence of slightly less than six grams of (uniformly dispersed) oxygen in the steam space. If gathered together in one volume, this is equivalent to a bubble approximately two inches in diameter.

However, the hydrogen and oxygen are less dense than steam at the same temperature. The densities work out as follows:

Hydrogen gas	0.38	lb/ft^3.	(Based on ideal gas law)
Oxygen gas	6.02	lb/ft^3.	(Based on ideal gas law)
Saturated steam	6.373	lb/ft^3.	(Based on steam tables)

Depending on the degree of stagnation in the pressurizer steam space, these gases, because of their buoyancy, will tend to collect at the top of the pressurizer steam space, rather than being evenly distributed. If the mixture of gases is quiescent, with no turbulence or convection currents, the hydrogen will rise to the top. The oxygen is less buoyant, but will still tend to rise and form a layer at the interface between the hydrogen and the steam. This will tend to reduce the partial pressure of both gases at the liquid surface, causing more gases to come out of solution. Because more dissolved hydrogen (and oxygen) is continuously being introduced into the primary coolant, the collection of a significant amount of these gases at the top of the pressurizer steam space does appear to be credible. Moreover, because of the tendency to stratify, there will be a locus somewhere in the hydrogen/oxygen interface where the ratio of the two gases permits deflagration.

Conversely, if this volume has enough circulation to remain well mixed, the hydrogen and oxygen will be highly diluted in steam, and are unlikely to form a combustible mixture in this volume. However, even if this volume is well-mixed, any connected piping can still collect pockets of hydrogen and oxygen if this piping leads in an upward direction and is slightly cooler than the main pressurizer steam space. The two combustible gases will tend to rise and collect in the upper portions of the piping, and the steam will tend to condense on the pipe walls and run back into the main volume.

In reality, the pressurizer steam space is likely to have some convection currents whenever the heaters are in operation. Also, normally a small amount of flow is maintained in the spray line to keep the boron concentration in the liquid-filled pressurizer volume equal to that in the rest of the reactor coolant system. Thus, there is likely to be some mixing, such that the hydrogen and oxygen are unlikely to be present in well-defined, horizontal layers, but the presence of a combustible mixture, particularly in attached piping, cannot be ruled out.

The top of the pressurizer does have some locations where collection and ignition of a hydrogen-oxygen mixture could pose a problem. For a typical Westinghouse design, these include:

- The top of the head itself. However, detonation of a gas stratum here would not be likely to do much damage, since the gas volume would not be confined.

- A bolted personnel access hatch ("manway"). Again, detonation of a gas stratum here would not be likely to do much damage, since the gas volume would not be confined.

- The pressurizer spray nozzles (supplied by a four-inch line), which are located at the top of the upper head in the Westinghouse design. However, normal operational practice is to maintain a small flow through these lines all the time, in order to ensure a uniform boron concentration throughout the primary system. Thus, the line is normally liquid-filled, which greatly reduces the likelihood of noncondensible gases accumulating in this volume.

- Safety Valves: These are connected via 6-inch pipes, but the pipes are shaped in the form of a loop seal, keeping liquid in contact with the valve seat. (In contrast to the case of BWRs, catalysis by the plating in the valve internals is not possible.) The only volume which

can accumulate gases is in the pipe upstream of the loop seal. A rupture of one of these lines would be an "intermediate" (i.e., up to six-inch equivalent diameter) LOCA.

- PORV Lines (Including Block Valves): The PWR training manual lists this as a six-inch line. However, the PORV capacity varies significantly from plant to plant, and presumably this pipe size can vary as well. However, a six-inch equivalent diameter LOCA should bound any PORV line break for any plant.

- Various instrument taps and small pipes, such as the level instrumentation, sampling lines, valve leakoffs, etc. A rupture of one of these lines would be a small break LOCA.

The design specifics can vary from product line to product line, and from reactor vendor to reactor vendor. For example, some B&W pressurizers have a vent valve at the very top, and the spray line penetration is off to the side, with a pipe running inside the steam space to the spray nozzle, which is located just below the center of the upper head. Nevertheless, all designs will have similar potential break locations.

Initiating Event Frequency Estimate

According to a private communication from the IAEA, PWR experience is approximately 6280 reactor-years at the time of this writing. An examination of the IAEA database has not found any instances of hydrogen deflagration in the primary system of a PWR.

In contrast to the PWR experience, there have been six such events in approximately 2325 BWR-years, as was discussed in GI-195. Thus, the credibility of such an event in a PWR is based on the experience of BWRs. There has been no PWR event which would indicate the possibility of such an event in a PWR. Clearly, the frequency of such an event in a PWR must be much less than the frequency of occurrence in BWRs, given that no PWR events have occurred, even though the accumulated experience of PWRs is almost triple that of BWRs. It will be necessary to make some assumptions regarding the combustion phenomena in order to extrapolate the BWR experience to PWRs.

One candidate explanation for the difference of event frequencies would be the PWR practice of intentionally operating with an excess of hydrogen in the coolant, with the express purpose of reducing the oxygen concentration. However, many BWRs use an analogous practice, adding hydrogen or noble metals to the feedwater and thereby reducing oxygen concentrations to as low as 0.005 ppm - actually less than oxygen concentrations in PWRs. This technique was being used in the Japanese plant that experienced a rupture of a steam condensation line. (See GI-195.) Thus, the PWR hydrogen strategy does not appear to provide a viable explanation for the difference in frequencies.

There are three other major differences between the steam spaces in a BWR and in a PWR. First, the PWR pressurizer steam space operates at double the pressure of a BWR, and consequently is about 100°F hotter. It is not obvious how this would affect the likelihood of a deflagration of hydrogen and oxygen. However, a higher temperature and pressure normally increase rather than decrease the likelihood of a spontaneous ignition. Thus, this difference also does not appear to provide a viable explanation for the difference in frequencies.

Second, the steam volume in a PWR pressurizer is fairly small - about 720 cubic feet. There is no readily-available data on the steam-filled volume in a boiling water reactor, but a rough calculation based on vessel dimensions, and including the steam lines, gave an estimate of approximately

7960 cubic feet - about eleven times that of a PWR. It does appear likely that more steam volume implies a higher likelihood of a combustion event. (Certainly, reducing the volume to zero would reduce the combustion probability to zero.)

Third, a PWR pressurizer does not have as many valve-isolated volumes connected to it. It should be noted that in all three BWR events that resulted in a pipe rupture, the rupture occurred in a pipe volume which was isolated by a check valve or isolation valve from the primary system. Although this (fortunately) meant that the events involved no significant coolant inventory loss, it also raises a question as to whether the pipes would have ruptured if the volume had not been confined. (Even a check valve, which would normally have relieved pressure by passing inventory back into the primary coolant system, might provide enough resistance to allow a deflagration event to transition into a detonation event.)

In the remaining three BWR events, combustible gases collected in the top works of a safety/relief valve. In these events, there was no piping rupture, but the safety/relief valve failed open. Such an event is less likely in a PWR, because of the use of loop seals, and because PWRs generally use spring safety valves which do not have the same top works as a BWR S/RV. It would be possible for a pressurizer PORV to be damaged and fail open by this mechanism, but the block valve would be available to stop the inventory loss.

Thus, the lower number of isolated volumes connected to a PWR steam space would also help explain the lower frequency of observed combustion events in PWRs.

In the absence of any better information, it will be assumed that the frequency of a combustion event in a PWR can be estimated from the BWR event frequency in linear proportion to the steam volumes of the two designs. For this purpose, the BWR safety/relief valve openings will not be included, since the analogous events in a PWR are rendered far less likely by the loop seals and the valve internals design. The BWR isolated-pipe-rupture event frequency used in the screening analysis of GI-195, based on three events in 2325 BWR-years, was 1.3E-3 event/ BWR-year, normally distributed with a standard deviation of 7.5E-5/BWR-year. Ratioing this by the steam space volumes gives an estimated PWR combustion event frequency of 1.17E-4 event/PWR-year, with a standard deviation of 6.8E-5/PWR-year.

A combustion event such as this does not necessarily lead to a loss of coolant, as the BWR experience shows. The screening analysis for GI-195 used an exponential distribution (with a mean of 0.21) to estimate the likelihood of a non-isolatable pipe rupture, given a combustion event. It should be noted that this is not a distribution with a peak at 0.21. Instead, this distribution has a maximum value at zero, a mean of 21%, and a tail that is "chopped off" with a value of 0.04 at unity.

In the absence of any better information, this same exponential distribution will be used in this analysis. However, it should be noted that, in the BWR screening analysis, this distribution was primarily a geometrical parameter used to describe the likelihood that a pipe rupture would be in a location not isolatable, i.e., the complementary likelihood would be the likelihood of a rupture in an isolatable location. In the present use for a PWR scenario, this is the likelihood that a combustion event will result in coolant inventory loss. The complementary likelihood is that of a combustion event that causes no significant damage.

Before going further, an examination of the reasonableness of these numbers is in order. Given that there are currently 6280 PWR-years of experience worldwide, and no observed events, it is reasonable to expect frequencies that are less than 1/6280, or about 1.6E-4. However, estimates

that are several orders of magnitude less than this upper limit would need considerable justification.

The mean PWR combustion event frequency distribution above is 1.17E-4/PWR-year, estimated from the BWR experience. This is not much less than what would be estimated if an actual event had been observed in the existing 6280 PWR-years of experience, which would be about 1.59 event/PWR-year. Thus, the estimate extrapolated from BWR experience is not highly non-conservative.

However, it could be argued that PWR combustion events might already have occurred but not have been noticed. To address this, a numerical calculation of the frequency of pipe ruptures was performed, using the normal combustion event frequency distribution and the exponential pipe rupture probability distribution discussed above. The results were a mean estimated pipe rupture frequency of 2.47E-5 event/PWR-year. (The number of significant figures does not imply accuracy to this degree, but instead are provided as an aid in following the calculation.) This is about 15% of the value corresponding to one actual event in the existing 6280 PWR-years of experience. Again, this appears to be reasonable.

Discussion

Again, there are two modeling assumptions in the approach to initiating event frequency above: that the PWR combustion event frequency can be extrapolated from the observed BWR frequency in proportion to steam space volumes, and that the likelihood that such a combustion event will result in a breach of the reactor coolant pressure boundary can be estimated by exponential assumption used in GI-195. The numerical effect of both of these assumptions will be explored in the section on "core damage frequency" below.

Both of these assumptions are being made because, if there were no BWR events, there would be no experience indicating that such an event would be expected in a PWR. There have been no recorded events of this nature in 6280 PWR-years of experience, worldwide. One other approach would be to focus only on the PWR experience and see if a bounding value can be inferred. This can be done by assuming an exponential distribution for the initiating event frequency, and choosing the exponential parameter such that 95% of the distribution lies below a frequency of one event in 6280 PWR-years, consistent with the usual use of a 95% confidence interval. This approach, which also will be explored, makes no assumptions based on BWR experience.

Core Damage Frequency

The event of interest is a breach in the reactor coolant pressure boundary caused by a detonation in the pressurizer steam space or associated piping. In view of the various pipe sizes where a combustible mixture might accumulate, the possible break sizes include:

- A very small break (S3 - 1/2 inch or less equivalent diameter) caused by a ruptured instrumentation line or a PORV leak.

- A small break (S2 - 1/2 inch to 2 inches equivalent diameter) caused by damage to the PORV and/or block valve internals.

- An intermediate-size break (S1 - 2 inches to 6 inches equivalent diameter) caused by rupture of the line leading to the PORV or by rupture of the pressurizer spray line.

Based on their positions in the upper portions of the pressurizer, the PORV and spray lines are the

most credible locations, although the spray line is normally liquid-filled. Thus, the probabilistic calculation will assume an intermediate-size ("S1") break. (This is somewhat conservative, in that instrument line breaks, sampling line breaks, etc. would normally be classified as S2 LOCAs, and evaluated using a separate event tree. This screening analysis therefore is bounding in the sense that all breaks are evaluated as S1 breaks.)

The NUREG-1150[1081] PRA for the Sequoyah plant was chosen for the analysis. This plant has a somewhat higher vulnerability to loss-of-coolant accidents because of its manual switchover to ECCS recirculation mode, and thus should bound most PWR plants.

As was done for the GI-195 screening analysis, this scenario was analyzed by constructing a new event tree. This new event tree was a simple copy of the existing event tree for the intermediate break "S1" LOCA, but the initiating event at the beginning of the tree was replaced by two top events - the detonation-induced pipe break frequency followed by the probability of not isolating the break, as described in the paragraphs above. The remainder of the event tree is exactly the same as that for the "S1" LOCA.

Five separate cases were run, to test first the event tree itself, and then to explore the effect of the various modeling assumptions described earlier. Case I was intended to see if the results of the calculation would match the published results for the SI LOCA event in the NUREG-1150[1081] analyses. To do this, the initiating event frequency was set to 10^{-3} S1 LOCA event/year, the probability of a non-isolatable break was set to unity, and the calculation was run using a sequence cutoff frequency of 10^{-7} and limited Latin hypercube sampling. The result was a mean core damage frequency (CDF) of 6.86E-06/year. The mean S1 LOCA sequences given on pp. 5-16 and 5-17 of NUREG/CR-4550,[1318] Vol. 5, Part 1, Rev. 1, sum to 6.8E-06, which compares quite well.

The sequence cutoff of 10^{-7} which was used in the NUREG-1150[1081] PRAs made the calculation much more practical by greatly reducing the number of cut sets to be calculated. Although such a cutoff tends to slightly reduce the CDF estimate, this is an appropriate numerical technique for the original PRA, since the S1 sequences contributed less than 10% of the total CDF. However, the sequences of interest for this GI are quite likely to be in the 10^{-7} range, and using this cutoff would significantly affect the results. Therefore, Case II is a repeat of Case I, i.e., a calculation of the S1 LOCA tree, but with the cutoff frequency lowered to 10^{-10}/year, and using regular Monte Carlo sampling instead of the limited Latin hypercube method.

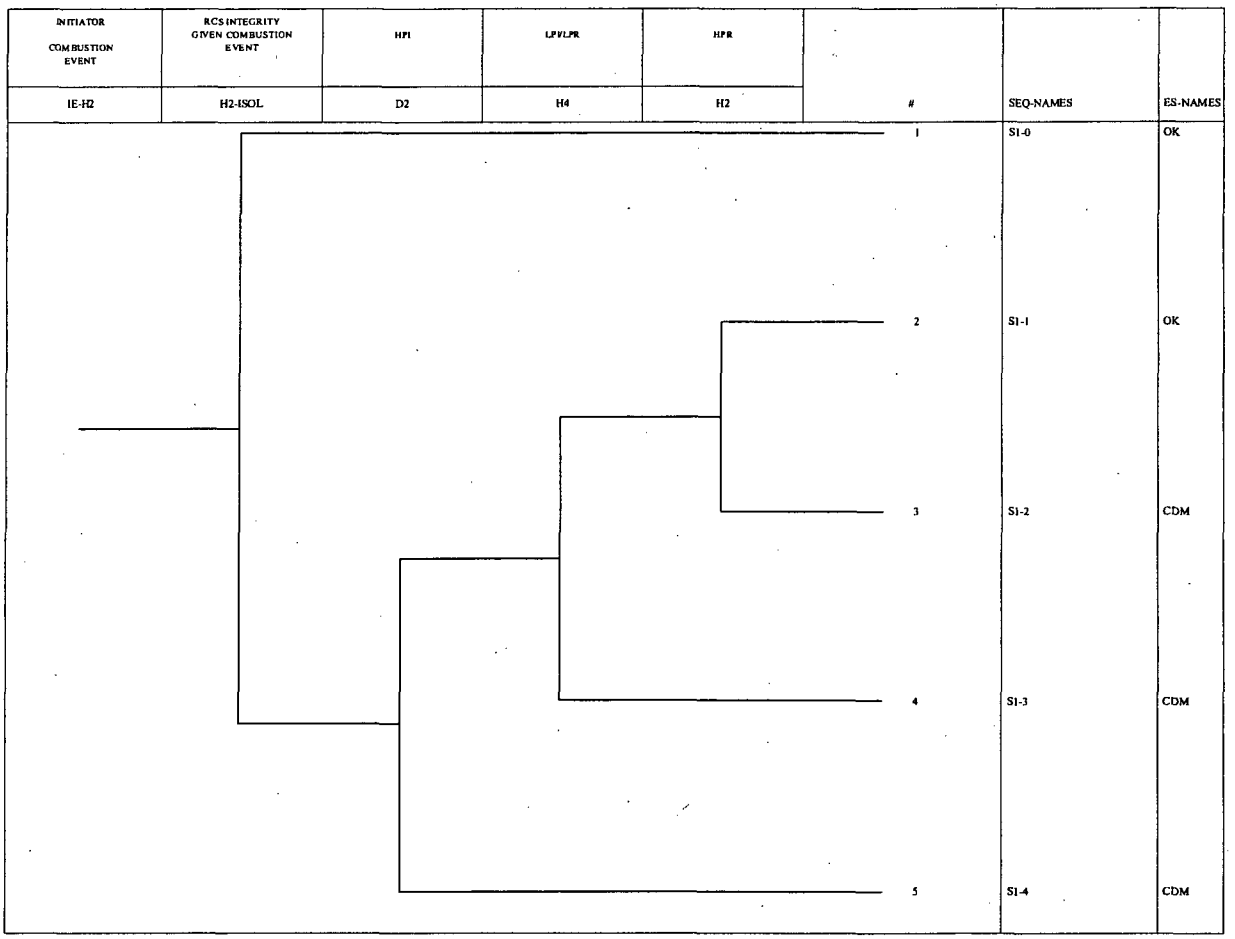
Case III is the same event tree, but with the initiating event frequency lowered to 1.17E-4/PWR-year, the estimate derived from BWR experience, and the probability of a non-isolatable break still set to unity. This calculation is not a "test case" like the first two cases, but instead corresponds to a CDF associated with this GI, although it assumes that every combustion event causes a LOCA.

Case IV is similar to Case III, but this time the probability of a non-isolatable break is calculated by means of the exponential distribution described earlier. This case is the "best" estimate of the safety significance of the GI, based on extrapolation from the experience in BWRs.

Finally, Case V uses the limiting exponential initiating event frequency based only on PWR experience, with the probability of a non-isolatable break set to unity. This case is intended as a check on the modeling assumptions used in Cases III and IV. The results of these cases are as shown in Table 3.198.1.

(Results in this and in subsequent tables are given to three significant figures for the convenience of the reader who wishes to follow the calculations, and are not intended to imply that these parameters are known to this accuracy, as the percentile range given in the table itself clearly shows.)

As can be seen by comparing the various cases tabulated above, the "limiting" case V CDF is only



about a factor of two greater than that estimated by Case IV. Also, the effect of the rather questionable extrapolation of pipe break probability given an ignition event can be seen by comparing the mean estimates based on Case IV vs. Case III: this assumption reduces the calculated CDF by about a factor of five. More importantly, all three cases associated with this GI (i.e., Cases III, IV, and V) have core damage frequencies below the GI screening criterion of 10^{-6} core damage event/RY.

In addition, an audit calculation was performed using a SPAR model for a PWR with the highest CDF for the medium LOCA. (A description of this calculation is attached to this report.) The audit calculation is similar to Case V, in that it is based on the PWR data only, but uses a more sophisticated Bayesian approach. This calculation gave a mean CDF of 8.3E-7, which agrees quite well with Case V above, and also is below the GI screening criterion of 10^{-6} core damage event/RY.

Table 3.198-1

Calculation	Mean	5th Percentile	Median	95th Percentile
Case I existing S1 initiating event frequency, cutoff = 10^{-7}, LHS sampling	6.860E-06	1.645E-07	1.931E-06	2.770E-05
Case II Existing S1 initiating event frequency, but cutoff = 10^{-10} and Monte Carlo sampling	1.003E-05	1.982E-07	2.415E-06	3.259E-05
Case III Initiating event frequency set to ratioed BWR ignition frequency, but assuming every ignition causes a break	1.232E-06	1.327E-07	7.018E-07	3.344E-06
Case IV Initiating event frequency set to ratioed BWR ignition frequency and including non-isolatable break probability	2.425E-07	4.413E-09	9.520E-08	8.662E-07
Case V Initiating event frequency set to 95% confidence distribution based on PWR experience.	5.113E-07	1.614E-08	2.354E-07	1.680E-06

Consequence Estimate

In the base PRA, the S1 LOCA sequences were placed in Plant Damage State Group 2 (see NUREG/CR-4551,[1795] page 2.9). A rough estimate of the consequences was made, using the CRIC-ET Code[1796] and the Sequoyah model, but using a consequence file loaded with the GIs program standard site parameters. The results, using the NUREG-1150[1081] technique of limited Latin hypercube sampling of 200 samples, were as follows:

	Person-rem per reactor-year
Mean	0.14
5th percentile	1.15E-03
95th percentile	0.813
Median	2.59E-02

The distribution is based on the uncertainties associated with the Level II and Level III analyses, and does not include the Level I distribution. (Again, the results in this table are given to three significant figures for the convenience of the reader who wishes to follow the calculations, and are not intended to imply that these parameters are known to this accuracy, as the percentile range given in the table itself clearly shows.) These estimates are well below the screening cutoff of 100 person-rem/RY given in the Handbook for Management Directive (MD) 6.4.

Large Early Release Frequency (LERF)

The NUREG-1150[1081] models loaded into the CRIC-ET code precede the development of large early release models, such as those described in NUREG/CR-6595.[1873] Nevertheless, an approximation can be made using the CRIC-ET code by selecting accident progression sequences which involve a large release. For this analysis, the following were selected:

Accident progression characteristic	Description	Mean frequency	Expected early fatalities, conditional on event
1C	Containment failure during core degradation	1.47E-08	10.9
1D	Containment failure at vessel breach	3.11E-08	17.9
6A	Induced Steam Generator Tube Rupture	1.20E-10	28.6
10C	Containment failed due to small hole, leak, failure to isolate	3.78E-08	0.77
Combination	all above sequences	6.74E-08	—

The frequencies and conditional early fatalities were calculated using the CRIC-ET code and the standard Generic Issues Program site parameters. Other accident progression sequences either were not applicable to a LOCA-initiated event (e.g., characteristics 1A and 1B are vessel rupture failures), or were not large and early, and (not surprisingly) had an estimated number of early fatalities below 0.5. The "combination" result in the bottom row is not a simple sum of the other rows, since there are some sequences which would be included in more than one row.

The combination row is a reasonable approximation for LERF. The statistical distribution is:

mean	6.74E-08	large early releases/RY
5th percentile	1.34E-09	
95th percentile	2.92E-07	
median	2.18E-08	

The distribution is based on the uncertainties associated with the Level II and Level III analyses, and does not include the Level I distribution. Again, the mean is well below the screening cutoff of 10^{-7} large early release/RY given in the Handbook for MD 6.4.

Cost Estimate

Because of the low CDF and risk, a cost estimate will not affect the conclusion of this analysis. Therefore, no cost analysis was performed.

Uncertainties

One of the more unusual aspects of the screening analysis calculations for this GI is that the initiating event is based on actual data, which results in an uncertainty distribution which, although quite wide by experimental standards, is much smaller than the uncertainty limits often seen in probabilistic analyses. Also, the distribution is normal rather than log-normal. The effect is to produce an estimate of CDF where the uncertainty bounds are not as wide as usual. However, some caution is called for in understanding these uncertainty limits, since they do not include the modeling uncertainty associated with extrapolating BWR experience to PWRs.

Discussion

The estimates above are all predicated on the fact that no hydrogen combustion events have thus far been observed thus far in PWRs. If such an event does occur in the future, this GI should be reevaluated. Similarly, if more BWR events occur, such that the estimated BWR initiating event frequency increases above the current estimate of 1.3E-3 combustion event/BWR-year, this GI

should be reevaluated.

CONCLUSION

The CDF, LERF, and risk associated with this GI are below the screening thresholds given in Appendix C of the Handbook for MD 6.4, "Generic Issues Program." Therefore, it was concluded that there was insufficient justification for this GI to continue to the technical assessment stage, and the issue was DROPPED from further consideration.[1874]

REFERENCES

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

1318. NUREG/CR-4550, "Analysis of Core Damage Frequency from Internal Events," U.S. Nuclear Regulatory Commission, (Vol. 1, Rev. 1) January 1990, (Vol. 2) April 1989, (Vol. 3, Rev. 1) April 1990, (Vol. 4, Rev. 1) August 1989, (Vol. 5, Rev. 1) April 1990, (Vol. 6) April 1987, (Vol. 7, Rev. 1) May 1990.

1795. NUREG/CR-4551, "Evaluation of Severe Accident Risks," U.S. Nuclear Regulatory Commission, (Vol. 1, Rev. 1) December 1993, (Vol. 4, Rev. 1, Part 1) December 1990, (Vol. 7, Rev. 1) March 1993.

1796. Letter Report, "NUREG-1150 Data Base Assessment Program: A Description of the Computational Risk Integration and Conditional Evaluation Tool (CRIC-ET) Software and the NUREG-1150 Data Base," T. D. Brown *et. al.*, March 1995.

1833. Memorandum to A. Thadani from J. Flack, "Results of Initial Screening of Generic Issue 195, 'Hydrogen Combustion in Foreign BWR Piping,'" February 23, 2004. [ML040850566]

1872. NUREG/CR-2726, "Light Water Reactor Hydrogen Manual," U.S. Nuclear Regulatory Commission, August 1983.

1873. NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," U.S. Nuclear Regulatory Commission, December 1999.

1874. Memorandum to B. Sheron from S. Bahadur, "Results of Initial Screening of Generic Issue 198, 'Hydrogen Combustion in PWR Piping,'" March 22, 2007. [ML070580447]

ISSUE 200: TIN WHISKERS

DESCRIPTION

Historical Background

This issue was identified by RES after concern was raised for the number of component failures in the solid state protection systems of nuclear power plants that were caused by the growth of tin whiskers.[1877]

In 1987, Dresden Nuclear Power Station Unit 2, while in Run Mode at 94% power, experienced a trip of the "B" channel of the RPS because of a high signal on an APRM channel (see LER #87-022-00). The APRM channels take LPRM output signals as their input. The root cause of the failure was determined to be the growth of metallic whiskers on the fission chamber outer electrode. An arc was generated between electrodes causing a transient short circuit which later burned away. After the whisker melted, the LPRM returned to normal operation.

On April 1, 1990, Duane Arnold Energy Center experienced an automatic reactor scram during reactor startup at 8% thermal power after an increase in flux greater than 15% was observed in APRM channels "C" and "D" (see LER #90-004-00). The intermediate cause of the event was a momentary LPRM spike cause by a transient short circuit between the anode and the cathode in the LPRM detector. This momentary spike in the LPRM output generated an APRM reading indicating a sudden power increase which in turn actuates the RPS logic that generates a scram. After the event, the affected LPRM was bypassed and the plant was started up. This event was not the only one of this nature at this plant. In February 1995, a similar event occurred during a controlled shutdown for a scheduled refueling outage. An upscale APRM reading occurred with the reactor shut down.

In December 1997, also at Dresden Unit 2, an unexpected full reactor scram occurred during an Instrumentation Surveillance due to an unexpected half-scram signal generated after RPS Channel "A" received a spiking LPRM reading in conjunction with an expected RPS Channel "B" half-scram signal (see LER #97-019-00). The unexpected LPRM reading was due to a short circuit caused by a whisker present within the detector. Actions taken after this event included complete I-V plots performance, and LPRM replacement with upgraded types.

On September 1, 1999, at South Texas Project Unit 2, a reactor pre-trip alarm occurred due to a low-low level indication in channel 2 of steam generator 2D while in Mode 1 at 100% power (see LER #1999-006-00). The failure of the input control relay that indicated the low-low level on the steam generator was attributed to tin whiskers. The corrective actions included the replacement of the relay.

On April 17, 2005, Millstone Unit 3 experienced a reactor trip from full power, and one of two trains of the Safety Injection and Main Steam Isolation actuated when low steam line pressure was sensed on one of four steam generators. The cause was the growth of a tin whisker between two components on a Westinghouse SSPS universal logic card, causing a short circuit to ground and triggering the single train of SI and the subsequent automatic reactor trip. A Westinghouse SSPS engineer revealed that the component where the whisker grew was a diode with a blue jacket.

On August 25, 2005, the NRC issued Information Notice 2005-25.[1878] which specifically discussed the inadvertent trip at Millstone Unit 3 caused by tin whiskers, notified licensees about recent operating experience related to the growth of tin whiskers in electronic circuits in nuclear power stations, and informed licensees to consider appropriate actions to avoid similar problems.

Safety Significance

Tin whiskers are electrically conductive, crystalline structures of tin that can grow from surfaces where tin (especially electroplated tin) is used as a final finish. Tin whiskers can range from several millimeters to as long as 10 mm in length. Their formation in electronic systems has been cited as the root cause for electronic system failure in military aviation and weapons systems, medical equipment, satellites, and in various instances within the electronics and power industries. Electronic component failures caused by tin whiskers have the potential to affect instrumentation and control (I&C) systems in nuclear power plants.

Until recently, this phenomenon has received relatively little attention in the nuclear industry. However, component failures attributed to tin whiskers have occurred at nuclear power plants, ranging in severity from spurious alarms and faulty signals in protection systems to reactor trips.

The use of electroplated tin has been driven by a movement to lead-free products. There is a possibility that the nuclear industry will be increasing the use of commercial off-the-shelf (COTS) -based I&C systems. Although formation of the tin whiskers have the potential to pose safety and reliability issues to all makers and users of high-reliability electronics and associated hardware, the electronics industry and COTS users have experience in documenting and tackling this problem.

Tin whisker incidents in nuclear power plants to date have not been safety significant. There have been no Accident Sequence Precursor (ASP) events, no impairment of redundant safety functions, no CCF events, no increasing trend of recent events, and no apparent decrease in reliability of systems/components due to tin whiskers. Lastly, this issue falls within the scope of the maintenance rule and no new rule/regulation is apparently needed.

Possible Solution

If the issue does not meet the criteria for pursuit, it could be dropped or be pursued as a compliance issue, as warranted. If further study is needed to better understand and establish the technical basis and safety impact of tin whiskers, NRR could make use of the User Need process to request RES assistance.

CONCLUSION

The low number of reported events associated with this issue, the lack of any increasing trend, the lack of any apparent decrease in reliability of systems or components due to tin whiskers, the existence of applicable regulatory requirements and programs (i.e., 10 CFR Part 21, the maintenance rule requirements, and the Reactor Oversight Program), and the issuance of Information Notice 2005-25[1878] to alert licensees collectively indicated that tin whiskers did not meet the requirements of NRC Management Directive 6.4. "Generic Issues Program," for further pursuit. Based on the considerations discussed above, RES recommended that the issue be returned to the originator to be evaluated for other possible options. As a result, the issue was DROPPED from further pursuit.[1879]

REFERENCES

1877. Memorandum to F. Eltawila from R. Barrett, "Proposed Generic Safety Issue: Tin Whiskers," August 18, 2005. [ML052080291]

1878. Information Notice 2005-25, "Inadvertent Reactor Trip and Partial Safety Injection Actuation Due to Tin Whisker," U.S. Nuclear Regulatory Commission, August 25, 2005. [ML052150404]

1879. Memorandum to B. Sheron from C. Lui, "Results of Initial Screening of Generic Isuue 200, 'Tin Whiskers,'" January 26, 2007. [ML070250399]

ISSUE 201: SMALL-BREAK LOCA AND LOSS OF OFFSITE POWER

DESCRIPTION

Historical Background

This issue was identified[1880] by NRR following an allegation that was submitted to the NRC in March 2006 that described a scenario in which a SBLOCA event in a PWR has progressed to the sump recirculation mode of core cooling with the ECCS aligned for high-pressure recirculation. The allegation also described the safety concern relative to "the plant response starting with this ECCS alignment, should a LOOP occur," and stated that for "some PWR designs and operating procedures, the plant response to a LOOP will cause the emergency diesel generators to start and loads to be automatically sequenced onto the emergency buses." This may cause the high-pressure ECCS pumps to be sequenced onto the emergency bus before the low-pressure pumps come onto the emergency bus – resulting in the high-pressure ECCS pumps starting with insufficient suction head, likely causing pump damage.

Safety Significance

In the event of a LOCA in a PWR, approximately 300,000 gallons of water are available in the RWST for post LOCA injection. Eventually a switchover to the recirculation mode is necessary after a LOCA as the RWST inventory is depleted. The timing of the switchover to recirculation mode depends on the size of the break, due to the flow from the centrifugal HPI pumps – increasing as the pressure decreases. For large breaks, the maximum injection pressure of the LPI pumps will be reached relatively quickly, and the HPI pumps are not needed for recirculation. The HPI pumps are needed during recirculation for small breaks. The scenario of interest is an SBLOCA event where the RWST is depleted and the ECCS is aligned in the HPI recirculation mode, approximately 4 hours into the event. At this lineup, the LPI pumps provide suction head to intermediate/or high head pumps, depending on the design. The potential safety concern is (the possibility) that if a LOOP occurs in this situation, the HPI pumps may be sequenced onto the EDG prior to the sequencing of the low head pumps. This would result in the HPI pumps being restarted with inadequate suction head, with likelihood of pump damage. It should be noted that there are potentially other scenarios, such as a transient relying upon primary system feed-and-bleed that result in conditions similar to the SBLOCA-LOOP scenario of interest.

SCREENING ANALYSIS

On detection of an SBLOCA, steam generators are used to cool the RCS. Centrifugal charging pumps are used for inventory control. Typically, HPI pumps (1500 psig discharge pressure) run with mini-flow valves to the RWST open – ensuring no damage. After a period of 2 to 4 hours, the RWST level will necessitate swap over to recirculation sump suction and allow recirculation (and closure of the HPI mini-flow valves).

If a LOOP occurs at this point, a valid recirculation signal will be present. Therefore, when the safety injection sequencer initiates after EDG breaker closing, the equipment's recirculation configuration will not change. Also, procedurally, operators are required to validate that the configuration is in its correct alignment for the plant condition (recirculation).

For most accident sequences, this issue is not expected to be a concern after about 4 hours following an SBLOCA; nevertheless, a 24-hour exposure period to account for uncertainties was used in this analysis. At this condition, the RCS temperature and pressure will be low enough to begin to transfer to long-term RHR. At this pressure and temperature (350°F, 350 psig), the HPI will inject enough water so that loss of mini-flow is not a problem. Similarly, since Westinghouse sometimes uses RHR to the SI pump piggy-back alignment to supply water from the sump to the RCS, and RHR spray is initiated to control the containment environment, RHR, mini-flow is not a concern in this situation.

Millstone 3 has a sub-atmospheric containment and relies on sump recirculation early in the event. As such, it is considered a bounding plant for this risk analysis. For most U.S. PWRs, following an SBLOCA, it is possible to use steam generator cooling to bring the plant pressure and temperature to a point at which RHR can be used for decay heat removal. In other words, most plants can use RHR without going to recirculation. Thus, the hypothetical frequency of going to recirculation at Millstone 3 is higher than for other plants. Also, based on our analysis of all SPAR PWR models, Millstone 3 has the same or a higher core damage frequency for the SBLOCA event than all others. Thus, it was used in the screening analysis.

At CE plants, LPI pumps are turned off with a recirculation signal, and the HPI pumps take suction directly from the sump. At Westinghouse plants, the low pressure pumps take suction from the sump and piggy-back the flow to the RCS through a combination of high pressure and low pressure pumps. For B&W plants following a LOCA, flow is initiated in the HPI and LPI systems from the borated water storage tank (BWST) to the reactor vessel. Flow is also initiated by the reactor building spray (RBS) system to the building spray headers. When the BWST inventory is depleted, recirculation from the reactor building sump is initiated by the operator for both the LPI flow and the reactor building sprays. If elevated RCS pressure requires piggyback operation, recirculation will also occur through the HPI System.

Long-term core cooling occurs by recirculation of injection water from the reactor building sump to the core through the LPI system for large breaks, or through the LPI system and the HPI system – in series – for small breaks where primary pressure remains above the shutoff head of the LPI pumps.

Although the HPI and the LPI systems operate to provide full protection across the entire range of break sizes, each system may operate individually and is initiated independently. The HPI system prevents uncovering of the core for small coolant piping leaks where high RCS pressure is maintained and delays uncovering of the core for intermediate sized breaks. The core flooding and LPI systems are designed to recover the core at intermediate to low RCS pressures and to insure adequate core cooling for break sizes ranging from intermediate breaks to the double-ended rupture of the largest pipe. The LPI system is also designed to permit long-term core cooling in the recirculation mode after a LOCA. The LPI system and the HPI system are designed to permit the recirculation mode at various system pressures following a LOCA. This is accomplished using LPI directly to the core for the low RCS pressures that exist following a large-break LOCA. The LPI discharge provides suction to the HPI in the "piggyback" mode of operation for higher RCS pressures which may occur following a small break. Pumped injection includes both HPI and LPI, each with separate and independent flow paths. One flow path from HPI system and one flow path from LPI system and the core flooding tanks are capable of providing 100% of necessary core injection.

During an SBLOCA, if system pressure remains above the LPI shutoff head upon depletion of the BWST, the LPI suction is manually realigned to the containment sump and the discharge is aligned to the suction of the HPI pumps (piggyback operation). Operation of the HPI system continues until the system operation is manually terminated.

To assure adequate makeup capability for the full range of SBLOCAs, each HPI pump is piped to all four injection lines. Redundant flow instrumentation and throttling globe valves were added to prevent pump run-out and to allow flow balancing among these four paths. Operation of this system does not normally depend on the operation of any other engineered safeguard; however, the system can be operated in series with the LPI system, in recirculation mode.

Frequency Estimate

A screening analysis was performed for sequences that require high pressure recirculation for continued core cooling during an SBLOCA event. The LOOP occurs during or after switchover to recirculation mode. The LPI system and the HPI system are designed to permit the recirculation mode at various system pressures following a LOCA. If the LPI pump loads before the HPI pump, the LPI discharge provides suction to the HPI in "piggyback" mode for higher RCS pressures which may exist following a small break. Therefore, there will be no pump runout and the HPI will have adequate suction flow. This concern is also applicable in sequences involving "feed and bleed" and stuck-open PORVs since plant behavior is equivalent to SBLOCA-initiated sequences.

Probability of a LOOP: We are concerned if a LOOP happens during the swapover to recirculation or during the recirculation. The LOOP could happen at random and could occur several times during recirculation. This random process can be represented by a Poisson process.

The probability of having failures (LOOPs) is:

$$P(n) = \frac{u^n e^{-u}}{n!}$$

$$P(n \geq 1) = 1 - P(0)$$

$$P(n \geq 1) = 1 - \frac{u^0 e^{-u}}{o!} = 1 - e^{-u}$$

where $u = (\lambda_{LOOP} + \lambda_{Weather} + \lambda_{Seismic})t = \lambda_{total}t = \lambda t$.

The LOOP frequency was 3.59E-02/year[1881] and exposure time was conservatively assumed to be 24 hours. Therefore:

$$u = \frac{3.59E-02}{year} * 24\ hours * \frac{1\ year}{8760\ hours} = 9.8E-05$$

Since u is much smaller than 1 and expanding the above equation using Taylor series, we get

$$P(n \geq 1) = 1 - (1 - u + \frac{u^2}{2!} - \frac{u^3}{3!} + \cdots).$$

If we ignore the small terms, we get:

$$P(n \geq 1) = u.$$

Therefore, the λt approximation for the occurrence probability of multiple LOOPs is equal to the exact expression.

Risk Calculation: A full train of ECCS is required for cooling during recirculation. It is conservatively assumed that at the occurrence of the LOOP the high pressure recirculation (HPR) is failed. The event trees for small LOCA and Transient are shown in figures below. We are only interested in sequences that involve recirculation, including those resulting from SBLOCA and consequential SBLOCA such as "feed and bleed" and stuck open PORVs. To determine the frequency of being in recirculation (i.e., being vulnerable to the GI-201 scenario), the probability of operator to initiate the HPR is set to one and the calculation was done using the SPAR (Standardized Plant Analysis Risk) model for Millstone 3 with a sequence cutoff frequency of 10^{-12}. The ΔCDF in Table 3.210-1 is the CDF increase for failed HPR function, which is equivalent to the frequency of entering HPR.

$$CDF = (Frequency\ being\ in\ \mathrm{Re}circulation) * P(LOOP)$$

Figures 3.201-1 and 3.201-2 show the transient and SBLOCA event trees. Other event trees referred to in Table 3.201-1 are similar to the transient (event) tree and are not included. The frequency of being in recirculation is the summation of being in recirculation during small LOCA, "feed and bleed" and stuck open PORVs modes (the scenarios are marked in the figures 1 and 2). Using the expression for the CDF – with frequency 4.1E-04 from Table 3.202-1, we get

CDF = (4.1E-04)/year * 9.8E-05 or, CDF = 4E-08/year.

This CDF is a bounding estimate of the risk associated with getting a LOOP during recirculation because:

(1) The analysis assumed all HPR pumps would start when the EDG started. This is not true if there is no SI signal present or if some HPR pumps are not designed to autostart.

(2) The EDG load sequencer may load differently for different operational modes. To ensure that we have bounded the issue, we assume that the most conservative loading pattern occurs, i.e., the loading sequence always causes all HPI pumps to fail.

(3) The analysis assumed that all HPR pumps will fail. They may survive until the LPI pumps are started and providing water.

(4) The analysis assumed that the plant was in HPR for 24 hours. The time in HPR varies for different sequences and, for the purpose of risk analysis, would always be less than 24 hours.

(5) The plant chosen for analysis has a simple AFW system and required HPR for successful mitigation of large and medium LOCAs.

(6) The plant chosen is one of the few plants that does not credit the ability to cool down and place the plant in RHR without going to recirculation.

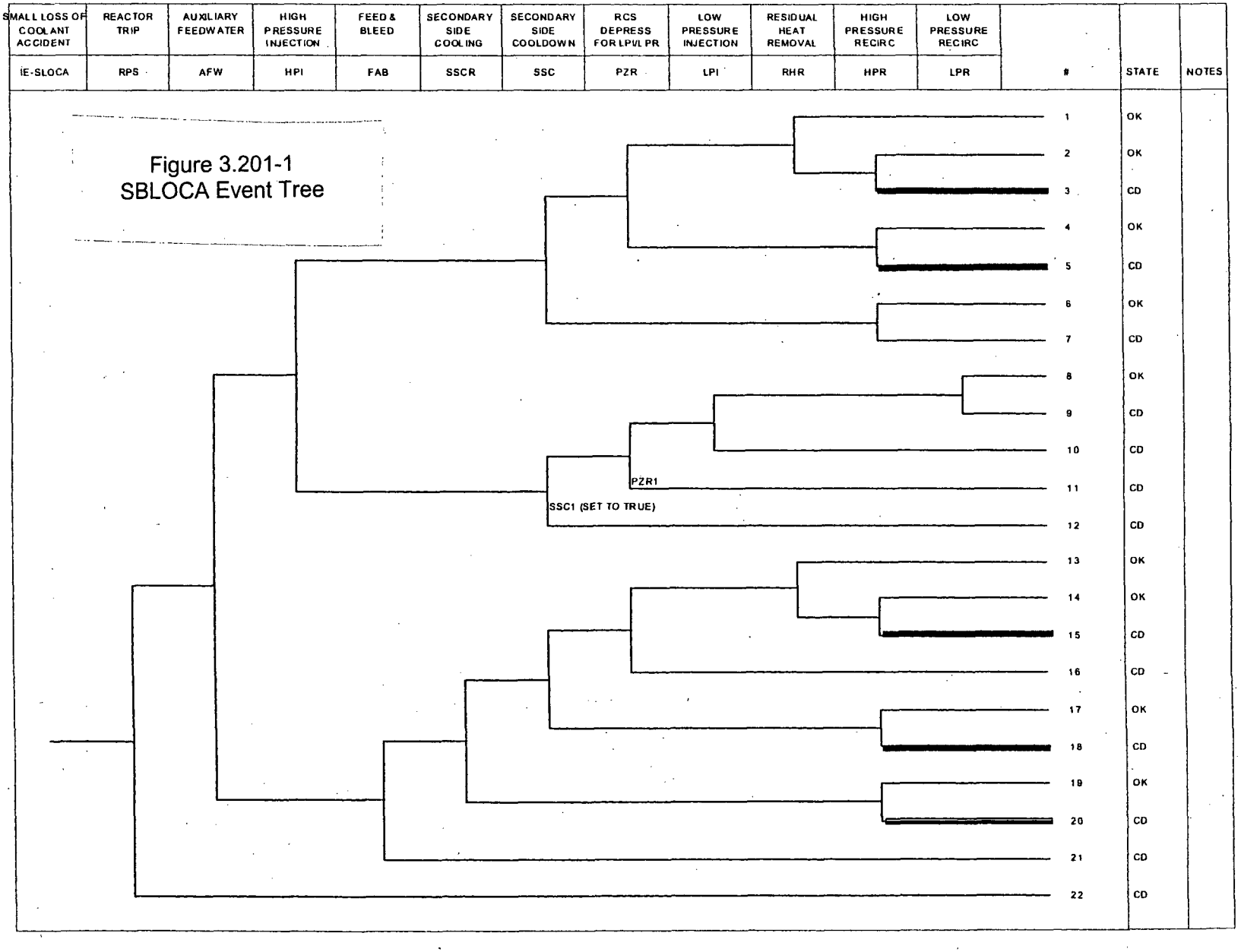


Figure 3.201-1
SBLOCA Event Tree

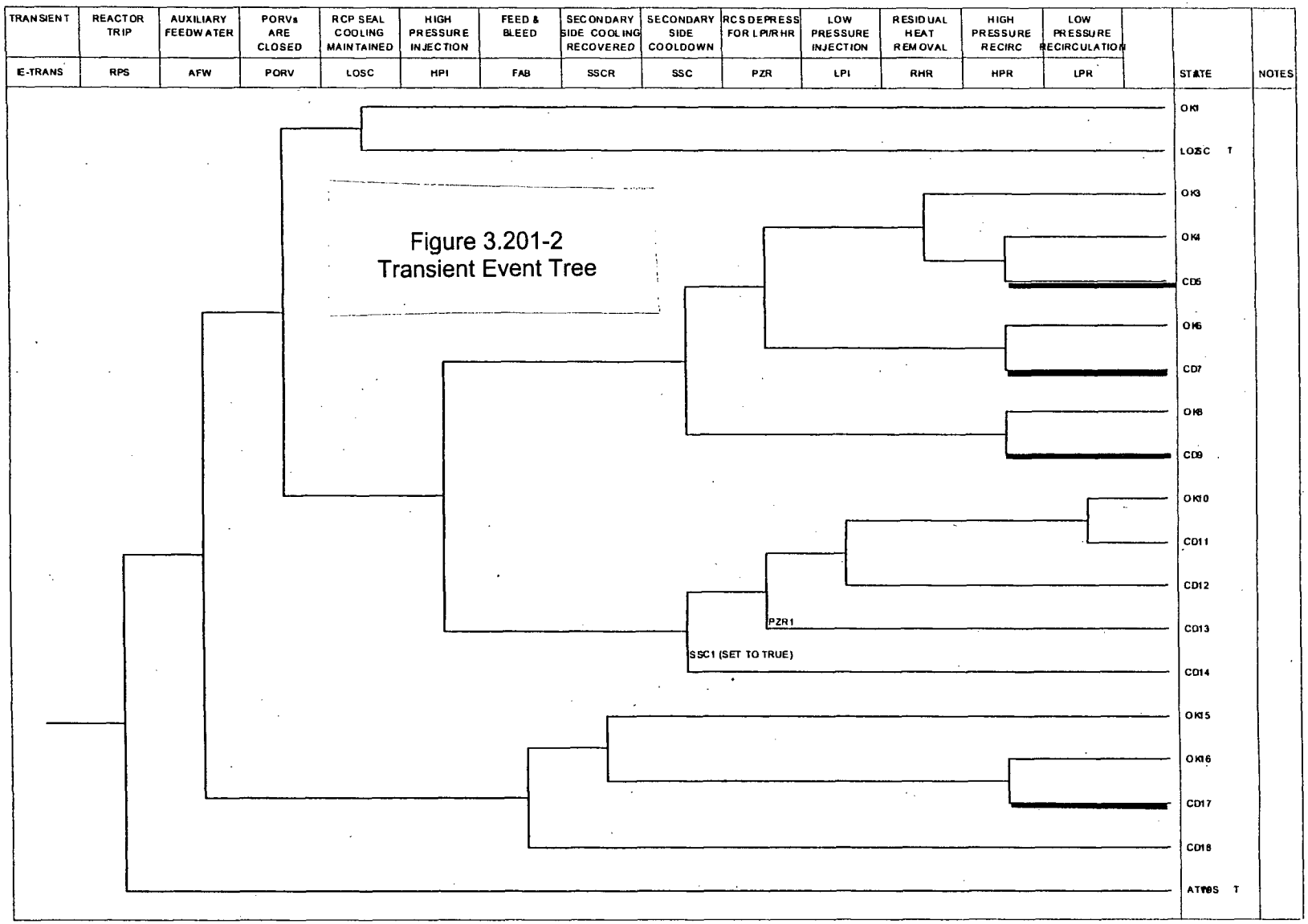


Figure 3.201-2
Transient Event Tree

With a CDF this low, calculations of LERF and person-rem/RY were not warranted. If it is assumed that all CDF ended in LERF (probability that CDF causes LERF is one), the LERF value is 4E-08, which is less than the LERF threshhold value of 1E-07.

Table 3.201-1

Event Tree	Sequence	CDF with HPR function Failed	Baseline CDF	ΔCDF
SLOCA	3	4.00E-04	2.40E-06	4.00E-04
TRANS	17	1.30E-06	7.50E-09	1.30E-06
SLOCA	5	1.10E-06	6.20E-09	1.10E-06
LOMFW	19	9.60E-07	2.00E-08	9.40E-07
LOCHS	19	8.60E-07	1.80E-08	8.40E-07
TRANS	5	5.10E-07	2.90E-09	5.00E-07
SLOCA	7	4.00E-07	2.20E-08	3.80E-07
LOMFW	18	1.90E-07	1.00E-09	1.90E-07
LOCHS	18	1.70E-07	9.10E-10	1.70E-07
LOCHS	5	6.50E-08	3.60E-10	6.50E-08
			Sum	4.05E-04

CONCLUSION

The negligible CDF increase associated with a SBLOCA and LOOP indicated that the issue did not require a technical assessment, in accordance with the guidelines of NRC Management Directive 6.4, "Generic Issues Program." Therefore, the issue was DROPPED from further pursuit.[1882]

REFERENCES

1880. Memorandum to F. Eltawila from M. Mayfield, "Small-Break Loss-of-Coolant Accident and Loss of Offsite Power Scenario (Allegation No. NRR-2006-A-0003)," August 7, 2006.

1881. NUREG/CR-6890, "Reevaluation of Station Blackout Risk at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, December 2005.

1882. Memorandum to B. Sheron from J. Monninger, "Results of Initial Screening of Generic Issue 201, 'Small-Break Loss-of-Coolant Accident with Loss of Offsite Power,'" March 28, 2007. [ML070820112]

ISSUE 202: SPENT FUEL POOL LEAKAGE IMPACTS

DESCRIPTION

Historical Background

Spent fuel pools (SFPs) at PWRs are Seismic Category I structures that contain borated water, maintain spent fuel temperatures, and provide radiation shielding. The SFPs are typically lined with stainless steel plates, joined by full-penetration welds, on the inner surface of the reinforced concrete SFP structures. The vast majority of SFPs also have leakage collection systems (channels embedded in concrete at weld seams) that are designed to collect any borated SFP water that might leak through the liner for processing as radioactive waste water. These systems also provide a means of monitoring SFP leakage.

This issue involves the potential for long-term leakage of borated water through the SFP liners, reactor cavities, and fuel transfer canals at PWRs, to degrade the concrete support structures and associated reinforcement steel (i.e., rebar) if the leak-off channels are clogged. If not properly drained, long-term leakage of borated water through the liners will accumulate, wet the concrete behind the liner, seep through cracks in the concrete, and reach the rebar within the concrete. If not corrected, over a long period of time this condition could degrade the concrete and rebar and potentially compromise the structural integrity.

In September 2002, Salem Unit 1 identified evidence of radioactive water leakage through the interior walls or penetrations in both the auxiliary and fuel handling buildings (FHB). Investigations of the leakage by the licensee in February 2003 revealed the radionuclide tritium in the groundwater near the FHB. Further licensee inspections identified long-term leakage of borated water from the SFP through cracks into the structural walls of the SFP. The licensee determined that the SFP leak-off channels (i.e., tell-tale drains) that drain borated SFP water collected between the pool liner and concrete walls were clogged. Since the Salem Unit 1 finding, similar conditions involving SFP structures were discovered at Indian Point and Seabrook (transfer canal). Based on these discoveries, NRR proposed[1883] that the issue be addressed as a generic issue, in accordance with NRC Management Directive (MD) 6.4, "Generic Issues Program."

Safety Significance

SFP leakage has the potential to degrade the integrity of SFP structures due to the adverse impact of borated water on concrete and its rebar. That is, borated water can seep through cracks in the concrete, reduce the inherent high alkaline environment of concrete, and possibly expose rebar in the vicinity of these cracks to mildly acidic conditions. The degree of potential degradation would be expected to vary among facilities as the SFP leak rates, concrete conditions (alkalinity, porosity, and cracking), and degree of leak-off channel flow restrictions vary. In accordance with their corrective action programs, the licensees of plants where this condition was discovered have evaluated the short-term and long-term safety concerns (e.g., potential degradation of the SFP basemat and potential reductions in design license margins of the SFP structures). In each case, the licensees' evaluations determined there was no immediate safety concern and no long-term degradation in the design license margins of the SFP structures from borated water leakage.[1883,1884] The potential adverse impacts of SFP leakage on groundwater and possible consequent effect of

the leakage on public health and safety was evaluated in 2006 as part of an NRC task force on ground water contamination from leakage of radioactive water from various sources at several licensee facilities.[1885]

Possible Solution

Information on the impact of borated water leakage through reinforced concrete structures at PWRs is limited. In many cases, the affected structures are subterranean and inaccessible, and leakage may not be readily detectable or repaired. The NRC issued generic communications to licensees to inform them of these conditions and the potential adverse consequences.[1886,1887] Licensees are required to maintain the design license basis of plant structures, systems, and components important to safety. The NRC monitors licensee performance and responds to conditions adverse to safety and quality at plants under the reactor oversight process (ROP). Accordingly, the existing regulatory framework provides sufficient means for licensees and the NRC to prevent, detect, and correct SFP leakage conditions that could adversely impact SFP structural integrity over the long term. If industry experience were to show an adverse trend developing in this area, then research to gain additional information on the impact of borated water leakage through reinforced concrete structures might be warranted.

SCREENING ANALYSIS

The PWR licensees that discovered SFP leakage conditions described above have taken corrective actions to preclude adverse impact to the long-term integrity of the Seismic Category I structures. For these few instances, the NRC used the ROP to review and assess the condition and the licensee actions to address the condition, and determined that the licensees' actions were adequate to maintain the plants' design license basis. The NRC also issued generic communications to inform plant licensees of these conditions and their potential adverse consequences. Accordingly, this issue was considered to be adequately monitored and addressed through existing regulatory programs. The following summarizes specific SFP leakage conditions discovered at Salem, Indian Point, and Seabrook plants.

Salem Unit 1: NRC and licensee reviews identified long-term leakage of the SFP through structural cracks to onsite groundwater. A visual inspection of the tell-tale leak-off drains revealed significant blockage with boric acid and calcium deposits. The licensee concluded that the blocked leak-off drains resulted in SFP water accumulation in the annulus area between the concrete pool wall and stainless steel liner, which eventually resulted in leakage of the water through construction joints and cracks in the SFP and FHB walls. The licensee initiated action to clear the drains to establish flow in the tell-tales and effectively drain the annulus area. Subsequently, the tell-tale drain rate held steady at about 100 gallons/day from liner leakage that was not specifically located or repaired. The leakage is being collected and processed as radioactive waste. Licensee monitoring of leakage did not proactively detect the leakage. Rather, a personnel contamination event prompted identification of the through-wall leakage. The NRC issued Information Notice No. 2004-05[1886] to inform the industry of this event.

The licensee initiated an assessment of potential long-term effects on the structure due to boric acid attack. The assessment covered an extended period of time to effectively simulate long-term boric acid effects on representative concrete specimens, including rebar. The licensee's assessment determined that the design license basis for the SFP structural integrity was maintained for the life of the plant, including during possible license extension. The licensee committed to routine monitoring of the leak-off flow of the tell-tale drain channels.

Indian Point 2: This plant does not have a SFP leak collection system, and has a history of leakage of borated water which may potentially affect the concrete and steel reinforcement. The licensee and NRC reviewed the SFP and concluded that the crack did not affect the SFP wall structural integrity. However, groundwater contamination was detected.

Seabrook: The licensee experienced sporadic leakage from its cask loading and transfer canal area during flood-up to support refueling. The area was drained after outages. The cause of this leakage was repaired, and a licensee analysis of the structural integrity of the canal did not identify any resulting adverse conditions. However, groundwater contamination was detected.

CONCLUSION

In addition to 10 CFR 50, Appendices A and B, industry standards existed to ensure a proper and timely evaluation of leaks that may develop in SFPs and the resultant impact on their concrete structures. The concerns of this issue involved long-term degradation of the structures and the adequacy of individual licensee corrective actions for this condition. Therefore, consistent with MD 6.4 and Generic Issues Program improvements described in SECY-07-0022,[1888] this issue was eliminated from further assessment as a generic issue. Additional research on the adverse impact of borated water on SFP concrete and its rebar might be warranted if the industry experienced an increasing adverse trend of this condition with the potential to degrade the design license basis of the structures.[1889]

REFERENCES

1883. Memorandum to F. Eltawila from M. Mayfield, "Proposed Generic Safety Issue: Spent Fuel Pool Leakage Limits," August 30, 2006. [ML062260014]

1884. Letter to W. Lewis (PSEG) from A. Burritt (NRC), "Salem Nuclear Generating Station – NRC Integrated Inspection Report 05000272/2007002 and 05000311/2007002," U.S. Nuclear Regulatory Commission, May 11, 2007. [ML071350075]

1885. "Liquid Radioactive Release Lessons Learned Task Force Final Report, U.S. Nuclear Regulatory Commission," September 1, 2006. [ML062650312]

1886. Information Notice No.2004-05, "Spent Fuel Pool Leakage to Onsite Groundwater," U.S. Nuclear Regulatory Commission, March 3, 2004. [ML040580454]

1887. Information Notice No. 2006-13, "Ground-Water Contamination Due to Undetected Leakage of Radioactive Water," U.S. Nuclear Regulatory Commission, July 10, 2006. [ML060540038]

1888. SECY-07-0022, "Status Report on Proposed Improvements to the Generic Issues Program," U.S. Nuclear Regulatory Commission, January 30, 2007. [ML063460239]

1889. Memorandum to B. Sheron from K. O'Brien, "Results of Initial Screening of Generic Issue 202, "Spent Fuel Pool Leakage Impacts,'" May 30, 2007. [ML071450125]

ISSUE 203: POTENTIAL SAFETY ISSUES WITH CRANES THAT LIFT SPENT FUEL CASKS

DESCRIPTION

This issue was identified by an NMSS engineer in December 2006 and addressed concerns about potential safety issues with cranes that are used to lift spent fuel casks at nuclear power plants. The areas of concern involved questions about the technical adequacy or programmatic effectiveness of existing regulatory programs and activities that are implemented through the NRC reactor oversight process (ROP). These were as follows:

(1) Cranes that do not conform to original design specifications: load drop analysis is not part of design basis, inadequate design basis documentation/information from parts vendors, and NRC inspection responsibilities for review of load drop analysis are unclear.

(2) Cranes that licensees modified without performing safety evaluation reviews required by 10 CFR 50.59, which may indicate that some licensees assume that these reviews are not required.

(3) Cranes that have had inadequate maintenance: overlooks important operating experience, invalidates single-failure proof capability, and tolerated by deficient NRC requirements for maintenance.

(4) Cranes that are not single-failure proof or lack credible validation for single-failure proof status have been used without adequate load path protection, or other mitigative measures, and this condition is not adequately considered in NRC's probabilistic risk assessment of crane events.

SCREENING ANALYSIS

RIS 2005-25[1890] describes the NRC's regulatory position with respect to the areas of concern identified in this issue, and identifies the design and inspection bases for cranes having the potential to impact SSCs important to safety. RIS 2005-25[1890] addresses: (1) single-failure-proof cranes, including guidelines for upgrading cranes to single-failure-proof status as well as for crane inspection, testing, and maintenance; and (2) conditions requiring load drop analyses. RIS 2005-25[1890] provides guidance on load drop analysis assumptions and methods, and incorporated operating experience from NUREG-1774,[1846] including results from staff work associated with Issue186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants," which formed the risk-informed basis for its use by the NRC to clarify NRC guidelines for licensees' programs for the control of heavy loads.

One possible net impact of the four areas of concern is that licensees have maintained inadequate control of changes to cranes over long periods of time. Sources of crane changes to consider include: design modifications, parts and information from vendors, aging, maintenance (or lack thereof), and incorporation of lessons learned from operating experience. Potential consequences from licensees' inadequate control of these changes may result in conditions where cranes no longer conform to original design specifications, conditions that invalidate load drop analysis or the single-failure-proof status of cranes. Licensees' control of the design basis of cranes that provide

functions important to safety is clearly within the purview of the ROP inspection procedures. In addition, guidance from RIS 2005-25[1890] describes the NRC's regulatory position that licensees can maintain adequate defense-in-depth through the use of single-failure-proof cranes, by providing various appropriate forms of load path protection, or by performing adequate load drop analyses that demonstrate acceptable consequences. Thus, licensees may choose among these alternatives to maintain adequate defense-in-depth (i.e., they are not specifically obligated to maintain original design basis, to perform load drop analysis, or to provide load path protection for cranes providing important to safety functions). Again, these considerations are part of the existing regulatory framework.

CONCLUSION

The four areas of concern identified in this issue are covered under existing inspection procedures of the ROP. The guidance provided to licensees in RIS 2005-25[1890] clearly describes NRC's regulatory position for licensees' control of heavy loads programs covering the areas of concern identified. Therefore, the areas of concern represented licensee compliance issues and, as such, were not suitable for further assessment in the Generic Issues Program, as delineated in MD 6.4. The concerns were entered into the ROP Feedback Program, in accordance with Inspection Manual Chapter 0801, and assigned the ROP Feedback Form Item Number 60854-1-1113. Thus, this issue was DROPPED from further pursuit as a generic issue.[1891]

REFERENCES

1846. NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," U.S. Nuclear Regulatory Commission, July 2003.

1890. Regulatory Issue Summary 2005-25, "Clarification of NRC Guidelines for Control of Heavy Loads," U.S. Nuclear Regulatory Commission, October 31, 2005. [ML05234485]

1891. Memorandum to M. Karmis from J. Foster, "Requested Generic Issue (GI)-203, Potential Safety Issues with Cranes that Lift Spent Fuel Casks," March 6, 2007. [ML070400565]

REFERENCES

* [Accession Numbers] are provided for easy retrieval of those documents that are accessible from the NRC Nuclear Documents System Advanced Design (NUDOCS/AD) or the Agencywide Documents Access and Management Systems (ADAMS)

1. SECY-81-513, "Plan for Early Resolution of Safety Issues," August 25, 1981. [8109140067]

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

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

4. NUREG-0572, "Review of Licensee Event Reports (1976-1978)," U.S. Nuclear Regulatory Commission, September 1979.

5. IE Circular No. 77-07, "Short Period During Reactor Startup," U.S. Nuclear Regulatory Commission, April 15, 1977. [9104240445]

6. IE Bulletin No. 79-12, "Short Period Scrams at BWR Facilities," U.S. Nuclear Regulatory Commission, May 31, 1979. [7906060168]

7. Memorandum for D. Ross from H. Richings, "RDA Statistical Analysis," June 17, 1975. [8105050833]

8. SECY-80-325, "Special Report to Congress Identifying New Unresolved Safety Issues," July 9, 1980. [8103180932]

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

10. NUREG/CR-3992, "Collection and Evaluation of Complete and Partial Losses of Off-Site Power at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1985.

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

12. Draft Regulatory Guide and Value/Impact Statement, Task SC 708-4, "Qualification and Acceptance Tests for Snubbers Used in Systems Important to Safety," U.S. Nuclear Regulatory Commission, February 1981. [9503290322]

13. NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, September 1980.

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

15. Nuclear Safety, Vol. 14, No. 3, 'Probability of Damage to Nuclear Components Due to Turbine Failure," S. H. Busch, 1973.

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

17. NUREG/CR-0255, "CONTEMPT-LT/028: A Computer Code for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission, March 1979.

18. Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," U.S. Atomic Energy Commission, May 1973. [7907100189]

19. NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," U.S. Nuclear Regulatory Commission, November 1979, (Rev. 1) July 1981.

20. Memorandum for R. Fraley from R. Mattson, "ACRS PWR Question Regarding Effect of Pressurizer Heater Uncovery on Pressurizer Pressure Boundary Integrity," November 5, 1979. [8004100530]

21. Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Rev. 1) June 1976. [7907100349]

22. Memorandum for H. Denton from C. Michelson, "BWR Jet Pump Integrity," May 23, 1980. [8006180872]

23. Memorandum for Distribution from W. Minners, "Generic Issues Screening Activity," September 30, 1981. [8110190695]

24. IE Bulletin No. 79-27, "Loss of Non-Class IE Instrumentation and Control Power System Bus During Operation," U.S. Nuclear Regulatory Commission, November 30, 1979. [7910250499]

25. Memorandum for F. Schroeder from T. Novak, "Application of SRP 15.4.6 Acceptance Criteria to Operating Reactors," December 12, 1980. [8102260305]

26. IE Information Notice 80-34, "Boron Dilution of Reactor Coolant During Steam Generator Decontamination," U.S. Nuclear Regulatory Commission, September 26, 1980. [8008220239]

27. Memorandum for R. Baer from A. Thadani, "RRAB Preliminary Assessment of the Reactor Coolant Pump Seal Failure Problem," December 12, 1980. [8103050765]

28. Memorandum for T. Novak from P. Check, "Spurious Automatic Switchover of ECCS from the Injection Mode to the Recirculation Mode," January 21, 1981. [8102280446]

29. Memorandum for T. Novak, et al., from A. Thadani, "Comparative Risk Assessment of ECCS Functional Switchover Options," April 1, 1981. [8104130436]

30. Memorandum for G. Lainas, et al., from P. Check, "BWR Scram Discharge System Safety Evaluation," December 1, 1980. [8101190514]

31. Memorandum for H. Denton from M. Ernst, "DST Evaluation of the Automatic Air Header Dump on Boiling Water Reactors," December 8, 1980. [8101230203]

32. NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff," U.S. Nuclear Regulatory Commission, November 1976.

33. Memorandum for B. Sheron from M. Srinivasan, "Probabilities and Consequences of LOCA/Loss of Offsite Power (LOOP) Sequences," April 13, 1982. [8206300420]

34. Memorandum for the Commissioners from W. Dircks, "Resolution of Issue Concerning Steam-line Break with Small LOCA," December 23, 1980. [8101150357]

35. Memorandum for S. Hanauer from T. Murley, "Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power," February 25, 1981. [8110190723]

36. Memorandum for C. Michelson from H. Denton, "Combination Primary/Secondary System LOCA," December 8, 1981. [8201200049]

37. NUREG/CR-2083, "Evaluation of the Threat to PWR Vessel Integrity Posed by Pressurized Thermal Shock Events," U.S. Nuclear Regulatory Commission, October 7, 1981.

38. "Generic Issues Tracking System Report," U.S. Nuclear Regulatory Commission, December 17, 1981.

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

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

41. Nuclear Safety, Vol. 11, No. 4, pp. 296-308, "Tornado Considerations for Nuclear Power Plant Structures Including the Spent Fuel Storage Pool," P. L. Doan, July 1970.

42. Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, April 1974. [7907100297]

43. Regulatory Guide 1.117, "Tornado Design Classification," U.S. Nuclear Regulatory Commission, June 1976, (Rev. 1) April 1978 [7907110104].

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

45. ANSI/ANS-58.8, "Time Response Design Criteria for Nuclear Safety Related Operator Actions," American Nuclear Society, 1984.

46. NRC Memorandum and Order CLI-80-21, May 27, 1980. [8007280084]

47. Memorandum for H. Denton from C. Michelson, "Degradation of Internal Appurtenances in LWR Piping," January 19, 1981. [8102020069]

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

49. ISA S67.04 (ANSI N719), Draft F, "Setpoints for Nuclear Safety-Related Instrumentation Used in Nuclear Power Plants," Instrument Society of America, May 22, 1979.

50. Draft Regulatory Guide and Value/Impact Statement, TASK IC 010-5, "Proposed Revision 2 to Regulatory Guide 1.105, Instrument Setpoints," U.S. Nuclear Regulatory Commission, December 1981. [8112230003]

51. Memorandum for C. Michelson from H. Denton, "BWR Jet Pump Integrity," July 11, 1980. [8009160606]

52. IE Bulletin No. 80-07, "BWR Jet Pump Assembly Failure," U.S. Nuclear Regulatory Commission, April 4, 1980. [8002280648]

53. SIL No. 330, "Jet Pump Beam Cracks," General Electric Company/BWR Product Service, June 9, 1980.

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

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, December 1975, (Rev. 1) August 1977 [8001240572], (Rev. 2) December 1980 [7912310387], (Rev. 3) May 1983 [8502060303].

56. Memorandum for R. Mattson, et al., from R. DeYoung, "Draft Report of Completion of Generic Activity A-34," March 28, 1979. [7904180060]

57. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," U.S. Nuclear Regulatory Commission, July 1979.

58. Memorandum for Commissioner Ahearne from H. Denton, "Instrumentation to Follow the Course of an Accident," September 4, 1979. [8005140362]

59. NUREG-0422, "SER for McGuire Nuclear Station Units 1 and 2," U.S. Nuclear Regulatory Commission, March 1978.

60. NUREG-0606, "Unresolved Safety Issues Summary," U.S. Nuclear Regulatory Commission, Vol. 7 No. 3, August 1985.

61. Memorandum for J. Murphy from B. Sheron, "Documentation of Generic Safety Issues on Degraded Voltage Protection," July 13, 1994. [9407250133]

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, (Supplement 1) May 1983, (Supplement 2) December 1983, (Supplement 3) September 1985, (Supplement 4) July 1986, (Supplement 5) July 1996.

65. Memorandum for H. Denton from R. Minogue, "Research Information Letter No. 117, 'Probability of Large LOCA Induced by Earthquakes,'" April 10, 1981. [8104220512]

66. Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Safety Guide 6)," U.S. Atomic Energy Commission, March 1971. [7907100064]

67. NRC Letter to All Power Reactor Licensees from B. Grimes, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978. [7910310568]

68. Memorandum for R. Fraley from K. Kniel, "Draft Task Action Plan for TASK A-45, Shutdown Decay Heat Removal Requirements," May 22, 1981. [8106010652]

69. NUREG-0880, "Safety Goals for Nuclear Power Plants: A Discussion Paper," U.S. Nuclear Regulatory Commission, February 1982, (Rev. 1) May 1983.

70. NUREG-0348, "Demographic Statistics Pertaining to Nuclear Power Reactor Sites," U.S. Nuclear Regulatory Commission, November 1979.

71. Memorandum for S. Hanauer from D. Eisenhut, "Proposed Recommendations for Improving the Reliability of Open Cycle Service Water Systems," March 19, 1982. [8204190039]

72. AEOD/C202, "Report on Service Water System Flow Blockages by Bivalve Mollusks at Arkansas Nuclear One and Brunswick," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, February 1982. [8202260124]

73. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. Atomic Energy Commission, March 23, 1962. [8202010067]

74. AEOD/C001, "Report on the Browns Ferry 3 Partial Failure to Scram Event on June 28, 1980," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, July 30, 1980. [8008140575]

75. Memorandum for H. Denton from C. Michelson, "Engineering Evaluation of the Salt Water System (SSWS) Flow Blockage at the Pilgrim Nuclear Power Station by Blue Mussels (Mytilus Edilus)," May 6, 1982. [8205130114]

76. NUREG/CR-2497, "Precursors to Potential Severe Core Damage Accidents: 1969-1979, A Status Report," U.S. Nuclear Regulatory Commission, June 1982.

77. IEEE Std 352, "IEEE Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Protection Systems," The Institute of Electrical and Electronics Engineers, Inc., 1976.

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

79. NUREG-0109, "Occupational Radiation Exposure at Light Water Cooled Power Reactors 1969-1975," U.S. Nuclear Regulatory Commission, August 1976.

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

81. IEEE Std 317, "Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," The Institute of Electrical and Electronics Engineers, Inc., 1976.

82. Regulatory Guide 1.63, "Electrical Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants," U.S. Nuclear Regulatory Commission, October 1973, (Rev. 1) May 1977, (Rev. 2) July 1978 [7907100240], (Rev. 3) February 1987.

83. Memorandum for K. Kniel from M. Srinivasan, "Generic Issues Tracking System (GITS) B-70; Power Grid Frequency Degradation and Effect on Primary Coolant Pumps," July 31, 1981. [8109140246]

84. Memorandum for C. Berlinger from E. Butcher, "Diagnostic Evaluation at Quad Cities Nuclear Power Station (TAC Nos. M88667/M88668)," June 8, 1994. [9406130249]

85. Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," U.S. Nuclear Regulatory Commission, June 1974. [8404100042, 9009140263]

86. Memorandum for T. Murley from M. Ernst, "Prioritization of New Requirements for PWR Feedwater Line Cracks," June 30, 1981. [8108030041]

87. Memorandum for R. Tedesco from T. Speis, "Supplement 2 to the Safety Evaluation Report for Grand Gulf Nuclear Station, Units 1 and 2," March 25, 1982. [8204080127]

88. Memorandum for All NRR Employees from H. Denton, "Regionalization of Selected NRR Functions," June 15, 1982. [9507280052]

89. NUREG/CR-1750, "Analysis, Conclusions, and Recommendations Concerning Operator Licensing," U.S. Nuclear Regulatory Commission, January 1981.

90. IEEE Std 323, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations," The Institute of Electrical and Electronics Engineers, Inc., 1974.

91. Regulatory Guide 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, November 1974 [7907100326], (Rev. 1) June 1984 [8407110475].

92. SECY-81-504, "Equipment Qualification Program Plan," August 20, 1981. [8109220949]

93. NUREG-0611, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plants," U.S. Nuclear Regulatory Commission, January 1980.

94. NUREG-0626, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications," U.S. Nuclear Regulatory Commission, January 1980.

95. NUREG-0635, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants," U.S. Nuclear Regulatory Commission, February 1980.

96. NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plant," U.S. Nuclear Regulatory Commission, January 1980.

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, November 1979.

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

99. ANSI/ANS 56.8, "Containment System Leakage Testing Requirements," American National Standards Institute, 1981.

100. Letter to General Electric Company from R. Tedesco (NRC), "Acceptance for Reference Topical Report NEDO-24712: Core Spray Design Methodology Confirmation Tests," January 30, 1981. [8103270130]

101. SECY-82-475, "Staff Resolution of the Reactor Coolant Pump Trip Issue," November 30, 1982. [8306030370]

102. NUREG/CR-0848, "Summary and Bibliography of Operating Experience with Valves in Light-Water-Reactor Nuclear Power Plants for the Period 1965-1978," U.S. Nuclear Regulatory Commission, August 1979.

103. Memorandum for M. Ernst from B. Fourest, "Review of ECCS Actuations on U.S. PWRs," June 11, 1981. [8107160006]

104. Memorandum for C. Michelson from H. Denton, "NRR Responses to AEOD Recommendations on the Arkansas Nuclear One Loss of Offsite Power Event of April 7, 1980," February 13, 1981. [8102270127]

105. Nuclear Power Experience, Volume BWR-2, Book-2, Section IX.A, Nuclear Power Experience, Inc., May 1982.

106. Memorandum for H. Denton from C. Michelson, "Lessons Learned from the Crystal River Transient of February 26, 1980 - Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power," May 23, 1980. [8009150079]

107. Memorandum for H. Denton from C. Michelson, "Potential for Unacceptable Interaction Between the Control Rod Drive System and Non-Essential Control Air System at the Browns Ferry Nuclear Plant," August 18, 1980. [8210120129]

108. Memorandum for R. Mattson from S. Hanauer, "Inadvertent Boron Dilution," March 10, 1982. [8205130278]

109. Memorandum for T. Murley from R. Mattson, "Inadvertent Boron Dilution," September 15, 1981. [8110080185]

110. NUREG/CR-2798, "Evaluation of Events Involving Unplanned Boron Dilutions in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1982.

111. Letter to R. Curtis (NRC) from N. DeMuth (LANL), "Analysis of Unmitigated Boron Dilution Events," November 18, 1981. [9507280099]

112. Transactions of the American Nuclear Society and the European Nuclear Society, 1984 International Conference on Nuclear Power - A Global Reality, November 11-16, 1984, Volume 47, pp. 254-256, "Analysis of Unmitigated Boron Dilution Events in Pressurized Water Reactors During Shutdown," B. Nassersharif, J. Wing (LANL).

113. EPRI NP-1194, "Operation and Design Evaluation of Main Coolant Pumps for PWR and BWR Service," Electric Power Research Institute, September 1979.

114. EPRI NP-2092, "Nuclear Unit Operating Experience, 1978 and 1979 Update," Electric Power Research Institute, October 1981.

115. NUREG/CR-3069, "Interaction of Electromagnetic Pulse with Commercial Nuclear Power Plant Systems," U.S. Nuclear Regulatory Commission, February 1983.

116. Letter to D. Switzer (Northeast Nuclear Energy Company) from G. Lear (NRC), "Millstone Nuclear Power Station Units Nos. 1 and 2," June 2, 1977. [9507280126]

117. NRC Letter to All Power Reactor Licensees (Except Humboldt Bay), "Adequacy of Station Electric Distribution Systems Voltages," August 8, 1979. [8005120354]

118. Memorandum for V. Stello from H. Denton, "Guidelines for Evaluating Qualification of Class IE Electrical Equipment in Operating Reactors," November 13, 1979. [7912190733]

119. Memorandum for K. Kniel from W. Gammill, "Need for Generic Issue B-67, Control and Monitoring of Radioactive Materials Released in Effluents and Performance of Radwaste Systems," November 20, 1981. [8201130493]

120. NUREG-0442, "Technical Report on Operating Experience with BWR Offgas Systems," U.S. Nuclear Regulatory Commission, April 1978.

121. IE Bulletin No. 78-03, "Potential Explosive Gas Mixture Accumulations Associated with BWR Offgas System Operations," U.S. Nuclear Regulatory Commission, February 8, 1978. [7909050232]

122. NRC Working Paper on Appendix J to 10 CFR Part 50, "Leak Tests for Primary and Secondary Containments of Light-Water-Cooled Nuclear Power Plants," May 17, 1982. [8401040228]

123. NRC Working Paper on Draft Regulatory Guide (MS021-5), "Containment System Leakage Testing," May 1982. [8405240527]

124. NUREG-0193, "FRANTIC - A Computer Code for Time-Dependent Unavailability Analysis," U.S. Nuclear Regulatory Commission, October 1977.

125. NRC Letter to the Northern States Power Company, "Order for Modification of License Concerning BWR Scram Discharge Systems," January 9, 1981. [8103250282]

126. Memorandum for R. Vollmer from T. Murley, "PWR Feedwater Line Cracks - New Regulatory Requirements," March 10, 1981. [8103250569]

127. Memorandum for H. Kouts from B. Rusche, "Quantification of Inherent Safety Margins in Seismic Design (SAFER-76-5)," June 7, 1976. [8003100435]

128. Memorandum for S. Levine from E. Case, "Quantification of Inherent Safety Margins in Seismic Design," June 16, 1977. [8103270937]

129. Memorandum for H. Denton from S. Levine, "RES Response to NRR User Request on Quantification of Inherent Safety Margins to Seismic Design," November 1, 1978. [8003100425]

130. Memorandum for S. Levine from H. Denton, "Seismic Safety Margins Research Program," February 23, 1979. [8003280774]

131. NUREG/CR-2015, "SSMRP Phase I Final Report," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1981, (Vol. 2) July 1981, (Vol. 3) January 1983, (Vol. 4) June 1982, (Vol. 5) August 1981, (Vol. 6) October 1981, (Vol. 8) September 1984, (Vol. 9) September 1981, (Vol. 10) July 1981.

132. Memorandum for R. Minogue from H. Denton, "NRR Research Needs in Seismic Analyses Methodology," April 8, 1982. [8706040051]

133. NUREG-0784, "Long Range Research Plan FY 1984-1988," U.S. Nuclear Regulatory Commission, August 1982.

134. SECY-82-53, "Possible Relocation of Design Controlling Earthquakes in the Eastern U.S.," U.S. Nuclear Regulatory Commission, February 5, 1982. [8203050077]

135. NUREG-0484, "Methodology for Combining Dynamic Responses," U.S. Nuclear Regulatory Commission, September 1978, (Rev. 1) May 1980.

136. Memorandum for W. Minners from R. Bosnak, "Comments on Generic Issue B-6," August 26, 1982. [8209280601]

137. Memorandum for W. Minners from F. Schauer, "Generic Issue B-6," September 2, 1982. [8401170090]

138. NUREG/CR-1924, "FRANTIC II - A Computer Code for Time Dependent Unavailability Analysis," U.S. Nuclear Regulatory Commission, April 1981.

139. Letter to W. Dickhoner (The Cincinatti Gas & Electric Company) from A. Giambusso (NRC), December 18, 1972. [8709240215]

140. Memorandum for R. Frahm from R. Emrit, "Summary Report on a Risk Based Categorization of NRC Technical and Generic Issues," June 30, 1989. [9507280169]

141. Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," U.S. Nuclear Regulatory Commission, June 1981 [8108040038], (Rev. 1) February 1983 [8808230046].

142. NRC Letter to Alabama Power Company, "Containment Purging During Normal Plant Operation," November 28, 1978. [7812140364]

143. NRC Letter to Nebraska Public Power District, "Containment Purging and Venting during Normal Operation," October 22, 1979. [7911190034]

144. Regulatory Guide 1.75, "Physical Independence of Electric Systems," U.S. Nuclear Regulatory Commission, February 1974, (Rev. 1) January 1975 [8605300425], (Rev. 2) September 1978 [7810050139].

145. Memorandum for D. Thatcher from R. Emrit, "Interim Criteria for Evaluating Steel Containment Buckling," June 21, 1982. [9507280196]

146. Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," U.S. Nuclear Regulatory Commission, September 1977, (Rev. 1) May 1981 [8106120320].

147. Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," U.S. Nuclear Regulatory Commission, December 1971, (Rev. 1) June 1975, (Rev. 2) May 1976 [7907100101].

148. "Memorandum of Agreement between the Institute of Nuclear Power Operations and the U.S. Nuclear Regulatory Commission," (Rev. 1) April 1, 1982. [8207010053]

149. Memorandum for J. Funches from R. Mattson, "Comments on Prioritization of Licensing Improvement Issues," February 2, 1983. [8401170099]

150. Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," U.S. Atomic Energy Commission, May 1973. [7907100191]

151. SECY-82-111, "Requirements for Emergency Response Capability," March 11, 1982. [8203180409]

152. NUREG/CR-2417, "Identification and Analysis of Human Errors Underlying Pump and Valve Related Events Reported by Nuclear Power Plant Licensees," U.S. Nuclear Regulatory Commission, February 1982.

153. NUREG/CR-3621, "Safety System Status Monitoring," U.S. Nuclear Regulatory Commission, March 1984.

154. NRC Letter to Construction Permit Holders of B&W Designed Facilities, October 25, 1979.

155. NUREG-0667, "Transient Response of Babcock & Wilcox Designed Reactors," U.S. Nuclear Regulatory Commission, May 1980.

156. Memorandum for H. Denton from D. Eisenhut, "NUREG-0667, Transient Response of Babcock & Wilcox Designed Reactors, Implementation Plan," June 3, 1981. [8510070181]

157. Memorandum for D. Eisenhut from G. Lainas, "Status Report on Implementation of NUREG-0667 Category A Recommendations," December 15, 1981. [8201190550]

158. Memorandum for H. Denton from R. Mattson, "Review of Final Report of the B&W Reactor Transient Response Task Force (NUREG-0667)," August 8, 1980. [8010270109, 8010240413]

159. Memorandum for S. Hanauer from R. Mattson, "Design Sensitivity of B&W Reactors, Item II.E.5.1 of NUREG-0660," February 26, 1982. [8203170235]

160. Memorandum for R. Mattson from S. Hanauer, "Design Sensitivity of B&W Reactors," June 21, 1982. [8207150195]

161. NUREG/CR-1250, "Three Mile Island: A Report to the Commission and to the Public," U.S. Nuclear Regulatory Commission, January 1980.

162. NRC Letter to All Light Water Reactors, "Containment Purging and Venting During Normal Operation - Guidelines for Valve Operability," September 27, 1979. [9705190209]

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

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

165. Memorandum for M. Srinivasan from O. Parr and B. Sheron, "Generic Issue (GI) A-30, Adequacy of Safety Related DC Power Supplies, Development of Licensing Guidelines," March 12, 1982. [8401170018]

166. Memorandum for E. Case from R. Mattson, "Task No. D-3 Control Rod Drop Accident (BWRs)," March 6, 1978. [8001140319]

167. Federal Register Notice 44 FR 68307, "Decommissioning and Site Reclamation of Uranium and Thorium Mills," November 28, 1979.

168. NRC Letter to Arkansas Power & Light Company, "Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves," (Docket No. 50-313), April 20, 1981. [8104270071]

169. Memorandum for A. Ungaro (NRC) from F. Clark (ORNL), "Report on Standards and Requirements for Electrical Penetration Assemblies for Nuclear Reactor Containment Structures," December 13, 1978. [9507280225]

170. NUREG/CR-1345, "Nuclear Power Plant Design Concepts for Sabotage Protection," U.S. Nuclear Regulatory Commission, 1981.

171. Bulletin of the Atomic Scientists, Vol. 32, No. 8, pp. 29-36, "Nuclear Sabotage," M. Flood, October 1976.

172. Federal Register Notice 43 FR 10370, "[10 CFR Parts 30, 40, 50, and 70] Decommissioning Criteria for Nuclear Facilities, Advance Notice of Proposed Rulemaking," March 13, 1978.

173. NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," U.S. Nuclear Regulatory Commission, August 1988.

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

175. ZAR-791030-01, "Report of the President's Commission on the Accident at Three Mile Island," J. G. Kemeny, et al., November 30, 1979.

176. Memorandum for J. Ahearne from M. Carbon, "Comments on the Pause in Licensing," December 11, 1979. [8001080218]

177. Memorandum for N. Moseley from J. Allan, "Operations Team Recommendations - IE/TMI Unit 2 Investigation," October 16, 1979. [8007160815]

178. EPRI NP-801, "ATWS: A Reappraisal, Part III, Frequency of Anticipated Transients," Electric Power Research Institute, July 1978.

179. NUREG-0020, "Licensed Operating Reactors, Status Summary Report," U.S. Nuclear Regulatory Commission, (Vol. 6, No. 2) February 1982.

180. NUREG-0580, "Regulatory Licensing Status Summary Report," U.S. Nuclear Regulatory Commission, (Vol. 11, No. 5) June 1982.

181. SECY-82-155, "Public Law 96-295, Section 307(B), Study of the Feasibility and Value of Licensing of Nuclear Plant Managers and Senior Licensee Officers," April 12, 1982. [8205050080]

182. NUREG/CR-0672, "Technology, Safety, and Costs of Decommissioning a Reference Boiling Water Reactor Power Station," U.S. Nuclear Regulatory Commission, June 1980.

183. NUREG-0153, "Staff Discussion of 12 Additional Technical Issues Raised by Responses to the November 3, 1976 Memorandum from Director, NRR to NRR Staff," December 1976.

184. Memorandum for R. Vollmer from D. Eisenhut, "Transmittal of Report on Threaded Fastener Experience in Nuclear Power Plants," August 25, 1982. [8209210482]

185. Memorandum for H. Denton from C. Michelson, "AEOD Report on the St. Lucie Natural Circulation Cooldown on June 11, 1980," December 24, 1980. [8101120011]

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

187. NUREG/CR-2300, "PRA Procedures Guide," U.S. Nuclear Regulatory Commission, (Vols. 1 and 2) January 1983.

188. NUREG/CR-2644, "An Assessment of Offsite, Real-Time Dose Measurements for Emergency Situations," U.S. Nuclear Regulatory Commission, April 1982.

189. Memorandum for K. Goller from R. Mattson, "Proposed Changes to Regulatory Guide 1.97," July 29, 1982. [8208060339]

190. Memorandum for W. Dircks from S. Chilk, "Staff Requirements - Affirmative Session, 11:50 a.m., Friday July 16, 1982," July 20, 1982. [8208040248, 8209010068]

191. NUREG-0799, "Draft Criteria for Preparation of Emergency Operating Procedures," U.S. Nuclear Regulatory Commission, June 1981.

192. NUREG-0899, "Guidelines for Preparation of Emergency Operating Procedures - Resolution of Comments on NUREG-0799," U.S. Nuclear Regulatory Commission, September 3, 1982.

193. Memorandum for J. Martin, et al., from L. Shao, "Division Review Request: Amendments to 10 CFR Parts 30, 40, 50, 70, and 72 on Decommissioning Criteria for Nuclear Facilities," July 7, 1982. [8209140007]

194. IEEE Std 500, "IEEE Guide to the Collection and Presentation of Electrical, Electronic, and Sensing Component Reliability Data for Nuclear Power Generating Stations," The Institute of Electrical and Electronics Engineers, Inc., 1977.

195. Memorandum for E. Adensam from R. Riggs, "Status on Reactor Coolant Pump Seal Degradation Review," December 9, 1980. [8102280212]

196. Memorandum for H. Denton from S. Hanauer, "Preliminary Ranking of NRR Generic Safety Issues," March 26, 1982. [8204280036]

197. Federal Register Notice 45 FR 37011, "Decommissioning of Nuclear Facilities Regulation (10 CFR Parts 30, 40, 50, and 70)," May 30, 1980.

198. NUREG-0698, "NRC Plans for Cleanup Operations at Three Mile Island Unit 2," U.S. Nuclear Regulatory Commission, July 1980.

199. NUREG-0683, "Final Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting from the March 28, 1979 Accident at Three Mile Island Nuclear Station, Unit 2," U.S. Nuclear Regulatory Commission, March 1981.

200. IEEE Std 603, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations," The Institute of Electrical and Electronics Engineers, Inc., 1980.

201. IE Bulletin No. 80-24, "Prevention of Damage Due to Water Leakage Inside Containment (October 17, 1980 Indian Point 2 Event)," U.S. Nuclear Regulatory Commission, November 21, 1980. [8008220270]

202. Memorandum for G. Cunningham et al., from K. Goller, "Proposed Amendment to Part 50 on Radiation Programs, Including ALARA," September 10, 1982. [8209300046]

203. SECY-82-157A, 'Status Report on the NRR Investigation of the Effects of Electromagnetic Pulse (EMP) on Nuclear Power Plants," July 16, 1982. [8205050108]

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. Rubenstein from M. Ernst, "Proposed Position Regarding Containment Purge/Vent Systems," April 17, 1981. [8105260251]

207. IE Bulletin No. 81-03, "Flow Blockage of Cooling Water to Safety System Components by CORBICULA SP. (Asiatic Clam) and MYTILUS SP. (Mussel)," U.S. Nuclear Regulatory Commission, April 10, 1981. [8011040289]

208. Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," U.S. Nuclear Regulatory Commission, June 1973, (Rev. 1) July 1976, (Rev. 2) March 1978 [7907100211].

209. Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1978, (Rev. 1) October 1979 [7911090195].

210. NUREG-0885, "U.S. Nuclear Regulatory Commission Policy and Planning Guidance," U.S. Nuclear Regulatory Commission, (Issue 1) January 1982, (Issue 2) January 1983, (Issue 3) January 1984, (Issue 4) February 1985, (Issue 5) February 1986, (Issue 6) September 1987.

211. Federal Register Notice 46 FR 764, "NRC Policy Statement on Cleanup of the Three Mile Island Plant," May 1, 1981.

212. NUREG-0772, "Technical Bases for Estimating Fission Product Behavior During LWR Accidents," U.S. Nuclear Regulatory Commission, June 1981.

213. Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors," U.S. Nuclear Regulatory Commission, November 1970, (Rev. 1) June 1973, (Rev. 2) June 1974 [7907100054].

214. Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, November 1970, (Rev. 1) June 1973, (Rev. 2) June 1974 [7907100058].

215. Memorandum for E. Sullivan from R. Bosnak, "Generic Issues," September 17, 1982. [8312290147]

216. Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, August 1976, (Rev. 1) August 1977. [7907100397]

217. NUREG/CR-0660, "Enhancement of On-site Emergency Diesel Generator Reliability," U.S. Nuclear Regulatory Commission, February 1979.

218. Memorandum for D. Eisenhut, et al., from S. Hanauer, "Diesel Generator Reliability at Operating Plants," May 6, 1982. [8205280490]

219. Memorandum for S. Hanauer from R. Mattson, "Request for Prioritization of BWR Main Steam Line Isolation Valve Leakage as a Generic Issue," July 30, 1982. [8209130423]

220. IE Bulletin No. 82-23, "Main Steam Isolation Valve (MSIV) Leakage," U.S. Nuclear Regulatory Commission, July 16, 1982. [8204210393]

221. Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category 1 Fluid System Components," U.S. Atomic Energy Commission, May 1973. [7907100195]

222. NUREG-0479, "Report on BWR Control Rod Drive Mechanical Failures," U.S. Nuclear Regulatory Commission, January 1979.

223. NUREG-0462, "Technical Report on Operating Experience with BWR Pressure Relief Valves," U.S. Nuclear Regulatory Commission, July 1978.

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, (Rev. 1) November 1980.

225. Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)," U.S. Nuclear Regulatory Commission, November 1972, (Rev. 1) February 1977, (Rev. 2) February 1978 [7907100144].

226. Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1971, (Rev. 1) September 1975 [8801130111], (Rev. 1-R) May 1977 [7907100073], (Rev. 2) April 1987 [8907180147].

227. NUREG/CR-0130, "Technology, Safety, and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station," U.S. Nuclear Regulatory Commission, June 1978.

228. SECY-81-450, "Development of a Selective Absorption System Emergency Unit," July 27, 1981. [8108140094]

229. Memorandum for T. Speis from R. Houston, "Containment Venting and Purging - Completion of TMI Action Plan Item II.E.4.4(4)," March 3, 1982. [8401170023, 8203240149]

230. Memorandum for R. Mattson from T. Speis, "Containment Purge and Venting -Completion of TMI Action Plan Item II.E.4.4(5)," April 9, 1982. [8204260021]

231. Memorandum for W. Dircks from R. Mattson, "Status Report on Containment Purge Evaluations," May 13, 1982. [8401170021]

232. SECY-81-168B, "Response to Commission Request for Information on Financial Considerations in Licensing Proceedings," July 13, 1981. [8107310227]

233. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1972, (Rev. 1) September 1974, (Rev. 2) June 1975, (Rev. 3) February 1976.

234. Federal Register Notice 47 FR 9987, "10 CFR Part 2, General Statement of Policy and Procedure for Enforcement Actions," March 9, 1982.

235. Memorandum for H. Denton from R. DeYoung, "TMI Action Plan Items Still Pending," June 10, 1982. [8401170101]

236. Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan - Completed Items," June 30, 1982. [8208110023]

237. SECY-80-366, "NRC Legislative Program for 97th Congress," August 6, 1980. [8101050634]

238. Memorandum for Chairman Hendrie, et al., from W. Dircks, "Memorandum of Agreement with INPO and NSAC on a Cooperative Relationship for the Collection and Feedback of Operational Data," June 16, 1981. [8106260511, 8106260514]

239. Memorandum for W. Dircks from V. Stello, "TMI Action Plan - Status Report," December 19, 1980. [8205260193]

240. SECY-81-153, "Nuclear Data Link," March 11, 1981. [8103240155]

241. NUREG/CR-1440, "Light Water Reactor Status Monitoring During Accident Conditions," U.S. Nuclear Regulatory Commission, May 1980.

242. NUREG/CR-2100, "Boiling Water Reactor Status Monitoring During Accident Conditions," U.S. Nuclear Regulatory Commission, May 1981.

243. NUREG/CR-2278, "Light Water Reactor Engineered Safety Features Status Monitoring," U.S. Nuclear Regulatory Commisson, October 1981.

244. NUREG/CR-2147, "Nuclear Control Room Annunciators," U.S. Nuclear Regulatory Commission, October 1981.

245. Memorandum for H. Denton, et al., from R. Minogue, "Research Information Letter #RIL-124, 'Control Room Alarms and Annunciators,'" October 20, 1981. [8111130045]

246. RIL-98, "Light Water Reactor Status Monitoring During Accident Conditions," U.S. Nuclear Regulatory Commission, August 18, 1980. [8104230867]

247. NUREG/CR-5669, "Evaluation of Exposure Limits to Toxic Gases for Nuclear Reactor Control Room Operators," U.S. Nuclear Regulatory Commission, July 1991.

248. Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan - Completed Items," December 28, 1981. [8205260197]

249. NUREG/CR-6210, "Computer Codes for Evaluation of Control Room Habitability (HABIT)," U.S. Nuclear Regulatory Commission, June 1996.

250. SECY-81-440, "Nuclear Power Plant Staff Working Hours," July 22, 1981. [8107290183]

251. SECY-79-330E, "Qualifications of Reactor Operators," July 30, 1979. [7910020256, 7910020279]

252. NRR-80-117, "Study of Requirements for Operator Licensing," February 4, 1982. [8203180234]

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

254. Letter to N. Palladino (NRC) from M. Udall (Chairman, Committee on Interior and Insular Affairs, U.S. House of Representatives), June 4, 1982. [8207120246]

255. Letter to M. Udall (Chairman, Committee on Interior and Insular Affairs, U.S. House of Representatives) from N. Palladino (NRC), June 30, 1982. [8206130067]

256. Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan - Completed Items," June 2, 1982. [8401170114]

257. NUREG-0728, "Report to Congress - NRC Incident Response Plan," U.S. Nuclear Regulatory Commission, September 1980.

258. NUREG-0845, "Agency Procedure for the NRC Incident Response Plan," U.S. Nuclear Regulatory Commission, March 1982.

259. Memorandum for J. Sniezek from J. Taylor, "TMI Action Plan Item II.J.1.2, Modification of Vendor Inspection Program," October 13, 1982. [8301050485]

260. SECY-81-494, "Integrated Operational Experience Reporting System," August 18, 1981. [8109110483]

261. Federal Register Notice 46 FR 53594, "NRC Regulatory Agenda," October 29, 1981.

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.

265. Memorandum for the Commissioners from W. Dircks, "Enforcement Policy," March 18, 1980. [8005160508]

266. SECY-80-139A, "NRC Enforcement Program," August 27, 1980. [8009180277]

267. Memorandum for R. Purple from R. Minogue, "TMI Action Plan," October 24, 1980. [8011120511]

268. Memorandum for W. Dircks from V. Stello, "Assignment of Resident Inspectors to Nuclear Steam System Suppliers and Architect-Engineers," September 14, 1981. [8111030559]

269. IE Circular No. 80-15, "Loss of Reactor Coolant Pump Cooling and Natural Circulation Cooldown," U.S. Nuclear Regulatory Commission, June 20, 1980. [8005050073]

270. Memorandum for C. Michelson from H. Denton, "Report on St. Lucie Natural Circulation Cooldown," April 6, 1981. [8104150248]

271. Memorandum for J. Taylor from E. Beckjord, "Closeout of TMI Action Plan Task I.D.5(5), Research on Disturbance Analysis Systems," April 17, 1995. [9705190216]

272. Memorandum for J. Gagliardo from D. Eisenhut, "Potential Failure of Turbine Driven Auxiliary Feedwater Pump Steam Supply Line - Fort Calhoun," October 8, 1982. [8210290122]

273. Memorandum for H. Denton from C. Michelson, "Technical Review Report, Postulated Loss of Auxiliary Feedwater System Resulting from Turbine Driven Auxiliary Feedwater Pump Steam Supply Line Rupture," February 16, 1983. [8303040296]

274. Letter to G. Knighton (NRC) from K. Baskin (Southern California Edison Company), "Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station Units 2 and 3," October 29, 1982. [8211020483]

275. NUREG/CR-1614, "Approaches to Acceptable Risk: A Critical Guide," U.S. Nuclear Regulatory Commission, September 1980.

276. NUREG/CR-1539, "A Methodology and a Preliminary Data Base for Examining the Health Risks of Electricity Generation from Uranium and Coal Fuels," U.S. Nuclear Regulatory Commission, August 1980.

277. NUREG/CR-1930, "Index of Risk Exposure and Risk Acceptance Criteria," U.S. Nuclear Regulatory Commission, February 1981.

278. NUREG/CR-1916, "A Risk Comparison," U.S. Nuclear Regulatory Commission, February 1981.

279. NUREG/CR-2040, "A Study of the Implications of Applying Quantitative Risk Criteria in the Licensing of Nuclear Power Plants in the U.S.," U.S. Nuclear Regulatory Commission, March 1981.

280. SECY-80-331, "NRC Training Program," July 14, 1980. [8009100166]

281. Memorandum for H. Denton, et al., from C. Michelson, "Case Study Report - Failure of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand," August 4, 1982. [8208240007]

282. Memorandum for C. Michelson from H. Denton, "AEOD Preliminary Report on Failures of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand," September 23, 1982. [8210290150]

283. Federal Register Notice 47 FR 36099, "Executive Order 12379 of August 17, 1982, Termination of Boards, Committees, and Commissions," August 19, 1982.

284. Letter to N. Palladino (NRC) from G. Keyworth (OSTP), July 21, 1982. [9705190213]

285. Letter to G. Keyworth (OSTP) from N. Palladino (NRC), July 23, 1982. [9705190203]

286. Letter to T. Pestorius (OSTP) from R. Minogue (NRC), August 27, 1982. [9104170201]

287. SECY-81-600A, "Revised General Statement of Policy and Procedure for Enforcement Actions," December 14, 1981. [8201190600]

288. NEDO-10174, "Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor," General Electric Company, October 1977, (Rev. 1) May 1980.

289. NUREG/CR-2075, "Standards for Psychological Assessment of Nuclear Facility Personnel," U.S. Nuclear Regulatory Commission, July 1981.

290. NUREG/CR-2076, "Behavioral Reliability Program for the Nuclear Industry," U.S. Nuclear Regulatory Commission, July 1981.

291. Memorandum for E. Jordan, et al., from R. Bernero, "Proposed Rule Review Request - 10 CFR Part 21, 'Reporting of Defects and Noncompliance,'" September 28, 1982. [8210150634]

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

293. Federal Register Notice 47 FR 18508, "NRC Regulatory Agenda," April 29, 1982.

294. Federal Register Notice 47 FR 48960, "NRC Regulatory Agenda," October 28, 1982.

295. BNL-NUREG-31940, "Postulated SRV Line Break in the Wetwell Airspace of Mark I and Mark II Containments - A Risk Assessment," Brookhaven National Laboratory, October 1982. [8212070471]

296. Letter to T. Kress from J. Taylor, "Resolution of Generic Safety Issue 83, 'Control Room Habitability,'" September 13,1995. [9605130222, 9605150092]

297. Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan - Completed Item," October 29, 1982. [8401170104]

298. Memorandum for W. Dircks from V. Stello, "TMI Action Plan - Status Report," April 17, 1981. [8205260194]

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, (Vol. 1) January 1982, (Vol. 2) May 1982.

301. Memorandum for R. Emrit from P. Goldman, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," December 29, 1982. [8312290171]

302. Memorandum for H. Denton from C. Michelson, "Operational Restrictions for Class IE 120 VAC Vital Instrument Buses," July 15, 1980. [8210120114]

303. Memorandum for C. Michelson from H. Denton, "LCO for Class IE Vital Instrument Buses in Operating Reactors," September 29, 1980. [8010220035]

304. UCID-19469, "120 VAC Vital Instrument Buses and Inverter Technical Specifications," Lawrence Livermore National Laboratory, October 28, 1982. [8405180177]

305. Memorandum to Distribution from J. Davis, "NMSS Procedure for Review of Routine Inspection Operational Data and Licensee Event Reports," March 9, 1982. [8312290164]

306. IEEE Catalog No. TH0073-7, "Record of the Working Conference on Advanced Electrotechnology Applications to Nuclear Power Plants, January 15-17, 1980, Washington, D.C.," The Institute of Electrical and Electronics Engineers, Inc.

307. EPRI NP-2230, "ATWS: A Reappraisal, Part 3," Electric Power Research Institute, 1982.

308. SECY-82-352, "Assurance of Quality," August 20, 1982. [8209160068]

309. SECY-82-1, "Severe Accident Rulemaking and Related Matters," January 4, 1982. [8201190416]

310. Memorandum for W. Dircks from S. Chilk, "Staff Requirements - Briefing on Status and Plan for Severe Accident Rulemaking (SECY-82-1)," January 29, 1982. [8202160202]

311. SECY-82-1A, "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation," July 16, 1982. [8208040432]

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

313. NUREG/CR-2063, "Effects of the Accident of Three Mile Island on Property Values and Sales," U.S. Nuclear Regulatory Commission, March 1981.

314. NUREG/CR-2749, "Socioeconomic Impacts of Nuclear Generating Stations - Three Mile Island Case Study," U.S. Nuclear Regulatory Commission, (Vol. 12) July 1982.

315. Memorandum of Understanding Between the Federal Emergency Management Agency and the Nuclear Regulatory Commission, "Incident Response," October 22, 1980. [8011170793]

316. Memorandum of Understanding Between the Nuclear Regulatory Commission and the Federal Emergency Management Agency, "Radiological Emergency Planning and Preparedness," November 4, 1980. [8012110538]

317. Memorandum for G. Lainas from F. Miraglia, "CRGR Package for MPA B-71, 120 VAC Vital Instrument Buses and Inverter Technical Specifications," November 23, 1982. [8212160596]

318. Memorandum for H. Denton, et al., from C. Michelson, "An Analysis of the Abnormal Transient Operating Guidelines (ATOG) as Applied to the April 1981 Overfill Event at Arkansas Nuclear One - Unit 1," April 9, 1982. [8204220005]

319. Memorandum for C. Michelson from D. Eisenhut, "Review of Abnormal Transient Operating Guidelines (ATOG) as Applied to the April 8, 1981 Overfill Event at Arkansas Nuclear One - Unit 1," July 30, 1982. [8208180173]

320. Memorandum for C. Michelson from H. Denton, "Review of the Case Study of the Abnormal Transient Operating Guidelines (ATOG) as Applied to the April 1981 Overfill Event at Arkansas Nuclear One - Unit 1," October 7, 1982. [8211030575]

321. Memorandum for Commissioner Ahearne from W. Dircks, "AEOD Report on Arkansas Unit 1 Overfill Event," November 1, 1982. [8211190330]

322. AEOD/C201, "Report on The Safety Concern Associated with Reactor Vessel Level Instrumentation in Boiling Water Reactors," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, January 1982. [8202180432]

323. Memorandum for C. Michelson from H. Denton, "AEOD January 1982 Report on Safety Concern Associated with Reactor Vessel Level Instrumentation in Boiling Water Reactors," March 19, 1982. [8204190068]

324. NUREG-0785, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System," U.S. Nuclear Regulatory Commission, April 1981.

325. NRC Letter to All BWR Licensees, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System (Generic Letter 81-20)," April 10, 1981. [8112170367]

326. NEDO-24342, "GE Evaluation in Response to NRC Request Regarding BWR Scram System Pipe Breaks," General Electric Company, April 1981. [8105070251]

327. Letter to D. Eisenhut (NRC) from G. Sherwood (GE), "NRC Report, 'Safety Concerns Associated with Pipe Breaks in the BWR Scram System,'" April 30, 1981. [8105070249]

328. NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," U.S. Nuclear Regulatory Commission, August 1981.

329. NRC Letter to All GE BWR Licensees (Except Humboldt Bay), "Safety Concerns Associated with Pipe Breaks in the BWR Scram System (Generic Letter 81-34)," August 31, 1981. [8110150121]

330. AEOD/C003, "Report on Loss of Offsite Power Event at Arkansas Nuclear One, Units 1 and 2, on April 7, 1980," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, October 15, 1980. [8011170099]

331. Memorandum for C. Michelson from H. Denton, "NRR Responses to AEOD Recommendations on the Arkansas Loss of Offsite Power Event of April 7, 1980," February 13, 1981. [8102270127]

332. NRC Letter to All BWR Applicants for CPs, Holders of CPs, and Applicants for OLs, "Safety Concerns Associated with Pipe Breaks in the BWR Scram System (Generic Letter 81-35)," August 31, 1981. [8112170388]

333. SECY-82-445, "Proposal to Assign Two Resident Inspectors to Each Reactor Construction Site," November 1, 1982. [8211190003]

334. SECY-82-478, "Resident Inspection Program," December 6, 1982. [8401250359]

335. Memorandum for J. Taylor from D. Morrison, "Resolution of Generic Safety Issue 83, 'Control Room Habitability,'" June 17, 1996. [9607250277]

336. NUREG-0834, "NRC Licensee Assessments," U.S. Nuclear Regulatory Commission, August 1981.

337. NUREG/CR-2672, "SBLOCA Outside Containment at Browns Ferry Unit One - Accident Sequence Analysis," U.S. Nuclear Regulatory Commission, November 1982.

338. NUREG/CR-2744, "Human Reliability Data Bank for Nuclear Power Plant Operations," U.S. Nuclear Regulatory Commission, November 1982.

339. NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," U.S. Nuclear Regulatory Commission, October 1983.

340. Memorandum for H. Denton from J. Fouchard, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," January 17, 1983. [8302030055]

341. NUREG/CR-2255, "Expert Estimation of Human Error Probabilities in Nuclear Power Plant Operations: A Review of Probability Assessment and Scaling," U.S. Nuclear Regulatory Commission, May 1982.

342. NUREG/CR-2743, "Procedures for Using Expert Judgment to Estimate Human Error Probabilities in Nuclear Power Plant Operations," U.S. Nuclear Regulatory Commission, February 1983.

343. NUREG/CR-2254, "Workbook for Conducting Human Reliability Analysis," U.S. Nuclear Regulatory Commission, February 1983.

344. NUREG/CR-1205, "Data Summaries of Licensee Event Reports of Pumps at U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, January 1980, (Rev. 1) January 1982.

345. Memorandum for R. Vollmer from T. Murley, "Prioritization of New Requirements for PWR Feedwater Line Cracks," July 21, 1981. [8108180001]

346. NUREG/CR-1363, "Data Summaries of Licensee Event Reports of Valves at U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, June 1980, (Rev. 1) October 1982.

347. NUREG/CR-6316, "Guidelines for the Verification and Validation of Expert System Software and Conventional Software," U.S. Nuclear Regulatory Commission, (Vols. 1, 2, 3, 4, 5, 6, 7, and 8) March 1995.

348. NUREG/CR-1362, "Data Summaries of Licensee Event Reports of Diesel Generators at U.S. Commercial Nuclear Power Plants, January 1, 1976 through December 31, 1978," U.S. Nuclear Regulatory Commission, March 1980.

349. NUREG/CR-1331, "Data Summaries of Licensee Event Reports of Control Rods and Drive Mechanisms at U.S. Commercial Nuclear Power Plants, January 1, 1972 through April 30, 1978," U.S. Nuclear Regulatory Commission, February 1980.

350. NUREG/CR-1730, "Data Summaries of Licensee Event Reports of Primary Containment Penetrations at U.S. Commercial Nuclear Power Plants, January 1, 1972 through December 31, 1978," U.S. Nuclear Regulatory Commission, September 1980.

351. NUREG/CR-1740, "Data Summaries of Licensee Event Reports of Selected Instrumentation and Control Components at U.S. Commercial Nuclear Power Plants from January 1, 1976 to December 31, 1978," U.S. Nuclear Regulatory Commission, May 1981.

352. Memorandum for C. Michelson from E. Brown, "Internal Appurtenances in LWRs," December 24, 1980. [8101150319]

353. NUREG/CR-2641, "The In-Plant Reliability Data Base for Nuclear Power Plant Components: Data Collection and Methodology Report," U.S. Nuclear Regulatory Commission, July 1982.

354. NUREG/CR-2886, "The In-Plant Reliability Data Base for Nuclear Power Plant Components: Interim Data Report - The Pump Component," U.S. Nuclear Regulatory Commission, January 1983.

355. EGG-EA-5502, "User's Guide to BFR, A Computer Code Based on the Binomial Failure Rate Common Cause Model," EG&G, Inc., July 1982.

356. EGG-EA-5623, "Common Cause Fault Rates for Instrumentation and Control Assemblies: Estimates Based on Licensee Event Reports at U.S. Commercial Nuclear Power Plants, 1976-1978," EG&G, Inc., (Rev. 1) September 1982.

357. EGG-EA-5485, "Common Cause Fault Rates for Valves: Estimates Based on Licensee Event Reports at U.S. Commercial Nuclear Power Plants, 1976-1980," EG&G, Inc., (Rev. 1) September 1982.

358. NUREG/CR-2099, "Common Cause Fault Rates for Diesel Generators: Estimates Based on Licensee Event Reports at U.S. Commercial Nuclear Power Plants, 1976-1978," U.S. Nuclear Regulatory Commission, (Rev. 1) June 1982.

359. NUREG/CR-1401, "Estimators for the Binomial Failure Rate Common Cause Model," U.S. Nuclear Regulatory Commission, April 1980.

360. EGG-EA-5289, "Common Cause Fault Rates for Pumps: Estimates Based on Licensee Event Reports at U.S. Commercial Nuclear Power Plants, January 1, 1972 through September 30, 1980," EG&G, Inc., (Rev. 1) August 1982.

361. JBFA-101-82, "Common Cause Screening Methodology Project (FY 81 Technical Progress Report)," JBF Associates, Inc., February 1982.

362. NUREG/CR-2542, "Sensitivity Study Using the FRANTIC II Code for the Unavailability of a System to the Failure Characteristics of the Components and the Operating Conditions," U.S. Nuclear Regulatory Commission, February 1982.

363. NUREG/CR-2332, "Time Dependent Unavailability of a Continuously Monitored Component," U.S. Nuclear Regulatory Commission, August 1981.

364. Memorandum for W. Dircks from S. Chilk, "Systematic Assessment of Licensee Performance," October 20, 1981. [8210080207]

365. NUREG/CR-2515, "Crystal River 3 Safety Study," U.S. Nuclear Regulatory Commission, December 1981.

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.

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

368. Memorandum for ACRS Members from C. Michelson, "Failure of a Feedwater Flow Straightener at San Onofre Nuclear Station, Unit 1," June 13, 1979. [7910180473]

369. SECY-82-396A, "Withdrawal of SECY-82-396 (Federal Policy Statement on Use of Potassium Iodide)," October 15, 1982. [8211040047]

370. SECY-81-676, "Delegation of Rulemaking Authority to the EDO," December 3, 1981. [8201110403]

371. SECY-82-187, "Revised Guidelines for Value-Impact Analyses," May 7, 1982. [8205130275]

372. SECY-82-447, "Draft Report of the Regulatory Reform Task Force," November 3, 1982. [8211160547]

373. NUREG-0499, "Preliminary Statement on General Policy for Rulemaking to Improve Nuclear Power Plant Licensing," U.S. Nuclear Regulatory Commission, December 1978.

374. Memorandum for J. Hendrie from L. Bickwit, "Review of Commission Delegation of Authority," October 4, 1979. [8001150518]

375. Memorandum for R. Minogue from R. Bernero, "Charter of the Regulatory Analysis Branch," October 9, 1981. [8110280720]

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. [8212060349]

377. Memorandum for W. Minners from B. Snyder, "Schedule for Resolving and Completing Generic Issues," December 16, 1982. [8312290162]

378. Memorandum for S. Boyd from M. Srinivasan, "FY 1983-FY 1984 Office of Nuclear Reactor Regulation Operating Plan," November 17, 1982. [8301100332]

379. Memorandum for H. Denton from R. DeYoung, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," January 24, 1983. [8401160474]

380. SECY-93-049, "Implementation of 10 CFR Part 54, 'Requirements for Renewal of Operating Licenses for Nuclear Power Plants,'" March 1, 1993. [9303030337, 9303100375].

381. Memorandum for W. Minners from O. Parr, "Prioritization of Proposed Generic Issue on CRD Accumulator Check Valve Leakage," August 13, 1984. [8408280264]

382. Memorandum for W. Minners from R. Mattson, "Schedules for Resolving and Completing Generic Issues," January 21, 1983. [8301260532]

383. Memorandum for W. Dircks from R. Mattson, "Closeout of TMI Action Plan I.C.1(4), Confirmatory Analyses of Selected Transients," November 12, 1982. [8212080586]

384. Memorandum for T. Speis from R. Vollmer, "Schedules for Resolving and Completing Generic Issues," February 1, 1983. [8401170076]

385. Memorandum for T. Murley from D. Ross, "Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis," March 10, 1981. [8103240798, 9804090138]

386. Memorandum for T. Novak from R. Frahm, "Summary of Meeting with General Electric on the Use of Non-Safety Grade Equipment," March 7, 1979. [7903220463]

387. NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," U.S. Nuclear Regulatory Commission, January 1978.

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

389. "Indian Point Probabilistic Safety Study," Power Authority of the State of New York and Consolidated Edison Company of New York, Inc., 1982.

390. NUREG-0850, "Preliminary Assessment of Core Melt Accidents at the Zion and Indian Point Nuclear Power Plants and Strategies for Mitigating Their Effects," U.S. Nuclear Regulatory Commission, November 1981.

391. Memorandum for E. Reeves from J. Knight, "Zion Liquid Pathway Analysis," August 8, 1980. [8008210647]

392. Memorandum for J. Funches from R. Mattson, "Request for Approval to Work on Low Priority Generic Safety Issues," November 5, 1982. [8211160524]

393. "TMI-2 Recovery Program Estimate," General Public Utilities Corp., (Rev. 1) July 1981.

394. Memorandum for S. Hanauer, et al., from D. Eisenhut, "Operating Reactor Event Memorandum No. 81-31: Loss of Direct Current (DC) Bus at Millstone Unit 2," March 31, 1981. [8104100493]

395. Memorandum for H. Denton from C. Michelson, "Millstone Unit 2 - Reactor Trip Following De-Energization of a 125 V DC Bus," November 5, 1981. [8112010276]

396. Memorandum for C. Michelson from H. Denton, "AEOD November 1981 Report on the Millstone Unit 2 Loss of 125 V DC Bus Event," January 4, 1982. [8202040017]

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

398. Memorandum for R. Tedesco from T. Speis, "Identification of Protection System Instrument Sensing Lines," April 29, 1982. [8205270511]

399. Memorandum to C. Rossi from A. Thadani, "Prioritization of and Transfer of Responsibility for Generic Safety Issue 156.6.1, 'Pipe Break Effects on Systems and Components Inside Containment,'" July 16, 1999. [9908300234]

400. Memorandum for V. Stello from H. Denton, "Standard Review Plan Guidance for Identification of Protection System Instrument Lines," December 29, 1982. [8301070067]

401. Memorandum for H. Denton from V. Stello, "Proposed Standard Review Plan Guidance for Identification of Protection System Instrument Lines," January 27, 1983. [8302180526]

402. Letter to D. Eisenhut (NRC) from T. Dente (BWR Owners' Group), "Analysis of Scram Discharge Volume System Piping Integrity, NEDO-22209 (Prepublication Form)," August 23, 1982. [8208310340]

403. Letter to K. Eccleston (NRC) from T. Dente (BWR Owners' Group), "Transmittal of Supporting Information on Application of Scram Time Fraction to Scram Discharge Volume (SDV) Pipe Break Probability as Used in NEDO-22209," January 28, 1983. [8302010525]

404. Letter to S. Israel (NRC) from J. Hickman (SNL), "Review and Evaluation of the Indian Point Probabilistic Safety Study," August 25, 1982. [8209230166]

405. Memorandum for W. Minners from A. Thadani, et al., "Probability of Core Melt Due to Component Cooling Water System Failures," January 19, 1983. [8301270522]

406. Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan - Status Report," March 4, 1982. [8204290601]

407. Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan - Completed Item," May 11, 1982. [8401170108]

408. NUREG-1509, "Radiation Effects on Reactor Pressure Vessel Supports," U.S. Nuclear Regulatory Commission, May 1996.

409. Memorandum for W. Minners from W. Mills, "Prioritization of Generic Issue III.D.3.5, Radiation Worker Data Base," February 22, 1983. [9705190229]

410. Memorandum for T. Speis from R. Browning, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," April 1, 1983. [8304200629, 9705190233]

411. SLI-8211, "Review of BWR Reactor Vessel Water Level Measurement Systems," S. Levy, Inc., July 1982.

412. Memorandum for T. Speis from J. Funches, "Prioritization of Generic Issues - Environmental and Licensing Improvements," February 24, 1983. [8303090540]

413. Memorandum for D. Eisenhut from E. Jordan, "Main Steam Isolation Valve (MSIV) Survey," July 1, 1982. [8209240107]

414. Memorandum for W. Minners from L. Hulman, "Consequence Analyses for BWR Main Steam System Leakage Pathway Generic Issue Evaluation," December 9, 1982. [8301050058]

415. Memorandum for W. Minners from L. Hulman, "MSIV Leakage Consequences," December 23, 1982. [8312290172]

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.

420. IE Bulletin No. 80-14, "Degradation of BWR Scram Discharge Volume Capability," U.S. Nuclear Regulatory Commission, June 12, 1980. [8005050056]

421. IE Bulletin No. 80-17, "Failure of 76 of 185 Control Rods to Fully Insert During Scram at BWR," U.S. Nuclear Regulatory Commission, July 3, 1980. [8005050076]

422. NRC Letter to All BWR Licensees, "BWR Scram Discharge System," December 9, 1980. [8102190299]

423. Memorandum for R. Mattson from D. Eisenhut, "Status of Long-Term Followup of the Indian Point Unit 2 Flooding Event," May 13, 1982. [8205240153]

424. Memorandum for F. Schroeder from T. Speis, "Designation of Inadvertent Containment Flooding as a Generic Issue," August 5, 1982. [8208120379]

425. "Zion Probabilistic Safety Study," Commonwealth Edison Company, 1981.

426. Memorandum for T. Novak from G. Lainas and V. Noonan, "NRR Input to SER on 'Indian Point Unit No. 2 Flood in Containment Due to Containment Cooler Service Water Leaks on 10/17/80,'" April 3, 1981. [8104090906]

427. Memorandum for T. Speis from R. Mattson, "Close-out of TAP-A-16, Steam Effects on BWR Core Spray Distribution (TACS-40066)," March 29, 1983. [8304130488]

428. Memorandum for W. Minners from P. Hayes, "Generic Safety Issue No. 51, Improved Reliability of Open Service Water Systems," April 5, 1983. [9705190249]

429. Memorandum for J. Knight from E. Sullivan, "Review ACRS Consultant Report," January 10, 1980. [8105150033]

430. Memorandum for K. Seyfrit from E. Imbro, "Flow Blockage in Essential Raw Cooling Water System Due to Asiatic Clam Intrusion," March 28, 1983. [8305230539]

431. EPRI NP-1138, "Limiting Factor Analysis of High-Availability Nuclear Plants," Electric Power Research Institute, September 1979.

432. SECY-82-296, "Resolution of AEOD Combination LOCA Concern," July 13, 1982. [8207230202]

433. Memorandum for C. Michelson from E. Brown, "Degradation of Internal Appurtenances and/or Loose Parts in LWRs," June 15, 1982. [8207280317]

434. Memorandum for H. Denton, et al., from C. Michelson, "Flow Blockage in Essential Equipment at ANO Caused by Corbicula sp. (Asiatic Clams)," October 21, 1980. [8011060029]

435. Letter to N. Palladino from P. Shewmon, "Control Room Habitability," August 18, 1982. [8207180073]

436. Letter to J. Ray from W. Dircks, "August 18, 1982, ACRS Letter on Control Room Habitability," January 31, 1983. [8302100196]

437. Memorandum for H. Denton from R. Minogue, "Draft Report on the Prioritization of Non-NRR TMI Action Plan Items," March 29, 1983. [8401160475]

438. Memorandum for G. Cunningham, et al., from W. Dircks, "NRC Actions Required by Enactment of the Nuclear Waste Policy Act of 1982," January 19, 1983. [8507110762]

439. Regulatory Guide 1.149, "Nuclear Power Plant Simulators for Use in Operator Training," U.S. Nuclear Regulatory Commission, April 1981 [8105220400], (Rev. 1) April 1987 [8704300503, 8601160291], (Rev. 2) April 1996 [9604170117].

440. Memorandum for W. Minners from D. Ziemann, "Schedules for Resolving and Completing Generic Issues," April 5, 1983. [8304180758]

441. Memorandum for H. Denton from R. DeYoung, "Commission Paper on the Prioritization of Generic Safety Issues," April 20, 1983. [9705190224]

442. Memorandum for R. Emrit from T. Rothschild, "Establishing Priorities for Generic Safety Issues," April 21, 1983. [8312290167]

443. Memorandum for W. Dircks from R. Mattson, "Closeout of NUREG-0660 Item II.E.5.1, Design Sensitivity of B&W Plants for Operating Plants," March 15, 1983. [8304080415]

444. "Letter to Public Service Electric and Gas Company from D. Fischer (NRC) 'Meeting Summary - Salem Unit - 1 Failure of Reactor Trip Breakers,'" March 14, 1983. [8303210160]

445. NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1983, (Vol. 2) August 1983.

446. Memorandum for Chairman Ahearne from C. Michelson, "New Unresolved Safety Issues," August 4, 1980. [8010240206]

447. NUREG-0977, "NRC Fact Finding Task Force Report on the ATWS Events at Salem Nuclear Generating Station Unit 1 on February 22 and 25, 1983," U.S. Nuclear Regulatory Commission, March 1983.

448. Memorandum for F. Rowsome from S. Bryan, "Reliability Assurance - Reactor Protection System," July 23, 1981. [8109100509]

449. Memorandum for S. Hanauer from D. Eisenhut, "Potential Generic Issue: BWR Control Rod Test Requirements Following Maintenance," November 26, 1982. [8212160790]

450. Memorandum for R. Mattson from T. Speis, "Potential Generic Issues Related to Scram Systems," April 7, 1983. [8304200351]

451. Memorandum for H. Denton from C. Heltemes, "Potential Design Deficiency in Westinghouse Reactor Protection System," March 10, 1983. [8303230335]

452. Memorandum for C. Heltemes from H. Denton, "Westinghouse Reactor Protection System Design Conformance to IEEE Standard 279," May 2, 1983. [8305180398]

453. Memorandum for H. Denton, et al., from R. Mattson, "Recommended Generic Actions," April 27, 1983. [8305250017]

454. SECY-83-98E, "Salem Restart Evaluation," April 11, 1983. [8304220308]

455. NUREG-0771, "Regulatory Impact of Nuclear Reactor Accident Source Term Assumptions," U.S. Nuclear Regulatory Commission, June 1981.

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

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

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

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

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

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

462. NUREG/CR-0022,""Need for Power: Determination in the State Decisionmaking Process," U.S. Nuclear Regulatory Commission, March 1978.

463. NUREG/CR-0250, "Regional Econometric Model for Forecasting Electricity Demand by Sector and State," U.S. Nuclear Regulatory Commission, October 1978.

464. NUREG-0555, "Environmental Standard Review Plans for the Environmental Review of Construction Permit Applications for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, May 1979.

465. NUREG-0398, "Federal-State Cooperation in Nuclear Power Plant Licensing," U.S. Nuclear Regulatory Commission, March 1980.

466. NUREG-0942, "Conducting Need-for-Power Review for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, December 1982.

467. NUREG/CR-2423, "Mathematical Simulation of Sediment and Radionuclide Transport in Estuaries," U.S. Nuclear Regulatory Commission, November 1982.

468. NUREG/CR-2823, "A Review of the Impact of Copper Released into Marine and Estuarine Environments," U.S. Nuclear Regulatory Commission, November 1982.

469. NUREG/CR-0892, "Chronic Effects of Chlorination Byproducts on Rainbow Trout, Salmo gairdneri," U.S. Nuclear Regulatory Commission, November 1980.

470. NUREG/CR-0893, "Acute Toxicity and Bioaccummulation of Chloroform to Four Species of Freshwater Fish," U.S. Nuclear Regulatory Commission, August 1980.

471. NUREG/CR-2750, "Socioeconomic Impacts of Nuclear Generating Stations," U.S. Nuclear Regulatory Commission, July 1982.

472. NUREG/CR-2861, "Image Analysis for Facility Siting: A Comparison of Low and High-Attitude Image Interpretability for Land Use/Land Cover Mapping," U.S. Nuclear Regulatory Commission, November 1982.

473. NUREG/CR-2550, "Charcoal Performance Under Simulated Accident Conditions," U.S. Nuclear Regulatory Commission, July 1982.

474. NUREG-0700, "Guidelines for Control Room Design Reviews," U.S. Nuclear Regulatory Commission, September 1981.

475. Memorandum for W. Minners from F. Congel, "Prioritization of Generic Issue 58, Containment Flooding," May 19, 1983. [8306080295]

476. Regulatory Guide 4.2, "Preparation of Environmental Reports for Nuclear Power Stations," U.S. Nuclear Regulatory Commission, July 1976. [7908310195]

477. NUREG/CR-2692, "An Integrated System for Forecasting Electric Energy and Load for States and Utility Service Areas," U.S. Nuclear Regulatory Commission, May 1982.

478. Regulatory Guide 1.57, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components," U.S. Nuclear Regulatory Commission, June 1973. [7907100220]

479. Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission, November 1978. [7812270049]

480. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," U.S. Nuclear Regulatory Commission, October 1977. [7907100401]

481. Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments," U.S. Nuclear Regulatory Commission, February 1973, (Rev. 1) June 1974, (Rev. 2) January 1976 [7907100149], (Rev. 3) July 1990 [7809180004].

482. Regulatory Guide 1.90, "Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons," U.S. Nuclear Regulatory Commission, August 1977. [7907100329]

483. ORNL-5470, "CONCEPT-5 User's Manual," Oak Ridge National Laboratory, December 1978.

484. ORNL/TM-6467, "A Procedure for Estimating Nonfuel Operation and Maintenance Costs for Large Steam-Electric Power Plants," Oak Ridge National Laboratory, January 1979.

485. NUREG/CR-2844, "Nonfuel Operation and Maintenance Costs for Large Steam-Electric Power Plants - 1982," U.S. Nuclear Regulatory Commission, September 1982.

486. Memorandum for Z. Rosztoczy, et al., from W. Anderson, "Seismic Scram," January 20, 1983. [8302100005]

487. Memorandum for G. Arndt from G. Burdick, "Review of Seismic Scram Report, UCRL-53037," March 3, 1983. [8303160092]

488. NUREG-0610, "Draft Emergency Action Level Guidelines for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, September 1979.

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

490. Memorandum for H. Denton from C. Michelson, "Potential Generator Missiles - Generator Rotor Retaining Rings," March 16, 1982. [8203300270]

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. [8212140103]

492. Memorandum for C. Michelson from H. Denton, "H. B. Robinson RCS Leak on January 29, 1981," June 15, 1981. [8107010140]

493. Memorandum for C. Michelson from H. Denton, "January 19, 1981, Memorandum on Degradation of Internal Appurtenances in LWR," April 30, 1981. [8105150032]

494. Memorandum for C. Michelson from H. Denton, "AEOD Preliminary Report on Calvert Cliffs Unit 1 Loss of Service Water," August 5, 1981. [8108170221]

495. Memorandum for C. Michelson from H. Denton, "Steam Generator Overfill and Combined Primary and Secondary Blowdown," May 27, 1981. [8106100241]

496. Memorandum for H. Denton from C. Michelson, "Concerns Relating to the Integrity of a Polymer Coating for Surfaces Inside Containment (IE Draft Bulletin No. 80-21)," August 29, 1980. [8210120370, 8009110599]

497. Memorandum for H. Denton and V. Stello from C. Michelson, "Immediate Action Memo: Common Cause Failure Potential at Rancho Seco - Desiccant Contamination of Air Lines," September 15, 1981. [8109280036]

498. Memorandum for C. Michelson from H. Denton, "AEOD Immediate Action Memo on Contamination of Instrument Air Lines at Rancho Seco," October 26, 1981. [8111300391]

499. Memorandum for H. Denton, et al., from C. Michelson, "Case Study Report on San Onofre Unit 1 Loss of Salt Water Cooling Event on March 10, 1980," August 12, 1980. [8208270684]

500. Memorandum for C. Michelson from H. Denton, "NRR Comments on AEOD Final Report: Case Study Report on San Onofre Unit 1 Loss of Salt Water Cooling Event of March 10, 1980," October 8, 1982. [8211030304]

501. IE Bulletin No. 79-24, "Frozen Lines," U.S. Nuclear Regulatory Commission, September 27, 1979. [7908220114]

502. Memorandum for H. Denton and V. Stello from C. Michelson, "Inoperability of Instrumentation Due to Extreme Cold Weather," June 15, 1981. [8107010161]

503. Memorandum for C. Michelson from H. Denton, "AEOD Memorandum on the Inoperability of Instrumentation Due to Extreme Cold Weather," August 14, 1981. [8109110138]

504. Draft Regulatory Guide and Value/Impact Statement, Task IC 126-5, "Instrument Sensing Lines," U.S. Nuclear Regulatory Commission, March 1982. [8204190028]

505. Regulatory Guide 1.151, "Instrument Sensing Lines," U.S. Nuclear Regulatory Commission, July 1983. [8808230051]

506. Federal Register Notice 48 FR 36029, "Regulatory Guide; Issuance, Availability," August 8, 1983.

507. Memorandum for C. Michelson from H. Denton, "Interlocks and LCO's for Redundant Class 1E Tie Breakers (Point Beach Nuclear Plant Units 1 and 2)," October 16, 1980. [8011050312]

508. Memorandum for F. Schroeder from L. Rubenstein, "Review of General Electric Topical Report NEDO-10174, Revision 1," August 18, 1982. [8209030003]

509. Memorandum for C. Michelson from H. Denton, "NRR Comments on AEOD Final Report: Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980," August 19, 1982. [8404110426]

510. Memorandum for C. Michelson from H. Denton, "Effects of Fire Protection System Actuation on Safety-Related Equipment," August 27, 1982. [8506050357]

511. "Value-Impact Analysis of Recommendations Concerning Steam Generator Tube Degradations and Rupture Events," Science Applications, Inc., February 2, 1983.

512. Memorandum for D. Eisenhut from T. Speis, "DST Prioritization of Steam Generator Requirements," May 4, 1983. [8305230682]

513. SECY-82-186A, "Make-up Nozzle Cracking in Babcock and Wilcox (B&W) Plants," July 23, 1982. [8209300376]

514. Letter to J. Stolz (NRC) from G. Westafer (Florida Power Corporation), "Crystal River Unit 3, Docket No. 50-302, Operating License No. DPR-72, Safe End Task Force Action Plan," February 28, 1983. [8303040544]

515. Memorandum for W. Minners from D. Dilanni, "Proposed Generic Issue 'PORV and Block Valve Reliability,'" June 6, 1983. [8307050513]

516. Memorandum for W. Johnston and L. Rubenstein from T. Speis, "Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety," August 6, 1982. [8208230475]

517. Memorandum for the Atomic Safety & Licensing Boards for: Callaway Plant, Unit 1; Comanche Peak Steam Electric Station, Units 1 & 2; and the Atomic Safety & Licensing Appeal Board for Virgil C. Summer Nuclear Station, Unit 1, from T. Novak, "Board

Notification - Control Rod Drive Guide Tube Support Pin Failures at Westinghouse Plants (Board Notification No. 82-81)," August 16, 1982. [8209290318]

518. Memorandum for D. Eisenhut from J. Crews, "NRC Lead Responsibility for Possibly Detached Thermal Sleeves - Trojan Nuclear Plant - Docket No. 50-344," June 18, 1982. [8710220041]

519. Memorandum for W. Minners from L. Hulman, "Generic Issue on Iodine Coolant Activity Limiting Conditions for Operation," June 10, 1983. [8307080562]

520. NRC Letter to All Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, "Required Actions Based on Generic Implications of Salem ATWS Events," (Generic Letter No. 83-28), July 8, 1983 [8307080169], (Supplement 1) October 7, 1992 [9210050243].

521. SECY-83-248, "Generic Actions for Licensees and Staff in Response to the ATWS Events at Salem Unit 1," June 22, 1983. [8307110103]

522. AEOD/P301, "Report on the Implications of the ATWS Events at the Salem Nuclear Power Plant on the NRC Program for Collection and Analysis of Operational Experience," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, July 1983. [8307140507]

523. Memorandum for C. Heltemes from H. Denton, "AEOD Final Report on the Implications of the ATWS Events at the Salem Nuclear Power Plant on the NRC Program for Collection and Analysis of Operational Experience," July 21, 1983. [8307290045]

524. Memorandum for R. Mattson from T. Speis, "Draft CRGR Package on A-30, DC Power," May 24, 1983. [8306030487]

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. [8303290078]

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 [8305110502], (Supplement 1) August 30, 1990 [9008240057].

528. Memorandum for B. Liaw from H. Berkow, "OMB Clearance Renewal - Monitoring of Fatigue Transient Limits for Reactor Coolant System," May 13, 1983. [9705190223]

529. Memorandum for H. Berkow from W. Minners, "OMB Clearance Renewal -Monitoring of Fatigue Transient Limits for Reactor Coolant System," June 1, 1983. [8306090456]

530. Letter to R. DeYoung (NRC) from J. Taylor (B&W), "Unanalyzed Reactor Vessel Thermal Stress During Cooldown," March 18, 1983. [8303250020]

531. Memorandum for R. Vollmer from W. Minners, "B&W Notification Concerning an Unanalyzed Reactor Vessel Thermal Stress During Cooldown," April 7, 1983. [8304140390]

532. Regulatory Guide 1.93, "Availability of Electric Power Sources," U.S. Nuclear Regulatory Commission, December 1974. [7907100337]

533. Memorandum for W. Minners from R. Bosnak, "B&W Notification Concerning an Unanalyzed Reactor Vessel Thermal Stress During Cooldown," April 26, 1983. [8305240235]

534. Memorandum for N. Palladino, et al., from D. Eisenhut, "Unanalyzed Reactor Vessel Thermal Stress During Cooldown (Board Notification #BN-83-42)," April 12, 1983. [8304220651]

535. CE-NPSD-154, "Natural Circulation Cooldown, Task 430 Final Report," Combustion Engineering, Inc., October 1981. [8304280091]

536. B&W Document No. 86-1140819-00, "Reactor Vessel Head Cooldown During Natural Circulation Cooldown Transients," Babcock & Wilcox Company, February 8, 1983. [8302160171]

537. Memorandum for W. Dircks from R. Fraley, August 18, 1982. [8207180092]

538. Memorandum for R. Fraley from H. Denton, "ACRS Inquiry on Pipe Break Effects on CRD Hydraulic Lines," October 29, 1982. [8211120045]

539. Letter to W. Dircks from J. Ebersole, "ACRS Comments Regarding Potential Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments," March 16, 1983. [8303290428]

540. BNL-NUREG-28109, "Thermal-Hydraulic Effects on Center Rod Drop Accidents in a Boiling Water Reactor," Brookhaven National Laboratory, July 1980. [8101220642]

541. Memorandum for B. Sheron from C. Berlinger, "ACRS Request for Information Related to LOCA Effects on CRD Hydraulic Lines," October 19, 1982. [8211040446]

542. Memorandum for R. Mattson, et al., from D. Eisenhut, "Potential Safety Problems Associated With Locked Doors and Barriers in Nuclear Power Plants," May 31, 1983. [8306200435]

543. Memorandum for T. Speis from R. Mattson, "Proposed Generic Issue on Beyond Design Basis Accidents in Spent Fuel Pools," August 10, 1983. [8308180730]

544. NUREG/CR-0649, "Spent Fuel Heatup Following Loss of Water During Storage," U.S. Nuclear Regulatory Commission, May 1979.

545. Memorandum for Z. Rosztoczy from P. Williams, "Trip Report: International Meeting on Severe Fuel Damage and Visit to Power Burst Facility," April 25, 1983. [8305060661]

546. Memorandum to C. Rossi from D. Cool, "NMSS Input for Second Quarter FY-2000 Update of the Generic Issues Management Control System," April 18, 2000.

547. SECY-95-245, "Completion of the Fatigue Action Plan," September 25, 1995. [9509290040]

548. Memorandum for W. Dircks from R. DeYoung, "TMI Action Plan Completed Items," January 26, 1983. [8303090323]

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, June 1982.

551. Letter to N. Palladino from J. Ray, "Need for Rapid Depressurization Capability in Newer Combustion Engineering, Inc. Plants," October 18, 1983. [8311010118]

552. Memorandum for W. Minners from B. Siegel, "Proposed Generic Issue 'Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments,'" November 3, 1983. [8312140360]

553. Memorandum for D. Eisenhut from J. Olshinski, "Loss of High Head Injection Capability at McGuire Unit 1 and Reconsideration of Technical Specification 3.0.3 and 3.5.2," April 12, 1982. [8802120046]

554. Memorandum for D. Eisenhut, et al., from H. Denton, "Development of Generic Recommendations Based on the Review of the January 25, 1982 Steam Generator Tube Rupture at Ginna," May 3, 1982. [8205280089]

555. Letter to D. Eisenhut (NRC) from D. Waters (BWR Owners' Group), "BWR Owners' Group Evaluations of NUREG-0737 Requirements II.K.3.16 and II.K.3.18," March 31, 1981. [8104200300]

556. Memorandum for G. Lainas, et al., from W. Houston, "Evaluation of BWR Owners' Group Generic Response to Item II.K.3.16 of NUREG-0737, 'Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification,'" April 1, 1983. [8711060070]

557. Memorandum for H. Denton and V. Stello from C. Michelson, "Calvert Cliffs Unit 1 Loss of Service Water," June 19, 1981. [8107060505]

558. Memorandum for H. Denton and R. DeYoung from C. Michelson, "Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980," December 17, 1981. [8201150431]

559. Memorandum for C. Michelson from H. Denton, "NRR Comments on AEOD Final Report: Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980," September 23, 1982. [8210180230]

560. Memorandum for H. Denton from C. Heltemes, "Response to NRR Comments on AEOD Report, 'Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980,'" May 2, 1983. [8305110577]

561. Memorandum for W. Houston and L. Rubenstein from F. Miraglia, "Response to NRR Comments on AEOD Report, 'Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980,'" June 2, 1983. [8306080036]

562. Memorandum for F. Miraglia from W. Houston and L. Rubenstein, "Comments to AEOD Memo dated May 2, 1983, on Calvert Cliffs, Unit 1, Loss of Service Water on May 20, 1980," July 22, 1983. [8308030493]

563. Memorandum for C. Heltemes from H. Denton, "Response to NRR Comments on AEOD Report, 'Calvert Cliffs Unit 1 Loss of Service Water on May 20, 1980,'" September 15, 1983. [8309270470]

564. Memorandum for R. Baer from K. Seyfrit, "Case Study, 'Calvert Cliffs Unit 1 Loss of Service Water on May 29, 1980,'" August 18, 1983. [8308290487]

565. IE Information Notice No. 83-77, "Air/Gas Entrainment Events Resulting in System Failures," U.S. Nuclear Regulatory Commission, November 14, 1983. [8311010015]

566. Memorandum for G. Holahan from W. Minners, "Prioritization of Issue 36: Loss of Service Water at Calvert Cliffs Unit 1," November 10, 1983. [8311180373]

567. Letter to A. Lundvall (Baltimore Gas and Electric Company) from D. Eisenhut (NRC), Docket No. 50-317, September 15, 1983. [8309270504]

568. Memorandum for W. Houston and L. Rubenstein from F. Schroeder, "Request for Reactor Systems Branch and Auxiliary Systems Branch Support for Plant Visits on USI A-45," November 28, 1983. [8312150068]

569. AEOD/C102, "Engineering Evaluation of the H. B. Robinson Reactor Coolant System Leak on January 29, 1981," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, March 23, 1981. [8104150060]

570. Memorandum for V. Stello from H. Denton, "Issuance of Revised Section 7.1, Appendix A to this Section, Section 7.5 and Section 7.7 of the Standard Review Plan, NUREG-0800," March 9, 1984. [8404160228]

571. Memorandum for H. Denton from V. Stello, "SRP Changes Concerning Resolution of Generic Issue 45, Inoperability of Instrumentation due to Extreme Cold Weather," April 3, 1984. [8404180510]

572. Memorandum for G. Lainas from F. Rowsome, "Safety Evaluation of the Westinghouse Licensees' Responses to TMI Action Item II.K.3.2," July 22, 1983. [8308040054]

573. Memorandum for G. Lainas from F. Rowsome, "Safety Evaluation of the B&W Licensees' Responses to TMI Action Item II.K.3.2," August 24, 1983. [8308310422]

574. Memorandum for G. Lainas from F. Rowsome, "Safety Evaluation of the CE Licensees' Responses to TMI Action Item II.K.3.2," August 26, 1983. [8309060394]

575. Memorandum for W. Minners from B. Sheron, "Proposed Generic Issue on PORV and Block Valve Reliability," June 27, 1983. [8307180224]

576. Memorandum for R. Riggs from F. Cherny, "Comments on Draft Write-up of Prioritization of Generic Issue 70 'PORV and Block Valve Reliability,'" December 21, 1983. [8401030003]

577. Memorandum for H. Denton, et al., from C. Heltemes, "Case Study Report - Low Temperature Overpressure Events at Turkey Point Unit 4," September 26, 1983. [8310060171]

578. NUREG-0748, "Operating Reactors Licensing Actions Summary," U.S. Nuclear Regulatory Commission, (Vol. 5, No. 11) February 1986.

579. NUREG-0694, "TMI-Related Requirements for New Operating Licenses," U.S. Nuclear Regulatory Commission, June 1980.

580. NUREG-0645, "Report of the Bulletins & Orders Task Force of the Office of Nuclear Reactor Regulation," U.S. Nuclear Regulatory Commission, (Vol. 1) January 1980, (Vol. 2) January 1980.

581. NUREG-0909, "NRC Report on the January 25, 1982 Steam Generator Tube Rupture at R.E. Ginna Nuclear Power Plant," U.S Nuclear Regulatory Commission, April 1982.

582. NUREG-0713, "Occupational Radiation Exposure at Commerical Nuclear Power Reactors - 1981," U.S Nuclear Regulatory Commission, (Vol. 1) March 1981, (Vol. 2) December 1981, (Vol. 3) November 1982, (Vol. 4) November 1983, (Vol. 5) March 1985, (Vol. 6) September 1986, (Vol. 7) April 1988.

583. EPRI NP-2292, "PWR Safety and Relief Valve Test Program," Electric Power Research Institute, December 1982.

584. EPRI NP-1139, "Limiting Factor Analysis of High Availability Nuclear Plants," Electric Power Research Institute, August 1979.

585. EPRI P-2410-SR, "Technical Assessment Guide," Electric Power Research Institute, May 1982.

586. WCAP-9804, "Probabilistic Analysis and Operational Data in Response to NUREG-0737 Item II.K.3.2 for Westinghouse NSSS Plants," Westinghouse Electric Corporation, February 1981. [8103160257]

587. "Accident Sequence Evaluation Program, Phase II Workshop Report," Sandia National Laboratories, EG&G Idaho, Inc., and Science Applications, Inc., September 1982.

588. Letter to Director (NRR) from K. Cook (Louisiana Power & Light), "Waterford SES Unit 3, Docket No. 50-382, Depressurization and Decay Heat Removal," October 27, 1983. [8311010259]

589. Letter to W. Dircks (NRC) from E. Van Brunt (Arizona Public Service Company), "Palo Verde Nuclear Generating Station (PVNGS), Units 1, 2, and 3, Docket Nos. STN-50-528/529/530," November 7, 1983. [8312230233]

590. ALO-75 (TR-3459-1), "Pilot Program to Identify Valve Failures Which Impact the Safety and Operation of Light Water Nuclear Power Plants," Teledyne Engineering Services, January 11, 1980.

591. IE Information Notice No. 82-45, "PWR Low Temperature Overpressure Protection," U.S. Nuclear Regulatory Commission, November 19, 1982. [8208190253]

592. IE Information Notice No. 82-17, "Overpressurization of Reactor Coolant System," U.S. Nuclear Regulatory Commission, June 10, 1982. [8204210383]

593. SECY-84-76, "Proposed Rulemaking for Operator Licensing and for Training and Qualifications of Civilian Nuclear Power Plant Personnel," February 13, 1984. [8403260357]

594. Letter to E. Wilkinson (INPO) from W. Dircks (NRC), November 23, 1983. [8312090099]

595. SECY-83-52A, "Final Rulemaking Concerning Licensed Operator Staffing at Nuclear Power Units and Draft Policy Statement on Shift Crew Qualifications," March 14, 1983. [8304010029]

596. Memorandum for W. Dircks from S. Chilk, "Staff Requirements - Affirmation/ Discussion and Vote, 3:35 p.m., Thursday, April 21, 1983, Commissioners' Conference Room (Open to Public Attendance)," April 28, 1983. [9705190263]

597. Federal Register Notice 48 FR 33850, "Licensee Event Report System," July 26, 1983.

598. Memorandum for W. Dircks from R. Mattson, "Closeout of TMI Action Plan Task III.D.2.5, 'Offsite Dose Calculation Manual,'" January 17, 1984. [8402020114]

599. NUREG/CR-3332, "Radiological Assessment - A Textbook on Environmental Dose Analysis," U.S. Nuclear Regulatory Commission, September 1983.

600. NUREG-0978, "Mark III LOCA-Related Hydrodynamic Load Definition," U.S. Nuclear Regulatory Commission, August 1984.

601. Memorandum for T. Combs from H. Denton, "Revised SRP Section 6.2.1.1.C of NUREG-0800," September 10, 1984. [8409180459]

602. Memorandum for T. Speis from R. Mattson, "Status of Generic Issues 40 and 65 Assigned to DSI," December 27, 1983. [8401170445]

603. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," U.S. Nuclear Regulatory Commission, May 1973. [7907100185]

604. SECY-81-641, "Electromagnetic Pulse (EMP) - Effects on Nuclear Power Plants," November 5, 1981. [8202090418, 8111250553]

605. SECY-82-157, "Status Report on the Evaluation of the Effects of Electromagnetic Pulse (EMP) on Nuclear Power Plants," April 13, 1982. [8205050108]

606. SECY-83-367, "Staff Study of Electromagnetic Pulse (EMP) Effects on Nuclear Power Plants and Discussion of Related Petitions for Rulemaking (PRM-50-32, 32A, and 32B)," September 6, 1983. [8312210152]

607. Memorandum for W. Dircks from S. Chilk, "SECY-83-367 - Staff Study of Electromagnetic Pulse (EMP) Effects on Nuclear Power Plants and Discussion of Related Petition for Rulemaking (PRM-50-32, 32A, and 32B)," November 15, 1983. [8402270019]

608. IE Information Notice No. 82-39, "Service Degradation of Thick-Walled Stainless Steel Recirculation Systems at BWR Plants," U.S. Nuclear Regulatory Commission, September 21, 1982. [8208190229]

609. IE Bulletin No. 82-03, "Stress Corrosion Cracking in Thick-Wall Large Diameter, Stainless Steel Recirculation System Piping at BWR Plants," U.S. Nuclear Regulatory Commission, October 14, 1982 [8208190238], (Rev. 1) October 28, 1982 [8208190240].

610. IE Bulletin No. 83-02, "Stress Corrosion Cracking in Large Diameter Stainless Steel Recirculation System Piping at BWR Plants," U.S. Nuclear Regulatory Commission, March 4, 1983. [8212060368]

611. NUREG-1061, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," U.S. Nuclear Regulatory Commission, (Vol. 1) August 1984, (Vol. 2) April 1985, (Vol. 3) November 1984, (Vol. 4) December 1984, (Vol. 5) April 1985.

612. SECY-83-267, "Status Report on Observation of Pipe Cracking at BWRs," July 1, 1983. [8307250565]

613. SECY-83-267A, "Update of Status Report on Observation of Pipe Cracking at BWRs (SECY-83-267)," July 11, 1983. [8307250578]

614. SECY-83-267B, "Update of Status Report on Observation of Pipe Cracking at BWRs (SECY-83-267 and 267A)," August 8, 1983. [8308230648]

615. SECY-83-267C, "Staff Requirements for Reinspection of BWR Piping and Repair of Cracked Piping," November 7, 1983. [8311160350]

616. SECY-84-9, "Report on the Long Term Approach for Dealing with Stress Corrosion Cracking in BWR Piping," January 10, 1984. [8402230344]

617. SECY-84-9A, "Update of Status Report on BWR Pipe Cracks and Projection of Upcoming Licensee Actions," January 27, 1984. [8402230347]

618. SECY-84-166, "Update of Status Report on BWR Pipe Cracks and Projection of Upcoming Licensee Actions," April 20, 1984. [8405180011]

619. SECY-84-301, "Staff Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping," July 30, 1984. [8408090406]

620. NRC Letter to All Licensees of Operating Reactors, Applicants for Operating License, and Holders of Construction Permits for Boiling Water Reactors, "Inspections of BWR Stainless Steel Piping," (Generic Letter 84-11), April 19, 1984. [8404230029]

621. NUREG-0992, "Report of the Committee to Review Safeguards Requirements at Power Reactors," U.S. Nuclear Regulatory Commission, May 1983.

622. Memorandum for T. Speis from R. Mattson, "Fuel Crumbling During LOCA,' February 2, 1983. [8302170511]

623. Memorandum for H. Denton from D. Eisenhut, "Potential Safety Problems Associated with Locked Doors and Barriers in Nuclear Power Plants," December 22, 1983. [8401130140]

624. Memorandum for D. Eisenhut from H. Denton, "Safety-Safeguards Interface," January 16, 1984. [8402010286]

625. Memorandum for H. Thompson from D. Eisenhut, "Potential Safety Problems Associated with Locked Doors and Barriers in Nuclear Power Plants," January 30, 1984. [8402140525]

626. Memorandum for T. Speis from H. Thompson, "Submittal of Potential Generic Issue Associated with Locked Doors and Barriers," June 8, 1984. [8407060042]

627. SECY-83-311, "Proposed Insider Safeguards Rules," July 29, 1983. [8308190179]

628. IE Information Notice No. 83-36, "Impact of Security Practices on Safe Operations," U.S. Nuclear Regulatory Commission, June 9, 1983. [8305110464]

629. Memorandum for H. Thompson from D. Morrison, "Closeout of Generic Safety Issue 78, 'Monitoring of Fatigue Transient Limits for Reactor Coolant System (RCS)' and Generic Safety Issue 166, 'Adequacy of Fatigue Life of Metal Components,'" February 5, 1997. [9703050391]

630. Memorandum for W. Minners from F. Miraglia, "Proposed Generic Issue - Technical Specifications for Anticipatory Trips," February 23, 1984. [8403080271]

631. Memorandum for F. Miraglia from W. Houston, "Task Interface Agreement Task No. 83-77 (TAC 40002, PA-157)," November 29, 1983. [8401060510]

632. NUREG/CR-6117, "Neutron Spectra at Different High Flux Isotope Reactor (HFIR) Pressure Vessel Surveillance Locations," U.S. Nuclear Regulatory Commission, December 1993.

633. Memorandum for P. Check from H. Richings, "Some Notes on PWR (W) Power Distribution Probabilities for LOCA Probabilistic Analyses," July 5, 1977.

634. NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis," U.S. Nuclear Regulatory Commission, April 1980.

635. Memorandum for G. Holahan and W. Minners from R. Mattson, "Disposition of AEOD Engineering and Technical Evaluation Reports," April 10, 1984. [9705190219]

636. Memorandum for R. DeYoung and H. Denton from C. Heltemes, "Vapor Binding of Auxiliary Feedwater Pumps," November 21, 1983. [8312070028]

637. AEOD/C404, "Steam Binding of Auxiliary Feedwater Pumps," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, July 1984. [8408060083]

638. Memorandum for H. Denton from C. Michelson, "Tie Breaker Between Redundant Class 1E Buses - Point Beach Nuclear Plant, Units 1 and 2," August 27, 1980. [8009150214, 8009160668]

639. Letter to J. Keppler (NRC) from C. Fay (Wisconsin Electric Power Company), "Docket No. 50-301, Point Beach Nuclear Plant Unit 2 Licensee Event Report No. 80-005/03L-0," June 27, 1980. [8007080381]

640. Memorandum for H. Denton from C. Heltemes, "Special Study Report - Human Error in Events Involving Wrong Unit or Wrong Train," January 13, 1984. [8401310079]

641. IE Information Notice No. 84-51, "Independent Verification," U.S. Nuclear Regulatory Commission, June 26, 1984. [8406250214]

642. IE Information Notice No. 84-58, "Inadvertent Defeat of Safety Function Caused by Human Error Involving Wrong Unit, Wrong Train, or Wrong System," U.S. Nuclear Regulatory Commission, July 25, 1984. [8407230079]

643. Memorandum for H. Denton from C. Heltemes, "Human Error in Events Involving Wrong Unit or Wrong Train," August 8, 1984. [9705190238]

644. Memorandum for D. Eisenhut, et al., from H. Thompson, "Maintenance and Surveillance Program Implementation Plan," July 7, 1984. [8407160259]

645. Memorandum for C. Heltemes from H. Denton, "Special Study Report - Human Errors in Events Involving Wrong Unit or Wrong Train," May 2, 1984. [8405170027]

646. Memorandum for C. Heltemes from H. Denton, "Human Error in Events Involving Wrong Unit or Wrong Train," September 17, 1984. [8410040282]

647. Memorandum for T. Speis from H. Denton, "Resolution of Generic Issue B-26, 'Structural Integrity of Containment Penetrations,'" September 27, 1984. [8410120090]

648. Memorandum for T. Speis from H. Denton, "Closeout of Generic Issue B-54, 'Ice Condenser Containments,'" October 22, 1984. [8411050142]

649. NUREG/CR-3716, "CONTEMPT 4/MOD 4," U.S. Nuclear Regulatory Commission, March 1984.

650. NUREG/CR-4001, "CONTEMPT 4/MOD 5," U.S. Nuclear Regulatory Commission, September 1984.

651. NUREG-0985, "U.S. Nuclear Regulatory Commission Human Factors Program Plan," U.S. Nuclear Regulatory Commission, August 1983, (Rev. 1) September 1984, (Rev. 2) April 1986.

652. Memorandum for W. Dircks from R. DeYoung, "Elimination of Duplicative Tracking Requirements for Revision of Regulatory Guide 1.33," July 26, 1984. [9705190264]

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.

654. Memorandum for W. Dircks from H. Thompson, "Closeout of TMI Action Plan Task I.G.2, 'Scope of Test Program,'" October 5, 1984. [8410160524]

655. Memorandum for W. Dircks from H. Denton, "Generic Issue II.A.1, 'Siting Policy Reformulation,'" September 17, 1984. [8410090175]

656. Memorandum for W. Dircks from H. Denton, "Closeout of TMI Action Plan Task II.E.5.2, Transient Response of B&W Designed Reactors," September 28, 1984. [8410110596]

657. Memorandum for D. Crutchfield from D. Eisenhut, "TMI Action Plan Task II.E.5.2," November 6, 1984. [8411270129]

658. NUREG-1054, "Simplified Analysis for Liquid Pathway Studies," U.S. Nuclear Regulatory Commission, August 1984.

659. Memorandum for H. Denton from R. Vollmer, "ESRP 7.1.1 'Environmental Impacts of Postulated Accidents Involving Radioactive Materials - Releases to Groundwater,'" September 25, 1984. [8410100758]

660. Memorandum for W. Dircks from H. Denton, "Generic Issue III.D.2.3 'Liquid Pathway Radiological Control,'" October 29, 1984. [8411190057]

661. Memorandum for H. Denton from C. Heltemes, "Failures of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand," April 29, 1983. [8305230511]

662. Memorandum for C. Heltemes from H. Denton, "AEOD April 1983 Report on Failures of Class 1E Safety-Related Switch Gear Circuit Breakers to Close on Demand," June 17, 1983. [8306280125]

663. IE Information Notice No. 83-50, "Failures of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand," August 1, 1983. [8306270418]

664. Memorandum for D. Eisenhut from R. Spessard, "Unmonitored Failures of Class 1E Safety-Related Switchgear Circuit Breakers and Power Supplies (AITS-F03052383)," June 1, 1984. [8408230490]

665. NUREG/CR-2989, "Reliability of Emergency AC Power System at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1983.

666. Memorandum for T. Speis from H. Denton, "Resolution of Generic Issue B-12: BWR Jet Pump Integrity," September 25, 1984. [8410030458]

667. Memorandum for T. Speis from H. Denton, "Resolution of Generic Issue 69: Make-up Nozzle Cracking in B&W Plants," September 27, 1984. [8410150536]

668. Memorandum for H. Denton from R. Minogue, "Comments on Generic Issue 79, 'Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown,'" October 5, 1983. [8310260398]

669. Letter to P. Kadambi (NRC) from F. Miller (B&W Owners Group Analysis Committee), "Transmittal of RV Head Stress Evaluation Program Results," October 15, 1984. [8410190186]

670. Memorandum for H. Denton from R. Mattson, "Generic Issue B-60, Loose Parts Monitoring Systems for Operating Reactors (TACS 52325)," January 10, 1984. [8401180046]

671. Letter to N. Palladino from P. Shewmon, "Control Room Habitability," August 18, 1982. [8207180073]

672. Memorandum for J. Larkins from J. Murphy, "Proposed Resolution of GSI-15, 'Radiation Effects on Reactor Pressure Vessel Supports,'" June 22, 1994. [9407140032]

673. Letter to W. Dircks from J. Ebersole, "ACRS Subcommittee Report on Control Room Habitability," May 17, 1983. [8305260104]

674. Memorandum for W. Dircks from H. Denton, "Control Room Habitability," July 27, 1983. [8308180433]

675. Memorandum for H. Denton from W. Dircks, "Control Room Habitability," August 15, 1983. [8309160034]

676. Memorandum for T. Murley, et al., from H. Denton, " Control Room Habitability," September 19, 1983. [8310120463]

677. Letter to W. Milstead (NRC) from T. Powers (PNL), "A Probabilistic Examination of Nuclear Power Plant Control Room Habitability During Various Accident Scenarios," December 3, 1984. [8412050472]

678. Memorandum for W. Dircks from H. Denton, "Control Room Habitability," June 29, 1984. [8407100196]

679. Memorandum for T. Speis from R. Bernero, "Revised Schedule for Generic Issue 83, Control Room Habitability," September 28, 1984. [8410110484]

680. NUREG/CR-2258, "Fire Risk Analysis for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, September 1981.

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

682. Note to W. Kane from G. Holahan, "Background Information Relating to the Assessment of the Offsite Consequences of Non-Core Melt, Steam Generator Tube Rupture Events," October 24, 1983. [9705190255]

683. Memorandum for W. Johnston from R. Ballard, "Disputed Procedures for Estimating Probable Maximum Precipitation," January 13, 1984. [8401260466]

684. Hydrometeorological Report No. 52, "Application of Probable Maximum Precipitation Estimates - United States East of the 105th Meridian," U.S. Department of Commerce, National Oceanic and Atmospheric Administration, August 1982.

685. Hydrometeorological Report No. 51, "Probable Maximum Precipitation Estimates, United States East of the 105th Meridian," U.S. Department of Commerce, National Oceanic and Atmospheric Administration, June 1978.

686. Hydrometeorological Report No. 33, "Seasonal Variation of the Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1,000 Square Miles and Durations of 6, 12, 24 and 48 Hours," U.S. Department of Commerce, April 1956.

687. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Rev. 2) August 1977. [7907100225]

688. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Rev. 1) September 1976. [7907100372]

689. Memorandum for V. Stello from H. Denton, "Potential Generic Requirement Concerning Design for Probable Maximum Precipitation," June 25, 1984. [8407100105]

690. Memorandum for V. Stello from H. Denton, "Generic Requirements Regarding Design for Probable Maximum Precipitation," October 10, 1984. [8503140522, 8410190029]

691. Memorandum for H. Denton from V. Stello, "Generic Requirements Regarding Design for Probable Maximum Precipitation," August 8, 1984. [8408160442]

692. Memorandum for T. Speis from H. Denton, "Generic Issue A-41; 'Long Term Seismic Program,'" October 10, 1984. [9705200066]

693. Memorandum for H. Denton from R. Bernero, "Resolution of Generic Issue No. 22, Inadvertent Boron Dilution Events (BDES)," September 17, 1984. [8410020424]

694. Memorandum for T. Speis from H. Denton, "Closeout of Generic Issue No. 22, 'Inadvertent Boron Dilution Events (BDE),'" October 15, 1984. [8410310592]

695. Memorandum for T. Speis from H. Denton, "Closeout of Generic Issue 50, 'Reactor Vessel Level Instrumentation in BWRs,'" October 17, 1984. [8411030745]

696. NRC Letter to All Boiling Water Reactor (BWR) Licensees of Operating Reactors (Except LaCrosse, Big Rock Point, Humboldt Bay and Dresden-1), "Reactor Vessel Water Level Instrumentation in BWRs (Generic Letter No. 84-23," October 26, 1984. [8410290050]

697. Memorandum for D. Eisenhut from R. Bernero, "Resolution of Generic Issue 50, Reactor Vessel Level Instrumentation in BWRs," September 6, 1984. [8410010093]

698. NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Rev. 1) March 1984.

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

700. NRC Letter to All Operating PWR Licenses, Construction Permit Holders, and Applicants for Construction Permits, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops (Generic Letter 84-04)," February 1, 1984. [8402010410]

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

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

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

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

705. Memorandum for C. Thomas from O. Parr, "CRD Accumulators - Proposed Improved Technical Specification," August 13, 1984. [8408270516]

706. NUREG-0123, "Standard Technical Specifications for General Electric Boiling Water Reactors (BWR/5)," U.S. Nuclear Regulatory Commission, (Rev. 3) December 1980.

707. Memorandum for H. Denton, et. al., from C. Michelson, "Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980," December 23, 1981. [8202040039]

708. Memorandum for C. Michelson from H. Denton, "NRR Comments on AEOD Draft Report: Survey of Valve Operator-Related Events Occurring During 1978, 1979 and 1980," March 5, 1982. [8203240048]

709. AEOD/C203, "Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, May 1982. [8206180032]

710. Memorandum for C. Michelson from E. Brown and F. Ashe, "AEOD Assessment of Program Office Responses to the Report AEOD/C203, 'Survey of Valve Operator-Related Events Occurring During 1978, 1979, and 1980,'" December 23, 1982. [8301250189, 8301120496]

711. Memorandum for H. Denton from C. Michelson, "AEOD Assessment of Program Office Responses to AEOD Case Study (C-203), 'Survey of Valve Operator Related Events Occurring During 1978, 1979, and 1980,'" January 12, 1983. [8301250183]

712. Memorandum for C. Michelson from H. Denton, "AEOD Assessment of Program Office Responses to AEOD Case Study (C203), 'Survey of Valve Operator Related Events Occurring During 1978, 1979, and 1980,'" February 23, 1983. [8303100567]

713. Memorandum for K. Seyfrit from E. Brown and F. Ashe, "Engineering Evaluation Report AEOD/E305 Inoperable Motor Operated Valve Assemblies Due to Premature Degradation of Motors and/or Improper Limit Switch/Torque Switch Adjustment," April 13, 1983. [8305050353]

714. Memorandum for W. Minners from R. Bosnak, "Status of Potential Generic Issue 54, 'Valve Operator Related Events Occurring During 1978, 1979, and 1980,'" March 26, 1984. [8404110417]

715. Memorandum for R. Vollmer from R. Bosnak, "MEB Task Action Plan for Resolution of Generic Issue II.E.6.1, 'In Situ Testing of Valves,'" July 30, 1984. [8408070139]

716. Memorandum for D. Eisenhut from D. Muller, "PWR Reactor Cavity Uncontrolled Exposures, Generic Letter Implementing a Generic Technical Specification," July 12, 1984. [8407230356]

717. Memorandum for A. Thadani from W. Minners, "CRAC2 Computer Runs in Support of USI A-43," February 1, 1983. [8302090275]

718. Memorandum for W. Minners from F. Congel, "Prioritization of Generic Issue 97: PWR Reactor Cavity Uncontrolled Exposures," February 8, 1985. [8502250136]

719. Memorandum for H. Denton from R. Bernero, "PWR Reactor Cavity Uncontrolled Exposures," November 28, 1984. [8412180620]

720. Memorandum for T. Speis from R. Bernero, "Request for Prioritization of Generic Safety Issue - Break Plus Single Failure in BWR Water Level Instrumentation," October 10, 1984. [8410290282]

721. Memorandum for H. Denton and V. Stello from C. Michelson, "Case Study Report - Safety Concern Associated with Reactor Vessel Instrumentation in Boiling Water Reactors," September 2, 1981. [8109220940]

722. Memorandum for B. Sheron from A. Thadani, "Reactor Vessel Level Instrumentation in BWR's (Generic Issue 50)," August 2, 1984. [8408090089]

723. Memorandum for H. Denton from T. Speis, "Reactor Vessel Level Instrumentation in BWRs (Generic Issue 50)," August 2, 1984. [8408090386, 8408090094]

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

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

726. NEDO-21506, "Stability and Dynamic Performance of the General Electric Boiling Water Reactor," General Electric Company, January 1977.

727. Memorandum for D. Crutchfield from L. Rubenstein, "Staff Evaluation of GE Topical Report NEDE-24011 (GESTAR) Amendment 8," April 17, 1985. [8504290470]

728. XN-NF-691(P)(A) & Supplement 1, "Stability Evaluation of Boiling Water Reactor Cores Sensitivity Analyses & Benchmark Analysis," Exxon Nuclear Company, Inc., August 22, 1984.

729. Memorandum for D. Eisenhut from R. Mattson, "Board Notification - BWR Core Thermal Hydraulic Stability," February 27, 1984. [8403020299]

730. Memorandum for T. Novak from L. Rubenstein, "Susquehanna 1 and 2 - Thermal Hydraulic Stability Technical Specification Change (TACS 55021 and 55022)," July 11, 1984. [8407170149]

731. Memorandum for G. Lainas from L. Rubenstein, "SER Input for Peach Bottom-3 Technical Specification Changes for Cycle 6 Operation with Increased Core Flows and Decreased Feedwater Temperatures (TACS #55123)," October 23, 1984. [8411010312]

732. NEDO-21078, "Test Results Employed by GE for BWR Containment and Vertical Vent Loads," General Electric Company, October 1975.

733. NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," U.S. Nuclear Regulatory Commission, November 1978, (Supplement 1) September 1980.

734. NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments," U.S. Nuclear Regulatory Commission, November 1981.

735. Letter to T. Novak (NRC) from T. Pickens (BWR Owners' Group), "Agreements from BWROG/NRC Meeting on Suppression Pool Temperature Limit," October 16, 1984. [8410220072]

736. Memorandum for T. Speis from R. Bernero, "Proposed Generic Issue 'BWR Suppression Pool Temperature Limits,'" November 21, 1984. [8412030526]

737. Memorandum for W. Minners from W. Butler, "Comments on Prioritization of Generic Issue 108, 'BWR Suppression Pool Temperature Limits,'" January 10, 1985. [8501160095]

738. NUREG-1044, "Evaluation of the Need for a Rapid Depressurization Capability for CE Plant," U.S. Nuclear Regulatory Commission, December 1984.

739. SECY-84-134, "Power Operated Relief Valves for Combustion Engineering Plants," March 23, 1984. [8404180339]

740. "Draft Maintenance Program Plan," U.S. Nuclear Regulatory Commission, May 8, 1984.

741. NUREG/CR-3543, "Survey of Operating Experience from LERs to Identify Aging Trend," U.S. Nuclear Regulatory Commission, January 1984.

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

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

744. NRC Letter to All Power Reactor Licensees (Except Ft. St. Vrain), "NUREG-0744 Rev. 1; Generic Letter No. 82-26) - Pressure Vessel Material Fracture Toughness," November 12, 1982. [8211160047]

745. EPRI NP-3967, "Classification and Analysis of Reactor Operating Experience Involving Dependent Events," Electric Power Research Institute, June 1985.

746. NUREG-0224, "Final Report on Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, September 1978.

747. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants Resolution of Generic Technical Activity A-36," U.S. Nuclear Regulatory Commission, July 1980.

748. NUREG-0763, "Guidelines for Confirmatory Inplant Tests of Safety Relief Valve Discharges for BWR Plants," U.S. Nuclear Regulatory Commission, May 1981.

749. NUREG-0802, "Safety/Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments," U.S. Nuclear Regulatory Commission, October 1982.

750. NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," U.S. Nuclear Regulatory Commission, July 1977, (Rev. 1) July 1980, (Rev. 2) January 1988.

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

752. Memorandum for S. Hanauer from D. Eisenhut, "Value/Impact Assessment of Proposed Steam Generator Generic Requirements," October 12, 1982. [8211110465]

753. SECY-84-13, "NRC Integrated Program for the Resolution of Steam Generator USI's," January 11, 1984. [8401310036]

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. [8303090047]

757. SECY-84-13A, "NRC Integrated Program for the Resolution of Steam Generator USIs," September 7, 1984. [8409140060]

758. SECY-84-13B, "NRC Integrated Program for the Resolution of Steam Generator USI's - Response to Commissioner Comments (Memo from Chilk to Dircks dated September 13, 1984)," November 5, 1984. [8411210357]

759. AEOD/C005, "AEOD Observations and Recommendations Concerning the Problem of Steam Generator Overfill and Combined Primary and Secondary Blowdown," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, December 17, 1980. [8101150366]

760. NUREG/CR-2883, "Study of the Value and Impact of Alternative Decay Heat Removal Concepts for Light Water Reactors," U.S. Nuclear Regulatory Commission, (Vol. 1) June 1983, (Vol. 2) June 1983, (Vol. 3) June 1983.

761. AEOD/E414, "Stuck Open Isolation Check Valve on the Residual Heat Removal System at Hatch Unit 2," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, May 31, 1984. [8406190101]

762. Memorandum for W. Minners from G. Holahan, "Prioritization of Interfacing System LOCA at Boiling Water Reactors," October 25, 1984. [8411050292]

763. NUREG-0677, "The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes," U.S. Nuclear Regulatory Commission, May 1980.

764. SECY-85-129, "Maintenance and Surveillance Program Plan," April 12, 1985. [8509190696]

765. SECY-85-62, "NRC Integrated Program for the Resolution of Steam Generator USI's - Response to Commissioner Comments (Memo from Chilk to Dircks Dated January 23, 1985)," February 22, 1985. [8504080388]

766. Memorandum for W. Dircks from S. Chilk, "SECY-85-62 - NRC Integrated Program for the Resolution of Steam Generator USIs - Response to Commissioners Comments (Memo from Chilk and Dircks Dated January 23, 1985)," March 15, 1985.

767. Memorandum for W. Dircks from H. Denton, "Final Rule - Applicability of License Conditions and Technical Specifications in an Emergency," February 17, 1983. [8303300333]

768. Memorandum for T. Speis from H. Denton, "Formation of a Technical Specification Improvement Project Group," December 31, 1984. [8501150417]

769. Memorandum for V. Stello from H. Denton, "Close Out Generic Issue #B-19 -Thermal-Hydraulic Stability," May 21, 1985. [8506040556]

770. Letter from P. Crane (Pacific Gas and Electric Company) to Director, Division of Licensing, U.S. Nuclear Regulatory Commission, "Report on June 7, 1975 Ferndale Earthquake," August 4, 1975. [8602070315, 993280104, ML993280111]

771. Memorandum for W. Minners from L. Reiter, "Generic Issue No. B-50 Post Operating Basis Earthquake Inspection," June 7, 1985.

772. Letter to A. Schwencer (NRC) from C. Dunn (Duquesne Light Company), "Beaver Valley Power Station, Unit No. 1, Docket No. 50-334, Request for Amendment to the Operating License - No. 35," October 27, 1978. [7811030107]

773. Letter to J. Carey (Duquesne Light Company) from S. Varga (NRC), "Beaver Valley Unit No. 1 - Operation With Two Out of Three Reactor Coolant Loops - Safety Evaluation," July 20, 1984. [8408010218]

774. Memorandum for D. Eisenhut from D. Wigginton, Closeout of MPA E-05; Westinghouse N-1 Loop Operation," January 11, 1985. [8501300565]

775. Memorandum for R. Emrit from A. Murphy, "Generic Issue Management Control System, Issue No. 119.3, Decouple OBE from SSE," February 21, 1992. [9803260147]

776. Memorandum for R. Bernero from D. Eisenhut, "BWR Thermal-Hydraulic Stability Technical Specifications," November 16, 1984. [8411290326]

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. [8507020587]

778. Memorandum for W. Dircks from H. Denton, "TMI Action Item I.A.3.4," February 12, 1985. [8502260084]

779. Memorandum for W. Dircks from J. Taylor, "TMI Action Plan - Completed Item," June 26, 1985. [8507080034]

780. IE Information Notice No. 83-58, "Transamerica DeLaval Diesel Generator Crankshaft Failure," U.S. Nuclear Regulatory Commission, August 30, 1983. [8308040044]

781. IE Information Notice No. 83-51, "Diesel Generator Events," U.S. Nuclear Regulatory Commission, August 5, 1983. [8306270425]

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. [8505130221]

783. SECY-84-34, "Emergency Diesel Generators Manufactured by Transamerica DeLaval, Inc.," January 25, 1984. [8403010451]

784. Letter to D. Bixby (TDI) from D. Eisenhut (NRC), February 14, 1984. [8402290333]

785. TDI Diesel Generators Owners' Group Program Plan, March 2, 1984.

786. SECY-84-155, "Section 208 Report to the Congress on Abnormal Occurrences for October-December, 1983," April 11, 1984. [8405140043]

787. Letter to J. George (Transamerica Delaval, Inc., Owners' Group) from D. Eisenhut (NRC), "Safety Evaluation Report, Transamerica Delaval, Inc. Diesel Generator Owners' Group Program Plan," August 13, 1984. [8408240115]

788. Memorandum for W. Minners from B. Sheron, "Additional Low-Temperature-Overpressure Protection Issues for Light-Water Reactors," August 1, 1984. [8408130012]

789. IE Information Notice No. 83-26, "Failure of Safety/Relief Valve Discharge Line Vacuum Breakers," U.S. Nuclear Regulatory Commission, May 3, 1983. [8303040028]

790. NUREG/CR-3384, "VISA - A Computer Code for Predicting the Probability of Reactor Pressure Vessel Failure," U.S. Nuclear Regulatory Commission, September 1983.

791. Memorandum for K. Seyfrit from C. Hsu, "EE No. AEOD/E322 Damage to Vacuum Breaker Valves as a Result of Relief Valve Lifting," September 21, 1983. [8310060353]

792. AEOD/C401, "Low Temperature Overpressure Events at Turkey Point Unit 4," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, March 1984. [8404050445]

793. Memorandum for B. Sheron from B. Liaw, "Additional Low-Temperature-Overpressure Protection (LTOP) Issues for Light-Water Reactors," August 30, 1984. [8409130397]

794. Memorandum for K. Seyfrit from E. Imbro, 'Single Failure Vulnerability of Power Operated Relief Valve Actuation Circuitry for Low Temperature Overpressure Protection (LTOP)," October 24, 1984. [8411070245]

795. AEOD/C403, "Edwin I. Hatch Unit No. 2 Plant Systems Interaction Event on August 25, 1982," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, May 1984. [8405300746]

796. Memorandum for R. Mattson from T. Dunning, "RHR Interlocks for Westinghouse Plants," April 17, 1984. [8404300085]

797. Memorandum for F. Rowsome from W. Houston, "RCS/RHR Suction Line Valve Interlock on PWRs," August 27, 1984. [8409070331]

798. NSAC-52, "Residual Heat Removal Experience Review and Safety Analysis, Pressurized Water Reactor," Nuclear Safety Analysis Center, January 1983.

799. Memorandum for W. Dircks from H. Denton, "Resolution of Generic Issue III.D.2.3 -- Liquid Pathway Studies," August 28, 1985. [8509050212]

800. NUREG/CR-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. [8507220495]

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. [8510030079]

805. Memorandum for T. Combs from H. Denton, "Revised SRP Section 13.5.2 and Appendix A to SRP Section 13.5.2 of NUREG-0800," July 17, 1985. [8508050283]

806. Memorandum for W. Dircks from H. Denton, "Closeout of TMI Action Plan, Task II.B.6, 'Risk Reduction for Operating Reactors at Sites With High Population Densities,'" September 25, 1985. [8510030342]

807. Memorandum for W. Dircks from R. Minogue, "Closeout of TMI Action Plan Task II.B.8 'Rulemaking Proceeding on Degraded Core Accidents - Hydrogen Control,'" July 19, 1985. [8508010066]

808. Memorandum for W. Dircks from H. Denton, "Close Out of TMI Action Plan, Task II.B.8," August 12, 1985. [8508210316]

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

810. NUREG/CR-3085, "Interim Reliability Evaluation Program: Analysis of the Millstone Point Unit 1 Nuclear Power Plant," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1983, (Vol. 2) August 1983, (Vol. 3) July 1983, (Vol. 4) July 1983.

811. NUREG/CR-3511, "Interim Reliability Evaluation Program: Analysis of the Calvert Cliffs Unit 1 Nuclear Power Plant," U.S. Nuclear Regulatory Commission, (Vol. 1) May 1984, (Vol. 2) October 1984.

812. NUREG/CR-2728, "Interim Reliability Evaluation Program Procedures Guide," U.S. Nuclear Regulatory Commission, March 1983.

813. Memorandum for W. Dircks from R. Minogue, "Closeout of TMI Action Plan, Task II.C.1, 'Interim Reliability Evaluation Program,'" July 9, 1985. [8507180593]

814. SECY-84-133, "Integrated Safety Assessment Program (ISAP)," March 23, 1984. [8404100072]

815. SECY-85-160, "Integrated Safety Assessment Program - Implementation Plan," May 6, 1985. [8505230571]

816. Memorandum for W. Dircks from H. Denton, "Close-out of Generic Issues II.C.2, 'Continuation of IREP,' and IV.E.5, 'Assess Currently Operating Reactors,'" September 25, 1985. [9909290069]

817. Memorandum for W. Dircks from R. Minogue, "Closeout of TMI Action Plan Task II.E.2.2, 'Research on Small Break LOCA's and Anomalous Transients,'" July 25, 1985. [9909290072]

818. Memorandum for W. Dircks from J. Taylor, "TMI Action Plan - Completed Item," August 15, 1985. [8508200726]

819. EPRI EL-3209, "Workshop Proceedings: Retaining Rings for Electric Generators," Electric Power Research Institute, August 1983.

820. Memorandum for R. Fraley from R. Vollmer, "Proposed NRR Revisions to Review Procedures for Turbine Missile Issue," May 12, 1983. [8305250286]

821. Memorandum for W. Johnson from T. Novak, "Midland SSER #3 - Turbine Missile Review," November 1, 1983. [8311140470]

822. Memorandum for V. Stello from H. Denton, "NRR Plans for Approval of WCAP-10271," January 11, 1985. [8501220433, 8501220440]

823. Letter to J. Sheppard (Westinghouse Owners Group) from C. Thomas (NRC), "Acceptance for Referencing of Licensing Topical Report WCAP-10271, 'Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation Systems,'" February 21, 1985. [8503010427]

824. Memorandum for T. Speis from R. Mattson, "Request for Prioritization of Generic Safety Issue - Failure of HPCI Steam Line Without Isolation," October 18, 1983. [8311020209]

825. Memorandum for K. Seyfrit from P. Lam, "Failure of an Isolation Valve of the Reactor Core Isolation Cooling System to Open Against Operating Reactor Pressure," August 23, 1984. [8411010554, 8411010453]

826. Letter to A. Schwencer (NRC) from J. Kemper (Philadelphia Electric Company), "Limerick Generating Station, Units 1 & 2, Request for Additional Information from NRC Equipment Qualification Branch (EQB)," February 27, 1984. [9909290076]

827. NEDO-24708A, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," General Electric Company, August 1979 [7909130302, 7909130304], December 1980.

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

829. "An Evaluation of Unisolated LOCA Outside the Drywell in the Shoreham Nuclear Power Station," Brookhaven National Laboratory, June 1985. [9909290080]

830. Memorandum for W. Minners from A. Thadani, "Comments on Generic Issue No. 87 - Failure of HPCI Steam Line Without Isolation," June 28, 1985. [8507170422]

831. NUREG/CR-1433, "Examination of the Use of Potassium Iodide (KI) as an Emergency Protective Measure for Nuclear Reactor Accidents," U.S. Nuclear Regulatory Commission, October 1980.

832. SECY-83-362, "Emergency Planning - Predistribution/Stockpiling of Potassium Iodide for the General Public," August 30, 1983. [8309080120]

833. SECY-85-167, "Federal Policy Statement on the Distribution and Use of Potassium Iodide," May 13, 1985. [8505310621]

834. Memorandum for H. Denton and R. Minogue from W. Dircks, "Review of NRC Requirements for Nuclear Power Plant Piping," August 1, 1983. [8308300212]

835. Memorandum for W. Dircks from R. Minogue, "Plan to Implement Piping Review Committee Recommendations," July 30, 1985. [9705050005]

836. Memorandum for T. Murley, et al., from J. Taylor, "Results of Regional Survey of Plant Specific Information Relating to the Potential for Uncontrolled Radiation Exposures in PWR Reactor Cavities," June 18, 1985. [8506250113]

837. Note to R. Vollmer from T. Speis, "Proposed Request to Perform Research on the Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments," January 7, 1985. [9909290082]

838. NUREG-1165, "Environmental Standard Review Plan for ES Section 7.1.1," U.S. Nuclear Regulatory Commission, November 1985.

839. Letter to J. Bayne (PASNY) from S. Varga (NRC), "Steam Generator Tube and Girth Weld Repairs at the Indian Point Nuclear Generating Plant, Unit No. 3 (IP-3)," May 27, 1983. [8306150627]

840. "Value-Impact Analysis of Recommendations Concerning Steam Generator Tube Degradations and Rupture Events," Science Applications, Inc., February 2, 1983.

841. Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, July 1975, (Rev. 1) April 1977 [7907100362], (Rev. 2) May 1988 [8907270187].

842. IE Information Notice No. 82-37, "Cracking in the Upper Shell to Transition Cone Girth Weld of a Steam Generator at an Operating Pressurized Water Reactor," U.S. Nuclear Regulatory Commission, September 16, 1982. [8208190220]

843. Letter to D. Smith (NRC) from E. Rahe (Westinghouse), January 17, 1982. [9909290085]

844. NUREG/CR-3281, "Investigation of Shell Cracking on the Steam Generators at Indian Point Unit No. 3," U.S. Nuclear Regulatory Commission, June 1983.

845. NUREG/CR-3614, "Constant Extension Rate Testing of SA302 Grade B Material in Neutral and Chloride Solutions," U.S. Nuclear Regulatory Commission, February 1984.

846. Letter to W. Hazelton (NRC) from H. Watanabe (GE), "'Laboratory Examination of Garigliano Secondary Steam Generator-B Core Samples,' NEDE-25162, July 1979," December 13, 1979. [7912130566]

847. EPRI NP-1136, "Limiting Factor Analysis of High Availability Nuclear Plants (Boiling Water Reactors)," Electric Power Research Institute, (Vol. 1) August 1979.

848. Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," U.S. Nuclear Regulatory Commission, (Rev. 1) July 1978.

849. NUREG/CR-3842, "Steam Generator Group Project Task 8 - Selective Tube Unplugging," U.S. Nuclear Regulator Commission, July 1984.

850. NRC Letter to All PWR Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, and Ft. St. Vrain, "Staff Recommended Actions Stemming from NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity (Generic Letter 85-02)," April 17, 1985. [8504120031]

851. NUREG/CP-0058, "Twelfth Water Reactor Safety Research Information Meeting," U.S. Nuclear Regulatory Commission, (Vol. 4) January 1985.

852. NUREG/CP-0044, "Proceedings of the International Atomic Energy Agency Specialists' Meeting on Subcritical Crack Growth," U.S. Nuclear Regulatory Commission, (Vol. 1) May 1983, (Vol. 2) May 1983.

853. "Corrosion Fatigue Crack Growth in Reactor Pressure Vessel Steels - Structural Integrity of Light Water Reactor Components," Scott, P. et al., Elsevier Science Publishing Co., Inc., 1982.

854. NUREG/CR-4121, "The Effects of Sulfur Chemistry and Flow Rate on Fatigue Crack Growth Rates in LWR Environments," U.S. Nuclear Regulatory Commission, February 1985.

855. NUREG-0975, "Compilation of Contract Research for the Materials Engineering Branch, Division of Engineering Technology," U.S. Nuclear Regulatory Commission, (Vol. 2) March 1984.

856. PNO-II-85-41, "Small Steam Generator Surface Cracks," U.S. Nuclear Regulatory Commission, April 23, 1985. [8504290412]

857. Memorandum for W. Minners from B. Liaw, "Prioritization of Generic Issue No. (111) Stress Corrosion Cracking of RCPB Ferritic Steels and Steam Generator Vessels," June 7, 1985. [8506170320]

858. IE Information Notice No. 85-65, "Crack Growth in Steam Generator Girth Welds," U.S. Nuclear Regulatory Commission, July 31, 1985. [8507290456]

859. Memorandum for H. Thompson from J. Knight, "Steam Generator Shell Transition Joint Cracking," July 10, 1985. [8507190409]

860. NUREG-0937, "Evaluation of PWR Response to Main Steamline Break With Concurrent Steam Generator Tube Rupture and Small-Break LOCA," U.S. Nuclear Regulatory Commission, December 1982. [8412190335]

861. SECY-83-357B, "Status of Hydrogen Control Issue and Rulemaking Recommendations in SECY-83-357A," December 3, 1984. [8412190335]

862. IE Bulletin No. 79-13, "Cracking in Feedwater System Piping," June 25, 1979 [7906250348], (Rev. 1) August 29, 1979 [7908220101], (Rev. 2) October 17, 1979 [7908220135].

863. Memorandum for T. Speis from H. Denton, "Closeout of Generic Issues B-58 and C-11," July 9, 1985. [8507180530]

864. AEOD/C301, "Failures of Class 1E Safety-Related Switchgear Circuit Breakers to Close on Demand," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, April 1983. [8305230531]

865. Memorandum for T. Speis from H. Denton, "Resolution of Generic Issue 14, 'PWR Pipe Cracks,'" October 4, 1985. [9909290092]

866. Federal Register Notice 47 FR 7023, "Proposed Policy Statement on Safety Goals for Nuclear Power Plants," February 17, 1982.

867. Federal Register Notice 48 FR 10772, "Safety Goal Development Program," March 14, 1983.

868. Letter to J. Ahearne from M. Plesset, "Recommendations of President's Commission on ACRS Role," January 15, 1980. [8002150071]

869. Federal Register Notice 46 FR 22358, "10 CFR Part 2, ACRS Participation in NRC Rulemaking," April 17, 1981.

870. Memorandum for Commissioner Ahearne, et al., from L. Bickwit, et al., "TMI Action Plan, Chapter V, Formal Procedures for Ensuring Periodic Public Interaction," October 2, 1980.

871. Memorandum for W. Dircks, et al., from J. Hoyle, "Staff Requirements - Discussion of Action Plan, Chapter V (See SECY-80-230B), 2:00 p.m. Monday, July 7, 1980, Commissioners' Conference Room, D.C. Office (Open to Public Attendance)," July 9, 1980. [8012030928]

872. Federal Register Notice 45 FR 49535, "10 CFR Part 2, Procedural Assistance in Adjudicatory Licensing Proceedings," July 25, 1980.

873. Federal Register Notice 46 FR 13681, "10 CFR Part 2, Domestic Licensing Proceedings; Procedural Assistance Program," February 24, 1981.

874. Memorandum for L. Bickwit from S. Chilk, "SECY-81-391 - Provision of Free Transcripts to All Full Participants in Adjudicatory Proceedings: May 11, 1981 Comptroller General Decision," February 25, 1982.

875. Federal Register Notice 45 FR 34279, "10 CFR Parts 2, 50, Possible Amendments to 'Immediate Effectiveness Rule,'" May 22, 1980.

876. Federal Register Notice 47 FR 47260, "10 CFR Part 2, Commission Review Procedures for Power Reactor Construction Permits; Immediate Effectiveness Rule," October 25, 1982.

877. Federal Register Notice 51 FR 10393, "10 CFR Parts 0 and 2, Revision of Ex Parte and Separation of Functions Rules Applicable to Formal Adjudicatory Proceedings," March 26, 1986.

878. NUREG-0632, "NRC Views and Analysis of the Recommendations of the President's Commission on the Accident at Three Mile Island," U.S. Nuclear Regulatory Commission, November 1979.

879. Federal Register Notice 46 FR 28533, "Statement of Policy on Conduct of Licensing Proceedings," May 27, 1981.

880. Memorandum to All Employees from N. Palladino, "Regulatory Reform Task Force," November 17, 1981.

881. Letter to the Honorable Thomas P. O'Neill, Jr. from N. Palladino, February 21, 1983.

882. Federal Register Notice 48 FR 44173, "10 CFR Part 50, Revision of Backfitting Process for Power Reactors," September 28, 1983.

883. Federal Register Notice 48 FR 44217, "10 CFR Part 50, Revision of Backfitting Process for Power Reactors," September 28, 1983.

884. Federal Register Notice 50 FR 38097, "10 CFR Parts 2 and 50, Revision of Backfitting Process for Power Reactors," September 20, 1985.

885. Memorandum for H. Thompson from D. Crutchfield, "Potential Immediate Generic Actions as a Result of the Davis-Besse Event of June 9, 1985," August 5, 1985. [8508090679]

886. NUREG-1154, "Loss of Main and Auxiliary Feedwater Event at the Davis-Besse Plant on June 9, 1985," U.S. Nuclear Regulatory Commission, July 1985.

887. Memorandum for T. Speis from H. Thompson, "Short Term Generic Actions as a Result of the Davis-Besse Event of June 9, 1985," August 19, 1985. [8508270246]

888. Memorandum for H. Denton from T. Speis, "Adequacy of the Auxiliary Feedwater System at Davis-Besse," July 23, 1985. [8508010086]

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

890. NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, June 1988.

891. Letter to T. Novak (NRC) from R. Crouse (Toledo Edison Company), December 31, 1981. [8201060607]

892. NUREG/CR-2770, "Common Cause Fault Rates for Valves," U.S. Nuclear Regulatory Commission, February 1983.

893. NUREG/CR-2098, "Common Cause Fault Rates for Pumps," U.S. Nuclear Regulatory Commission, February 1983.

894. Memorandum for O. Parr from A. Thadani, "Auxiliary Feedwater System - CRGR Package," November 9, 1984. [8411280233]

895. Memorandum for H. Denton, et al., from W. Dircks, "Staff Actions Resulting from the Investigation of the June 9 Davis-Besse Event (NUREG-1154)," August 5, 1985. [8508090534]

896. SECY-86-56, "Status of Staff Study to Determine if PORVs Should be Safety Grade," February 18, 1986. [8611100428]

897. Memorandum for G. Lainas from F. Rowsome, "Safety Evaluation of the CE Licensees' Responses to TMI Action Item II.K.3.2," August 26, 1983. [8309060394]

898. Memorandum for G. Lainas from F. Rowsome, "Safety Evaluation of the B&W Licensees' Responses to TMI Action Item II.K.3.2," August 24, 1983. [8308310422]

899. Memorandum for G. Lainas from F. Rowsome, "Safety Evaluation of the Westinghouse Licensees' Responses to TMI Action Item II.K.3.2," July 22, 1983. [8308040054]

900. Memorandum for H. Thompson from W. Russell, "Comments on Draft List of Longer Term Generic Actions as a Result of the Davis-Besse Event of June 9, 1985," September 19, 1985. [8509240326]

901. Memorandum for T. Combs from H. Denton, "Revised SRP Section 9.2.1 and SRP Section 9.2.2 of NUREG-0800," June 24, 1986. [8607080481]

902. Memorandum for J. Sniezek and R. Fraley from H. Denton, "Resolution of Generic Issue No. 36, 'Loss of Service Water,'" May 13, 1986. [8605300159]

903. Memorandum for T. Speis from H. Denton, "Resolution of Generic Issue 3, 'Setpoint Drift in Instrumentation,'" May 19, 1986. [8606110638]

904. SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," July 9, 1983. [8308080642]

905. Memorandum for T. Speis from R. Bernero, "Enhancement of the Reliability of Westinghouse Solid State Protection System (SSPS)," April 5, 1985. [8504160610]

906. NUREG/CR-3971, "A Handbook for Cost Estimating," U.S. Nuclear Regulatory Commission, October 1984.

907. Memorandum for W. Minners from B. Sheron, "Generic Issues C-4, C-5, C-6," May 29, 1985. [8506100882]

908. SECY-83-472, "Emergency Core Cooling System Analysis Methods," November 17, 1983. [8401060169]

909. AEOD/C503, "Decay Heat Removal Problems at U.S. Pressurized Water Reactors," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, December 1985. [8601060316]

910. Memorandum for H. Denton from C. Heltemes, "Case Study Report - Decay Heat Removal Problems at U.S. Pressurized Water Reactors," December 23, 1985. [8601060315]

911. Memorandum for C. Heltemes from H. Denton, "AEOD's Report on Decay Heat Removal Problems at U.S. PWRs," February 10, 1986. [8602200004]

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.

914. Memorandum for H. Thompson and T. Speis from R. Bernero, "Request for Comments on Draft CRGR Package with Requirements for Upgrading Auxiliary Feedwater Systems in Certain Operating Plants," October 3, 1985. [8510090228]

915. Memorandum for W. Minners from A. Thadani, "Seismic Induced Relay Chatter Issue," March 22, 1985.

916. Regulatory Guide 1.29, "Seismic Design Classification," U.S. Nuclear Regulatory Commission, June 1972, (Rev. 1) August 1973 [8003280778], (Rev. 2) February 1976, (Rev. 3) September 1978 [7810030052].

917. Regulatory Guide 1.100, "Seismic Qualification of Electrical Equipment for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1976, (Rev. 1) August 1977.

918. NUREG/CP-0070, "Proceeding of the Workshop on Seismic and Dynamic Fragility of Nuclear Power Plant Components," U.S. Nuclear Regulatory Commission, August 1985.

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

920. ANSI/ANS 5.1, "Decay Heat Power in Light Water Reactors," American National Standards Institute, 1979.

921. Letter to the Honorable Morris K. Udall from Joseph M. Hendrie, August 7, 1978. [7901030172, 8001230259]

922. Letter to Joseph Hendrie from Morris K. Udall, January 27, 1978. [8007210279, 8007180431]

923. Memorandum for J. Taylor from D. Morrison, "Resolution of Generic Safety Issue 15, 'Radiation Effects on Reactor Vessel Supports,'" May 29, 1996. [9606190081]

924. SECY-96-107, "Uniform Tracking of Agency Generic Technical Issues," May 14, 1996. [9605230140]

925. Memorandum for E. Beckjord from T. Murley, "Regulatory Guide 1.44," April 30, 1992. [9205110015]

926. Memorandum for Record from E. McGregor, "SECY-80-366 - NRC Legislative Program for 97th Congress," April 8, 1981.

927. Memorandum for Chairman Palladino, et al., from A. Kenneke, "TMI Action Plan, Chapter V," May 18, 1984.

928. Memorandum for A. Thadani from T. Speis, "Generic Safety Issue (GSI)-166, 'Adequacy of Fatigue Life of Metal Components,'" August 26, 1996. [9808210022]

929. Regulatory Guide 1.139, "Guidance for Residual Heat Removal," U.S. Nuclear Regulatory Commission, May 1978.

930. NUREG-0957, "The Price-Anderson Act - The Third Decade," U.S. Nuclear Regulatory Commission, December 1983.

931. NUREG-0689, "Potential Impact of Licensee Default on Cleanup of TMI-2," U.S. Nuclear Regulatory Commission, November 1980.

932. SECY-83-64A, "10 CFR 140: Proposed Rule to Revise the Criteria for Determination of an Extraordinary Nuclear Occurrence," August 9, 1983. [8308250291]

933. Memorandum for A. Kenneke from W. Olmstead, "Chapter 5 of TMI Action Plan," March 16, 1984. [8404040211]

934. Letter to the Honorable Alan Simpson from Joseph Hendrie, March 24, 1981. [8104030556]

935. NUREG/CR-1368, "Development of a Checklist for Evaluating Maintenance, Test and Calibration Procedures Used in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, May 1980.

936. NUREG/CR-1369, "Procedures Evaluation Checklist for Maintenance, Test and Calibration Procedures," U.S. Nuclear Regulatory Commission, May 1980.

937. Memorandum for Chairman Ahearne from W. Dircks, "Manual Chapters Delegation of Authority to Staff Office Directors," December 23, 1980.

938. SECY-80-497, "Review of Delegations of Authority and Other Documentation," November 10, 1980. [8011190612]

939. Federal Register Notice 51 FR 28044, "Safety Goals for the Operations of Nuclear Power Plants," August 4, 1986.

940. Memorandum for T. Speis from H. Thompson, "Longer-Term Generic Actions as a Result of the Davis-Besse Event of June 9, 1985," November 6, 1985. [8511120162]

941. Memorandum for B. Morris from D. Basdekas, "Concerns Related to the Davis-Besse Incident on June 9, 1985," August 13, 1985. [8508230349]

942. Memorandum for F. Gillespie from D. Basdekas, "Concerns Related to the Davis-Besse Incident on June 9, 1985," September 27, 1985. [9909290115]

943. Memorandum for A. DeAgazio from D. Crutchfield, "Davis-Besse Restart Safety Evaluation (TAC No. 59702)," December 17, 1985. [8512230373]

944. Letter to G. Ogeka (BNL) from T. Speis (NRC), "BNL Technical Assistance to the Division of Safety Review and Oversight, Office of Nuclear Reactor Regulation, NRC - 'Reduction of Risk Uncertainty' (FIN A-3846)," April 28, 1986. [9909290117]

945. Memorandum for K. Kniel from R. Riggs, "OTSG Thermal Stress (GI-125.II.4)," June 17, 1986. [8608070348]

946. Memorandum for H. Thompson from R. Bernero, "Auxiliary Feedwater Systems," August 23, 1985. [8509030040]

947. Memorandum for B. Boger from A. Gody, "Implementation of the Resolution for Generic Issue 142, 'Leakage Through Electrical Isolators,'" May 28, 1993. [9803260145]

948. Memorandum for H. Thompson from G. Edison, "Recommendation for Longer Term Generic Action as a Result of Davis-Besse Event of June 9, 1985," September 11, 1985. [9909290121]

949. Memorandum for F. Miraglia from G. Edison, "Prioritization of Generic Issue 125.II.I.D," April 25, 1986. [8605050358]

950. BAW-1919, "B&W Owners' Group Trip Reduction and Transient Response Improvement Program," May 31, 1986. [8606020079, 8605190153]

951. Memorandum for H. Thompson and W. Minners from F. Rowsome, "Another Generic Safety Issue Suggested by the Davis-Besse Incident of June 9, 1985," September 9, 1985. [8509110328]

952. Memorandum for W. Minners from K. Kniel, "Value/Impact Assessment for Draft CRGR Package Requiring Upgrading of Auxiliary Feedwater Systems in Certain Operating Plants," January 16, 1986. [8601240311]

953. Memorandum for G. Mazetis from A. Marchese, "Revised Outline of Regulatory Analysis for USI A-45," January 14, 1986. [9909290124]

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

955. Memorandum for W. Dircks from H. Denton, "Close Out of Completed TMI Action Plan Item I.C.9, 'Long-Term Program Plan for Upgrading of Procedures,'" June 7, 1985. [8506200155]

956. Memorandum for V. Stello from H. Denton, "Close-out of the Division of Human Factors Technology TMI Action Plan Items," January 6, 1987. [8701140115]

957. Federal Register Notice 49 FR 46428, "10 CFR Parts 50 and 55, Operator's Licenses and Conforming Amendment," November 26, 1984.

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. [8605130362]

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 & Constructors), "Decay Heat Removal Systems Evaluations Feasibility and Cost Evaluations of Special Issues Related to Decay Heat Removal," January 20, 1986. [9910200312]

961. NUREG/CR-4627, "Generic Cost Estimates," U.S. Nuclear Regulatory Commission, June 1986, (Rev. 1) February 1989, (Rev. 2) February 1992.

962. NUREG-1021, "Operator Licensing Examiner Standards," U.S. Nuclear Regulatory Commission, October 1983.

963. SECY-85-21, "Policy Statement on Fitness for Duty of Nuclear Power Plant Personnel," January 17, 1985. [8502280427]

964. SECY-85-21A, "Withdrawal Notice: Fitness for Duty of Nuclear Power Plant Personnel," April 12, 1985. [8505030703]

965. SECY-85-21B, "Fitness for Duty of Nuclear Power Plant Personnel," August 26, 1985. [8510150472]

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.

967. Federal Register Notice 51 FR 27921, "Commission Policy Statement on Fitness for Duty of Nuclear Power Plant Personnel," August 4, 1986.

968. Memorandum for J. Roe from R. Minogue, "Nuclear Plant Analyzer (NPA) Management Plan," December 12, 1985. [9909290129]

969. NUREG/CR-3403, "Criteria and Test Method for Certifying Air-Purifying Respirator Cartridges and Canisters Against Radioiodine," U.S. Nuclear Regulatory Commission, November 1983.

970. NUREG/CR-3568, "A Handbook for Value-Impact Assessment," U.S. Nuclear Regulatory Commission, December 1983.

971. NUREG/CR-4330, "Review of Light Water Reactor Regulatory Requirements," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1986, (Vol. 2) June 1986.

972. SECY-80-230B, "Update of Chapter V of TMI Action Plan: NRC Policy, Organization, and Management," June 20, 1980. [8009160065]

973. Memorandum for T. Speis from W. Minners, "Schedule for Resolving Generic Issue No. 125.II.1.b, 'Review Existing AFW Systems for Single Failure,'" December 10, 1986. [8612180094]

974. NUREG-1122, "Knowledges and Abilities Catalog for Nuclear Power Plant Operators: Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, July 1985.

975. IE Circular No. 80-02, "Nuclear Power Plant Staff Work Hours," U.S. Nuclear Regulatory Commission, February 1, 1980. [7912190657]

976. NRC Letter to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, "Interim Criteria for Shift Staffing (Generic Letter 80-72)," July 31, 1980. [8009020297]

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. [8206160341]

980. NRC Letter to All Pressurized Power Reactor Licensees, "NUREG-0737 Technical Specifications (Generic Letter No. 82-16)," September 20, 1982. [8209210027]

981. NRC Letter to All Boiling Water Reactor Licensees, "NUREG-0737 Technical Specifications (Generic Letter No. 83-02)," January 10, 1983. [8301110134]

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. [8303040005]

983. Memorandum for W. Dircks from J. Hoyle, "Updating NRC Policy Statements," September 30, 1985. [8611190084]

984. Memorandum for J. Tourtelotte, et al., from S. Chilk, "Addendum to SRM M841218 - Briefing and Discussion on the Hearing Process, 2:00 p.m., Tuesday, December 18, 1984, Commissioners' Conference Room, D.C. Office (Open to Public Attendance)," January 31, 1985. [8502060511]

985. Federal Register Notice 48 FR 50550, "10 CFR Part 2, Rules of Practice for Domestic Licensing Proceedings; Role of NRC Staff in Adjudicatory Licensing Hearings," November 2, 1983.

986. Federal Register Notice 51 FR 36811, "10 CFR Part 2, Rules of Practice for Domestic Licensing Proceedings; Role of NRC Staff in Adjudicatory Licensing Hearings," October 16, 1986.

987. Federal Register Notice 49 FR 14698, "10 CFR Parts 2 and 50, Request for Public Comment on Regulatory Reform Proposal Concerning the Rules of Practice, Rules for Licensing of Production and Utilization Facilities," April 12, 1984.

988. Federal Register Notice 51 FR 24365, "10 CFR Part 2, Rules of Practice for Domestic Licensing Proceedings - Procedural Changes in the Hearing Process," July 3, 1986.

989. Federal Register Notice 50 FR 13978, "10 CFR Part 140, Criteria for an Extraordinary Nuclear Occurrence," April 9, 1985.

990. Memorandum for J. Funches from F. Rowsome, "Handling of DHFT Issues in GIMCS," June 6, 1986. [8606120789]

991. Memorandum for T. Speis from R. Bernero, "Resolution of Comment No. 9 of CRGR/OIA Issues on Potential Generic Concerns Regarding BWR Drywell Coolers," July 31, 1986. [8608190656]

992. Federal Register Notice 50 FR 42145, "10 CFR Part 1, Statement of Organization and General Information," October 18, 1985.

993. NUREG-1220, "Training Review Criteria and Procedures," U.S. Nuclear Regulatory Commission, July 1986.

994. Federal Register Notice 48 FR 31611, "10 CFR Part 50, Licensed Operator Staffing at Nuclear Power Plants," July 11, 1983.

995. Regulatory Guide 1.114, "Guidance on Being Operator at the Controls of a Nuclear Power Plant," U.S. Nuclear Regulatory Commission, February 1976 [8012110846], (Rev. 1) November 1976 [8307070393], (Rev. 2) May 1989 [8906200342].

996. Federal Register Notice 50 FR 43621, "Commission Policy Statement on Engineering Expertise on Shift," October 28, 1985.

997. Memorandum for W. Dircks from H. Denton, "Human Factors Program Plan (HFPP)," December 6, 1984. [8501080482]

998. Memorandum for T. Speis from H. Denton, "Resolution of Generic Safety Issue 61, 'SRV Line Break Inside the Wetwell Airspace of Mark I and II Containments,'" August 8, 1986. [8608180209]

999. NUREG/CR-4594, "Estimated Safety Significance of Generic Safety Issue 61," U.S. Nuclear Regulatory Commission, June 1986.

1000. Memorandum for T. Speis, et al., from R. Mattson, "Generic Issue 23, 'Reactor Coolant Pump Seal Failures' - Task Action Plan," October 26, 1983. [8311080469]

1001. Memorandum for H. Denton from T. Speis, "Integration of Electrical Power Issues into Proposed Generic Issue 128, 'Electrical Power Reliability,'" November 28, 1986. [8612080528]

1002. Memorandum for H. Clayton from B. Sheron, "Criteria for Initiating Feed and Bleed," September 13, 1985. [8509180314]

1003. Memorandum for W. Russell from K. Perkins, "Generic Issue 125.I.8, 'Procedures and Staffing for Reporting to NRC Operations Center,'" November 25, 1986. [8612050442]

1004. Memorandum for G. Lainas and D. Crutchfield from F. Rowsome, "Davis-Besse Restart Considerations," August 13, 1985. [8508210208]

1005. Memorandum for V. Stello from D. Ward, "ACRS Comments on Proposed Resolution of Generic Issue 124, 'Auxiliary Feedwater System Reliability,'" September 17, 1986. [8609230137]

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. [8605200493]

1008. Memorandum for E. Jordan from G. Holahan, "Proposed IE Information Notice," June 6, 1986. [8606110821]

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. [8607240026]

1011. NUREG-1177, "Safety Evaluation Report Related to the Restart of Davis-Besse Nuclear Power Station, Unit 1, Following the Event of June 9, 1985," U.S. Nuclear Regulatory Commission, June 1986.

1012. Federal Register Notice 50 FR 29937, "10 CFR Part 50, Analysis of Potential Pressurized Thermal Shock Events," July 23, 1985.

1013. NUREG-1212, "Status of Maintenance in the U.S. Nuclear Power Industry 1985," U.S. Nuclear Regulatory Commission, (Volumes 1 and 2), June 1986.

1014. Memorandum for F. Schroeder from D. Crutchfield, "Dynamic Qualification Testing of Large Bore Hydraulic Snubbers," March 6, 1985. [8503180471]

1015. Memorandum for R. DeYoung, et al., from C. Heltemes, "Failure of Large Hydraulic Snubbers to Lock-up," September 21, 1984. [8410290312, 8410290114]

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 Licenses to Operate Power Reactors, "Technical Specification for Snubbers (Generic Letter 84-13)," May 3, 1984. [8405040043]

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.

1023. SECY-86-231, "Survey on Engineering Expertise on Shift," August 6, 1986. [8608200375]

1024. Memorandum for K. Kniel from C. Ferrell, "Modification of Generic Issue No. 106, 'Highly Combustible Gases in Vital Areas,'" February 20, 1986. [8602280811]

1025. IE Information Notice No. 83-41, "Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment," U.S. Nuclear Regulatory Commission, June 22, 1983. [8305110477]

1026. Letter to D. Farrar (Commonwealth Edison Co.) from J. Zwolinski (NRC), "Technical Specifications Relating to the Use of a Mobile Volume Reduction System (MVRS) at Dresden Station (TAC 56373, 56374)," August 13, 1986. [8608210177]

1027. Memorandum for D. Eisenhut from G. Lainas, "Summary of the Operating Reactor Events Meeting," January 28, 1982. [8310260053]

1028. Memorandum for R. Vollmer and E. Jordan from C. Michelson, "Effects of Fire Protection System Actuation on Safety Related Equipment," January 28, 1982. [8202220663]

1029. "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations," BWR Owners Group for IGSCC Research, Hydrogen Installation Subcommittee, Electric Power Research Institute, 1987.

1030. NASA TMX-71565, "Review of Hydrogen Accidents and Incidents in NASA Operation," National Aeronautics and Space Administration, August 1974.

1031. Memorandum for T. Murley from E. Beckjord, "A New Generic Issue: Multiple Steam Generator Tube Leakage," June 16, 1992. [9212040356]

1032. Memorandum for H. Denton from T. Speis, "Earthquakes and Emergency Planning," January 18, 1984. [8402020014]

1033. Letter to W. Dircks (NRC) from S. Sholly (Union of Concerned Scientists), December 22, 1983. [8502270371]

1034. Letter to J. Asselstine (NRC) from S. Sholly (Union of Concerned Scientists), December 22, 1983. [8502090516]

1035. SECY-85-283, "Final Amendments to 10 CFR Part 50, Appendix E; Consideration of Earthquakes in Emergency Planning," August 21, 1985. [8508300319]

1036. IE Bulletin No. 85-03, "Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings," U.S. Nuclear Regulatory Commission, November 15, 1985 [8511130441], (Supplement 1) April 27, 1988 [8804210018].

1037. SECY-83-484, "Requirements for Emergency Response Capability," November 29, 1983. [8312130459]

1038. IE Information Notice No. 86-10, "Safety Parameter Display System Malfunctions," U.S. Nuclear Regulatory Commission, February 13, 1986. [8602100408]

1039. Memorandum for H. Denton from T. Speis, "Prioritization of Selected MPAs (Operating Plan, Item VI.B.6.b)," October 19, 1984. [8411010640]

1040. NUREG/CR-3246, "The Effect of Some Operations and Control Room Improvements on the Safety of the Arkansas Nuclear One, Unit One, Nuclear Power Plant," U.S. Nuclear Regulatory Commission, June 1983.

1041. Memorandum for K. Kniel from R. Bosnak, "Request for Subsumption of Generic Issue B-6 (GI B-6) Into Generic Isssue 119.1 (GI 119.1)," January 8, 1987. [8701200186]

1042. SECY-87-101, "Issues and Proposed Options Concerning Degree Requirement for Senior Operators," April 16, 1987. [8706030157]

1043. SECY-86-348, "Final Rulemaking for Revisions to Operator Licensing - 10 CFR 55 and Conforming Amendments," November 21, 1986. [8701020003]

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. [8710280270]

1046. Memorandum for V. Stello from E. Beckjord, "Closeout of TMI Action Plan Item," February 27, 1987. [9704150146]

1047. Memorandum for K. Kniel from B. Sheron, "Request for the Prioritization of a Generic Issue on the Reliability of PWR Main Steam Safety Valves," May 27, 1986. [8604030313]

1048. IE Information Notice No. 86-05, "Main Steam Safety Valve Test Failures and Ring Setting Adjustments," U.S. Nuclear Regulatory Commission, January 31, 1986 [8601290054], (Supplement 1) October 16, 1986 [8610100107].

1049. Memorandum for F. Cherny from R. Baer, "50.55(e) Report on Crosby Main Steam Valve Ring Settings," February 5, 1985. [8502140267, 9704090262]

1050. Memorandum for R. Bosnak from F. Cherny, "Trip Report - Meeting of ASME Section III Subgroup on Pressure Relief, February 11, 1987," March 13, 1987. [8703190114]

1051. INPO 82-025, "Review of NRC Report: Precursors to Potential Severe Core Damage Accidents: 1969-1979 A Status Report, NUREG/CR-2497," Institute for Nuclear Power Operations, September 1982.

1052. NUREG/CR-2228, "Containment Response During Degraded Core Accidents Initiated by Transients and Small Break LOCA in the Zion/Indian Point Reactor Plants," U.S. Nuclear Regulatory Commission, July 1981.

1053. NUREG/CR-4752, "Coincident Steam Generator Tube Rupture and Stuck-Open Safety Relief Valve Carryover Test," U.S. Nuclear Regulatory Commission, March 1987.

1054. Memorandum for W. Russell, et al., from R. Starostecki, "Request for Regional Inspection to Verify Adequate Flow Capacity of Main Steam Code Safety Valves and Proper Ring Adjustments," November 8, 1987. [8711120155]

1055. AEOD/C204, "San Onofre Unit 1 Loss of Salt Water Cooling Event on March 10, 1980," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, July 1982. [8208260403]

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

1057. NUREG-0897, "Containment Emergency Sump Performance," U.S. Nuclear Regulatory Commission, (Rev. 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, June 30, 1974 [7902090041], (Rev. 1) November 30, 1985 [8512100138], (Rev. 2) May 31, 1996 [9605210504].

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. [8511270253]

1060. SECY-85-349, "Resolution of Unresolved Safety Issue A-43, 'Containment Emergency Sump Performance,'" October 31, 1985. [8511070302]

1061. NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1980, (Rev. 1) September 1984.

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

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.

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.

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. [8702200135]

1070. NUREG-1216, "Safety Evaluation Report Related to the Operability and Reliability of Emergency Diesel Generators Manufactured by Transamerica Delaval, Inc.," U.S. Nuclear Regulatory Commission, August 1986.

1071. Memorandum for T. Speis, et al., from C. Berlinger, "Closeout of Generic Issue 91 - TDI Emergency Diesel Generator Reliability," September 3, 1987. [8709080427]

1072. Memorandum for W. Russell from T. Speis, "Generic Issue 125.II.13 - Operator Job Aids," June 12, 1986. [8606250128]

1073. SECY-83-288, "Pressurized Thermal Shock (PTS) Rule," July 15, 1983. [8307270206]

1074. Memorandum for W. Dircks from S. Chilk, "SECY-83-288, 'Proposed Pressurized Thermal Shock (PTS) Rule,'" January 13, 1984. [8402100267]

1075. Memorandum for K. Kniel from R. Bosnak, "Integration of NUREG-0933 Issues," May 27, 1986. [8606090491]

1076. NUREG-1230, "Compendium of ECCS Research for Realistic LOCA Analysis," U.S. Nuclear Regulatory Commission, December 1988. [8903030340]

1077. Federal Register Notice 52 FR 9453, "10 CFR Parts 50 and 55, Operators' Licenses and Conforming Amendments," March 25, 1987.

1078. AEOD/C701, "Air Systems Problems at U.S. Light Water Reactors," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, March 1987. [8707240066]

1079. NUREG-1275, "Operating Experience Feedback Report," U.S. Nuclear Regulatory Commission, (Vol. 1) July 1987, (Vol. 2) December 1987, (Vol. 3) November 1988, (Vol. 4) March 1989, (Vol. 5) March 1989, (Vol. 5, Addendum) August 1989, (Vol. 6) February 1991, (Vol. 7) September 1992, (Vol. 8) December 1992, (Vol. 9) March 1993.

1080. NUREG/CR-4374, "A Review of the Oconee-3 Probabilistic Risk Assessment," U.S. Nuclear Regulatory Commission, (Vol. 1) March 1986, (Vol. 2) March 1986, (Vol. 3) June 1986.

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

1082. NUREG/CR-3673, "Economic Risks of Nuclear Power Reactor Accidents," U.S. Nuclear Regulatory Commission, May 1984.

1083. Memorandum for T. Speis from F. Gillespie, "Review of RES Proposed Prioritization of Generic Issue (GI) 125.II.11, 'Recovery of Main Feedwater as an Alternative to Auxiliary Feedwater,'" April 27, 1988. [8805120322]

1084. NUREG-1258, "Evaluation Procedure for Simulation Facilities Certified Under 10 CFR 55," U.S. Nuclear Regulatory Commission, December 1987.

1085. NRC Letter to All Operating Reactor Licensees, Applicants for an Operating License and Holders of Construction Permits for Babcock & Wilcox Pressurized Water Reactors, "Safety Evaluation of 'Abnormal Transient Operating Guidelines,' (Generic Letter 83-31)," September 19, 1983. [8309190017]

1086. Memorandum for B. Morris from B. Sheron, "LOCA Concern of SCE Employee," April 28, 1987. [9704150141]

1087. Federal Register Notice 50 FR 27006, "10 CFR Part 50, Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures," July 1, 1985.

1088. UCID-20397, "Assessment of Value-Impact Associated with the Elimination of Postulated Pipe Ruptures from the Design Basis for Nuclear Power Plants," Lawrence Livermore National Laboratory, March 29, 1985.

1089. Letter to the Honorable Edward J. Markey (Committee on Energy and Commerce, U.S. House of Representatives) from L. Zech (NRC), March 20, 1987. [8703270224]

1090. GAO/RCED-88-73, "Nuclear Regulation - Action Needed to Ensure that Utilities Monitor and Repair Pipe Damage," U.S. General Accounting Office, March 1988.

1091. Memorandum for D. Morrison from H. Thompson, "Generic Issue Management Control System," January 17, 1997. [9803260111]

1092. EPRI NP-5410, "Nondestructive Evaluation of Ferritic Piping for Erosion-Corrosion," Electric Power Research Institute, September 1987.

1093. NRC Bulletin No. 87-01, "Thinning of Pipe Walls in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 9, 1987. [8707020018]

1094. SECY-88-50, "Status Report on Pipe Wall Thinning (Responses to NRC Bulletin 87-01 on Pipe Wall Thinning in Nuclear Power Plants)," February 22, 1988. [8809090066]

1095. NRC Information Notice No. 88-17, "Summary of Responses to NRC Bulletin 87-01, Thinning of Pipe Walls in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, April 22, 1988. [8804180039]

1096. SECY-88-50A, "Report on the Meeting with NUMARC, EPRI, and INPO on Status of Industry's Erosion/Corrosion Program," May 10, 1988. [8805230074]

1097. Memorandum to J. Taylor and W. Parler from S. Chilk, "COMSECY-93-029 - Draft Rulemaking Package on License Renewal; SECY-93-049 - Implementation of 10 CFR Part 54, 'Requirements for Renewal of Operating Licenses for Nuclear Power Plants'; SECY-93-113 - Additional Implementation Information for 10 CFR Part 54, 'Requirements for Renewal of Operating Licenses for Nuclear Power Plants,'" June 28, 1993. [9409010107]

1098. Memorandum for V. Stello from T. Murley, "Resolution of Generic Issue I.A.4.2(4) 'Review Simulators for Conformance to Criteria,'" May 28, 1988. [8806020275]

1099. Memorandum for B. Morris from B. Sheron, "Updated GIMCS for GI I.D.5(5)," February 2, 1988. [9704150145]

1100. Memorandum for V. Stello from E. Beckjord, "Redesignation of Generic Issue I.D.5(5), 'Disturbance Analysis Systems,'" February 22, 1988. [8809190312]

1101. Memorandum for V. Stello from E. Beckjord,""Closure of Generic Issue I.D.4 'Control Room Design Standard,'" March 28, 1988. [9704160014]

1102. Memorandum for T. Speis from R. Houston, Integration of Generic Issue Resolution," November 4, 1987. [9704150161]

1103. Memorandum for V. Stello from E. Beckjord, "Resolution of Generic Safety Issue II.E.4.3, 'Containment Integrity Check,'" March 22, 1988. [8809150125]

1104. NUREG-1273, "Technical Findings and Regulatory Analysis for Generic Safety Issue II.E.4.3, 'Containment Integrity Check,'" U.S. Nuclear Regulatory Commission, April 1988.

1105. Memorandum for T. Speis from G. Arlotto, "Generic Issues Program," January 14, 1988. [9704160053]

1106. Memorandum for R. Baer from G. Bagchi, "Proposed Resolution of Generic Issue B-5, 'Buckling of Steel Containment,'" March 1, 1988. [8804270290]

1107. Memorandum for E. Beckjord from G. Arlotto, "Closeout of Generic Issue B-5, Buckling Behavior of Steel Containments," April 28, 1988. [8805050117]

1108. NUREG-1109, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout," U.S. Nuclear Regulatory Commission, June 1988.

1109. Federal Register Notice 53 FR 23203, "10 CFR 50, Station Blackout," June 21, 1988.

1110. Regulatory Guide 1.155, "Station Blackout," U.S. Nuclear Regulatory Commission, June 1988. [8907270193]

1111. NRC Letter to All Licensees of Operating Boiling Water Reactors (BWRs), and Holders of Construction Permits, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping (Generic Letter 88-01)," January 25, 1988. [8801260537]

1112. IE Bulletin No. 85-01, "Steam Binding of Auxiliary Feedwater Pumps," U.S. Nuclear Regulatory Commission, October 29, 1985. [8510250539]

1113. NRC Letter to All Licensees, Applicants for Operating Licenses, and Holders of Construction Permits for Pressurized Water Reactors, "Resolution of Generic Safety Issue 93, 'Steam Binding of Auxiliary Feedwater Pumps' (Generic Letter 88-03)," February 17, 1988. [8802180267]

1114. Memorandum for E. Beckjord from T. Murley, "Resolution of Generic Safety Issue 93, 'Steam Binding of Auxiliary Feedwater Pumps,'" August 14, 1987. [8708210408]

1115. Memorandum for E. Beckjord from F. Gillespie, "Review of RES-Proposed Prioritization of Generic Issue No. 136, 'Storage and Use of Large Quantities of Cryogenic Combustibles on Site,'" March 25, 1988. [8804050182]

1116. Federal Register Notice 53 FR 9430, "Final Commission Policy Statement on Maintenance of Nuclear Power Plants," March 23, 1988.

1117. Memorandum for V. Stello from T. Murley, "Closeout of Generic Issue HF-08, 'Maintenance and Surveillance Program,'" May 4, 1988. [8805160004]

1118. SECY-88-248, "Implementation of the Severe Accident Policy for Future Light Water Reactors," September 6, 1988. [8809160019]

1119. NUREG/CR-4780, "Procedures for Treating Common Cause Failures in Safety and Reliability Studies," U.S. Nuclear Regulatory Commission, (Vol. 1) January 1988, (Vol. 2) January 1989.

1120. NUREG-1192, "An Investigation of the Contributors to Wrong Unit or Wrong Train Events," U.S. Nuclear Regulatory Commission, April 1986.

1121. Information Notice No. 87-25, "Potentially Significant Problems Resulting from Human Error Involving Wrong Unit, Wrong Train, or Wrong Components," U.S. Nuclear Regulatory Commission, June 11, 1987. [8706050211]

1122. Memorandum for V. Stello from T. Murley, "Final Resolution of Generic Issue (GI) 102: Human Error in Events Involving Wrong Unit or Wrong Train," September 12, 1988. [8810070118]

1123. Dam Failure Model, Pacific Northwest Laboratories, October 1983.

1124. "Analysis of Gradual Earth-Dam Failure," Journal of Hydraulic Engineering, Volume 114, No. 1, American Society of Civil Engineers, January 1988.

1125. "Use of A Dam Break Model to Assess Flooding at Haddam Neck Nuclear Power Plant," Water Resources Bulletin, Vol. 20, No. 6, American Water Resources Association, December 1984.

1126. Technical Evaluation Report, "Quabbin Dam Failure Flooding Consequences at Haddam Neck Plant," Franklin Research Center, August 25, 1983.

1127. "Dam Breach Parameters, Outflow Peaks, and Flood Stages," International Symposium on Hydrometeorology, American Water Resources Association, June 1982.

1128. PB82-224577, "Application of and Guidelines for Using Available DAM Break Models," Tennessee Water Resources Research Center, May 1981.

1129. IE Bulletin No. 82-02, "Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants," U.S. Nuclear Regulatory Commission, June 2, 1982. [8204210380]

1130. RIL 158, "Operational Safety Reliability Program," U.S. Nuclear Regulatory Commission, October 31, 1988. [8811070111]

1131. Memorandum for V. Stello from E. Beckjord, "Closure of Generic Issue II.C.4, 'Reliability Engineering,'" October 31, 1988. [8811150124]

1132. Memorandum for E. Beckjord from F. Gillespie, "Generic Issue 139, 'Thinning of Carbon Steel Piping in LWRs,'" December 27, 1988. [8901130015]

1133. NUREG-1332, "Regulatory Analysis for the Resolution of Generic Issue 125.II.7, 'Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break,'" U.S. Nuclear Regulatory Commission, September 1988.

1134. Memorandum for V. Stello from E. Beckjord, "Resolution of Generic Issue 125.II.7, 'Reevaluate Provision to Automatically Isolate Feedwater from Steam Generator During a Line Break,'" September 9, 1988. [8811290524]

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

1136. Memorandum to C. Miller from R. Borchardt, "Review of Temporary Instruction 2515/131, 'Licensee Offsite Communication Capabilities,' for Deletion from the NRC Inspection Manual," December 3, 1996. [9612060074]

1137. SECY-86-97, "Steam Generator USI Program - Utility Responses to Staff Recommendations in Generic Letter 85-02," March 24, 1986. [8609160048]

1138. NRC Bulletin No. 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," February 5, 1988. [8802020035]

1139. SECY-88-272, "Technical Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," September 27, 1988. [8811040042]

1140. NRC Information Notice No. 87-28, "Air Systems Problems at U.S. Light Water Reactors," June 22, 1987 [8706170115], (Supplement 1) December 28, 1987 [8712230003].

1141. NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants, "Instrument Air Supply System Problems Affecting Safety-Related Equipment (Generic Letter 88-14)," August 8, 1988. [8808120294]

1142. Memorandum for V. Stello from E. Beckjord, "Resolution of Generic Issue 43, Air Systems Reliability," September 30, 1988. [9704160039]

1143. SECY-88-260, "Shutdown Decay Heat Removal Requirements (USI A-45)," September 13, 1988. [8811040098]

1144. NUREG/CR-5015, "Improved Reliability of Residual Heat Removal Capability in PWRs as Related to Resolution of Generic Issue 99," U.S. Nuclear Regulatory Commission, May 1988.

1145. NRC Letter to All Holders of Operating Licenses or Construction Permits for Pressurized Water Reactors (PWRs), "Loss of Decay Heat Removal (Generic Letter No. 88-17) 10 CFR 50.54f," October 17, 1988. [8810180350]

1146. Memorandum for V. Stello from E. Beckjord, "Resolution of Generic Issue 99, 'Loss of RHR Capability in PWRs,'" November 2, 1988. [8811290361]

1147. Memorandum for V. Stello from T. Murley, "Final Resolution of Generic Issue (GI) 66, 'Steam Generator Requirements,'" November 28, 1988. [8812010081]

1148. Memorandum for W. Minners from F. Rowsome, "A Candidate Generic Issue," December 11, 1984. [8501080138]

1149. Regulatory Guide 1.65, "Materials and Inspections for Reactor Vessel Closure Studs," U.S. Atomic Energy Commission, October 1973. [7907100246]

1150. Memorandum for W. Minners from A. Thadani, "Prioritization of RHR Suction Valve Testing," May 7, 1984. [8405180403]

1151. NUREG/CR-2934, "Review and Evaluation of the Indian Point Probabilistic Safety Study," U.S. Nuclear Regulatory Commission, December 1982.

1152. NUREG/CR-3300, "Review and Evaluation of the Zion Probabilistic Safety Study," U.S. Nuclear Regulatory Commission, (Vol. 1) May 1984.

1153. Memorandum for F. Cherny from W. Minners, "Reactor Coolant System Pressure Isolation Valve (PIV) Leak Test Requirements," July 2, 1985. [8507120595]

1154. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," U.S. Atomic Energy Commission, March 1972. [7907100118]

1155. Memorandum for R. Bernero from T. Speis, "Relationship of TIA 84-72 (Haddam Neck Refueling Cavity Seal Failure) to Generic Issue No. 82 (Beyond Design Basis Accidents in Spent Fuel Pools)," April 11, 1985. [8504240705]

1156. Memorandum for K. Kniel from W. Minners, "Refueling Cavity Seal Failure," April 1, 1986. [8604080427]

1157. NUREG/CR-4982, "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82," U.S. Nuclear Regulatory Commission, July 1987.

1158. IE Bulletin No. 84-03, "Refueling Cavity Seal Failure," U.S. Nuclear Regulatory Commission, August 24, 1984. [8408240358]

1159. Letter to D. Crutchfield (NRC) from W. Counsil (Connecticut Yankee Atomic Power Company), "Haddam Neck Plant Reactor Cavity Seal Ring Failure," September 12, 1984. [8409250335]

1160. Memorandum for K. Kniel from W. Minners, "Refueling Cavity Seal Failure," May 8, 1986. [8605210217]

1161. Memorandum for K. Kniel from W. Minners, "Proposed Generic Issue - Fission Product Removal by Containment Sprays or Pools," March 10, 1987. [8703170451]

1162. Federal Register Notice 54 FR 3701, "[NUREG-0800] Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants; Issuance and Availability," January 25, 1989.

1163. Memorandum for T. Speis from K. Kniel, "Treatment of Lessons-Learned from Surry Event as Related to Generic Issues," March 31, 1987. [8704030542]

1164. Memorandum for T. Speis from R. Bernero, "Prioritization of Generic Issue - Valve Interlocks to Prevent Vessel Draining During Shutdown Cooling," May 21, 1986. [8606120635]

1165. AEOD/E609, "Inadvertent Draining of Reactor Vessel During Shutdown Cooling Operation," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, August 1986. [8608290176, 8608290040]

1166. Memorandum for T. King from K. Kniel, "Additional Comments Regarding Prioritization of Generic Issue-129, 'Residual Heat Removal System Valve Mis-alignment during Shutdown Cooling Operations,'" December 7, 1988. [8812210402]

1167. Memorandum for W. Minners from B. Sheron, "Proposed Generic Issue, 'Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Designed Plants,'" August 27, 1985. [8509050358]

1168. Letter to F. Miraglia (NRC) from G. Goering (Westinghouse Owners Group), "Potential Seismic Interaction Associated with the Flux Mapping System in Westinghouse Plants," June 10, 1985. [8509050363]

1169. NUREG/CR-2000, "Licensee Event Report (LER) Compilation," U.S. Nuclear Regulatory Commission, (Vol. 3, No. 7) August 1984.

1170. Memorandum for T. King from R. Riggs, "Computer Program 'SEALCOM' Used in Generic Issue 131," May 1, 1989. [9704160010]

1171. IE Information Notice No. 85-45, "Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Designed Plants," U.S. Nuclear Regulatory Commission, June 6, 1985. [8506060677]

1172. Letter to R. Engelken (NRC) from H. Ray (Southern California Edison Company), "Docket No. 50-361, Licensee Event Report, Numbers 82-002 and 82-003, San Onofre Nuclear Generating Station, Unit 2," March 30, 1982. [8204140262]

1173. Letter to R. Haynes (NRC) from C. Mathis (Boston Edison Company), "Docket No. 50-293, License DPR-35," September 15, 1982. [8209280087]

1174. NUREG-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," U.S. Nuclear Regulatory Commission, (Vols. I and II) April 1989.

1175. SECY-89-081, "Final Report on Chernobyl Implications," March 7, 1989. [8903200205]

1176. AEOD/S801, "Significant Events that Involved Procedures," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, March 1988. [8907310351, 8906090032]

1177. NRC Letter to All Holders of Operating Licenses, Applicants for Operating Licenses and Holders of Construction Permits for Power Reactors, "NRC Use of the Terms, 'Important to Safety' and 'Safety Related"(Generic Letter 84-01)," January 5, 1984. [8401050382]

1178. NRC Memorandum and Order CLI-84-9, June 6, 1984. [8406070146]

1179. SECY-85-119, "Issuance of Proposed Rule on the Important-to-Safety Issue," April 5, 1985. [8505030656]

1180. Memorandum for W. Dircks from S. Chilk, "Staff Requirements -- SECY-85-119 - 'Issuance of Proposed Rule on the Important-to-Safety Issue,'" December 31, 1985. [8601160559]

1181. SECY-86-164, "Proposed Rule on the Important-to-Safety Issue," May 29, 1986. [8607010004]

1182. Memorandum for V. Stello from E. Beckjord, "Resolution of Generic Issue I.F.1, "Expand QA List,'" January 12, 1989. [9704150147]

1183. Memorandum for W. Minners from L. Engle, "Generic Implications/LLNL Technical Evaluation Report on Seven Main Transformer Failures at the North Anna Power Station, Units 1 and 2," November 16, 1984. [8411270057]

1184. UCID-20053, "Technical Evaluation Report on the Seven Main Transformer Failures at the North Anna Power Station, Units 1 and 2," Lawrence Livermore National Laboratory, March 29, 1984. [8412120181, 8412070065]

1185. Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Draft) June 1976, (Draft) November 1977.

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

1187. Memorandum for V. Stello from E. Beckjord, "Closeout of Generic Issue II.F.5, 'Classification of Instrumentation, Control and Electrical Equipment,'" May 5, 1989. [8906270390]

1188. SECY-83-221, "Prioritization of Generic Safety Issues," June 7, 1983. [8306150099]

1189. Memorandum for W. Dircks from S. Chilk, "SECY-83-221 - Prioritization of Generic Safety Issues," December 9, 1983. [9704150148]

1190. Federal Register Notice 43 FR 1565, "Program for Resolution of Generic Issues Related to Nuclear Power Plants," January 10, 1978.

1191. Federal Register Notice 54 FR 24432, "Program for Resolution of Generic Issues Related to Nuclear Power Plants; Policy Statement," June 7, 1989.

1192. RES Office Letter No. 1, "Procedure for Identification, Prioritization, Resolution, and Tracking of Generic Issues," Office of Nuclear Regulatory Research, December 3, 1987 [9704150149], (Rev. 1) March 22, 1989 [9609200344], (Rev. 2) July 12, 1991 [9107250098], (Rev. 3) November 26, 1991 [9704150154], (Rev. 4) June 2, 1994 [9704150218].

1193. RES Office Letter No. 2, "Procedures for Obtaining Regulatory Impact Analysis Review and Support," Office of Nuclear Regulatory Research, November 18, 1988. [8901180069]

1194. RES Office Letter No. 3, "Procedure and Guidance for the Resolution of Generic Issues," May 10, 1988 [8809220069], (Rev. 1) December 21, 1988 [9704100054], (Rev. 2) March 27, 1989 [9609200351].

1195. NUREG/CR-1251, "Implications of the Accident at Chernobyl for Safety Regulation of Commercial Nuclear Power Plants in the United States," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1989, (Vol. 2) April 1989.

1196. NUREG/CR-5176, "Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants," U.S. Nuclear Regulatory Commission, January 1989.

1197. NUREG/CR-5281, "Value/Impact Analyses of Accident Preventive and Mitigative Options for Spent Fuel Pools," U.S. Nuclear Regulatory Commission, March 1989.

1198. NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82 'Beyond Design Basis Accidents in Spent Fuel Pools,'" U.S. Nuclear Regulatory Commission, April 1989.

1199. Memorandum for V. Stello from E. Beckjord, "Resolution of Generic Issue, 'Beyond Design Basis Accidents in Spent Fuel Pools,'" April 24, 1989. [9704100053]

1200. NUREG/CR-5197, "Evaluation of Generic Issue 115, 'Enhancement of the Reliability of Westinghouse Solid State Protection System,'" U.S. Nuclear Regulatory Commission, January 1989.

1201. NUREG-1341, "Regulatory Analysis for the Resolution of Generic Issue 115, 'Enhancement of the Reliability of the Westinghouse Solid State Protection System,'" U.S. Nuclear Regulatory Commission, May 1989.

1202. Memorandum for V. Stello from E. Beckjord, "Resolution of Generic Issue 115, 'Enhancement of the Reliability of Westinghouse Solid State Protection Systems,' NUREG-1341," April 17, 1989. [9608210072]

1203. Memorandum for V. Stello from T. Murley, "Plant-Specific Backfit for Improved Auxiliary Feedwater System Reliability at Arkansas Nuclear One, Unit 2 and Rancho Seco," January 31, 1989. [8902030163]

1204. Memorandum for V. Stello from T. Murley, "Final Resolution of Generic Issue (GI) 122.2, 'Initiating Feed and Bleed,'" April 26, 1989. [8905090075]

1205. NRC Letter to All Licensees of Operating Plants, Applicants for Operating Licenses, and Holders of Construction Permits, "Task Action Plan I.D.2 - Safety Parameter Display System - 10 CFR §50.54(f) - (Generic Letter No. 89-06)," April 12, 1989. [8904120042]

1206. NUREG-1342, "A Status Report Regarding Industry Implementation of Safety Parameter Display Systems," U.S. Nuclear Regulatory Commission, April 1989.

1207. Memorandum for V. Stello from T. Murley, "Final Resolution of Generic Issue 125.I.3, 'SPDS Availability,'" April 26, 1989. [8905050362]

1208. Memorandum for V. Stello from T. Murley, "Final Resolution of Generic Issue (GI) HF4.1, Inspection Procedure for Upgraded Emergency Operating Procedures," October 17, 1988. [8811070169]

1209. NUREG-1358, "Lessons Learned from the Special Inspection Program for Emergency Operating Procedures," U.S. Nuclear Regulatory Commission, April 1989.

1210. NRC Information Notice No. 86-64, "Deficiencies in Upgrade Programs for Plant Emergency Operating Procedures," August 14, 1986 [8608120028], (Supplement 1) April 20, 1987 [8704160062].

1211. NUREG/CR-5088, "Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues," U.S. Nuclear Regulatory Commission, January 1989.

1212. NUREG/CR-5112, "Evaluation of Boiling Water Reactor Water-Level Sensing Line Break and Single Failure," U.S. Nuclear Regulatory Commission, March 1989.

1213. NRC Letter to All Holders of Operating Licenses or Construction Permits for Boiling Water Reactors, "Resolution of Generic Issue 101, 'Boiling Water Reactor Water Level Redundancy' (Generic Letter 89-11)," June 30, 1989. [8906300178]

1214. Memorandum for V. Stello from E. Beckjord, "Closeout of GI 101, 'Boiling Water Reactor Water Level Redundancy,'" April 24, 1989. [9704100038]

1215. Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," U.S. Nuclear Regulatory Commission, November 1975, (Rev. 1) March 1977. [7907100392]

1216. NUREG-1296, "Thermal Overload Protection for Electric Motors on Safety-Related Motor-Operated Valves - Generic Issue II.E.6.1," U.S. Nuclear Regulatory Commission, June 1988.

1217. NRC Letter to All Licensees of Operating Power Plants and Holders of Construction Permits for Nuclear Power Plants, "Safety-Related Motor-Operated Valve Testing and Surveillance (Generic Letter No. 89-10) - 10 CFR 50.54(f)," June 28, 1989 [8906290082], (Supplement 1) June 13, 1990 [9201300217], (Supplement 2) August 3, 1990 [9007310052], (Supplement 3) October 25, 1990 [9010220146], (Supplement 4) February 12, 1992 [9202250311], (Supplement 5) June 28, 1993 [9306230099], (Supplement 6) March 8, 1994 [9402280155].

1218. Memorandum for V. Stello from E. Beckjord, "Close-out of Generic Issue II.E.6.1, 'In Situ Testing of Valves,'" June 30, 1989. [8907100275]

1219. Memorandum for F. Rowsome from D. Crutchfield, "Potential Generic Issue: Loss of Effective Volume for Containment Recirculation Spray," July 13, 1984. [8407240406]

1220. Memorandum for G. Lainas from R. Houston, "Task Interface Agreement (TIA) #83-144: Loss of Effective Volume for Containment Recirculation Spray for H.B. Robinson, Unit 2 (TAC #53223)," August 6, 1984. [8408130232]

1221. Memorandum for W. Minners from F. Rowsome, "Candidate Generic Safety Issue: Allowable Outage Times for Diverse, Simultaneous Equipment Outages," May 9, 1985. [8506030097]

1222. NRC Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR § 50.54(f), (Generic Letter No. 88-20)," November 23, 1988 [8811280048], (Supplement 1) August 29, 1989 [8908300001], (Supplement 2) April 4, 1990 [9003300127], (Supplement 3) July 6, 1990 [9007020114], (Supplement 4) June 28, 1991 [9106270324], (Supplement 5) September 8, 1995.

1223. Proceedings of the International Topical Meeting on Probability, Reliability, and Safety Assessment, PSA '89, p.48, "Potential Underestimation of Test and Maintenance Unavailabilities in Probabilistic Risk Assessments," American Nuclear Society, April 2-7, 1989.

1224. Memorandum for B. Morris from F. Gillespie, "Prioritization of GI-117, 'Allowable Outage Times for Diverse Simultaneous Equipment Outages,'" August 4, 1989. [9704100058]

1225. Federal Register Notice 46 FR 58484, "10 CFR Part 50, Interim Requirements Related to Hydrogen Control," December 2, 1981.

1226. Federal Register Notice 50 FR 3498, "10 CFR Part 50, Hydrogen Control Requirements," January 25, 1985.

1227. SECY-89-122, "Resolution of Unresolved Safety Issue (USI) A-48, 'Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment,'" April 19, 1989. [8905010149]

1228. NUREG-0943, "Threaded-Fastener Experience in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, January 1983.

1229. EPRI NP-3784, "A Survey of the Literature on Low-Alloy Steel Fastener Corrosion in PWR Power Plants," Electric Power Research Institute, December 1984.

1230. EPRI RP 2520-7, "Degradation and Failure of Bolting in Nuclear Power Plants," Electric Power Research Institute, June 1987.

1231. EPRI NP-2174, "A Study of Bolting Problems, Tools, and Practices in the Nuclear Industry," Electric Power Research Institute, December 1981.

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

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

1234. SECY-89-230, "Unresolved Safety Issue A-17, 'Systems Interactions in Nuclear Power Plants,'" August 1, 1989. [8908140127]

1235. NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants, "Resolution of Unresolved Safety Issue A-17, 'Systems Interactions in Nuclear Power Plants' (Generic Letter 89-18)," September 6, 1989. [8909070029]

1236. Federal Register Notice 54 FR 34836, "Issuance and Availability of NUREG-1174, 'Evaluation of Systems Interactions in Nuclear Power Plants: Technical Findings Related to Unresolved Safety Issue A17,' and NUREG-1229, 'Regulatory Analysis for Resolution of USI A-17, - Systems Interactions in Nuclear Power Plants,'" August 22, 1989.

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

1238. NUREG/CR-5437, "Recommendations for Resolution of Public Comments on USI A-40, 'Seismic Design Criteria,'" U.S. Nuclear Regulatory Commission, June 1989.

1239. IE Bulletin No. 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts," U.S. Nuclear Regulatory Commission, March 8, 1979 [7903140038], (Rev. 1) June 20, 1979 [7906200183], (Rev. 2) November 8, 1979 [7908220136].

1240. IE Bulletin No. 79-14, "Seismic Analysis for As-Built Safety-Related Piping Systems," U.S. Nuclear Regulatory Commission, July 2, 1979 [7907060295], (Rev. 1) July 18, 1979 [7907250430].

1241. IE Bulletin No. 80-11, "Masonry Wall Design," U.S. Nuclear Regulatory Commission, May 8, 1980. [7912190695]

1242. NUREG/CR-1161, "Recommended Revisions to Nuclear Regulatory Commission Seismic Design Criteria," U.S. Nuclear Regulatory Commission, May 1980.

1243. NUREG/CR-3480, "Value/Impact Assessment for Seismic Design Criteria USI A-40," U.S. Nuclear Regulatory Commission, August 1984.

1244. NUREG-1233, "Regulatory Analysis for USI A-40, 'Seismic Design Criteria,'" U.S. Nuclear Regulatory Commission, September 1989.

1245. SECY-89-296, "Unresolved Safety Issue A-40, 'Seismic Design Criteria,'" September 22, 1989. [8910060116]

1246. Federal Register Notice 54 FR 40220, "Issuance and Availability Final Resolution of Unresolved Safety Issue (USI) A-40; Seismic Design Criteria," September 29, 1989.

1247. NUREG-1217, "Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plants," U.S. Nuclear Regulatory Commission, June 1989.

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

1249. SECY-89-255, "Unresolved Safety Issue A-47, 'Safety Implications of Control Systems,'" August 23, 1989. [8908250318]

1250. NRC Letter to All Licensees of Operating Reactors, Applicants for Operating Licenses and Holders of Construction Permits for Light Water Reactor Nuclear Power Plants, "Request for Action Related to Resolution of Unresolved Safety Issue A-47, 'Safety Implication of Control Systems in LWR Nuclear Power Plants' Pursuant to 10 CFR 50.54(f) - Generic Letter 89-19,'" September 20, 1989. [8909200223]

1251. Federal Register Notice 54 FR 36922, "Issuance and Availability of NUREG-1217, 'Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plants - Technical Findings Related to USI A-47,' and NUREG-1218, 'Regulatory Analysis for Resolution of USI A-47,'" September 5, 1989.

1252. Memorandum for T. King from C. Serpan, "Reevaluation of Issue 15, 'Radiation Effects on Reactor Vessel Supports,'" September 30, 1988. [9704100071]

1253. ORNL/TM-10444, "Evaluation of HFIR Pressure-Vessel Integrity Considering Radiation Embrittlement," Oak Ridge National Laboratory, April 1988.

1254. NUREG/CR-5320, "Impact of Radiation Embrittlement on Integrity of Pressure Vessel Supports for Two PWR Plants," U.S. Nuclear Regulatory Commission, January 1989.

1255. UCLA-ENG-76113, "Some Probabilistic Aspects of the Seismic Risk of Nuclear Reactors," University of California, Los Angeles, December 1976.

1256. SECY-89-180, "Generic Safety Issue 15, 'Radiation Effects on Reactor Vessel Supports,'" June 13, 1989. [8906190110]

1257. NUREG/CR-5210, "Technical Findings Document for Generic Issue 51: Improving the Reliability of Open-Cycle Service-Water Systems," U.S. Nuclear Regulatory Commission, August 1988.

1258. NUREG/CR-5234, "Value/Impact Analysis for Generic Issue 51: Improving the Reliability of Open-Cycle Service-Water Systems," U.S. Nuclear Regulatory Commission, February 1989.

1259. NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants, "Service Water System Problems Affecting Safety-Related Equipment (Generic Letter 89-13)," July 18, 1989. [8907180211]

1260. Memorandum for J. Taylor from E. Beckjord, "Closeout of GI-51, 'Improving the Reliability of Open-Cycle Service Water Systems,'" August 10, 1989. [9704100044]

1261. Federal Register Notice 54 FR 31268, "'Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants'; Issuance and Availability Revised SRP Sections 2.4.2 and 2.4.3," July 27, 1989.

1262. NRC Letter to All Licensees of Operating Reactors and Holders of Construction Permits, "Potential for Increased Roof Loads and Plant Area Flood Runoff Depth at Licensed Nuclear Power Plants Due to Recent Change in Probable Maximum Precipitation Criteria Developed by the National Weather Service (Generic Letter 89-22)," October 19, 1989. [8910180273]

1263. Memorandum for J. Taylor from E. Beckjord, "Close-out of Generic Safety Issue No. 103, 'Design for Probable Maximum Precipitation,'" November 28, 1989. [8912180025]

1264. Memorandum for V. Stello from S. Chilk, "Degree Operators: Advance Notice of Rulemaking," January 23, 1986. [8601280245]

1265. Federal Register Notice 54 FR 33639, "Education for Senior Reactor Operators and Shift Supervisors at Nuclear Power Plants; Policy Statement," August 15, 1989.

1266. Federal Register Notice 54 FR 33568, "Education and Experience Requirements for Senior Reactor Operators and Supervisors at Nuclear Power Plants; Withdrawal of Proposed Rulemaking," August 15, 1989.

1267. NUREG-1267, "Technical Resolution of Generic Safety Issue A-29," U.S. Nuclear Regulatory Commission, September 1989.

1268. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Safety Issue A-29, 'Nuclear Power Plant Design for Reduction of Vulnerability to Industrial Sabotage,'" October 6, 1989. [8910190129]

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

1270. Memorandum for B. Morris from B. Sheron, "Proposed Generic Issue on Leakage Through Electrical Isolators," June 23, 1987. [9704100047]

1271. Memorandum for T. Speis from R. Bernero, "Request for Prioritization of Potential Generic Issue Per Office Letter No. 40," August 4, 1985. [8508120299]

1272. Memorandum for R. Mattson from F. Rosa, "Combustion Engineering Standard Technical Specifications (NUREG-0212) - Proposed Revision 3 - Relay Testing," October 8, 1982. [8211030387]

1273. NUREG-0693, "Analysis of Ultimate Heat Sink Cooling Ponds," U.S. Nuclear Regulatory Commission, November 1980.

1274. NUREG-0733, "Analysis of Ultimate Heat-Sink Spray Ponds," U.S. Nuclear Regulatory Commission, August 1981.

1275. NUREG-0858, "Comparison Between Field Data and Ultimate Heat Sink Cooling Pond and Spray Pond Models," U.S. Nuclear Regulatory Commission, September 1982.

1276. Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, (Rev. 2) January 1976.

1277. NUREG/CR-4120, "Mathematical Modeling of Ultimate Heat Sink Cooling Ponds," U.S. Nuclear Regulatory Commission, March 1985.

1278. "Performance Model for Ultimate Heat Spray Ponds," Journal of Energy Engineering, Vol. 112, No. 2, August 1986.

1279. "Method for Analysis of Ultimate Heat Sink Cooling Tower Performance," University of Illinois at Urbana-Champaign, April 1986. [9704090166]

1280. Memorandum for C. Ader from K. Kniel, "Request for Prioritization of New Generic Safety Issue 'Loss of Essential Service Water in LWRs,'" May 2, 1990. [9704090120]

1281. Memorandum for W. Minners from F. Rowsome, "A New Generic Safety Issue: Accident Management," April 16, 1985. [8505080417]

1282. SECY-88-147, "Integration Plan for Closure of Severe Accident Issues," May 25, 1988. [8806030338]

1283. Memorandum for C. Ader from W. Minners, "GI 116, Accident Management," May 9, 1990. [9704090138]

1284. Memorandum for J. Olshinski from D. Eisenhut, "Control Rod Guide Tube Pin Failures and Peening Damage on Integrity of Steam Generator Tube to Tubesheet Welds and Tube Ends -- North Anna Power Station, Unit No. 1 (NA-1)," December 13, 1982. [8212270164]

1285. EPRI NP-5544, "Nuclear Unit Operating Experience: 1985-1986 Update," Electric Power Research Institute, December 1987.

1286. Memorandum for M. Virgilio from S. Newberry, "Proposed Research Programs to Support SICB Regulation Needs," April 26, 1990. [9005090104]

1287. Memorandum for F. Gillespie, et. al., from T. Speis, "CRGR Combined Packages for the Proposed Resolution of Generic Issue 70, 'Power Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,'" December 7, 1988. [9507280258]

1288. SECY-90-232, "Evaluation of the Need for Primary System High Capacity Manual Venting Capability on Combustion Engineering (CE) Plants Without PORVs (GI-84)," June 28, 1990. [9007020274]

1289. Letter to K. Carr from C. Michelson, "Generic Issue - 84, Combustion Engineering Plants Without Power Operated Relief Valves," June 12, 1990. [9006220172]

1290. NRC Letter to All Pressurized Water Reactor Licensees and Construction Permit Holders, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f) (Generic Letter 90-06)," June 25, 1990. [9006200120]

1291. NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,'" U.S. Nuclear Regulatory Commission, December 1989.

1292. Memorandum for J. Taylor from E. Beckjord, "Close-out of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,'" July 26, 1990. [9507280267]

1293. NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70," U.S. Nuclear Regulatory Commission, December 1989.

1294. Memorandum for F. Gillespie from E. Beckjord, "Resolutions of Generic Issue 70, 'Power Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,'" November 16, 1989. [8911290064]

1295. SECY-90-153, "Staff Conclusions Relative to the Classification of PORVs as Safety Grade," April 27, 1990. [9005030123]

1296. Information Notice No. 90-19, "Potential Loss of Effective Volume for Containment Recirculation Spray at PWR Facilities," U.S. Nuclear Regulatory Commission, March 14, 1990. [9003080213]

1297. IE Bulletin No. 83-01, "Failure of Reactor Trip Breakers (Westinghouse DB-50) to Open on Automatic Trip Signal," U.S. Nuclear Regulatory Commission, February 25, 1983. [8212060367]

1298. IE Bulletin No. 83-04, "Failure of the Undervoltage Trip Function of Reactor Trip Breakers," U.S. Nuclear Regulatory Commission, March 11, 1983. [8212060380]

1299. IE Bulletin No. 83-08, "Electrical Circuit Breakers With an Undervoltage Trip Feature in Use in Safety-Related Applications Other that the Reactor Trip System," U.S. Nuclear Regulatory Commission, December 28, 1983. [8312120090]

1300. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Safety Issue 75, 'Generic Implications of Salem ATWS - QA,'" May 18, 1990. [9507280290]

1301. Memorandum for W. Dircks from R. Minogue, "Preliminary Survey of Requirements and Guidance by Functional Areas for Operating Nuclear Power Plants, Dated February 1984," August 21, 1984. [8409110305]

1302. Memorandum for E. Beckjord from W. Minners, "Generic Issue 131, 'Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants,'" July 18, 1990. [9007240192]

1303. NRC Letter to All Power Reactor Licensees and Applicants, "Relaxation of Staff Position in Generic Letter 83-28, Item 2.2 Part 2, 'Vendor Interface for Safety-Related Components' (Generic Letter No. 90-03)," March 20, 1990. [9003140089]

1304. IE Bulletin No. 77-02, "Potential Failure Mechanism in Certain Westinghouse (W) AR Relays With Latch Attachments," U.S. Nuclear Regulatory Commission, September 12, 1977. [7909050215]

1305. IE Bulletin No. 79-09, "Failures of GE Type AK-2 Circuit Breaker in Safety Related Systems," U.S. Nuclear Regulatory Commission, April 17, 1979. [7905010083]

1306. IE Circular No. 81-12, "Inadequate Periodic Test Procedure of PWR Protection System," U.S. Nuclear Regulatory Commission, July 22, 1981. [8103300406]

1307. NUREG-1372, "Regulatory Analysis for the Resolution of Generic Issue C-8: 'Main Steam Isolation Valve Leakage and LCS Failure,'" U.S. Nuclear Regulatory Commission, June 1990.

1308. NUREG-1169, "Resolution of Generic Issue C-8," U.S. Nuclear Regulatory Commission, August 1986.

1309. NUREG/CR-5397, "Value-Impact Analysis of Regulatory Options for Resolution of Generic Issue C-8: MSIV Leakage and LCS Failure," U.S. Nuclear Regulatory Commission, May 1990.

1310. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Safety Issue C-8," March 15, 1990. [9507280304]

1311. IE Information Notice No. 82-09, "Cracking in Piping of Makeup Coolant Lines at B&W Plants," U.S. Nuclear Regulatory Commission, March 31, 1982. [8202040131]

1312. IE Information No. 82-30, "Loss of Thermal Sleeves in Reactor Coolant System Piping at Certain Westinghouse PWR Power Plants," U.S. Nuclear Regulatory Commission, July 26, 1982. [8204210403]

1313. SECY-82-186, "Status of Make-up Nozzle Cracking in Babcock & Wilcox (B&W) Plants," May 7, 1982. [8205280495]

1314. Letter to R. Gridley (General Electric Company) from D. Eisenhut, "Safety Evaluation for the General Electric Topical Report NEDE-218121-02, 'BWR Feedwater Nozzle/Sparger Final Report, Supplement 2,'" January 14, 1980. [8002070141]

1315. Memorandum for D. Sternberg from R. Clark, "Degradation of Thermal Sleeves - Trojan Nuclear Plant," August 11, 1982. [8208250478]

1316. Memorandum for T. Novak, et al., from J. Knight, "Evaluation of Thermal Sleeve Problems in Westinghouse Plants," October 28, 1983. [8311140192]

1317. Letter to V. Stello from W. Kerr, "ACRS Comments on Nuclear Power Plant Air Cooling Systems," October 15, 1987. [8710210001]

1318. NUREG/CR-4550, "Analysis of Core Damage Frequency from Internal Events," U.S. Nuclear Regulatory Commission, (Vol. 1, Rev. 1) January 1990, (Vol. 2) April 1989, (Vol. 3, Rev. 1) April 1990, (Vol. 4, Rev. 1) August 1989, (Vol. 5, Rev. 1) April 1990, (Vol. 6) April 1987, (Vol. 7, Rev. 1) May 1990.

1319. NRC Information Notice 92-18, "Potential for Loss of Remote Shutdown Capability During a Control Room Fire," U.S. Nuclear Regulatory Commission, February 28, 1992. [9202240025]

1320. SECY-89-170, "Fire Risk Scoping Study: Summary of Results and Proposed Staff Actions," June 7, 1989. [8906260024]

1321. NRC Information Notice 91-53, "Failure of Remote Shutdown System Instrumentation Because of Incorrectly Installed Components," U.S. Nuclear Regulatory Commission, September 4, 1991. [9108280089]

1322. IE Information Notice 87-12, "Potential Problems with Metal Clad Circuit Breakers, General Electric Type AKF-2-25," U.S. Nuclear Regulatory Commission, February 13, 1987. [8702110132]

1323. Memorandum for W. Minners, et al., from F. Rowsome, "Generic Issue 123, 'Deficiencies in the Regulations Suggested by the Davis-Besse Incident,'" November 21, 1985. [8512100189]

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

1325. Regulatory Guide 1.152, "Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, November 1985. [8511220286]

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

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

1328. AEOD/E804, "Reliability of Non-Safety Related Field Breakers During ATWS Events," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, July 26, 1988. [8905020208]

1329. Memorandum for T. King from K. Kniel, "Request for Prioritization of New Generic Safety Issue 'Reliability of Recirculation Pump Trip (RPT) During an ATWS,'" March 17, 1989. [9507280112]

1330. Memorandum for T. King from W. Minners, "Overpressurization of Containment Penetrations," March 16, 1989. [9507280122]

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

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

1333. NSAC-148, "Service Water Systems and Nuclear Plant Safety," Electric Power Research Institute, May 1990.

1334. NUREG/CR-2797, "Evaluation of Events Involving Service Water Systems in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, November 1982.

1335. Memorandum for W. Minners from F. Rowsome, "A Candidate Generic Safety Issue," December 11, 1984. [8501090105]

1336. Memorandum for B. Morris from L. Shao, "Resolution of Generic Issue 119.2," July 16, 1990. [9507280130]

1337. Memorandum for J. Taylor from E. Beckjord, "Proposed Resolution and Closeout of Generic Issue 135, 'Steam Generator and Steam Line Overfill Issues,'" March 29, 1991. [9507280149]

1338. RES Office Letter No. 7, "Procedures for Identification, Prioritization, Resolution, and Tracking of Generic Issues," February 16, 1996. [9608070117]

1339. Memorandum for All RES Employees from E. Beckjord, "Withdrawal of RES Office Letter No. 3, 'Procedure and Guidance for the Resolution of Generic Issues,'" June 2, 1994. [9704100042]

1340. Federal Register Notice 51 FR 12502, "10 CFR Part 50, Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures," April 11, 1986.

1341. Federal Register Notice 51 FR 26393, "10 CFR Part 50, Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures," July 23, 1986.

1342. Federal Register Notice 52 FR 41288, "10 CFR Part 50, Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures," October 27, 1987.

1343. Federal Register Notice 53 FR 1968, "Standard Review Plan Revision," January 25, 1988.

1344. Federal Register Notice 52 FR 23376, "Standard Review Plan Issuance," June 19, 1987.

1345. NRC Letter to All Operating Licensees, Construction Permit Holders and Applicants for Construction Permits, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements (Generic Letter 87-11)," June 19, 1987. [8706230486]

1346. Memorandum for Distribution from G. Arlotto, "Termination of Proposed Revision to SRP 3.9.3," October 2, 1986. [8811180136]

1347. Regulatory Guide 1.84, "Design and Fabrication Code Case Acceptability -- ASME III, Division 1," U.S. Nuclear Regulatory Commission, (Rev. 30) October 31, 1994. [9411040236]

1348. Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," U.S. Nuclear Regulatory Commission, October 1973. [7907100231]

1349. Memorandum for J. Roe from J. Wermiel, "Closure of Generic Issue No. 133, 'Update Policy on Nuclear Plant Staff Working Hours,'" July 10, 1991. [9107230263]

1350. NRC Information Notice No. 91-36, "Nuclear Plant Staff Working Hours," June 10, 1991. [9106040339]

1351. SECY-90-343, "Status of the Staff Program to Determine How the Lessons Learned from the Systematic Evaluation Program Have Been Factored into the Licensing Bases of Operating Plants," October 4, 1990. [9010150030]

1352. Memorandum for J. Knight, et al., from H. Thompson, "Action Plan for Resolving Failure of Tendon Anchorage at Farley 2 and for Determining Need for Immediate Licensing Action on Other Facilities," June 25, 1985. [8507030479]

1353. NUREG/CR-4712, "Regulatory Analysis of Regulatory Guide 1.35 (Revision 3, Draft 2) - In-Service Inspection of Ungrouted Tendons in Prestressed Concrete Containments," U.S. Nuclear Regulatory Commission, February 1987.

1354. NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," U.S. Nuclear Regulatory Commission, June 1991.

1355. NRC Letter to All Licensees of Operating Pressurized Water Nuclear Power Reactors and Applicants for Operating Licenses (Except for St. Lucie, Unit No. 1), "Natural Circulation Cooldown (Generic Letter No. 81-21)," May 5, 1981. [8105140267]

1356. NRC Letter to All Operating Pressurized Water Reactors (PWR's) (Generic Letter 80-53), June 11, 1980. [8007230099]

1357. ANSI/ANS 2.3, "Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites," American National Standards Institute, Inc., October 17, 1983.

1358. Memorandum for H. Thompson from G. Arlotto, "RES Input - Action Plan for Resolving Failure of Tendon Anchorage at Farley-2 and for Determining Need for Immediate Licensing Action on Other Facilities," July 31, 1985. [9312220342]

1359. Memorandum for T. Speis and E. Jordan from J. Knight, "Tendon Anchor Head Failure - Needed Licensing Action at Other Facilities," December 6, 1985. [8512240278]

1360. Regulatory Guide 1.35.1, "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments," U.S. Nuclear Regulatory Commission, July 1990. [9503290310]

1361. Memorandum for E. Jordan from E. Beckjord, "CRGR Review of: 1. Regulatory Guide 1.35, Rev. 3, 'Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments,' 2. Regulatory Guide 1.35.1, 'Determining Prestressing Forces for Inspection of Prestressed Concrete Containments,'" July 28, 1989. [8908100273]

1362. Memorandum for E. Beckjord from F. Gillespie, "Generic Concerns Arising from TMI-2 Cleanup," February 21, 1991. [9103010101]

1363. Memorandum for E. Beckjord from F. Gillespie, "Request for Generic Rulemaking Concerning Decommissioning Issues," January 7, 1992. [9201150209]

1364. Federal Register Notice 53 FR 24018, "10 CFR Parts 30, 40, 50, 51, 70, and 72, General Requirements for Decommissioning Nuclear Facilities," June 27, 1988.

1365. Memorandum for Z. Rosztoczy from S. Bajwa, "Generic Issue 148: Smoke Control and Manual Fire Fighting Effectiveness; Generic Issue 149: Adequacy of Fire Barriers," April 3, 1991. [9104080111]

1366. NUREG-1286, "Safety Evaluation Report Related to the Restart of Rancho Seco Nuclear Generating Station, Unit 1 Following the Event of December 26, 1985," U.S. Nuclear Regulatory Commission, October 1987.

1367. Memorandum for W. Russell from A. Thadani, "Task Action Plan for Resolution of Service Water System Problems," June 27, 1991. [9107120290]

1368. NRC Letter to Licensees and Applicants of the Following Pressurized-Water Reactor Nuclear Power Plants: 1. Braidwood Units 1 and 2; 2. Byron Units 1 and 2; 3. Catawba Units 1 and 2; 4. Comanche Peak Units 1 and 2; 5. Cook Units 1 and 2; 6. Diablo Canyon Units 1 and 2; 7. McGuire Units 1 and 2, "Request for Information Related to the Resolution of Generic Issue 130, 'Essential Service Water System Failures at Multi-Unit Sites,' Pursuant to 10 CFR 50.54(f) - Generic Letter 91-13," September 19, 1991. [9109160253]

1369. NUREG-1269, "Loss of Residual Heat Removal System, Diablo Canyon Unit 2, April 10, 1987," U.S. Nuclear Regulatory Commission, June 1987.

1370. SECY-91-283, "Evaluation of Shutdown and Low Power Risk Issues," September 9, 1991. [9109120134]

1371. NUREG/CR-4960, "Control Room Habitability Survey of Licensed Commercial Nuclear Power Generating Stations," U.S. Nuclear Regulatory Commission, October 1988.

1372. Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations," U.S. Nuclear Regulatory Commission, September 1974, (Rev. 1) November 1975. [7907200072]

1373. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," U.S. Nuclear Regulatory Commission, June 1974. [8001240567]

1374. Regulatory Guide 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," U. S. Nuclear Regulatory Commission, January 1975, (Rev. 1) February 1978 [8808230010].

1375. Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," U.S. Nuclear Regulatory Commission, February 1975, (Rev. 1) January 1977 [8001240569].

1376. Letter to D. Solberg (NRC) from T. Charlton (INEL), "Transmittal of Letter Report on Turbine Trip Failure Events - TRC-28-88," April 27, 1988. [9502070128]

1377. Memorandum for R. Baer from C. Hrabal, "Prioritization of GI-144, 'SCRAM Without a Turbine/Generator Trip,'" September 24, 1991. [9312220339]

1378. Memorandum for T. King from K. Kniel, "Request for Prioritization of New Generic Safety Issue 'SCRAM Without a Turbine/Generator Trip,'" March 22, 1988. [9312220315]

1379. NUREG/CR-5653, "Recriticality in a BWR Following a Core Damage Event," U.S. Nuclear Regulatory Commission, November 1990.

1380. Memorandum for W. Minners from B. Sheron, "Request for Prioritization of Potential Generic Issues," September 4, 1984. [8409170085]

1381. Memorandum for W. Minners from B. Sheron, "Update of Generic Issue Management Control System (GIMCS)," July 5, 1991. [9312220300]

1382. NUREG-1365, "Revised Severe Accident Research Program Plan," U.S. Nuclear Regulatory Commission, August 1989, (Rev. 1) December 1992.

1383. Memorandum for R. Baer from S. Diab, "Supporting Analyses for Prioritization of Issue 110, 'Equipment Protective Devices on Engineered Safety Features,'" April 16, 1992. [9312220226]

1384. Memorandum for W. Dircks for R. Fraley, "Bolt Failures in Nuclear Power Plants," October 20, 1981. [8201200698]

1385. NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants, "Generic Safety Issue 29, 'Bolting Degradation or Failure in Nuclear Power Plants,' (Generic Letter 91-17)," October 17, 1991. [9110150302]

1386. NRC Letter to All Licensees of Operating PWRs and Holders of Construction Permits for PWRs, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants (Generic Letter 88-05)," March 17, 1988. [8803220364]

1387. NRC Letter to All Licensees, Applicants and Holders of Operating Licenses Not Required to be Reviewed for Seismic Adequacy of Equipment Under the Provisions of USI A-46, 'Seismic Qualification of Equipment in Operating Plants,' "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46 (Generic Letter 87-03)," February 27, 1987. [8703060307]

1388. NRC Bulletin No. 89-02, "Stress Corrosion Cracking of High-Hardness Type 410 Stainless Steel Internal Preloaded Bolting in Anchor Darling Model S350W Swing Check Valves or Valves of Similar Design," U.S. Nuclear Regulatory Commission, July 19, 1989. [8907110441]

1389. NRC Compliance Bulletin No. 87-02, "Fastener Testing to Determine Conformance with Applicable Material Specifications," U.S. Nuclear Regulatory Commission, November 6, 1987 [8711050040], (Supplement 1) April 22, 1988 [8804180142], (Supplement 2) June 10, 1988 [8806090301].

1390. NRC Information Notice No. 89-22, "Questionable Certification of Fasteners," U.S. Nuclear Regulatory Commission, March 3, 1989. [8902270158]

1391. NRC Information Notice No. 89-56, "Questionable Certification of Material Supplied to the Defense Department by Nuclear Suppliers," U.S. Nuclear Regulatory Commission, July 20, 1889 [8907140274], (Supplement 1) November 22, 1989 [8911160058], (Supplement 2) July 19, 1991 [9107120259].

1392. NRC Information Notice No. 89-70, "Possible Indications of Misrepresented Vendor Products," U.S. Nuclear Regulatory Commission, October 11, 1989 [8910040381], (Supplement 1) April 26, 1990 [9004200525].

1393. IE Information Notice No. 86-25, "Traceability and Material Control of Material and Equipment, Particularly Fasteners," U.S. Nuclear Regulatory Commission, April 11, 1986. [8604090451]

1394. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Safety Issue 29, 'Bolting Degradation or Failure in Nuclear Power Plants,'" October 25, 1991. [9312220296]

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

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

1397. SECY-91-150, "Proposed Amendments to 10 CFR Part 21, 'Reporting of Defects and Noncompliance' and 10 CFR 50.55(e), 'Conditions of Construction Permits,'" May 22, 1991. [9106040262]

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

1399. NRC Letter to All Holders of Operating Licenses, "Resolution of Generic Issue A-30, 'Adequacy of Safety-Related DC Power Supplies,' Pursuant to 10 CFR 50.54(f) (Generic Letter 91-06)," April 29, 1991. [9104170256]

1400. NRC Letter to All Holders of Operating Licenses, "Resolution of Generic Issues 48, 'LCOs for Class 1E Vital Instrument Buses,' and 49, 'Interlocks and LCOs for Class 1E Tie Breakers' Pursuant to 10 CFR 50.54(f) (Generic Letter 91-11)," July 18, 1991. [9107160296]

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

1402. Memorandum for J. Taylor from E. Beckjord, "Resolution of GI-128, Electrical Power Reliability,'" September 12, 1991. [9312220229]

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

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

1405. SECY-82-1B, "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation," November 24, 1982. [8301120513]

1406. Memorandum for J. Taylor from E. Beckjord, "Technical Resolution of Generic Issue 87, 'Failure of HPCI Steam Line Without Isolation,'" December 9, 1991. [9312220344]

1407. NUREG/CR-4681, "Enclosure Environment Characterization Testing for the Base Line Validation of Computer Fire Simulation Codes," U.S. Nuclear Regulatory Commission, March 1987.

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

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

1410. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Issue 130, 'Essential Service Water System Failures at Multi-Unit Sites,'" September 23, 1991. [9312220347]

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

1412. Memorandum for J. Taylor, et al., form S. Chilk "SECY-91-132 - Evaluation of the Feasibility of Initiating a Consensus Process to Address Issues Related to the Below Regulatory Concern Policy," June 28, 1991. [9109060094]

1413. SECY-92-045, "Enhanced Participatory Rulemaking Process," February 7, 1992. [9202130092]

1414. Memorandum for K. Kniel from G. Lainas, "Proposed Generic Issue Deinerting Upon Discovery of Reactor Coolant System Leakage," August 1, 1986. [8608110015]

1415. Letter to D. Basdekas (NRC) from J. Lambright (SNL), "Generic Issue 148, 'Smoke Control and Manual Fire Fighting Effectiveness,'" March 4, 1992. [9502070165]

1416. Memorandum for B. Morris from W. Minners, "Prioritization of Proposed New Generic Issue," December 4, 1989. [9312220350]

1417. NUREG/CR-5856 "Identification and Evaluation of PWR In-Vessel Severe Accident Management Strategies," U.S. Nuclear Regulatory Commission, March 1992.

1418. Memorandum for T. Murley from E. Beckjord, "A New Generic Issue: Multiple Steam Generator Tube Leakage," June 16, 1992. [9212040356]

1419. Memorandum for C. Serpan from J. Muscara, "Steam Generator Tube Inspection, Integrity and Plugging Issues," March 16, 1992. [9212040327]

1420. Letter to J. Cross (Portland General Electric Company) from L. Kokajko (NRC), "Issuance of Amendment for Trojan Nuclear Plant (TAC No. M82287)," February 5, 1992. [9202130137]

1421. Letter to the NRC from J. Cross (Portland General Electric Company), "Request for Additional Information Regarding Trojan Steam Generator Tube Structural Integrity Report

and License Change Application (LCA) 219 Dated January 3, 1992 (TAC No. M82287)," January 16, 1992. [9201220023]

1422. NUREG/CR-0718, "Steam Generator Tube Integrity Program Phase I Report," U.S. Nuclear Regulatory Commission, September 1979.

1423. NUREG-1350, "Nuclear Regulatory Commission Information Digest," U.S. Nuclear Regulatory Commission, (Vol. 4) March 1992, (Vol. 7) March 1995.

1424. EGG-PE-6670, "Generic Cost Analysis for Steam Generator Repairs and Replacement," Idaho National Engineering Laboratory, August 1984.

1425. SECY-91-270, "Interim Guidance on Staff Implementation of the Commission's Safety Goal Policy," August 27, 1991. [9109030213]

1426. Memorandum for R. Emrit from G. Burdick, "Multiple Steam Generator Tube Leakage," October 30, 1992. [9502070227]

1427. SECY-92-292, "Advance Notice of Proposed Rulemaking on Severe Accident Plant Performance Criteria for Future LWRs," August 21, 1992. [9208250010]

1428. NUREG/CR-4470, "Survey and Evaluation of Vital Instrumentation and Control Power Supply Events," U.S. Nuclear Regulatory Commission, August 1986.

1429. NUREG-1455, "Transformer Failure and Common-Mode Loss of Instrument Power at Nine Mile Point Unit 2 on August 13, 1991," U.S. Nuclear Regulatory Commission, October 1991.

1430. Memorandum for T. Martin et al., from T. Murley, "Preliminary Results from Individual Plant Examinations (IPE)," April 22, 1991. [9105020194]

1431. NRC Letter to All Holders of Operating Licenses or Construction Permits for Pressurized Water Reactors (PWRs), "Resolution of Generic Issue 79, 'Unanalyzed Reactor Vessel (PWR) Thermal Stress During Natural Convection Cooldown' (Generic Letter 92-02)," March 6, 1992. [9203030209]

1432. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Issue 79, 'Unanalyzed Reactor Vessel (PWR) Thermal Stress During Natural Convection Cooldown,'" May 4, 1992. [9312220157]

1433. Memorandum for C. Heltemes from F. Gillespie, "Generic Issue 163, 'Multiple Steam Generator Tube Leakage,'" November 24, 1992. [9212040320]

1434. Memorandum for E. Beckjord from L. Shao, "Interim Plugging Criteria for Trojan Nuclear Plant," December 9, 1992. [9212140066]

1435. Memorandum for F. Gillespie from C. Heltemes, "GI-163, 'Multiple Steam Generator Tube Leakage,'" September 28, 1992. [9212040379]

1436. Federal Register Notice 51 FR 27817, "10 CFR Parts 50 and 73, Miscellaneous Amendments Concerning Physical Protection of Nuclear Power Plants," August 4, 1986.

1437. NRC Letter to All Power Reactor Licensees, "Implementation of 10 CFR 73.55 Miscellaneous Amendments and Search Requirements (Generic Letter 87-08)," May 11, 1987. [8705110372]

1438. Regulatory Guide 5.65, "Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls," U.S. Nuclear Regulatory Commission, September 1986. [8610030129]

1439. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Issue 151 'Reliability of ATWS Recirculation Pump Trip in BWRs,'" September 29, 1992. [9312220159]

1440. NUREG/CR-2336, "Steam Generator Tube Integrity Program," U.S. Nuclear Regulatory Commission, August 1988.

1441. Memorandum for T. Murley from E. Beckjord, "Interim Plugging Criteria for Trojan Nuclear Plant," January 5, 1993. [9301110331]

1442. Memorandum for T. Murley from E. Beckjord, "Interim Plugging Criteria for Trojan Nuclear Plant," January 15, 1993. [9301250251]

1443. SECY-90-160, "Proposed Rule on Nuclear Power Plant License Renewal," May 3, 1990. [9005080305]

1444. NUREG-1412, "Foundation for the Adequacy of the Licensing Bases," U.S. Nuclear Regulatory Commission, December 1991.

1445. NUREG/CR-6010, "History and Current Status of Generation 3 Thermal Sleeves in Westinghouse Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1992.

1446. Memorandum for J. Taylor from E. Beckjord, "Technical Resolution of Generic Issue 73, 'Detached Thermal Sleeves,'" September 2, 1992. [9210070203]

1447. Memorandum for D. Eisenhut, et al., from R. Vollmer, "Evaluation of Allegations Regarding Class 1 Piping Design Deficiencies (TAC #49242)," September 1, 1983. [8309210477]

1448. IE Information Notice No. 83-80, "Use of Specialized 'Stiff' Pipe Clamps," November 23, 1983. [8311010020]

1449. NUREG/CR-2405, "Subsystem Fragility - Seismic Safety Margins Research Program (Phase 1)," U.S. Nuclear Regulatory Commission, February 1982.

1450. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Issue 113, 'Dynamic Qualification and Testing of Large Bore Hydraulic Snubbers,'" August 27, 1992. [9312220197]

1451. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Issue (GI) 121, 'Hydrogen Control for PWR Dry Containments,'" March 24, 1992. [9312220194]

1452. Memorandum for W. Minners from F. Gillespie, "Prioritization of Generic Issue 78, 'Monitoring of Design Basis Transient Fatigue Limits for Reactor Coolant System,'" June 10, 1992. [9312220188]

1453. NRC Information Notice 92-06, "Reliability of ATWS Mitigation System and Other NRC Required Equipment Not Controlled by Plant Technical Specifications," January 15, 1992 [9201080305], (Supplement 1) July 1, 1993 [9306250303].

1454. Memorandum for W. Minners from B. Sheron, "Proposed Generic Issue 'RHR Pumps Inside Containment,'" August 23, 1985. [8508290373]

1455. NUREG/CR-5300, "Integrated Reliability and Risk Analysis System (IRRAS) Version 2.5," U.S. Nuclear Regulatory Commission, (Vol. 1) March 1991.

1456. NUREG/CR-5303, "System Analysis and Risk Assessment System (SARA) Version 4.0," U.S. Nuclear Regulatory Commission, (Vol. 1) February 1992, (Vol. 2) January 1992.

1457. Letter to C. Rourk (NRC) from N. Anderson (INEL), "Transmittal of Final Report, 'Analysis of Plant Specific Responses for the Resolution of Generic Issue A-30, Adequacy of Safety-Related DC Power Supplies,' (FIN D6025) NRA-20-92," July 9, 1992. [9502070242]

1458. SECY-87-297, "MARK I Containment Performance Program Plan," December 8, 1987. [8803080354]

1459. SECY-89-017, "MARK I Containment Performance Improvement Program," January 23, 1989. [8903090205]

1460. Memorandum for V. Stello from S. Chilk, "SECY-89-017 - MARK I Containment Performance Improvement Program," July 11, 1989. [8907270013]

1461. SECY-91-316, "Status of Severe Accident Research," October 7, 1991. [9110160271]

1462. Letter to D. Grace (BWR Owners Group) from A. Thadani (NRC), "Safety Evaluation of 'BWR Owners' Group - Emergency Procedure Guidelines, Revision 4,' NEDO-31331, March 1987," September 12, 1988. [8809190198]

1463. NRC Letter to All Holders of Operating Licenses for Nuclear Power Reactors With Mark I Containments, "Installation of a Hardened Wetwell Vent (Generic Letter No. 89-16)," September 1, 1989. [8909010375]

1464. NUREG/CR-5662, "Hydrogen Combustion, Control, and Value-Impact Analysis for PWR Dry Containments," U.S. Nuclear Regulatory Commission, June 1991.

1465. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1995.

1466. NUREG/CR-5460, "A Cause-Defense Approach to the Understanding and Analysis of Common Cause Failures," U.S. Nuclear Regulatory Commission, March 1990.

1467. Federal Register Notice 56 FR 31306, "10 CFR 50, RIN 3150-AD00, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," July 10, 1991.

1468. Memorandum for E. Beckjord from F. Gillespie, "Potential Generic Issue - Adequacy of Emergency and Essential Lighting - (RES Office Letter No. 1, Rev. 1)," September 14, 1990. [9009210192]

1469. NUREG/CR-4834, "Recovery Actions in PRA for the Risk Methods Integration and Evaluation Program (RMIEP)," U.S. Nuclear Regulatory Commission, (Vol. 1) June 1987.

1470. NUREG/CR-4674, "Precursors to Potential Severe Core Damage Accidents," U.S. Nuclear Regulatory Commission, (Vols. 15 and 16) September 1992.

1471. NRC Information Notice No. 90-69, "Adequacy of Emergency and Essential Lighting," U.S. Nuclear Regulatory Commission, October 31, 1990. [9010250054]

1472. NUREG-1272, "Office for Analysis and Evaluation of Operational Data 1991 Annual Report," U.S. Nuclear Regulatory Commission, (Vol. 6, No. 1) August 1992.

1473. Memorandum for J. Taylor from S. Chilk, "SECY-89-102 - Implementation of the Safety Goals," June 15, 1990. [9007090094]

1474. Letter to W. Conway (Arizona Public Service Company) from C. Trammell (NRC), "Review of Eddy-Current Inspections of Steam Generator Tubes - Palo Verde Nuclear Generating Station, Unit No. 2 (TAC No. M86178)," June 8, 1993. [9306100267]

1475. NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes," U.S. Nuclear Regulatory Commission, (Draft) June 1993.

1476. Memorandum for T. Murley from E. Beckjord, "Recommendations Regarding Revision of Standard Review Plan Sections Related to 'Stiff Pipe Clamps,'" August 12, 1992. [9312220199]

1477. Memorandum for T. Speis from F. Gillespie, "Consideration of New Generic Issue on `Support Flexibility of Equipment and Components,'" January 30, 1989. [8903010215]

1478. NUREG/CR-2999, "Final Report USNRC Anchor Bolt Study: Data Survey and Dynamic Testing," U.S. Nuclear Regulatory Commission, December 1982.

1479. SECY-93-108, "Revised Guidelines for Prioritization of Generic Safety Issues," April 28, 1993. [9308230261]

1480. EPRI NP-6154, "Proceedings: EPRI/NRC/TPC Workshop on Seismic Soil-Structure Interaction Analysis Techniques Using Data From Lotung, Taiwan," Electric Power Research Institute, (Vol. 1) March 1989, (Vol. 2) March 1989.

1481. Memorandum for E. Beckjord from T. Murley, "Potential New Generic Issues," September 25, 1991. [9110250132]

1482. Memorandum for T. Murley from E. Beckjord, "Prioritization of Generic Issue 161, 'Associated Circuits,'" March 12, 1993. [9312220201]

1483. Regulatory Guide 1.9, "Selection, Design, and Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants," March 1971, (Rev. 1) November 1978, (Rev. 2) December 1979 [8001220580], (Rev. 3) July 1993 [9308180045].

1484. Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, June 1993 [9306250035], (Rev. 1) January 1995 [9501300137].

1485. Federal Register Notice 58 FR 41813, "Regulatory Guide; Withdrawal," August 5, 1993.

1486. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Safety Issue B-56, 'Diesel Generator Reliability,'" June 29, 1993. [9312220205]

1487. NRC Letter to All Licensees of Operating Reactors, Applicants for An Operating License, and Holders of Construction Permits, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability (Generic Letter 84-15)," July 2, 1984. [8407020206]

1488. Memorandum for E. Jordan from T. Novak, "Engineering Evaluation Report - Pump Damage Due to Low Flow Cavitation (AEOD/E807)," October 18, 1988. [9312220206, 8811170140, 8810250191]

1489. NRC Bulletin No. 88-04, "Potential Safety-Related Pump Loss," U.S. Nuclear Regulatory Commission, May 5, 1988. [8804290177]

1490. NUREG/CR-5706, "Potential Safety-Related Pump Loss: An Assessment of Industry Data," U.S. Nuclear Regulatory Commission, June 1991.

1491. Memorandum for J. Norberg from R. Jones, "Review of Responses to Bulletin 88-04," July 22, 1991. [9108010062]

1492. NUREG/CR-5404, "Auxiliary Feedwater System Aging Study," U.S. Nuclear Regulatory Commission, (Vol. 1) March 1990, (Vol. 2) July 1993.

1493. Memorandum for V. Stello from S. Chilk, "Staff Requirements - Briefing on Status of Unresolved Safety/Generic Issues, 10:00 a.m., Wednesday, October 21, 1987, Commissioners' Conference Room, D. C. Office (Open to Public Attendance)," November 6, 1987. [8711100418]

1494. NUREG/CR-5604, "Assessment of ISLOCA Risk - Methodology and Application to a Babcock and Wilcox Nuclear Power Plant," U.S. Nuclear Regulatory Commission, (Vol. 1) April 1992, (Vol. 2) April 1992, (Vol. 3) April 1992.

1495. NUREG/CR-5744, "Assessment of ISLOCA Risk - Methodology and Application to a Westinghouse Four-Loop Ice Condenser Plant," U.S. Nuclear Regulatory Commission, April 1992.

1496. NUREG/CR-5745, "Assessment of ISLOCA Risk - Methodology and Application to a Combustion Engineering Plant," U.S. Nuclear Regulatory Commission, April 1992.

1497. NUREG/CR-5603, "Pressure-Dependent Fragilities for Piping Components," U.S. Nuclear Regulatory Commission, October 1990.

1498. NUREG/CR-5862, "Screening Methods for Developing Internal Pressure Capacities for Components in Systems Interfacing With Nuclear Power Plant Reactor Coolant Systems," U.S. Nuclear Regulatory Commission, May 1992.

1499. NUREG/CR-5928, "ISLOCA Research Program Final Report," U.S. Nuclear Regulatory Commission, July 1993.

1500. NUREG/CR-5102, "Interfacing System LOCA: Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, February 1989.

1501. NUREG-1463, "Regulatory Analysis for the Resolution of Generic Safety Issue 105: Interfacing System Loss-of-Coolant Accident in Light-Water Reactors," U.S. Nuclear Regulatory Commission, July 1993.

1502. NRC Information Notice 92-36, "Intersystem LOCA Outside Containment," May 7, 1992 [9205010045], (Supplement 1) February 22, 1994 [9402150320].

1503. Memorandum for F. Gillespie from W. Minners, "Proposed Resolution of Generic Issue 105, `Interfacing Systems LOCA in LWRs," April 2, 1993. [9312220208]

1504. Memorandum for J. Taylor from E. Beckjord, "Technical Resolution of Generic Issue 105 (GI-105) `Interfacing Systems Loss of Coolant Accident (ISLOCA) in LWRs," June 3, 1993. [9312220210]

1505. Memorandum for J. Taylor from S. Chilk, "SECY-93-108 - Revised Guidelines for Prioritization of Generic Safety Issues," July 23, 1993. [9308270094]

1506. Memorandum for W. Minners from L. Shao, "Closeout of GSI 119.4," July 17, 1992. [9312220212]

1507. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," U.S. Atomic Energy Commission, May 1973. [7907100182]

1508. Memorandum for J. Taylor from E. Beckjord, "Final Technical Resolution of Generic Safety Issue 120, 'On-Line Testability of Protection System,'" March 4, 1993. [9502070269]

1509. Letter to J. Taylor from P. Shewmon, "Prioritization of Generic Issue 152, 'Design Basis for Valves that Might Be Subjected to Significant Blowdown Loads,'" April 23, 1993. [9305060143]

1510. Letter to J. Wilkins (ACRS) from J. Taylor (EDO) June 8, 1993. [9306210081, 9305200137]

1511. Memorandum for J. Taylor from E. Beckjord, "Resolution of GI-142, 'Leakage Through Electrical Isolators,'" March 9, 1993. [9312220214]

1512. NUREG-1461, "Regulatory Analysis for the Resolution of Generic Issue 153: Loss of Essential Service Water in LWRs," U.S. Nuclear Regulatory Commission, August 1993.

1513. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Issue 153 (GI-153), 'Loss of Essential Service Water in LWRs,'" June 14, 1993. [9312220216]

1514. NUREG/CR-5910, "Loss of Essential Service Water in LWRs (GI-153)," U.S. Nuclear Regulatory Commission, August 1992.

1515. Memorandum for E. Beckjord from T. Murley, "Proposed New Generic Issue: 'Determination of Neutron Fluence to PWR Pressure Vessels,'" October 8, 1992. [9210190215]

1516. Memorandum for T. Murley from E. Beckjord, "Proposed New Generic Issue: Determination of Neutron Fluence to PWR Pressure Vessels," November 30, 1992. [9312220218]

1517. Memorandum for J. Sniezek from T. Murley and E. Beckjord, "Resolution of Fatigue and Environmental Qualification Issues Related to License Renewal," April 1, 1993. [9304270324]

1518. Memorandum for The Chairman, et al., from J. Taylor, "Environmental Qualification of Electric Equipment," May 27, 1993. [9308180153]

1519. SEASF-LR-92-022, "Supplemental Study of Generic Issue No. 153, 'Loss of Essential Service Water in LWRs,'" Science and Engineering Associates, Inc., (Rev. 1) January 1993. [9502070279]

1520. Memorandum for E. Beckjord from T. Murley, "Request to Prioritize a New Generic Issue for Spring-Actuated Safety and Relief Valve Reliability," October 8, 1992. [9312280153]

1521. NUREG/CR-3696, "Potential Human Factors Deficiencies in the Design of Local Control Stations and Operator Interfaces in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, April 1984.

1522. NUREG/CR-3217, "Near-Term Improvements for Nuclear Power Plant Control Room Annunciator Systems," U.S. Nuclear Regulatory Commission, April 1983.

1523. NUREG/CR-3987, "Computerized Annunciator Systems," U.S. Nuclear Regulatory Commission, June 1985.

1524. NUREG/CR-5572, "An Evaluation of the Effects of Local Control Station Design Configurations on Human Performance and Nuclear Power Plant Risk," U.S. Nuclear Regulatory Commission, September 1990.

1525. Memorandum for J. Taylor from E. Beckjord, "Termination of Work on Generic Safety Issue HF5.1 'Local Control Stations,'" June 29, 1993. [9312220224]

1526. Memorandum for J. Taylor from E. Beckjord, "Resolution of Human Factors Generic Issue 5.2, 'Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation,'" June 29, 1993. [9312220225]

1527. NUREG/CR-5186, "Value/Impact Analysis of Generic Issue 94, 'Additional Low Temperature Overpressure Protection for Light Water Reactors,'" U.S. Nuclear Regulatory Commission, November 1988.

1528. NRC Information Notice No. 90-22, "Unanticipated Equipment Actuations Following Restoration of Power to Rosemount Transmitter Trip Units," U.S. Nuclear Regulatory Commission, March 23, 1990. [9003190349]

1529. NUREG-1422, "Summary of Chernobyl Followup Research Activities," U.S. Nuclear Regulatory Commission, June 1992.

1530. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Issue 155.1, 'More Realistic Source Term Assumptions,'" March 13, 1995. [9511090074]

1531. NUREG-0940, "Enforcement Actions: Significant Actions Resolved," U.S. Nuclear Regulatory Commission, (Vol. 14, No. 2, Parts 1, 2, and 3) August 1995.

1532. Memorandum for C. Serpan from W. Minners, "Identification of New Generic Issue: Hydrogen Storage Facility Separation," December 16, 1993. [9312290134]

1533. Letter to the NRC from M. Tuckman (Duke Power Company), "Oconee Nuclear Station, Docket Nos. 50-269, 50-270, and 50-287, Generic Letter 88-20," November 30, 1990. [9012060005]

1534. EGG-SSRE-9747, "Improved Estimates of Separation Distances to Prevent Unacceptable Damage to Nuclear Power Plant Structures from Hydrogen Detonation for Gaseous Hydrogen Storage," Idaho National Engineering Laboratory, (Draft) November 1993. [9502070287]

1535. SCIE-EGG-103-89, "Draft Technical Evaluation Report on U.S. Commercial Power Reactor Hydrogen Tank Farms and Their Compliance With Separation Distance Safety Criteria," Scientech, Inc., March 1990. [9502070289]

1536. Memorandum for J. Taylor from E. Beckjord, "Resolution of GI-I.D.3, 'Safety System Status Monitoring,'" August 20, 1993. [9502070295]

1537. Memorandum for T. Murley from E. Beckjord, "Research Information Letter Number 171, 'Continuous On-Line Reactor Surveillance System,'" May 4, 1993. [9305100271]

1538. Memorandum for J. Taylor from E. Beckjord, "Closure of Generic Issue I.D.5(3), 'On-Line Automated Continuous Reactor Surveillance Systems,'" November 12, 1993. [9502070301]

1539. SECY-93-119, "TMI-2 Vessel Investigation Project," May 5, 1993. [9305100253]

1540. Memorandum for J. Taylor from E. Beckjord, "Closure of Generic Issue II.H.2, 'Obtain Data on Conditions Inside TMI-2 Containment,'" February 9, 1994. [9502070304]

1541. NUREG-1472, "Regulatory Analysis for the Resolution of Generic Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment," U.S. Nuclear Regulatory Commission, October 1993.

1542. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Safety Issue (GSI)-57, 'Effects of Fire Protection System Actuation on Safety-Related Equipment,'" September 30, 1993. [9502070315]

1543. NRC Information Notice 94-12, "Insights Gained from Resolving Generic Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment," U.S. Nuclear Regulatory Commission, February 9, 1994. [9402030011]

1544. NUREG/CR-5759, "Risk Analysis of Highly Combustible Gas Storage, Supply, and Distribution Systems in Pressurized Water Reactor Plants," U.S. Nuclear Regulatory Commission, June 1993.

1545. NUREG-1364, "Regulatory Analysis for the Resolution of Generic Safety Issue 106: Piping and the Use of Highly Combustible Gases in Vital Areas," U.S. Nuclear Regulatory Commission, June 1993.

1546. Memorandum for J. Taylor from E. Beckjord, "Proposed Resolution of GSI-106, 'Piping and the Use of Highly Combustible Gases in Vital Areas,'" November 3, 1993. [9502070320]

1547. Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Reactors, "Research Results on Generic Safety Issue 106, 'Piping and the Use of Highly Combustible Gases in Vital Areas,' (Generic Letter 93-06)," U.S. Nuclear Regulatory Commission, October 25, 1993. [9310200286]

1548. Memorandum for F. Gillespie from E. Beckjord, "Generic Letter for Implementation of Resolution of Generic Safety Issue 106, 'Piping and the Use of Highly Combustible Gases in Vital Areas,'" December 14, 1992. [9502070322]

1549. NUREG-1427, "Regulatory Analysis for the Resolution of Generic Issue 143: Availability of Chilled Water System and Room Cooling," U.S. Nuclear Regulatory Commission, December 1993.

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

1551. Memorandum for J. Taylor from E. Beckjord, "Resolution of Generic Issue 143 (GI-143), 'Availability of Chilled Water System and Room Cooling,'" September 30, 1993. [9502070325]

1552. NRC Information Notice No. 89-44, "Hydrogen Storage on the Roof of the Control Room," U.S. Nuclear Regulatory Commission, April 27, 1989. [8904260247]

1553. Memorandum for A. Gody from G. Holahan, "Prioritization of Proposed Generic Issue 162, 'Inadequate Technical Specifications for Shared Systems at Multi-plant Sites When One Unit Is Shut Down,'" March 20, 1993. [9304070074]

1554. Memorandum for J. Taylor from E. Beckjord, "Resolution of GI 67.5.1, 'Reassessment of SGTR Radiological Consequences,'" June 30, 1994. [9407130262]

1555. Memorandum for E. Beckjord from J. Murphy, "Staff Review Guidance for Generic Safety Issue (GSI) 147, 'Fire-Induced Alternate Shutdown/Control Room Panel Interactions,'" March 9, 1994. [9502070329]

1556. Memorandum for C. Rossi, et al., from T. Novak, "Safety and Safety/Relief Valve Reliability," April 24, 1992. [9205060277]

1557. NRC Information Notice No. 90-05, "Inter-System Discharge of Reactor Coolant," U.S. Nuclear Regulatory Commission, January 29, 1990. [9001230126]

1558. NRC Information Notice 92-64, "Nozzle Ring Settings on Low Pressure Water-Relief Valves," U.S. Nuclear Regulatory Commission, August 28, 1992. [9208240139]

1559. NRC Information Notice 92-61, "Loss of High Head Safety Injection," U.S. Nuclear Regulatory Commission, August 20, 1992 [9208180039], (Supplement 1) November 6, 1992 [9211020211].

1560. NUREG/CR-6001, "Aging Assessment of BWR Standby Liquid Control Systems," U.S. Nuclear Regulatory Commission, August 1992.

1561. NRC Information Notice No. 90-18, "Potential Problems With Crosby Safety Relief Valves Used on Diesel Generator Air Start Receiver Tanks," U.S. Nuclear Regulatory Commission, March 9, 1990. [9003050043]

1562. Memorandum for J. Taylor from E. Beckjord, "Resolution of Human Factors Generic Issue 4.4, 'Guidelines for Upgrading Other Procedures,'" July 29, 1993. [9502070331]

1563. NUREG/CR-5382, "Screening of Generic Safety Issues for License Renewal Considerations," U.S. Nuclear Regulatory Commission, December 1991.

1564. Memorandum for W. Russell from E. Beckjord, "License Renewal Implications of Generic Safety Issues (GSIs) Prioritized and/or Resolved Between October 1990 and March 1994," May 5, 1994. [9406170365]

1565. Memorandum for T. Murley from W. Russell and J. Partlow, "Closeout of TMI Action Plan Items III.A.1.2 and III.A.2.2 (Multi-Plant Actions F-63, F-64, F-65, and F-68)," October 2, 1990. [9010160111]

1566. SECY-80-275, "Final Rulemaking on Emergency Preparedness," June 3, 1980. [8007090015]

1567. NUREG/CP-0011, "Proceedings to Workshops Held on Proposed Rulemaking on Emergency Planning for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, April 1980.

1568. Federal Register Notice 46 FR 11666, "10 CFR Parts 30, 40, 50, 70 and 72 Decommissioning Criteria for Nuclear Facilities; Notice of Availability of Draft Generic Environment Impact Statement," February 10, 1981.

1569. SECY-87-309, "Final Rule Amendments to 10 CFR Parts 30, 40, 50, 51, 70, and 72: General Requirements for Decommissioning Nuclear Facilities," December 17, 1987. [8801130361]

1570. SECY-88-94, "Final Rule Amendments to 10 CFR Parts 30, 40, 50, 51, 70, and 72: General Requirements for Decommissioning Nuclear Facilities (SECY-87-309)," April 5, 1988. [8804120065]

1571. SECY-94-179, "Notice of Proposed Rulemaking on Decommissioning of Nuclear Power Reactors," July 7, 1994. [9407180100]

1572. Memorandum for J. Taylor and K. Cyr from J. Hoyle, "SECY-94-179 - Notice of Propose Rulemaking on Decommissioning of Nuclear Power Reactors and COMKR-94-002 - Decommissioning of Nuclear Power Reactors and Comments on SECY-94-179," October 5, 1994. [9410270085]

1573. NRC Letter to All Pressurized Water Reactor Licensees, "Inadvertent Boron Dilution Events (Generic Letter 85-05)," January 31, 1985. [8502010366]

1574. Memorandum for J. Taylor from R. Bernero, "Resolution of Issue Numbert B-64, 'Decommissioning of Reactors,' of the Generic Issue Management Control System," September 26, 1994. [9410110028]

1575. Memorandum for C. Serpan and C. Ader from J. Greeves, "Reference to the U.S. Nuclear Regulatory Commission Dam Safety Program in NUREG-0933," August 12, 1994. [9409060217]

1576. "Approaches to Upgrading Procedures in Nuclear Power Plants," Pacific Northwest Laboratory, August 1994. [9507280167]

1577. NUREG/CR-6146, "Local Control Stations: Human Engineering Issues and Insights," U.S. Nuclear Regulatory Commission, September 1994.

1578. NUREG/CR-6105, "Human Factors Engineering Guidance for the Review of Advanced Alarm Systems," U.S. Nuclear Regulatory Commission, September 1994.

1579. Letter to L. Zech from F. Remick, "Resolution of Generic Issue 43, 'Air Systems Reliability,'" Januuary 19, 1989. [8901260092]

1580. Memorandum for J. Larkins from E. Beckjord, "Evaluation of Potential Safety Issues from the Multiple System Responses Program," June 3, 1994. [9406230143]

1581. Memorandum for T. Speis from A. Thadani, "Review of NUREG/CR-5420," April 30, 1995. [9505230058]

1582. NUREG/CR-5455, "Development of NRC's Human Performance Investigation Process (HPIP)," U.S. Nuclear Regulatory Commission, (Vol. 1) October 1993, (Vol. 2) October 1993, (Vol. 3) October 1993.

1583. NUREG-0711, "Human Factors Engineering Program Review Model," U.S. Nuclear Regulatory Commission, July 1994.

1584. NUREG/CR-5908, "Advanced Human-System Interface Design Review Guideline," U.S. Nuclear Regulatory Commission, (Vol. 1) July 1994, (Vol. 2) July 1994.

1585. Memorandum for L. Shao from B. Sheron, "Proposed Generic Issue on Safety Systems' Response to the Sequential Occurrence of LOCA and Loss of Offsite Power Events," February 17, 1995. [9502270179]

1586. NRC Information Notice 93-17, "Safety Systems Response to Loss of Coolant and Loss of Offsite Power," March 8, 1993 [9303020536], (Rev. 1) March 25, 1994 [9403220236].

1587. NUREG-1335, "Individual Plant Examination: Submittal Guidance," U.S. Nuclear Regulatory Commission, August 1989.

1588. NUREG/CR-5580, "Evaluation of Generic Issue 57: Effects of Fire Protection System Actuation on Safety-Related Equipment," (Vol. 1) December 1992, (Vol. 2) December 1992, (Vol. 3) December 1992, (Vol. 4) December 1992, (Vol. 5) December 1992.

1589. NUREG/CR-5720, "Motor-Operated Valve Research Update," U.S. Nuclear Regulatory Commission, June 1992.

1590. EGG-REQ-7297, "Summary of Valve Assemblies in High Energy BWR Systems Outside of Containment -- Interim Report," EG&G Idaho, Inc., June 1986. [9511090106]

1591. Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions," U.S. Nuclear Regulatory Commission, February 1972. [7907100108]

1592. Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems," U.S. Nuclear Regulatory Commission, June 1976, (Rev. 1) November 1977, (Rev. 2) June 1978 [7907110110], (Rev. 3) April 1995 [9505030214].

1593. NUREG-1453, "Regulatory Analysis for the Resolution of Generic Issue 142: Leakage Through Electrical Isolators in Instrumentation Circuits," U.S. Nuclear Regulatory Commission, September 1993.

1594. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, May 1974. [7907100299]

1595. Memorandum for E. Beckjord from T. Murley, "User Need for Assistance on High Burnup Fuels," October 4, 1993. [9310270186]

1596. Memorandum for W. Russell from E. Beckjord, "Fuel Damage Criteria for Reactivity Transients," April 29, 1994. [9511090065]

1597. NRC Information Notice 94-64, "Reactivity Insertion Transient and Accident Limits for High Burnup Fuel," August 31, 1994 [9408250234], (Supplement 1) April 6, 1995 [9503310049].

1598. Memorandum to the Chairman, et al., from J. Taylor, "Reactivity Transients and High-Burnup Fuel," September 13, 1994. [9409300142]

1599. Memorandum to the Chairman, et al., from J. Taylor, "Reactivity Transients and Fuel Damage Criteria for High Burnup Fuel," November 9, 1994. [9511090217]

1600. NUREG/CP-0139, "Transactions of the Twenty-Second Water Reactor Safety Information Meeting," U.S. Nuclear Regulatory Commission, October 1994.

1601. Memorandum to C. Serpan from A. Chaffee, "Nuclear Reactor Regulation (NRR) Input Into Research NUREG-0933 (WITS Item 9400213)," February 13, 1996. [9602260124]

1602. NUREG-1352, "Action Plans for Motor-Operated Valves and Check Valves," U.S. Nuclear Regulatory Commission, June 1990.

1603. Memorandum to A. Thadani from J. Strosnider, "Plan for Addressing Generic Reactor Pressure Vessel Issues," August 9, 1995. [9508150078]

1604. NUREG-1511, "Reactor Pressure Vessel Status Report," U.S. Nuclear Regulatory Commission, December 1994.

1605. Memorandum to A. Thadani from J. Strosnider, "Assessment of Impact of Increased Variability in Chemistry on the RT_{PTS} Value of PWR Reactor Vessels," May 5, 1995 [9505100187]

1606. SECY-91-161, "Schedules for the Advanced Reactor Reviews and Regulatory Guidance Revisions," May 31, 1991. [9106050174]

1607. Memorandum for W. Russell from F. Gillespie, "Action Plan for the Development of Draft SRP Revisions in the SRP-UDP," May 17, 1994. [9406280148, 9405270273]

1608. Memorandum to J. Taylor from C. Paperiello and W. Russell, "Dry Cask Storage Action Plan," July 28, 1995. [9508250186]

1609. Memorandum to J. Taylor from W. Russell and R. Bernero, "Realignment of Reactor Decommissioning Program," March 15, 1995. [9508250180]

1610. NUREG-1474, "Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Station from August 20 - 30, 1992," U.S. Nuclear Regulatory Commission and the Institute of Nuclear Power Operations, March 1993. [9307060041]

1611. Memorandum for J. Taylor from T. Murley, "Office of Nuclear Reactor Regulation (NRR) Plan for Generic Follow-on Actions - 'Report on the Effect of Hurricane Andrew on the Turkey Point Nuclear Generating Station from August 20 - 30, 1992,'" July 22, 1993. [9308160297]

1612. NRC Information Notice 93-53, "Effect of Hurricane Andrew on Turkey Point Nuclear Generating Station and Lessons Learned," July 20, 1993 [9307140056], (Supplement 1) April 29, 1994 [9404280023].

1613. NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," U.S. Nuclear Regulatory Commission, October 1995.

1614. NRC Information Notice 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR," September 30, 1992. [9209290014]

1615. NRC Bulletin No. 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," May 11, 1993. [9305110015]

1616. NRC Information Notice 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," April 26, 1993 [9304260085], (Supplement 1) May 6, 1993 [9305050002].

1617. NRC Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," October 17, 1995. [9510040059]

1618. NRC Information Notice No. 88-28, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," May 19, 1988. [8805130108]

1619. NRC Information Notice 92-85, "Potential Failures of Emergency Core Cooling Systems Caused by Foreign Material Blockage," December 23, 1992. [9212170209]

1620. NRC Information Notice 94-57, "Debris in Containment and the Residual Heat Removal System," August 12, 1994. [9408080111]

1621. NRC Information Notice 95-06, "Potential Blockage of Safety-Related Strainers by Material Brought Inside Containment," January 25, 1995. [9501190091]

1622. NRC Information Notice 95-47, "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," October 4, 1995 [9510030107], (Rev. 1) November 30, 1995 [9511270084].

1623. Memorandum to A. Thadani from G. Holahan, "Task Action Plan for Spent Fuel Storage Pool Safety," October 13, 1994. [9410190155]

1624. NRC Information Notice 94-38, "Results of a Special NRC Inspection at Dresden Nuclear Power Station Unit 1 Following a Rupture of Service Water Inside Containment," May 27, 1994. [9405240025]

1625. NRC Bulletin 94-01, "Potential Fuel Pool Draindown Caused by Inadequate Maintenance Practices at Dresden Unit 1," April 14, 1994. [9404120041]

1626. Memorandum for A. Thadani from G. Holahan, "Revision to Report on the Re-Assessment of the NRC Fire Protection Program," February 27, 1993. [9504190319]

1627. SECY-93-143, "NRC Staff Actions to Address the Recommendations in the Report on the Reassessment of the NRC Fire Protection Program," May 21, 1993. [9306030231]

1628. SECY-95-034, "Status of Recommendations Resulting from the Reassessment of the NRC Fire Protection Program," February 13, 1995. [9503060019]

1629. Memorandum for Chairman Jackson, et al., from J. Taylor, "Semiannual Report on the Status of the Thermo-Lag Action Plan and Fire Protection Task Action Plan," September 20, 1995. [9509250375]

1630. SECY-94-219, "Proposed Agency-Wide Implementation Plan for Probabilistic Risk Assessment (PRA)," August 19, 1994. [9409090234]

1631. SECY-95-079, "Status Update of the Agency-Wide Implementation Plan for Probabilistic Risk Assessment," March 30, 1995. [9504100180]

1632. SECY-95-126, "Final Policy Statement on the Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," May 18, 1995. [9506020152]

1633. SECY-95-280, "Framework for Applying Probabilistic Risk Analysis in Reactor Regulation," November 27, 1995. [9512180168, 9512040133]

1634. Memorandum to Chairman Jackson from J. Taylor, "Improvements Associated With Managing the Utilization of Probabilistic Risk Assessment (PRA) and Digital Instrumentation and Control Technology," January 3, 1996. [9601180203]

1635. Memorandum for J. Taylor from T. Murley, et al., "Agency Directions for Current and Future Uses of Probabilistic Risk Assessment (PRA)," November 2, 1993. [9311100145]

1636. Memorandum to A. Thadani from G. Holahan and R. Spessard, "Action Plan to Monitor, Review, and Improve Fuel and Core Components Operating Performance," October 7, 1994. [9411040040]

1637. NRC Letter to All Operating Reactor Licensees, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions (Generic Letter No. 83-11)," February 4, 1983. [8302080304]

1638. NRC Letter to All Power Reactor Licensees and Applicants, "Removal of Cycle-Specific Parameter Limits from Technical Specifications (Generic Letter 88-16)," October 4, 1988. [8810050058, 8810140007]

1639. NRC Information Notice No. 91-47, "Failure of Thermo-Lag Fire Barrier Material to Pass Fire Endurance Test," August 6, 1991. [9108020180]

1640. NRC Information Notice 91-79, "Deficiencies in the Procedures for Installing Thermo-Lag Fire Barrier Materials," December 6, 1991 [9112020091], (Supplement 1) August 4, 1994 [9408030006].

1641. NRC Information Notice 92-55, "Current Fire Endurance Test Results for Thermo-Lag Fire Barrier Material," July 27, 1992. [9207270345]

1642. NRC Information Notice 92-82, "Results of Thermo-Lag 330-1 Combustibility Testing," December 15, 1992. [9212090211]

1643. NRC Information Notice 94-22, "Fire Endurance and Ampacity Derating Test Results for 3-Hour Fire-Rated Thermo-Lag 330-1 Fire Barriers," March 16, 1994. [9403150511]

1644. NRC Information Notice 94-34, "Thermo-Lag 330-660 Flexi-Blanket Ampacity Derating Concerns," May 13, 1994. [9405090108]

1645. NRC Information Notice 94-86, "Legal Actions Against Thermal Science, Inc., Manufacturer of Thermo-Lag," December 22, 1994. [9412160132]

1646. NRC Information Notice 95-27, "NRC Review of Nuclear Energy Institute, 'Thermo-Lag 330-1 Combustibility Evaluation Methodology Plant Screening Guide,'" May 31, 1995. [9505240424]

1647. NRC Information Notice 95-32, "Thermo-Lag 330-1 Flame Spread Test Results," August 10, 1995. [9508040074]

1648. NRC Information Notice 95-49, "Seismic Adequacy of Thermo-Lag Panels," October 27, 1995. [9510240388]

1649. NRC Information Notice 95-03, "Loss of Reactor Coolant Inventory and Potential Loss of Emergency Mitigation Functions While in a Shutdown Condition," January 18, 1995 [9501110412], (Supplement 1) March 25, 1996 [9602050208].

1650. NRC Letter to All Power Reactor Licensees and Applicants for Power Reactor Licenses, "Policy Statement on Engineering Expertise on Shift (Generic Letter 86-04)," February 13, 1986. [8602240459]

1651. NRC Information Notice 91-77, "Shift Staffing at Nuclear Power Plants," November 26, 1991. [9111200123]

1652. NRC Information Notice 93-44, "Operational Challenges During a Dual-Unit Transient," June 15, 1993. [9306080170]

1653. NRC Information Notice 93-81, "Implementation of Engineering Expertise on Shift," October 12, 1993. [9310060239]

1654. NRC Information Notice 95-48, "Results of Shift Staffing Study," October 10, 1995. [9510040181]

1655. Letter to B. Boger (NRC) from R. Whitesel (NUMARC), December 29, 1992. [9301080124]

1656. SECY-93-184, "Shift Staffing at Nuclear Power Plants," June 29, 1993. [9307080273]

1657. SECY-93-193, "Policy on Shift Technical Advisor Position at Nuclear Power Plants," July 13, 1993. [9307200065]

1658. NRC Bulletin No. 90-01, "Loss of Fill-Oil in Transmitters Manufactured by Rosemount," March 9, 1990 [9003050148], (Supplement 1) December 22, 1992 [9212170002].

1659. Memorandum for R. Zimmerman, et al., from J. Sniezek, "Review of Rosemount Transmitter Issues," May 21, 1993. [9308090287, 9310010241]

1660. SECY-95-078, "Staff Actions to Address Recommendations Resulting from Recent Evaluations of the Notice of Enforcement Discretion (NOED) Policy and Process," March 29, 1995. [9504100173]

1661. NUREG-1600, "General Statement of Policy and Procedures for NRC Enforcement Actions," U.S. Nuclear Regulatory Commission, July 1995.

1662. Memorandum for E. Jordan, et al., from J. Taylor, "Unauthorized Forced Entry Into the Protected Area at Three Mile Island Unit 1 on February 7, 1993 (NUREG-1485)," June 18, 1993. [9308110317]

1663. NRC Letter to All Holders of Operating Licenses or Construction or Construction Permits for Nuclear Power Reactors, Except for Big Rock Point and Facilities Permanently or Indefinitely Shut Down, "Emergency Response Data System Test Program (Generic Letter 93-01)," March 3, 1993. [9302240242]

1664. NRC Information Notice 93-94, "Unauthorized Forced Entry Into the Protected Area at Three Mile Island Unit 1 on February 7, 1993," December 9, 1993. [9312030104]

1665. NUREG-1485, "Unauthorized Forced Entry Into the Protected Area at Three Mile Island Unit 1 on February 7, 1993," U.S. Nuclear Regulatory Commission, April 1993.

1666. NUREG/CR-6432, "Estimated Net Value and Uncertainty for Automating ECCS Switchover at PWRs," U.S. Nuclear Regulatory Commission, February 1996.

1667. Memorandum for J. Taylor from D. Morrison, "Technical Resolution of Generic Issue 24 (GI-24), 'Automatic ECCS Switch to Recirculation,'" October 31, 1995. [9511140037]

1668. NUREG/CR-5904, "Functional Issues and Environmental Qualification of Digital Protection Systems of Advanced Light-Water Nuclear Reactors," U.S. Nuclear Regulatory Commission, April 1994.

1669. NUREG/CR-5941, "Technical Basis for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related I&C Systems," U.S. Nuclear Regulatory Commission, April 1994.

1670. NRC Generic Letter 94-03, "Intergrannular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors," July 25, 1994. [9407210200]

1671. Memorandum for A. Thadani from B. Sheron, "Staff Action Plan for the Resolution of Issues Associated With Boiling Water Reactor Internals Cracking," April 26, 1995. [9505220070]

1672. NRC Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Plants (Except Yankee Atomic Electric Company, Licensee for the Yankee Nuclear Power Station), "Reactor Vessel Structural Integrity, 10 CFR 50.54(f) (Generic Letter 92-01)," February 28, 1992, (Rev. 1) March 6, 1992 [9203060147], (Rev. 1, Supplement 1) May 19, 1995 [9505090312].

1673. Memorandum for E. Beckjord from W. Russell, "NRR User Need Request for Support of Resolving Problem of Stress Corrosion Cracking of Reactor Vessel Internal Components," December 2, 1994. [9505090299]

1674. Memorandum for D. Morrison from W. Russell, "Request for Research on Reactor Pressure Vessel Integrity," August 11, 1995. [9508220323]

1675. Memorandum to L. Shao from M. Mayfield, "Summary, NRC/NEI Workshop on Nuclear RPV Integrity," September 6, 1995. [9509200141]

1676. NRC Administrative Letter 95-03, "Availability of Reactor Vessel Integrity Database," August 4, 1995. [9508010148]

1677. AEOD/S95-01, "Reactor Coolant System Blowdown at Wolf Creek on September 17, 1994," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, March 1995. [9503310036]

1678. Memorandum to A. Thadani from R. Jones, "Proposed Action Plan for the 'Wolf Creek Draindown Event,'" September 8, 1995. [9509140225]

1679. Memorandum for the Chairman from J. Taylor, "Commercial Contract for Technical Assistance to Support the Standard Review Plan Update and Development Program," November 18, 1991.

1680. Memorandum for J. Taylor from I. Selin, "Commercial Contract for Technical Assistance to Support the Standard Review Plan Update and Development Program," December 13, 1991.

1681. Memorandum for J. Taylor from T. Murley, "Planned Actions to Address the Issues from the Office of Inspector General's Report on the NRC Staff's Review and Acceptance of Thermo-Lag 330-1 Fire Barrier Material," August 21, 1992. [9209250288]

1682. NRC Bulletin No. 92-01, "Failure of Thermo-Lag 330 Fire Barrier System to Maintain Cabling in Wide Cable Trays and Small Conduits Free from Fire Damage," June 24, 1992. [9206240122]

1683. Memorandum for W. Russell from T. Murley, "Final Report - Special Review Team for the Review of Thermo-Lag Fire Barrier Performance," April 21, 1992. [9205120277]

1684. Mmemorandum for E. Beckjord from T. Murley, "Request for Prioritization of Potential Generic Safety Issue - BWR MSIV Common Mode Failure Due to Loss of Accumulator Pressure," May 25, 1993. [9308160285]

1685. Memorandum for T. Murley from E. Beckjord, "Request for Prioritization of Potential Generic Safety Issue - BWR MSIV Common Mode Failure Due to Loss of Accumulator Pressure," June 29, 1993. [9509050193]

1686. SEA No. 95-3101-01-A:1, "Technical Information for Prioritization of Generic Safety Issues," Science and Engineering Associates, Inc., June 1996. [9704090123]

1687. SEA No. 97-3701-010-A:1, "Issue 107, Main Transformer Failures," Science and Engineering Associates, Inc., March 28, 1997. [9704090149]

1688. NUREG/CR-5595, "FORECAST: Regulatory Effects Cost Analysis Software Manual," (Rev. 1) July 1996.

1689. Memorandum to J. Taylor from J. Hoyle, "COMSECY-95-033 - Proposed Dollar per Person-Rem Conversion Factor; Response to SRM Concerning Issuance of Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission and SRM Concerning the Need for a Backfit Rule for Materials Licensees (RES-950225) (WITS-9100294)," September 18, 1995. [9803260148]

1690. Memorandum for All RES Employees from D. Morrison, "RES Office Letter No. 3C -- Procedures for Obtaining Regulatory Impact Analysis Review and Support," February 23, 1996. [9803260238]

1691. Memorandum to D. Morrison from W. Russell, "Third Supplemental User Need Request Regarding Potential for Loss of Emergency Core Cooling in a Boiling Water Reactor due to Clogging of the Suction Strainers by Loss-of-Coolant Accident Generated Debris," December 7, 1995. [9512140237]

1692. Memorandum to L. Shao from M. Marshall, "Expansion of Work Being Performed Under GSI-191, `Assessment of Debris Accumulation on Pressurized Water Reactors Sump Performance,'" May 14, 1997. [9706060061]

1693. Memorandum to Chairman Jackson, et al., from J. Taylor, "Report on Survey of Refueling Practices," May 21, 1996. [9606030213]

1694. Memorandum to Chairman Jackson, et al., from J. Taylor, "Resolution of Spent Fuel Storage Pool Action Plan Issues," July 26, 1996. [9611180017]

1695. SECY-97-168, "Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation," July 30, 1997. [9708150168]

1696. Memorandum to L. Callan from J. Hoyle, "Staff Requirements - SECY-97-168 - Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation," December 11, 1997. [9712180222]

1697. Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," U.S. Nuclear Regulatory Commission, (Rev. 1) December 1975 [7907100079], (Draft Rev. 2) December 1981 [9803260142].

1698. Memorandum to E. Beckjord from E. Jordan, "Periodic Review of Low Priority Generic Safety Issues," April 7, 1995. [9701230176]

1699. Memorandum to D. Morrison from W. Russell, "Periodic Review of Low-Priority Generic Issues," April 11, 1996. [9604240169]

1700. Memorandum to L. Shao from D. Cool, "Submittal of Generic Safety Issues," April 12, 1996. [9605170029]

1701. Memoranadum to F. Coffman from J. Piccone, "Status of NMSS Generic Safety Issues," December 15, 1997. [9712180068]

1702. NRC Information Notice 96-21, "Safety Concerns Related to the Design of the Door Interlock Circuit on Nucletron High-Dose Rate and Pulsed Dose Rate Remote Afterloading Brachytherapy Devices," April 10, 1996. [9604040106]

1703. NRC Information Notice 96-51, "Residual Contamination Remaining in Krypton-85 Handling System After Venting," September 11,1996. [9609050281]

1704. NRC Information Notice 96-54, "Vulnerability of Stainless Steel to Corrosion When Sensitized," October 17, 1996. [9610100212]

1705. Memorandum to L. Shao from D. Cool, "Submittal of Generic Safety Issue," June 18, 1997. [9706240185]

1706. NRC Bulletin 97-01, "Potential for Erroneous Calibration, Dose Rate, or Radiation Exposure Measurements With Certain Victoreen Model 530 and 530SI Electrometer/ Dosemeters," April 30, 1997. [9704300128]

1707. Memorandum to R. Bangart, et al., from D. Cool, "Closeout Report for Bulletin 97-01, Potential for Erroneous Measurements With Certain Victoreen Electrometers," September 8, 1997. [9709170137]

1708. Memorandum to L. Shao from D. Cool, "Submittal of Generic Safety Issue," August 5, 1997. [9708130432]

1709. Memorandum to J. Craig from F. Combs, "Submittal of New Generic Issues for Tracking in the Generic Issues Management and Control System (GIMCS)," June 4, 1998. [9806090180]

1710. NRC Information Notice 96-70, "Year 2000 Effect on Computer System Software," December 24, 1996. [9612200319]

1711. NRC Information Notice 97-61, "U.S. Department of Health and Human Services Letter to Medical Device Manufacturers on the Year 2000 Problem," August 6, 1997. [9707310130]

1712. NRC Information Notice 97-91, "Recent Failures of Control Cables Used on Amersham Model 660 Posilock Radiography Systems," December 31, 1997 [9712310254], (Supplement 1) August 10, 1998 [9808050063].

1713. NRC Information Notice 96-52, "Cracked Insertion Rods on Troxler Model 3400 Series Portable Moisture Density Gauges," September 26, 1996. [9609200181]

1714. NRC Bulletin 97-02, "Puncture Testing of Shipping Packages Under 10 CFR Part 71," September 23, 1997. [9709180179]

1715. Memorandum to D. Morrison from T. Gwynn, "Periodic Review of Low-Priority Generic Safety Issues," April 16, 1997. [9909290132]

1716. Memorandum to T. Gwynn from T. Martin, "Periodic Review of Low-Priority Generic Safety Issues," July 13, 1998. [9909290134]

1717. UCRL-52156, "Advisability of Seismic Scram," Lawrence Livermore Laboratory, June 30, 1976. [8103270386]

1718. SECY-98-166, "Summary of Activities Related to Generic Safety Issues," July 6, 1998. [9807220129, 9807170226]

1719. NUREG-1631, "Source Disconnects Resulting from Radiography Drive Cable Failures," U.S. Nuclear Regulatory Commission, June 1998.

1720. Memorandum to J. Craig from F. Combs, "Closure of NMSS Generic Issues," October 13, 1998. [9810160185]

1721. NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," U.S. Nuclear Regulatory Commission, January 1997.

1722. Letter to E. Fuller (Sierra Nuclear Corporation) from M. Knapp (NRC), "Closure of Confirmatory Action Letter 97-7-001," July 22, 1998. [9807290363]

1723. Memorandum to J. Craig from F. Combs, "Submittal of New Generic Issues for Tracking in the Generic Issues Management and Control System (GIMCS)," July 23, 1998. [9807280039]

1724. SECY-98-001, "Staff Requirements Memorandum 951219A - Briefing on Mechanisms for Addressing Generic Safety Issues," January 2, 1998. [9801230188, 9801140109]

1725. Memorandum to E. Ten Eyck, et al., from C. Paperiello, "NMSS Policy and Procedures Letter 1-57, Rev. 1, 'NMSS Generic Issues Program,'" October 30, 1997. [9711050048]

1726. NUREG/CR-6538, "Evaluation of LOCA With Delayed Loop and Loop With Delayed LOCA Accident Scenarios," U.S.Nuclear Regulatory Commission, July 1997.

1727. Memorandum to W. Travers from A. Thadani, "Resolution of Generic Safety Issue (GSI) - 171, 'ESF Failure from LOOP Subsequent to LOCA,'" December 9, 1998. [9909290137]

1728. Letter to J. Birmingham, et al., (NRC) from W. Foster (The B&W Owners' Group), "Submittal of B&WOG Report 'Evaluation of Potential Boron Dilution following Small Break Loss of Coolant Accident,' 77-5002260-00, September 1998," September 11, 1998. [9809150094]

1729. Letter to W. Lyon (NRC) from J. Link (The B&W Owners' Group), "Transmittal of Report 'Status Report on Return to Criticality Following Small Break Loss of Coolant Accident,' June 1998, Document No. 47-5001848-00," June 15, 1998. [9806220211]

1730. Memorandum to A. Thadani from S. Collins, "Potential Need to Reprioritize/Reopen Aspects of Generic Safety Issue (GSI) 22 Pertaining to Boron Dilution Following Loss-of-Coolant Accidents," February 1, 1999. [9902160085]

1731. Memorandum for W. Russell from D. Morrison, "Prioritization of the NRR Action Plans Submitted to RES on February 13, 1996," June 24, 1996. [9606260260]

1732. Memorandum for W. Minners from E. Beckjord, "Generic Issue No. 165, 'Spring-Actuated Safety and Relief Valve Reliability,'" November 26, 1993. [9312090116]

1733. Memorandum to W. Travers from A. Thadani, "Closeout of Generic Safety Issue 165, Spring-Actuated Safety and Relief Valve Reliability," June 18, 1999.

1734. Memorandum for W. Travers from A. Thadani, "Resolution of Generic Safety Issue B-61, 'Analytically Derived Allowable Equipment Outage Periods,'" March 2, 1999. [9904050209]

1735. Regulatory Guide 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," U.S. Nuclear Regulatory Commission, August 1998. [9809110028]

1736. Memorandum to M. Knapp from L. Shao, "Generic Issue No. 169, 'BWR MSIV Common Mode Failure Due to Loss of Accumulator Pressure,'" March 10, 1998. [9804070430]

1737. Memorandum to NRR Division Directors from D. Matthews, "Director's Quarterly Status Report," January 26, 1999. [9902040247]

1738. Memorandum to L. Shao from D. Morrison, "Generic Issue No. 171, 'ESF Failure from LOOP Subsequent to LOCA,'" June 16, 1995. [9507030081]

1739. Memorandum for J. Murphy from E. Beckjord, "Generic Issue No. 158, 'Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions,'" January 26, 1994. [9402040031]

1740. Memorandum to J. Murphy from E. Beckjord, "Generic Issue No. 167, 'Hydrogen Storage Facility Separation,'" September 29, 1994. [9410250044]

1741. NUREG/CP-0123, "Proceedings of the Second NRC/ASME Symposium on Pump and Valve Testing," U.S. Nuclear Regulatory Commission, July 1992.

1742. AEOD/C603, "A Review of Motor-Operated Valve Performance," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, December 1986. [8612150167]

1743. Memorandum for Chairman Zech, et al., from V. Stello, "Case Study Report - A Review of Motor-Operated Valve Performance (AEOD/C603)," December 10, 1986.

1744. Memorandum to W. Travers from A. Thadani, "Closeout of Generic Safety Issue (GSI) - 158, 'Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions,'" August 2, 1999. [9910040224]

1745. Memorandum for W. Minners from E. Beckjord, "Generic Issue No. 148, 'Smoke Control and Manual Fire-Fighting Effectiveness,'" August 26, 1992. [9208310325]

1746. Memorandum to A. Thadani from T. King, "Staff Review Guidance for Generic Safety Issue (GSI) 148, 'Smoke Control and Manual Fire-Fighting Effectiveness,'" July 22, 1999. [9907270312]

1747. Memorandum to G. Holahan from J. Wermeil, "Closeout of Core Performance Action Plan (TAC Nos. M91256, M91602)," February 16, 1999. [9902190260]

1748. Memorandum to Chairman Jackson, et al., from L. Callan, "Agency Program Plan for High-Burnup Fuel," July 6, 1998. [9808060096]

1749. Memorandum to D. Morrison from W. Russell, "Periodic Review of Low-Priority Generic Issues," April 11, 1996. [9604240169]

1750. Memorandum to B. Sheron from J. Craig, "Periodic Review of Low-Priority Generic Safety Issues," March 5, 1999. [9904060275]

1751. Letter to Seismic Qualification Advisory Committee (SQAC) and Meeting Attendees from G. Sliter and R. Vasudevan (EPRI), "Summary of the EPRI Seismic Equipment Qualification Research Coordination Meeting at ANCO Engineers, Inc., Los Angeles, California, September 19 & 20, 1984," October 10, 1984.

1752. NUREG/CR-5500, "Reliability Study: Westinghouse Reactor Protection System, 1984-1995," U.S. Nuclear Regulatory Commission, (Vol. 2) April 1999.

1753. Memorandum to D. Matthews from E. Rossi, "Issue of Final Report - System Reliability: Westinghouse Reactor Protection System, 1984-1995 (NUREG/CR-5500, Volume 2)," March 24, 1999. [9904060063]

1754. Memorandum for W. Minners from E. Beckjord, "Generic Issue No. 145, 'Actions to Reduce Common Cause Failures,'" February 11, 1992. [9203170332]

1755. NUREG/CR-6268, "Common-Cause Failure Database and Analysis System," U.S. Nuclear Regulatory Commission, (Vols. 1, 2, 3, and 4) June 1998.

1756. NRC Administrative Letter 98-04, "Availability of Common-Cause Failure Database," July 30, 1998. [9807240296]

1757. NRC Regulatory Issue Summary 99-03, "Resolution of Generic Issue 145, Actions to Reduce Common-Cause Failures," October 13, 1999. [9910060044]

1758. Memorandum to W. Travers from A. Thadani, "Resolution of Generic Safety Issue 145, 'Actions to Reduce Common Cause Failures,'" October 18, 1999.

1759. NUREG/CR-5640, "Overview and Comparison of U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, September 1990.

1760. NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995," U.S. Nuclear Regulatory Commission, February 1999.

1761. Memorandum to A. Thadani from E. Beckjord, "Generic Issue 156-6.1, 'Pipe Break Effects on Systems and Components,'" October 31, 1994. [9412070254]

1762. Memorandum for J. Murphy from E. Beckjord, "Generic Issue No. 156.6.1, 'Pipe Break Effects on Systems and Components,'" April 29, 1994. [9406200193]

1763. Memorandum to W. Travers from A. Thadani, "Closeout of Generic Safety Issue 23, 'Reactor Coolant Pump Seal Failure,'" November 8, 1999. [ML993370509]

1764. Memorandum to W. Travers from A. Thadani, "Closeout of Generic Safety Issue 190, 'Fatigue Evaluation of Metal Components for 60-Year Plant Life,'" December 26, 1999.

1765. Memorandum to W. Travers from S. Collins, "Closeout of Generic Safety Issue B-55, 'Improved Reliability of Target Rock Safety Relief Valves,'" December 17, 1999.

1766. Memorandum to W. Travers from A. Thadani, "Proposed Resolution of Generic Issue B-17, 'Criteria for Safety-Related Operator Actions,'" March 27, 2000. [ML003695959]

1767. NRC Regulatory Issue Summary 2000-02, "Closure of Generic Safety Issue 23, Reactor Coolant Pump Seal Failure," February 15, 2000. [ML003680402]

1768. NRC Regulatory Issue Summary 2000-03, "Resolution of Generic Safety Issue 158: Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions," March 15, 2000. [ML003686003]

1769. NRC Regulatory Issue Summary 2000-05, "Resolution of Generic Safety Issue 165, Spring-Actuated Safety and Relief Valve Reliability," March 16, 2000. [ML003689694]

1770. Memorandum for W. Russell from C. Miller, "Licensee Offsite Communication Capabilities; Results of Information Gathering Using Temporary Instruction," September 26, 1996.

1771. Memorandum to C. Rossi from D. Cool, "Status of NMSS Issues in the Generic Issue Management and Control System," June 25, 1999. [9907010194]

1772. Memorandum to C. Rossi from D. Cool, "Closure of NMSS Generic Issue," May 18, 1999.

1773. Memorandum for E. Beckjord from J. Milhoan, "Periodic Review of Low-Priority Generic Safety Issues," June 4, 1993.

1774. Memorandum to A. Thadani from S. Bahadur, "Reprioritization of GSI-71, 'Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety,'" December 20, 2000. [ML003779066]

1775. Memorandum to A. Thadani from M. Mayfield, "Closeout of GSI-152, 'Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads,'" April 4, 2001. [ML010740024]

1776. Memorandum to A. Thadani from J. Rosenthal, "Initial Screening of Candidate Generic Issue 187, 'The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump,'" April 30, 2001. [ML011210348]

1777. Memorandum to A. Thadani from B. Sheron, "Proposed Generic Safety Issue - The Potential Impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump," December 16, 1999. [ML993610109]

1778. Memorandum to W. Travers from A. Thadani, "Closure of Generic Issue 170, Reactivity Transients and Fuel Damage Criteria for High Burnup Fuel," May 4, 2001. [ML011280414]

1779. NUREG/CR-6395, "Enhanced Prioritization of Generic Safety Issue 156-6.1 Pipe Break Effects on Systems and Components Inside Containment," (Draft) September 1999. [ML010460480]

1780. Memorandum to F. Eltawila from D. Cool, "Submittal of Generic Issues for Tracking in the Generic Issue Management Control System (GIMCS)," November 14, 2000. [ML003763127]

1781. NRC Information Notice 99-26, "Safety and Economic Consequences of Misleading Marketing Information," August 24, 1999. [9908180183]

1782. NRC Information Notice 99-09, "Problems Encountered When Manually Editing Treatment Data on the Nucletron MicroSelectron-HDR (New) Model 105.999," March 24, 1999. [9903190227]

1783. NRC Information Notice 99-23, "Safety Concerns Related to Repeated Control Unit Failures of the Nucletron Classic Model High-Dose-Rate Remote Afterloading Brachytherapy Devices," July 6, 1999. [9907010001]

1784. Memorandum to W. Travers from W. Kane, "Closure of Two NMSS Generic Issues," January 26, 2001. [ML010240165]

1785. Memorandum to W. Travers from W. Kane, "Closure of NMSS Generic Issue Relating to Gamma Stereotactic Radiosurgery," February 12, 2001. [ML010390357]

1786. NRC Information Notice 2000-22, "Medical Misadministrations Caused by Human Errors Involving Gamma Stereotactic Radiosurgery (Gamma Knife)," December 18, 2000. [ML003761619]

1787. Memorandum to F. Eltawila from D. Cool, "NMSS Input for Second Quarter FY-2001 Update of the Generic Issue Management Control System," April 12, 2001. [ML011000117]

1788. NUREG-1090, "U.S. Nuclear Regulatory Commission 1983 Annual Report," June 1984.

1789. Memorandum to A. Thadani from S. Collins, "Proposed Generic Safety Issue Related to Secondary Containment Drawdown Time," December 3, 2001. [ML013330114]

1790. NUREG/CR-5640, "Overview and Comparison of U.S. Commercial Nuclear Power Plants," U.S. Nuclear Regulatory Commission, September 1990.

1791. Memorandum to J. Flack from M. Cunningham, "Information Concerning Generic Issue on Combustible Gas Control for PWR Ice Condenser and BWR Mark III Containment Designs," August 15, 2001. [ML012330522]

1792. SECY-00-0198, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-Informed Changes to 10 CFR 50.44 (Combustible Gas Control)," September 14, 2000. [ML003747699]

1793. NUREG/CR-6427, "Assessment of the DCH Issue for Plants with Ice Condenser Containments," U.S. Nuclear Regulatory Commission, April 2000.

1794. Memorandum to M. Snodderly (NRC) from M. Zavisca, et al. (ERI), "Combustible Gas Control Risk Calculations (DRAFT) for Risk-Informed Alternative to Combustible Gas Control Rule for PWR Ice Condenser, BWR Mark I, and BWR Mark III (10 CFR 50.44)," October 22, 2001.

1795. NUREG/CR-4551, "Evaluation of Severe Accident Risks," U.S. Nuclear Regulatory Commission, (Vol. 1, Rev. 1) December 1993, (Vol. 4, Rev. 1, Part 1) December 1990, (Vol. 7, Rev. 1) March 1993.

1796. Letter Report, "NUREG-1150 Data Base Assessment Program: A Description of the Computational Risk Integration and Conditional Evaluation Tool (CRIC-ET) Software and the NUREG-1150 Data Base," T. D. Brown *et. al.*, March 1995.

1797. Letter to H. VanderMolen (NRC) from V. Mubayi (BNL), "NUREG-1150 Consequence Calculations," July 20, 1994.

1798. NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program, Main Report," (Volumes 1 and 2) April 2002.

1799. Memorandum for A. Thadani from J. Wiggins, "Staff Member Concern Regarding the Potential for Resonance Vibrations of Steam Generator Tubes During a Main Steam Line Break Event," June 27, 2000.

1800. Letter to W. Travers from D. Powers, "Differing Professional Opinion on Steam Generator Tube Integrity," February 1, 2001. [ML010780125]

1801 Letter to W. F. Conway (Arizona Public Service Company) from J. B. Martin (NRC), "NRC Inspection Report 50-529/93-14," April 16, 1993. [9305030083]

1802. Letter to A. A. Blind (Consolidated Edison Company of New York, Inc.) from H. J. Miller (NRC), "NRC Augmented Inspection Team - Steam Generator Tube Failure - Report No. 05000247/2000-002," April 28, 2000. [ML011930057]

1803. NUREG/IA-0137, "A Study of Control Room Staffing Levels for Advanced Reactors," U.S. Nuclear Regulatory Commission, November 2000. [ML003774060]

1804. NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," August 3, 1995. [9507310085]

1805. Memorandum to A. Thadani from N. Chokshi, "Initial Screening of Candidate Generic Issue 188, 'Steam Generator Tube Leaks or Ruptures, Concurrent with Containment Bypass from Main Steam Line or Feedwater Line Breaches," May 21, 2001. [ML011410572]

1806. Memorandum to W. Travers from A. Thadani, "Closeout of Generic Safety Issue 172, 'Multiple System Responses Program,'" January 22, 2002. [ML020230162]

1807. Memorandum to W. Travers from S. Collins, "Resolution of Generic Safety Issue (GSI) 173A, 'Spent Fuel Storage Pool for Operating Facilities,'" December 19, 2001. [ML013520142]

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

1809. Memorandum to S. Collins from A. Thadani, "Generic Issue 80, 'Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR MARK I and II Containments,'" February 14, 2003. [ML030550470]

1810. Memorandum to M. Knapp from S. Collins, "Periodic Review of Low-Priority Generic Safety Issues," March 25, 1998. [9803310320]

1811. Memorandum to F. Eltawila from M. Mayfield, "Transfer of Responsibility for Generic Issue 80, 'Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR MARK I and II Containments,'" April 3, 2001. [ML010470257, ML010670420]

1812. NRC Generic Letter 2003-01, "Control Room Habitability," June 12, 2003. [ML031620248]

1813. Memorandum to A. Thadani from H. Vandermolen, "Results of Initial Screening of Generic Issue 192, 'Secondary Containment Drawdown Time,'" July 2, 2002, [ML021840788]

1814. Memorandum to S. Newberry from J. Grobe,"Potentially Generic Safety Issue - BWR ECCS Suction Concerns," May 10, 2002. [ML021340802]

1815. Memorandum to F. Eltawila from S. Newberry, "Potentially Generic Safety Issue - BWR ECCS Suction Concerns," May 28, 2002. [ML021480496]

1816. NRC Bulletin 96-03, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors," May 6, 1996.

1817. AEOD-E218, "Potential for Air Binding or Degraded Performance of BWR RHR System Pumps During the Recirculation Phase of a LOCA," U.S. Nuclear Regulatory Commission, March 31, 1982.

1818. NEDE-24539-P, "Mark I Containment Program: Full Scale Test Program Final Report," General Electric Company, April 1979.

1819. Memorandum to J. Dyer from S. Reynolds, "Ad Hoc Review Panel Recommendation Regarding a Differing Professional View on BWR ECCS Suction Concerns," April 8, 2002.

1820. CEN 420-P, Volume 1, "Small Break LOCA Realistic Evaluation Model: Calculational Models," October 1993. [9310080210]

1821. NUREG/CR-2792, "An Assessment of Residual Heat Removal and Containment Spray Pump Performance Under Air and Debris Ingesting Conditions," U.S. Nuclear Regulatory Commission, September 1982.

1822. Memorandum to J. Flack from F. Eltawila, "Information on Generic Issue 195, 'Hydrogen Combustion in Foreign BWR Piping,'" March 13, 2003. [ML030720615]

1823. NRC Information Notice 2002-15, "Hydrogen Combustion Events in Foreign BWR Piping," April 12, 2002 [ML020980466], (Supplement 1) May 6, 2003 [ML031210054].

1824. Memorandum to A. Thadani from F. Eltawila, "Results of Initial Screening of Generic Safety Issue 193, 'BWR ECCS Suction Concerns,'" October 16, 2003. [ML032940708]

1825. Information Notice No. 88-23, "Potential for Gas Binding of High-Pressure Safety Injection Pumps During a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission, May 12, 1998 [8805060246], (Supplement 1) January 5, 1989 [8812300186], (Supplement 2) January 31, 1990 [9001250020], (Supplement 3) December 10, 1990 [9012040239], (Supplement 4) December 18, 1992 [9212150017], (Supplement 5) April 23, 1999 [9904200058].

1826. Information Notice No. 89-80, "Potential for Water Hammer, Thermal Stratification, and Steam Binding in High-Pressure Coolant Injection Piping," U.S. Nuclear Regulatory Commission, December 1, 1989. [8911270002]

1827. Information Notice No. 90-64, "Potential for Common-Mode Failure of High Pressure Safety Injection Pumps or Release of Reactor Coolant Outside Containment During a Loss-of-Coolant Accident," U.S. Nuclear Regulatory Commission, October 4, 1990. [9111040293]

1828. NRC Letter to All Nuclear Power Reactor Licensees and Applicants, "Information to Licensees Regarding Two NRC Inspection Manual Sections on Resolution of Degraded and Nonconforming Conditions and on Operability (Generic Letter 91-18)," November 7, 1991 [9111040293], (Revision 1) October 8, 1997 [9710060322].

1829. Memorandum to F. Eltawila from G. Holahan, "Hydrogen Detonations in BWRs," March 11, 2003.

1830. AEOD/E910, "Potential for Gas Binding of High Head Safety Injection Pumps Resulting From Inservice Testing of VCT Outlet Isolation Valves," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, December 1989.

1831. AEOD/T515, "Residual Heat Removal Service Water Booster Pump Air Binding at Brunswick Unit 1," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, December 5, 1985.

1832. AEOD/T927, "Follow-up on Steam Binding of AFW Pumps," Office for Analysis and Evaluation of Operational Data, U.S. Nuclear Regulatory Commission, December 29, 1989.

1833. Memorandum to A. Thadani from J. Flack, "Results of Initial Screening of Generic Issue 195, 'Hydrogen Combustion in Foreign BWR Piping,'" February 23, 2004. [ML040850566]

1834. NUREG/CR-1582, "Seismic Hazard Analysis - Overview and Executive Summary," U.S. Nuclear Regulatory Commission, (Vol. 1) April, 1983.

1835. NUREG/CR-5250, "Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains," U.S. Nuclear Regulatory Commission, (Vols. 1 to 8) January 1989.

1836. NUREG-1488, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains," U.S. Nuclear Regulatory Commission, (Draft) October 1993.

1837. Memorandum to J. Flack from D. Dorman, "Proposed Generic Safety Issue on the Implications of Updated Probabilistic Seismic Hazard Estimates," June 6, 2002. [ML021580151]

1838. NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts," U.S. Nuclear Regulatory Commission, April, 1997. [9705280207]

1839. NUREG/CR-6607, "Guidance for Performing Probabilistic Seismic Hazard Analysis for a Nuclear Plant Site: Example Application to the Southeastern United States," U.S. Nuclear Regulatory Commission, October 2002.

1840. NUREG/CR-6728, "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-consistent Ground Motion Spectra Guidelines," U.S. Nuclear Regulatory Commission, October 2001.

1841. Memorandum to A. Thadani from N. Chokshi, "Results of Initial Screening of Generic Issue 194, 'Implications of Updated Probabilistic Seismic Hazard Estimates,'" September 12, 2003. [ML032680979]

1842. NRC Letter to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, "Control of Heavy Loads," (Generic Letter 80-113) December 22, 1980.

1843. NRC Letter to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, "Control of Heavy Loads Generic Letter 81-07)," February 3, 1981.

1844. NRC Letter to All Licensees for Operating Reactors, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants,' NUREG-0612 (Generic Letter 85-11)" June 28, 1985. [8506270216]

1845. Memorandum to A. Thadani from B. Sheron, "Proposed Generic Safety Issue - Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants," April 19, 1999. [ML003714155]

1846. NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," U.S. Nuclear Regulatory Commission, July 2003.

1847. NRC Letter to All Holders of Operating Licenses, Applicants for Operating Licenses and Holders of Construction Permits for Power Reactors (Generic Letter 83-42)," December 19, 1983. [8312190365]

1848. NRC Bulletin 96-02, "Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment," April 11, 1996. [9604080259]

1849. NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, May 1979.

1850. Memorandum to J. Dyer from A. Thadani, "Proposed Recommendations for Generic Issue (GI)-186, 'Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants,'" November 12, 2003. [ML033170301]

1851. Memorandum to M. Mayfield from F. Eltawila, "Proposed Recommendations for Generic Issue (GI)-186, 'Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants,'" November 21, 2003. [ML033250654]

1852. NRC Regulatory Issue Summary 2003-09, "Environmental Qualification of Low-Voltage Instrumentation and Control Cables," May2, 2003. [ML031220078]

1853. Memorandum to W. Travers from R. Borchardt, "Closeout of Generic Safety Issue (GSI) 168, 'Environmental Qualification of Low-Voltage Instrumentation and Control Cables,'" August 14, 2003. [ML032060326]

1854. Memorandum to F. Eltawila from D. Carlson, "Proposed Generic Safety Issue: Boral Degradation," November 4, 2003. [ML033090600]

1855. Memorandum to D. Carlson from F. Eltawila, "Generic Issue 196: Boral Degradation," November 10, 2003, [ML033160580]

1856. "Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM Cask System)," Docket 72-1014, July 19, 2000. [ML003777866]

1857. NUREG-75/110, "Safety Evaluation Report for Preliminary Design Approval for GESSAR-238 - Nuclear Island Standard Design," U.S. Nuclear Regulatory Commission, December 1975.

1858. Management Directive 6.4, "Generic Issues Program," U.S. Nuclear Regulatory Commission, December 4, 2001.

1859. SECY-00-0038, "The 1999 NRC Annual Report," February 15, 2000.

1860. Memorandum to F. Eltawila from M. Gamberoni, "Proposed Generic Issue: Iodine Spiking Phenomena," July 22, 2004. [ML042090538]

1861. NUREG-1740, "Voltage-Based Alternative Repair Criteria: A Report to the Advisory Committee on Reactor Safeguards by the Ad Hoc Subcommittee on a Differing Professional Opinion," U.S. Nuclear Regulatory Commission, March 2001.

1862. Letter to W. Travers from M. Bonaca, "Resolution of Certain Items Identified by the ACRS in NUREG-1740, 'Voltage-Based Alternative Repair Criteria,'" May 21, 2004. [ML0414202370]

1863. NUREG-1542, "Performance and Accountability Report," U.S. Nuclear Regulatory Commission.

1864. NUREG/BR-0184, "Regulatory Analysis Technical Information Handbook," U.S. Nuclear Regulatory Commission, January 1997.

1865. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," U.S. Nuclear Regulatory Commission, July 2000.

1866. Memorandum to A. Thadani, et al., from M. Hodges, "Reassessment of the Assumptions and Proposed Alternative Method for Determining Radiological Consequences of Main Steam Line Break and Steam Generator Tube Rupture," June 7, 1996. [ML003702950]

1867. Memorandum to C. Paperiello from J. Uhle, "Results of Initial Screening of Generic Issue 197, 'Iodine Spiking Phenomena,'" May 8, 2006. [ML061100331]

1868. Memorandum to L. Reyes from C. Paperiello, "Closure of Generic Issue 80, 'Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments,'" November 17, 2005. [ML053000179]

1869. Memorandum to L. Reyes from C. Paperiello, "Closure of Generic Safety Issue 185, 'Control of Recriticality Following Small-Break LOCAs in PWRs,'" September 23, 2005. [ML052590135]

1870. Memorandum to L. Reyes from C. Paperiello, "Completion of Generic Safety Issue 188, 'Steam Generator Tube Leaks or Ruptures Concurrent with Containment Bypass from Main Steam or Feedwater Line Breaches,'" December 16, 2005. [ML052150154]

1871. Memorandum to L. Reyes from J. Larkins, "Resolution of Generic Safety Issue 188, 'Steam Generator Tube Leaks or Ruptures Concurrent with Containment Bypass from Main Steamline or Feedwater Line Breaches,'" March 17, 2006. [ML060870089]

1872. NUREG/CR-2726, "Light Water Reactor Hydrogen Manual," U.S. Nuclear Regulatory Commission, August 1983.

1873. NUREG/CR-6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," U.S. Nuclear Regulatory Commission, December 1999.

1874. Memorandum to B. Sheron from S. Bahadur, "Results of Initial Screening of Generic Issue 198, 'Hydrogen Combustion in PWR Piping,'" March 22, 2007. [ML070580447]

1875. Memorandum to C. Paperiello from J. Rosenthal, "Results of Initial Screening of Generic Issue 196, 'Boral Degradation,'" November 19, 2004. [ML042670379]

1876. Memorandum to L. Reyes from B. Sheron, "Closure of Generic Issue 196, 'Boral Degradation,'" February 22, 2007. [ML070100403]

1877. Memorandum to F. Eltawila from R. Barrett, "Proposed Generic Safety Issue: Tin Whiskers," August 18, 2005. [ML052080291]

1878. Information Notice 2005-25, "Inadvertent Reactor Trip and Partial Safety Injection Actuation Due to Tin Whisker," U.S. Nuclear Regulatory Commission, August 25, 2005. [ML052150404]

1879. Memorandum to B. Sheron from C. Lui, "Results of Initial Screening of Generic Isuue 200, 'Tin Whiskers,'" January 26, 2007. [ML070250399]

1880. Memorandum to F. Eltawila from M. Mayfield, "Small-Break Loss-of-Coolant Accident and Loss of Offsite Power Scenario (Allegation No. NRR-2006-A-0003)," August 7, 2006.

1881. NUREG/CR-6890, "Reevaluation of Station Blackout Risk at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, December 2005.

1882. Memorandum to B. Sheron from J. Monninger, "Results of Initial Screening of Generic Issue 201, 'Small-Break Loss-of-Coolant Accident with Loss of Offsite Power,'" March 28, 2007. [ML070820112]

1883. Memorandum to F. Eltawila from M. Mayfield, "Proposed Generic Safety Issue: Spent Fuel Pool Leakage Limits," August 30, 2006. [ML062260014]

1884. Letter to W. Lewis (PSEG) from A. Burritt (NRC), "Salem Nuclear Generating Station – NRC Integrated Inspection Report 05000272/2007002 and 05000311/2007002," U.S. Nuclear Regulatory Commission, May 11, 2007. [ML071350075]

1885. "Liquid Radioactive Release Lessons Learned Task Force Final Report, U.S. Nuclear Regulatory Commission," September 1, 2006. [ML062650312]

1886. Information Notice No.2004-05, "Spent Fuel Pool Leakage to Onsite Groundwater," U.S. Nuclear Regulatory Commission, March 3, 2004. [ML040580454]

1887. Information Notice No. 2006-13, "Ground-Water Contamination Due to Undetected Leakage of Radioactive Water," U.S. Nuclear Regulatory Commission, July 10, 2006. [ML060540038]

1888. SECY-07-0022, "Status Report on Proposed Improvements to the Generic Issues Program," U.S. Nuclear Regulatory Commission, January 30, 2007. [ML063460239]

1889. Memorandum to B. Sheron from K. O'Brien, "Results of Initial Screening of Generic Issue 202, "Spent Fuel Pool Leakage Impacts,'" May 30, 2007. [ML071450125]

1890. Regulatory Issue Summary 2005-25, "Clarification of NRC Guidelines for Control of Heavy Loads," U.S. Nuclear Regulatory Commission, October 31, 2005. [ML05234485]

1891. Memorandum to M. Karmis from J. Foster, "Requested Generic Issue (GI)-203, Potential Safety Issues with Cranes that Lift Spent Fuel Casks," March 6, 2007. [ML070400565]

1892. NUREG/CR-6870, "Consideration of Geochemical Issues in Groundwater Restoration at Uranium In-Situ Leach Mining Facilities," U.S. Nuclear Regulatory Commission, January 2007.

1893. Memorandum to L. Reyes from C. Miller, "Closeout of Generic Safety Issue NMSS-0014, 'Surety Estimates for Groundwater Restoration at In-Situ Leach Facilities,'" May 25, 2007. [ML070790303]

1894. Memorandum to L. Reyes from C. Miller, "Closeout of Generic Safety Issue NMSS-016, 'Adequacy of 0.05 Weight Percent Limit in 10 CFR 40,'" November 27, 2006. [ML062000507]

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

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

Legend

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

Appendix B (Continued)

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

Appendix B (Continued)

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

Appendix B (Continued)

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

Appendix B (Continued)

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

Appendix B (Continued)

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

Appendix B (Continued)

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

Appendix B (Continued)

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

Appendix B (Continued)

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

Appendix B (Continued)

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

Appendix B (Continued)

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

Appendix B (Continued)

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

Appendix B (Continued)

Action Plan Item/Issue No.	Title	Safety Priority/Status	Affected NSSS Vendor		Operating Plants-MPA No	Operating Plants - Effective Date	Future Plants-Effective Date
			BWR	PWR			
	Sites						
155	Generic Concerns Arising from TMI-2 Cleanup	-	-	-	-	-	-
155.1	More Realistic Source Term Assumptions	NOTE 3(a)	All	All	NA	NA	02/--/95
156	Systematic Evaluation Program	-	-	-	-	-	-
156.6.1	Pipe Break Effects on Systems and Components	HIGH	All	All		TBD	TBD
163.	Multiple Steam Generator Tube Leakage	HIGH	NA	All		TBD	TBD
177.	Vehicle Intrusion at TMI	NOTE 3(a)	All	All		08/01/94	08/01/94
186.	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants	CONTINUE	All	All		TBD	TBD
189.	Susceptibility of Ice Condenser Containments to Early Failure from Hydogen Combustion During A Severe Accident	CONTINUE	All	All		TBD	TBD
191.	Assessment of Debris Accumulation on PWR Sump Performance	HIGH	NA	All		TBD	TBD
193.	BWR ECCS Suction Concerns	CONTINUE	All	NA		TBD	TBD
199.	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States	NOTE 4	All	All		TBD	TBD
		HUMAN FACTORS ISSUES					
HF1	STAFFING AND QUALIFICATIONS						
HF.1.1	Shift Staffing	NOTE 3(a)	All	All		01/--/84	01/--/84

APPENDIX F

NUCLEAR MATERIAL SAFETY AND SAFEGUARDS GSIs

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

TABLE F.1

LISTING OF NMSS GSIs

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

Legend

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

Issue No.	Title	Priority Engineer	LeadOffice/ Division/Branch	Safety Priority Ranking	Latest Revision	Latest Issuance Date
NMSS-0001	Door Interlock Failure Resulting from Faulty MicroSelectron-High Dose Rate Remote Afterloader	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0002	Significant Quantities of Fixed Contamination Remain in Krypton-85 Leak-Detection Devices After Venting	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0003	Corrosion of Sealed Sources Caused by Sensitization of Stainless Steel Source Capsules During Shipment	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0004	Overexposures Caused by Sources Stolen from Facility of Bankrupt Licensee	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)		12/31/1998
NMSS-0005	Potential for Erroneous Calibration, Dose Rate, or Radiation Exposure Measurements With Victoreen Electrometers	Ramsey	NMSS/IMNS/IMOB	NOTE 3(a)		12/31/1998
NMSS-0006	Criticality in Low-Level Waste	Ramsey	NMSS/IMNS/IMOB	NOTE 3(b)	1	06/30/2000
NMSS-0007	Criticality Benchmarks Greater Than 5% Enrichment	Ramsey	NMSS/FCSS	HIGH	1	06/30/2001
NMSS-0008	Year 2000 Computer Problem - Non-Reactor Licensees	Ramsey	NMSS/IMNS	NOTE 3(b)	1	06/30/2000
NMSS-0009	Amersham Radiography Source Cable Failures	Ramsey	NMSS/IMNS	NOTE 3(b)		12/31/1998
NMSS-0010	Troxler Gauge Source Rod Weld Failures	Ramsey	NMSS/IMNS	NOTE 3(b	1	06/30/2002

Revision 5

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

NMSS-0014: SURETY ESTIMATES FOR GROUNDWATER RESTORATION AT IN-SITU LEACH FACILITIES

DESCRIPTION

This issue was identified[1723] by NMSS to pursue research to provide a methodology to calculate surety for groundwater restoration activities at in situ leach uranium extraction facilities and a post-restoration groundwater quality stability monitoring methodology. The following tasks were envisioned: (1) review approaches used to estimate pore volumes and to calculate surety amounts and obtain data to evaluate these approaches; (2) develop a pore volume estimation methodology and document it in a NUREG report; (3) develop cost estimation methodology for use in evaluating the level of financial surety required; (4) brief regulators on the surety methodology; (5) review the existing approaches used to determine an appropriate time period for post-restoration monitoring period and obtain datasets to evaluate the methodologies; (6) use the datasets to develop and test the methodologies; (7) develop a robust methodology; and (8) transfer the methodology to regulators through briefings and a NUREG report.

CONCLUSION

This issue was given a medium priority ranking and resolution was pursued.[1723] The staff's work resulted in the publication of NUREG/CR-6870[1892] which provides a useful perspective on the costs associated with groundwater restoration at in-situ leach facilities, and describes several methods for estimating these costs at new facilities, including a rigorous approach for development of a site-specific model that considers groundwater flow, solute transport, and geochemical reactions in the uranium ore zone of interest. With this modeling approach, groundwater restoration volumes and, correspondingly, their associated costs, can be estimated. As such, NUREG/CR-6870[1892] provides the methodology originally requested by the Uranium Recovery Program. Thus, the issue was resolved with no new requirements or guidance for licensees.[1893]

REFERENCES

1723. Memorandum to J. Craig from F. Combs, "Submittal of New Generic Issues for Tracking in the Generic Issues Management and Control System (GIMCS)," July 23, 1998.

1892. NUREG/CR-6870, "Consideration of Geochemical Issues in Groundwater Restoration at Uranium In-Situ Leach Mining Facilities," U.S. Nuclear Regulatory Commission, January 2007.

1893. Memorandum to L. Reyes from C. Miller, "Closeout of Generic Safety Issue NMSS-0014, 'Surety Estimates for Groundwater Restoration at In-Situ Leach Facilities,'" May 25, 2007. [ML070790303]

NMSS-0016: ADEQUACY OF 0.05 WEIGHT PERCENT LIMIT IN 10 CFR 40

DESCRIPTION

Exposure to the "unimportant quantities" of source material, defined in 10 CFR 40.13(a) as less than 0.05 wt% uranium or thorium, could result in annual doses to the public of hundreds of millirem exceeding NRC's public dose limit of 100 mem/year from all sources. In July 1996, NMSS developed a Draft User Need memorandum requesting development of a regulation to limit the transfer of source material meeting the "unimportant quantity" limit or to revise the definition of source material. Discussions in 1996 and 1997 with RES and OGC indicated that there were several options available to the staff to revise the definition of source material. However, the User Need memorandum was never finalized.

Subsequently, the Division of Fuel Cycle Safety (FCSS)/NMSS received a licensee request to transfer baghouse dust containing less than 0.05 wt% uranium and thorium to an exempt person per 10 CFR 40.51(b)(3) and 40.13(a). Some conservative dose estimates indicated that the transfer could result in doses exceeding the public dose limit. This issue was identified[1723] by NMSS to pursue rulemeking to immediately cease transfers under 10 CFR 40.51(b)(3) and 40.51(b)(4) of source material exempted under 10 CFR 40.13(a). By eliminating these provisions, any future transfers would have to meet existing general license conditions or be specifically approved on a case-by-case basis.

CONCLUSION

This issue was given a medium priority ranking and resolution was pursued.[1723] The staff initially pursued a modification to 10 CFR 40.51, "Transfer of source or byproduct material," to codify the Commission's direction given in various SRMs. However, in November 2006, NMSS concluded that the existing NRC policy for handling the source material satisfied the intent of the Commission directives, and the issue was closed by the Office of Federal and State Materials and Environmental Management Programs (FSME)/NMSS with no new requirements or guidance for licensees. The staff plans to take further action related to the proposed rule (i.e., publish the final rule or rescind the proposed rule) when the EPA finalizes activities related to its ANPR, and/or the recommendation of the Interagency Jurisdictional Working Group(IJWG) is implemented.[1894]

REFERENCE

1723. Memorandum to J. Craig from F. Combs, "Submittal of New Generic Issues for Tracking in the Generic Issues Management and Control System (GIMCS)," July 23, 1998.

1894. Memorandum to L. Reyes from C. Miller, "Closeout of Generic Safety Issue NMSS-016, 'Adequacy of 0.05 Weight Percent Limit in 10 CFR 40,'" November 27, 2006. [ML062000507]

NRC FORM 335
(9-2004)
NRCMD 3.7

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER
(Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)

NUREG-0933
Supplement 31

2. TITLE AND SUBTITLE

A Prioritization of Generic Safety Issues

3. DATE REPORT PUBLISHED

MONTH	YEAR
June	2007

4. FIN OR GRANT NUMBER

5. AUTHOR(S)

R. Emrit, et al.

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

07-01-2006 to 06-30-2007

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Risk Assessment and Special Projects
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Division of Risk Assessment and Special Projects
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report presents the safety priority rankings for generic safety issues related to nuclear power plants. The purpose of these rankings is to assist in the timely and efficient allocation of NRC resources for the resolution of those safety issues that have a significant potential for reducing risk. The safety priority rankings are HIGH, MEDIUM, LOW, DROP, and CONTINUE, and have been assigned on the basis of risk significance estimates, the ratio of risk to costs and other impacts estimated to result if resolution of the safety issues were implemented, and the consideration of uncertainties and other quantitative or qualitative factors. To the extent practical, estimates are quantitative.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

generic safety issues

13. AVAILABILITY STATEMENT

unlimited

14. SECURITY CLASSIFICATION

(This Page)
unclassified

(This Report)
unclassified

15. NUMBER OF PAGES

218

16. PRICE

PRINTED ON RECYCLED PAPER

Printed
on recycled
paper

Federal Recycling Program