1.8 Conformance with Standard Review Plan and Applicability of Codes and Standards

1.8.1 Conformance with Standard Review Plan

This subsection provides the information required by 10CFR50.34(h) showing conformance with the Standard Review Plan (SRP) in effect six months prior to the docket date of the application (September 29, 1987). The summary of differences from the SRP section is presented by SRP section in Tables 1.8-1 through 1.8-18. (See Subsection 1.8.4.1 for COL license information.)

Table 1.8-18a shows conformance with the SRP in effect six months prior to the docket date of the renewal for the Tier 1, Tier 2*, and related Tier 2 information changed from the original design certification rule's Design Control Document.

SRP Chapter 19 did not exist at the time of the initial ABWR design certification. The probabilistic risk assessment (PRA) performed for the initial certification generally met the intent of the current SRP, although the PRA analysis used the methodology and guidance applicable at the time. The limited update of the PRA (see 19.2.3) for the renewal is consistent with the original PRA analysis, but provides a more modern treatment of select areas of the analysis.

1.8.2 Applicability of Codes and Standards

Standard Review Plans, Branch Technical Positions, Regulatory Guides and Industrial Codes and Standards which are applicable to the ABWR design are provided in Tables 1.8-19, 1.8-20 and 1.8-21. Applicable revisions are also shown.

1.8.3 Applicability of Experience Information

1.8.3.1 Operating Experience Information for Initial ABWR Design Certification

Experience information was routinely made available and distributed to design personnel in the design process. Nuclear field experience was maintained in hard copy form in functional component and library files and in the GE world-wide computer retrieval system.

Generic Letters and IE Bulletins, Information Notices and Circulars covering the decade including 1980 through late 1991 were reviewed for applicability to the ABWR design. The review was enhanced by associating related experiences and tracing referenced occurrences. This was accomplished starting with the then current issues of the Generic Letters and proceeding back into the decade. The Circulars, Bulletins and Notices were reviewed in that order. Interfacing experience was included in the review. The selection of ABWR information was based on the significance to future design and operation guidance. Included is a list of NUREGs related to the closing of current safety issues. Experience that resulted in applicable rules, codes and standards was not repeated. Table 1.8-22 lists the experience information that

has been included in the ABWR design or impacts the COL applicant. (See Subsection 1.8.4.2 for COL license information.)

A systematic procedure encompassing available resources was used to identify the applicability of experience information resulting in Table 1.8-22. Engineering management surveyed the indices of annual experience information to identify those very likely to be applicable to the ABWR. The remaining potentially applicable experiences were reviewed individually. Experience information not deemed applicable to the ABWR design (issues pertaining to other reactor types, scram discharge volume, etc.) were not included in Table 1.8-22. The experience information categories applicable to the ABWR design in Table 1.8-22 include experience information accommodated by a design change, covered by review of USIs/GSIs or an issue that impacts the ABWR design but must be addressed by the COL applicant. This latter category is included as COL license information.

Experiences related to identified regulatory or industry developed resolutions were eliminated to avoid repetition except for selected experiences that have a nuisance potential for reoccurring. Lead system engineers classified the more complex experiences.

Reference to the new or novel design features used in the ABWR are provided below:

Feature	Tier 2 Section
Fine Motion Control Rod Drive	4.6
Internal Reactor Pumps	5.4.1
Multiplexing	7A.2
Digital/Solid-State Control	7A.7
Overpressure Protection System	6.2.5.2.6, 6.2.5.3, 6.2.5.4
AC-Independent Water Addition System	5.4.7.1.1.10
Lower Drywell Flooder	9.5.1.2
Alternate Feedwater Injection	9.5.14

1.8.3.2 Operating Experience Information for the ABWR Design Certification Renewal

NRC Generic Letters, Information Notices, Bulletins and NUREGs covering the time period of January 1992 through May 2010 were reviewed for applicability of operating experience to the Renewal of the ABWR design certification.

When applicable to the ABWR, the review determined if the operating experience was related to the design, construction, testing and/or operation phase of the ABWR. The relative safety significance (high, medium or low) was also assessed. Table 1.8-23 summarizes high safety significant experience related to the ABWR design that has been included in the ABWR design or impacts the COL applicant. Note that in some cases where COL applicant is indicated in Table 1.8-23, the issue has been addressed in the ABWR design and additional requirements are imposed on the COL applicant such as a requirement to provide material test results. (See Subsection 1.8.4.2 for COL license information.)

1.8.3.3 Japanese Operating Experience Information for the ABWR Design Certification Renewal

The ABWR was developed as an evolutionary design, incorporating U.S., European and Japanese BWR experience. Toshiba has been engaged in the development, design, construction and operation of boiling water reactors since 1966. Toshiba experience is shown in Table 1.4-1. Toshiba has been engaged in the design and construction of three ABWRs currently in operation in Japan: Kashiwazaki-Kariwa 6 and 7 since the mid-1990s, and Hamaoka-5 since 2005.

The Japan Nuclear Energy Safety Organization (JNES) is a technical support organization for Japan's Nuclear and Industrial Safety Agency (NISA). JNES collects information and issues reports that describe operating events and failures that occur at commercial nuclear power plants and are reportable to NISA under Japanese law ("the Act for the Regulation on Nuclear Source Material, Nuclear Fuel Material and Reactors" and "the Electric Utilities Industry Law"). The JNES reports are publicly available on its website.

Information in Table 1.8-24 is cited from the JNES website and the table summarizes the reportable events that occurred at Japanese ABWRs during the time period from January 1997 through May 2010. The table includes events from all four of the ABWR units that were operating during this time period - Hamaoka-5, Kashiwazaki-Kariwa 6 and 7, and Shika-2. Toshiba reviewed the JNES reports for these ABWR events to determine the impact of each event on the standard ABWR design, including the need for design change(s) to address lessons learned and prevent reoccurrence. In the table, the entries in columns "Facility," "Date," "Title" and "Description" are based on information taken from reports on the JNES website. The entries in the column "Toshiba Comment" indicate whether the event was considered to have a design impact. Based on this review, Toshiba concluded that none of the listed events impacted the ABWR design and no changes to the standard design are required.

Toshiba also reviewed the JNES reports for operating events and failures that occurred at Japanese BWRs and PWRs during the same time period. Based on this review, Toshiba concluded that only one of the events (hydrogen gas accumulation and combustion at Hamaoka Unit 1 in November 2001) impacted the ABWR design. This event was reported by the NRC in Information Notice 2002-15 and evaluated in NRC GI-195, “Hydrogen Combustion in Foreign BWR Piping.” To address this issue, the renewal of the ABWR Design Certification adds to the Reactor Water Cleanup System a vent line for the reactor pressure vessel head spray line in order to preclude the accumulation of hydrogen gas in the head spray line (see Figures 5.1-3 and 5.4-12). Toshiba concluded that none of the other BWR and PWR events impacted the ABWR design and no other changes to the standard design are required.

1.8.4 COL License Information

1.8.4.1 SRP Deviations

The SRP sections to be addressed by the COL applicant are indicated in the comments column of Table 1.8-19 as “COL Applicant”. Where applicable the COL applicant will provide the information required by 10CFR50.34(h) similar to Tables 1.8-1 through 1.8-18 (see Subsection 1.8.1).

1.8.4.2 Experience Information

The experience information to be addressed by the COL applicant is indicated in the comment column of Table 1.8-22 and Table 1.8-23 as “COL Applicant” (see Subsection 1.8.3).

Table 1.8-1 Summary of Differences from SRP Section 1

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

Table 1.8-2 Summary of Differences from SRP Section 2

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
2.2.1-2.2.2 2.2.3 2.3.1 2.3.4 2.4.1 2.4.4 2.4.5 2.4.6 2.4.8 2.4.11.6 2.4.12	See Table 2.1-1.	Limits imposed on selected SRP Section II acceptance criteria by (1) the envelope of the ABWR Standard Plant site parameters and (2) evaluations assumptions.	2.1
2.5.2.7		OBE is not a design requirement.	

Table 1.8-3 Summary of Differences from SRP Section 3

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
3.6.1 and 3.6.2	ASB 3-1 and MEB 3-1 Consider 1/2 SSE for postulating pipe ruptures.	Earthquake stresses considered only in cumulative usage factor calculations when postulating pipe ruptures	3.6.1.1, 3.6.2.1.4.2, 3.6.2.1.4.3, 3.6.2.1.4.4., 3.6.2.1.5.2, 3.6.2.1.5.3
3.6.2	MEB 3-1, B.1.c.(1).(b) - Pipe ruptures must be postulated if Eq.(10) of NB-3653 exceeds 2.4 Sm.	Pipe ruptures postulated only if, in NB-3653, Eq. (10) and either (12) or Eq. (13) exceed 2.4 Sm.	3.6.1.1 and 3.6.2.1.4.3
3.7.1 and 3.7.3	II.2 - Two earthquakes, the SSE and the OBE shall be considered in the design.	The ABWR will be based on a single earthquake (SSE) design.	3.6, 3.7, 3.9
3.9.2	II-E.2.g - For multiply supported equipment use envelope RS and;	Independent Support Motion Response Spectrum methods acceptable for use.	3.7.3.8.1.10
	Combine responses from inertia effects with anchor displacements by Absolute Sum.	Combine responses from inertia effects with anchor displacements by SRSS.	3.7.3.8.1.8
3.7.3	II.2.b—Determination of number of OBE cycles.	The ABWR is based on a single earthquake (SSE) design, two SSE events with 10 peak stress cycles per event are used.	3.7.3.2

Table 1.8-4 Summary of Differences from SRP Section 4

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

Table 1.8-5 Summary of Differences from SRP Section 5

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
5.2.3	II.3.b.(3)—Reg Guide 1.71, Welding Qualification for Areas of Limited Accessibility.	Alternate position employed.	5.2.3.4.2.3
5.2.4	II.1—Inspection of Class 1 pressure-containing components.	Some welds inaccessible for volumetric examination.	5.2.4.2.2
5.4.6	Deleted		
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	6.2.1.1.5.6.1
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
6.2.1.1	Monthly vacuum valve operability test	Operability tests only performed during refueling outages	6.2.1.1.5.6.3

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: 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 remote 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
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) Appendix 9A
8.1	Table 8-1: 2f Reg Guide 1.75	4.572 m cable marking intervals.	8.3.3.5.1.3
8.1	Table 8-1: 2f IEEE-384	LDS divisional separation in steam tunnel.	8.3.3.6.1.2 (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 micrometer.	Instrument air is filtered to 5 micrometer.	9.3.6.2
9.3.2	II.k.5—Capable of sampling liquid of 370,000 MBq/cm^3.	Capable of sampling liquids of 37,000 MBq/cm^3.	9.3.2.3.1
9.4.1	GDC 19	Site specific.	6.4.7.1
9.5.1	Section 7.b	Control Room Complex 1. Peripheral rooms 2. Underfloor (subfloor) 3. Consoles & cabinets	9.5.1
	Section 7.j	Diesel fuel oil tank capacity	9.5.1
	Section 7.i	Diesel Generator operation	9.5.1

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 1.8-11 Summary of Differences from SRP Section 11

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
11.1	II.9—BWR GALE Code	Alternate computer code.	20.3.7 (Response to Question 460.1)

Table 1.8-12 Summary of Differences from SRP Section 12

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

Table 1.8-13 Summary of Differences from SRP Section 13

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

Table 1.8-14 Summary of Differences from SRP Section 14

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

Table 1.8-15 Summary of Differences from SRP Section 15

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
15.1.1-15.1.4	Acceptable analytical model.	3D simulator instead of REDY Code is used.	15.1.1.3.2
15.2.6	All recirculation pumps are tripped simultaneously by the initiating event.	Only four of ten RIPs are tripped. This is based on ABWR design.	15.2.6.1.1
15.3.1-15.3.2	Complete recirculation pumps trip is considered as a moderate-frequency transient.	Trip of all RIPs is classified as an infrequent low probability event with special acceptance for fuel failure.	15.3.1.1.2
15.3.3-15.3.4	II.10—coincident turbine trip, loss of offsite power and coastdown of undamaged pumps.	Not analyzed with the assumption. If the assumption is made, the consequence would be similar to the event shown in 15.2.6.	15.3.3.2.2.
15.4.2	Analysis of uncontrolled control rod withdrawal at power.	No quantitative analysis is provided because ABWR's ATLM design prevents this transient from occurring.	15.4.2.2
15.4.4-15.4.5	II.2.(b)—Fuel cladding integrity.	MCPR not calculated, since transients are very mild.	15.4.4.3 15.4.5.3.2.1 and 15.4.5.3.2.2
15.4.9	Not applicable SRP for BWR.	Discussion is provided to show this event cannot occur with ABWR FMCRD design.	15.4.9
15.4.10	Analysis of rod drop accidents.	No quantitative analysis is provided because ABWR's FMCRD design prevents this accident from occurring.	15.4.10.1 & 15.4.10.2
15.6.5	II.2—Use of assumptions outlined in Reg Guide 1.3.	ABWR LOCA analysis incorporates suppression pool scrubbing IAW SRP 6.5.5 and in variance from R.G. 1.3. Fission product plateout and removal is incorporated in the analysis of leakage sources through the main steamlines and into the turbine condenser based upon BWROG analysis of acceptability of the steamlines and condenser to mitigate releases without requiring Seismic Category I structures.	15.6.5.5

Table 1.8-16 Summary of Differences from SRP Section 16

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

Table 1.8-17 Summary of Differences from SRP Section 17

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
17.1	II.1 - Applicant is responsible for overall QA program.	Toshiba & its major technical associates are responsible for their own QA programs. GE & its major technical associates were responsible for their own QA programs during preparation of the initial ABWR design certification.	17.0 17.1.1 17.1.2
17.1	II.2,3,4,7,13,17,18-Meet identified quality related Reg Guides.	Reg Guide 1.28, Rev. 3 and alternative positions employed.	Table 17.0-1 17.1.2, 17.1.3 17.1.4, 17.1.7 17.1.13, 17.1.17 17.1.18
17.1	II.2 - Meet identified regulations and codes.	Differences between domestic and international designs are identified in a controlled list.	17.1.3

Table 1.8-18 Summary of Differences from SRP Section 18

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

Table 1.8-18a Conformance of Tier 1, Tier 2* , and Related Tier 2 Section Renewal Changes with the March 2007 SRP[1]

Subject	Affected Sections	Applicable SRP Sections	Conformance with Applicable SRP/ Justification for Differences
RIP Motor Casing Cladding	Tier 1 Section 2.1 Tier 2 Section 5.3	5.3.1 5.3.3	Conforms with applicable SRP sections
Control System Changes	Tier 1 Section 2.2 Tier 1 Section 2.15 Tier 2 Section 7.2 Tier 2 Section 7.6 Tier 2 Section 7.7 Tier 2 Section 10.1 Tier 2 Section 16.3.3.1.1 Tier 2 Section 16B.3.3.1.1	1.0 App 7.0-A 7.1 7.2 7.3 7.6 7.7	Conforms with applicable SRP sections
Hi Rad MSIV Closure	Tier 1 Section 2.3 Tier 1 Section 2.7 Tier 2 Section 1.2 Tier 2 Appendix 1A Tier 2 Section 3.4 Tier 2 Section 5.2 Tier 2 Section 7.1 Tier 2 Section 7.2 Tier 2 Section 7.3 Tier 2 Section 7.5 Tier 2 Section 15.2 Tier 2 Section 16.3.3.1.1 Tier 2 Section 16.3.3.6.1 Tier 2 Section 16B.3.3.1.1 Tier 2 Section 16B.3.3.6.1	1.0 3.4.1 5.2.5 7.1 Table 7.1 App 7.1-A 7.2 7.3 7.5 11.5 13.3 15.2.1-15.2.5 16.0	Conforms with applicable SRP sections
RHR Spent Fuel Pool Cooling	Tier 1 Section 2.4 Tier 1 Section 2.6 Tier 2 Appendix 1AA Tier 2 Section 3.1 Tier 2 Section 3.9 Tier 2 Appendix 3MA Tier 2 Section 5.4 Tier 2 Section 6.3 Tier 2 Section 6.6 Tier 2 Section 7.3 Tier 2 Section 7.4 Tier 2 Section 9.1 Tier 2 Appendix 19L Tier 2 Appendix 19Q	3.2.1 3.2.2 5.4 5.4.7 6.3 6.6 7.3 7.4 9.1.1 9.1.2 9.1.3 9.1.4 9.1.5 13.4 19.1 19.2	Conforms with applicable SRP sections

Table 1.8-18a Conformance of Tier 1, Tier 2* , and Related Tier 2 Section Renewal Changes with the March 2007 SRP[1]

Subject	Affected Sections	Applicable SRP Sections	Conformance with Applicable SRP/ Justification for Differences
FW Line Break Mitigation	Tier 1 Section 2.4 Tier 2 Section 5.2 Tier 2 Section 6.2 Tier 2 Section 7.3 Tier 2 Section 8.1 Tier 2 Section 8.3 Tier 2 Section 10.2 Tier 2 Appendix 19L Tier 2 Section 16.3.3.1.1 Tier 2 Section 16.3.3.1.4 Tier 2 Section 16B.3.3.1.1 Tier 2 Section 16B.3.3.1.4	3.2.2 3.6.2 3.6.3 3.8.3 3.11 5.2.2 5.2.3 5.2.4 5.2.5 6.2.1.1C 6.2.1.2 6.2.1.3 6.2.1.4 6.2.2 6.2.3 6.2.4 6.5.2 6.6 9.2.2 9.2.3 10.2 16.0	Conforms with applicable SRP sections
RCIC Pump	Tier 1 Section 2.4 Tier 2 Appendix 1A Tier 2 Section 3.2 Tier 2 Section 3.9 Tier 2 Appendix 3B Tier 2 Appendix 3MA Tier 2 Section 5.4 Tier 2 Section 6.2 Tier 2 Section 7.3 Tier 2 Section 14.2 Tier 2 Section 16.3.3.1.1 Tier 2 Section 16.3.3.1.4 Tier 2 Section 16B.3.3.1.1 Tier 2 Section 16B.3.3.1.4 Tier 2 Section 19.3 Tier 2 Section 19.9 Tier 2 Section 19.11 Tier 2 Section 19.13 Tier 2 Appendix 19K Tier 2 Appendix 19M	3.2.2 5.4 5.4.6 5.4.7 6.2.1.2 6.2.1.3 6.2.1.4 6.2.1.5 6.2.2 6.2.4 7.3 14.2 16.0 16.1 19.1 19.2	Conforms with applicable SRP sections

Table 1.8-18a Conformance of Tier 1, Tier 2* , and Related Tier 2 Section Renewal Changes with the March 2007 SRP[1]

Subject	Affected Sections	Applicable SRP Sections	Conformance with Applicable SRP/ Justification for Differences
New Fuel Storage Racks	Tier 1 Section 2.5 Tier 2 Section 1.2 Tier 2 Section 3.1 Tier 2 Section 9.1 Tier 2 Section 12.3 Tier 2 Section 16.4	9.1.1 9.1.2	Conforms with applicable SRP Sections
Addition of Condensate Booster Pumps	Tier 1 Section 2.10	10.4.7	Conforms with applicable SRP sections
Breaker/Fuse Coordination	Tier 1 Section 2.12	14.3 14.3.6	Conforms with applicable SRP sections
I&C Power Division	Tier 1 Section 2.12 Tier 2 Section 8.1 Tier 2 Section 8.3	App 7.1-C	Conforms with applicable SRP section

Table 1.8-18a Conformance of Tier 1, Tier 2* , and Related Tier 2 Section Renewal Changes with the March 2007 SRP[1]

Subject	Affected Sections	Applicable SRP Sections	Conformance with Applicable SRP/ Justification for Differences
Hydrogen Recombiner Elimination	Tier 1 Section 2.2 Tier 1 Section 2.3 Tier 1 Section 2.4 Tier 1 Section 2.7 Tier 1 Section 2.11 Tier 1 Section 2.14 Tier 1 Section 2.15 Tier 2 Section 1.2 Tier 2 Appendix 1A Tier 2 Appendix 1AA Tier 2 Section 3.2 Tier 2 Section 3.9 Tier 2 Appendix 3I Tier 2 Appendix 3MA Tier 2 Section 5.2 Tier 2 Section 5.4 Tier 2 Section 6.2 Tier 2 Section 6.5 Tier 2 Section 6.6 Tier 2 Section 7.1 Tier 2 Section 7.3 Tier 2 Section 7.4 Tier 2 Section 7.5 Tier 2 Section 7.6 Tier 2 Section 9.2 Tier 2 Section 9.4 Tier 2 Appendix 9A Tier 2 Section 14.2 Tier 2 Appendix 15A Tier 2 Section 16.3.3.6.1 Tier 2 Section 16.3.3.6.2 Tier 2 Section 16.3.6.3.1 Tier 2 Section 16.3.6.3.2 Tier 2 Section 16.5.0 Tier 2 Section 16B.3.3.6.1 Tier 2 Section 16B.3.3.6.2 Tier 2 Section 16B.3.6.3.1 Tier 2 Section 16B.3.6.3.2 Tier 2 Appendix 18A Tier 2 Appendix 18B Tier 2 Appendix 18F	3.2.2 5.4 5.4.6 5.4.7 6.5.2 6.5.3 6.5.5 7.1 7.3 7.4 7.5 7.6 16.0 16.1	Conforms with applicable SRP sections

Table 1.8-18a Conformance of Tier 1, Tier 2* , and Related Tier 2 Section Renewal Changes with the March 2007 SRP[1]

Subject	Affected Sections	Applicable SRP Sections	Conformance with Applicable SRP/ Justification for Differences
Hydrogen Recombiner Elimination (continued)	Tier 2 Appendix 18H Tier 2 Appendix 19A		
Radwaste Building Substructure Seismic Classification	Tier 1 Section 2.15 Tier 2 Section 2.0 Tier 2 Section 2.4S.13 Tier 2 Section 2.5S.4 Tier 2 Section 3.1 Tier 2 Section 3.2 Tier 2 Section 3.3 Tier 2 Section 3.4 Tier 2 Section 3.7 Tier 2 Section 3.8 Tier 2 Appendix 3C Tier 2 Appendix 3H Tier 2 Section 11.4 Tier 2 Section 12.2 Tier 2 Section 14.3 Tier 2 Section 15.7 Tier 2 Section 19.4 Tier 2 Appendix 19H Tier 2 Section 21.0 Tier 2 Section 21.1	2.0 2.4.13 2.5.4 3.2.1 3.2.2 3.3.1 3.3.2 3.4.1 3.7.1 3.7.2 3.7.3 3.8.3 3.8.4 11.4 12.2 14.3 15.7.3 15.7.4 15.7.5 19.0	Conforms with applicable SRP sections
RBSRDG HVAC	Tier 1 Section 2.15 Tier 2 Section 9.4	9.4.1 9.4.3 9.4.4	Conforms with applicable SRP sections

Table 1.8-18a Conformance of Tier 1, Tier 2* , and Related Tier 2 Section Renewal Changes with the March 2007 SRP[1]

Subject	Affected Sections	Applicable SRP Sections	Conformance with Applicable SRP/ Justification for Differences
Safety-Related I&C Architecture	Tier 1 Section 2.2 Tier 1 Section 2.7 Tier 1 Section 3.4 Tier 2 Section 1.2 Tier 2 Section 3.2 Tier 2 Section 6.2 Tier 2 Section 7.1 Tier 2 Section 7.2 Tier 2 Section 7.3 Tier 2 Section 7.4 Tier 2 Section 7.6 Tier 2 Section 7.7 Tier 2 Appendix 7A Tier 2 Appendix 7C Tier 2 Section 10.1 Tier 2 Section 10.4 Tier 2 Section 11.5 Tier 2 Section 12.3 Tier 2 Section 14.2 Tier 2 Section 15.0 Tier 2 Section 15.1S Tier 2 Appendix 15B Tier 2 Appendix 15E Tier 2 Section 16.1.0 Tier 2 Section 16B.3.3.3.1 Tier 2 Section 16B.3.3.4.1 Tier 2 Section 16B.3.3.5.1 Tier 2 Section 16B.3.3.6.1 Tier 2 Section 16B.3.3.6.2 Tier 2 Section 16B.3.8.4 Tier 2 Section 18.4 Tier 2 Section 18.6 Tier 2 Section 18.8 Tier 2 Appendix 18C Tier 2 Appendix 18E Tier 2 Section 19.3 Tier 2 Section 19.9 Tier 2 Section 19.11 Tier 2 Appendix 19K Tier 2 Appendix 19L Tier 2 Appendix 19N Tier 2 Appendix 19Q	3.2.1 3.2.2 6.2.4 App 7.0-A 7.1 7.2 7.3 7.4 7.6 7.7 7.9 10.3 10.4.5 10.4.7 11.5 12.3-12.4 14.2 15.8 16.0 18.0 19.0	Conforms with applicable SRP sections

Table 1.8-18a Conformance of Tier 1, Tier 2* , and Related Tier 2 Section Renewal Changes with the March 2007 SRP[1]

Subject	Affected Sections	Applicable SRP Sections	Conformance with Applicable SRP/ Justification for Differences
Site Parameters	Tier 1 Section 5.0 Tier 2 Section 2.2 Tier 2 Section 3.1 Tier 2 Section 3.4 Tier 2 Appendix 3H Tier 2 Section 9.4 Tier 2 Section 19.3 Tier 2 Section 19.9 Tier 2 Section 19.11 Tier 2 Section 19.13 Tier 2 Appendix 19K Tier 2 Appendix 19Q Tier 2 Appendix 19R	2.0 2.2.1-2.2.2 2.4.1 2.4.2 2.4.3 2.4.4 2.4.10 3.4.1 3.4.2 6.4 6.5.1 9.2.1 9.2.5 9.4.1 9.4.2 9.4.3 9.4.4 9.4.5	Conforms with applicable SRP sections Conforms with FSER 19.1.3.3.4
Codes, Standards, and RGs	Tier 2 Section 1.8 Tier 2 Section 5.2 Tier 2 Appendix 7A	1.8 3.2.1 3.2.2 3.8.1 3.8.3 3.8.4 3.8.5 3.12 5.2.3 5.2.4 5.2.5 5.3.1 5.3.2	Conforms with applicable SRP sections

1 Tier 2 information changed in renewal and not related to the Table's Tier 1 and Tier 2* sections is not included.

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

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
Chapter 1 Introduction and General Description of Plant					
1.8	Interfaces for Standard Design	1	7/81	Yes	
Chapter 2 Site Characteristics					
2.1.1	Site Location and Description	2	7/81	—	COL Applicant
2.1.2	Exclusion Area Authority and Control	2	7/81	—	COL Applicant
2.1.3	Population Distribution	2	7/81	—	COL Applicant
2.2.1–2.2.2	Identification of Potential Hazards in Site Vicinity	2	7/81	—	COL Applicant
2.2.3	Evaluation of Potential Accidents	2	7/81	—	COL Applicant
2.3.1	Regional Climatology	2	7/81	—	COL Applicant
2.3.2	Local Meteorology	2	7/81	—	COL Applicant
2.3.3	Onsite Meteorological Measurements Programs	2	7/81	—	COL Applicant
	Appendix A	2	7/81	—	COL Applicant
2.3.4	Short-Term Diffusion Estimates for Accidental Atmospheric Releases	1	7/81	—	COL Applicant
2.3.5	Long-Term Diffusion Estimates	2	7/81	—	COL Applicant
2.4.1	Hydrologic Description	2	7/81	—	COL Applicant
	Appendix A	2	7/81	—	COL Applicant
2.4.2	Floods	2	7/81	—	COL Applicant
2.4.3	Probable Maximum Flood (PMF) on Streams and Rivers	2	7/81	—	COL Applicant

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

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
2.4.4	Potential Dam Failures	2	7/81	—	COL Applicant
2.4.5	Probable Maximum Surge and Seiche Flooding	2	7/81	—	COL Applicant
2.4.6	Probable Maximum Tsunami Flooding	2	7/81	—	COL Applicant
2.4.7	Ice Effects	2	7/81	—	COL Applicant
2.4.8	Cooling Water Canals and Reservoirs	2	7/81	—	COL Applicant
2.4.9	Channel Diversions	2	7/81	—	COL Applicant
2.4.10	Flood Protection Requirements	2	7/81	—	COL Applicant
2.4.11	Cooling Water Supply	2	7/81	—	COL Applicant
2.4.12	Groundwater	2	7/81	—	COL Applicant
	BTP HGEB 1	2	7/81	—	COL Applicant
2.4.13	Accidental Releases of Liquid Effluents in Ground and Surface Waters	2	7/81	—	COL Applicant
2.4.14	Technical Specifications and Emergency Operation Requirements	2	7/81	—	COL Applicant
2.5.1	Basic Geologic and Seismic Information	2	7/81	—	COL Applicant
2.5.2	Vibratory Ground Motion	4	3/07	—	COL Applicant
2.5.3	Surface Faulting	2	7/81	—	COL Applicant
2.5.4	Stability of Subsurface Materials and Foundations	2	7/81	—	COL Applicant
2.5.5	Stability of Slopes	2	7/81	—	COL Applicant
Chapter 3 Design of Structures, Components, Equipment, and Systems					
3.2.1	Seismic Classification	1	7/81	Yes	

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

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
3.2.2	System Quality Group Classification	1	7/81	Yes	
	Appendix A (Formerly BTP RSB 3-1)	1	7/81	Yes	
	Appendix B (Formerly BTP RSB 3-2)	1	7/81	Yes	
3.3.1	Wind Loadings	2	7/81	Yes	
3.3.2	Tornado Loadings	2	7/81	Yes	
3.4.1	Flood Protection	2	7/81	Yes	
3.4.2	Analysis Procedures	2	7/81	Yes	
3.5.1.1	Internally Generated Missiles (Outside Containment)	2	7/81	Yes	
3.5.1.2	Internally Generated Missiles (Inside Containment)	2	7/81	Yes	
3.5.1.3	Turbine Missiles	2	7/81	Yes	
3.5.1.4	Missiles Generated by Natural Phenomena	2	7/81	Yes	
3.5.1.5	Site Proximity Missiles (Except Aircraft)	1	7/81	Yes	
3.5.1.6	Aircraft Hazards	2	7/81	Yes	
3.5.2	Structures, Systems, and Components to be Protected from Externally Generated Missiles	2	7/81	Yes	
3.5.3	Barrier Design Procedures	1	7/81	Yes	
	[*Appendix A*	*0*	*7/81*](1)	Yes	
3.6.1	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	1	7/81	Yes	
	BTP ASB-3-1	1	7/81	Yes	
3.6.2	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	1	7/81	Yes	
	BTP MEB-3-1	2	6/87	Yes	
3.7.1	Seismic Design Parameters	2	8/89	Yes	
3.7.2	Seismic System Analysis	2	8/89	Yes	
3.7.3	Seismic Subsystem Analysis	2	8/89	Yes	
3.7.4	Seismic Instrumentation	1	7/81	Yes	
3.8.1	Concrete Containment	1	7/81	Yes	
	[*Appendix*	*0*	*7/81*](1)	Yes	

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

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
3.8.2	Steel Containment	1	7/81	Yes	
3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments	1	7/81	Yes	
3.8.4	Other Seismic category I Structures	1	7/81	Yes	
	Appendix A	0	7/81	Yes	
	Appendix B	0	7/81	Yes	
	Appendix C	0	7/81	Yes	
	Appendix D	0	7/81	Yes	
3.8.5	Foundations	1	7/81	Yes	
3.9.1	Special Topics for Mechanical Components	2	7/81	Yes	
3.9.2	Dynamic Testing and Analysis of Systems, Components, and Equipment	2	7/81	Yes	
3.9.3	ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures	1	7/81	Yes	
	Appendix A	1	4/84	Yes	
3.9.4	Control Rod Drive Systems	2	4/84	Yes	
3.9.5	Reactor Pressure Vessel Internals	2	7/81	Yes	
3.9.6	Inservice Testing of Pumps and Valves	2	7/81	Yes	
3.10	Seismic Qualification of Category I Instrumentation and Electrical Equipment	2	7/81	Yes	
3.11	Environmental Design of Mech. and Elec. Equip.	2	7/81	Yes	
Chapter 4 Reactor					
4.2	Fuel System Design	2	7/81	Yes	
	[*Appendix A*	*0*	*7/81*](2)	Yes	
4.3	Nuclear Design	2	7/81	Yes	
	BTP CPB 4.3-1	2	7/81	Yes	
4.4	Thermal and Hydraulic Design	1	7/81	Yes	
4.5.1	Control Rod Drive Structural Materials	2	7/81	Yes	
4.5.2	Reactor Internal and Core Support Materials	2	7/81	Yes	
4.6	Functional Design of Control Rod Drive System	1	7/81	Yes	

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

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
Chapter 5 Reactor Coolant System and Connected Systems					
5.2.1.1	Compliance with the codes and Standard Rule, 10CFR50.55a	2	7/81	Yes	
5.2.1.2	Applicable Code Cases	2	7/81	Yes	
5.2.2	Overpressure Protection	2	7/81	Yes	
	BTP RSB 5-2	0	7/81	No	PWR only
5.2.3	Reactor Coolant Pressure Boundary Materials	2	7/81	Yes	
	BTP MTEB 5-7 (Superseded by NUREG-0313)				
5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	1	7/81	Yes	
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	1	7/81	Yes	
5.3.1	Reactor Vessel Materials	1	7/81	Yes	
5.3.2	Pressure-Temperature Limits	1	7/81	Yes	
	BTP MTEB 5-2	1	7/81	Yes	
5.3.3	Reactor Vessel Integrity	1	7/81	Yes	
5.4	Deleted				
5.4.1.1	Pump Flywheel Integrity (PWR)	1	7/81	No	PWR only
5.4.2.1	Steam Generator Materials	2	7/81	No	PWR only
	BTP MTEB 5-3	2	7/81	No	
5.4.2.2	Steam Generator Tube Inservice Inspection	1	7/81	No	PWR only
5.4.6	Reactor Core Isolation Cooling System (BWR)	3	7/81	Yes	
5.4.7	Residual Heat Removal (RHR) System	3	7/81	Yes	
	BTP RSB 5-1				
5.4.8	Reactor Water Cleanup System (BWR)	2	7/81	Yes	
5.4.11	Pressurizer Relief Tank	2	7/81	No	PWR only
5.4.12	Reactor Coolant System High Point Vents	0	7/81	Yes	
Chapter 6 Engineered Safety Features					
6.1.1	Engineered Safety Features Materials	2	7/81	Yes	
	BTP MTEB 6-1	2	7/81	Yes	

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

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
6.1.2	Protective Coating Systems (Paints)—Organic Materials	2	7/81	Yes	
6.2.1	Containment Functional Design	2	7/81	Yes	
6.2.1.1.A	PWR Dry Containments, Including Subatmospheric Containments	2	7/81	No	PWR only
6.2.1.1.B	Ice Condenser Containments	2	7/81	No	PWR only
6.2.1.1.C	Pressure Suppression Type BWR Containments	6	8/84	Yes	
	Appendix A	2	1/83	Yes	
	Appendix B	0	1/83	Yes	
6.2.1.2	Subcompartment Analysis	2	7/81	Yes	
6.2.1.3	Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents	1	7/81	Yes	
6.2.1.4	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	1	7/81	Yes	
6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	2	7/81	No	PWR Only
	BTP CSB 6-1	2	7/81	No	PWR only
6.2.2	Containment Heat Removal Systems	4	10/85	Yes	
6.2.3	Secondary Containment Functional Design	2	7/81	Yes	
	BTP CSB 6-3	2	7/81	Yes	
6.2.4	Containment Isolation System	2	7/81	Yes	
	BTP CSB 6-4	2	7/81	Yes	
6.2.5	Combustible Gas Control in Containment	2	7/81	Yes	
	Appendix A	2	7/81	Yes	
	BTP CSB 6-2 (Superseded by Reg. Guide 1.7)				
6.2.6	Containment Leakage Testing	2	7/81	Yes	
6.2.7	Fracture Prevention of Containment Pressure Boundary	0	7/81	Yes	
6.3	Emergency Core Cooling System	2	4/84	Yes	
	BTP RSB 6-1	1	7/81	Yes	
6.4	Control Room Habitability Systems	2	7/81	Yes	

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

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	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	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 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR (Continued)

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
	BTP ICSB 3	2	7/81	Yes	
	BTP ICSB 4 (PSB)	2	7/81	Yes	
	BTP ICSB 5 (Superseded by Std. Tech Specs)				
	BTP ICSB 9 (Superseded by Std. Tech Specs)				
	BTP ICSB 12	2	7/81	Yes	
	BTP ICSB 13	2	7/81	Yes	
	BTP ICSB 14	2	7/81	Yes	
	BTP ICSB 16 (Deleted)				
	BTP ICSB 19 (Deleted)				
	BTP ICSB 20	2	7/81	Yes	
	BTP ICSB 21	2	7/81	Yes	
	BTP ICSB 22	2	7/81	Yes	
	BTP ICSB 25 (Superseded by Std. Tech Specs)				
	BTP ICSB 26	2	7/81	Yes	
	Appendix 7-B General Agenda, Station Site Visits	1	7/81	Yes	
Chapter 8 Electric Power					
8.1	Electric Power-Interaction	2	7/81	Yes	
	Table 8-1 Acceptance Criteria and Guidelines for Electric Power Systems	2	7/81	Yes	
8.2	Offsite Power System	3	7/83	Yes	ABWR and COL Applicant
	Appendix A	0	7/83	Yes	ABWR and COL Applicant
8.3.1	AC Power Systems (Onsite)	2	7/81	Yes	
	Appendix (Superseded by BTP PSB-2)				
8.3.2	DC Power Systems (Onsite)	2	7/81	Yes	
	Appendix 8 — A Branch Technical Positions (PSB)	2	7/81	Yes	
	BTP ICSB 2 (PSB) (Superseded by IEEE-387)				
	BTP ICSB 4 (PSB)	2	7/81	Yes	

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

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
	BTP ICSB 8 (PSB)	2	7/81	Yes	
	BTP ICSB 11 (PSB)	2	7/81	Yes	
	BTP ICSB 15 (PSB) (Deleted)				
	BTP ICSB 17 (PSB) (Superseded by Reg. Guide 1.9)				
	BTP ICSB 18 (PSB)	2	7/81	Yes	
	BTP ICSB 21 (PSB)	2	7/81	Yes	
	BTP PSB 1	0	7/81	Yes	
	BTP PSB 2	0	7/81	Yes	
	Appendix 8 — B General Agenda, Station Site Visits	0	7/81	Yes	
Chapter 9 Auxiliary Systems					
9.1.1	New Fuel Storage	2	7/81	Yes	
9.1.2	Spent Fuel Storage	3	7/81	Yes	
9.1.3	Spent Fuel Pool Cooling and Cleanup System	1	7/81	Yes	
9.1.4	Light Load Handling System (Related to Refueling)	2	7/81	Yes	
	BTP ASB 9-1 (Superseded by NUREG-0554)				
9.1.5	Overhead Heavy Load Handling Systems	0	7/81	Yes	
9.2.1	Station Service Water System	4	6/85	Yes	ABWR and COL Applicant
9.2.2	Reactor Auxiliary Cooling Water Systems	3	6/86	Yes	ABWR and COL Applicant
9.2.3	Demineralized Water Makeup System	2	7/81	Yes	ABWR and COL Applicant
9.2.4	Potable and Sanitary Water Systems	2	7/81	Yes	ABWR and COL Applicant
9.2.5	Ultimate Heat Sink	2	7/81	—-	COL Applicant
	BTP ASB 9-2	2	7/81	Yes	

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

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
9.2.6	Condensate Storage Facilities	2	7/81	Yes	
9.3.1	Compressed Air System	1	7/81	Yes	
9.3.2	Process and Post-Accident Sampling Systems	2	7/81	Yes	
9.3.3	Equipment and Floor Drainage System	2	7/81	Yes	ABWR and COL Applicant
9.3.4	Chemical and Volume Control System (PWR) (Including Boron Recovery System)	2	7/81	No	PWR only
9.3.5	Standby Liquid Control System (BWR)	2	7/81	Yes	
9.4.1	Control Room Area Ventilation System	2	7/81	Yes	ABWR and COL Applicant
9.4.2	Spent Fuel Pool Area Ventilation System	2	7/81	Yes	
9.4.3	Auxiliary and Radwaste Area Ventilation System	2	7/81	Yes	ABWR and COL Applicant
9.4.4	Turbine Area Ventilation System	2	7/81	Yes	
9.4.5	Engineered Safety Feature Ventilation System	2	7/81	Yes	
9.5.1	Fire Protection Program	3	7/81	Yes	
	BTP CMEB 9.5-1	2	7/81	Yes	
	Appendix A (Deleted)				
9.5.2	Communication Systems	2	7/81	Yes	ABWR and COL Applicant
9.5.3	Lighting Systems	2	7/81	Yes	
9.5.4	Emergency Diesel Engine Fuel Oil Storage and Transfer System	2	7/81	Yes	
9.5.5	Emergency Diesel Engine Cooling Water System	2	7/81	Yes	
9.5.6	Emergency Diesel Engine Starting System	2	7/81	Yes	
9.5.7	Emergency Diesel Engine Lubrication System	2	7/81	Yes	
9.5.8	Emergency Diesel Engine Combustion Air Intake and Exhaust System	2	7/81	Yes	

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

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
Chapter10	**Steam and Power Conversion System**				
10.2	Turbine Generator	2	7/81	Yes	
10.2.3	Turbine Disk Integrity	1	7/81	Yes	
10.3	Main Steam Supply System	3	4/84	Yes	
10.3.6	Steam and Feedwater System Materials	2	7/81	Yes	
10.4.1	Main Condensers	2	7/81	Yes	
10.4.2	Main Condenser Evacuation System	2	7/81	Yes	
10.4.3	Turbine Gland Sealing System	2	7/81	Yes	
10.4.4	Turbine Bypass System	2	7/81	Yes	
10.4.5	Circulating Water System	2	7/81	Yes	ABWR and COL Applicant
10.4.6	Condensate Cleanup System	2	7/81	Yes	
10.4.7	Condensate and Feedwater System	3	4/84	Yes	
	BTP ASB 10-2	3	4/84	Yes	
10.4.8	Steam Generator Blowdown System (PWR)	2	7/81	No	PWR only
10.4.9	Auxiliary Feedwater System (PWR)	2	7/81	No	PWR only
	BTP ASB 10-1	2	7/81	No	PWR only
Chapter 11	**Radioactive Waste Management**				
11.1	Source Terms	2	7/81	Yes	
11.2	Liquid Waste Management Systems	2	7/81	Yes	
11.3	Gaseous Waste Management Systems	2	7/81	Yes	
	BTP ETSB 11-5	0	7/81	No	
11.4	Solid Waste Management Systems	2	7/81	Yes	
	BTP ETSB 11-3	2	7/81	Yes	
	Appendix 11.4-A	0	7/81	Yes	
11.5	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	3	7/81	Yes	
	Appendix 11.5-A	1	7/81	Yes	

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

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
Chapter 12 Radiation Protection					
12.1	Assuring That Occupational Radiation Exposures Are as Low as Reasonably Achievable	2	7/81	Yes	
12.2	Radiation Sources	2	7/81	Yes	
12.3–12.4	Radiation Protection Design Features	2	7/81	Yes	
12.5	Operational Radiation Protection Program	2	7/81	—	COL Applicant
Chapter 13 Conduct of Operations					
13.1.1	Management and Technical Support Organization	2	7/81	—	COL Applicant
13.1.2–13.1.3	Operating Organization	2	7/81	—	COL Applicant
13.2	Training (Replaced by SRP Sections 13.2.1 and 13.2.2)				
13.2.1	Reactor Operator Training	0	7/81	—	COL Applicant
13.2.2	Training For Non-Licensed Plant Staff	0	7/81	—	COL Applicant
13.3	Emergency Planning	2	7/81	—	COL Applicant
13.4	Operational Review	2	7/81	—	COL Applicant
13.5	Plant Procedures (Replaced by SRP Sections 13.5.1 and 13.5.2)				
13.5.1	Administration Procedures	0	7/81	—	COL Applicant
13.5.2	Operating and Maintenance Procedures	1	7/85	—	COL Applicant
	Appendix A	0	7/85	—	COL Applicant
13.6	Physical Security	2	7/81	Yes	ABWR and COL Applicant

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

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
Chapter 14 Initial Test Program					
14.1	Initial Plant Test Programs—PSAR (Deleted)				
14.2	Initial Plant Test Programs—FSAR	2	7/81	Yes	
14.3	Standard Plant Design, Initial Test Program—Final Design Approval (FDA) (Deleted)				
Chapter 15 Accident Analysis					
15.0	Introduction	2	7/81	Yes	
15.1.1–15.1.4	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	1	7/81	Yes	
15.1.5	Steam System Piping Failures Inside and Outside of Containment (PWR)	2	7/81	No	PWR only
	Appendix A	2	7/81	No	PWR only
15.2.1–15.2.5	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)	1	7/81	Yes	
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries	1	7/81	Yes	
15.2.7	Loss of Normal Feedwater Flow	1	7/81	Yes	
15.2.8	Feedwater system Pipe Breaks Inside and Outside Containment (PWR)	1	7/81	No	PWR only
15.3.1–15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump and Flow Controller Malfunctions	1	7/81	Yes	
15.3.3–15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	2	7/81	Yes	
15.4.1	Uncontrolled control Rod Assembly Withdrawal from a Subcritical of Low Power Startup Condition	2	7/81	Yes	
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	2	7/81	Yes	
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	2	7/81	Yes	

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

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
15.4.4–15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	1	7/81	Yes	
15.4.6	Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant (PWR)	1	7/81	No	PWR only
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	1	7/81	Yes	
15.4.8	Spectrum of Rod Ejection Accidents (PWR)	2	7/81	No	PWR only
	Appendix A	1	7/81	No	PWR only
15.4.9	Spectrum of Rod Drop Accidents (BWR)	2	7/81	Yes	
	Appendix A	2	7/81	Yes	
15.5.1–15.5.2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory	1	7/81	Yes	
15.6.1	Inadvertent Opening of a PWR Pressurizer Relief Valve or a BWR Relief Valve	1	7/81	Yes	
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	2	7/81	Yes	
15.6.3	Radiological Consequences of Steam Generator Tube Failure (PWR)	2	7/81	No	PWR only
15.6.4	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	2	7/81	Yes	
15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	2	7/81	Yes	
	Appendix A	1	7/81	Yes	
	Appendix B	1	7/81	Yes	
	Appendix C (Deleted)				
	Appendix D	1	7/81	Yes	
15.7.1	Waste Gas System Failure (Deleted)				
15.7.2	Radioactive Liquid Waste System Leak or Failure (Released to Atmosphere) (Deleted)				

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

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
15.7.3	Postulated Radioactive Release Due to Liquid-Containing Tank Failures	2	7/81	Yes	
15.7.4	Radiological Consequences of Fuel Handling Accidents	1	7/81	Yes	
15.7.5	Spent Fuel Cask Drop Accidents	2	7/81	Yes	
15.8	Anticipated Transients Without Scram	1	7/81	Yes	
	Appendix (Deleted)				
Chapter 16 Technical Specifications					
16.0	Technical Specifications	1	7/81	Yes	
Chapter 17 Quality Assurance					
17.1	Quality Assurance During the Design and Construction Phases	2	7/81	Yes	
17.2	Quality Assurance During the Operations Phase	2	7/81	—	COL Applicant
17.5	Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants	0	3/07	Yes	COL Applicant
Chapter 18 Human Factors Engineering					
18.0	Human Factors Engineering/Standard Review Plan Development	1	9/84	Yes	
18.1	Control Room	0	9/84	Yes	
	Appendix A	0	9/84	Yes	
18.2	Safety Parameter Display System	0	11/84	Yes	
	Appendix A	0	11/84	Yes	

Notes:

(1) See Subsection 3.8.3.2

(2) See Subsection 4.2

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps	0	11/70	Yes	
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors	2	6/74	Yes	
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors	2	6/74	No	PWR only
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steamline Break Accident for Boiling Water Reactors	0	3/71	Yes	
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	0	3/71	Yes	
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident	3	3/07	Yes	
1.8	Personnel Selection and Training	--	--	--	See Table 17.0-1
1.9	Selection, Design, Qualification, and Testing of Emergency Diesel-Generator Units Used As Class 1E Onsite Electric Power Systems at Nuclear Plants	3	7/93	Yes	
1.11	Instrument Lines Penetrating Primary Reactor Containment	0	3/71	Yes	
1.12	Instrumentation for Earthquakes	1	4/74	No	NA
1.13	Spent Fuel Storage Facility Design Basis	1	12/75	Yes	
1.14	Reactor Coolant Pump Flywheel Integrity	1	8/75	No	PWR only
1.16	Reporting of Operating Information —Appendix A Technical Specifications	4	8/75	---	COL Applicant
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	2	5/76	Yes	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.21	Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light Water Nuclear Power Plants	1	6/74	Yes	
1.22	Periodic Testing of Protection System Actuation-Functions	0	2/72	Yes	
1.23	Onsite Meteorological Programs	0	2/72	Yes	
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure	0	3/72	No	PWR only
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	0	3/72	Yes	
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	---	---	---	See Table 17.0-1
1.27	Ultimate Heat Sink for Nuclear Power Plants	2	1/76	Yes	
1.28	Quality Assurance Program Requirements (Design and Construction)	---	---	---	See Table 17.0-1
1.29	Seismic Design Classification	---	---	---	See Table 17.0-1
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	---	---	---	See Table 17.0-1
1.31	Control of Ferrite Content in Stainless Steel Weld Metal	3	4/78	Yes	
1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants	2	2/77	Yes	
1.33	Quality Assurance Program Requirements (Operations)	2	2/78	---	COL Applicant
1.34	Control of Electroslag Weld Properties	0	12/72	Yes	
1.35	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures	2	1/76	Yes	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.36	Non-Metallic Insulation for Austenitic Stainless Steel	0	2/73	Yes	
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	---	---	---	See Table 17.0-1
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	---	---	---	See Table 17.0-1
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	---	---	---	See Table 17.0-1
1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants	0	3/73	Yes	
1.41	Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments	0	3/73	Yes	
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components	0	5/73	Yes	
1.44	Control of Use of Sensitized Stainless Steel	0	5/73	Yes	
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems	0	5/73	Yes	
[*1.47*	*Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems*	*0*	*5/73*	*Yes*][(4)]	
1.49	Power Levels of Nuclear Power Plants	1	12/73	Yes	
1.50	Control of Preheat Temperature Welding of Low-Alloy Steel	0	5/73	Yes	
1.52	Design, Testing, Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants	2	3/78	Yes	
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems[(7)]	0	6/73	Yes	
1.54	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants	0	6/73	Yes	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.56	Maintenance of Water Purity in Boiling Water Reactors	1	7/78	Yes	
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	0	6/73	Yes	
1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel		Superceded		See Table 17.0-1
1.59	Design Basis Floods for Nuclear Power Plants	2	8/77	Yes	
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	1	12/73	Yes	
1.61	Damping Values for Seismic Design of Nuclear Power Plants	0	10/73	Yes	
1.62	Manual Initiation of Protective Actions	0	10/73	Yes	
1.63	Electric Penetration Assemblies in Containment Structures of Nuclear Power Plants	3	2/87	Yes	
1.64	Quality Assurance Requirements for the Design of Nuclear Power Plants		Superceded		See Table 17.0-1
1.65	Materials and Inspections for Reactor Vessel Closure Studs	0	10/73	Yes	
1.68	Initial Test Programs for Water-Cooled Reactor Power Plants	2	8/78	Yes	
1.68.1	Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants	1	1/77	Yes	
1.68.2	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants	1	7/78	Yes	
1.68.3	Preoperational Testing of Instrument and Control Air Systems	0	4/82	Yes	
1.69	Concrete Radiation Shields for Nuclear Power Plants	0	12/73	Yes	
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants	3	11/78	Yes	
1.71	Welder Qualifications for Areas of Limited Accessibility	0	12/73	---	COL Applicant

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.72	Spray Pond Piping Made From Fiberglass-Reinforced Thermosetting Resin	2	11/78	Yes	
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	0	1/74	Yes	
1.74	Quality Assurance Terms and Definitions		Super-ceded		See Table 17.0-1
[*1.75*	*Physical Independence of Electric Systems*	*3*	*2/05*	*Yes*][(4)]	
1.76	Design Basis Tornado for Nuclear Power Plants	0	4/74	Yes	
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors	0	5/74	No	PWR only
1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	0	6/74	Yes	
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	1	9/75	No	PWR only
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Power Plants	1	1/75	Yes	
1.82	Water Sources for Long-Term Recirculation Cooling Following Loss-of-Coolant Accident	3	11/03	Yes	
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes	1	7/75	No	PWR only
[*1.84*	*Design and Fabrication Code Case Acceptability, ASME Section III, Division 1*	*33*	*8/05*	*Yes*][(1)]	
1.86	Termination of Operating Licenses for Nuclear Reactors	0	6/74	----	COL Applicant
1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596)	1	6/75	No	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.88	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records		Super-ceded		See Table 17.0-1
[*1.89*	*Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants*	*1*	*6/84*	*Yes*](2)	
1.90	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons	1	8/77	---	COL Applicant
1.91	Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants	2	2/78	Yes	
[*1.92*	*Combining Modal Responses and Spatial Components in Seismic Response Analysis*	*1*	*2/76*	*Yes*](1)	
1.93	Availability of Electric Power Sources	0	12/74	Yes	
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Steel During the Construction Phase of Nuclear Power Plants	---	---	---	See Table 17.0-1
1.95	Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release	1	1/77	Yes	
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	1	6/76	Yes	
1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident	3	5/83	Yes	
1.98	Assumptions for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor	0	3/76	Yes	
1.99	Radiation Embrittlement of Reactor Vessel Materials	2	5/88	Yes	
[*1.100*	*Seismic Qualification of Electric Equipment for Nuclear Power Plants*	*2*	*6/88*	*Yes*](2)	
1.101	Emergency Planning and Preparedness for Nuclear Power Reactors	3	8/92	Yes	
1.102	Flood Protection for Nuclear Power Plants	1	9/76	Yes	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
[*1.105*	*Instrument Setpoints for Safety-Related Systems*	*2*	*2/86*	*Yes*](3)	
1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves	1	3/77	Yes	
1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures	1	2/77	Yes	
1.108	Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants		superseded		Replaced by RG 1.9
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	1	10/77	Yes	
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Plants	0	3/76	Yes	
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	1	7/77	Yes	
1.112	Calculation for Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors	0	5/77	Yes	
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I	1	4/77	Yes	
1.114	Guidance On Being Operator At the Controls of a Nuclear Power Plant	1	11/76	---	COL Applicant
1.115	Protection Against Low-Trajectory Turbine Missiles	1	7/77	Yes	
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems	---	---	---	See Table 17.0-1
1.117	Tornado Design Classification	1	4/78	Yes	
1.118	Periodic Testing of Electric Power and Protection Systems(7)	2	6/78	Yes	
1.120	Fire Protection Guidelines for Nuclear Power Plants	1	11/87	Yes	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes	0	8/76	No	PWR only
1.122	Development of floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	1	2/78	Yes	
1.123	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants		Superceded		See Table 17.0-1
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports	1	1/78	Yes	
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants	1	11/78	Yes	
1.126	An Acceptable Model and Related Statistical Methods for the Analysis for Fuel Densification	1	3/78	Yes	
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants	1	3/78	---	COL Applicant
1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants	1	10/78	Yes	
1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants	1	2/78	Yes	
1.130	Service Limits and Loading Combination for Class 1 Plate-and-Shell-Type Component Supports	1	10/78	Yes	
1.131	Qualification Tests of Electric Cable, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants	0	8/77	Yes	
1.132	Site Investigations for Foundations of Nuclear Power Plants	1	3/79	Yes	
1.133	Loose-Part Detection Program for the Primary Systems of Light-Water-Cooled Reactors	1	6/81	Yes	
1.134	Medical Evaluation of Licensed Personnel for Nuclear Power Plants	2	5/87	---	COL Applicant
1.135	Normal Water Level and Discharge at Nuclear Power Plants	0	9/77	Yes	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.136	Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments	3	3/07	Yes	
1.137	Fuel-Oil Systems for Standby Diesel Generators	1	10/79	Yes	
1.138	Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants	0	4/78	Yes	
1.139	Guidance for Residual Heat Removal	0	5/78	Yes	
1.140	Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants	1	10/79	No	No charcoal filtration required for normal operation
1.141	Containment Isolation Provisions for Fluid Systems	0	4/78	Yes	
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)	2	11/01	Yes	
1.143	Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants	2	11/01	Yes	
1.144	Auditing of Quality Assurance Programs Nuclear Power Plants		Super-ceded		See Table 17.0-1
1.145	Atmospheric Dispersion Models for Potential Accident Consequences Assessments at Nuclear Power Plants	1	12/82	Yes	
1.146	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants		Super-ceded		See Table 17.0-1
1.147	Inservice Inspection Code Case Acceptability-ASME Section XI, Division 1	8	11/90	Yes	
1.148	Functional Specifications for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants	0	4/81	Yes	
1.149	Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations	1	5/87	---	COL Applicant
1.151	Instrument Sensing Lines	0	7/83	Yes	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
[*1.152*	*Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants*(5)	*2*	*1/06*	*Yes*](4)	
[*1.153*	*Criteria for Power, Instrumentation, and Control Portions of Safety Systems*	*1*	*6/96*	*Yes*](4)	
1.154	Format and Contents of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors	0	3/87	No	PWR only
1.155	Station Blackout	0	8/88	Yes	
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	0	6/93	Yes	
1.168	Verification, Validation, Reviews and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	1	2004	Yes	
1.169	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	0	9/97	Yes	
1.170	Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	0	9/97	Yes	
1.171	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	0	9/97	Yes	
1.172	Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	0	9/97	Yes	
1.173	Developing Software Life Cycle Process for Digital Computer Software Used in Safety Systems of Nuclear Power Plants	0	9/97	Yes	
1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems Instrumentation and Control Systems(6)	1	10/03	Yes	
1.209	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants	0	03/07	Yes	
5.1	Serial Numbering of Fuel Assemblies for Light-Water-Cooled Nuclear Power Plants	0	12/72	Yes	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
5.7	Control of Personnel Access to Protected Areas, Vital Areas, and Material Access Areas	1	5/80	Yes	
5.12	General use of Locks in the Protection and Control of Facilities and Special Nuclear Materials	0	11/73	Yes	
5.44	Perimeter Intrusion Alarm Systems	2	6/80	Yes	
5.65	Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls	0	9/86	Yes	
8.5	Criticality and Other Interior Evacuation Signals	0	2/73	Yes	
8.8	Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable	3	6/78	Yes	
8.12	Criticality Accident Alarm Systems	1	2/81	Yes	
8.19	Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates	1	6/79	Yes	

Table 1.8-20 Notes:

(1) See Subsection 3.9.1.7 for restriction of change to this revision. The change restriction to R.G 1.84 applies only in regard to Code Case N-420 (See DCD/Introduction, Table 7).
(2) See Section 3.10 for restriction of change to this revision.
(3) See Subsection 7.1.2.10.9 for restriction to change this revision.
(4) See Section 7A.1(1).
(5) The DI&C Systems will be evaluated for compliance with revised cyber security guidance being developed by the NRC and industry as computer security guidance to be issued as Secure Development and Operational Environment (SDOE) in the context of requirements in Revision 3 to Regulatory Guide 1.152 (DG-1249).
(6) RG 1.180 endorses IEEE 1050-1996. The digital instrumentation and controls systems conform to IEEE 1050-2004 as shown in Table 1.8-21.
(7) The DI&C Systems will comply with current RG 1.53 Rev. 2 (11/03), RG 1.118 Rev. 3 (1995), IEEE 323-2003, IEEE 338-1987, and IEEE 379-2000.

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR

Code or Standard Number	Year	Title
American Concrete Institute (ACI)		
211.1†	1981	Practice for Selecting Proportions for Normal, Heavy Weight, and Mass Concrete.
212	1981	Guide for Admixtures in Concrete
214	1977	Recommended Practice for Evaluation of Strength Test Results of Concrete
301†	1984	Specifications for Structural Concrete for Buildings
304	1973	Practice for Measuring, Mixing, Transporting, and Placing of Concrete
305	1977	Recommended Practice for Hot Weather Concreting
306	1978	Recommended Practice for Cold Weather Concreting
307	1979	Specification for the Design and Construction of Reinforced Concrete Chimneys
308†	1981	Practice for Curing Concrete
309	1972	Practice for Consolidation of Concrete
311.1R	1981	ACI Manual of Concrete Inspection
311.4R	1981	Guide for Concrete Inspection
315†	1980	Details and Detailing of Concrete Reinforcement
318†	1989	Building Code Requirements for Reinforced Concrete
[*349†*	*1997*	*Code Requirements for Nuclear Safety-Related Concrete Structures*][(1)]
359		(See ASME BPVC Section III)
American Institute of Steel Construction (AISC)		
[*N690†*	*1984*	*Specifications for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities*][(1)]
--	--	Manual of Steel Construction
American Iron and Steel Insitute		
SG-673	1986	Cold-Formed Steel Design Manual

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

Code or Standard Number	Year	Title
American Nuclear Society (ANS)		
2.3†	1983	Standard for Estimating Tornado and Other Extreme Wind Characteristics at Nuclear Power Sites
2.8†	1981	Determining Design Basis Flooding at Power Reactor Sites
4.5†	1988	Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors
5.1†	1979	Decay Heat Power in LWRs
[*7-4.3.2*†	*1982*	*Application Criteria for Programmable Digital Computer Systems in Safety Systems of NPGS*](3)(4)
18.1 (ANSI N237)	1984	Radioactive Source Term for Normal Operation of LWRs
52.1†	1983	Nuclear Safety Design Criteria for the Design of Stationary Boiling Water Reactor Plants
55.4	1979	Gaseous Radioactive Waste Processing Systems for Light Water Reactors
56.5	1979	PWR and BWR Containment Spray System Design Criteria
56.11†	1988	Standard Design Criteria for Protection Against the Effects of Compartment Flooding in Light Water Reactor Plants
57.1†(ANSI N208)	1980	Design Requirements for LWR Fuel Handling Systems
57.2(ANSI N210)	1976	Design Requirements for LWR Spent Fuel Storage Facilities at NPP
57.3	1983	Design Requirements for New Fuel Storage Facilities at LWR Plants
[*57.5*†	*1981*	*Light Water Reactor Fuel Assembly Mechanical Design and Evaluation*](2)
[*58.2*†	*1988*	*Design Basis for Protection of Light Water NPP Against Effects of Postulated Pipe Rupture*](8)
58.8†	1984	Time Response Design Criteria for Nuclear Safety Related Operator Actions
59.51 (ANSI N195)	1976	Fuel Oil Systems for Standby Diesel-Generators
American National Standards Institute (ANSI)‡		
A40	1993	National Plumbing Code
A58.1	1982	Minimum Design Loads for Buildings and other Structures, revised and redesigned as ASCE 7-1988
AG-1		(See ASME AG-1)
B3.5	1960	American Standard Tolerance for Ball and Roller Bearings
B30.2		(See ASME B30.2)

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

Code or Standard Number	Year	Title
B30.9		(See ASME B30.9)
B30.10		(See ASME B30.10)
B30.11		(See ASME B30.11)
B30.16		(See ASME B30.16)
B31.1		(See ASME B31.1)
B96.1		(See ASME B96.1)
C1 /ASQC	1985	Specifications of General Requirements for a Quality Program
C37.01		(See IEEE C37.01)
C37.04		(See IEEE C37.04)
C37.06	1987	Preferred Ratings of Power Circuit Breakers
C37.09		(See IEEE C37.09)
C37.11	1979	Power Circuit Breaker Control Requirements
C37.13		(See IEEE C37.13)
C37.16	1988	Preferred Ratings and Related Requirements for Low Voltage AC Power Circuit Breakers
C37.17	1979	Trip Devices for AC and General-Purpose DC Low-Voltage Power Circuit Breakers
C37.20		(See IEEE C37.20)
C37.50	1989	Test Procedures for Low Voltage AC Power Circuit Breakers Used in Enclosures
C37.100		(See IEEE C37.100)
C57.12		(See IEEE C57.12)
C57.12.11		(See IEEE C57.12.11)
C57.12.80		(See IEEE C57.12.80)
C57.12.90		(See IEEE C57.12.90)
C62.41		(See IEEE C62.41)
C62.45		(See IEEE C62.45)
C63.12		(See IEEE C63.12)
D975 /ASTM	1981	Diesel Fuel Oils, Specifications for
HE I	1970	Standards for Steam Surface Condenser, 6th Ed., Heat Exchangers Institute
[*HFS-100*	*1988*	*Human Factors Engineering of Visual Display Terminal Workstations*](5)

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

Code or Standard Number	Year	Title
MC11.1	1976	Quality Standard for Instrument Air
N5.12	1972	Protective Coatings (Paint) for Nuclear Industry
N13.1	1969	Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities
N14.6	1986	Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials
N18.7	1976	Administrative Controls and Quality Assurance for the Operation Phase of Nuclear Power Plants
N45.2.1[f] (RG 1.37)	1973	Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants
N45.2.2[f] (RG 1.38)	1972	Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants During the Construction Stage
N45.2.3	1973	Housekeeping During the Construction Phase of Nuclear Power Plants
N45.2.4	1972	Quality Assurance Program Requirements for Nuclear Power Plants
N45.2.5	1974	Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants
N45.2.8[f] (RG 1.116)	1976	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems
N45.4		(See ASME N45.4)
N101.2	1972	Protective Coatings (Paints) for Light Water Nuclear Containment Facilities
N101.4	1972	QA for Protective Coatings Applied to Nuclear Facilities
N195		(See ANS 59.51)
N237		(See ANS 18.1)
N270		(See ANS 52.2)
N509		(See ASME N509)
N510		(See ASME N510)
N690		(See AISC N690)
NQA-1		(See ASME NQA-1)
NQA-1a		(See ASME NQA-1a)
NQA-2a		(See ASME NQA-2a)
OM3	1990	Requirements for preoperational and Initial Startup Vibration Test Program for Water-Cooled Power Plants

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

Code or Standard Number	Year	Title
OM7	1986	Requirements for Thermal Expansion Testing of Nuclear Plant Piping Systems [September 1986 (Draft-Revision 7)]
[X3.139	*1987*	*Fiber Distributed Data Interface (FDDI) - Token Ring Media Access Control (MAC)]*(3)(4)
[X3.148	*1988*	*Fiber Distributed Data Interface (FDDI) - Token Ring Physical Layer Protocol (PHY)]*(3)(4)
[X3.166	*1990*	*Fiber Distributed Data Interface (FDDI) - Physical Layer Medium Dependent (PMD)]* (3)(4)
[X3T9.5/84-49	*Rev. 7.1 May 7, 1992*	*FDDI Station Mangement (SMT), Preliminary Draft]*(3)(4)
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†	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

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

Code or Standard Number	Year	Title
[NQA-2a†	*1990*	*Quality Assurance Requirements of Computer Software for Nuclear Facility Application]*(3)(4)
OMa	1988	Operation and Maintenance of Nuclear Power Plants (Addenda to OM-1987)
Sec II	1989	BPVC Section II, Material Specifications
NOG-1	2004	Rules for Construction of Overhead and Gantry Cranes
[*Sec III*	*1989*	*BPVC Section III, Division 1, Rules for Construction of Nuclear Power Plant Components]*(8)
[Sec III	*2001 with 2003 Addenda*	*BPVC Section III, Division 2, Rules for Construction of Nuclear Power Plant Components]*(6)
Sec VIII	1989	BPVC Section VIII, Rules for Construction of Pressure Vessel
Sec IX	1989	BPVC Section IX, Qualification Standard for Welding and Brazing Procedures Welder, Brazers and Welding and Brazing Operators
Sec XI	1989	BPVC Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components
American Society for Testing and Materials (ASTM)		
[*C776*	*1979*	*Sintered Uranium Dioxide Pellets*](2)
[*C934*	*1980*	*Design and Quality Assurance Practices for Nuclear Fuel Rods*](2)
E84 REV. A	1991	Methods of Test of Surface Burning Characteristics of Building Materials
E119	1988	Standard Test Methods for Fire Tests of Building Construction and Materials
E152	1981	Standard Methods of Fire Tests of Door Assemblies
		(See ASME BPVC Section III for ASTM Material Specifications)
American Welding Society (AWS)		
A4.2†	1986	Procedures for Calibrating Magnetic Instruments to Measure the Delta Ferrite content of Anstenitic Stainless Steel Weld Metal
D1.1†	1986	Steel Structural Welding Code
D14.1†	1985	Welding of Industrial and Mill Cranes and other Material Handling Equipment
American Water Works Association (AWWA)		
D100†	1984	Welded Steel Tanks for Water Storage
CMAA70	1983	Specification for Electric Overhead Traveling Cranes

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

Code or Standard Number	Year	Title
Insulated Cable Engineer Association (ICEA)		
P-46-426/IEEE S-135	1982	Ampacities Including Effect of Shield Losses for Single Conductor Solid-Dielectric Power Cable 15 kV through 69 kV
P-54-440/NEMA WC-51	1987	Ampacities of Cables in Open-Top Cable Trays
S-61-402/NEMA WC-5	1973	Thermoplastic Insulated Wire & Cable for the Transmission and Distribution of Electrical Energy
S-66-524/NEMA WC-7	1982	Cross Linked Thermosetting Polyethylene Insulated Wire and Cable for Transmission and Distributor of Electrical Energy
Institute of Electrical and Electronics Engineers (IEEE)		
C37.01†	1979	Application Guide for Power Circuit Breakers
C37.04†	1979	AC Power Circuit Breaker Rating Structure
C37.09†	1979	Test Procedure For Power Circuit Breakers
C37.13†	1989	Low Voltage Power Circuit Breakers
C37.20†	1987	Switchgear Assemblies and Metal-Enclosed Bus
[*C37.90.2*	*1987*	*Trial-Use Standard, Withstand Capability of Relay Systems to Radiated Electromagnetic Interference form Transceivers*](3)(4)
C37.100†	1992	Definitions for Power Switchgear Transformers
C57.12†	1987	General Requirements for Distribution, Power, and Regulating Transformers
C57.12.11†	1980	Guide for Installation of Oil-immersed Transformers(10MVA & Larger, 69-287 kV Rating)
C57.12.80†	1978	Terminology for Power and Distribution Transformers
C57.12.90†	1987	Test Code for Distribution, Power, and Regulating Transformers
[*C62.41†*	*1991*	*Guide for Surge Voltage in Low-Voltage AC Power Circuits*](3)(4)
[*C62.45†*	*1987*	*Guide on Surge Testing for Equipment Connected to Low-Voltage AC Power Curcuits*](3)(4)
[*C63.12†*	*1987*	*American National Standard for Electromagnetic Compatibility Limits-Recommended Practice*](3)(4)
7-4.3.2	2003	"Standard Criteria for Digital Computers Used in Safety Systems of Nuclear Power Generation Stations")
80†	1986	Guide for Safety in AC Substation Grounding
81†	1983	Guide for Measuring Earth Resistivity, Ground Impedance, and Earth Surface Potentials of a Ground System
S-135		(See ICEA P-46-426)

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

Code or Standard Number	Year	Title
141†	1986	Recommended Practice for Electric Power Distribution for Industrial Plants (IEEE Red Book)
242†	1986	Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems
308†	1980	Criteria for Class 1E Power Systems for NPGS
317†	1983	Electrical Penetration Assemblies in Containment Structures for NPGS
[323†	*1974*	*Qualifying Class 1E Equipment for NPGS]*(3)(4)(10)
334†	1974	Motors for NPGS, Type Tests of Continuous Duty Class 1E
[338†	*1977*	*Criteria for the Periodic Testing of NPGS Safety Systems*](3)(9)(10)
[344†	*1987*	*Recommended Practices for Seismic Qualifications of Class 1E Equipment for NPGS*](7)
352†	1987	General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems
379†	1977	Standard Application of the Single-Failure Criterion to NPGS Safety Systems(10)
382†	1985	Qualification of Actuators for Power-Operated Valve Assemblies with Safety-Related Functions for NPP
383†	1974	Type Test of Class 1E Cables; Field Splices and Connections for NPGS
[384†	*1992*	*Criteria for Independence of Class 1E Equipment and Circuits*](3)
387†	1984	Criteria for Diesel-Generator Units Applied as Standby Power Supplies for NPGS
399†	1990	Recommended Practice for Industrial and Commercial Power Systems Analysis (IEEE Brown Book)
450†	1987	Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations
484†	1987	Recommended Practice for the Design and Installation of Large Lead Storage Batteries for NPGS
485†	1983	Recommended Practice for Sizing Large Lead Storage Batteries for NPGS
500	1984	Guide to the Collection and Presentation of Electronic, Sensing Component, and Mechanical Equipment Reliability Data for Nuclear Power Generating Stations.
[518	*1982*	*Guide for the Installation of Electrical Equipment to Minimize Electrical Noise Inputs to Controllers from External Sources]*(3)(4)
519†	1981	IEEE Standard Recommended Practices and Requirements for Harmonic Control in Electrical Power Systems

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

Code or Standard Number	Year	Title
[*603*†]	*1991*	*IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations, including the corrective sheet dated January 30, 1995*](3)/(4)
622†	1987	Recommended Practice for the Design and Installation of Electric Heat Tracing Systems in Nuclear Power Generating Stations
622A†	1984	Recommended Practice for the Design and Installation of Electric Pipe Heating Control and Alarm Systems in Nuclear Power Generating Stations
665	1995	IEEE Guide for Generating Station Grounding
[*730*	*1984*	*Standard for Software Quality Assurance Plans*](3)(4)
741†	1986	Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations
765†	1983	Standard for Preferred Power Supply for Nuclear Power Generating Stations
[*802.2*†	*1985*	*Standards for Local Area Networks: Logic Link Control*](3)
[*802.5*†	*1985*	*Token Ring Access Method and Physical Layer Specifications*](3)
[*828*†	*1990*	*Standard for Software Configuration Management Plans*](3)(4)
[*829*†	*1983*	*Standard for Software Test Documentation*](3)(4)
[*830*†	*1993*	*Standard for Software Requirements Specifications*](3)(4)
[*845*†	*1988*	*Guide to Evaluation of Man-Machine Performance in Nuclear Power Generating Station Control Rooms and Other Peripheries*](5)
944†	1986	Recommended Practice for the Application and Testing of Uninterruptable Power Supplies for Power Generating Station
946†	1985	Recommended Practice for the Design of Safety-Related DC Auxiliary Power Systems for Nuclear Power Generating Stations
1008	1987	Standard for Software Unit Testing
[*1012*†	*1998*	*Standard for Software Verification and Validation*](3)(4)
[*1023*†	*1988*	*IEEE Guide to the Application of Human Factors Engineering to Systems, Equipment and Facilities of Nuclear Power Generating Stations*](5)
1028	1997	Standard for Software Reviews and Audits
[*1042*	*1987*	*Guide to Software Configuration Management*](3)(4)
[1050	*2004*	*Guide for Instrumentation and Control Equipment Grounding in Generating Stations](3)(4)*
1074	1995	Standard for Developing Software Life Cycle Processes
[*1228 (Draft)*	*1992*	*Standard for Software Safety Plans*](3)(4)

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

Code or Standard Number	Year	Title
Instrument Society of America (ISA)		
S7.3†	1981	Quality Standard for Instrument Air
S67.02-80	1980	Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants
National Electrical Manufacturers Association (NEMA)		
AB 1	1986	Molded Case Circuit Breakers
FB1	1977	Fittings and Support for Conduit and Cable Assemblies
ICS 1†	1983	General Standards for Industrial Control
ICS 2†	1988	Standards for Industrial Control Devices, Controllers and Assemblies
MG 1	1987	Motors and Generators
WC-5		(See ICEA S-61-402)
WC 7		(See ICEA S-66-524)
WC 51		(See ICEA P-54-440)
National Fire Protection Association (NFPA)		
10†	1981	Portable Fire Extinguishers - Installation
10A	1973	Portable Fire Extinguishers - Maintenance and Use
11†	1988	Low Expansion Foam and Combined Agent Systems-Foam Extinguishing System
12†	1985	Carbon Dioxide Extinguishing Systems
13†	1985	Installation of Sprinklers Systems
14†	1986	Installation of Standpipe and Hose Systems
15†	1985	Standard for Water Spray Fixed Systems
16†	1991	Deluge Foam-Water Sprinkler and Foam-Water Spray Systems
16A†	1988	Recommended Practice for the Installation of Closed Head Foam-Water Sprinkler Systems
20†	1990	Standard for the Installation of Centrifugal Fire Pumps
24†	1984	Private Service Mains and their Appurtenances
26†	1988	Recommended Practice for the Supervision of Valves Controlling Water Supplies for Fire Protection
37†	1984	Stationary Combustion Engines and Gas Turbines
70†	1987	National Electrical Code-Handbook 1987
72†	1990	Protective Signaling Systems
72D	1986	Proprietary Protective Signaling Systems

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

Code or Standard Number	Year	Title
78†	1986	Lightning Protection Code
80†	1986	Fire Doors and Windows
80A†	1993	Protection of Buildings from Exterior Fire Exposures
90A†	1985	Installation of Air Conditioning and Ventilating Systems
91†	1983	Blower and Exhaust Systems
92A†	1988	Smoke Control Systems
101†	1985	Life Safety Code
251†	1985	Fire Test, Building Construction and Materials
252†	1984	Fire Tests, Door Assemblies
255†	1984	Building Materials, Test of Surface Burning Characteristics
321†	1987	Classification of Flammable Liquids
801†	1986	Facilities Handling Radioactive Materials
802†	1988	Nuclear Research Reactors
803†	1993	Fire Protection for Light Water Nuclear Power Plants
1961†	1979	Fire Hose
1963†	1985	Screw Threads and Gaskets for Fire Hose Connections
Steel Structures Painting Council (SSPC)		
PA-1	1972	Shop, Field and Maintenance Painting
PA-2	1973	Measurements of Paint Film Thickness with Magnetic Gages
SP-1	1982	Solvent Cleaning
SP-5	1985	White Metal Blast Cleaning
SP-6	1986	Commercial Blast Cleaning
SP-10	1985	Near-White Blast Cleaning
U.S. Department of Defense (DOD)		
[*5000.2*	*1991*	*Defense Acquisition Management Policies and Procedures*](5)
[*AD/A223168*	*1990*	*System Engineering Management Guide*](5)
[*AR602-1*	*1983*	*Human Factors Engineering Program*](5)
[*DI-HFAC-80740*	*1989*	*Human Factors Engineering Program Plan*](5)
[*ESD-TR-86-278*	*1986*	*Guidelines for Designing User Interface Software*](5)
[*HDBK-761A*	*1990*	*Human Engineering Guidelines for Management Information Systems*](5)
[*HDBK-763*	*1991*	*Human Engineering Procedures Guide, Ch. 5-7 & Appendix. A&B*](5)

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

Code or Standard Number	Year	Title
[*STD-2167A*	*1988*	*Defense System Software Development*](3)(4)
[*TOP 1-2-610*	*1990*	*Test Operating Procedure Part 1*](5)
U.S. Military (MIL)		
F-51068	Latest Edition	Filter, Particulate High-Efficiency, Fire-Resistant
[*H-46855B*	*1979*	*Human Engineering Requirements for Military Systems, Equipment and Facilities*](5)
[*HDBK-217*	*Latest Edition*	*Reliability Prediction of Electronic Equipment*](3)
[*HDBK-251*	*Latest Edition*	*Reliability/Design: Thermal Applications*](3)
[*HDBK-759A*	*1981*	*Human Factors Engineering Design for Army Material*](5)
STD-282	1956	Filter Units, Protective Clothing Gas-Mask Components and Related Products: Performance-Test Methods
[*STD-461E*	*1999*	*Electromagnetic Emission and Susceptibility Requirements for the Control of Electromagnetic Interference*](3)(4)
[*STD-462E*	*1999*	*Measurement of Electromagnetic Interference Characteristics*](3)(4)
[*STD-1472D*	*1989*	*Human Engineering Design Criteria for Military Systems, Equipment and Facilities*](5)
Others		
ASCE 7	1988	Minimum Design Loads for Buildings and Other Structures
ERDA 76-21	1976	Testing of Ventilation Systems, Section 9 of Industrial Ventilation Systems
[*IEC 61000*	*2001*	*Electromagnetic Compatibility (EMC) - Part 4-2: Testing and Measurement Techniques - Electrostatic Discharge Immunity Test*](3)
[*IEC 880*	*1986*	*Software for Computers in the Safety Systems of Nuclear Power Stations*](3)(4)
[*IEC 964*	*1989*	*Design for Control Rooms of Nuclear Power Plants, Bureau Central de la Commission Electrotechnique Internationale*](5)
[*ISO 7498*	*1984*	*Open Systems Interconnection-Basic Refence Model, as the Data Link Layer and Physical Layer*](3)
OSHA 1910.179	1990	Overhead and Gantry Cranes
TEMA C	1978	Standards of Tubular Exchanger Manufacturers Association
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

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

Code or Standard Number	Year	Title
--	--	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
IBC	2006	International Building Code

* The listing of a code or standard does not necessarily mean that it is applicable in its entirety.

† 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.

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

Notes:

(1) See Subsection 3.8.3.2 for restriction to use of these.
(2) See Subsection 4.2.
(3) See section 7A.1(1).
(4) See Section 7A.1(2).
(5) See Section 18E.1 for required use of this document.
(6) See Subsection 3.8.1.1.1 for specific restriction of change to this edition.
(7) See Section 3.10 for restriction of change to this revision.
(8) See Subsection 3.9.1.7 for specific restriction of change to this edition in application to piping design. See Table 3.2-3 for the restricted Subsections of this Code as applied to piping design only.
(9) See Subsection 7.1.1.2.
(10) The DI&C Systems will comply with current RG 1.53 Rev. 2 (11/03), RG 1.118 Rev. 3 (1995), IEEE 323-2003, IEEE 338-1987, and IEEE 379-2000.

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
Type: Generic Letters			
80-06	4/25/80	Clarification of NRC Requirement for Emergency Response Facilities at Each Site	
80-30	12/15/80	Periodic Updating of Final Safety Analysis Reports (FSARs)	COL Applicant
80-31	12/22/80	Control of Heavy Loads	
81-03	2/26/81	Implementation of NUREG-0313m, Rev. 1	
81-04	2/25/81	Emergency Procedures and Training for Station Blackout Events	COL Applicant
81-07	2/3/81	Control of Heavy Loads	
81-10	2/18/81	Post-TMI Requirements for the Emergency Operations Facility	
81-11	2/22/81	Error in NUREG-0619	
81-20	4/1/81	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	
81-37	12/29/81	ODYN Code Reanalysis Requirements	
81-38	11/10/81	Storage of Low-Level Radioactive Wastes at Power Reactor Sites	COL Applicant
82-09	4/20/82	Environmental Qualification of Safety-Related Electrical Equipment	
82-21	10/6/82	Technical Specifications for Fire Protection Audits	COL Applicant
82-22	10/30/82	Inconsistency Between Requirements of 10CFR73.40(d) and Standard Technical Specifications for Performing Audits of Safeguard Contingency Plans	
82-27	11/15/82	Transmittal of NUREG-0763, “Guidelines for Confirmatory In-Plant Tests of Safety-Relief Valve Discharges for BWR Plants,” and NUREG-0783, “Suppression Pool Temperature Limits for BWR Containments.”	
82-33	12/17/82	Supplement 1 to NUREG-0737	
82-39	12/22/82	Problems with the Submittals of 10CFR73.21 Safeguards Information Licensing Review	COL Applicant
83-05	2/83	Safety Evaluation of “Emergency Procedure Guidelines,” Revision 2, NEDO-24934, June 1982	COL Applicant
83-07	2/16/83	The Nuclear Waste Policy Act of 1982	COL Applicant
83-13	3/2/83	Clarification of Surveillance Requirements for HEPA Filters and Charcoal Absorber Units in Standard Technical Specifications on ESF Cleanup Systems	

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
83-28	7/8/83	Required Actions Based on Generic Implications of Salem ATWS Events	
83-33	10/19/83	NRC Positions on Certain Requirements of Appendix R to 10 CFR 50	COL Applicant
84-15	7/2/84	Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability	
84-23	10/26/84	Reactor Vessel/Water Level Instrumentation in BWRs	
85-01	1/9/85	Fire Protection Policy Steering Committee Report	
86-10	4/24/86	Implementation of Fire Protection Requirements	
87-06	3/13/87	Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves	COL Applicant
87-09	6/4-87	Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operations and Surveillance Requirements	
88-01	1/25/88	NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping	
88-02	1/20/88	Integrated Safety Assessment Program II (ISAP II)	
88-14	8/8/88	Instrument Air Supply System Problems Affecting Safety-Related Equipment Past Related Correspondence: IE Notice 87-28, Supp. 1 NUREG-1275, Volume 2	
88-15	9/12/88	Electric Power Systems — Inadequate Control Over Design Process Past Related Correspondence: IE Notice 88-45	
88-16	10/4/88	Removal of Cycle-Specific Parameter Limits from Technical Specifications	
88-18	10/20/88	Plant Record Storage on Optical Disks Past Related Correspondence: NUREG-0800 Reg. Guide 1.28, Rev. 3	COL Applicant
88-20	11/23/88	Individual Plant Examination for Severe Accident Vulnerabilities-10CFR Para. 50.54(f)	
88-20	8/29/89	Generic 88-20 Supplement No. 1	
89-01	1/31/89	Implementation of programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
89-02	3/21/89	Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products Past Related Correspondence: EPRI-NP-5652, “Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety-Related Applications”. Bulletins 87-02 and Supplements 1 and 2, 88-05 and Supplements 1 and 2, 88-10 IE Notices 87-66, 88-19, 88-35, 88-46 and Supplements 1 and 2, 88-48 and Supplement 1, 88-97	COL Applicant
89-04	4/3/89	Guidance on Developing Acceptable Inservice Testing Program	COL Applicant
89-06	4/12/89	Task Action Plan Item I.D.2 – Safety Parameter Display System CFR 50.54(f)	1A.2.3
89-07	4/28/89	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	
89-07 Supp I	4/21/89	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	
89-08	5/2/89	Erosion/Corrosion-Induced Pipe Wall Thinning	
89-10	6/28/89	Safety-Related Motor-Operated Valve Testing and Surveillance	COL Applicant
89-13	7/18/89	Service Water System Problems Affecting Safety-Related Equipment	COL Applicant
89-14	8/21/89	Line Item Improvements in Technical Specifications Removal of the 3.25 Limit on Extending Surveillance Intervals	
89-15	8/21/89	Emergency Response Data System	COL Applicant
89-16	9/1/89	Installation of a Hardened Wetwell Vent	
89-18	9/6/89	Resolution of USI A-17, Systems Interactions	Subsection 19B.2.59
89-19	9/20/89	Request for Action Related to Resolution of Unresolved Safety Issue A-47, “Safety Implication of Control Systems in LWR Nuclear Power Plants”, Pursuant to 10CFR50.54(f)	Subsection 19B.2.17
89-22	10/19/89	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	
90-09	12/11/90	Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions	
91-03	03/06/91	Reporting of Safeguards Events	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
91-04	04/02/91	Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle	
91-05	04/04/91	Licensee Commercial Grade Procurement and Dedication Programs	
91-06	04/29/91	Resolution of Generic Issue A-30, “Adequacy of Safety-Related DC Power Supplies”, Pursuant to 10CFR50.54(f)	Subsection 19B.2.52
91-10	07/08/91	Explosive Searches at Protected Area Portals	COL Applicant
91-11	07/19/91	Resolution of Generic Issue 48, “LCOs for Class 1E Tie Breakers”, Pursuant to 10CFR50.54(f)	Subsection 19B.2.52
91-14	09/23/91	Emergency Telecommunications	
91-16	10/03/91	Licensed Operators' and Other Nuclear Facility Personnel Fitness for Duty	COL Applicant
91-17	10/17/91	Generic Safety Issue 29, “Bolting Degradation or Failure in Nuclear Power Plants”	Subsection 19B.2.62
92-04	8/19/92	Resolution of the Issues Related to Reactor Vessel Level Instrumentation in BWRs Pursuant to 10CFR50.54(f)	
Type: IE Bulletins			
79-02	3/8/79	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	
79-08	4/14/79	Events Relevant to BWR Identified During TMI Incident	
80-01	1/11/80	ADS Valve Pneumatic Supply	
80-03	2/6/80	Loss of Charcoal from Absorber Cells	
80-05	3/10/80	Vacuum Condition Resulting in Damage to Chemical and Volume Control System (CVCS) Holdup Tanks	COL Applicant
80-06	3/13/80	ESF Reset Controls	
80-08	4/7/80	Containment Lines Penetration Welds	COL Applicant
80-10	5/6/80	Non-Radioactive System – Potential for Unmonitored Release	COL Applicant
80-12	5/9/80	Decay Heat Removal System Operability	COL Applicant
80-13	5/12/80	Cracking in Core Spray Spargers	
80-15	6/18/80	Possible Loss of Emergency Notification System with Loss of Offsite Power	
80-20	7/31/80	Westinghouse Type W-2 Switch Failures	

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
80-21	11/6/80	Valve Yokes Supplied by Mole	COL Applicant
80-22	9/11/80	Automatic Industries, Model 200-500-008 Sealed Source Containers	COL Applicant
80-24	11/21/80	Prevention of Damage due to H2O Leakage Inside Containment	NUREG/CR-4524
80-25	12/19/80	Operating Problems with Target Rock SRVs at BWRs	
81-01	1/27/81	Surveillance of Mechanical Snubbers	
81-02	4/9/81	Failure of Gate Type Valves to Close	COL Applicant
81-02, Supp 1	8/19/81	Failure of Gate Type Valves to Close Against Differential Pressure	COL Applicant
81-03	4/10/81	Flow Blockage of Cooling Water to Safety System	COL Applicant
82-04	12/3/82	Deficiencies in Primary Containment Electrical Penetration Assemblies	COL Applicant
83-06	7/22/83	Non-Conforming Materials Supplied by Tube-Line Corp.	COL Applicant
84-01	2/3/84	Cracks in Boiling Water Reactor Mark I Containment Vent Header	
84-03	8/24/84	Refueling Cavity Water Seal	
85-03	11/15/85	Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings	COL Applicant
85-03, Supp 1	4/27/88	Motor-Operated Valve Common Mode Failure During Plant Transients Due to Improper Switch Settings Past Related Correspondence: IE Bulletin 85-03, IE Notice 86-29, and IE Notice 87-01	COL Applicant
86-01	5/23/86	Minimum Flow Logic Problems That Could Disable RHR Pumps	
86-03	10/8/86	Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line	
87-01	7/9/87	Thinning of Pipe Walls in Nuclear Power Plants	
87-02	11/6/87	Fastener Testing to Determine Conformance with Applicable Material Specifications	COL Applicant
87-02, Supp 1	4/22/88	Fastener Testing to Determine Conformance with Applicable Material Specifications Past Related Correspondence: IE Notice 88-17	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
87-02, Supp 2	6/10/88	Fastener Testing to Determine Conformance with Applicable Material Specifications	COL Applicant
88-04	5/5/88	Potential Safety-Related Pump Loss Past Related Correspondence: IE Notice 87-59	
88-07	6/15/88	Power Oscillations in Boiling Water Reactors (BWRs) Past Related Correspondence: IE Notice 88-39	
88-07, Supp 1	12/30/88	Power Oscillations in Boiling Water Reactors (BWRs)	Subsections 7.1.2.6.1.4 and 7.1.2.1.1.2.2
90-01	03/09/90	Loss of Fill-Oil in Transmitters Manufactured by Rosemount	
90-02	03/20/90	Loss of Thermal Margin Caused by Channel Box Bow	
91-01	10/18/91	Reporting Loss of Criticality Safety Controls	
Type: IE Information Notices			
79-22	9/14/79	Qualifications of Control Systems	COL Applicant
80-12	3/31/80	Instrumentation Failure Causes PORV Opening	
80-21	5/16/80	Anchorage and Support of Safety-Related Electrical Equipment	
80-22	5/28/80	Breakdowns in Contamination Control Programs	COL Applicant
80-40	11/7/80	Excessive N_2 Supply Pressure	
80-42	11/24/80	Effect of Radiation on Hydraulic Snubber Fluid	
81-05	3/13/81	Degraded DC Systems at Palisades	COL Applicant
81-07	3/16/81	Potential Problem with Water Soluble Purge Dam Materials Used During Inert Gas Welding	COL Applicant
81-10	3/25/81	Inadvertent Containment Spray	COL Applicant
81-20	7/13/81	Test Failures of Electrical Penetrations	
81-21	7/21/81	Potential Loss of Direct Access to Ultimate Heat Sink	COL Applicant
81-31	10/8/81	Failure of Safety Injection Valves	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
81-38	12/17/81	Potential Significant Equipment Failures Resulting from Contamination of Air-Operated Systems	COL Applicant
82-03	3/22/82	Environmental Tests of Electrical Terminal Block	
82-10	3/3/82	Following Up Symptomatic Repairs	COL Applicant
82-12	4/21/82	Surveillance of Hydraulic Snubbers	
82-22	7/9/82	Failures in Turbine Exhaust Lines	
82-23	7/16/82	Main Steam Isolation Valve Leakage	
82-25	7/20/82	Failures of Hiller Actuators Upon Gradual Loss of Air Pressure	
82-32	8/19/82	Contamination of Reactor Coolant System by Organics	COL Applicant
82-40	9/22/82	Deficiencies in Primary Containment Electrical Penetration Assemblies	
82-43	11/16/82	Deficiencies in LWR Air Filtration/Vent System	
82-49	12/16/82	Correction for Sample Conditions for Air & Gas Monitor	COL Applicant
83-03	1/28/83	Calibration of Liquid Level Instruments	COL Applicant
83-07	3/7/83	Nonconformities with Materials Supplied by Tube Line Corp.	COL Applicant
83-08	3/9/83	Component Failures Caused by Elevated DC Control Voltage	
83-17	3/31/83	Electrical Control Logic Problem Resulting in Inoperable Auto Start of Emergency Diesel Generator	
83-30	5/11/83	Misapplication of Generic EOP Guidelines	COL Applicant
83-35	5/31/83	Fuel Movement with Control Rods Withdrawn at BWRs	COL Applicant
83-44	7/1/83	Damage to Redundant Safety Equipment from Backflow Through the Equipment	
83-46	7/11/83	Common Mode Valve Failures Degrade Surry's Recirculation Spray Subsystem	COL Applicant
83-50	8/1/83	Failure of Class 1E Circuit Breakers to Close	
83-51	8/5/83	Diesel Generator Events	
83-62	9/26/83	Failure of Toxic Gas Detectors	Subsection 19B.2.40

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
83-64	9/29/83	Lead Shielding Attached to Safety-Related Systems	COL Applicant
83-70	10/25/83	Vibration-Induced Valve Failures	
83-70, Supp 1	3/4/85	Vibration-Induced Valve Failures	
83-72	10/28/83	Environmental Qualification Testing Experience	
83-75	11/3/83	Improper Control Rod Manipulation	COL Applicant
83-80	11/23/83	Use of Specialized “Stiff” Pipe Clamps	
84-09	2/13/84	Lessons Learned from NRC Inspections of Fire Protection Safe Shutdown Systems (10CFR50, App. R)	
84-09, Rev. 1	3/7/84	Lessons Learned from NRC Inspections of Fire Protection Safe Shutdown Systems (10CFR50, App. R)	
84-10	2/24/84	Motor-Operated Valve Torque Switches Set Below the Manufacturer’s Recommended Value	COL Applicant
84-17	3/5/84	Problems with Liquid Nitrogen Cooling Components Below the Nil Ductility Temperature	
84-22	3/29/84	Deficiency in Comsip, Inc. Standard Bed Catalyst	
84-23	4/5/84	Results of the NRC-Sponsored Qualification Methodology on ASCO Solenoid Valves	
84-32	4/18/84	Auxiliary Feedwater Sparger and Pipe Hanger Damage	
84-35	4/23/84	BWR Post-Scram Drywell Pressurization	
84-38	5/17/84	Problems With Design, Maintenance, and Operation of Offsite Power Systems	
84-47	6/15/84	Environmental Qualification Tests of Electrical Terminal Blocks	
84-67	8/17/84	Recent Snubber Inservice Testing With High Failure Rates	COL Applicant
84-69	8/29/84	Operation of Emergency Diesel Generators	COL Applicant
84-69, Supp. 1	2/24/86	Operation of Emergency Diesel Generators	COL Applicant
84-70	9/4/84	Reliance on Water Level Instrumentation with a Common Reference Leg	COL Applicant
84-70, Supp. 1	8/26/85	Reliance on Water Level Instrumentation with a Common Leg	COL Applicant
84-76	10/19/84	Loss of All AC Power	

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
84-87	12/3/84	Piping Thermal Deflection Induced by Stratified Flow	
84-88	12/3/84	Standby Gas Treatment System Problems	
84-93	12/17/84	Potential for Loss of Water from the Refueling Cavity	
85-08	1/30/85	Industry Experience on Certain Materials Used in Safety-Related Equipment	
85-13	2/21/85	Consequences of Using Soluble Dams	COL Applicant
85-17	4/1/85	Possible Sticking of ASCO Solenoid Valves	
85-17, Supp. 1	10/1/85	Possible Sticking of ASCO Solenoid Valves	
85-24	3/26/85	Failures of Protective Coatings in Pipes and Heat Exchangers	COL Applicant
85-25	4/2/85	Consideration of Thermal Conditions in the Design and Installation of Supports for Diesel Generator Exhaust Silencers	
85-28	4/9/85	Partial Loss of AC Power and Diesel Generator Degradation	
85-30	4/19/85	Microbiologically Induced Corrosion of Containment Service Water System	
85-32	4/22/85	Recent Engine Failures of Emergency Diesel Generators	
85-33	4/22/85	Undersized Nozzle-to-Shell Welded Joints in Tanks and Heat Exchangers Constructed Under the Rules of the ASME Boiler and Pressure Vessel Code	
85-34	4/30/85	Heat Tracing Contributes to Corrosion Failure of Stainless Steel Piping	COL Applicant
85-35	4/30/85	Failure of Air Check Valves to Seat	
85-35, Supp. 1	5/17/88	Failure of Air Check Valves to Seat	
85-47	6/18/85	Potential Effect of Line-Induced Vibration on Certain Target Rock Solenoid-Operated Valves	
85-51	7/10/85	Inadvertent Loss of Improper Actuation of Safety-Related Equipment	COL Applicant
85-59	7/17/85	Valve Stem Corrosion Failures	
85-66	8/7/85	Discrepancies Between As-Built Construction Drawings and Equipment Installations	COL Applicant
85-76	9/19/85	Recent Water Hammer Events	
85-77	9/20/85	Possible Loss of Emergency Notification System Due to Loss of AC Power	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
85-81	10/17/85	Problems Resulting in Erroneously High Reading With Thermoluminscent Dosimeters	COL Applicant
85-84	10/30/85	Inadequate Inservice Testing of Main Steam Isolation Valves	
85-85	10/31/85	Systems Interaction Event Resulting in Reactor System Safety/Relief Valve Opening Following a Fire-Protection Deluge System Malfunction	
85-86	11/5/85	Lightning Strikes at Nuclear Power Generating Stations	
85-87	11/18/85	Hazards of Inerting Atmospheres	COL Applicant
85-89	11/19/85	Potential Loss of Solid-State Instrumentation Following Failure or Control Room Cooling	Subsection 19B.2.40
85-90	11/19/85	Use of Sealing Compounds in an Operating Plant	COL Applicant
85-91	11/27/85	Load Sequencers for Emergency Diesel Generators	COL Applicant
85-92	12/2/85	Surveys of Wastes Before Disposal From Nuclear Reactor Facilities	COL Applicant
85-94	12/13/85	Potential for Loss of Minimum Flow Paths Leading to ECCS Pump Damage During a LOCA	
85-96	12/23/85	Temporary Strainers Left Installed in Pump Suction Piping	COL Applicant
86-01	1/6/86	Failure of Main Feedwater Check Valves Causes Loss of Feedwater System Integrity and Water-Hammer Damage	
86-03	1/14/86	Potential Deficiencies in Environmental Qualification of Limitorque Motor Valve Operator Wiring	
86-09	2/3/86	Failure of Check and Stop Valves Subjected to Low Flow Conditions	
86-10	2/13/86	Safety Parameter Display System Malfunctions	
86-29	4/25/86	Effects of Changing Valve Motor-Operator Switch Settings Past Related Correspondence: IE Bulletin 85-03	COL Applicant
86-30	4/29/86	Design Limitations of Gaseous Effluent Monitoring System	
86-39	5/20/86	Failures of RHR Pump Motors and Pump Internals	
86-43	6/10/86	Problems with Silver Zeolite Sampling of Airborne Radioiodine	COL Applicant
86-48	6/13/86	Inadequate Testing of Boron Solution Concentration in the Standby Liquid Control System	

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
86-50	6/18/86	Inadequate Testing to Detect Failures of Safety-Related Pneumatic Components or Systems Past Related Correspondence: IE Notices 82-25, 85-35, 85-84, 85-94	
86-51	6/18/86	Excessive Pneumatic Leakage in the Automatic Depressurization System Past Related Correspondence: IE Bulletins 80-01, 80-25; IE Notice 85-35; IE Inspection Report 50-458/84-18 (8/16/84)	
86-53	6/26/86	Improper Installation of Heat Shrinkable Tubing	COL Applicant
86-57	7/11/86	Operating Problems With Solenoid-Operated Valves at Nuclear Power Plants	
86-60	7/28/86	Unanalyzed Post-LOCA Release Paths Past Related Correspondence: NUREG-0737	
86-68	8/15/86	Stuck Control Rod	
86-70	8/18/86	Potential Failure of All Emergency Diesel Generators	
86-71	8/19/86	Recent Identified Problems With Limitorque Motor Operators Past Related Correspondence: IE Notice 86-03	
86-76	8/20/86	Problems Noted in Control Room Emergency Ventilation Systems Past Related Correspondence: Item III D.3.4 of NUREG-0737 Generic Issue 83, IE Notice 85-89	Subsection 19B.2.40
86-83	9/16/86	Underground Pathways into Protected Areas, Vital Areas, Material Access Areas, and Controlled Access Areas Past Related Correspondence: NUREG-0908, ANSI 3.3	COL Applicant
86-87	10/10/86	Loss of Offsite Power Upon An Automatic Bus Transfer	
86-89	10/16/86	Uncontrolled Rod Withdrawal Because of A Single Failure	
86-96	11/20/86	Heat Exchanger Fouling Can Cause Inadequate Operability of Service Water Systems Past Related Correspondence: IE Bulletin 81-03, IE Notice 81-21	COL Applicant
86-100	12/12/86	Loss of Offsite Power to Vital Buses at Salem 2	
86-104	12/16/86	Unqualified Butt Splice Connectors Identified in Qualified Penetrations	
86-106	12/16/86	Feedwater Line Break	

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
86-106, Supp. 1	2/13/87	Feedwater Line Break Past Related Correspondence: E Notice 82-22 EPRI Report NP-3944, 4/85	
86-106, Supp. 2	3/18/87	Feedwater Line Break	
86-106, Supp. 3	10/10/88	Feedwater Line Break	
86-109	12/29/86	Diaphragm Failure in Scram Outlet Valve Causing Rod Insertion Past Related Correspondence: IE Notice 86-08	COL Applicant
87-06	1/30/87	Loss of Suction to Low-Pressure Service Water System Pumps Resulting From Loss of Siphon	COL Applicant
87-08	2/4/87	Degraded Motor Leads in Limitorque DC Motor Operators Past Related Correspondence: (Unrelated problems involving wiring installed in Limitorque motor actuators) IE Notices 83-72, 86-03 and 86-71	
87-09	2/5/87	Emergency Diesel Generator Room Cooling Deficiency Past Related Correspondence: IE Notice 86-50, 86-51 and 86-89	
87-10	2/11/87	Potential for Water Hammer During Restart of Residual Heat Removal Pumps Past Related Correspondence: AEOD/E309, 4/83	
87-13	2/24/87	Potential For High Radiation Fields Following Loss of Water From Fuel Pool Past Related Correspondence: IE Notice 84-93, IE Bulletin 84-03	
87-14	3/23/87	Actuation of Fire Suppression System Causing Inoperability of Safety-Related Ventilation Equipment Past Related Correspondence: IE Notice 83-41, 85-85, 86-106 Supp. 2	
87-28	6/22/87	Air Systems Problems at U.S. Light Water Reactors Past Related Correspondence: AEOD-C701	
87-28, Sup. 1	12/28/88	Air Systems Problems at U.S. Light Water Reactors Past Related Correspondence: AEOD-C701 NUREG-1275 Vol. 2	
87-36	8/4/87	Significant Unexpected Erosion of Feedwater Lines Past Related Correspondence: IE Notice 82-22, 86-106 plus Supp. 1&2 IE Bulletin 87-01	

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
87-43	9/8/87	Gaps in Neutron-Absorbing Material in High-Density Spent Fuel Storage Racks Past Related Correspondence: EPRI NP-4724	
87-49	10/9/87	Deficiencies in Outside Containment Flooding Protection	
87-50	10/9/87	Potential LOCA at High- and Low-Pressure COL Applicants from Fire Damage	
87-59	11/17/87	Potential RHR Pump Loss	
88-01	1/27/88	Safety Injection Pipe Failure	
88-04	2/5/88	Inadequate Qualification and Documentation of Fire Barrier Penetration Seals Past Related Correspondence: 10CFR50 Appendix R, Appendix A to BTP APCSB 9.5-1, NUREG-0800, ASTM E-119, BTP CMEB 9.5-1, Generic Letter 86-10	
88-04, Supp. 1	8/9/88	Inadequate Qualification and Documentation of Fire Barrier Penetration Seals	
88-05	2/12/88	Fire in Annunciator Control Cabinets	
88-12	4/12/88	Overgreasing of Electrical Motor Bearings Past Related Correspondence: LER 387/84-036	COL Applicant
88-13	4/18/88	Water Hammer and Possible Piping Damage Caused by Misapplication of Kerotest Packless Metal Diaphragm Globe Valves	
88-17	4/22/88	Summary of Responses to NRC Bulletin 87-01, “Thinning of Pipe Walls in Nuclear Power Plants” Past Related Correspondence: IE Bulletin 87-01; IE Notice 82-22, 86-106, 87-36	
88-21	5/9/88	Inadvertent Criticality Events at Oskarshamn and at U.S. Nuclear Power Plants	COL Applicant
88-24	5/13/88	Failures of Air-Operated Valves Affecting Safety-Related Systems Past Related Correspondence: IE Notice 87-28 & Supp. 1, NUREG-1275	
88-27	5/18/88	Deficient Electrical Terminations Identified in Safety-Related Components	COL Applicant
88-35	6/3/88	Inadequate Licensee Performed Vendor Audits Past Related Correspondence: IE Bulletin 88-05	COL Applicant
88-37	6/14/88	Flow Blockage of Cooling Water to Safety System Components Past Related Correspondence: IE Notice 81-21, 86-96; IE Bulletin 81-03	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
88-39	6/15/88	LaSalle Unit 2 Loss of Recirculation Pumps With Power Oscillation Event Past Related Correspondence: Generic Issue B-19, Generic Letter 86-02	
88-43	6/23/88	Solenoid Valve Problems Past Related Correspondence: IE Notices 85-17 & Supp. 1, 86-57; IE Circular 81-14	
88-51	7/21/88	Failures of Main Steam Isolation Valves	
88-61	8/11/88	Control Room Habitability-Recent Reviews of Operating Experience	Subsection 19B.2.40
88-63	8/15/88	High Radiation Hazards from Irradiated Incore Detectors and Cables	COL Applicant
88-65	8/18/88	Inadvertent Drainings of Spent Fuel Pools	
88-70	8/29/88	Check Valve Inservice Testing Program Deficiencies Past Related Correspondence: IE Notice 86-01, Generic Letter 87-06	
88-72	9/2/88	Inadequacies in the Design of DC Motor-Operated Valves	
88-76	9/19/88	Recent Discovery of a Phenomenon Not Previously Considered in the Design of Secondary Containment Pressure Control Past Related Correspondence: NUREG-0800	
88-77	9/22/88	Inadvertent Reactor Vessel Overfill	
88-81	10/7/88	Failure of AMP Window Indent Kynar Splices and Thomas and Betts Nylon Wire Caps During Environmental Qualification Testing	
88-85	10/14/88	Broken Retaining Block Studs on Anchor Darling Check Valves	
88-86	10/21/88	Operating with Multiple Grounds in Direct Current Distribution Systems and Supplement 1	
88-89	11/21/88	Degradation of Kapton Electrical Insulation Past Related Correspondence: IE Notices 87-08, 87-16	
88-92	11/22/88	Potential for Spent Fuel Pool Draindown	
88-95	12/8/88	Inadequate Procurement Requirements Imposed by Licensees on Vendors	COL Applicant
89-01	1/4/89	Valve Body Erosion Past Related Correspondence: IE Notice 88-17	
89-04	1/17/89	Potential Problems from the Use of Space Heaters	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
89-07	1/25/89	Failures of Small-Diameter Tubing in Control Air, Fuel Oil, and Lube Oil Systems Which Render Emergency Diesel Generators Inoperable	
89-08	1/26/89	Pump Damage Caused by Low-Flow Operation	
89-10	1/27/89	Undetected Installation Errors in Main Steam Line Pipe Tunnel Differential Temperature Sensing Elements at Boiling Water Reactors	
89-11	2/2/89	Failure of DC Motor-Operated Valves to Develop Rated Torque Because of Improper Cabling Sizing	
89-14	2/16/89	Inadequate Dedication Process for Commercial Grade Components Which Could Lead to Common Mode Failure of a Safety System	
89-16	2/16/89	Excessive Voltage Drop in DC Systems Past Related Correspondence: Generic Letter 88-15	
89-17	2/22/89	Contamination and Degradation of Safety-Related Battery Cells	
89-20	2/24/89	Weld Failures in a Pump of Byron-Jackson Design	
89-21	2/27/89	Changes in Performance Characteristics of Molded Case Circuit Breakers	
89-26	3/7/89	Instrument Air Supply to Safety-Related Equipment Past Related Correspondence: Generic Letter 88-14	
89-30	3/15/89	High Temperature Environments at Nuclear Power Plants	
89-36	4/4/89	Excessive Temperatures in Emergency Core Cooling System Piping Located Outside Containment	
89-37	4/4/89	Proposed Amendments to 40CFR Part 61, Air Emission Standards for Radionuclides	
89-39	4/5/89	List of Parties Excluded from Federal Procurement of Non-procurement Programs	COL Applicant
89-52	6/8/89	Potential Fire Damper Operational Problems	
89-61	8/30/89	Failure of Borg-Warner Gate Valves to Close Against Differential Pressure	
89-63	9/5/89	Possible Submergence of Electrical Circuits Located Above the Flood Level Because of Water Intrusion and Lack of Drainage	COL Applicant
89-64	9/7/89	Electrical Bus Bar Failures	COL Applicant
89-66	9/11/89	Qualification Life of Solenoid Valves	

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
89-68	9/25/89	Evaluation of Instrument Setpoints During Modifications	COL Applicant
89-69	9/29/89	Loss of Thermal Margin Caused by Channel Box Bow	COL Applicant
89-70	10/11/89	Possible Indications of Misrepresented Vendor Products	COL Applicant
89-71	10/19/89	Diversion of the Residual Heat Removal Pump Seal Cooling Water Flow During Recirculation Operation Following a Loss-of-Coolant Accident	
89-72	10/24/89	Failure of Licensed Senior Operators to Classify Emergency Events Properly	COL Applicant
89-73	11/1/89	Potential Overpressurization of Low Pressure Systems	COL Applicant
89-76	11/21/89	Biofouling Agent: Zebra Mussel	COL Applicant
89-77	11/21/89	Debris in Containment Emergency Sumps and Incorrect Screen Configurations	
89-79	12/1/89	Degraded Coatings and Corrosion of Steel Containment Vessels	
89-80	12/1/89	Potential for Water Hammer, Thermal Stratification, and Steam Binding in High-Pressure Coolant Injection Piping	
89-81	12/6/89	Inadequate Control of Temporary Modifications to Safety-Related Systems	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
Type: IE Information Notices			
89-83	12/11/89	Sustained Degraded Voltage on the Offsite Electrical Grid and Loss of Other Generating Stations as a Result of a Plant Trip	COL Applicant
89-87	12/19/89	Disabling of Emergency Diesel Generators by Their Neutral Ground-Fault Protection Circuitry	
89-88	12/16/89	Recent NRC-Sponsored Testing of Motor-Operated Valves	
90-02	01/22/90	Potential Degradation of Secondary Containment	
90-05	01/29/90	Inter-System Discharge of Reactor Coolant	
90-07	01/30/90	New Information Regarding Insulation Material Performance and Debris Blockage of PWR Containment Sumps	
90-8	02/01/90	KR-85 Hazards From Decayed Fuel	
90-13	03/05/90	Importance of Review and Analysis of Safeguards Event Logs	COL Applicant
90-20	03/22/90	Personnel Injuries Resulting From Improper Operation of Radwaste Incinerators	COL Applicant
90-21	03/22/90	Potential Failure of Motor-Operated Butterfly Valves to Operate Because Valve Seat Friction was Underestimated	COL Applicant
90-22	03/23/90	Unanticipated Equipment Actuation Following Restoration of Power to Rosemount Transmitter Trip Units	COL Applicant
90-25	04/16/90	Loss of Vital AC Power With Subsequent Reactor Coolant System Heatup	COL Applicant
90-25 Supp.1	03/11/90	Loss of Vital AC Power With Subsequent Reactor Coolant System Heatup	COL Applicant
90-26	04/24/90	Inadequate Flow of Essential Service Water to Room Coolers and Heat Exchangers for Engineered Safety-Feature Systems	COL Applicant
90-30	05/01/90	Ultrasonic Inspection Techniques for Dissimilar Metal Welds	
90-33	05/09/90	Sources of Unexpected Occupational Radiation Exposure at Spent Fuel Storage Pools	COL Applicant
90-39	06/01/90	Recent Problems with Service Water Systems	COL Applicant
90-40	06/05/90	Results of NRC-Sponsored Testing of Motor-Operated Valves	COL Applicant
90-42	06/19/90	Failure of Electrical Power Equipment Due to Solar Magnetic Disturbances	
90-47	07/27/90	Unplanned Radiation Exposures to Personnel Extremities Due to Improper Handling of Potentially Highly Radioactive Sources	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
90-50	08/08/90	Minimization of Methane Gas in Plant Systems and Radwaste Shipping Containers	COL Applicant
90-53	08/16/90	Potential Failures of Auxiliary Steam Piping and the Possible Effects on the Operability of Vital Equipment	
90-54	08/28/90	Summary of Requalification Program Deficiencies	COL Applicant
90-61	09/20/90	Potential for Residual Heat Removal Pump Damage Caused by Parallel Pump Interaction	
90-63	10/03/90	Management Attention to the Establishment and Maintenance of a Nuclear Criticality Safety Program	COL Applicant
90-67	10/29/90	Potential Security Equipment Weaknesses	
90-68	10/30/90	Stress Corrosion Cracking of Reactor Coolant Pump Bolts	
90-69	10/31/90	Adequacy of Emergency and Essential Lighting	
90-70	11/06/90	Pump Explosions Involving Ammonium Nitrate	
90-72	11/28/90	Testing of Parallel Disc Gate Valves in Europe	
90-74	12/04/90	Information on Precursors to Severe Accidents	
90-78	12/18/90	Previously Unidentified Release Path From Boiling Water Reactor Control Rod Hydraulic Units	
90-81	12/24/90	Fitness For Duty	COL Applicant
90-82	12/31/90	Requirements For Use of Nuclear Regulatory Commission-(NRC)-Approved Transport Packages For Shipment of Type A Quantities of Radioactive Material	COL Applicant
91-04	01/28/91	Reactor Scram Following Control Rod Withdrawal Associated With Low Power Turbine Testing	
91-06	01/31/91	Lockup of Emergency Diesel Generator and Load Sequencer Control Circuits Preventing Restart of Tripped Emergency Diesel Generator	
91-12	02/15/91	Potential Loss of Net Positive Suction Head (NPSH) of Standby Liquid Control System Pumps	
91-13	03/04/91	Inadequate Testing of Emergency Diesel Generators (EDGs)	
91-14	03/05/91	Recent Safety-Related Incidents at Large Irradiators	
91-17	03/11/91	Fire Safety of Temporary Installation of Services	COL Applicant
91-19	03/12/91	High-Energy Piping Failures Caused by Wall Thinning	

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
91-22	03/19/91	Four Plant Outage Events Involving Loss of AC Power or Coolant Spills	
91-23	03/26/91	Accident Radiation Overexposures to Personnel Due to Industrial Radiography Accessory Equipment Malfunctions	COL Applicant
91-29	04/15/91	Deficiencies Identified During Electrical Distribution System Functional Inspections	
91-33	05/31/91	Reactor Safety Information for States During Exercises and Emergencies	COL Applicant
91-34	06/03/91	Potential Problems in Identifying Causes of Emergency Diesel Generator Malfunctions	
91-37	06/10/91	Compressed Gas Cylinder Missile Hazards	COL Applicant
91-38	06/13/91	Thermal Stratification in Feedwater System Piping	
91-40	06/19/91	Contamination of Nonradioactive System and Resulting Possibility for Unmonitored, Uncontrolled Release to the Environment	COL Applicant
91-41	06/27/91	Potential Problems with the Use of Freeze Seals	COL Applicant
91-42	07/27/91	Plant Outage Events Involving Poor Coordination Between Operations and Maintenance Personnel During Valve Testing and Manipulations	COL Applicant
91-46	07/18/91	Degradation of Emergency Diesel Generator Fuel Oil Delivery Systems	COL Applicant
91-47	08/06/91	Failure of Thermo-Lag Fire Barrier Material to Pass Fire Endurance Test	
91-49	08/15/91	Enforcement of Safety Requirements for Radiographers	COL Applicant
91-50	08/20/91	A Review of Water Hammer Events After 1985	
91-51	08/20/91	Inadequate Fuse Control Programs	COL Applicant
91-57	09/19/91	Operational Experience on Bus Transfers	
91-58	09/20/91	Dependency of Offset Disc Butterfly Valve's Operation of Orientation With Respect to Flow	
91-59	09/23/91	Problems With Access Authorization Programs	COL Applicant
91-60	11/01/91	Reissuance of Information Notice 91-60: False Alarms of Alarm Ratemeters Because of Radio Frequency Interference	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
91-61	09/30/91	Preliminary Results of Validation Testing of Motor-Operated Valve Diagnostic Equipment	
91-63	10/03/91	Natural Gas Hazards at Fort St. Vrain Nuclear Generating Station	COL Applicant
91-64	10/09/91	Site Area Emergency Resulting From a Loss of Non-Class 1E Uninterruptable Power Supplies	
91-65	10/17/91	Emergency Access to Low-Level Radioactive Waste Disposal Facilities	COL Applicant
91-66	10/18/91	(1) Erroneous Date in “Nuclear Safety Guide, TID-7016, Revision 2,” (NUREG/CR-0095, ORNL/NUREG/CSD-6 (1978) And (2) Thermal Scattering Data Limitation in the Cross-Section Sets Provided With the Keno and Scale Codes	
91-68	10/28/91	Careful Planning Significantly Reduces the Potential Adverse Impacts of Loss of Offsite Power Events During Shutdown	COL Applicant
91-72	11/19/91	Issuance of a Revision to the EPA Manual of Protective Action Guides and Protective Actions for Nuclear Incidents	
Type: IE Circulars			
80-03	3/6/80	Protection from Toxic Gas Hazards	COL Applicant
80-05	4/1/80	Emergency D/G Lube Oil	COL Applicant
80-08	4/18/80	RPS Response Time	
80-09	4/28/80	Problems with Plant Internal Communications Systems	COL Applicant
80-10	4/29/80	Failure to Maintain Environmental Qualification of Equipment	COL Applicant
80-11	5/13/80	Emergency Diesel Generator Lube Oil Cooler Failures	COL Applicant
80-14	6/24/80	Radioactive Contamination of Demin Water System	COL Applicant
80-18	8/22/80	10 CFR 50.59 Safety Evaluation for Changes to Radioactive Waste Treatment Systems	COL Applicant
81-03	3/2/81	Inoperable Seismic Monitoring Instrument	COL Applicant
81-05	3/31/81	Self-Aligning Rod End Bushing for Pipe Supports	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

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
0654	10/80	Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants	COL Applicant
0696	12/80	Functional Criteria for Emergency Response Facilities	COL Applicant
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	

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
0744 Rev. 1	10/82	Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue	
0800	7/81	Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition	
0808	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	
0933	4/93	A prioritization of Generic Safety Issues	Appendix 19B
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	
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

Table 1.8-22 Experience Information Applicable to ABWR (Initial Certification)

No.	Issue Date	Title	Comment
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
CR-4261	3/86	Assessment of Systems Interactions in Nuclear Power Plants	Subsection 19B.2.59
CR-4262	5/85	Effects of Control System Failures on Transients, Accidents at a GE BWR, Vol. 1 and 2	
CR-4387	12/85	Effects of Control System Failures on Transient and Accidents and Core-Melt Frequencies at a GE BWR	
CR-4470	5/86	Survey and Evaluation of Vital Instrumentation and Control Power Supply Events	
CR-5055	5/88	Atmospheric Diffusion for Control Room Habitability Assessment	Subsection 19B.2.40
CR-5088	1/89	Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues.	
CR-5230	4/89	Shutdown Decay Heat Removal Analysis: Plant Case Studies and Special Issues	
CR-5347	6/89	Recommendations for Resolution of Public Comments on USI A-40	Subsection 19B.2.14
CR-5458	12/89	Value-Impact Assess for Candidate Operating Procedure Upgrade Program	
CR-4674	84/89	Precusors to Potential Severe Core Damage Accidents: Series	

Table 1.8-23 Significant Operating Experience Information Applicable to ABWR (Certification Renewal)

No.	Issue Date	Title	Comment
Type: NRC Generic Letters			
94-03	7/25/94	Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors	
95-07	8/17/95	Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves	COL Applicant
96-06, Including Supplement 1	9/30/96, 11/13/97	Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions	COL Applicant
97-04	10/7/97	Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps	COL Applicant
98-04	7/14/98	Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-Of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment	COL Applicant
2003-01	6/12/03	Control Room Habitability	COL Applicant
2008-01	1/11/08	Managing Gas Accumulation in Emergency Core Cooling, Residual Heat Removal, and Containment Spray Systems	
Type: NRC Bulletins			
93-02, Including Supplement 1	5/11/93, 2/18/94	Debris Plugging of Emergency Core Cooling Suction Strainers	COL Applicant
93-03	5/28/93	Resolution of Issues Related to Reactor Vessel Water Level Instrumentation in BWRs	COL Applicant
96-03	5/6/96	Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors	COL Applicant

Table 1.8-23 Significant Operating Experience Information Applicable to ABWR (Certification Renewal)

No.	Issue Date	Title	Comment
Type: NRC Information Notices			
92-26	4/2/92	Pressure Locking of Motor-Operated Flexible Wedge Gate Valves	COL Applicant
92-36, Including Supplement 1	5/7/92, 2/22/94	Intersystem LOCA Outside Containment	
92-71	9/30/92	Partial Plugging of Suppression Pool Strainers at a Foreign BWR	COL Applicant
92-85	12/23/92	Potential Failures of Emergency Core Cooling Systems Caused by Foreign Material Blockage	COL Applicant
93-27	4/8/93	Level Instrumentation Inaccuracies Observed During Normal Plant Depressurization	COL Applicant
93-34, Including Supplement 1	4/26/93, 5/6/93	Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment	COL Applicant
93-35	5/12/93	Insights from Common-Cause Failure Events	COL Applicant
93-53, Including Supplement 1	7/20/93, 4/29/94	Effect of Hurricane Andrew on Turkey Point Nuclear Generating Station and Lessons Learned	COL Applicant
93-82	10/12/93	Recent Fuel and Core Performance Problems in Operating Reactors	
93-89	11/26/93	Potential Problems with BWR Level Instrumentation Backfill Modifications	COL Applicant
95-10, Including Supplements 1 and 2	2/3/95, 2/10/95, 8/11/95	Potential for Loss of Automatic Engineered Safety Features Actuation	
95-16	3/9/95	Vibration Caused by Increased Recirculation Flow in a Boiling Water Reactor	

Table 1.8-23 Significant Operating Experience Information Applicable to ABWR (Certification Renewal)

No.	Issue Date	Title	Comment
95-18, Including Supplement 1	3/15/95, 3/31/95	Potential Pressure-Locking of Safety-Related Power-Operated Gate Valves	COL Applicant
95-30	8/3/95	Susceptibility of Low-Pressure Coolant Injection and Core Spray Injection Valves to Pressure Locking	COL Applicant
97-19	4/18/97	Safety Injection System Weld Flaw at Sequoyah Nuclear Power Plant, Unit 2	COL Applicant
97-22	4/25/97	Failure of Welded-Steel Moment-Resisting Frames During the Northridge Earthquake	COL Applicant
97-81	11/24/97	Deficiencies in Failure Modes and Effects Analyses for Instrumentation and Control Systems	
97-84	12/11/97	Rupture in Extraction Steam Piping as a Result of Flow-Accelerated Corrosion	
2000-20	12/11/00	Potential Loss Of Safety-Related Equipment Because of the Lack of High-Energy Line Break Barriers	
2001-15	10/29/01	Non-Conservative Errors in Minimum Critical Power Ratio Limits	
2005-30	11/7/05	Safe Shutdown Potentially Challenged by Unanalyzed Internal Flooding Events and Inadequate Design	
2006-21	9/21/06	Operating Experience Regarding Entrainment of Air Into Emergency Core Cooling and Containment Spray Systems	
2009-16	9/15/09	Spurious Relay Actuations Result in Loss of Power to Safeguards Buses	COL Applicant
2009-29	11/24/09	Potential Failure of Fire Water Supply Pumps to Automatically Start Due to a Fire	COL Applicant
2010-02	1/28/10	Construction-Related Experience with Cables, Connectors, and Junction Boxes	COL Applicant

Table 1.8-23 Significant Operating Experience Information Applicable to ABWR (Certification Renewal)

No.	Issue Date	Title	Comment
Type: NRC NUREGs			
1275, Volume 9	3/93	Operating Experience Feedback Report - Pressure Locking and Thermal Binding of Gate Valves	COL Applicant
1544	3/96	Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components	
CP-0146	7/95	Proceedings of the Workshop on Gate Valve Pressure Locking and Thermal Binding	COL Applicant
CR-4667, Volumes 13-36	3/92 through 8/07	Environmentally Assisted Cracking in Light Water Reactors, Semiannual Reports	COL Applicant
CR-5675	3/99	Residual Stresses and Associated Stress Intensity Factors in Core Shroud Weldments	
CR-5754	9/93	Boiling-Water Reactor Internals Aging Degradation Study, Phase 1	
CR-6049	8/93	Piping Benchmark Problems for the General Electric Advanced Boiling Water Reactor	
CR-6121	7/94	Component Evaluation for Intersystem Loss-of-Coolant Accidents in Advanced Light Water Reactors	
CR-6223	6/94	Review of the Proposed Materials of Construction for the SBWR and AP600 Advanced Reactors	COL Applicant
CR-6224	10/95	Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris	COL Applicant
CR-6677	7/00	Evaluation of Risk Associated with Intergranular Stress Corrosion Cracking in Boiling Water Reactor Internals	
CR-6687	10/00	Irradiation-Assisted Stress Corrosion Cracking of Model Austenitic Stainless Steel Alloys	
CR-6717	5/01	Environmental Effects on Fatigue Crack Initiation in Piping and Pressure Vessel Steels	COL Applicant

Table 1.8-23 Significant Operating Experience Information Applicable to ABWR (Certification Renewal)

No.	Issue Date	Title	Comment
CR-6721	4/01	Effects of Alloy Chemistry, Cold Work, and Water Chemistry on Corrosion Fatigue and Stress Corrosion Cracking of Nickel Alloy Welds	
CR-6787	7/02	Mechanism and Estimation of Fatigue Crack Initiation in Austenitic Stainless Steels in LWR Environments	COL Applicant
CR-6826	8/03	Fracture Toughness and Crack Growth Rates of Irradiated Austenitic Stainless Steels	
CR-6878	7/05	Effect of Material Heat Treatment on Fatigue Crack Initiation in Austenitic Stainless Steels in LWR Environments	COL Applicant
CR-6891	1/06	Crack Growth Rates of Irradiated Austenitic Stainless Steel Weld Heat Affected Zone in BWR Environments	
CR-6892	1/06	Irradiation-Assisted Stress Corrosion Cracking Behavior of Austenitic Stainless Steels Applicable to LWR Core Internals	
CR-6909	2/07	Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials	COL Applicant
CR-6960	3/08	Crack Growth Rates and Fracture Toughness of Irradiated Austenitic Stainless Steels in BWR Environments	
CR-6965	9/08	Irradiation-Assisted Stress Corrosion Cracking of Austenitic Stainless Steels and Alloy 690 from Halden Phase-II Irradiations	
CR-6967	7/08	Cladding Embrittlement during Postulated Loss-of-Coolant Accidents	COL Applicant
CR-6983	11/08	Seismic Analysis of Large-Scale Piping Systems for the JNES-NUPEC Ultimate Strength Piping Test Program	
CR-6991	9/09	Design Practices for Communications and Workstations in Highly Integrated Control Rooms	COL Applicant
CR-7011, Including Errata	5/10	Evaluation of Treatment of Effects of Debris in Coolant on ECCS and CSS Performance in Pressurized Water Reactors and Boiling Water Reactors	COL Applicant

Table 1.8-24 Japanese Operating Experience Information for the ABWR

Facility	Date	Title	Description	Toshiba Comment
Kashiwazaki-Kariwa Unit-7	May 21, 1997	Break of Pressure Sensing Piping for Performance Confirmation inside Low Pressure Turbine	With Unit-7 under trial operation at rated power, an abnormal noise was heard around the low pressure turbine (B). An investigation confirmed that the pressure sensing piping for performance (turbine inlet pressure) monitoring installed inside the low pressure turbine (B) casing was broken at a fillet weld. A similar crack was also observed at the same location in low pressure turbine (A). The cause of this event was attributed to thermal stress due to the temperature difference between the inner and outer side of the inner casing of the low pressure turbine, as well as cyclic stress due to mechanical vibration caused by turbine rotation and flowing steam around the location of the break. The inadequate treatment of the weld toe led to the concentration of these stresses with its level reaching the fatigue limit, resulting in the break of the concerned piping. Contact between the broken piping faces generated the abnormal noise. There was no release of radioactive material outside the reactor facilities, and no unplanned radiation exposure was received by employees.	The probable cause of this event is fatigue failure of fillet welds due to faulty workmanship. This event does not affect the standard ABWR design.
Kashiwazaki-Kariwa Unit-6	August 29, 1998	Automatic Reactor Shutdown due to Actuation of 500kV Pilot Wire Protection Relay	With Unit-6 operating at rated power, the reactor was automatically shut down because of generator and turbine trips that occurred due to the actuation of a protection relay for the cable (500kV) connecting the main transformer of Unit-6 to the station switchyard. The relay was actuated when lightning struck a transmission line (Kita-Tochigi Trunk Line). An investigation found that one of current detectors of the relevant relay was connected with reversed polarity. This mistake occurred because the installation of the relay was performed using an incorrect wiring diagram. The relay was erroneously actuated due to the input current to the associated cable caused by the lightning strike. There was no release of radioactive materials to the environment or unexpected personnel exposure.	The probable cause of this event is incorrect fabrication drawings. This event does not affect the standard ABWR design.
Kashiwazaki-Kariwa Unit-7	March 31, 1999	Leakage of Radioactive Material from Fuel Assembly	With Unit-7 operating at rated power, the indication of an off-gas radiation monitor and the concentration of radioactive iodine in the reactor coolant gradually increased. Since the iodine might have leaked from a fuel assembly, it was decided to perform a detailed inspection and investigation. During leakage testing of all (872) fuel assemblies loaded in the reactor core, the leakage of radioactive material from one fuel assembly was detected. No abnormality was found during a visual inspection of the fuel assembly; therefore, it was concluded that the leakage was due to a defect that was generated incidentally. There was no release of radioactive materials to the environment or unexpected personnel exposure.	The probable cause of this event is a defect of the fuel assembly generated incidentally. The specific nature and cause of the defect are not known based on the JNES report. However, the affected fuel assembly type is not used in the U.S. ABWR. Also, no fuel-related failures have occurred in Japanese ABWRs during the last 10 years. Therefore, this event does not affect the standard ABWR design.

Table 1.8-24 Japanese Operating Experience Information for the ABWR

Facility	Date	Title	Description	Toshiba Comment
Kashiwazaki-Kariwa Unit-6	May 25, 1999	Automatic Reactor Shutdown due to Generator Exciter Trip	With Unit-6 operating at rated power, the "generator exciter failure trip" alarm occurred. The generator and the turbine tripped and the reactor was automatically shut down. An investigation found that one of five power converters installed in the generator exciter (which supplies direct current to the exciter circuit of the generator) was tripped for inspection, and after control power of the relevant power converter was cut off, the exciter monitoring program recognized this cutoff of the control power as the stop of the second power converter, resulting in the exciter trip. It was confirmed that this event occurred because the trip signal of the power converter and the stop signal of the control power were input at the same time, and they were therefore processed as two signals. There was no release of radioactive materials to the environment or unexpected personnel exposure.	The probable cause of this event is an inadequate exciter monitoring program. This event does not affect the standard ABWR design.
Kashiwazaki-Kariwa Unit-7	July 28, 1999	Automatic Stop of Reactor Coolant Recirculation Pump	With Unit-7 operating at rated power, one of ten reactor coolant recirculation pumps stopped. An investigation determined that an electrical abnormality had occurred in either the concerned pump motor or a connecting cable. Therefore, it was decided to perform a detailed inspection and investigation. These found that one of the cable terminals of the concerned reactor coolant recirculation pump was broken. One of similar terminals was also found to be cracked for each of the other nine recirculation pumps. Examination of fracture surfaces found that there was a pattern (striations) consistent with a fatigue fracture. The cable holding plate of the broken terminal was also found loosened. Vibration analysis and fatigue testing found that the cause was vibration generated by rotation of the pump, which was transferred to the cable through the terminal box. The cable resonance with the vibrating frequency induced repeated stress on the terminal, leading to cracking and eventual failure. There was no release of radioactive materials to the environment or unexpected personnel exposure.	The probable cause of this event is fatigue failure caused by the vibration of cable with a loosened holding fixture. This event does not affect the standard ABWR design.

Table 1.8-24 Japanese Operating Experience Information for the ABWR

Facility	Date	Title	Description	Toshiba Comment
Kashiwazaki-Kariwa Unit-6	May 28, 2000	Leakage of Radioactive Material from Fuel Assemblies	With Unit-6 operating at rated power, the indication of an off-gas radiation monitor (which is a monitor installed between the condenser and the charcoal hold-up device) showed an increase. When the iodine concentration in the reactor coolant was measured, it was also recognized that the iodine concentration also showed an increase. Leakage from one or more fuel assemblies was suspected. Therefore, it was decided to perform an inspection and investigation. The reactor was manually shut down. As a result of a leak test of all (872) fuel assemblies loaded in the reactor core, it was recognized that radioactive materials were leaking from two fuel assemblies. Visual inspection of these two fuel assemblies revealed no abnormality. The leakage was attributed to defects that were generated accidentally. There was no release of radioactive materials to the environment or unexpected personnel exposure.	The probable cause of this event is a defect of the fuel assemblies generated accidentally. The specific nature and cause of the defects are not known based on the JNES report. However, the affected fuel assembly type is not used in the U.S. ABWR. Also, no fuel-related failures have occurred in Japanese ABWRs during the last 10 years. Therefore, this event does not affect the standard ABWR design.
Kashiwazaki-Kariwa Unit-6	June 18, 2001	Leakage of Reactor Auxiliary Water in Containment	With Unit-6 operating at rated power on May 20, 2001, the amount of high conductivity liquid waste in the reactor containment was observed to be gradually increasing. As a result of a cause investigation, it was presumed that the cooling water of the reactor auxiliary cooling water system in the reactor containment had leaked. Therefore, it was decided to manually shut down the reactor and inspect and repair the concerned portion. The reactor was shut down on June 18, 2001. An inspection revealed that cooling water had leaked from the gland portion of the valve that feeds the cooling water (the reactor auxiliary cooling water system) to one of three air conditioners in the reactor containment. During overhaul of the valve, the sealing function of the gland portion of the valve was found to be deteriorated due to degradation of the gland packing, which resulted in leakage. There was no release of radioactive materials to the environment or unexpected personnel exposure.	The probable cause of this event is the degradation of valve gland packing. This event does not affect the standard ABWR design.

Table 1.8-24 Japanese Operating Experience Information for the ABWR

Facility	Date	Title	Description	Toshiba Comment
Kashiwazaki-Kariwa Unit-7	April 26, 2002	Leakage of Radioactive Material from Fuel Assemblies	With Unit-7 operating at rated power, the indication of an off-gas radiation monitor (a monitor installed between the condenser and the charcoal hold-up device) was determined to have slightly increased since July 21, 2001. Since there was possibility of leakage of radioactive material from fuel assemblies to the inside of the reactor, all 872 fuel assemblies were inspected. This inspection found that radioactive material was leaking from two fuel assemblies. A review of operating history and visual inspection of the concerned fuel assemblies revealed no abnormality. Therefore, the leakage from the two fuel assemblies was attributed to defects that were generated accidentally, not due to deficiency of fuel design/manufacturing, fuel handling, and plant operation control. There was no release of radioactive materials to the environment or unexpected personnel exposure.	The probable cause of this event is a defect of the fuel assemblies generated accidentally. The specific nature and cause of the defects are not known based on the JNES report. However, the affected fuel assembly type is not used in the U.S. ABWR. Also, no fuel-related failures have occurred in Japanese ABWRs during the last 10 years. Therefore, this event does not affect the standard ABWR design.
Shika Unit-2	January 27, 2006	Manual Reactor Shutdown due to Malfunction of Steam Supply Isolation Valve of Reactor Core Isolation Cooling System	During the opening-and-closing test of steam supply isolation valves of the reactor core isolation cooling system, it was confirmed that an outer isolation valve was not fully closed. Although closing operation of the valve was confirmed, the reactor was manually shut down in order to conduct a detailed inspection of the valve from a preventive maintenance point of view. The inspection revealed that the electromagnetic contactor used to operate the motor for driving the valve was stuck at the "valve-open signal transmission" position and did not send the signal to initiate the closing motion of the valve. An investigation determined the cause to be surface deposits due to larger current flow with smaller contact point (smaller than the normal condition) and the prolonged chattering. Also the investigation found the contactor spring was out of position (displaced during cleaning work conducted prior to the test operation). Consequently, the signal of "valve-close" was not transmitted to the motor to operate the valve and the valve did not operate. There was no release of radioactive materials into the environment.	The probable cause of this event is the failure of an electromagnetic contactor due to poor workmanship. It does not indicate any impact on the standard ABWR design.

Table 1.8-24 Japanese Operating Experience Information for the ABWR

Facility	Date	Title	Description	Toshiba Comment
Hamaoka Unit-5	June 15, 2006	Automatic Reactor Shutdown due to Steam Turbine Failure	With Unit-5 operating at rated power, the turbine shaft vibration amplitude became large and then the turbine stopped. This resulted in automatic shutdown of the reactor. The turbine casing was opened and a visual inspection was conducted. The inspection found that one blade of the turbine wheel on the third stage from the outside of the generator side (the twelfth stage) of the low pressure turbine (B) was disconnected from the turbine shaft and had dropped into the lower part of the turbine. Breaks and cracks were recognized in the fork-shaped installation portions of the blades of the same stages of the other low pressure turbines. Blade cracking was attributed to a fluid excitation force (random vibration) that was generated during non-steady flow at no load and low load operation during the test operations. The fluid excitation force was also generated due to reverse steam flow (flush back) from the extraction pipe at the load rejection test during the test operations. These forces led to the generation of excessive repeated stress in the fork-shaped installment portions of the blades of the twelfth stage. Two rejection tests caused cyclic stresses in excess of the fatigue limit in the affected portion, and resulted in the initiation and growth of cracks in the blades. Regarding the one blade that failed during operation, the blade could not endure the centrifugal load during rated power operation. The sheared blade breakage occurred and the blade fell off from the circular plate portion. There was no radiological impact on the outside or inside of the power plant.	The probable cause of this event is the inadequate design of a vendor-specific turbine system. This event does not affect the standard ABWR design.
Hamaoka Unit-5	July 5, 2007	Malfunction of Average Power Range Monitor Channel	With Unit-5 under an adjustment operation, the alarm showing inoperability of an average power range monitor (APRM-B) was generated. A deviation from the limiting conditions for operation specified in the safety preservation rules of the reactor facility was simultaneously declared. The signal from the affected monitor was excluded, and then recovery from the limiting conditions for operation was declared. An investigation found that a signal showing “rapid reduction of reactor coolant flow rate”, whose exclusion was not allowed for, was also excluded at the same time the signal from the power range monitor was excluded. As a result, a deviation from the limiting conditions for operation was again declared, and the reactor thermal power was reduced to less than 75% pursuant to the requirements of the safety preservation rules. Following this, recovery from deviation from the limiting conditions for operation was again declared. An investigation attributed that the initial alarm of the malfunction in the APRM unit was caused either by the failure of the programmable logic device (PLD) or the failure of the watchdog time (WDT) elements in the CPU board. Both the APRM and CPU board were replaced. There was no radiological impact on the outside or inside of the power plant.	The probable cause of this event is the failure of a central processing unit (CPU), and does not indicate any impact on the standard ABWR design.

Table 1.8-24 Japanese Operating Experience Information for the ABWR

Facility	Date	Title	Description	Toshiba Comment
Kashiwazaki-Kariwa Unit-6	July 16, 2007	Overflow of Water to Refueling Floor of Reactor Building due to 2007 Niigataken Chuetsu-oki Earthquake	A walkdown conducted after the occurrence of the 2007 Niigataken Chuetsu-oki Earthquake discovered an overflow of water onto the refueling floor (controlled area) of the reactor building in Units 1 through 7. An inspection and investigation, which included observations by plant workers, surveillance video evidence, and an inspection of refueling floor equipment that could have leaked, concluded that the water on the refueling floor had flowed out from the spent fuel pool by sloshing due to the earthquake. There was no radiological impact on the outside or inside of the power plant.	The probable cause of this event is the inadequate sealing around the refueling floor. This event does not affect the standard ABWR design.
		Leakage of Water Containing Radioactive Material to Non-controlled Area in Reactor Building due to 2007 Niigataken Chuetsu-oki Earthquake	During a periodic inspection at Unit-6 after the occurrence of the 2007 Niigataken Chuetsu-oki Earthquake, a water puddle was identified in non-controlled areas of the reactor building. A sample was collected and the radioactivity was measured. It was confirmed that the leaked water contained radioactive materials. The total amount of leakage was approximately 1.5 liters, with an activity of approximately 16.3 x 10^3 Bq. An inspection and investigation concluded that water from the spent fuel pool leaked through electrical wire conduit and discharged into the sea via station drainage. The amount of discharged water was about 1.2 m^3, with an activity of approximately 9 x 10^4 Bq. The sea water monitor value did not show any significant fluctuation and the amount of discharged radioactivity was below the level provided by law; therefore, it was concluded that there was no impact to the environment. There was no radiological impact on the outside or inside of the power plant.	
Kashiwazaki-Kariwa Unit-6	July 24, 2007	Damage of Overhead Crane in Reactor Building due to 2007 Niigataken Chuetsu-oki Earthquake	During a periodic inspection at Unit-6 after the occurrence of the 2007 Niigataken Chuetsu-oki Earthquake, it was confirmed that two cross pins on the wheel side of the travel gears for the overhead crane in the reactor building were broken. As a result of a detailed investigation, damage to one cross pin at the electric motor side was also confirmed. The damage to the cross pins in the overhead crane was attributed to excessive forces on the crane transmission joints due to seismic motion. Since the overhead crane had not derailed, it was concluded that there was no risk of the crane falling from the ceiling. There was no radiological impact on the outside or inside of the power plant.	The probable cause of this event is the unintended motion of the overhead crane forced by earthquake. This event does not affect the standard ABWR design.
Kashiwazaki-Kariwa Unit-6	June 27, 2008	Defective Coupling between Control Rod Drive Mechanism and Control Rod	Unit-6 had been under its periodic inspection. During an operation test of the control rod drive mechanisms, it was confirmed that one control rod drive mechanism was not coupled with a control rod, and it was concluded that the concerned control rod drive mechanism did not have necessary functions. An investigation determined that the cause of the event was the inadequate coupling of the control rod drive mechanism to the corresponding control rod, as well as an inadequate confirmation test of the coupling. There was no radiological impact to the outside or inside of the power plant.	The probable cause of this event is inadequate procedure for the coupling of a control rod drive mechanism with a control rod. This event does not affect the standard ABWR design.

Table 1.8-24 Japanese Operating Experience Information for the ABWR

Facility	Date	Title	Description	Toshiba Comment
Hamaoka Unit-5	November 5, 2008	Manual Reactor Shutdown due to Temperature Increase of Noble Gas Hold-up Equipment in Off-gas Treatment System	Unit-5 had been in the reactor start-up operation. On November 5, 2008, it was confirmed that the temperature of the noble gas hold-up equipment in the off-gas treatment system had increased. Consequently, the reactor was manually shut down. Based on an investigation, the causes of the increase of temperature of the noble gas hold-up equipment in the off-gas treatment system were determined to be the following. The start-up operation had been conducted in an unstable region where [oxygen concentration] / [hydrogen concentration] ratio in the off-gas treatment system was below the threshold. Although oxygen supply was controlled to increase when hydrogen concentration increased, a recombination reaction rate did not increase and the hydrogen concentration exceeded the combustion limit (4%). A review meeting was held to review potential abnormal symptoms. Based on input from this meeting, an operation that did not comply with the alarm response manual etc. was continued in order to increase the recombination reaction rate in the off-gas recombiner, resulting in an increase in the hydrogen concentration to approximately 50%. As gas flow rate increased in the downstream of the off-gas recombiner, hydrogen ignited in the piping of the off-gas treatment system due to the reaction with ferrioxide etc. The combusting hydrogen spread to the charcoal in the noble gas hold-up equipment, resulting in the temperature increase. The temperature increase of the charcoal terminated due to the cut-off of the oxygen supply through isolation of the system after the reactor manual shutdown. There was no radiological impact to the outside or inside of the power plant.	The probable cause of this event is inadequate plant operation. This event does not affect the standard ABWR design.
Hamaoka Unit-5	December 30, 2008	Manual Reactor Shutdown due to Increase of Hydrogen Concentration in Off-gas Treatment System	Unit-5 had been in the reactor start-up operation for the adjustment operation during its periodic inspection. The performance of the off-gas recombiner was determined to be degraded as the hydrogen concentration of the off-gas treatment system increased and the outlet temperature of the off-gas recombiner tended to decrease. The reactor was manually shut down. An investigation determined the causes of the increased hydrogen concentration in the off-gas treatment system to be the result of (1) contamination of a platinum catalyst (used to recombine hydrogen and oxygen) by boehmite during manufacturing, and (2) further performance degradation of the catalyst by siloxane, which was used as a sealing material for the low pressure turbine and flowed into the off-gas treatment system. The increase in hydrogen concentration occurred as a result of the catalyst degradation. There was no radiological impact to the outside or inside of the power plant.	The probable cause of this event is degradation of the catalyst for the off-gas recombiner due to contamination during manufacturing and plant operation. This event does not affect the standard ABWR design.

Table 1.8-24 Japanese Operating Experience Information for the ABWR

Facility	Date	Title	Description	Toshiba Comment
Shika Unit-2	November 13, 2009	Emergency Diesel Generators Declared Inoperable due to Oil Contamination and Leakage	Unit-2 had been in the adjustment operation at the rated electric power during its periodic inspection. When a surveillance test of the emergency diesel generator (EDG)-A was carried out, approximately 100 cc of lubricating oil leaked from the valve used for checking for the presence of water or oil in a cylinder of the diesel engine. As a result, the EDG was determined to not be operable and "the deviation from the limiting condition for operation" was declared based on the safety regulation. During an investigation of the operability of the remaining two EDGs, approximately 4 cc of lubricating oil leaked from the similar location of EDG-B. The reactor was manually shut down as a result of the two EDGs being inoperable. The leakage of the lubricating oil was attributed to a decrease in the reclosing pressure of a pressure control check valve (which allowed the lubricating oil to flow into the cylinder and then leak out) that resulted from the contamination of the lubricant oil with a fine metallic powder (which caused wear of the sliding surface(s) of the valve disc of the pressure control check valve and resulted in an increase in sliding resistance). The lubricating oil leakage occurred due to the degradation of the pressure control check valve. There was no radiological impact to the outside or inside of the nuclear facility.	The probable cause of this event is performance degradation of the pressure control check valve due to lubricating oil contamination. This event does not affect the standard ABWR design.

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