

February 10, 1994

Docket No. 52-001

Mr. Joseph Quirk
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Dear Mr. Quirk:

SUBJECT: ROUND THREE OF STAFF FEEDBACK ON THE ADVANCED BOILING WATER REACTOR
(ABWR) AMENDMENT 33 TO THE STANDARD SAFETY ANALYSIS REPORT (SSAR),
AND TECHNICAL SPECIFICATIONS

I am providing the third round of staff comments on GE's SSAR
Amendment 33. They include additional Plant Systems Branch, Standardization
Project Branch and a markup of one technical specification (TS) page generated
by the staff audit. I expect to provide a few remaining comments on the TS by
early next week. If you have any questions on these comments please contact
me on 301-504-1132.

(Original signed by)

Chester Poslusny, Project Manager
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Enclosure:
As stated

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DATE	02/10/94	02/10/94	02/10/94

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Docket No. 52-901

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SPLB COMMENTS ON TABLE 1.8
ABWR SSAR

Page 1.8-5, Table 1.8-6, SRP Section 6.2.1.1. The reference to the subsection where this is discussed should be 6.2.1.1.5.6.1.

Page 1.8-18, Table 1.8-19. Item 9.3.3 should add a note similar to 9.2.4 above it which states that the system is partly ABWR and partly COL. This should also be done for all other interfacing systems.

Page 1.8-19, Table 1.8-19. Item 9.5.8 should be revision 2, not 3.

Page 1.8-26, Table 1.8-20. Item 1.26 should say Rev. 3, 2/76 (Issue date), and applicable to the ABWR.

Page 1.8-30, Table 1.8-20. Item 1.89 should say 6/84, not 7/84.

Page 1.8-36, Table 1.8-21. Item 57.1 should have in parentheses (ANSI N208)

Page 1.8-36, Table 1.8-21. Item 57.2 is ANSI N210, not 270. Also, the title is "Design Objectives", not "Design Requirements." Finally, a new entry should follow this:

57.3, "Design Requirements for New LWR Fuel Storage Facilities."

Page 1.8-50, Table 1.8-22. Item 80-03 should be followed by a new entry 80-05, "Vacuum Condition Resulting in Damage to Chemical and Volume Control System (CVCS) Holdup Tanks."

The issue date for Revision 1 of Regulatory Guide 1.89 is June 1984, not July 1984 as in the current version of the SSAR.

Enclosure

Table 1.8-5 Summary of Differences from SRP Section 5 (Continued)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
5.4.7	Branch Technical Position RSB 5-1, B.1.(b) and (c)—Diverse interlocks for RHR suction isolation valves.	No diversity of interlocks.	5.4.7.1.1.7

Table 1.8-6 Summary of Differences from SRP Section 6

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
6.2.1.1	Design provision for automatic actuation of wetwell spray 10 minutes following a LOCA signal	Manual actuation of wetwell spray 30 minutes following a LOCA signal	6.2.1.1.5.5.1 <i>6.2.1.1.5.6.1</i>
6.2.4	One isolation valve inside and one isolation valve outside containment	Both isolation valves located outside the containment	6.2.4.3.2.2.2.3
	Purge and vent valves to close in ≤5 seconds	Purge and vent valves will close in ≤20 seconds	6.2.4.3.2.2.2.3

Table 1.8-7 Summary of Differences from SRP Section 7

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
7.1	Table 7-1: 1a IEEE-279 4.19	RHR Annunciation at loop level.	7.3.2.3.2 (1) 7.3.2.4.2 (1) 7.4.2.3.2 (1)
7.1	Table 7-1: 2i GDC 20	Some modes of RHR are not automatic.	7.3.2.3.2 (2)(b) 7.3.2.4.2 (2)(b) 7.4.2.3.2 (1)
7.1	Table 7-1: 3a Reg Guide 1.22	Clarification of requirements.	7.3.2.1.2. (3)(a)
7.1	Table 7-1: 3a Reg Guide 1.22	HP/LP interlocks cannot be tested during power operation.	7.6.2.3.2 (3)
7.1	Table 7-1: 3c Reg Guide 1.53	Continuity testing of certain solenoids.	7.3.2.1.2. (3)(c)
7.1	Table 7-1: 3c Reg Guide 1.53	Some components are not redundant.	7.3.2.5.2 (3) 7.4.2.2.2 (3)
7.1	Table 7-1: 3c Reg Guide 1.53	Limited redundancy of re: vote shutdown.	7.4.2.4.2 (1) 7.4.2.4.2 (3)
7.1	Table 7-1: 3e Reg Guide 1.75	Alternate positions employed.	7.1.2.10.5

Table 1.8-7 Summary of Differences from SRP Section 7 (Continued)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
7.1	Table 7-1: 3h Reg Guide 1.118	Some sensors cannot be tested at power operation.	7.2.2.2.1 (7) 7.2.2.2.3.1. (10) 7.2.2.2.3.1 (21)
7.1	Table 7-1: 4i BTP ICSB 22	Some actuators cannot be exercised during power operation.	7.3.2.1.2 (4)(d) 7.4.2.3.2 (4)(c)

Table 1.8-8 Summary of Differences from SRP Section 8

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
8.1	Table 8-1: 2f Reg Guide 1.75	Exception to LOCA trip for certain non-1E loads.	8.1.3.1.2.2 (6) 8.3.1.4.2.2.4 Appendix 9A
8.1	Table 8-1: 2f Reg Guide 1.75	15-ft cable marking intervals.	8.3.1.3.1
8.1	Table 8-1: 2f IEEE-384	LDS divisional separation in steam tunnel.	8.3.1.4.2.2.2

Table 1.8-9 Summary of Differences from SRP Section 9

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
9.3.1	II.1—Particles shall not exceed 3 microns.	Instrument air is filtered to 5 microns.	9.3.6.2
9.3.2	II.k.5—Capable of sampling liquid of 10 ci/cm ³ .	Cable of sampling liquids of 1 ci/cm ³ .	9.3.2.3.1
9.4.1	GDC 19	Site specific.	6.4.7.1 6.4.7.1 ←

Table 1.8-10 Summary of Differences from SRP Section 10

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
None	None	None	None

Table 1.8-19 Standard Review Plans and Branch Technical Positions
Applicable to ABWR (Continued)

SRP No.		Appl. Rev.	Issued Date	ABWR Applicable?	Comments
	Appendix A	2	7/81	Yes	
6.5.1	ESF Atmosphere Cleanup Systems	2	7/81	Yes	
6.5.2	Containment spray as a Fission Product Cleanup System	1	7/81	Yes	
6.5.3	Fission Product Control Systems and Structures	2	7/81	Yes	
6.5.4	Ice Condenser as a Fission Product Cleanup System	2	7/81	No	PWR only
6.5.5	Pressure Suppression Pool as a Fission Product Cleanup System	0	12/88	Yes	
6.6	Inservice Inspection of Class 2 and 3 Components	1	7/81	Yes	
6.7	Main Steam Isolation Valve Leakage Control System (BWR)	2	7/81	Yes	NO?
Chapter 7 Instrumentation and Controls					
7.1	Instrumentation and Controls Introduction	3	2/84	Yes	
	Table 7-1 Acceptance Criteria and Guidelines for Instrumentation and Controls Systems Important to Safety	3	2/84	Yes	
	Table 7-2 TMI Action Plan Requirements for Instrumentation and Controls Systems important to Safety	0	7/81	Yes	
	Appendix A	1	2/84	Yes	
	Appendix B	0	7/81	Yes	
7.2	Reactor Trip System	2	7/81	Yes	
	Appendix A (Superseded by SRP 7.1 App. B)				
7.3	Engineered Safety Features Systems	2	7/81	Yes	
	Appendix A (Superseded by SRP 7.1 App. B)				
7.4	Safe Shutdown Systems	2	7/81	Yes	
7.5	Information Systems Important to Safety	3	2/84	Yes	
7.6	Interlock systems Important to Safety	2	7/81	Yes	
7.7	Control Systems	3	2/84	Yes	
	Appendix 7-A Branch Technical Positions (ICSB)	2	7/81	Yes	
	BTP ICSB 1 (DOR) (Deleted)				

Table 1.8-21 Industrial Codes and Standards Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
OM3	1987	Requirements for preoperational and Initial Startup Vibration Test Program for Water-Cooled Power Plants
OM7	1986	Requirements for Thermal Expansion Testing of Nuclear Plant Piping Systems (September 1986 (Draft-Revision 7))
American Petroleum Institute (API)		
620*	1986	Rules for Design and Construction of Large, Welded, Low-Pressure Storage Tanks
650*	1980	Welded Steel Tanks for Oil Storage
American Society of Heating, Refrigerating and Air-Conditioning Engineers, Inc. (ASHRAE)		
30	1978	Methods of Testing Liquid Chilling Packages
33	1978	Methods of Testing Forced Circulation Air Cooling and Air Heating Coils
American Society of Mechanical Engineers (ASME)		
AG-1*	1988 1991	Code on Nuclear Air and Gas Treatment
B30.2*	1983	Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Grider, Top Running Trolley Hoist)
B30.9*	1984	Slings
B30.10*	1982	Hooks
B30.11*	1980	Monorails and Underhung Cranes
B30.16*	1981	Overhead Hoists
B31.1*	1986	Power Piping
B96.1*	1986	Specification for Welded Aluminum-Alloy Storage Tanks
N45.4	1972	Leakage-Rate Testing of Containment Structures for Nuclear Reactors
N509*	1989	Nuclear Power Plant Air-Cleaning Units and Components
N510*	1989	Testing of Nuclear Air-Cleaning Systems
NQA-1*	1983	Quality Assurance Program Requirements for Nuclear Facilities
NQA-1A*	1983	Addenda to ANSI/ASME NQA-1-1983
Sec II	1989	BPVC Section II, Material Specifications
Sec III	1989	BPVC Section III, Rules for Construction of Nuclear Power Plant Components

Table 1.8-21 Industrial Codes and Standards Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
IEC 880	1986	Software for Computers in the Safety Systems of Nuclear Power Stations
OSHA 1910.179	1990	Overhead and Gantry Cranes
TEMA C	1978	Standards of Tubular Exchanger Manufacturers Association
TOP	1990	Test Operating Procedure, Part 1, Department of Defense
UL-44*	1983	Rubber-Insulated Wires and Cables
UL-489*	1991	Molded-Case Circuit Breakers and Circuit Breaker Enclosures
UL-845*	1988	Standard for Safety Motor Control Centers - Low Voltage Circuit Breakers
--	--	Crane Manufacturers Association of America, Specification No. 70
--	--	Aluminum Construction Manual by Aluminum Association
NCIG-01	Rev. 2	Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants

* Also an ANSI code (i.e. ANSI/ASME, ANSI/ANS, ANSI/IEEE etc.).

† ANSI, ANSI/ANS, ANSI/ASME, and ANSI/IEEE codes are included here. Other codes that approved by ANSI and another organization are listed under the latter.

‡ As modified by NRC accepted alternate positions to the related Regulatory Guide and identified in Table 2-1 of Reference 1 to Chapter 17.

• ERDA 76-21

• "Testing of Ventilation Systems," Section 9
of "Industrial Ventilation System"

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
81-07	5/14/81	Control of Radioactivity Contaminated Material	COL Applicant
81-08	5/29/81	Foundation Materials	COL Applicant
81-09	7/10/81	Containment Effluent Water	
81-11	7/24/81	Inadequate Decay Heat Removal	COL Applicant
81-13	9/25/81	Torque Switch Electrical Bypass Circuit	COL Applicant
81-14	11/5/81	Main Steam Isolation Valve Failures to Close	COL Applicant
NUREG			
0313 Rev. 2	6/88	Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping	
0371	10/78	Task Action Plans for Generic Activities Category A	
0471	6/78	Generic Task Problem Description: Category B, C & D Tasks	
0578	9/80	Performance Testing of BWR and PWR Relief and Safety Valves.	
0588	12/79	Interim Staff Position On Environmental Qualification of Safety-Related Electrical Equipment	
0619	4/80	BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking	
0626	1/80	Generic Evaluation of Feedwater Transients and Small Break LOCA in GE-Designed Operating Plants and Near-Term Operating License Applications	
0660	5/80	NRC Action Plan Developed as a Result of the TMI-2 Accident	
0661 Supp. 1	8/82	Safety Evaluation Report - Mark I Containment Long-Term Program - Resolution of Generic Technical Activity A-7	Subsection 19B.2.3
0710 Rev. 1	6/81	Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License	
0737 Supp.1	12/82	Clarification of TMI Action Plan Requirements	
0744 Rev. 1	10/82	Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue	
→ 0800	8/81	Mark II Containment Program Load Evaluation and Acceptance Criteria	
0813	9/81	Draft Environmental Statement: Related to the Operation of Calloway Plant, Unit No. 1	

JRP for the review of SAR for Nuclear Power Plants

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
0977	3/83	NRC Fact-Finding Task Force Report on the ATWS Events at the Salem Nuclear Generating Station, Unit 1, on February 22 and 25, 1983	
0933	4/93	<i>A prioritization of Generic Safety Issues</i>	
1150	6/89	Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, Vol. 1 & 2	
1161	5/80	Recommended Revisions to USNRC-Seismic Design Criteria	Subsection 19B.2.14
1174	5/89	Evaluation of Systems Interactions in Nuclear Power Plants	Subsection 19B.2.59
1212	6/86	Status of Maintenance in the US Nuclear Power Industry, 1985, Vol. 1, 2	
1216	8/86	Safety Evaluation PP2 Related to Operability and Reliability of Emergency Diesel Generators	
1217	4/88	Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plants-Technical Findings Related to USI A-47	Subsection 19B.2.17
1218	4/88	Regulatory Analysis for Proposed Resolution of USI A-47	Subsection 19B.2.17
1229	8/89	Regulatory Analysis for Resolution of USI A-17	Subsection 19B.2.59 & 19B.2.14
1233	9/89	Regulatory Analysis for USI A-40	Subsection 19B.2.14
1273	4/88	Containment Integrity Check-Technical Finds Regulatory Analysis	
1296	2/88	Peer Review of High Level Nuclear Waste	
1341	5/89	Regulatory Analysis for Resolution of Generic Issue 115, Enhancement	
1353	4/89	Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools"	Subsection 19B.2.63
1370	9/89	Resolution of USI A-48	Subsection 19B.2.18
1275	2/91	Volume 6, Operating Experience Feedback Report Solenoid Operated Valve Problems	
1339	6/90	Resolution of Generic Safety Issue 29: Bolting Degradation of Failure in Nuclear Power Plants	Subsection 19B.2.62
CR-3922	1/85	Survey and Evaluation of System Interaction Events and Sources, Vol. 1, 2	Subsection 19B.2.59

*New
mark-up*

Safety Issues Index (Continued)

Title	NRC Priority	SSAR Subsection
I.C.8 Pilot-Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	Resolved	COL App.
I.D.1 Control Room Design Reviews	Resolved	1A.2.2
I.D.2 Plant Safety Parameter Display Console	Resolved	1A.2.3
I.D.3 Safety System Status Monitoring	Medium	19A.2.17
I.D.5(2) Plant Status and Post-Accident Monitoring	Resolved	19B.2.65
I.D.5(3) On-Line Reactor Surveillance System	Near Res.	19B.2.66
I.F.2(2) Include QA Personnel in Review and Approval of Plant Procedures	Resolved	19A.2.43
I.F.2(3) Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	Resolved	19A.2.43
I.F.2(6) Increase the Size of Licensees' QA Staff	Resolved	19A.2.43
I.F.2(9) Clarify Organizational Reporting Levels for the QA Organization	Resolved	19A.2.43
I.G.1 Training Requirements	Resolved	1A.2.4
I.G.2 Scope of Test Program	Resolved	19B.2.67
II.B.1 Reactor Coolant System Vents	Resolved	1A.2.5
II.B.2 Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	Resolved	COL App. 1A.2.6 ←
II.B.3 Post-Accident Sampling	Resolved	1A.2.7
II.B.4 Training for Mitigating Core Damage	Resolved	COL App.
II.B.8 Rulemaking Proceeding on Degraded Core Accidents	Resolved	19A.2.1
II.D.1 Testing Requirements	Resolved	1A.2.9
II.D.3 Relief and Safety Valve Position Indication	Resolved	1A.2.10
II.E.1.3 Update Standard Review Plan and Develop Regulatory Guide	Resolved	COL App.
II.E.4.1 Dedicated Penetrations	Resolved	1A.2.13
II.E.4.2 Isolation Dependability	Resolved	1A.2.14
II.E.4.4 Purging	Resolved	19A.2.27
II.E.6.1 Test Adequacy Study	Resolved	19B.2.68 COL App.
II.F.1 Additional Accident Monitoring Instrumentation	Resolved	1A.2.15
II.F.2 Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	Resolved	1A.2.16

*New
Mark-ups*

Safety Issues Index (Continued)

Title	NRC Priority	SSAR Subsection
II.F.3 Instruments for Monitoring Accident Conditions	Resolved	1A.2.17
II.J.4.1 Revise Deficiency Reporting Requirements	Resolved	COL App.
II.K.1(5) Safety-Related Valve Position Description	Resolved	1A.2.18 18.8.7
II.K.1(10) Review and Minimize Procedures for Removing Safety-Related Systems from Service	Resolved	1A.3.2
II.K.1(13) Propose Technical Specifications Changes Reflecting Implementation of All Bulletin Items	Resolved	19B.2.69
II.K.1(22) Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	Resolved	1A.2.20
II.K.1(23) Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	Resolved	1A.2.21
II.K.3(3) Report Safety and Relief Valve Failures Promptly and Challenges Annually	Resolved	1A.3.4
II.K.3(11) Control Use of PORV Supplied by Control Components, Inc. Until Further Review Complete	Resolved	19B.2.70
II.K.3(13) Separation of HPCI and RCIC System Initiation Levels	Resolved	1A.2.22
II.K.3(15) Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	Resolved	1A.2.23 COL App. ←
II.K.3(16) Reduction of Challenges and Failures of Relief Valves—Feasibility Study and System Modification	Resolved	1A.2.24
II.K.3(17) Report and Outage of ECC Systems—Licensee Report and Technical Specification Changes	Resolved	1A.2.25
II.K.3(18) Modification of ADS Logic—Feasibility Study and Modification for Increased Diversity for Some Event Sequences	Resolved	1A.2.26
II.K.3(21) Restart of Core Spray and LPCI Systems on Low Level-Design and Modification	Resolved	1A.2.27
II.K.3(22) Automatic Switchover of RCIC System Suction—Verify Procedures and Modify Design	Resolved	1A.2.28
II.K.3(24) Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	Resolved	1A.2.29
II.K.3(25) Effect of Loss of AC Power on Pump Seals	Resolved	1A.2.30
II.K.3(27) Provide Common Reference Level for Vessel Level Instrumentation	Resolved	1A.2.21
II.K.3(28) Study and Verify Qualification of Accumulators on ADS Valves	Resolved	1A.2.31

Table 3.1.4-1
Control Rod Scram Times

NOTES

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."
2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod Operability," for control rods with scram times > [] seconds to 60% rod insertion position, are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow." These control rods

ROD POSITION PERCENT INSERTION (%)	SCRAM TIMES(a) (seconds)		
	REACTOR STEAM DOME PRESSURE(b) 0 Kg/cm ² g	REACTOR STEAM DOME PRESSURE(b) 66.8 Kg/cm ² g	REACTOR STEAM DOME PRESSURE(b) 73.8 Kg/cm ² g
10	(c)	[]	[]
40	(c)	[]	[]
60		[]	[]

- (a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids as time zero.
- (b) For intermediate reactor steam dome pressures, the scram time criteria are determined by linear interpolation.
- (c) For reactor steam dome pressure ≤ 66.8 Kg/cm²g, only 60% rod insertion position scram time limit applies.