NUREG–1557

Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

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Manuscript Completed: June 1996
Date Published: October 1996

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Abstract

In about 1990, the Nuclear Management and Resources Council (NUMARC) submitted for NRC review ten industry reports (IRs) addressing aging issues associated with specific structures and components of nuclear power plants and one IR addressing the screening methodology for integrated plant assessment. The NRC staff had been reviewing the ten NUMARC IRs; their comments on each IR and NUMARC responses to the comments have been compiled as public documents. This report provides a brief summary of the technical information and NUMARC/NRC agreements from the ten IRs, except for the Cable License Renewal IR. The technical information and agreements documented herein represent the status of the NRC staff's review when the NRC staff and industry resources were redirected to address rule implementation issues. The NRC staff plans to incorporate appropriate technical information and agreements into the draft standard review plan for license renewal.

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Executive Summary

In about 1990, the Nuclear Management and Resources Council (NUMARC) submitted for NRC review ten industry reports (IRs) addressing aging issues associated with specific structures and components of nuclear power plants and one IR addressing the screening methodology for integrated plant assessment (IPA). The ten IRs for specific structures and components included eight IRs on pressurized water reactor (PWR) and boiling water reactor (BWR) reactor vessel, containment, primary coolant pressure boundary, and vessel internals, and two IRs on class 1 structures and low-voltage, in-containment, environmentally qualified cable. The original intent of the IRs for specific structures and components was to serve as a referenceable surrogate for carrying out the IPA requirements of the license renewal rule.

In 1992, the NRC staff and industry resources were redirected to address implementation issues of the license renewal rule. The NRC staff recommended that appropriate technical information and agreements from the NUMARC IRs be incorporated into the draft standard review plan for license renewal (SRP-LR).

This report provides a brief summary of the technical information and NUMARC/NRC agreements from the ten IRs, except for the cable IR. The cable IR addresses the issue of environmental qualification (EQ) of electric equipment, which was superseded by the EQ action plan. The technical information and agreements documented herein represent the status of the NRC staff's review when the NRC staff and industry resources were redirected to address rule implementation issues.

The ten IRs, except for the cable IR, submitted by NUMARC addressing aging issues associated with specific structures and components of nuclear power plants have been reviewed. The technical information and NUMARC/NRC agreements for each IR have been compiled into tables. The information presented in each of the tables includes the following: specific structures and components and their materials of construction; ARDMs and their effects on structures and components; relevant comments of the NRC staff; and the NUMARC/NRC agreements or proposals and their technical basis, including assumptions and references. The technical information and agreements documented herein represent the status of the NRC staff's review when the NRC staff and industry resources were redirected to address rule implementation issues. The NRC staff plans to incorporate appropriate technical information and agreements into the draft SRP-LR.

Abbreviations, Acronyms, and Designations

ACI	American Concrete Institute
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARD	Aging–Related Degradation
ARDM	Aging–Related Degradation Mechanism
ASA	American Standards Association
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
B–A	ASME Section XI, Table IWB–2500–1, Examination Category for Pressure Retaining Welds in Reactor Vessel
B–B	ASME Section XI, Table IWB–2500–1, Examination Category for Pressure Retaining Welds in Vessels Other Than Reactor Vessel
B–D	ASME Section XI, Table IWB–2500–1, Examination Category for Full Penetration Welds of Nozzles in Vessels
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B–G–1	ASME Section XI, Table IWB–2500–1, Examination Category for Pressure Retaining Bolting, Greater Than 2 in. in Diameter
B–G–2	ASME Section XI, Table IWB–2500–1, Examination Category for Pressure Retaining Bolting, 2 in. and Less in Diameter
B–H	ASME Section XI, Table IWB–2500–1, Examination Category for Integral Attachments for Vessels
B–J	ASME Section XI, Table IWB–2500–1, Examination Category for Pressure Retaining Welds in Piping
B–K–1	ASME Section XI, Table IWB–2500–1, Examination Category for Integral Attachments for Piping, Pumps, and Valves
B–L–1	ASME Section XI, Table IWB–2500–1, Examination Category for Pressure Retaining Welds in Pump Casing
B–L–2	ASME Section XI, Table IWB–2500–1, Examination Category for Pump Casing
B–M–1	ASME Section XI, Table IWB–2500–1, Examination Category for Pressure Retaining Welds in Valve Bodies
B–M–2	ASME Section XI, Table IWB–2500–1, Examination Category for Valve Bodies
B–N–1	ASME Section XI, Table IWB–2500–1, Examination Category for Interior of Reactor Vessel
B–N–2	ASME Section XI, Table IWB–2500–1, Examination Category for Integrally Welded Core Support Structures and Interior Attachments to Reactor Vessels
B–N–3	ASME Section XI, Table IWB–2500–1, Examination Category for Removable Core Support Structures
B–O	ASME Section XI, Table IWB–2500–1, Examination Category for Pressure Retaining Welds in Control Rod Housing

B–P	ASME Section XI, Table IWB–2500–1, Examination Category for All Pressure Retaining Components
B&W	Babcock & Wilcox Nuclear Service Company
BWR	Boiling Water Reactor
CASS	Cast Austenitic Stainless Steel
CAV	Crack Arrest Verification (System)
CE	ABB Combustion Engineering
CEA	Control Element Assembly
CFR	Code of Federal Regulations
CFS	Core Flood System
Cl	Class
CLB	Current Licensing Basis
CRA	Control Rod Assembly
CRDM	Control Rod Drive Mechanism
CS	Carbon Steel
CRDM	Control Rod Drive Mechanism
DBA	Design Basis Accident
DHRS	Decay Heat Removal System
DOE	United States Department of Energy
E–A	ASME Section XI, Table IWE–2500–1, Examination Category for Containment Surfaces
E–B	ASME Section XI, Table IWE–2500–1, Examination Category for Containment Surfaces
E–C	ASME Section XI, Table IWE–2500–1, Examination Category for Pressure Retaining Welds
E–D	ASME Section XI, Table IWE–2500–1, Examination Category for Seals, Gaskets, and Moisture Barriers
E–F	ASME Section XI, Table IWE–2500–1, Examination Category for Pressure Retaining Dissimilar Metal Welds
E–G	ASME Section XI, Table IWE–2500–1, Examination Category for Pressure Retaining Bolting
E–P	ASME Section XI, Table IWE–2500–1, Examination Category for all Pressure Retaining Components
E/C	Erosion/Corrosion
ECCS	Emergency Core Cooling System
ECP	Electrochemical Potential
EFPY	Effective Full Power Years
EPA	United States Environmental Protection Agency
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
FSAR	Final Safety Analysis Report
FW	Feedwater
GDS	General Design Criteria
Gr	Grade
HAZ	Heat Affected Zone
HPCI	High–Pressure Coolant Injection
HPCS	High–Pressure Core Spray
HSW	Heat Sink Welding
HVAC	Heating, Ventilation, and Air Conditioning

HWC	Hydrogen Water Chemistry
IASCC	Irradiation Assisted Stress Corrosion Cracking
IGSCC	Intergranular Stress Corrosion Cracking
IHSI	Induction Heating Stress Improvement
INPO	Institute of Nuclear Power Operations
IPA	Integrated Plant Assessment
IR	Industry Report
IRM	Intermediate Range Monitor
ISI	Inservice Inspection
IWB	Subsection of ASME Code, Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components," containing "Requirements for Class 1 Components of Light–Water Cooled Power Plants"
IWB–2500	Table of ASME Code, Section XI, Subsection IWB, dealing with "Examination and Pressure Test Requirements"
IWE	Subsection of ASME Code, Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components," containing "Requirements for Class MC and Metallic Liners of Class CC Components of Light–Water Cooled Power Plants"
IWL	Subsection of ASME Code, Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components," containing "Requirements for Class CC Concrete Components of Light–Water Cooled Power Plants"
KSB	Klein, Schanzlin, and Becker
L–A	ASME Section XI, Table IWL–2500–1, Examination Category for Concrete
LER	Licensee Event Report
LOCA	Loss of Coolant Accident
LPCI	Low–Pressure Coolant Injection
LPCS	Low–Pressure Core Spray
LPRM	Low–Power Range Monitor
LTOP	Low–Temperature Overpressure Protection
LWR	Light Water Reactor
MS	Main Steam
MSIP	Mechanical Stress Improvement Process
MSIV	Main Steam Isolation Valve
NB	Subsection of ASME Code, Section III "Rules for Construction of Nuclear Power Plant Components," dealing with Class 1 Components
NEI	Nuclear Energy Institute
NG	Subsection of ASME Code, Section III "Rules for Construction of Nuclear Power Plant Components," dealing with Core Support Structure
NPAR	Nuclear Plant Aging Research
NRC	Nuclear Regulatory Commission
NRCA	National Roofing Contractors Association
NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Management and Resources Council
PCM	Pipe Crack Monitor (System)
PCPB	Primary Coolant Pressure Boundary
PORV	Power–Operated Relief Valve
ppb	Parts per billion
ppm	Parts per million
PT	Pressure Temperature (Limits)
PTS	Pressurized Thermal Shock

PRA	Probabilistic Risk Analysis
PWR	Pressurized Water Reactor
RBCCW	Reactor Building Closed Cooling Water (System)
RCC	Rod Control Cluster
RCCA	Rod Control Cluster Assembly
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RPV	Reactor Pressure Vessel
SAW	Submerged ARC Welding
SCC	Stress Corrosion Cracking
SCM	Stress Corrosion Monitoring (System)
SRM	Source Range Monitor
SRP-LR	Standard Review Plan for License Renewal
SRV	Safety/Relief Valve
SS	Stainless Steel
TGSCC	Transgranular Stress Corrosion Cracking
USAS	United States of America Standard
UT	Ultrasonic Testing
VT-1	Visual examination specified in ASME Section XI inservice inspection conducted to determine the condition of the part, component, or surface examined, including conditions such as cracks, wear, corrosion, erosion, or physical damage
VT-2	Visual examination specified in ASME Section XI inservice inspection conducted to locate evidence of leakage from pressure-retaining components, as required by system pressure or functional tests
VT-3	Visual examination specified in ASME Section XI inservice inspection conducted to determine the mechanical and structural integrity, such as the examination of conditions that could affect functionality of components and their supports

1 Introduction

In about 1990, the Nuclear Management and Resources Council (NUMARC), now the Nuclear Energy Institute (NEI), submitted for NRC review, ten industry reports (IRs) addressing aging issues associated with specific structures and components of nuclear power plants,[1-10] and one IR addressing the screening methodology for integrated plant assessment (IPA).[11] The ten IRs for specific structures and components are:

1. Pressurized Water Reactor (PWR) Reactor Vessel
2. Boiling Water Reactor (BWR) Reactor Vessel
3. PWR Containment
4. BWR Containment
5. PWR Reactor Coolant Pressure Boundary
6. BWR Reactor Coolant Pressure Boundary
7. PWR Reactor Vessel Internals
8. BWR Reactor Vessel Internals
9. Class I Structures
10. Low-Voltage, In-Containment, Environmentally Qualified Cable.

The original intent of the IRs for specific structures and components was to serve as a referenceable surrogate for carrying out the IPA requirements of the license renewal rule. The NRC staff had been reviewing the IRs. Public documents associated with the NRC staff's review of each of the ten IRs are listed in chronological order in Appendix A.

However, in 1992, the NRC staff and industry resources were redirected to address implementation issues of the license renewal rule.[12,13] In SECY-93-049,[12] the NRC staff recommended that appropriate technical information and agreements from the NUMARC IRs be incorporated into the draft standard review plan for license renewal (SRP-LR). In a staff requirements memorandum dated June 28, 1993, the Commission approved the NRC staff's recommendations.* NUMARC agreed with the NRC staff's approach in a letter dated March 3, 1994.**

This report provides a brief summary of the technical information and NUMARC/NRC agreements from the ten IRs, except for the cable IR.[10] The cable IR addresses the issue of environmental qualification (EQ) of electric equipment, which was superseded by the EQ action plan.†,†† The technical information and agreements documented herein represent the status of the NRC staff's review when the NRC staff and industry resources were redirected to address rule implementation issues. The technical information and NUMARC/NRC agreements have been compiled for each of the ten IRs, except for the cable IR. A draft version of this report (NUREG/????) was presented during a public meeting held on December 12, 1994, and placed in the NRC Public Document Room on December 22, 1994.# The NEI submitted comments

* Staff Requirements Memorandum from Samuel J. Chilk, dated June 28, 1993.

** Letter from William H. Rasin of the Nuclear Management and Resources Council to William T. Russel of NRC, dated March 3, 1993.

† Letter from William T. Russel of NRC to William H. Rasin of the Nuclear Management and Resources Council, dated July 30, 1993.

†† "Environmental Qualification of Electric Equipment," Memorandum for the Commissioners from James M. Taylor, dated April 8, 1994.

December 12, 1994, Meeting Summary "License Renewal Industry Report," dated December 22, 1994.

dated December 20, 1995,## comparing the draft summary report with a similar NEI effort performed by EPRI. These comments were discussed and resolved during a public meeting held on May 14, 1996,* and appropriate revisions were incorporated into the draft summary report. The NRC staff plans on incorporating appropriate technical information and agreements into the draft SRP-LR.**

2 Technical Information and Agreements

The NRC staff had been reviewing the ten NUMARC IRs; their comments on each IR and NUMARC responses to the comments have been compiled as public documents listed in Appendix A. The technical information and agreements for each IR, except for the cable IR, have been compiled into tables, and are presented in Appendix B. The following approach was used in identifying technical information and agreements for this report:

1. There is an agreement if there is no NRC comment on the aging issue addressed in the IR. The technical basis for the agreement was obtained from a review of the IRs and their associated public documents and is described in the tables.

2. During the IR review, the staff reached many "agreements in principle" (AIPs) on issues. An AIP was for NUMARC to revise the IRs in an agreed upon way to address the NRC comment. If the NUMARC response appropriately addressed the NRC comment as discussed in the AIP, there was an agreement. The technical basis for the agreement was obtained from a review of the IRs and their associated public documents and is described in the tables.

3. During the IR review, many issues were identified as "open." These open issues were not reviewed for this report and remain identified as open. NUMARC's and NRC's proposals and bases are briefly described in the tables.

4. Chapter 6 in the IRs discusses NUMARC's recommendations on ongoing management options. Chapter 6 was not the focus of the NRC review during the IR review process and issues/comments relating to Chapter 6 information are not discussed in this report.

Each table consists of seven columns. The first two columns list the aging-related degradation mechanisms (ARDMs) addressed in the IRs and their effects on structures and components; the effcts of ARDMs were based primarily on information in the IRs. A complete list of the ARDMs and their effects is given in Appendix C.

The specific structures and/or components, and their construction materials based on the IRs are listed in the third and fourth columns, respectively. For each IR, a complete list of components is included at the end of the table. In general, the IRs present only representative examples and do not provide a comprehensive list of the type, grade, and specification of

Letter from Douglas J. Walters of the Nuclear Energy Institute to Scott F. Newberry of the NRC, dated December 20, 1995.

* "Summary of Meeting with NEI on the Industry and Nuclear Regulatory Commission (NRC) Efforts to Summarize Technical Information and Agreements Reached During Prior Review of License Renewal Industry Reports," dated May 24, 1996.

** "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," (Previous draft report NUREG-1299).

materials used for various reactor components. For most IRs, only material categories such as stainless steel (SS), cast austenitic stainless steel (CASS), Ni alloy, or carbon steel (CS), are listed in column four of the tables. A detailed list of material type and grade is provided in the PWR and BWR reactor pressure vessel IRs.

The NRC staff comments on each IR are referenced in the fifth column. The designation for the comment numbers used in documents in Appendix A is as follows: G– general comment, S– specific comment, and (S1), (S2), or (S3) represent supplemental list of comments, Set 1, 2, or 3. Comments have been resolved unless identified as open issues.

The sixth column presents the NUMARC/NRC agreements or proposals on whether a ARDM or ARDM/component combination is potentially significant, and if it is potentially significant, a brief description of the program that can adequately manage the effects of aging is presented in the same column. The information in column six represents NUMARC/NRC agreements, unless identified as proposals. The technical basis for these agreements or proposals, including assumptions and references, are described in the seventh (and final) column. A few examples of the information included in these two columns are given below.

1. For a specific ARDM or ARDM/component combination, if the effects of aging are not potentially significant, then "non–significant" is listed in the agreements column. The technical basis, assumptions, and references for the agreement are presented in column seven. For example, the effect of creep is non–significant for BWR primary coolant piping and fittings fabricated from CS or SS because the reactor operating temperatures are significantly lower than the temperatures at which creep is a concern for CS and for SS components. Also, if the effects of aging are not potentially significant when certain bounding conditions are met, then "for components that meet the basis requirements, this ARDM is *non–significant*" is listed in the agreements column. For example, the effects of freeze–thaw is non–significant for Class 1 concrete structures that meet the following criteria: located in geographic regions of negligible weathering conditions (weathering index <100 day–inch/yr); and if located in severe weathering conditions (weathering index 100–500 day–inch/yr) the concrete mix design meets the air content and water–to–cement ratio requirements of ACI 318–63 or ACI–349–85. This information is primarily included in Chapter 4 of the IRs.

2. If a specific ARDM/component combination is potentially significant and the effects of aging are adequately addressed by current management programs, then a brief description of the program is provided in column six. For example, the program delineated in NUREG–0313 and implemented through NRC generic letter 88–01, is a current and adequate program to manage the effects of intergranular stress corrosion cracking (IGSCC) of SS piping and fittings of BWR primary coolant pressure boundary. This information is primarily included in Chapter 5 of the IRs.

3. If a specific ARDM/component combination is potentially significant and the current programs are not adequate for managing the effects of aging, then column six simply states "current practices to be enhanced, select plant–specific aging management." For these cases, the NUMARC recommended aging management options are described in Chapter 6 of the IRs. However, Chapter 6 was not the focus of NRC review of the IRs.

4. An ARDM or ARDM/component combination is listed as "unresolved issue" if no agreement was reached between NUMARC and the NRC staff. For these cases, both the NUMARC and the NRC proposals are briefly described in the agreements column. An example of an unresolved issue is the effects of thermal aging embrittlement of PWR primary coolant system components fabricated from cast austenitic stainless steel (CASS). The NUMARC proposal considers a ferrite content screening criterion and ASME Code Section XI, Subsection IWB inspection to be an adequate program for managing the effects of thermal embrittlement. The NRC proposal, however, considers that the ferrite content criterion is inadequate for screening and VT-3 visual examination is not intended or reliable for detecting tight cracks.

3 Conclusions

The ten IRs, except for the cable IR, submitted by NUMARC addressing aging issues associated with specific structures and components of nuclear power plants have been reviewed. The technical information and NUMARC/NRC agreements for each IR have been compiled into tables. The information presented in each of the tables includes specific structures and components and their materials of construction; ARDMs and their effects on structures and components; relevant comments of the NRC staff; and the NUMARC/NRC agreements or proposals and their technical basis, including assumptions and references. The technical information and agreements documented herein represent the status of the NRC staff's review at the time when the NRC staff and industry resources were redirected to address rule implementation issues. The NRC staff plans on incorporating appropriate technical information and agreements into the draft SRP-LR.

References

1. "Pressure Water Reactor Reactor Vessel License Renewal Industry Report," NUMARC Report Number 90-04, Nuclear Management and Resource Council, May 1990; Revision 1, Sept. 1992.

2. "Boiling Water Reactor Vessel License Renewal Industry Report," NUMARC Report Number 90-02, Nuclear Management and Resource Council, Oct. 1989; Revision 1, Sept. 1992.

3. "Pressurized Water Reactor Containment Structures License Renewal Industry Report," NUMARC Report Number 90-01, Nuclear Management and Resource Council, Aug. 1989; Revision 1, Sept. 1991.

4. "Boiling Water Reactor Containments License Renewal Industry Report," NUMARC Report Number 90-10, Nuclear Management and Resource Council, July 1990; Revision 1, Dec. 1991.

5. "PWR Reactor Coolant System License Renewal Industry Report," NUMARC Report Number 90-07, Nuclear Management and Resource Council, Oct. 1990; Revision 1, May 1992.

6. "BWR Primary Coolant Pressure Boundary License Renewal Industry Report," NUMARC Report Number 90–09, Nuclear Management and Resource Council, Sept. 1990; Revision 1, April 1992.

7. "Pressurized Water Reactor Vessel Internals License Renewal Industry Report," NUMARC Report Number 90–05, Nuclear Management and Resource Council, Sept. 1990; Revision 1, Dec. 1992.

8. "Boiling Water Reactor Vessel Internals License Renewal Industry Report," NUMARC Report Number 90–03, Nuclear Management and Resource Council, Feb. 1990; Revision 1, June 1992.

9. "Class I Structures License Renewal Industry Report," NUMARC Report Number 90–06, Nuclear Management and Resource Council, June 1990; Revision 1, Dec. 1991.

10. "Low–Voltage, In–Containment, Environmentally–Qualified Cable License Renewal Industry Report," NUMARC Report Number 90–08, Nuclear Management and Resource Council, July 1990; Revision 1, March 1993.

11. "Methodology to Evaluate Plant Equipment for License Renewal," Nuclear Management and Resource Council, Oct. 6, 1989.

12. Implementation of 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," SECY–93–049, March 1, 1993.

13. Additional Implementation Information for 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," SECY–93–113, April 30, 1993.

Appendix A: Public Documents

A1 Pressure Water Reactor Vessel License Renewal Industry Report NUMARC Report Number 90–04

The public documents associated with NRC staff's review of the IR on "Pressurized Water Reactor Vessel" are listed below in chronological order.

05/25/90 Letter from E. Griffing (NUMARC) to W. Travers (NRC): Transmittal Letter for NUMARC Report Number 90–04, "Pressure Water Reactor Vessel License Renewal Industry Report."

05/90 NUMARC Report Number 90–04, "Pressure Water Reactor Vessel License Renewal Industry Report" (attachment to 05/25/90 letter).

08/13/90 Letter from E. Griffing (NUMARC) to J. Craig (NRC); NUMARC – Summary of (07/24/90) meeting with License Renewal Project Directorate to discuss the industry report procedural process.

09/14/90 Letter from J. Craig (NRC) to E. Griffing (NUMARC); Transmittal of staff comments and request for additional information related to "Pressure Water Reactor Vessel License Renewal Industry Report."

11/01/90 Letter from J. Craig (NRC) to E. Griffing (NUMARC); Transmittal of supplemental NRC staff comments related to "PWR Vessel License Renewal Industry Report."

12/26/90 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal of NUMARC responses to 09/14/90 NRC staff comments and request for additional information, and requested technical references related to "PWR Vessel License Renewal Industry Report."

01/03/91 Letter from J. Craig (NRC) to E. Griffing (NUMARC); Transmittal of additional supplemental staff comments related to "Pressure Water Reactor Vessel License Renewal Industry Report."

02/14/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal letter of NUMARC responses to two sets of staff supplemental comments related to "PWR Vessel License Renewal Industry Report."

02/15/91 Memo from D. Jackson to J. Craig; Summary of 02/14/91 teleconference –status of NRC evaluation of 12/16/90 & 02/14/91 NUMARC responses to NRC staff comments related to "PWR Vessel Industry Report."

02/28/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); NUMARC commitment to use standardized language in the "PWR Vessel IR."

03/15/91 Memo from S. Lee to J. Craig; Summary of 02/22/91 meeting with NUMARC on resolution of NRC staff comments on PWR Vessel License Renewal Industry Report.

04/04/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal Letter of referenced Westinghouse Yankee Pressure Vessel Cladding Paper (WCAP–2588, November 1965).

05/09/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal of NUMARC responses to 11/01/90 NRC staff supplemental comments related to "PWR Vessel License Renewal Industry Report."

05/29/91 Letter from J. Craig (NRC) to E. Griffing (NUMARC): Incorporation of Appendices VII and VIII of section XI of ASME code in NUMARC license renewal industry reports.

07/11/91 Letter from J. Craig (NRC) to E. Griffing (NUMARC); Status of (05/09/91) NUMARC responses to 11/01/90 NRC staff supplemental comments related to "PWR Vessel License Renewal Industry Report."

09/16/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC): Industry Implementation of ASME Code for effective management of aging. Response to staff correspondence of 05/29/91.

10/20/92 Letter from R. Ng (NUMARC) to F. Akstulewicz (NRC): Transmittal letter for September 1992, "Pressure Water Reactor Vessel License Renewal Industry Report," NUMARC Report Number 90-04.

09/92 NUMARC Report Number 90-04, "Pressure Water Reactor Reactor Vessel License Renewal Industry Report," Revised September 1992 (attachment to 10/20/92 letter).

09/10/92 NUMARC revised responses to NRC comments and supplemental comments and documented AIPs on PWR Reactor Vessel IR (attachment to 10/20/92 letter).

A2 Boiling Water Reactor Vessel License Renewal Industry Report NUMARC Report Number 90-02

The public documents associated with NRC staff's review of the IR on "Pressurized Water Reactor Vessel" are listed below in chronological order.

10/16/89 Letter from W. Rasin (NUMARC) to W. Houston (NRC); Transmittal Letter for "Boiling Water Reactor Vessel License Renewal Industry Report."

10/89 NUMARC "Boiling Water Reactor Vessel Industry Report" (attachment to 10/16/89 letter).

04/02/90 Letter from W. Travers (NRC) to E. Griffing (NUMARC); Transmittal of staff comments and request for additional information related to "Boiling Water Reactor Vessel Industry Report."

05/29/90 Letter from E. Griffing (NUMARC) to W. Travers (NRC); Transmittal of DRAFT NUMARC responses to 04/02/90 NRC staff comments related to "BWR Reactor Vessel Industry Report."

06/11/90 Memo from R.Parkhill to J. Craig; Summary of 06/05/90 meeting with NUMARC regarding NUMARC's response to NRC comments on "Boiling Water Reactor License Renewal Report" dated October 1989.

09/20/90 Memo from F. Akstulewicz to J. Craig; Summary of 8/31/90 meeting between LRPD and NUMARC which discussed industry reports.

10/26/90 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal of NUMARC responses to 04/02/90 NRC comments and request for additional information related to "Boiling Water Reactor Vessel Industry Report."

11/30/90 Letter from J. Craig (NRC) to E. Griffing (NUMARC); Transmittal of supplemental staff comments related to "Boiling Water Reactor Vessel Industry Report."

01/02/91	Memo from R. Parkhill to J. Craig; Summary of (12/21/90) teleconference – status of NRC evaluation of 10/26/90 NUMARC responses to 04/02/90 staff comments related to "BWR Reactor Vessel IR."
02/25/91	Letter from J. Craig (NRC) to E. Griffing (NUMARC); Status of the resolution of NRC comments on industry reports for the BWR Reactor Vessel (90–02) and BWR Reactor Vessel Internals (90–03).
03/15/91	Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal Letter for 3/12/91 NUMARC responses to supplemental staff comments related to "Boiling Water Reactor Vessel Industry Report."
03/12/91	NUMARC response to 11/30/90 supplemental NRC staff comments on "Boiling Water Reactor Vessel Industry Report" (attachment to 03/12/91 letter).
03/29/91	Letter from E. Griffing (NUMARC) to J. Craig (NRC); NUMARC response to staff proposal of the use of ASME Section XI, Appendices VII and VIII code (personnel qualification).
04/23/91	Memo from R. Parkhill to J. Craig; Summary of 01/22–24/91 meeting with NUMARC on resolution of NRC staff comments on license renewal industry reports for the BWR Reactor Vessel (90–02) and BWR Reactor Vessel Internals (90–03).
05/29/91	Letter from J. Craig (NRC) to E. Griffing (NUMARC); Incorporation of Appendices VII and VIII of Section XI of ASME code in NUMARC License Renewal Industry Reports.
06/10/91	Memo from R. Parkhill to J. Craig; Summary of 05/28/91 teleconference – status of NRC evaluation of 03/15/91 NUMARC responses to 11/30/90 supplemental NRC staff comments on "Boiling Water Reactor Vessel Industry Report."
07/01/91	Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal Letter for referenced GE RICSIL–055 Paper.
07/03/91	Memo from R. Parkhill to J. Craig; Summary of 06/17/91 meeting with NUMARC on resolution supplemental comments on license renewal industry reports form the BWR Reactor Vessel (90–02) and BWR Reactor Vessel Internals (90–03).
09/16/91	Letter from E. Griffing (NUMARC) to J. Craig (NRC); Industry implementation of ASME code for effective management of aging. Response to staff correspondence of 05/29/91.
10/20/92	Letter from R. Ng (NUMARC) to F. Akstulewicz (NRC); Transmittal Letter for September 1992, "Boiling Water Reactor; Reactor Pressure Vessel License Renewal Industry Report," NUMARC Report Number.
09/92	NUMARC Report Number 90–02, "Boiling Water Reactor Reactor Vessel License Renewal Industry Report," revised September 1992 (attachment to 10/20/92 letter).
09/16/92	NUMARC revised responses to NRC comments and supplemental comments and documented AIPs on BWR Vessel IR (attachment to 10/20/92 letter).

A3 Pressurized Water Reactor Containment Structures License Renewal Industry Report NUMARC Report Number 90–01

The public documents associated with NRC staff's review of the IR on "Pressurized Water Reactor Containment Structures" are listed below in chronological order.

08/30/89 Letter from W. Rasin (NUMARC) to R. Houston (NRC); Transmittal Letter for "Pressure Water Reactor Containment Structures; License Renewal Industry Report."

08/89 NUMARC "Pressure Water Reactor Containment Structures; License Renewal Industry Report" (attachment to 08/30/89 letter).

06/04/90 Letter from J. Craig (NRC) to E. Griffing (NUMARC); Transmittal of staff comments and request for additional information related to "Pressure Water Reactor Containment Structures; License Renewal Industry Reports."

10/04/90 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal of NUMARC responses to 06/04/90 NRC comments and request for additional information related to "Pressure Water Reactor Containment Structures; License Renewal Industry Reports."

11/09/90 Memo from D. Tang to J. Craig; Summary of 10/26/90 teleconference with NUMARC – PWR Containment Industry Report comments/responses resolution status.

11/15/90 Memo from D. Tang to J. Craig; Summary of 10/30/90 meeting with NUMARC – comments/responses resolution for PWR Containment Structure Industry Report.

12/05/90 Memo from T. Tang to J. Craig; Summary of 11/08/90 teleconference with NUMARC – comments/responses resolution for PWR Containment Structure Industry Report.

12/19/90 Letter from J. Craig (NRC) to E. Griffing (NUMARC); Summary of staff's conclusions concerning comments/responses for "PWR Containment Structures Industry Report."

09/17/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal Letter for September 1991, "Pressurized Water Reactor Containment Structures, License Renewal Industry Report," NUMARC Report 90–01.

09/91 NUMARC Report 90–01, "Pressurized Water Reactor Containment Structures, License Renewal Industry Report," Revision 1, September 1991 (attachment to 09/17/91 letter).

08/28/91 NUMARC revised responses to NRC comments and documented AIPs on PWR Containment IR (attachment to 09/17/91 letter).

10/30/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal letter for revised NUMARC responses to NRC comments related to "PWR Containment Structures IR."

10/29/91 NUMARC revised responses to NRC comments and documented AIPs on "PWR Containment IR."

11/27/91 Letter from K.Cozens (NUMARC) to J. Craig (NRC); Transmittal of missing pages to September 1991, "Pressurized Water Reactor Containment Structures, License Renewal Industry Report," Revision 1.

04/13/92 Memo from D. Tang (NRC); Summary of 02/26/92 meeting with NUMARC –Discussion of Revision 1, PWR Containment Structures License Renewal Industry Report.

04/15/92 Memo from D. Tang (NRC); Summary of 03/12/92 meeting with NUMARC –Discussion of marked–up comments on Revision 1, PWR Containment Structures License Renewal Industry Report.

A4 BWR Containments License Renewal Industry Report NUMARC Report Number 90–10

The public documents associated with NRC staff's review of the IR on "BWR Containments" are listed below in chronological order.

07/25/90	Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal Letter of NUMARC Report Number 90–10, "BWR Containments License Renewal Industry Report."
07/90	NUMARC Report Number 90–10, "BWR Containments License Renewal Industry Report" (attachment to 07/25/90 letter).
10/24/90	Letter from J. Craig (NRC) to E. Griffing (NUMARC); Transmittal of staff comments and request for additional information related to "Boiling Water Reactor Containment License Renewal Industry Report."
01/27/91	Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal of NUMARC responses to 10/24/90 NRC comments and request for additional information related to "BWR Containment License Renewal Industry Report."
04/11/91	Memo from D. Jackson to J. Craig; Summary of 02/20/91 meeting with NUMARC – resolution of staff comments on industry report for the BWR Containments (90–10).
12/27/91	Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal Letter for December 1991, "BWR Containments License Renewal Industry Report."
12/91	NUMARC Report Number 90–10, "BWR Containments License Renewal Industry Report," Revision 1, December 1991 (attachment to 12/27/91 letter).
12/20/91	NUMARC revised responses to NRC comments and documented AIPs on BWR Containment IR (attachment to 12/27/91 letter).

A5 PWR Reactor Coolant System License Renewal Industry Report NUMARC Report Number 90–07

The public documents associated with NRC staff's review of the IR on "PWR Reactor Coolant System" are listed below in chronological order.

10/10/90	Letter from E. Griffing (NUMARC) to NRC; Transmittal letter for NUMARC Report Number 90–07, "PWR Reactor Coolant System License Renewal Industry Report."
10/90	NUMARC Report Number 90–07, "PWR Reactor Coolant System License Renewal Industry Report" (attachment to 10/10/90 letter).
02/06/91	Letter from J. Craig (NRC) to E. Griffing (NUMARC); Transmittal of NRC staff comments and request for additional information related to "PWR Reactor Coolant System License Renewal Industry Report."
05/10/91	Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal of NUMARC responses to 02/06/91 NRC comments and request for additional information related to "PWR Reactor Coolant System License Renewal Industry Report."
07/12/91	Memo from T. J. Kim to J. Craig; Summary of 06/12/91 meeting with NUMARC on resolution of NRC staff comments on the PWR Reactor Coolant System License Renewal Industry Report (90–07).

09/13/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); NUMARC response to staff's proposal for the use of NUREG/CR-4513 – fracture mechanics analysis for the screening criteria of thermal embrittlement of CASS.

09/16/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Industry Implementation of ASME Code for effective management of aging. Response to staff correspondence of 05/29/91.

10/04/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Submittal of EPRI Report NP-3614, Volumes 1 and 2, as requested by the staff during the 06/12/91 meeting.

06/11/92 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal letter for May 1992, "PWR Reactor Coolant System License Renewal Industry Report," NUMARC Report Number 90–07,

05/92 NUMARC Report Number 90–07, "PWR Reactor Coolant System License Renewal Industry Report," revised May 1992 (attachment to 06/11/92 letter).

05/28/92 NUMARC revised responses to NRC comments and documented AIPs on PWR Reactor Coolant System IR (attachment to 06/11/92 letter).

A6 BWR Primary Coolant Pressure Boundary License Renewal Industry Report NUMARC Report Number 90–09

The public documents associated with NRC staff's review of the IR on "BWR Primary Coolant Pressure Boundary" are listed below in chronological order.

09/18/90 Letter from E. Griffing (NUMARC) to NRC; Transmittal letter of NUMARC Report Number 90–09, "BWR Primary Coolant Pressure Boundary License Renewal Industry Report."

09/90 NUMARC Report Number 90–09, "BWR Primary Coolant Pressure Boundary License Renewal Industry Report" (attachment to 09/18/90 letter).

01/18/91 Letter from J. Craig (NRC) to E. Griffing (NUMARC); Transmittal of staff comments and request for additional information related to "BWR Primary Coolant Pressure Boundary License Renewal Industry Report."

04/26/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal of NUMARC responses to 01/18/91 NRC comments and request for additional information related to "BWR Primary Coolant Pressure Boundary License Renewal Industry Report."

05/29/91 Letter from J. Craig (NRC) to E. Griffing (NUMARC); Incorporation of Appendices VII and VIII of Section XI of ASME code in NUMARC license renewal industry reports.

06/18/91 Memo from T. J. Kim (NRC) to J. Craig (NRC); Summary of 05/30/91 meeting with NUMARC on resolution of NRC staff comments on the BWR Primary Coolant Pressure Boundary License Renewal Industry Report (90–09).

09/06/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal of AIP references requested by the NRC during May 30th meeting concerning PWR RCS and BWR PCPB Industry Reports (90–07 & 90–09).

09/09/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); NUMARC schedule for submittal of revised license renewal industry reports.

09/13/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); NUMARC response to staff's proposal for the use of NUREG/CR-4513 – fracture mechanics analysis for the screening criteria of thermal embrittlement of CASS.

09/16/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Industry Implementation of ASME Code for effective management of aging. Response to staff correspondence of 05/29/91.

06/11/92 Transmittal Letter from E. Griffing (NUMARC) to J. Craig (NRC); transmittal letter for April 1992, "BWR Primary Coolant Pressure Boundary License Renewal Industry Report," NUMARC Report Number 90-01.

04/92 NUMARC Report Number 90-01, "BWR Primary Coolant Pressure Boundary License Renewal Industry Report," revised April 1992 (attachment to 06/11/92 letter).

05/15/92 NUMARC revised responses to NRC comments and documented AIPs on "BWR Primary Coolant Pressure Boundary IR (attachment to 06/11/92 letter).

A7 Pressurized Water Reactor Vessel Internals License Renewal Industry Report NUMARC Report Number 90–05

The public documents associated with NRC staff's review of the IR on "Pressurized Water Reactor Vessel Internals" are listed below in chronological order.

09/18/90 Letter from E. Griffing (NUMARC) to NRC; Transmittal letter for NUMARC Report Number 90-05, "Pressurized Water Reactor Vessel Internals License Renewal Industry Report."

09/90 NUMARC Report Number 90-05, "Pressurized Water Reactor Vessel Internals License Renewal Industry Report" (attachment to 09/18/90 letter).

01/31/91 Letter from J. Craig (NRC) to E. Griffing (NUMARC); Transmittal of staff comments and request for additional information related to "Pressurized Water Reactor Vessel Internals License Renewal Industry Report."

02/03/92 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal of NUMARC response to (01/29[sic.31]/92) NRC's comments on "PWR Vessel Internals License Renewal Industry Report."

08/04/93 Letter from E. Griffing (NUMARC) to J. Craig (NRC): Transmittal Letter of December 1992, "Pressurized Water Reactor Vessel Internals License Renewal Industry Report," NUMARC Report Number 90-05.

12/92 NUMARC Report Number 90-05, "Pressurized Water Reactor Vessel Internals License Renewal Industry Report," Revision 1, December 1992 (attachment to 08/04/93 letter).

12/23/92 NUMARC revised responses to NRC comments and documented AIPs on PWR Internals IR (attachment to 08/04/93 letter).

7/10/96 Letter from Douglas J. Walters (NEI) to Scott F. Newberry (NRC); regarding the list of components within the scope of the PWR Reactor Pressure Vessel Internals IR that are susceptible to IASCC.

A8 Boiling Water Reactor Vessel Internals License Renewal Industry Report NUMARC Report Number 90–03

The public documents associated with NRC staff's review of the IR on "Boiling Water Reactor Vessel Internals" are listed below in chronological order.

02/23/90 Letter from E. Griffing (NUMARC) to W. Travers (NRC); Transmittal letter for "Boiling Water Reactor Vessel Internals License Renewal Industry Report."

02/90 NUMARC "Boiling Water Reactor Vessel Internals License Renewal Industry Report" (attachment to 02/23/90 letter).

07/06/90 Letter from J. Craig (NRC) to E. Griffing (NUMARC); Transmittal of staff comments and request for additional information related to "Boiling Water Reactor Vessel Internals License Renewal Industry Report."

11/20/90 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal of NUMARC responses to 07/06/90 NRC comments and request for additional information related to "Boiling Water Reactor Vessel Internals License Renewal Industry Report."

11/30/90 Letter from J. Craig (NRC) to E. Griffing (NUMARC); Transmittal of supplemental staff comments and request for additional information related to "Boiling Water Reactor Vessel Internals License Renewal Industry Report."

01/11/91 Memo from R. Parkhill to J. Craig; Summary of 01/10/91 teleconference – status of NRC evaluation of 11/20/90 NUMARC responses to staff comments related to "Boiling Water Reactor Vessel Internals License Renewal Industry Report."

02/25/91 Letter from J. Craig (NRC) to E. Griffing (NUMARC); Summary of 01/22/91 meeting – Status of the resolution of NRC comments on industry reports for the BWR Reactor Vessel (90–02) and BWR Reactor Vessel Internals (90–03).

03/15/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal of NUMARC responses to 11/30/90 supplemental NRC staff comments and request for additional information related to BWR Reactor Vessel (90–02) and BWR Reactor Vessel Internals (90–03) Industry Reports.

03/12/91 NUMARC responses to NRC comments and supplemental comments related to "BWR Reactor Vessel Internals IR" (attachment to 03/15/91 letter).

03/29/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); NUMARC response to staff proposal of the use of ASME Section XI, Appendices VII and VIII code (personnel qualification).

04/23/91 Memo from R. Parkhill to J. Craig; Summary of 01/22/91 meeting with NUMARC on resolution of staff comments on license renewal industry reports for the BWR Reactor Vessel (90–02) and BWR Reactor Vessel Internals (90–03).

06/10/91 Memo from R. Parkhill to J. Craig; Summary of 05/28/91 teleconference – status of NRC evaluation of 03/15/91 NUMARC response to 11/30/90 supplemental NRC staff comments.

07/01/91 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal letter for referenced GE Report RICSIL 055.

07/03/91	Memo from R. Parkhill to J. Craig; Summary of 06/17/91 meeting with NUMARC on resolution of supplemental comments on license renewal industry reports for the BWR Reactor Vessel (90-02) and BWR Reactor Vessel Internals (90-03).
06/29/92	Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal letter for June 1, 1992, "BWR Reactor Vessel Internals Industry Report."
06/92	NUMARC "Boiling Water Reactor Vessel Internals Industry Report," revised June 1, 1992 (attachment to 06/29/92 letter).
06/11/91	NUMARC revised responses to NRC comments and supplemental comments and documented AIPs on BWR Vessel Internals IR (attachment to 06/29/92 letter).

A9 Class I Structures License Renewal Industry Report NUMARC Report Number 90-06

The public documents associated with NRC staff's review of the IR on "Class 1 Structures" are listed below in chronological order.

06/11/90	Letter from E. Griffing (NUMARC) to W. Travers (NRC); Transmittal Letter for NUMARC Report Number 90-06, "Class I Structures License Renewal Industry Report."
06/90	NUMARC Report Number 90-06, "Class I Structures License Renewal Industry Report" (attachment to 06/11/90 letter).
10/17/90	Letter from J. Craig (NRC) to E. Griffing (NUMARC); Transmittal of staff comments and request for additional information related to "Class I Structures License Renewal Industry Report."
01/21/91	Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal of NUMARC responses to 10/17/90 NRC staff comments and request for additional information related to "Class I Structures License Renewal Industry Report."
02/15/91	Memo from D. Jackson to J. Craig; Summary of 02/14/91 teleconference -status of NRC evaluation of 01/12/91 NUMARC responses to NRC staff comments related to "Class I Structures License Renewal Industry Report."
03/18/91	Memo from D. Tang to J. Craig; Summary for (02/20/91) meeting with Nuclear Management and Resources Council (NUMARC) - comments/responses resolution for Class I Structures Industry Report.
04/04/91	Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal Letter for referenced NBS Papers on Underground Corrosion of Steel, by W. J. Schwerdtfeger & M. Romanoff, NBS MN-127, March 1972.
05/08/91	Memo from D. Tang to J. Craig; Summary of 04/12/91 teleconference with NUMARC - resolution of remaining comments/responses for Class I Structures Industry Report.
05/09/91	Letter from J. Craig (NRC) to E. Griffing (NUMARC); Resolution status of NRC staff comments on Class I Structures Industry Report.
12/27/91	Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal letter for December 1991, "Class I Structures License Renewal Industry Report," NUMARC Report Number 90-06.

12/91 NUMARC Report Number 90-06, "Class I Structures License Renewal Industry Report," Revision 1, December 1991 (attachment to 12/27/91 letter).

12/20/91 NUMARC revised responses to NRC comments and documented AIPs on Class 1 Structures IR (attachment to 12/27/91 letter).

A10 Low-Voltage In-Containment Environmentally Qualified Cable License Renewal Industry Report NUMARC Report Number 90-08

The public documents associated with NRC staff's review of the IR on "Low-Voltage In-Containment, Environmentally Qualified Cable" are listed below in chronological order.

07/31/90 Letter from E. Griffing (NUMARC) to J. Craig (NRC); Transmittal Letter for NUMARC Report Number 90-08, "Low-voltage In-containment Environmentally Qualified Cable License Renewal Industry Report."

7/90 NUMARC Report Number 90-08, "Low-Voltage In-Containment, Environmentally Qualified Cable License Renewal Industry Report" (attachment to 07/31/90 letter).

11/14/90 Letter from J. Craig (NRC) to E. Griffing (NUMARC); Transmittal of NRC staff comments and request for additional information related to "Low-Voltage In-Containment, Environmentally Qualified Cable License Renewal Industry Report."

03/15/91 Letter from E. Griffing (NUMARC) to J. Craig (NUMARC); Transmittal of NUMARC responses to 11/14/90 NRC comments and request for additional information related to "Low-Voltage In-Containment, Environmentally Qualified Cable License Renewal Industry Report."

05/16/91 Memo from P. Shemanski to J. Craig; Summary of 04/16/91 meeting with NUMARC on resolution of NRC staff comments on Cable In-Containment License Renewal Industry Report (90-08).

11/20/91 Letter from K. Cozens (NUMARC) to J. Craig (NRC); Transmittal of industry response to NRC "G-2 Comments," related to "Low-Voltage In-Containment, Environmentally Qualified Cable License Renewal Industry Report."

11/27/91 Letter from K. Cozens (NUMARC) to J. Craig (NRC); Transmittal of annotated response to "G-2 Comments," submitted in the 11/20/91 letter.

08/03/92 Letter from E. Griffing (NUMARC) to E. Igne (ACRS); Transmittal letter for March 1993, "Low-Voltage In-Containment, Environmentally Qualified Cable License Renewal Industry Report." and documentation delineating Industry Position on NRC "G-2 Comments."

08/04/93 Letter from R. Ng (NUMARC) to W. Travers (NRC); Transmittal letter for March 1993, "Low-Voltage Environmentally-Qualified Cable License Renewal Industry Report," NUMARC Report Number 90-08.

03/93 NUMARC Report Number 90-08, "Low-Voltage Environmentally-Qualified Cable License Renewal Industry Report," revised March 1993 (attachment to 08/04/93 letter).

04/07/93 NUMARC Revised responses to NRC comments and documented AIPs on Low-Voltage Environmentally-Qualified Cable IR (attachment to 08/04/93 letter).

Appendix B: NUMARC/NRC Agreements

Table B1. Brief summary of technical information and NUMARC/NRC agreements from PWR pressure vessel industry report

Aging–Related Degradation Mechanism	Aging Effects	Components[a]	Materials[b]	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron Irradiation Embrittle-ment	Loss of fracture toughness	Closure Head Dome	SA302-Gr B, SA533-Gr B	None	Non-significant	Total fast neutron fluence level is low & is less than 1×10^{17} n/cm^2, above which a surveillance program is required under Appendix H of 10CFR Part 50.
		CRD Mechanism Housing	SB-166, SA182 Type 304 or 316			
		Refueling Seal Ledge	SA212-Gr B, SA516-Gr 70, SA533-Gr B			
		Closure Head Lifting Lugs	SA302-Gr B, SA533-Gr B			
		Shroud Support Ring	SA212-Gr B, SA516-Gr 70			
		Closure Head Flange	SA336, SA508			
		Closure Stud Assembly	SA-540-B23 or B24, SA-320-L43			
		Vessel Flange	SA336, SA508			
		Leakage Monitoring Tubes	SB-166, SB-167, SA-312 Type 316			
		Core Support Pads (Lugs)	SB-166, SB-168			
		Bottom Head Dome	SA302-Gr B, SA533-Gr B			
		Instrumentation Tubes/Penetrations*	SB-166, SB-167			

* Includes the vent pipe on the closure head dome.

Table B1. Brief summary of technical information and NUMARC/NRC agreements from PWR pressure vessel industry report

Aging–Related Degradation Mechanism	Aging Effects	Components[a]	Materials[b]	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron Irradiation Embrittlement	Loss of fracture toughness	Intermediate & Lower Shell	SA302-Gr B, SA533-Gr B, SA336, SA508	1, 11 to 13, (S1)S-15, (S2) 2, (S2) 4, (S2) 5, (S2) 7, (S3)S-1, (S3)S-4 (S3)S-6, (S3)S-7, (S3)S-8 Open issue 3, 4, (S1)S-1, (S2) 3, (S2) 6	Unresolved issue *NUMARC proposal*: Verification of pressure vessel integrity by requirements of fracture toughness & materials surveillance program delineated in Appendices G & H of 10CFR50;[1] guidelines of RG 1.99[3] to estimate embrittlement & 1.154[4] for PTS safety analysis. If requirements of upper-shelf toughness can not be met, then 100% volumetric inspection of beltline; supplemental fracture toughness testing; fracture mechanics analysis showing equivalent margins of safety, & methodology of NUREG 0744 for operation; NUREG 0244,[6] SRP Sect. 5.2.2,[7] & BTP RSB 5-2[8] provide guidance on low temp over pressure protection. If above acceptance criteria can not be met then select plant-specific aging management program that may include mitigation by flux reduction or thermal annealing. *(Contd. next page)*	*NUMARC basis*: Reactor vessel material surveillance program in accordance with 10CFR50, Appendix H[1] implementing recommendations of ASTM STD E185-82,[2] & guidelines of RG 1.99, Rev. 2,[3] ensure that the beltline material is not excessively irradiated during the license renewal term; pressurized thermal shock (PTS) rule 10CFR 50.61 & RG 1.154[4] ensure that vessel integrity is maintained; Appendix G of 10CFR50,[1] using guidelines of ASME Sect. III,[5] Appendix G, assures that beltline materials will maintain adequate upper-shelf toughness; & 10CFR50, Appendix G,[1] with recommendations of NUREG 0244,[6] SRP Sect. 5.2.2,[7] & BTP RSB 5-2,[8] assure that PT limits will not adversely affect vessel integrity. If upper-shelf requirements cannot be met, then 100% volumetric inspection of beltline region in accordance with ASME Sect. XI, Table IWB-2500-1;[9] supplemental fracture toughness tests; fracture mechanics analysis that conservatively demonstrate *(Contd. next page)*

Table B1. Brief summary of technical information and NUMARC/NRC agreements from PWR pressure vessel industry report

Aging–Related Degradation Mechanism	Aging Effects	Components[a]	Materials[b]	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron Irradiation Embrittle-ment	Loss of fracture toughness	Intermediate & Lower Shell	SA302-Gr B, SA533-Gr B, SA336, SA508	Same as previous page	*(Contd. from previous page)* *NRC proposal*: The definition of beltline will be consistent with the regulations; Effectiveness of ISI of vessel components should be ad-dressed & should incorporate requirements of ASME Sect. XI, Apendices VII & VIII.	*(Contd. from previous page)* *NUMARC basis (Contd.)*: adequate margins of safety; & methodology of NUREG 0744[10] for operation. If acceptance criteria of above programs can not be met then select plant-specific aging management program that may include flux reduction by fuel management or shielding, or thermal annealing.
Neutron Irradiation Embrittle-ment	Loss of fracture toughness	Upper (Nozzle) Shell	SA302-Gr B, SA533-Gr B, SA336, SA508	(S3)S-2 Open issue 3, 4, (S1)S-1	Unresolved issue *NUMARC proposal*: Non-significant *NRC proposal*: Fluence level 1×10^{17} n/cm^2 is the level above which a surveillance program is re-quired under Appendix H of 10CFR Part 50 & not the threshold for irradiation damage; & the definition of beltline will be consistent with the regulations.	*NUMARC basis*: The total fast neutron fluence within the license renewal term is less than 1×10^{17} n/cm^2, the reference threshold fluence of concern for radiation dam-age identified in 10CFR50 Appendix H[1] requiring a ma-terials surveillance program & operating temperatures are in the range 274-302°C (525-575°F).
		Primary Coolant Nozzles*	SA336, SA508			

* Includes safety injection nozzles on some vessels.

Table B1. Brief summary of technical information and NUMARC/NRC agreements from PWR pressure vessel industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials[b]	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IGSCC	Crack initiation & growth	Closure Stud Assembly	Alloy 4340	None	ASME Sect. XI,[9] Subsect. IWB, exam. category B-G-1 visual VT-1, surface, & volumetric exam. of closure stud assemblies, in accordance with RG 1.65;[11]	ASME Sect. XI,[9] exam. category B-G-1 & guidelines of RG 1.65[11] for closure stud assemblies is current & effective program.
IGSCC	Crack initiation & growth	Closure Head Dome	SA302-Gr B, SA533-Gr B	16, (S1)S-2, (S1)S-6, (S1)S-9 (S3)S-2, (S3)S-3 Open issues 7b, (S1)S-13, (S3)S-5	Unresolved issue *NUMARC proposal*: Non-significant *NRC proposal*: Low-temperature sensitization of SS cladding is possible.[12-14] Evaluate the effects of oxygen injection during cool down. Although, SCC of low-alloy steel is unlikely in "typical" PWR environment, it may not be true under crevice conditions; consider the information in NUREG/CR-5020.[15]	*NUMARC basis*: Low alloy steels & SS cladding with >5% ferrite content are not susceptible to SCC in PWR environment; implementation of RG 1.43[16] to prevent underclad cracking & guidelines of RG 1.44[17] to avoid sensitization of SS; control of halogens & oxygen in the primary water to <5 & <0.01 ppm, respectively;[18] & monitor & control of water chemistry during shutdown to mitigate the potential of SCC; or the components are not subjected to corrosive environment.
		CRD Mechanism Housing	SA182 Type 304 or 316			
		Refueling Seal Ledge	SA212-Gr B, SA516-Gr 70, SA533-Gr B			
		Closure Head Lifting Lugs	SA302-Gr B, SA533-Gr B			
		Shroud Support Ring	SA212-Gr B, SA516-Gr 70			
		Closure Head Flange	SA336, SA508			
		Vessel Flange	SA336, SA508			
		Upper (Nozzle) Shell	SA302-Gr B, SA533-Gr B, SA336, SA508			
		Primary Coolant Nozzles	SA336, SA508			
		Leakage Monitoring Tubes	SA-312 Type 316			
		Intermediate & Lower Shell	SA302-Gr B, SA533-Gr B, SA336, SA508			
		Bottom Head Dome	SA302-Gr B, SA533-Gr B			

Table B1. Brief summary of technical information and NUMARC/NRC agreements from PWR pressure vessel industry report

Aging–Related Degradation Mechanism	Aging Effects	Components[a]	Materials[b]	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IGSCC	Crack initiation & growth	CRD Mechanism Housing	SB-166	2, 7a, (S1)S-2, (S1)S-11, (S2) 1a	Unresolved issue *NUMARC proposal:* ASME Sect. XI,[9] Subsect. IWB, exam. category B-O requires volumetric & surface exam. of CRD housings; B-E calls for visual VT-2 of the external surfaces of partial penetration welds for leakage during hydrotests; B-N-2 covers visual VT-3 exam. of interior attachment welds for core support pads; plant specific review of component materials; & evaluation, defect repair , & replacement. *NRC proposal:* Alloy 600 should be further evaluated; evaluate the potential of cracking of Inconel 182 based on recent experience of Arkansas Nuclear One Unit 1 described in LER 90-021-00. Effectiveness of ISI of vessel components should be addressed & should incorporate requirements of ASME Sect. XI, Apendices VII & VIII.	*NUMARC basis:* ASME Sect. XI,[9] Subsect. IWB, inspection & testing programs that include exam. category B-O for welds in CRD housing; B-E for partial penetration welds; B-N-2 for integrally-welded interior attachment welds for core support pads; & plant-specific review of component material, e.g., Alloy 600 applications, & fabrication records, component stress reports, & component service temperature; & evaluation, defect repair, & replacement are current & effective to manage SCC damage.
		Leakage Monitoring Tubes	SB-166, SB-167,			
		Core Support Pads (Lugs)	SB-166, SB-168	Open issue Inconel 182*		
		Instrumentation Tubes/Penetrations	SB-166, SB-167			

* Meeting summary, dated March 15, 1991.

Table B1. Brief summary of technical information and NUMARC/NRC agreements from PWR pressure vessel industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials[b]	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion/ Boric Acid Wastage of External Surfaces	Loss of material	Refueling Seal Ledge	SA212-Gr B, SA516-Gr 70, SA533-Gr B	(S3)S-2, (S3)S-3 9	Non-significant	Internal surfaces of reactor vessel components, whether clad or fabricated of SS or Alloy 600, are not subject to corrosive attack in a PWR environment; & corrosion of reactor vessel base metal due to removal of protective cladding results in very low corrosion rates; or components are not exposed to corrosive environment or are not susceptible to potential boric acid leak.
		Closure Head Lifting Lugs	SA302-Gr B, SA533-Gr B			
		Shroud Support Ring	SA212-Gr B, SA516-Gr 70			
		Upper (Nozzle) Shell	SA302-Gr B, SA533-Gr B, SA336, SA508			
		Primary Coolant Nozzles	SA336, SA508			
		Leakage Monitoring Tubes	SB-166 & 167 SA-312-316			
		Intermediate & Lower Shell	SA302-Gr B, SA533-Gr B, SA336, SA508			
		Core Support Pads (Lugs)	SB-166 & 168			
		Bottom Head Dome	SA302-Gr B, SA533-Gr B			
		Instrumentation Tubes/Penetrations	SB-166, SB-167			
Corrosion/ Boric Acid Wastage of External Surfaces	Loss of material	Closure Head Dome	SA302-Gr B, SA533-Gr B	(S3)S-4, (S3)S-6, (S3)S-7	Implementation of Generic Letter 88-05.[19] Exam. category B-P, Subsect. IWB[9] requires VT-2 inspection during leakage & hydrostatic tests & VT-3 inspection of bolt in case of leakage; exam. category B-E provides visual VT-2 of partial penetration welds during system hydrotesting, & visual VT-1 of closure nuts, washers, & bushings.	Recommendations of Generic Letter 88-05[19] are current & effective program to monitor & control primary coolant leakage.
		CRD Mechanism Housing	SB-166, SA182 Type 304 or 316			
		Closure Head Flange	SA336, SA508			
		Closure Stud Assembly	SA-540-B23 or B24, SA-320-L43			
		Vessel Flange	SA336, SA508			

Table B1. Brief summary of technical information and NUMARC/NRC agreements from PWR pressure vessel industry report

Aging–Related Degradation Mechanism	Aging Effects	Components[a]	Materials[b]	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Erosion/ Corrosion (E/C)	Wall thinning	Closure Head Dome	SA302-Gr B, SA533-Gr B	(S3)S-2, (S3)S-3	Non-significant	Austenitic SS & Ni alloys on inside reactor vessel surfaces are resistant to E/C; moderate fluid velocities & single phase flow; and controls on reactor coolant system chemistry.
		CRD Mechanism Housing	SB-166, SA182 Type 304 or 316			
		Refueling Seal Ledge	SA212-Gr B, SA516-Gr 70, SA533-Gr B			
		Closure Head Lifting Lugs	SA302-Gr B, SA533-Gr B			
		Shroud Support Ring	SA212-Gr B, SA516-Gr 70			
		Closure Stud Assembly	SA-540-B23 or B24, SA-320-L43			
		Upper (Nozzle) Shell	SA302-Gr B, SA533-Gr B, SA336, SA508			
		Primary Coolant Nozzles	SA336, SA508			
		Leakage Monitoring Tubes	SB-166 & 167 SA-312-316			
		Intermediate & Lower Shell	SA302-Gr B, SA533-Gr B, SA336, SA508			
		Core Support Pads (Lugs)	SB-166 & 168			
		Bottom Head Dome	SA302-Gr B, SA533-Gr B			
		Instrumentation Tubes/Penetrations	SB-166, SB-167			
Erosion/ Corrosion (E/C)	Wall thinning	Closure Head Flange	SA336, SA508	(S3)S-4, (S3)S-6, (S3)S-7	ASME Sect. XI exam. category B-P requires VT-2 visual system leakage & hydrotests.	ASME Sect. XI,[9] Subsect. IWB exam. category B-P is current & effective program to manage the effects of E/C.
		Vessel Flange	SA336, SA508			

Table B1. Brief summary of technical information and NUMARC/NRC agreements from PWR pressure vessel industry report

Aging–Related Degradation Mechanism	Aging Effects	Components[a]	Materials[b]	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Wear	Attrition	Closure Head Dome	SA302-Gr B, SA533-Gr B	10, (S1)S-14 (S3)S-2, (S3)S-3	Non-significant	Not subjected to motion relative to other components.
		CRD Mechanism Housing	SB-166, SA182 Type 304 or 316			
		Refueling Seal Ledge	SA212-Gr B, SA516-Gr 70, SA533-Gr B			
		Closure Head Lifting Lugs	SA302-Gr B, SA533-Gr B			
		Shroud Support Ring	SA212-Gr B, SA516-Gr 70			
		Upper (Nozzle) Shell	SA302-Gr B, SA533-Gr B, SA336, SA508			
		Primary Coolant Nozzles	SA336, SA508			
		Leakage Monitoring Tubes	SB-166 & 167 SA-312-316			
		Intermediate & Lower Shell	SA302-Gr B, SA533-Gr B, SA336, SA508			
		Bottom Head Dome	SA302-Gr B, SA533-Gr B			
		Instrumentation Tubes/Penetrations	SB-166, SB-167			
Wear	Attrition	Closure Head Flange	SA336, SA508	(S3)S-4, (S3)S-6, (S3)S-7	ASME Sect. XI exam. category B-P requires VT-2 during system leakage & hydrotests, B-G-1 requires visual, surface, & volumetric exam. of closure stud assemblies, & B-N-1 requires VT-3 of support pads.	ASME Sect. XI,[9] Subsect. IWB exam. category B-P for closure head & vessel flanges, B-G-1 for closure studs, & B-N-1 for core support pads are current & effective programs to manage the effects of wear.
		Closure Stud Assembly	SA-540-B23 or B24, SA-320-L43			
		Vessel Flange	SA336, SA508			
		Core Support Pads (Lugs)	SB-166 & 168			

Table B1. Brief summary of technical information and NUMARC/NRC agreements from PWR pressure vessel industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials[b]	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Creep	Change in dimension	Closure Head Dome	SA302-Gr B, SA533-Gr B	(S3)S-2, (S3)S-3	Non-significant	The PWR reactor vessel operating temperatures are <343°C (<650°F) which is well below the creep range; creep is not a concern below 427°C (800°F) for low alloy steels & below 538°C (1000°F) for SS. No significant effect of irradiation on creep of vessel materials has been identified.[20]
		CRD Mechanism Housing	SB-166, SA182 Type 304 or 316			
		Refueling Seal Ledge	SA212-Gr B, SA516-Gr 70, SA533-Gr B			
		Closure Head Lifting Lugs	SA302-Gr B, SA533-Gr B			
		Shroud Support Ring	SA212-Gr B, SA516-Gr 70			
		Closure Head Flange	SA336, SA508			
		Closure Stud Assembly	SA-540-B23 or B24, SA-320-L43			
		Vessel Flange	SA336, SA508			
		Upper (Nozzle) Shell	SA302-Gr B, SA533-Gr B, SA336, SA508			
		Primary Coolant Nozzles	SA336, SA508			
		Leakage Monitoring Tubes	SB-166 & 167 SA-312 Type 316			
		Intermediate & Lower Shell	SA302-Gr B, SA533-Gr B, SA336, SA508			
		Core Support Pads (Lugs)	SB-166 & 168			
		Bottom Head Dome	SA302-Gr B, SA533-Gr B			
		Instrumentation Tubes/Penetrations	SB-166, SB-167			

Table B1. Brief summary of technical information and NUMARC/NRC agreements from PWR pressure vessel industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials[b]	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Thermal Embrittlement	Loss of fracture toughness	Closure Head Dome	SA302-Gr B, SA533-Gr B	8, (S1)S-7, (S1)S-8, (S3)S-2, (S3)S-3	Non-significant	None of the reactor vessel components are susceptible to the various forms of thermal aging, e.g., thermal embrittlement of CASS, temper embrittlement, or strain aging embrittlement, because there is no CASS component (CRD housing made of CASS is outside the scope of this Industry Report), the relatively low operating temperatures of commercial PWRs & the material controls used during manufacturing.
		CRD Mechanism Housing	SB-166, SA182 Type 304 or 316			
		Refueling Seal Ledge	SA212-Gr B, SA516-Gr 70, SA533-Gr B			
		Closure Head Lifting Lugs	SA302-Gr B, SA533-Gr B			
		Shroud Support Ring	SA212-Gr B, SA516-Gr 70			
		Closure Head Flange	SA336, SA508			
		Closure Stud Assembly	SA-540-B23 or B24, SA-320-L43			
		Vessel Flange	SA336, SA508			
		Upper (Nozzle) Shell	SA302-Gr B, SA533-Gr B, SA336, SA508			
		Primary Coolant Nozzles	SA336, SA508			
		Leakage Monitoring Tubes	SB-166 & 167 SA-312-316			
		Intermediate & Lower Shell	SA302-Gr B, SA533-Gr B, SA336, SA508			
		Core Support Pads (Lugs)	SB-166 & 168			
		Bottom Head Dome	SA302-Gr B, SA533-Gr B			
		Instrumentation Tubes/Penetrations	SB-166 & 167			

Table B1. Brief summary of technical information and NUMARC/NRC agreements from PWR pressure vessel industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials[b]	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	Closure Head Dome	SA302-Gr B, SA533-Gr B	5(a)-(c), 6, (S3)S-2, (S3)S-3 Open issues 5(d), 14, (S1)S-3, (S1)S-12, (S1)S-16, (S1)S-17, (S1)S-18	Unresolved issue *NUMARC proposal*: Non-significant *NRC proposal*: Until an agreement is reached on the draft staff discussion paper on fatigue, the issue is unresolved.	*NUMARC basis*: Fatigue usage factors are <0.4 for the current license term, & are anticipated to be significantly less than ASME Code limit of 1.0 for the entire license renewal term.
		Refueling Seal Ledge	SA212-Gr B, SA516-Gr 70, SA533-Gr B			
		Closure Head Lifting Lugs	SA302-Gr B, SA533-Gr B			
		Shroud Support Ring	SA212-Gr B, SA516-Gr 70			
		Closure Head Flange	SA336, SA508			
		Vessel Flange	SA336, SA508			
		Upper (Nozzle) Shell	SA302-Gr B, SA533-Gr B, SA336, SA508			
		Leakage Monitoring Tubes	SB-166, SB-167, SA-312 Type 316			
		Intermediate & Lower Shell	SA302-Gr B, SA533-Gr B, SA336, SA508			
		Core Support Pads (Lugs)	SB-166, SB-168			
		Bottom Head Dome	SA302-Gr B, SA533-Gr B			

Table B1. Brief summary of technical information and NUMARC/NRC agreements from PWR pressure vessel industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials[b]	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	CRD Mechanism Housing	SB-166, SA182 Type 304 or 316	5(a)-(c), 6 (S3)S-4, (S3)S-6, (S3)S-7 Open issues 5(d), 14, (S1)S-3, (S1)S-12, (S1)S-16, (S1)S-17, (S1)S-18	Unresolved issue *NUMARC proposal:* ASME Sect. III[5] Subsect. NB reanalysis of usage factor, regrouping design-basis transients, actual plant transients, cycle monitoring, & partial cycle counting; ASME Sect. XI,[9] Subsect. IWB, inspection requirements of IWB-2500-1; & evaluation, defect repair & replacement. *NRC proposal:* Until an agreement is reached on the draft staff discussion paper on fatigue, the issue is unresolved.	*NUMARC basis:* Verification of continued adequacy of fatigue design basis through reanalysis of usage factor. ASME Sect. XI,[9] Subsect. IWB, inspection & testing programs that include, exam. category B-A & Supplement 6, Appendix VIII for volumetric inspection of pressure retaining welds; exam. category B-D & Supplement 7, Appendix VIII for volumetric inspection of full penetration nozzle welds; exam. category B-E for visual VT-2 inspection of external surfaces of partial penetration welds; & exam. category B-G-1, Appendix VI, & Supplement 8 of Appendix VIII for volumetric & surface inspection of closure studs, replacement in accordance with RG 1.65;[11] & evaluation, defect repair, & replacement are current & effective to manage fatigue degradation.
		Closure Stud Assembly	SA-540-B23 or B24, SA-320-L43			
		Primary Coolant Nozzles	SA336, SA508			
		Instrumentation Tubes (Penetrations)	SB-166, SB-167			

[a] Nozzle safe ends and dissimilar metal welds for the nozzles, as well as CRD upper housing flanges fabricated from CASS, are not included within the scope.
[b] The vessel shell, closure head dome, bottom head dome, and primary coolant nozzles are clad with weld-deposited SS, usually Type 308 or 309. The closure head and vessel flanges have SS clad mating surface. Some reactor vessels have areas of Alloy 600 weld clad.
[c] Comment (S1) S-5 was not included in the table because it deals with clarification, and comment (S1) S-4 was excluded because it deals with scope of IR.

REFERENCES:

1. 10CFR50 - "Code of Federal Regulations, Title 10, Part 50: Domestic Licensing of (Nuclear Power) Production and Utilization Facilities," Office of Federal Register, National Archives and Records Administration, Washington DC.
   (1) 10CFR Part 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."
   (2) Appendix G: "Fracture Toughness Requirements"
   (3) Appendix H: "Reactor Vessel Surveillance Program Requirements."
2. ASTM STD E185-82, "Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels," Annual Book of ASTM Standards, American Society of Testing and Materials, Philadelphia, PA, July 1982.
3. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, Washington, DC, May 1988.
4. Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, Washington, DC, Jan. 1987.
5. ASME B & PV Code, "Boiler and Pressure Vessel Code of Design and Construction Practices," Section III: "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017.
   (1) Appendix G: "Protection Against Non-Ductile Failure"
6. NUREG-0244, "Reactor Vessel Pressure Transient Protection for Pressurized Water Reactors," U.S. Nuclear Regulatory Commission, Washington, DC, 1978.
7. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," Section 5.2.2: Overpressure Protection, U.S. Nuclear Regulatory Commission, Washington, DC, June 1987.
8. Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures," US Nuclear Regulatory Commission.
9. ASME B & PV Code - "Boiler and Pressure Vessel Code of Design and Construction Practices," Section XI: "Rules for In-Service Inspection (ISI) of Nuclear Power Plant Components," American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017.
10. NUREG-0744, "Resolution of Reactor Vessel Material Toughness Safety Issue," U.S. Nuclear Regulatory Commission, Washington, DC, Sept. 1981.
11. Regulatory Guide 1.65, "Materials and Inspection for Reactor Vessel Closure Studs," U.S. Nuclear Regulatory Commission, Washington, DC, Oct. 1973.
12. EPRI NP 1823, "Basic Study on the Variabilities of Fabrication Related Sensitization Phenomenon in Stainless Steel," P. L. Andersen, H. D. Solomen, and M. J. Fox, Electric Power Research Institute, Palo Alto, CA, May 1981.
13. UCRL-15619, "Overview of Low Temperature Sensitization," M. J. Fox and R. D. McWright, 1983.
14. "Low Temperature Sensitization of Type 304 Stainless Steel Pipe Weld Heat Affected Zone," C. G. Schmidt, et al., *Met. Tran. A*, 18A, 1483, Aug. 1987.
15. NUREG/CR-5020, "Summary of Environmentally Assisted Crack-Growth Studies Performed at Westinghouse Electric Corporation Under Funding from the Heavy-Section Steel Technology Program," Oak Ridge National Laboratory, May 1988.

16. Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," U.S. Nuclear Regulatory Commission, Washington, DC.

17. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," U.S. Nuclear Regulatory Commission, Washington, DC, May 1973.

18. EPRI NP-5960-SR, "PWR Primary Water Chemistry Guidelines: Revision 1," C. J. Wood, et al., Electric Power Research Institute, Palo Alto, CA, 1988.

19. NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," March 17, 1988.

20. D. S. Clark and W. R. Varney, *Physical Metallurgy for Engineers*, D. Van Nostrand Co., New York, 1962.

LIST OF PWR PRESSURE VESSEL COMPONENTS:

PWR PRESSURE VESSEL

Closure Head Dome
Control Rod Drive Mechanism Housing
Refueling Seal Ledge
Closure Head Lifting Lugs
Shroud Support Ring
Closure Head Flange
Closure Stud Assembly
Vessel Flange
Upper (Nozzle) Shell
Primary Coolant Nozzles
Leakage Monitoring Tubes
Intermediate & Lower Shell
Core Support Pads (Lugs)
Bottom Head Dome
Instrumentation Tubes/Penetrations

Table B2. Brief summary of technical information and NUMARC/NRC agreements from BWR pressure vessel industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron Irradiation Embrittle-ment	Loss of fracture toughness	Attachment Welds	SS, Alloy 182	G-7, S-23, (S1)G-2, (S1)G-5, (S1)S-1, (S1)S-9	Non-significant	The total fast neutron fluence within the license renewal term is less than 1×10^{17} n/cm^2, the level identified in 10CFR50 Appendix H[1] requiring a materials surveillance program; or the components are made of SS or Ni-Cr-Fe alloys that are not susceptible to neutron embrittlement at fluences less than 1×10^{20} n/cm^2.
		Bottom Head	SA302-Gr B, SA533-Gr B			
		Closure Studs	SA-193, SA-540			
		Nozzles				
		Feedwater	SA508-Cl 2			
		BWR/2 CRD return line (RL)	SA508-Cl 2			
		All Other	SA508-Cl 2			
		Penetrations				
		CRD Stub Tubes	SS, SB-167			
		All Other	SB-167			
		Safe Ends				
		BWR/5 LPCI	SS, SB-166			
		Feedwater	CS, SB-166			
		BWR/2 CRDRL	CS, SB-166			
		All Other	CS, SB-166			
		Vessel Flange	SA336 SA508-Cl 2			
		Vessel Shell				
		Other than beltline	SA302-Gr B, SA533-Gr B			
		Top Head	SA302-Gr B, SA533-Gr B			
Neutron Irradiation embrittle-ment	Loss of fracture toughness	Support Skirt	SA533-Gr B	G-7, (S1)G-2, (S1)G-5, (S1)S-1, S-60	Non-significant	During the license renewal term, shift in reference temp. due to neutron exposure is <11°C (<20°F) & irradiation embrittlement due to thermal neutrons is not significant.

Table B2. Brief summary of technical information and NUMARC/NRC agreements from BWR pressure vessel industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron Irradiation Embrittlement	Loss of fracture toughness	*Nozzles*		G-4, G-5, G-18, S-5, S-7, S-9, S-48 to S-52, S-54, S-55, S-60, S-63, S-67, (S1)G-10, (S1)G-11, (S1)S-2, (S1)S-12, (S1)S-13 Open issue S-53	Agreement that inspection will be performed in accordance with ASME Sect. XI, Appendices VII and VIII. Unresolved issue related to the extent of inspection. *NUMARC proposal:* Verification of pressure vessel integrity by operating & surveillance requirements of Appendices G & H of 10CFR50.[1] In the event that fracture toughness requirements reach acceptance criteria, then current practices to be enhanced, select plant-specific management that may include complete volumetric inspection of beltline, supplemental fracture toughness testing, & fracture mechanics analysis showing equivalent margins of safety for operation. *NRC proposal:* A 100% volumetric inspection of all beltline & all other accessible welds required by ASME Sect. XI.[6] Exemptions for license renewal will be reviewed on a case by case basis.	ASME Section XI, Appendices VII and VIII provide an adequate process for qualifying inspection personnel, equipment, and procedures. *NUMARC basis:* Pressure vessel integrity is assured by operating requirements of Appendix G of 10CFR50,[1] using guidelines of ASME Sect. III,[2] Appendix G, MTEB 5-2,[3] & Reg. Guide 1.99, Rev. 2;[4] & surveillance requirements of Appendix H of 10CFR50, implementing guidelines of ASTM STD E185-82.[5] If requirements of 10CFR50 cannot be met, then select a plant-specific management plan that may include a complete volumetric inspection of beltline region in accordance with ASME Sect. XI, Table IWB-2500-1;[6] supplemental tests to obtain additional evidence of change in fracture toughness; fracture mechanics analysis that conservatively demonstrate adequate margins of safety; & methodology of NUREG-0744[7] for operation.
		BWR/5 LPCI	SA508-Cl 2			
		Vessel Shell				
		Beltline Weld	Low-alloy steel (LAS) weldment			
		Beltline Shell	SA302-Gr B, SA533-Gr B			

Table B2. Brief summary of technical information and NUMARC/NRC agreements from BWR pressure vessel industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IGSCC	Crack initiation & growth	Closure Studs	SA-193, SA-540	S-1, S-3, S-38, (S1)S-19,	ASME Sect. XI,[6] Subsect. IWB, inspection & testing requirements of IWB-2500-1, augmented by NUREG 0313[10] & Generic letter 88-01[11] for nozzles & safe ends, RICSIL 055[8] & RG 1.65[9] for closure studs; & evaluation, defect repair, & replacement.	ASME Sect. XI,[6] Subsect. IWB, inspection & testing programs; exam. category B-G-1 for closure studs, recommendations of RICSIL-055,[8] & replacement in accordance with RG 1.65;[9] categories B-D & B-F for nozzles & safe ends, additional requirements of NUREG 0313[10] implemented by Generic letter 88-01;[11] category B-E for penetrations; & analytical evaluation, defect repair, & replacement are current & effective programs.
		Nozzles		S-10, S-28, S-29, S-36 S-37		
		BWR/5 LPCI	SA508-Cl 2			
		Feedwater	SA508-Cl 2			
		BWR/2 CRDRL	SA508-Cl 2			
		All Other	SA508-Cl 2			
		Penetrations		S-6, S-11, S-12, S-19, (S1)S-14		
		CRD Stub Tubes	SS, SB-167			
		All Other	SB-167			
		Safe Ends				
		BWR/5 LPCI	SS, SB-166	S-13, S-14		
		Feedwater	CS, SB-166			
		BWR/2 CRDRL	CS, SB-166			
		All Other	CS, SB-166			
				Common to above S-25, S-26, S-31, S-32, S-42 to S-45, S-66		

Table B2. Brief summary of technical information and NUMARC/NRC agreements from BWR pressure vessel industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IGSCC	Crack initiation & growth	Attachment Welds	SS, Alloy 182	Common items previous page & S-30, (S1)S-17, (S1)S-22, (S1)S-23	Current practices to be enhanced, select plant-specific aging management program.*	Select plant-specific aging management plan comprising of qualified inspection & monitoring;[12] water chemistry control;[13] & analytical evaluation-repair-replacement.
IGSCC	Crack initiation & growth	Bottom Head	SA302-Gr B, SA533-Gr B	G-5, G-7, G-18, (S1)G-2, (S1)G-5, (S1)S-1, S-41 S-27, S-33 to S-35, S-24, (S1)S-3	Non-significant	Low-alloy steel & SS clad with >5% ferrite are not susceptible to SCC,[14] and/or applied & residual stresses are low, or are not subjected to corrosive environment.
		Vessel Flange	SA336, SA508-Cl 2			
		Vessel Shell				
		Beltline Shell	SA302-Gr B, SA533-Gr B			
		Other than beltline shell & weld	SA302-Gr B, SA533-Gr B			
		Top Head	SA302-Gr B, SA533-Gr B			
		Support Skirt	SA533-Gr B			
IGSCC	Crack initiation & growth	*Vessel Shell* Beltline Weld	LAS weldment	See above & Open issue S-53	Unresolved issue *NUMARC proposal*: Non-significant *NRC proposal*: See neutron irradiation embrittlement of beltline welds.	*NUMARC basis*: Weld metal with at least 5% ferrite is not susceptible to SCC, & control of water chemistry such that oxygen is <10 ppb & halogen level <5 ppm.

* Items concerning chapter six were not the focus of the NRC staff review.

Table B2. Brief summary of technical information and NUMARC/NRC agreements from BWR pressure vessel industry report

<table>
<tr><th>Aging-Related Degradation Mechanism</th><th>Aging Effects</th><th>Components</th><th>Materials[a]</th><th>NRC Comment Number[b]</th><th>NUMARC/NRC Agreement or Proposal</th><th>Basis for Agreement or Proposal</th></tr>
<tr><td rowspan="24">IASCC</td><td rowspan="24">Crack initiation & growth</td><td>Attachment Welds</td><td>SS,
Alloy 182</td><td rowspan="24">G-7,
S-39,
(S1)G-2,
(S1)G-5,
(S1)S-1</td><td rowspan="24">Non-significant</td><td rowspan="24">IASCC is non-significant in low-alloy or CS components subjected to neutron fluences typical of BWR vessel service; IASCC is non-significant for SS and Ni-Cr-Fe alloy components because the total fast neutron fluence within the license renewal term is <1 x 10^{20} n/cm^2 for highly stressed components & <5 x 10^{20} n/cm^2 for components that are subjected to stresses <68 MPa (<10 ksi).</td></tr>
<tr><td>Bottom Head</td><td>SA302-Gr B,
SA533-Gr B</td></tr>
<tr><td>Closure Studs</td><td>SA-193,
SA-540</td></tr>
<tr><td>Nozzles</td><td></td></tr>
<tr><td>BWR/5 LPCI</td><td>SA508-Cl 2</td></tr>
<tr><td>Feedwater</td><td>SA508-Cl 2</td></tr>
<tr><td>BWR/2 CRDRL</td><td>SA508-Cl 2</td></tr>
<tr><td>All Other</td><td>SA508-Cl 2</td></tr>
<tr><td>Penetrations</td><td></td></tr>
<tr><td>CRD Stub Tubes</td><td>SS, SB-167</td></tr>
<tr><td>All Other</td><td>SB-167</td></tr>
<tr><td>Safe Ends</td><td></td></tr>
<tr><td>BWR/5 LPCI</td><td>SS, SB-166</td></tr>
<tr><td>Feedwater</td><td>CS, SB-166</td></tr>
<tr><td>BWR/2 CRDRL</td><td>CS, SB-166</td></tr>
<tr><td>All Other</td><td>CS, SB-166</td></tr>
<tr><td>Vessel Flange</td><td>SA336
SA508-Cl 2</td></tr>
<tr><td>Vessel Shell</td><td></td></tr>
<tr><td>Beltline Weld</td><td>LAS weldment</td></tr>
<tr><td>Beltline Shell</td><td>SA302-Gr B,
SA533-Gr B</td></tr>
<tr><td>All Other</td><td>SA302-Gr B,
SA533-Gr B</td></tr>
<tr><td>Top Head</td><td>SA302-Gr B,
SA533-Gr B</td></tr>
<tr><td>Support Skirt</td><td>SA533-Gr B</td></tr>
<tr><td></td><td></td></tr>
</table>

Table B2. Brief summary of technical information and NUMARC/NRC agreements from BWR pressure vessel industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion	Loss of material/ corrosion product buildup	Attachment Welds	SS, Alloy 182	G-7, (S1)G-2, (S1)G-5, (S1)S-1	Non-significant	The pressure vessel components are internally clad with SS or fabricated of SS or Ni-Cr-Fe alloy which are very resistant to corrosion; for unclad regions corrosion rates in typical BWR environments are very low.
		Bottom Head	SA302-Gr B, SA533-Gr B			
		Closure Studs	SA-193, SA-540			
		Nozzles				
		BWR/5 LPCI	SA508-Cl 2			
		Feedwater	SA508-Cl 2			
		BWR/2 CRDRL	SA508-Cl 2			
		All Other	SA508-Cl 2			
		Penetrations				
		CRD Stub Tubes	SS, SB-167			
		All Other	SB-167			
		Safe Ends				
		BWR/5 LPCI	SS, SB-166			
		Feedwater	CS, SB-166			
		BWR/2 CRDRL	CS, SB-166			
		All Other	CS, SB-166			
		Vessel Flange	SA336 SA508-Cl 2			
		Vessel Shell				
		Beltline Weld	LAS weldment			
		Beltline Shell	SA302-Gr B, SA533-Gr B			
		All Other	SA302-Gr B, SA533-Gr B			
		Top Head	SA302-Gr B, SA533-Gr B			
		Support Skirt	SA533-Gr B			

Table B2. Brief summary of technical information and NUMARC/NRC agreements from BWR pressure vessel industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Erosion/ Corrosion (E/C)	Wall thinning	Attachment Welds	SS, Alloy 182	G-7, S-40, (S1)G-2, (S1)G-5, (S1)S-1, (S1)S-10	Non-significant	Most of the carbon & low-alloy steel components are clad with austenitic SS. Austenitic SSs & Ni-Cr-Fe alloys are resistant to E/C and/or relatively low flow.[15]
		Bottom Head	SA302-Gr B, SA533-Gr B			
		Closure Studs	SA-193, SA-540			
		Nozzles				
		BWR/5 LPCI	SA508-Cl 2			
		Feedwater	SA508-Cl 2			
		BWR/2 CRDRL	SA508-Cl 2			
		All Other	SA508-Cl 2			
		Penetrations				
		CRD Stub Tubes	SS, SB-167			
		All Other	SB-167			
		Safe Ends				
		BWR/5 LPCI	SS, SB-166			
		Feedwater	CS, SB-166			
		BWR/2 CRDRL	CS, SB-166			
		All Other	CS, SB-166			
		Vessel Flange	SA336 SA508-Cl 2			
		Vessel Shell				
		Beltline Weld	LAS weldment			
		Beltline Shell	SA302-Gr B, SA533-Gr B			
		All Other	SA302-Gr B, SA533-Gr B			
		Top Head	SA302-Gr B, SA533-Gr B			
Erosion/ Corrosion (E/C)	Wall thinning	Support Skirt	SA533-Gr B	Same as above	Non-significant	Not exposed to flowing liquid.

Table B2. Brief summary of technical information and NUMARC/NRC agreements from BWR pressure vessel industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	*Nozzles*		G-7, S-15 to S-17, S-22, S-46, (S1)G-2, (S1)G-5, (S1)S-1 (S1)S-4 Open issues: G-8 to G-11, & S-47	Unresolved issue *NUMARC proposal:* Non-significant *NRC proposal:* Licensee verifies that plant-specific analyses, based on a conservative extrapolation of an enveloping set of actual plant transients, demonstrate that fatigue cumulative usage factor will be <1; & analysis should consider the effects of BWR environment, e.g., coolant chemistry, loading frequency, & temp.	*NUMARC basis:* No fatigue cracking under expected operating conditions; & design-basis or plant-specific fatigue usage factor is <0.25 for CS in high stress & high oxygen service, & is <0.4 for all other material & service conditions.
		Other than Feedwater & Uncapped BWR/2 CRDRL	SA508-Cl 2			
		Penetrations				
		Other than CRD Stub Tubes	SB-167			
		Safe Ends				
		Other than Feedwater & BWR/2 CRDRL	SS, SB-166			
		Vessel Shell				
		Beltline Weld	LAS weldment			
		Beltline Shell	SA302-Gr B, SA533-Gr B			
		All Other	SA302-Gr B, SA533-Gr B			
		Top Head	SA302-Gr B, SA533-Gr B			
		Attachment Welds	SS, Alloy 182			
Fatigue	Cumulative fatigue damage	Bottom Head	SA302-Gr B, SA533-Gr B	Same as above & G-12, S-17	Unresolved issue *NUMARC proposal:* Non-significant for all BWRs except BWR-2 for which fatigue is non-significant if the temp. difference of 63°C (145°F) between top & bottom head has not been exceeded on a continuing basis. *NRC proposal:* Same as above	*NUMARC basis:* Same as above

Table B2. Brief summary of technical information and NUMARC/NRC agreements from BWR pressure vessel industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	Vessel Flange	SA336, SA508-Cl 2	Same as for Vessel shell & S-17 Open issue S-21	Unresolved issue *NUMARC proposal*: Non-significant *NRC proposal*: See Vessel shell; & ASME Sect. XI task group has identified fatigue as a significant ARDM for BWR vessel flange.	*NUMARC basis*: Same as for Vessel shell
Fatigue	Cumulative fatigue damage	Closure Studs	SA-193, SA-540	G-18, S-41, (S1)S-4, (S1)S-7, (S1)G-12, G-13, G-14, S-10, S-57 to S-59, (S1)S-15, (S1)S-16, (S1)S-8, (S1)S-20 Open issues G-8 to G-11, & S-47	Unresolved issue *NUMARC proposal*: ASME Sect. III,[2] Subsect. NB reanalysis of usage factor, re-grouping design-basis tran-sients, actual plant tran-sients, cycle monitoring, & partial cycle counting; ASME Sect. XI,[6] Subsect. IWB, in-spection requirements of IWB-2500-1, augmented by RICSIL 055 & RG 1.65 for closure studs, NUREG 0313, Generic letter 88-01, & NUREG 0619 for safe ends & nozzles; & evaluation, defect repair, & replacement. *NRC proposal*: Same as for vessel shell	*NUMARC basis*: Verification of continued adequacy of fatigue design basis through reanalysis of usage factor. ASME Sect. XI,[6] Subsect. IWB, inspection & testing programs that include, exam. category B-G-1 for closure studs, rec-ommendations of RICSIL-055,[8] & replacement in accordance with Reg. Guide 1.65;[9] cate-gories B-D & B-F for nozzles & safe ends, additional require-ments of NUREG 0313[10] im-plemented by Generic letter 88-01,[11] leakage monitoring, establishing plant-specific re-furbishment period for feed-water sparger, & conformity with guidelines of NUREG 0619;[16] & evaluation, defect repair, and replacement.
		Nozzles				
		Feedwater	SA508-Cl 2			
		Safe Ends				
		Feedwater	SA508-Cl 2			
		Uncapped BWR/2 CRDRL	SA508-Cl 2			

Table B2. Brief summary of technical information and NUMARC/NRC agreements from BWR pressure vessel industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	*Penetrations* CRD Stub Tubes	SS, SB-167	See items previous page & Open issue S-20	Unresolved issue *NUMARC proposal*: ASME Sect. III,[2] Subsect. NB reanalysis; ASME Sect. XI,[6] Subsect. IWB, inspection requirements; & evaluation, repair, and replacement. (Acceptable current practice) *NRC proposal*: Fatigue usage factor of stub tubes could be as high as 0.67 during 40-yr life. More frequent inspections may be needed.	*NUMARC basis*: Fatigue reanalysis of CRD stub tubes shows that fatigue usage factors are <0.1. Also, system leakage & hydrotests combined with ASME Sect. XI, UT & visual inspection are effective to manage degradation.
Fatigue	Cumulative fatigue	Support Skirt	SA533-Gr B	G-18 S-41, S-60, (S1)S-4, (S1)S-7, (S1)S-18 Open issue See vessel shell	Unresolved issue *NUMARC proposal*: ASME Sect. III,[3] Subsect. NB reanalysis of usage factor using actual system transients. *NRC proposal*: Same as for vessel shell	*NUMARC basis*: Verification of continued adequacy of the fatigue design basis through reanalysis of fatigue usage factor are current & effective programs.

Table B2. Brief summary of technical information and NUMARC/NRC agreements from BWR pressure vessel industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	*Nozzles*		G-13, G-14, S-10, S-57 to S-59, (S1)S-15, (S1)S-16	Current practices to be enhanced, select plant-specific management program.*	Select plant-specific aging management that may include the following: verification of fatigue design basis through reanalysis of usage factor; online fatigue monitoring; enhanced inspection;[6] water chemistry control; flaw evaluation-repair; & capping the nozzle.
		Uncapped BWR/2 CRDRL	SA508-Cl 2			

* Items concerning chapter six were not the focus of the NRC staff review.

[a] The vessel shell and bottom head are clad with weld-deposited SS. On earlier vessels, the top head was clad, but on BWR/6 the top head was left unclad.

[b] The following comments were not included in the table because they deal with clarification or modification of contents of the IR: G-3, G-15, G-19, G-20, S-2, S-4, S-18, S-61, S-62, (S1)G-6, (S1)G-13, and (S1)S-11. The following comments were also excluded because they deal with scope of the IR: G-1, G-2, G-6, G-16, G-17, (S1)G-1, (S1)G-7, (S1)G-9, & (S1)S-6.

REFERENCES:

1. 10CFR50 - "Code of Federal Regulations, Title 10, Part 50: Domestic Licensing of (Nuclear Power) Production and Utilization Facilities," Office of Federal Register, National Archives and Records Administration, Washington DC.
   (1) Appendix G: "Fracture Toughness Requirements"
   (2) Appendix H: "Reactor Vessel Surveillance Program Requirements."
2. ASME B & PV Code, "Boiler and Pressure Vessel Code of Design and Construction Practices," Section III: "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017.
3. NRC BTP MTEB 5-2, Revision 1, "Fracture Toughness Requirements," US Nuclear Regulatory Commission, Branch Technical Position, July 1981.
4. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, Washington, DC, May 1988.
5. ASTM STD E185-82, "Standard Practice for Conducting Surveillance Tests for Light Water Cooled Nuclear Power Reactor Vessels," Annual Book of ASTM Standards, American Society of Testing and Materials, Philadelphia, PA, July 1982.
6. ASME B & PV Code - "Boiler and Pressure Vessel Code of Design and Construction Practices," Section XI: "Rules for In-Service Inspection (ISI) of Nuclear Power Plant Components," American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017.
7. NUREG-0744, "Resolution of Reactor Vessel Material Toughness Safety Issue," U.S. Nuclear Regulatory Commission, Washington, DC, September 1981.

8. RICSIL 055R1, "RPV Head Stud Cracking," GE Nuclear Energy, San Jose, CA, September 30, 1991.

9. Regulatory Guide 1.65, "Materials and Inspection for Reactor Vessel Closure Studs," U.S. Nuclear Regulatory Commission, Washington, DC, October 1973.

10. NUREG-0313, Rev. 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," W. S. Hazelton and W. H. Koo, January 1988.

11. NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," January 1988.

12. "On-Line Monitoring to Assure Reactor Components," in Special Issue on *Assuming Structural Integrity of Steel Reactor Boundary Components*, J. of Pressure Vessel Piping, Vol. 34, Nos. 1-5, Elsevier Science Publishing, Ltd., Essex, England, pp. 109-110, 1987.

13. EPRI NP-4946-SR, "BWR Normal Water Chemistry Guidelines: 1986 Revision," Electric Power Research Institute, Palo Alto, CA, September 1988.

14. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," U.S. Nuclear Regulatory Commission, Washington, DC, May 1973.

15. NUREG-1344, "Erosion-Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants," April 1989.

16. NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," U.S. Nuclear Regulatory Commission, Washington, DC, November 1980.

LIST OF BWR PRESSURE VESSEL COMPONENTS:

BWR PRESSURE VESSEL

Top Head
  Vessel Shell
  Beltline Shell
Beltline Weld
  All Other
  Vessel Flange
  Closure Studs
  Attachment Welds
Bottom Head
Nozzles
  BWR/5 LPCI
  Feedwater
  BWR/2 CRDRL
  All Other
Penetrations
CRD Stub Tubes
All Other
Safe Ends
BWR/5 LPCI
Feedwater
BWR/2 CRDRL
All Other
Support Skirt

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
General	Age related degradation effects	Concrete- & steel- containment components	Concrete & steel	G-2, S-8, S-31, S-32, S-34, S-71 Open issues: G-7, S-7 S-42, S-67, S-70	Unresolved issue (One-time inspection) *NUMARC proposal*: Resolution of the effects of ARD mechanisms is based upon the review/evaluation of plant-specific features, including appropriate CLB documents/information. General baseline inspection are not warranted if the criteria used in the evaluations are validated. *NRC proposal*: A one time focused inspection of containment is proposed to provide a reasonable level of assurance for continued satisfactory performance of the containment, and to identify existing degradation mechanisms (if any) & take necessary corrective actions so that the containments are able to take the challenges during the license renewal term.	*NUMARC basis*: The ARD mechanisms are evaluated for significance using the available research & industry data. If acceptance criteria (including a review of plant performance history, to assure that contradictory evidence does not exist) are satisfied, then the inspection for that mechanism/component combination is not needed.

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Freeze-thaw	Scaling, cracking, & spalling	Concrete Containments Reinforced/Prestressed •Concrete Dome •Concrete Containment Wall Above Grade •Concrete Containment Wall Below Grade •Concrete Basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser •Concrete Basemat	Concrete	G-12, S-5, S-38 to S-40 Open issue S-10*	For concrete containment structures that meet the basis requirements, freeze-thaw is non-significant ARDM. Open issue: NRC considers that potential freeze-thaw damage of the dome of the concrete containments should be addressed.	Freeze-thaw is non-significant for concrete containment structures located in a geographic regions of negligible weathering conditions (weathering index <100 day-inch/yr);[1] and if located in severe (weathering index >500 day-inch/yr) or moderate (100-500 day-inch/yr) weathering conditions the concrete mix design meets the air content & water-to-cement ratio requirements of ASTM C260[2] or equivalently, the ASME Sect. III, Division 2,[3] paragraph CC 2231.7.1.[3]
Leaching of Calcium Hydroxide	Increase of porosity & permeability	Concrete Containments Reinforced/Prestressed •Concrete Dome •Concrete Containment Wall Above Grade •Concrete Containment Wall Below Grade •Concrete Basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser •Concrete Basemat	Concrete	G-12, S-5, S-11, S-38 to S-40	For concrete containment structures that meet the basis requirements, leaching of calcium hydroxide is non-significant ARDM.	Leaching of calcium hydroxide is non-significant for concrete containment structures not exposed to flowing water; and for structures that are exposed to flowing water but are constructed using the guidance of ACI 201.2R-77[4] to ensure dense, well-cured concrete with low permeability and control cracking through proper arrangement & distribution of reinforcement.

* See also NUMARC/NRC agreement concerning Freeze-thaw pages B-46 (Table B4) and B-135 (Table B9).

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Aggressive Chemical Attack	Increase of porosity & permeability, cracking, & spalling	Concrete Containments Reinforced/Prestressed •Concrete Dome •Concrete Containment Wall Above Grade	Concrete	G-12, S-5, S-38 to S-41	For concrete containment structures that meet the basis requirements, aggressive chemical attack is non-significant ARDM.	Degradation caused by aggressive chemical attack is non-significant for concrete containment structures not exposed to aggressive environment (pH <5.5), or to chloride or sulfate solutions beyond defined limits (>500 ppm chloride,[5] and 1500 ppm sulfate);[6] or if exposed to ground water that exceeds the pH, chloride, sulfate limits the exposure is for intermittent periods only.
Aggressive Chemical Attack	Increase of porosity & permeability, cracking, & spalling	Concrete Containments Reinforced/Prestressed •Concrete Containment Wall Below Grade •Concrete Basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser •Concrete Basemat	Concrete	G-10, G-13, G-15, S-25, S-36, S-37, S-41, S-65, S-66, S-69, S-72, S-75	Accessible concrete surfaces are periodically examined in accordance with the procedures of Type A[7] integrated leak rate test, or in accordance with ASME Sect. XI, Subsect. IWL.[8] Management for the effects of aggressive chemical attack of concrete surfaces that are not periodically examined due to inaccessibility requires further plant-specific evaluation.	In cases where containment concrete is exposed to aggressive groundwater (pH <5.5, chloride >500 ppm, & sulfate >1500 ppm), periodic inspection of accessible concrete surfaces as part of Type A integrated leak rate test performed under Appendix J, 10CFR50,[7] or in accordance with ASME Sect. XI, Subsect. IWL,[8] exam. category L-A, & guidelines of ACI 201.1.[9] Further evaluation for management of inaccessible areas is to be justified on a plant-specific basis.

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Reaction with Aggregates	Expansion & cracking	Concrete Containments Reinforced/Prestressed •Concrete Dome •Concrete Containment Wall Above Grade •Concrete Containment Wall Below Grade •Concrete Basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser •Concrete Basemat	Concrete	G-12, S-5, S-38 to S-40 Open issue S-12*	Unresolved issue *NUMARC proposal*: For concrete containment structures that meet the basis requirements, reaction with aggregates is non-significant ARDM. *NRC proposal*: Alkaline-aggregate reactions can not be ruled out. Tests involving aggregates alone are not satisfactory in predicting aggregate performance. Alkaline-aggregate reaction may occur after 25 or more years. Use of pozzolans & low alkali content cement may not control reactions for concrete fabricated using sand-gravel aggregates.	*NUMARC basis*: Reactions with aggregates are non-significant for concrete containment structures constructed either from aggregate taken from geographic regions other than those known to yield aggregates suspected of or known to cause alkali-aggregate reactions;[4,10] or from aggregate that was investigated, tested, & subject to petrographic exam. conducted in accordance with ASTM C295,[11] or ASTM C227,[12] which showed that the aggregate is non-reactive; or if the aggregate was examined & found potentially reactive, the provisions of ACI 201.2R-77[4] were followed.

* See also NUMARC/NRC agreement concerning Reaction with Aggregates pages B-49 (Table B4) and B-139 (Table B9).

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Elevated Temp.	Loss of strength & modulus	Concrete Containments Reinforced/Prestressed •Concrete Dome •Concrete Containment Wall Above Grade •Concrete Containment Wall Below Grade •Concrete Basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser •Concrete Basemat	Concrete including embedded CS & reinforcing CS (rebar) in concrete	G-12, S-5, S-19, S-38 to S-40 S-44, S-45	Non-significant if it meets the basis requirements.	Degradation from exposure to elevated temperatures is non-significant for concrete containment structures maintained at operating temperatures<66°C 150°F) and local area temperatures <93°C (200°F);[3,13] or for structures that operate above these limits, plant-specific justification is provided in accordance with ACI 349-85.[13]
Elevated Temp.	Loss of strength & modulus	Concrete Containments Reinforced/Prestressed •Dome Reinforcing Steel •Cont. Wall Reinforcing Steel Above Grade •Cont. Wall Reinforcing Steel Below Grade •Basemat Reinforcing Steel Free-Standing Steel Containment with Flat Bottom & an Ice Condenser •Basemat Reinforcing Steel	CS	Same as above	Non-significant	Normal operating temperatures within PWR containment structures are 49-66°C (120-150°F) which are well below the 371°C (700°F) level at which the structural integrity of rebar/concrete combination begins to be significantly affected.[14]
Elevated Temp.	Loss of strength & modulus	Concrete Containments Prestressed •Prestressing Tendons	CS	Same as above	Non-significant	PWR containment prestressing tendons are subjected to temperatures <60°C (140°F).

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Elevated Temp.	Loss of strength & modulus	Concrete Containments Reinforced/Prestressed	CS	Same as for rebar	Non-significant	Normal operating temperatures within PWR containment structures are 49-66°C (120-150°F) which are well below the 371°C (700°F) level at which the structural integrity of rebar/concrete combination begins to be significantly affected.[14]
		•Containment Liner Int. Surface				
		•Containment Liner Above Grade Exterior Surface				
		•Containment Liner Below Grade Exterior Surface				
		•Basemat Liner Interior Surface				
		•Basemat Liner Exterior Surface				
		•Liner Anchors Above Gr.				
		•Liner Anchors Below Gr.				
		Free-Standing Cylindrical & Spherical Steel Containment with Elliptical Bottom				
		•Containment Shell Int. Surface				
		•Containment Shell Ext. Surf.				
		•Embedded Shell Region				
		•Sand Pocket Region				
		Free-Standing Steel Cont. with Flat Bottom & an Ice Condenser				
		•Dome Shell Interior Surface				
		•Dome Shell Exterior Surface				
		•Cylindrical Shell Int. Surface				
		•Cylindrical Shell Ext. Surface				
		•Embedded Shell Region				
		•Basemat Liner				
		•Liner Anchors				
		Common Components				
		•Penetration Sleeves				
		•Penetration Bellows	SS, CS			
		•Personnel Airlock	CS			
		•Equipment Hatches				

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Irradiation of Concrete	Loss of strength & modulus	Concrete Containments Reinforced/Prestressed •Concrete Dome •Concrete Containment Wall Above Grade •Concrete Containment Wall Below Grade •Concrete Basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser •Concrete Basemat	Concrete including embedded CS & reinforcing CS (rebar) in concrete	G-12, S-5, S-38 to S-40, S-46, S-51, S-55	Non-significant	The neutron fluence levels & maximum integrated gamma doses incurred by PWR containment concrete during the license renewal term do not exceed the level at which measurable degradation of concrete strength properties occurs (10^{19} n/cm^2 & 10^{10} rads, respectively).[5,15]
Irradiation of Steel	Loss of fracture toughness	Concrete Containments Reinforced/Prestressed •Dome Reinforcing Steel •Containment Wall Reinforcing Steel Above Grade •Containment Wall Reinforcing Steel Below Grade •Basemat Reinforcing Steel Free-Standing Steel Containment with Flat Bottom & an Ice Condenser •Basemat Reinforcing Steel	CS	G-12, S-5, S-38 to S-40, S-46, S-51, S-55	Non-significant	The cumulative radiation exposure experienced by reinforced concrete PWR containment structures during the license renewal term is far below the level of 10^{19} n/cm^2 for degradation of reinforcing steel.[16]
Irradiation of Steel	Loss of fracture toughness	Concrete Containments Prestressed •Prestressing Tendons	CS	Same as above	Non-significant	PWR containment tendons & corrosion inhibitors will not receive enough radiation exposure during the license renewal term to incur age related degradation ($<4 \times 10^{16}$ n/cm^2, & 10^{10} rads, respectively).[13]

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Irradiation of Steel	Loss of fracture toughness	Concrete Containments Reinforced/Prestressed	CS	G-12, S-5, S-38 to S-40, S-46, S-51, S-55	Non-significant	The cumulative radiation exposure experienced by concrete PWR containment liners or free-standing steel containment shells throughout the license renewal term is far below the level of 2×10^{17} n/cm^2 (>1 MeV) which could cause a change in mechanical or physical properties.[17]
		•Containment Liner Int. Surface				
		•Containment Liner Above Grade Exterior Surface				
		•Containment Liner Below Grade Exterior Surface				
		•Basemat Liner Interior Surface				
		•Basemat Liner Exterior Surface				
		•Liner Anchors Above Gr.				
		•Liner Anchors Below Gr.				
		Free-Standing Cylindrical & Spherical Steel Containment with Elliptical Bottom				
		•Containment Shell Int. Surface				
		•Containment Shell Ext. Surf.				
		•Embedded Shell Region				
		•Sand Pocket Region				
		Free-Standing Steel Cont. with Flat Bottom & an Ice Condenser				
		•Dome Shell Interior Surface				
		•Dome Shell Exterior Surface				
		•Cylindrical Shell Int. Surface				
		•Cylindrical Shell Ext. Surface				
		•Embedded Shell Region				
		•Basemat Liner				
		•Liner Anchors				
		Common Components				
		•Penetration Sleeves				
		•Penetration Bellows	SS, CS			
		•Personnel Airlock	CS			
		•Equipment Hatches				

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion of Embedded Steel	Cracking, spalling, loss of bond, & loss of material	Concrete Containments Reinforced/Prestressed •Concrete Dome •Concrete Containment Wall Above Grade •Dome Reinforcing Steel •Containment Wall Reinforcing Steel Above Grade	Embedded CS & reinforcing CS (rebar) in concrete	G-12, S-5, S-13, S-38 to S-40, S-43, S-49, S-60	For concrete containment structures that meet the basis requirements, corrosion of embedded steel or rebar is non-significant ARDM.	Non-significant for concrete not exposed to aggressive environment, pH <11.5 or chlorides >500 ppm;[18] or if exposed to aggressive environment concrete has relatively high strength [27.6 MPa (4 ksi)], low water-to-cement ratio (0.35-0.45), adequate air entrainment (3-6%), low permeability, and designed in accordance with ACI 318[5] or ASME Sect. III, Div. 2.[3]
Corrosion of Embedded Steel	Cracking, spalling, loss of bond, & loss of material	Concrete Containments Reinforced/Prestressed •Concrete Containment Wall Below Grade •Concrete Basemat •Containment Wall Reinforcing Steel Below Grade •Basemat Reinforcing Steel Free-Standing Steel Containment with Flat Bottom & an Ice Condenser •Concrete Basemat •Basemat Reinforcing Steel	Embedded CS & reinforcing CS (rebar) in concrete	G-10, G-13, G-15, S-23, S-25, S-36, S-37, S-49, S-60, S-65, S-66, S-69, S-72 Open issue S-42*	Accessible concrete surfaces are periodically examined in accordance with procedures of Type A[7] integrated leak rate test, or ASME Sect. XI, Subsect. IWL.[8] Concrete surfaces that are not periodically examined due to inaccessibility require further plant-specific evaluation. Open issue: NRC considers that potential degradation due to chloride corrosion of the PWR containments should be addressed.	In cases where containment concrete is exposed to aggressive groundwater (pH <5.5, chloride >500 ppm, & sulfate >1500 ppm) periodic inspection of accessible concrete surfaces as part of Type A integrated leak rate test performed under Appendix J, 10CFR50,[7] or in accordance with ASME Sect. XI, Subsect. IWL,[8] exam. category L-A, & guidelines of ACI 201.1.[9] Further evaluation for management of inaccessible areas is to be justified on a plant-specific basis.

* See also NUMARC/NRC agreement concerning Corrosion of Embedded Steel pages B-52 (Table B4) and B-141 (Table B9).

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion of Structural Steel & Liner	Loss of material	Concrete Containments Reinforced/Prestressed •Containment Liner Interior Surface •Containment Liner Above Grade Exterior Surface •Basemat Liner Interior Surface •Liner Anchors Above Grade Common Components •Penetration Sleeves •Dissimilar Metal Welds •Personnel Airlock •Equipment Hatches	CS	G-5, G-12, S-5, S-16, S-38 to S-40, S-62	For PWR containment liners that meet the basis requirements, corrosion is non-significant ARDM.	Galvanic corrosion & corrosion due to aggressive aqueous solutions will not occur if dissimilar metals are not used in construction & if aggressive ground water (chlorides >500 ppm) is not present. SCC is not significant because PWR containment liners only experience compressive stresses due to dead load & prestress.
Corrosion of Structural Steel & Liner	Loss of material	Free-Standing Cylindrical & Spherical Steel Containment with Elliptical Bottom •Containment Shell Interior Surface •Containment Shell Exterior Surface Free-Standing Steel Containment with Flat Bottom & an Ice Condenser •Dome Shell Interior Surface •Dome Shell Exterior Surface •Cylindrical Shell Interior Surface •Cylindrical Shell Exterior Surface Common Components •Penetration Bellows	CS SS	G-5, G-12, S-5, S-6, S-16, S-38 to S-40	For PWR free standing steel containment that meet the basis requirements, corrosion is non-significant ARDM.	Galvanic corrosion & SCC are not significant ARD mechanisms if dissimilar metals are not used in the construction of PWR free-standing steel containment; & in the case of SS bellows assemblies for CS vent lines or pipe sleeves if the materials are protected by shields from corrosive environment.

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion of Structural Steel & Liner	Loss of material	Concrete Containments Reinforced/Prestressed •Containment Liner Below Grade Exterior Surface •Basemat Liner Exterior Surface •Liner Anchors Below Grade Free-Standing Cylindrical & Spherical Steel Containment with Elliptical Bottom •Embedded Shell Region •Sand Pocket Region Free-Standing Steel Containment with Flat Bottom & an Ice Condenser •Embedded Shell Region •Basemat Liner •Liner Anchors	CS	G-5, G-8, G-9, G-11, G-15, G-16, S-24, S-25, S-36, S-37, S-54, S-64, S-66, S-69, S-72, S-73, S-74, S-77	ASME Sect. XI,[8] Subsect. IWE, requires visual examination of accessible surfaces prior to any Type A test to uncover evidence of structural degradation; supplementary methods for condition monitoring confirmation of minimum required wall thickness by UT methods;[19] affected areas evaluated in accordance with criteria of ASME Sect. III,[20] & repair & replacement in accordance with ASME Sect. XI,[8] Subsect. IWE-4000 & 7000. For inaccessible areas, further plant-specific evaluation is required.	Periodic examination & monitoring of accessible areas in accordance with ASME Sect. XI, Subsect. IWE,[8] exam. categories E-D, E-F, & E-P; & areas exempt from inspection monitored to maintain required wall thickness minimums by UT performed[19] in accordance with existing standards; are effective programs. Further evaluation for management of inaccessible areas is to be justified on a plant-specific basis.

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion of Tendons	Loss of material	Concrete Containments Prestressed				
		•Prestressing Tendons	CS	G-8, G-9, G-19, G-11, G-16, S-9, S-18, S-50, S-64 Open issue S-7, S-61	Unresolved issue *NUMARC proposal:* RG 1.35 & ASME Sect XI, Subsect. IWL require testing & examination of tendons & leakage of corrosion protection medium; VT-1 includes anchor head, bearing plates, wedges, buttonheads, shims, & concrete; acceptance criteria IWL-3221.2 include absence of physical damage, corrosion limits; & minimum specified material properties; IWL-2525-1 examines corrosion protection medium & any free water; repair & replacement. *NRC proposal:* Large amount of grease leakage can degrade concrete strength. IWL lacks certain criteria in RG 1.35. Also, anchor heads have failed in prestressed concrete containments.	*NUMARC basis:* Examination of tendon anchorage hardware in accordance with the provisions of RG 1.35[21] or the requirements of ASME Sect. XI,[8] Subsect. IWL, including visual examination of tendon anchorage hardware, evaluation of corrosion protection medium, & identification & testing of any free water; repair & replacement; are effective in managing degradation by corrosion of prestressing tendons & anchor heads.

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	Concrete Containments Reinforced/Prestressed	Concrete including embedded CS & reinforcing CS (rebar) in concrete	G-12, S-5, S-21, S-38 to S-40	Non-significant	Containment concrete, reinforcing steel, prestressing systems, steel liners, & free-standing steel containments are designed to have good fatigue strength properties (10^5 cycles) of below yield load in accordance with ASME Sect. III, Division 2,[3] & ACI 215R-74[23] codes. Potential low-cycle fatigue due to localized elevated temperatures are not anticipated to be significant.
		•Concrete Dome				
		•Concrete Containment Wall Above Grade				
		•Concrete Containment Wall Below Grade				
		•Concrete Basemat				
		•Dome Reinforcing Steel				
		•Containment Wall Reinforcing Steel Above Grade				
		•Containment Wall Reinforcing Steel Below Grade				
		•Basemat Reinforcing Steel				
		Free-Standing Cylindrical & Spherical Steel Containment with Elliptical Bottom	CS			
		•Containment Shell Int. Surface				
		•Containment Shell Ext. Surf.				
		Free-Standing Steel Cont. with Flat Bottom & an Ice Condenser				
		•Dome Shell Interior Surface				
		•Dome Shell Exterior Surface				
		•Cylindrical Shell Int. Surface				
		•Cylindrical Shell Ext. Surface				
		•Concrete Basemat				
		•Basemat Reinforcing Steel				
		Common Components				
		•Personnel Airlock				
		•Equipment Hatches				

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	Common Components •Penetration Sleeves •Penetration Bellows	 CS SS, CS	G-11, G-16, S-14, S-64, Open issue G-3, G-4, S-68	Unresolved issue *NUMARC proposal*: Fatigue reanalysis of penetrations in accordance with ASME Sect. III, Subsect. NB;[20] & ISI in accordance with ASME Sect. XI, Subsect. IWE,[8] exam. category E-B requires visual VT-1 of containment penetration welds, including bellows seal circumferential weld. *NRC proposal*: Sensitivity evaluations & appropriate references should be included for fatigue of bellows assemblies. Fatigue of penetration sleeve anchors can be induced by thermal cyclic loading & may not be detectable by the leak rate tests.	*NUMARC basis*: Fatigue reanalysis conducted in accordance with ASME Sect. III,[20] Subsect. NB, to show that fatigue usage factors are maintained below unity throughout the license renewal term, monitoring of penetration temperatures may be required to establish the magnitude & frequency of transients; ISI in accordance with ASME Sect. XI,[8] Subsect. IWE, to ensure that component integrity is maintained in the presence of known or suspected fatigue damage, including a flaw; are effective to manage the effects of fatigue damage accumulation or fatigue crack growth.
Loss of Prestress	Reduction of design margin	Concrete Containments Prestressed •Prestressing Tendons*	 CS	G-9, G-14, S-18, S-33, S-45, S-47, S-48, S-52, S-53	Inspection & load monitoring to detect progressive reductions in the levels of prestress; evaluation for the license renewal term using RG 1.35;[21] & corrective action.	Periodic monitoring of prestressing losses in accordance with tendon lift-off test of RG 1.35;[21] validation with predictions of prestressing loss; identification of reportable conditions of RG1.35; documentation of RG 1.16;[22] & plant-specific evaluation & corrective actions are effective in managing the effects of pre-stressing loss.

* Rock anchors are outside the scope of this review.

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Concrete Interaction with Aluminum	Loss of strength	Concrete Containments Reinforced/Prestressed •Concrete Dome •Concrete Containment Wall Above Grade •Concrete Containment Wall Below Grade •Concrete Basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser •Concrete Basemat	Concrete	S-59	For concrete containment structures that meet the basis requirements, concrete interaction with aluminum is non-significant ARDM.	Adverse effects of concrete interactions with aluminum would have been identified during the initial acceptance test prior to initial operation. If no degradation of concrete strength was noted during initial structural testing, or if aluminum piping were not used for concrete placement, then concrete interaction with aluminum is not significant.
Settlement	Cracks, distortion, increase in component stress level	Concrete Containments Reinforced/Prestressed •Concrete Basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser •Concrete Basemat	Concrete	Open issue S-63*	Unresolved issue *NUMARC proposal*: Structure settlement monitoring during construction, & continued monitoring during operation for sites with soft soil and/or significant changes in ground water conditions. *NRC proposal*: Effect of settlement of the PWR containments need to be evaluated.	*NUMARC basis*: Structure settlement monitoring initiated during construction phase to confirm that actual settlement is consistent with the allowances included in design basis, & continued settlement monitoring during operation for sites with soft soil and/or significant changes in ground water conditions.

* See also NUMARC/NRC agreement concerning Settlement pages B-62 (Table B4) and B-154 (Table B9).

Table B3. Brief summary of technical information and NUMARC/NRC agreements from PWR containment structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Strain Aging	Loss of fracture toughness	Free-Standing Cylindrical & Spherical Steel Containment with Elliptical Bottom	CS	None	For containment structures that meet the basis requirements, strain aging is non-significant ARDM.	Dynamic strain aging is non-significant for free standing steel containment structures that do not allow loads to exceed the elastic limit. Static strain aging is non-significant for free standing steel containment structures that were not cold worked; or if cold worked during the forming process, the plates were normalized or stress relieved or both after forming with minimal (<5%) subsequent cold working.
		•Containment Shell Int. Surf.				
		•Containment Shell Ext. Surf.				
		•Embedded Shell Region				
		•Sand Pocket Region				
		Free-Standing Steel Containment with Flat Bottom & an Ice Condenser				
		•Dome Shell Interior Surface				
		•Dome Shell Exterior Surface				
		•Cylindrical Shell Int. Surface				
		•Cylindrical Shell Ext. Surface				
		Common Components				
		•Penetration Sleeves				
		•Penetration Bellows	CS			
		•Personnel Airlock	CS			
		•Equipment Hatches				

[a] Designation for steel materials are as follows: SS = stainless steel, CS = carbon steel.

[b] The following comments were not included in the table because they deal with clarification or modification of contents of the IR: S-1, S-2, S-4, S-15, S-26, and S-78. The following comments were also excluded because they deal with scope of the IR: G-1, S-3, S-17, S-20, S-22, S-27 to S-30, S-35, S-56 to S-58, and S-76.

REFERENCES:

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2. ASTM C260-77, "Specification for Air Entraining Admixture for Concrete," American Society for Testing and Materials, Philadelphia, PA, 1977.
3. ASME B & PV Code, "Boiler and Pressure Vessel Code of Design and Construction Practices," Section III, Division 2: "Code for Concrete Reactor Vessel and Containments," American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017, 1986 Edition. Subsection CC: "Concrete Containments."
4. ACI 201.2R-77, "Guide to Durable Concrete," American Concrete Institute.

5. ACI 318, "Building Code Requirements for Reinforced Concrete," American Concrete Institute.

6. "Concrete Degradation Monitoring and Evaluation," N. Prasad et al., NUREG/CP-0100, Proc. Intl. Nuclear Power Plant Aging Symposium, U.S. Nuclear Regulatory Commission, Washington DC.

7. 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Office of the Federal Register National Archives and Records Administration, US Government Printing Office, Washington, DC.

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Subsection IWL: "Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants."

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13. ACI 349-85, "Code Requirements for Nuclear Safety Related Concrete Structures," American Concrete Institute.

14. "Resistance to High Temperatures," P. Smith, in *Significance of Tests and Properties of Concrete - Making Materials*, American Society for Testing and Materials, STP 169B, Chapter 25, 1978.

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16. "Concrete, Cements, Mortars, and Grouts," H. E. Hungerford, et al., *Engineering Compendium on Radiation Shielding*, Section 9.1.12, Volume II, Springer-Verlag New York, Inc., NY, 1975.

17. "Neutron Irradiation Embrittlement of Reactor Pressure Vessel Steels," L. E. Steele, International Atomic Energy Agency, Vienna, Austria, 1975.

18. "Composition and Properties of Concrete," Second Edition, G. E Troxell, H. E. Davis, and J. W. Kelly, McGraw-Hill, 1968.

19. ASTM E797-81, "Practice for Measuring Thickness by Manual Ultrasonic Pulse-Echo Contact Method," American Society of Testing and Materials, Philadelphia, PA, 1981.

20. ASME B & PV Code - "Boiler and Pressure Vessel Code of Design and Construction Practices," Section III: "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017.
Subsection NB: "Class 1 Components."
Subsection NE: "Class MC Components."

21. Regulatory Guide 1.35, Revision 3, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures," U.S. Nuclear Regulatory Commission, July11, 1990.

22. Regulatory Guide 1.16, Revision 4, "Reporting of Operating Information - Appendix A, Technical Specification," U.S. Nuclear Regulatory Commission, August 1975.

23. ACI 215 R-74, "Consideration for Design of Concrete Structures Subjected to Fatigue Loading," American Concrete Institute, 1986.

LIST OF PWR CONTAINMENT COMPONENTS:

CONCRETE CONTAINMENTS REINFORCED/PRESTRESSED

Concrete Dome
Dome Reinforcing Steel
Concrete Containment Wall Above Grade
Containment Wall Reinforcing Steel Above Grade
Concrete Containment Wall Below Grade
Containment Wall Reinforcing Steel Below Grade
Concrete Basemat
Basemat Reinforcing Steel
Containment Liner Interior Surface
Containment Liner Above Grade Exterior Surface
Containment Liner Below Grade Exterior Surface
Basemat Liner Interior Surface
Basemat Liner Exterior Surface
Liner Anchors Above Grade
Liner Anchors Below Grade

COMMON COMPONENTS

Penetration Sleeves
Penetration Bellows
Personnel Airlock
Equipment Hatches

FREE-STANDING STEEL CONTAINMENT WITH FLAT BOTTOM & AN ICE CONDENSER

Dome Shell Interior Surface
Dome Shell Exterior Surface
Cylindrical Shell Interior Surface
Cylindrical Shell Exterior Surface
Embedded Shell Region
Concrete Basemat
Basemat Reinforcing Steel
Basemat Liner
Liner Anchors

FREE-STANDING CYLINDRICAL & SPHERICAL STEEL CONTAINMENT WITH ELLIPTICAL BOTTOM

Containment Shell Interior Surface
Containment Shell Exterior Surface
Embedded Shell Region
Sand Pocket Region

CONCRETE CONTAINMENTS PRESTRESSED Only

Prestressing Tendons

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Freeze-thaw	Scaling, cracking, & spalling	Mark III Concrete Containments* •Concrete Containment Walls Above Grade •Concrete Containment Walls Below Grade •Concrete Dome	Concrete	G-1, G-2, G-3, G-4, G-11, G-16, G-17, G-19, G-21, S-74, S-81,	For Mark III concrete containment components that meet the basis requirements, freeze-thaw is a non-significant ARDM.§	Freeze-thaw is non-significant for Mark III concrete containment components located in a geographic region of negligible weathering conditions (weathering index <100 day-inch/yr); and if located in severe (weathering index >500 day-inch/yr) or moderate (100-500 day-inch/yr) weathering conditions the concrete mix design meets the air content and water-to-cement ratio requirements of ASTM-C260-77[1] or ASME Sect. III, Division 2,[2] CC-2231.7.1; or the susceptible surfaces are protected by shielding.

* The Mark I and II concrete containments are protected from freezing by the secondary containment.
§ See also NUMARC/NRC agreement concerning Freeze-thaw page B-135 (Table B9) and comment S–10, page B-29 (Table B3).

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Leaching of Calcium Hydroxide	Increase of porosity & permeability	Mark I Concrete Containments •Drywell Concrete •Torus Concrete Mark II Concrete Containments •Containment Concrete •Concrete Basemat Mark III Concrete Containments •Concrete Containment Wall Above Grade •Concrete Containment Wall Below Grade •Concrete Dome •Concrete Basemat Mark III Steel Containments •Concrete Basemat •Concrete Fill in Annulus	Concrete	G-1, G-2, G-3, G-4, G-11, G-16, G-17, G-19, G-21, S-74, S-81	For containment components that meet the basis requirements, leaching of calcium hydroxide is a non-significant ARDM.	Leaching of calcium hydroxide is non-significant for containment concrete components not exposed to flowing water; and for structures that are exposed to flowing water but are constructed using the guidance of ACI 201.2R-77[3] to ensure dense, well-cured concrete with low permeability and control cracking through proper arrangement and distribution of reinforcement.

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Aggressive Chemical Attack	Increase of porosity & permeability, cracking, & spalling	Mark I Concrete Containment •Drywell Concrete •Torus Concrete Mark III Concrete Containments •Concrete Containment Walls Above Grade •Concrete Dome	Concrete	G-1, G-2, G-3, G-4, G-11, G-16, G-17, G-19, G-21, S-71, S-74, S-81, S-98	For containment concrete components that meet the basis requirements, aggressive chemical attack is a non-significant ARDM.	Degradation caused by aggressive chemical attack is non-significant for containment components not exposed to aggressive environment (pH <5.5), or to chloride or sulfate solutions beyond defined limits (>500 ppm chloride,[4] and >1500 ppm sulfate);[5] or if exposed to groundwater that exceeds the pH, chloride, sulfate limits the exposure is for intermittent periods only.
Aggressive Chemical Attack	Increase of porosity & permeability, cracking, & spalling	Mark II Concrete Containments •Concrete Basemat Mark III Concrete Containments •Concrete Containment Walls Below Grade •Concrete Basemat Mark III Steel Containments •Concrete Basemat	Concrete	G-1 to G-4, G-11, G-16, G-17, G-19 to G-21, S-71, S-98	Management for the effects of aggressive chemical of concrete surfaces that are not periodically examined due to inaccessibility requires plant-specific evaluation.	Plant-specific program is to be justified for the inaccessible areas.

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Reaction with Aggregates	Expansion & cracking	Mark I Concrete Containments	Concrete	G-1, G-2, G-3, G-4, G-11, G-16, G-17, G-19, G-21, S-74, S-81	For containment concrete components that meet the basis requirements, reaction with aggregates is a non-significant ARDM.§	Reaction with aggregates is non-significant for BWR containment constructed either from aggregate taken from geographic regions other than those known to yield aggregates suspected of or known to cause alkali-aggregate reactions,[6] or from aggregates was investigated, tested, and subject to a petrographic examination conducted in accordance with ASME Section III, Division 2, Class CC,[2] ASTM C295,[7] or ASTM C227,[8] which showed that the aggregate is nonreactive; or the aggregate was examined & found to be potentially reactive, but the provisions of ACI 201.2R-77[3] were followed.
		•Drywell Concrete				
		•Torus Concrete				
		Mark II Concrete Containments				
		•Containment Concrete				
		•Concrete Basemat				
		Mark III Concrete Containments				
		•Concrete Containment Wall Above Grade				
		•Concrete Containment Wall Below Grade				
		•Concrete Dome				
		•Concrete Basemat				
		Mark III Steel Containments				
		•Concrete Basemat				
		•Concrete Fill in Annulus				

§ See also NUMARC/NRC agreement concerning Reaction with Aggregates page B-139 (Table B9) and comment S-12, page B-31 (Table B3).

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Atmospheric Corrosion	Loss of material	Common Components		G-1, G-2, G-3, G-4, G-7, G-11, G-16, G-17, G-19, G-21, S-3, S-9, S-12 to S-15, S-21, S-32 to S-35, S-38, S-43, S-46, S-48, S-61, S-63	For containment steel components that meet basis requirements, atmospheric corrosion is a non-significant ARDM.	Atmospheric corrosion is not a significant ARD for steel containment components fabricated from stainless steel, or for components having intact protective coatings, or for components having a corrosion allowance ≥1/32 inch. Austenitic SS is corrosion resistant. The atmospheric corrosion for carbon and low alloy steels without protective coatings is less than 0.5 mils per year or <1/32 inches (0.03125 inches) for a 60-year period.[9,10]
		•Penetration Sleeves	CS			
		•Penetration Bellows	SS			
		•Personnel Airlock	CS			
		•Equipment Hatches				
		•CRD Hatch				
		Mark I Steel Containment				
		•Drywell Interior Surface	CS			
		•Drywell Head				
		•Torus Interior Surface				
		•Torus Exterior Surface				
		•Vent Lines				
		•Ring Girder				
		•Vent Line Bellows	SS			
		•Vent Header	CS			
		•Downcomer & Bracing				
		•Vent System Supports				
		•Torus Seismic Restraints				
		•Torus Support Columns/Saddles	CS, graphite			
		Mark II Steel Containments				
		•Drywell Interior Surface	CS			
		•Drywell Head				
		•Suppression Chamber Interior Surface				
		•Downcomer Pipes & Bracing				
		Mark III Steel Containments	CS			
		•Cont. Shell Interior Surface				
		•Cont. Shell Exterior Surface				
		•Supp. Chamber Shell Int. Surf.	SS			
		•Supp. Chamber Shell Ext. Surf.			Continued on next page	Continued on next page

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Atmospheric Corrosion	Loss of material	Mark III Steel Containments		Same as Mark I Cont.	Same as Mark I Steel Containment	Same as Mark I Steel Containment
		•Basemat Liner	SS			
		Mark I Concrete Containments				
		•Vent Lines	CS			
		•Vent Headers				
		•Vent Line Bellows	SS			
		•Vent System Supports	CS			
		•Drywell Head				
		Mark II Concrete Containments				
		•Drywell Head				
Atmospheric Corrosion	Loss of material	Mark I Steel Containments		G-5, G-10, G-12, G-19, S-6, S-18, S-37, S-85	The Examination Categories E-A, E-P, & E-C of ASME Sect. XI, Subsect. IWE[11] are required to be performed in conjunction with 10CFR50, Appendix J, Type A leak rate test.[12]	Appendix J of 10CFR50 requires general inspection of containment & component surfaces prior to Type A leak rate test. If there is any evidence of degradation, Type A tests shall not be formed until corrective action is taken. Exam. Category E-P of ASME Sect. IX, Subsect. IWE provides VT-3 examination on accessible containment pressure boundary & E-A provides VT-3 exam. for the containment shell welds. Exam. E-C provides VT-1 inspection on surface areas likely to experience accelerated corrosion. If the areas are found to be defective, volumetric examination is required.
		•Drywell Exterior Surface	CS			
		•ECCS Suction Header*				
		•Ocean Plants with Uncoated CS Component Surfaces				
		•Uncoated Submerged CS Components				
		Mark II Steel Containments				
		•Drywell Exterior Surface				
		•Ocean Plants† with Uncoated CS Surfaces				
		•Uncoated Submerged CS Components				
		•Suppression Chamber Exterior Surface				

* Components submerged in water are treated as having no corrosion allowance.
† A plant located within 1,000 feet distance from the ocean, uncoated CS components are treated as having no corrosion allowance.

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion of Embedded Steel	Cracking, spalling, loss of bond, & loss of material	Mark I Concrete Containments •Drywell Concrete Reinforcing Steel •Torus Concrete Reinforcing Steel Mark II Concrete Containments •Containment Concrete Reinforcing Steel Mark III Concrete Containments •Containment Dome Reinforcing Steel	Reinforcing & embedded CS in concrete	G-1, G-2, G-3, G-4, G-11, G-16, G-17, G-19, G-21, S-74, S-81	For containment concrete components that meet basis requirements, corrosion of embedded steel & rebar is non-significant ARDM.	Degradation due to corrosion of embedded and reinforcing steel is non-significant for concrete structures not exposed to aggressive environment (pH <11.5 or chlorides >500 ppm);[13] or for structures exposed to aggressive environment but have low water-to-cement ratio (0.35-0.45), adequate air entrainment (3-6%), low permeability, and are designed in accordance with ACI 318-63[4] or ASME Sect. III, Division 2.
Corrosion of Embedded Steel	Cracking, spalling, loss of bond, & loss of material	Mark II Concrete Containments •Basemat Reinforcing Steel Mark III Concrete Containments •Cont. Wall Below Grade Reinforcing Steel •Basemat Reinforcing Steel Mark III Steel Containments •Basemat Reinforcing Steel	Reinforcing & embedded CS in concrete	Same as above	Further evaluation for management of inaccessible areas is to be justified on a plant-specific basis.[§]	For inaccessible areas plant-specific evaluation is required.

§ See also NUMARC/NRC agreement concerning Corrosion of Embedded Steel page B-141 (Table B9) and comment S-42, page B-36 (Table B3)

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Local Corrosion	Loss of material	Common Components		G-5, G-10, G-12, S-6, S-10, S-15, S-44, S-47, S-80, S-85, S-88	Periodically examined by the Exam. Category E-C under the provisions of IWE-1240 of ASME Sect. XI, Subsect. IWE.[11]	IWE-1240 of ASME Sect. XI, Subsect. IWE provides for the identification of accessible surface areas likely to experience is accelerated corrosion. These areas are included in the inspection plan, subject to VT-1 visual examination and ultrasonic thickness measurements. If abnormal conditions are identified, the area should be repaired, replaced, or justified by engineering evaluation. Reexamination required for 100% of the suspect area at every 10 yrs of inspection interval.
		•Dissimilar Metal Welds*	CS welded with SS			
		Mark I Steel Containments				
		•Torus Interior Surface at Waterline	CS			
		•Downcomers and Bracing				
		•Drywell Exterior Shell with Compressible Material	CS, polyurethane			
		Mark II Steel Containments				
		•Suppression Chamber Interior Surface at Waterline	CS			
		•Downcomer Pipes & Bracing				
		•Drywell Exterior Shell with Compressible Material	CS, polyurethane			
Local Corrosion	Loss of material	Mark I Steel Containment		G-12, G-19, G-20, S-20, S-87, S-97	A plant-specific aging program is required to manage the local corrosion of these inaccessible and/or embedded carbon steel containment components.	The evaluation for management of inaccessible areas is to be justified on a plant-specific basis.
		•Embedded Shell Region	CS			
		•Drywell Support Skirt				
		•Sand Pocket Region				
		Mark II Steel Containment				
		•Embedded Shell Region				
		•Support Skirt				
		•Sand Pocket Region				
		•Region Shielded by Diaphragm Floor				
		Mark III Steel Containment				
		•Embedded Shell Region				

* Galvanic corrosion potential areas: Vent line or penetration bellow locations. The bellows are SS & the rest of the pipe lines or pipe sleeves are CS.

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Local Corrosion	Loss of material	Common Components	CS	G-1, G-2, G-3, G-4, G-11, G-16, G-17, G-19, G-21, S-74, S-81	For concrete containment liner and anchors that meet the basis requirements, liner plate corrosion is non-significant ARDM.	The liners are typically constructed of mild carbon steel, with a coating applied to the surface to protect it from corrosion effects. Corrosion of the liner plate is mitigated by protective coatings on the interior surface, and the alkaline environment on the exterior surface. SS is corrosion resistant.
		•Penetration Sleeves				
		Mark I Concrete Containments				
		•Drywell Liner Interior Surface				
		•Drywell Liner Exterior Surface				
		•Torus Liner Interior Surface				
		•Torus Liner Exterior Surface				
		•Liner Anchors				
		Mark II Concrete Containments				
		•Drywell Liner Interior Surface				
		•Drywell Liner Exterior Surface				
		•Torus Liner Interior Surface	CS or SS			
		•Basemat Liner	CS			
		•Liner Anchors				
		Mark III Concrete Containments	SS in pool region, CS rest			
		•Containment Liner Interior Surface				
		•Containment Liner Exterior Surface				
		•Suppression Chamber Liner Interior Surface or Cladding Surface	SS			
		•Basemat Liner	SS			
		•Liner Anchors	CS			
		•Suppression Chamber Liner Exterior Surface	SS			

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Local Corrosion	Loss of material	Common Components •Dissimilar Metal Welds Mark I Concrete Containments •Torus Liner Interior Surface at Waterline •Downcomers & bracing Mark II Concrete Containments •Suppression Chamber Liner Interior Surface at Waterline •Downcomers & bracing	CS, SS CS	G-10, G-12, G-19, S-6, S-15, S-85	Periodical examination under the provisions of IWE-1240 (Exam. Category E-C) of ASME Sect. XI, Subsect. IWE.[11]	Underwater surfaces are considered as accessible by the rules of IWE-1240 of ASME Sect. XI; Subsect. IWE requires the identification, assessment, & mitigation of local corrosion of containment liner components. Visual VT-1 and ultrasonic thickness measurements followed by engineering evaluation are required.
Local Corrosion	Loss of material	Mark II Concrete Containment •Region Shielded by Diaphragm Floor	CS	G-5, G-12, G-19	Plant-specific management program is required for inaccessible areas.	Evaluation is to be justified on a plant-specific basis.
Local Corrosion	Loss of material	Mark II Concrete Containment •Prestressed Tendons	CS & Concrete	G-10, G-13, G-14, G-19, S-2, S-6, S-16, S-17, S-75, S-78, S-85	Prestressing tendons and tendon anchorage hardware should be examined in accordance with the provisions of RG 1.35.[14]	Corrosion of prestressed tendons can be identified and managed by established programs for periodic visual examination of the tendon anchor heads as well as frequent examination of the corrosion protection medium to ensure absence of corrosive fluids, as prescribed in RG 1.35.

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Elevated Temp-erature	Loss of strength & modulus	Common Components		G-1, G-2, G-3, G-4, G-11, G-16 to G-18, G-21, S-1, S-65	Non-significant ARDM if it meets the basis requirements.	Degradation from exposure to elevated temperatures is non-significant for containment components maintained at operating temperatures <66°C (150°F) and local area temperatures <93°C (200°F)[15] or for structures that operate above these limits, plant-specific justification is provided in accordance with ACI 349-85,[16] or ASME Sect. III, Division 2, Class CC.[2] Degradation from exposure to elevated temperatures is non-significant for reinforcing steel (rebar) maintained at temperatures <316°C (600°F).[15]
		•Penetration Sleeves	CS			
		Mark I Concrete Containments	CS, concrete			
		•Drywell & Torus Concrete				
		•Drywell & Torus Reinforcing Steel				
		Mark II Concrete Containments				
		•Containment Concrete				
		•Containment Concrete Reinforcing Steel				
		•Concrete Fill in Annulus				
		•Concrete Basemat and Reinforcing Steel				
		Mark III Concrete Containments				
		•Containment Wall Concrete Above Grade & Dome				
		•Containment Wall Concrete Below Grade				
		•Containment Wall Concrete & Dome Reinforcing Steel				
		•Concrete Basemat & Reinforcing Steel				
		Mark III Steel Containments				
		•Concrete Basemat				
		•Concrete Fill in Annulus				

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Irradiation	Loss of fracture toughness for steel & loss of strength & modulus for concrete	All Components of Mark I, II, and III Steel and Concrete Containments	Concrete, CS, SS	G-1, G-2, G-3, G-4, G-11, G-16, G-17, G-18, G-21, S-72	Non-significant ARDM.	The neutron fluence levels & maximum integrated gamma doses incurred by containment components, including rebars & prestressed tendons for both the current and license renewal period do not exceed the level at which measurable degradation occurs. (4 x 10^{16} n/cm^2 for prestressed tendons; 2 x 10^{17} n/cm^2 for all components made of CS, SS including rebar, linear plate; 1 x 10^{19} n/cm^2 neutron radiation & 1 x 10^{10} rads gamma radiation for concrete).[17,18]

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative Fatigue Damage	Common Components •Penetration Sleeves Mark I Steel Containment •Vent Header† •Downcomers & Bracing Mark II Steel Containment •Unbraced Downcomers Mark I and Mark II Concrete Containment •Vent Header* •Downcomers & Bracing	CS	S-5 S-7, S-8, S-51, S-52A, S-53, S-56, S-67, S-68, S-70, S-90, S-91 Open issue G-6, G-15, S-50, S-52B, S-91	Unresolved issue *NUMARC proposal*: Perform ASME Code Sect. III fatigue reanalysis to ensure that the fatigue usage factors can be maintained <1 throughout the license renewal term; or ASME Sect. XI, Subsect. IWE, inspection to ensure that component integrity is maintained throughout the license renewal term, including the continued service of a component with an otherwise rejectable flaw, as justified by an engineering evaluation. *NRC proposal*: Damage of bellows by low-cycle fatigue is credible, especially when environmentally assisted, & should be included in the management program. Environment-accelerated fatigue due to periodic steam & condensate & environmental effects on crack initiation & growth should be considered. Additional information is needed on the basis of the determination of the maximum expected load cycles for 60 yrs of operation.	*NUMARC basis*: Existing component fatigue analyses may be used to prorate the calculated usage factors for the license renewal period. For components having no fatigue design basis, ASME Code Sect. III provides guidance for supplemental plant-unique fatigue assessment. Exam. Category E-B provides a VT-1 examination for containment penetration welds, including penetration sleeve.

* At vent header and downcomer intersection.

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cum-ulative Fatigue Damage	All Other Components of All Containments	CS, SS, concrete	G-1 to G-4, G-11, G-16, G-17, G-18, S-5, S-7, S-51, S-52A, S-53, S-56, S-67, S-68 Open issue G-6, G-15, S-50, S-52B, S-91	Unresolved issue *NUMARC proposal*: Non-significant *NRC proposal*: Same as previous page	*NUMARC basis*: Containment concrete components subjected to repeated load are designed in accordance with ACI 318 or an equivalent code which limits the maximum design stress level to < 50% of static strength in working stress design and 71% in ultimate strength design; concrete structures can resist $>10^6$ cycles of loading in this stress range.[19] Containment steel components subjected to repeated loading are designed in accordance with ASME Code or its equivalent, which limits the stress ranges in steel components and connections.

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Mechanical Wear	Lockup	Common Components	CS	G-10, G-11, G-16 to G-19, G-21, S-49, S-55, S-85	Conduct inspection and mitigation of mechanical wear in accordance with the provisions of ASME Sect. XI, Subsect. IWE[11] & IWF,[20] as applicable, to ensure that component integrity is maintained throughout the license renewal term.	The pressure retaining components such as airlock, equipment hatches, CRD hatch and drywell head are required to be examined once every inspection interval (10 yrs) according to ASME XI, Subsect. IWE[11] Exam. Category E-G specifies a VT-1 examination for bolted connections. Category E-D specifies a VT-3 examination of the seals and gaskets required for leaktight integrity. The supporting components such as downcomer bracing, vent system supports and seismic restraints are required to be examined by VT-3 examination for the identification of wear, corrosion, loose parts, deformation and other degradation, in accordance with ASME Sect. XI, Subsect. IWF.[20]
		•Personnel Airlock				
		•Equipment Hatches				
		•CRD Hatch				
		Mark I Steel Containments	CS			
		•Drywell Head				
		•Downcomers & Bracing				
		•Vent System Supports	CS			
		•Torus Support Columns/Saddles	Saddle: CS lubron plate: graphite			
		•Torus Seismic Restraints	CS			
		Mark II Steel Containments				
		•Drywell Head				
		•Downcomer Pipes & Bracing				
		Mark I Concrete Containments				
		•Drywell Head				
		•Downcomers & Bracing				
		•Vent System Supports				
		Mark II Concrete Containments				
		•Drywell Head				
		•Downcomer Pipes & Bracing				

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Strain Aging	Loss of fracture toughness	Common Components	CS	G-1 to G-4, G-11, G-16 to G-18, G-21	Strain aging is a non-significant ARDM for containment steel components that meet the basis requirements.	Strain aging is a non-significant ARDM for containment steel components having service stress in the elastic region and without severely cold working in the forming process.[21] If severe cold working was used in the forming process, but the plates are normalized, or stress relieved or both after forming with minimal (<5%) subsequent cold working, then strain aging is not significant.
		•All other except dissimilar metal welds				
		Mark I Steel Containments				
		•All components except: Vent Header, Downcomers & Bracing, Vent System Supports, Torus Seismic Restraint, & ECCS Suction Header				
		Mark II Steel Containments				
		•All components except downcomer pipes and bracing				
		Mark III Steel Containments				
		•Containment Shell Interior Surface				
		•Containment Shell Exterior Surface				
		•Suppression Chamber Shell Interior Surface				
		•Suppression Chamber Shell Exterior Surface				
		•Embedded Shell Region				
		Mark I Concrete Containments				
		•Vent Lines				
		•Vent Line Bellows				
		•Drywell Head				
		Mark II Concrete Containments				
		•Drywell Head				

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Loss of Prestress	Reduction of design margin	Mark II Concrete Containments •Containment Concrete •Prestress Tendons	Concrete, CS	G-10, G-11, G-13, G-16 to G-19, G-21, S-2, S-17	Periodic monitoring of loss of prestress in accordance with the tendon lift-off test provisions of RG 1.35[14] to ensure satisfactory comparison with predictions of prestressing loss for the license renewal term.	Periodic monitoring of loss of prestress in accordance with the tendon lift-off test of RG 1.35 is effective.
Settlement	Cracking, distortion, increase in component stress level	Mark II Concrete Containments •Basemat (bearing on soil, or piles) Mark III Steel Containments •Basemat (bearing on soil, or piles) Mark III Concrete Containments •Basemat (bearing on soil, or piles)	CS, concrete	G-1 to G-4, G-11, G-16 to G-18, G-21, S-3, S-77, S-93	Settlement is non-significant ARDM for containments bearing on bedrock. For BWR containments bearing on soil or piles, a settlement monitoring program is required to ensure that the differential settlement does not exceed the design criteria for the containment throughout the license renewal term.§	Long-term settlement due to variations of the water table and consolidation of clay soils can be determined from the current plant settlement monitoring program. Early indications of potentially significant settlement can be detected using the widely accepted methods.[22] When settlement approaches the acceptance criteria, reevaluation of the containment is necessary.

§ See also NUMARC/NRC agreement concerning Settlement page B-154 (Table B9) and comment S-63, page B-42 (Table B3).

Table B4. Brief summary of technical information and NUMARC/NRC agreements from BWR containments license renewal industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials[a]	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Stress Corrosion Cracking (SCC)	Crack initiation & growth	Common Components •Penetration Sleeves •Penetration Bellows Mark I Steel Containments •Vent Line Bellows Mark I Concrete Containments •Vent Line Bellows	 CS SS	G-1 to G-4, G-16 to G-18, G-21, S-19	SCC is a non-significant ARDM for the components that meet the basis requirements.	For austenitic SS containment components, SCC is not a significant ARDM if they are only exposed to the containment or reactor building environment or their normal operational stress levels are less than materials yield strength or fracture mechanics analysis has established that cracks do not propagate. Additionally, SCC is not significant for high strength bolts if material yield strength is <1034 MPa (<150 ksi).
SCC	Crack initiation & growth	Mark III Steel Containments •Suppression Chamber Shell Interior Surface or Cladding Surface Mark II Concrete Containments •Suppression Chamber Interior SS Liner Mark III Concrete Containments •Suppression Chamber Interior SS Liners	SS	G-5, G-10, G-21, S-19, S-59, S-85	Detection of liner leakage through 10CFR50, Appendix J integrated leak rate test to ensure that linear integrity is maintained throughout the license renewal term.	Cracks originating on the liner surface due to SCC must first propagate through the liner thickness in order to affect the leakage integrity. Any leakage due to through-wall cracks would be detected by periodic 10CFR50 Appendix J, leak rate test & remains within the limits established by plant's technical specifications. If any detected leakage exceeds the acceptance criteria of the plant's technical specifications or the ASME Sect. XI, IWE, mitigation via repair or replacement & retesting.

[a] Designation for steel materials are as follows: SS = stainless steel, CS = carbon steel.
[b] The following comments were not included in the table because they deal with clarification, scope, or modification of contents of the IR: G-8, G-9, S-4, S-22 to 24, S-27 to 29, S-31, S-36, S-39 to 41, S-45, S-57, S-60, S-62, S-66, S-69, S-79, S-82 to 84, and S-92.

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1. Specification for Air Entraining Admixture for Concrete, ASTM C260-77, American Society for Testing and Materials, Philadelphia, PA, 1977.
2. Code for Concrete Reactor Vessels and Containments, ASME Boiler and Pressure Vessel Code, Section III, Division 2, American Society of Mechanical Engineers, New York, NY.
3. "Guide to Durable Concrete," ACI 201.2R-77, American Concrete Institute.
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LIST OF BWR CONTAINMENT COMPONENTS:

MARK I STEEL CONTAINMENT
Drywell Interior Surface
Drywell Exterior Surface
Drywell Head
Embedded Shell Region
Drywell Support Skirt
Sand Pocket Region
Torus Interior Surface
Torus Interior Surface at Waterline
Torus Exterior Surface
Torus Ring Girder
Vent Lines
Vent Line Bellows
Vent Header
Downcomers and Bracing
Vent System Supports
Torus Seismic Restraints
Torus Support Columns/Saddles
ECCS Suction Header
Ocean Plant with Uncoated CS Surfaces
Uncoated Submerged CS Surfaces

MARK II STEEL CONTAINMENTS
Drywell Interior Surface
Drywell Exterior Surface
Drywell Head
Suppr. Chamber Exterior Surface
Suppr. Chamber Interior Surface
Suppr. Chamber Interior Surface at Waterline
Region Shielded by Diaphragm Floor
Embedded Shell Region
Sand Pocket Region
Support Skirt
Downcomer Pipes and Bracing
Ocean Plant with Uncoated CS Surfaces
Uncoated Submerged CS Surfaces

MARK I CONCRETE CONTAINMENT
Drywell Liner Interior Surface
Drywell Liner Exterior Surface
Torus Liner Interior Surface
Torus Liner Interior Surface at Waterline
Torus Liner Exterior Surface
Liner Anchors
Drywell Concrete
Torus Concrete
Drywell Concrete Reinforcing Steel
Torus Concrete Reinforcing Steel
Vent Lines
Vent Line Bellows
Vent Headers
Downcomers and Bracing
Vent System Supports
Drywell Head

MARK II CONCRETE CONTAINMENTS
Drywell Liner Interior Surface
Drywell Liner Exterior Surface
Suppr. Chamber Liner Interior Surface
Suppr. Chamber Liner Interior Surface at Waterline
Suppr. Chamber Liner Exterior Surface
Liner Anchors
Liner Region Shielded by Diaphragm Floor
Containment Concrete
Concrete Containment Reinforcing Steel
Drywell Head
Downcomer Pipes and Bracing
Concrete Basemat
Basemat Liner
Basemat Reinforcing Steel
Prestressing Tendons and Ducts

MARK III STEEL CONTAINMENTS
Containment Shell Interior Surface
Containment Shell Exterior Surface
Suppr. Chamber Shell Interior Surface
Suppr. Chamber Shell Exterior Surface
Basemat Liner
Liner Anchors
Concrete Basemat
Concrete Fill in Annulus
Embedded Shell Region

MARK III CONCRETE CONTAINMENTS
Containment Liner Interior Surface
Containment Liner Exterior Surface
Suppr. Chamber Liner or Cladding Interior Surface
Suppr. Chamber Liner Exterior Surface
Concrete Containment Wall Above Grade
Concrete Containment Wall Below Grade
Concrete Dome
Basemat Liner
Concrete Basemat
Liner Anchors
Containment Wall Reinforcing Steel
Dome Reinforcing Steel
Basemat Reinforcing Steel

COMMON COMPONENTS
Penetration Sleeves
Dissimilar Metal Welds
Penetration Bellows
Personnel Airlock
Equipment Hatches
CRD Hatch

Table B5. Brief summary of technical information and NUMARC/NRC agreements from PWR reactor coolant system industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
SCC	Crack initiation & growth	Reactor Coolant Pump		G-1, G-2, G-3, G-7, S-V-36, S-V-37 Open issues S-V-38, S1 S-1	Unresolved issue *NUMARC proposal*: Non-significant ARDM *NRC proposal*: IGSCC can occur under the operating conditions (water chemistry) during shutdown because oxygen is introduced to primary coolant during cool down to control CRUD-bursts, & coolant is exposed to air during many shutdowns. The potential of cracking in cladding remote from welds should be addressed. SS cladding may have regions of low delta ferrite that have been sensitized during PWHT & thus susceptible to IGSCC; ASME Sect. XI requires inspection of weld & weld regions.	*NUMARC basis*: Components fabricated of CASS or CS internally clad with SS (>5% ferrite) have reduced susceptibility to SCC (see S1 S-1),[1] underlying CS base metal is not susceptible to decohesion;[2] & concentrations of oxygen, halogens, & sulfates are monitored & controlled in the coolant (see S-V-38);[3] and/or not subjected to corrosive environment.
		Casing†	CASS			
		Cover	CASS			
		Casing Flange	CASS			
		Cover Flange	CASS			
		Nozzles	SS			
		Pressurizer				
		Shell/Heads	CS			
		Spray Line Nozzle	CS			
		Valve Nozzle	CS			
		Manway	CS			
		Instrument Nozzle	CS			
		Surge Line Nozzle	CS, CASS			
		Support Skirt	CS			
		Safety & Relief Valves				
		Valve Body	SS, CASS			
		Bonnet	SS, CASS			
		Body Flange	SS, CASS			
		Bonnet Flange	SS, CASS			
		Nozzles	SS			
		Seats & Disks	Stellite, SS			
		Piping & Fittings				
		Cold-Leg	CS, SS, CASS			
		Hot-Leg	CS, SS, CASS			
		Surge Line	SS, CASS			
		Spray Line	SS			
		Nozzles & Safe Ends	CS, SS, CASS			
		Auxiliary Piping				
		DHRS	SS			
		CFS	SS			
		Fittings, Nozzles, & Safe Ends	SS, CASS			
		Integral Support	CS, SS			

† Only CASS pump casings are included in this review, which excludes the CE-KSB Type F pump casings that are low alloy internally clad with SS.

Table B5. Brief summary of technical information and NUMARC/NRC agreements from PWR reactor coolant system industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
SCC	Crack initiation & growth	Pressurizer		S-V-39	Evaluate Ni-alloy applications in pressurizers; ASME Sect. XI Subsect. IWB[4] requirements of VT-2 supplemented by augmented inspection programs based on NRC Inf. notice No. 90-10.[5]	Review of use of Ni-alloys & ASME Sect. XI, Subsect. IWB[4] exam. category B-E supplemented by augmented inspection programs based on NRC Inf. notice No. 90-10[5] are current & effective programs for detection, sizing, evaluation, & remediation.
		Instrument Nozzle	Ni-Alloy			
		Heater Sleeves	Ni-Alloy			
SCC	Crack initiation & growth	Pressurizer		None	Surface &/or volumetric exam. of ASME Sect. XI, Subsect. IWB.[4]	ASME Sect. XI, Subsect. IWB[4] exam. category B-F, are current & effective programs for detection, sizing, evaluation, & remediation.
		Safe Ends	SS			
SCC	Crack initiation & growth	Reactor Coolant Pump		None	ASME Sect. XI, Subsect. IWB, includes VT-1 of flange, nuts, bushing, & washer surfaces, & volumetric exam. of bolts & studs.[4]	ASME Sect. XI, Subsect. IWB[4] exam. category B-G-1 & -2, are current & effective programs for detection, sizing, evaluation, & remediation.
		Closure Bolting	HSLAS†			
		Pressurizer				
		Manway Bolting	HSLAS†			
		Safety & Relief Valves				
		Closure Bolting	HSLAS†			

† High-strength low-alloy steel.

Table B5. Brief summary of technical information and NUMARC/NRC agreements from PWR reactor coolant system industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Thermal Embrittlement	Loss of fracture toughness	Reactor Coolant Pump		G-4, S-I-12, S-III-32, S-III-34 Open issues S-III-28, S-III-33, S1 S-2	Unresolved issue. *NUMARC proposal*: ASME Sect. XI, Subsect. IWB requires visual VT-3 exam. & analytical evaluation procedures for flaw tolerance. *NRC proposal*: Ferrite criteria is inadequate tool for screening & VT-3 as referenced in Code Case N-481 is not intended for detection of cracks. Fracture toughness may be estimated based on NUREG/CR-4513 Rev. 1.[6]	*NUMARC basis*: Components with ferrite content >20% for all centrifugally cast & static-cast CF-3 & CF-8, or >14% for static -cast CF-3M & CF-8M (S-III-28), ASME Sect. XI, Subsect. IWB,[4] exam categories B-L-1 & -2, B-M-1 & -2, & B-P with flaw acceptance standard IWB-3500, and UT procedures of supplements to mandatory Appendix VIII; & flaw tolerance evaluation of ASME Code Case N-481 (S-III-33 & S1 S-2); are current & effective programs for detection & evaluation-repair-replacement.
		Casing	CASS			
		Cover	CASS			
		Casing Flange	CASS			
		Cover Flange	CASS			
		Pressurizer				
		Surge Line Nozzle	CASS			
		Safety & Relief Valves				
		Valve Body	CASS			
		Bonnet	CASS			
		Body Flange	CASS			
		Bonnet Flange	CASS			
		Piping & Fittings				
		Cold-Leg	CASS			
		Hot-Leg	CASS			
		Surge Line	CASS			
		Nozzles & Safe Ends	CASS			
		Auxiliary Piping				
		Fittings, Nozzles, & Safe Ends	CASS			

Table B5. Brief summary of technical information and NUMARC/NRC agreements from PWR reactor coolant system industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Thermal Embrittlement	Loss of fracture toughness	Reactor Coolant Pump		G-1, G-2, G-3, G-7, S-III-29, S-III-30, S-III-31	Non-significant	Proper material selection and relatively low PWR operating temperatures.
		Nozzles	SS			
		Closure Bolting	HSLAS			
		Pressurizer				
		Shell/Heads	CS			
		Spray Line Nozzle	CS			
		Valve Nozzle	CS			
		Manway	CS			
		Instrument Nozzle	CS, Ni-Alloy			
		Surge Line Nozzle	CS			
		Heater Sleeves	Ni-Alloy			
		Safe Ends	SS			
		Support Skirt	CS			
		Manway Bolting	HSLAS			
		Safety & Relief Valves				
		Valve Body	SS			
		Bonnet	SS			
		Body Flange	SS			
		Bonnet Flange	SS			
		Nozzles	SS			
		Seats & Disks	Stellite, SS			
		Closure Bolting	HSLAS			
		Piping & Fittings				
		Cold-Leg	CS, SS			
		Hot-Leg	CS, SS			
		Surge Line	SS			
		Spray Line	SS			
		Nozzles & Safe Ends	CS, SS			
		Auxiliary Piping				
		DHRS	SS			
		CFS	SS			
		Fittings, Nozzles, & Safe Ends	SS			
		Integral Support	CS, SS			

Table B5. Brief summary of technical information and NUMARC/NRC agreements from PWR reactor coolant system industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion	Loss of material	Reactor Coolant Pump		G-1, G-2, G-3, G-7, G-9 c, S-VI-41, S-VI-42, S-VI-44	Non-significant	Components are fabricated of or are internally clad with SS or Ni-alloys; hydrogen overpressure provides protection against crevice corrosion; and/ or components not in contact with primary coolant.
		Nozzles	SS			
		Pressurizer				
		Shell/Heads	CS			
		Spray Line Nozzle	CS			
		Valve Nozzle	CS			
		Manway	CS			
		Instrument Nozzle	CS, Ni-Alloy			
		Surge Line Nozzle	CS, CASS			
		Heater Sleeves	Ni-Alloy			
		Safe Ends	SS			
		Support Skirt	CS			
		Safety & Relief Valves				
		Nozzles	SS			
		Seats & Disks	Stellite, SS			
		Piping & Fittings				
		Cold-Leg	CS, SS, CASS			
		Hot-Leg	CS, SS, CASS			
		Surge Line	SS, CASS			
		Spray Line	SS			
		Nozzles & Safe Ends	CS, SS, CASS			
		Integral Support	CS, SS			
		Auxiliary Piping				
		DHRS	SS			
		CFS	SS			
		Fittings, Nozzles, & Safe Ends	SS, CASS			

Table B5. Brief summary of technical information and NUMARC/NRC agreements from PWR reactor coolant system industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion/ Boric Acid Wastage of external surfaces	Loss of material	Reactor Coolant Pump		S-VI-43	Program developed & implemented based on Generic Letter 88-05.[7]	Recommendations of Generic Letter 88-05[7] are current & effective program to monitor & control primary coolant leakage.
		Casing	CASS			
		Cover	CASS			
		Casing Flange	CASS			
		Cover Flange	CASS			
		Closure Bolting	HSLAS			
		Pressurizer				
		Top head	CS			
		Manway Bolting	HSLAS			
		Safety & Relief Valves				
		Valve Body	SS, CASS			
		Bonnet	SS, CASS			
		Body Flange	SS, CASS			
		Bonnet Flange	SS, CASS			
		Closure Bolting	HSLAS			

Table B5. Brief summary of technical information and NUMARC/NRC agreements from PWR reactor coolant system industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Erosion/ Corrosion (E/C)	Wall thinning	Safety & Relief Valves		G-1, G-2, G-3, G-7, S-VI-40	Non-significant	High-alloy steels, nickel-base alloys, & SSs are resistant to E/C, and/or relatively low flow & pH control in PWR environments.
		Valve Body & Body Flange	SS, CASS			
		Bonnet & Bonnet Flange	SS, CASS			
		Nozzles	SS			
		Seats & Disks	Stellite, SS			
		Closure Bolting	HSLAS			
		Piping & Fittings				
		Cold-Leg	CS, SS, CASS			
		Hot-Leg	CS, SS, CASS			
		Surge Line	SS, CASS			
		Spray Line	SS			
		Nozzles & Safe Ends	CS, SS, CASS			
		Auxiliary Piping				
		DHRS	SS			
		CFS	SS			
		Fittings, Nozzles, & Safe Ends	SS, CASS			
		Integral Support	CS, SS			
		Reactor Coolant Pump				
		Casing & Casing Flange	CASS			
		Cover & Cover Flange	CASS			
		Nozzles	SS			
		Closure Bolting	HSLAS			
		Pressurizer				
		Shell/Heads	CS			
		Spray Line Nozzle	CS			
		Valve Nozzle	CS			
		Manway	CS			
		Instrument Nozzle	CS, Ni-Alloy			
		Surge Line Nozzle	CS, CASS			
		Heater Sleeves	Ni-Alloy			
		Safe Ends	SS			
		Support Skirt	CS			
		Manway Bolting	HSLAS			

Table B5. Brief summary of technical information and NUMARC/NRC agreements from PWR reactor coolant system industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron Irradiation Embrittle-ment	Loss of fracture toughness	Reactor Coolant Pump		G-1, G-2, G-3, G-7 S-IV-35	Non-significant	Neutron irradiation embrittle-ment is non-significant be-cause of low fluence level.[9,10]
		Casing & Casing Flange	CASS			
		Cover & Cover Flange	CASS			
		Nozzles	SS			
		Closure Bolting	HSLAS			
		Pressurizer				
		Shell/Heads	CS			
		Spray Line Nozzle	CS			
		Valve Nozzle	CS			
		Manway	CS			
		Instrument Nozzle	CS, Ni-Alloy			
		Surge Line Nozzle	CS, CASS			
		Heater Sleeves	Ni-Alloy			
		Safe Ends	SS			
		Support Skirt	CS			
		Manway Bolting	HSLAS			
		Safety & Relief Valves				
		Valve Body & Body Flange	SS, CASS			
		Bonnet & Bonnet Flange	SS, CASS			
		Nozzles	SS			
		Seats & Disks	Stellite, SS			
		Closure Bolting	HSLAS			
		Piping & Fittings				
		Cold-Leg	CS, SS, CASS			
		Hot-Leg	CS, SS, CASS			
		Surge Line	SS, CASS			
		Spray Line	SS			
		Nozzles & Safe Ends	CS, SS, CASS			
		Auxiliary Piping				
		DHRS	SS			
		CFS	SS			
		Fittings, Nozzles, & Safe Ends	SS, CASS			
		Integral Support	CS, SS			

Table B5. Brief summary of technical information and NUMARC/NRC agreements from PWR reactor coolant system industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Creep	Change in dimension	Reactor Coolant Pump		G-1, G-2, G-3, G-7	Non-significant	Operating temperatures are <371°C (<700°F) for CS, <538°C (<1000°F) for SS.
		Casing & Casing Flange	CASS			
		Cover & Cover Flange	CASS			
		Nozzles	SS			
		Closure Bolting	HSLAS			
		Pressurizer				
		Shell/Heads	CS			
		Spray Line Nozzle	CS			
		Valve Nozzle	CS			
		Manway	CS			
		Instrument Nozzle	CS, Ni-Alloy			
		Surge Line Nozzle	CS, CASS			
		Heater Sleeves	Ni-Alloy			
		Safe Ends	SS			
		Support Skirt	CS			
		Manway Bolting	HSLAS			
		Safety & Relief Valves				
		Valve Body & Body Flange	SS, CASS			
		Bonnet & Bonnet Flange	SS, CASS			
		Nozzles	SS			
		Seats & Disks	Stellite, SS			
		Closure Bolting	HSLAS			
		Piping & Fittings				
		Cold-Leg	CS, SS, CASS			
		Hot-Leg	CS, SS, CASS			
		Surge Line	SS, CASS			
		Spray Line	SS			
		Nozzles & Safe Ends	CS, SS, CASS			
		Auxiliary Piping				
		DHRS	SS			
		CFS	SS			
		Fittings, Nozzles, & Safe Ends	SS, CASS			
		Integral Support	CS, SS			

Table B5. Brief summary of technical information and NUMARC/NRC agreements from PWR reactor coolant system industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Stress Relaxation	Loss of preload	Reactor Coolant Pump		G-1, G-2, G-3, G-7	Non-significant	These components do not depend on preload for functionality.
		Casing & Casing Flange	CASS			
		Cover & Cover Flange	CASS			
		Nozzles	SS			
		Pressurizer				
		Shell/Heads	CS			
		Spray Line Nozzle	CS			
		Valve Nozzle	CS			
		Manway	CS			
		Instrument Nozzle	CS, Ni-Alloy			
		Surge Line Nozzle	CS, CASS			
		Heater Sleeves	Ni-Alloy			
		Safe Ends	SS			
		Support Skirt	CS			
		Safety & Relief Valves				
		Valve Body & Body Flange	SS, CASS			
		Bonnet & Bonnet Flange	SS, CASS			
		Nozzles	SS			
		Seats & Disks	Stellite, SS			
		Piping & Fittings				
		All Components	CS, SS, CASS			
		Auxiliary Piping				
		All Components	SS, CASS			
		Integral Support	CS, SS			
Stress Relaxation	Loss of preload	Reactor Coolant Pump		S-II-17	ASME Sect. XI Table IWB-2500-1 includes VT-1 of nuts bushings, & washer surfaces, & volumetric exam. of bolts & studs;[4] corrective measure IWA-5250; acceptance criteria IWA-3142.	ASME Sect. XI, Subsect. IWB, exam. categories B-G-1 & -2, & B-P for leakage,[4] & corrective measure IWA-5250, acceptance criteria IWA-3142; are current & effective for detection & correction of preload.
		Closure Bolting	HSLAS			
		Pressurizer				
		Manway Bolting	HSLAS			
		Safety & Relief Valves				
		Closure Bolting	HSLAS			

Table B5. Brief summary of technical information and NUMARC/NRC agreements from PWR reactor coolant system industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	Reactor Coolant Pump		G-4 G-9 b,d S-I-14 S-I-15 S-II-16 through S-II-27 Open issues G-8 G-9 a G-10 G-11 S-I-13	Unresolved issue *NUMARC proposal:* Verification of continued adequacy of the fatigue design basis through reanalysis of fatigue usage factor, actual plant transients, & partial cycle counting; assurance provided by ASME Sect. XI ISI and system testing programs, including commitments to enhance or augment inspection as a result of plant experience or regulatory action; and defect repair & component replacement. *NRC proposal:* Fatigue issues are unresolved until an agreement is reached in the ongoing discussions on fatigue evaluation for license renewal between NUMARC and staff (G-8, G-9a, G-10, & G-11). Potentially significant fatigue damage due to cold water spraying onto the pressurizer shell should be evaluated (S-I-13).	*NUMARC basis:* ASME Sect. III, Subsect. NB[11] reanalysis of usage factor (see G-9 a & G-11); actual plant transient, cycle monitoring, & partial cycle counting; ASME Sect. XI, Subsect. IWB[4] inspection requirements of IWB-2500-1 (see G-10) & UT procedures of supplements to mandatory Appendix VIII (see G-8); continuation of NRC Bulletin 88-08[12] program for unisolable piping & NRC Bulletin 88-11[13] program for pressurizer surge line thermal stratification in accordance with NSSS-specific operational procedures (see S-I-13); are current & effective programs for detection & evaluation-repair-replacement.
		Casing & Casing Flange	CASS			
		* Cover & Cover Flange	CASS			
		* Nozzles	SS			
		* Closure Bolting	HSLAS			
		Pressurizer				
		* Shell/Heads	CS			
		Spray Line Nozzle	CS			
		Valve Nozzle	CS			
		Manway	CS			
		Instrument Nozzle	CS, Ni-Alloy			
		* Surge Line Nozzle	CS, CASS			
		Heater Sleeves	Ni-Alloy			
		Safe Ends	SS			
		Support Skirt	CS			
		Manway Bolting	HSLAS			
		Safety & Relief Valves				
		* Valve Body & Body Flange	SS, CASS			
		Bonnet & Bonnet Flange	SS, CASS			
		Nozzles	SS			
		Seats & Disks	Stellite, SS			
		* Closure Bolting	HSLAS			
		Piping & Fittings				
		* Cold- & Hot-Leg	CS, SS, CASS			
		* Surge Line	SS, CASS			
		Spray Line	SS			
		* Nozzles & Safe Ends	CS, SS, CASS			
		Auxiliary Piping				
		* DHRS	SS			
		* CFS	SS			
		* Fittings, Nozzles, & Safe Ends	SS, CASS			
		Integral Support	CS, SS			

* These components can have significant fatigue damage (NUMARC proposal).

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Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Wear	Attrition	Reactor Coolant Pump		G-1, G-2, G-3, G-7, S-VII-45	Non-significant	Not subjected to relative motion or does not incorporate clamped joints.
		Casing	CASS			
		Cover	CASS			
		Nozzles	SS			
		Pressurizer				
		Shell/Heads	CS			
		Spray Line Nozzle	CS			
		Valve Nozzle	CS			
		Manway	CS			
		Instrument Nozzle	CS, Ni-Alloy			
		Surge Line Nozzle	CS, CASS			
		Heater Sleeves	Ni-Alloy			
		Safe Ends	SS			
		Support Skirt	CS			
		Manway Bolting	HSLAS			
		Safety & Relief Valves				
		Valve Body	SS, CASS			
		Bonnet	SS, CASS			
		Nozzles	SS			
		Piping & Fittings				
		All Components	CS, SS, CASS			
		Auxiliary Piping				
		All Components	SS, CASS			
		Integral Support	CS, SS			

Table B5. Brief summary of technical information and NUMARC/NRC agreements from PWR reactor coolant system industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Wear	Attrition	Reactor Coolant Pump		None	ASME Sect. XI Table IWB-2500-1[4] includes VT-1 of nuts bushings, & washer surfaces, & volumetric exam. of bolts & studs; & inservice & functional testing of ASME/ANSI[14] OM Part 1 for safety & relief valves & Part 6 for pumps.	ASME Sect. XI, Subsect. IWB,[4] exam. categories B-G-1 & -2, & B-P for system leakage/testing.
		Closure Bolting	HSLAS			
		Casing Flange	CASS			
		Cover Flange	CASS			
		Safety & Relief Valves				
		Body Flange	SS, CASS			
		Bonnet Flange	SS, CASS			
		Seats & Disks	Stellite, SS			
		Closure Bolting	HSLAS			

[a] The following comments were not included in the table because they deal with scope of the IR and, therefore, have no relevance for NUMARC/NRC agreements but would be important for preparing the SRP document: G-5 and S-I-12.

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LIST OF PWR PRIMARY COOLANT SYSTEM COMPONENTS:

PWR Primary Coolant System

Reactor Coolant Pump
- Casing
- Cover
- Casing Flange
- Cover Flange
- Nozzles
- Closure Bolting

Pressurizer
- Shell/Heads
- Spray Line Nozzle
- Valve Nozzle
- Manway
- Instrument Nozzle
- Surge Line Nozzle
- Heater Sleeves
- Safe Ends
- Support Skirt
- Manway Bolting

Safety & Relief Valves
- Valve Body
- Bonnet
- Body Flange
- Bonnet Flange
- Nozzles
- Seats & Disks
- Closure Bolting

Piping & Fittings
- Cold-Leg
- Hot-Leg
- Surge Line
- Spray Line
- Nozzles & Safe Ends

Auxiliary Piping
- DHRS
- CFS
- Fittings, Nozzles, & Safe Ends

Integral Support

Table B6. Brief summary of technical information and NUMARC/NRC agreements from BWR primary coolant pressure boundary industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IGSCC	Crack initiation & growth	Piping & Fittings		G-2, S-III-1, S-III-2	Non-significant	Wrought & cast CS are resistant to sensitization, and/or applied & residual stresses are low, &/ or are not subjected to corrosive environment. Bolting degradation or failure has been addressed in Generic Safety Issue 29.[1]
		MS	CS			
		FW	CS			
		RHR	CS			
		RCIC	CS			
		HPCI	CS			
		LPCI	CS			
		LPCS	CS			
		HPCS	CS			
		Relief & In-Line Valves				
		Valve Body	CS			
		Bonnet	CS			
		Seal Flange	CS			
		Nuts & Bolts	CS			
		Integral Support	CS			
IGSCC	Crack initiation & growth	Relief & In-Line Valves		G-2	Non-significant	Applied & residual stresses are low &/or not subjected to corrosive environment. Bolting degradation or failure has been addressed in Generic Safety Issue 29.[1]
		Seal Flange	SS			
		Nuts & Bolts	SS			
		Recirculation Pump				
		Heat Exchanger	SS			
		Seal Flange	SS			
		Nuts & Bolts	SS			
		Integral Support	SS			

Table B6. Brief summary of technical information and NUMARC/NRC agreements from BWR primary coolant pressure boundary industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IGSCC	Crack initiation & growth	Relief & In-Line Valves		G-1, G-3, S-III-3, Open issue: S-III-4	Unresolved issue: *NUMARC proposal:* CASS materials are resistant to IGSCC if C & ferrite content meet the boundaries of Hughes et al.[7] *NRC proposal:* CASS materials that meet the NUREG-0313, Rev. 2[5] guidelines of ≤0.035% C & ≥7.5% ferrite have reduced susceptibility to IGSCC.	*NUMARC basis:* CASS materials that meet the C & ferrite content criteria of Hughes et al[7] are resistant to IGSCC.
		Valve Body	CASS			
		Bonnet	CASS			
		Recirculation Pump				
		Bowl	CASS			
		Cover	CASS			
IGSCC	Crack initiation & growth	Piping & Fittings		G-1, G-3, S-III-1, S-III-3, S-III-5, S-III-6, S-III-7	Program delineated in NUREG-0313, Rev. 2,[5] & implemented through NRC Generic letter 88-01.[8]	Implementation of effective inspection, mitigation, & repair techniques are adequate programs.
		Recirc.	SS			
		RHR	SS			
		LPCI	SS			
		LPCS	SS			
		HPCS	SS			

Table B6. Brief summary of technical information and NUMARC/NRC agreements from BWR primary coolant pressure boundary industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
TGSCC	Crack initiation & growth	Piping & Fittings		G-2, G-4	Non-significant	CSs do not suffer TGSCC under BWR operating conditions of temperature, dissolve oxygen, & stress;[9,10] SSs (N <0.12%) do not suffer TGSCC under BWR operating conditions of temp., DO, impurity level, and design stress;[11,12] nitrogen concentrations of >0.12% are not in BWR application.
		MS	CS			
		FW	CS			
		Recirc.	SS			
		RHR	CS, SS			
		RCIC	CS			
		HPCI	CS			
		LPCI	CS, SS			
		LPCS	CS, SS			
		HPCS	CS, SS			
		Relief & In-Line Valves				
		Valve Body	CS, CASS			
		Bonnet	CS, CASS			
		Seal Flange	CS, SS			
		Nuts & Bolts	CS, SS			
		Recirculation Pump				
		Bowl	CASS			
		Cover	CASS			
		Heat Exchanger	SS			
		Seal Flange	SS			
		Nuts & Bolts	SS			
		Integral Support	CS, SS			

Table B6. Brief summary of technical information and NUMARC/NRC agreements from BWR primary coolant pressure boundary industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IASCC	Crack initiation & growth	Piping & Fittings		G-2	Non-significant	IASCC is non-significant because the total fast neutron fluence within the license renewal term is less than 1×10^{20} n/m^2.
		MS	CS			
		FW	CS			
		Recirc.	SS			
		RHR	CS, SS			
		RCIC	CS			
		HPCI	CS			
		LPCI	CS, SS			
		LPCS	CS, SS			
		HPCS	CS, SS			
		Relief & In-Line Valves				
		Valve Body	CS, CASS			
		Bonnet	CS, CASS			
		Seal Flange	CS, SS			
		Nuts & Bolts	CS, SS			
		Recirculation Pump				
		Bowl	CASS			
		Cover	CASS			
		Heat Exchanger	SS			
		Seal Flange	SS			
		Nuts & Bolts	SS			
		Integral Support	CS, SS			

Table B6. Brief summary of technical information and NUMARC/NRC agreements from BWR primary coolant pressure boundary industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion	Loss of material	Piping & Fittings		G-2, S-IV-1, S-IV-2, S-IV-3	Non-significant	Water quality & chemistry are controlled according to technical specifications requirements and corrosion allowances are defined according to pressure integrity requirements.
		MS	CS			
		FW	CS			
		Recirc.	SS			
		RHR	CS, SS			
		RCIC	CS			
		HPCI	CS			
		LPCI	CS, SS			
		LPCS	CS, SS			
		HPCS	CS, SS			
		Relief & In-Line Valves				
		Valve Body	CS, CASS			
		Bonnet	CS, CASS			
		Seal Flange	CS, SS			
		Nuts & Bolts	CS, SS			
		Recirculation Pump				
		Bowl	CASS			
		Cover	CASS			
		Heat Exchanger	SS			
		Seal Flange	SS			
		Nuts & Bolts	SS			
		Integral Support	CS, SS			

Table B6. Brief summary of technical information and NUMARC/NRC agreements from BWR primary coolant pressure boundary industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Erosion/ Corrosion (E/C)	Wall thinning	Piping & Fittings		G-2	Non-significant	SS components are resistant to E/C, CS components operate in low temperature (< 79°C) and/ or low flow range, components not in contact with primary coolant.
		Recirc.	SS			
		RHR	CS, SS			
		RCIC	CS			
		HPCI	CS			
		LPCI	CS, SS			
		LPCS	CS, SS			
		HPCS	CS, SS			
		Relief & In-Line Valves				
		Valve Body	CASS			
		Bonnet	CS, CASS			
		Seal Flange	CS, SS			
		Nuts & Bolts	CS, SS			
		Recirculation Pump				
		Bowl	CASS			
		Cover	CASS			
		Heat Exchanger	SS			
		Seal Flange	SS			
		Nuts & Bolts	SS			
		Integral Support	CS, SS			
E/C	Wall thinning	Piping & Fittings		G-1, G-3, S-VI-1	Appendix A of NUREG-1344 for single-phase lines.[13] CHECMATE Code for two-phase lines.[14]	NUREG-1344[13] recommends industry program for control of E/C in single-phase systems & CHECMATE[14] predicts E/C in two-phase systems.
		MS	CS			
		FW	CS			

Table B6. Brief summary of technical information and NUMARC/NRC agreements from BWR primary coolant pressure boundary industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
E/C	Wall thinning	Relief & In-Line Valves Valve Body	 CS	G-1, G-3, S-VI-2	ASME Sect. XI requires VT-3 of valve body internal surfaces & VT-2 of pressure retaining boundary & system leakage & hydrostatic tests. Also, compliance with NEDC 31743[15] is necessary.	ASME Sect XI, Subsect. IWB, exam. categories B-M-1 & -2, & B-P; and guidelines of NEDC-31743;[15] are current & effective ISI and testing programs for detection and evaluation-repair-replacement of valve components.
Thermal Embrittlement of CASS	Loss of fracture toughness	Relief & In-Line Valves Valve Body Bonnet Recirculation Pump Bowl Cover	 CASS CASS CASS CASS	G-1, G-3, S-V-2, S-V-3, S-V-4 Open issues S-V-1, S-V-5, S-V-6	Unresolved issue. *NUMARC position*: ASME Sect. XI, Subsect. IWB, requires visual VT-3 exam. & analytical evaluation procedures for flaw tolerance. *NRC position*: Ferrite criteria is inadequate tool for screening & VT-3 can not reliably detect tight cracks. Fracture toughness may be estimated based on NUREG/CR-4513 Rev. 1.[17]	*NUMARC basis*: Components with ferrite content >20% for all centrifugally cast & static-cast CF-3 & CF-8, or >14% for static -cast CF-3M & CF-8M, ASME Sect. XI, Subsect. IWB,[16] exam categories B-L-1 & -2, B-M-1 & -2, & B-P; & flaw tolerance evalution of ASME Code Case 481 is current & effective program for detection & evaluation-repair-replacement.

Table B6. Brief summary of technical information and NUMARC/NRC agreements from BWR primary coolant pressure boundary industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron Irradiation Embrittlement	Loss of fracture toughness	Piping & Fittings		G-2	Non-significant	Total fast neutron fluence within the license renewal term is $<1 \times 10^{17}$ n/m^2 for the PCPB components.
		MS	CS			
		FW	CS			
		Recirc.	SS			
		RHR	CS, SS			
		RCIC	CS			
		HPCI	CS			
		LPCI	CS, SS			
		LPCS	CS, SS			
		HPCS	CS, SS			
		Relief & In-Line Valves				
		Valve Body	CS, CASS			
		Bonnet	CS, CASS			
		Seal Flange	CS, SS			
		Nuts & Bolts	CS, SS			
		Recirculation Pump				
		Bowl	CASS			
		Cover	CASS			
		Heat Exchanger	SS			
		Seal Flange	SS			
		Nuts & Bolts	SS			
		Integral Support	CS, SS			

Table B6. Brief summary of technical information and NUMARC/NRC agreements from BWR primary coolant pressure boundary industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Creep	Change in dimension	Piping & Fittings		G-2	Non-significant	Operating temperatures are <371°C (<700°F) for CS, <538°C (<1000°F) for SS.
		MS	CS			
		FW	CS			
		Recirc.	SS			
		RHR	CS, SS			
		RCIC	CS			
		HPCI	CS			
		LPCI	CS, SS			
		LPCS	CS, SS			
		HPCS	CS, SS			
		Relief & In-Line Valves				
		Valve Body	CS, CASS			
		Bonnet	CS, CASS			
		Seal Flange	CS, SS			
		Nuts & Bolts	CS, SS			
		Recirculation Pump				
		Bowl	CASS			
		Cover	CASS			
		Heat Exchanger	SS			
		Seal Flange	SS			
		Nuts & Bolts	SS			
		Integral Support	CS, SS			

Table B6. Brief summary of technical information and NUMARC/NRC agreements from BWR primary coolant pressure boundary industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Stress Relaxation	Loss of preload	Piping & Fittings		G-2	Non-significant	These components do not depend on preload for functionality.
		MS	CS			
		FW	CS			
		Recirc.	SS			
		RHR	CS, SS			
		RCIC	CS			
		HPCI	CS			
		LPCI	CS, SS			
		LPCS	CS, SS			
		HPCS	CS, SS			
		Relief & In-Line Valves				
		Valve Body	CS, CASS			
		Bonnet	CS, CASS			
		Seal Flange	CS, SS			
		Recirculation Pump				
		Bowl	CASS			
		Cover	CASS			
		Heat Exchanger	SS			
		Seal Flange	SS			
		Integral Support	CS, SS			
Stress Relaxation	Loss of preload	Relief & In-Line Valves		G-1, G-3, S-II-1	ASME Sect. XI, Table IWB-2500-1, ISI includes VT-1 of flange, nuts, bushing, and washer surfaces, & volumetric exam. of bolts & studs,[16] corrective measure IWA-5250, & acceptance criteria IWA-3142.	ASME Sect. XI, Subesct. IWB, exam. categories B-G-1 & -2, & testing category B-P for system leakage,[16] are current & effective ISI and testing programs for detection & correction of loss of bolting preload.
		Nuts & Bolts	CS, SS			
		Recirculation Pump				
		Nuts & Bolts	SS			

Table B6. Brief summary of technical information and NUMARC/NRC agreements from BWR primary coolant pressure boundary industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Wear	Attrition	Piping & Fittings		G-2	Non-significant	Not subjected to relative motion or does not incorporate clamped joints.
		MS	CS			
		FW	CS			
		Recirc.	SS			
		RHR	CS, SS			
		RCIC	CS			
		HPCI	CS			
		LPCI	CS, SS			
		LPCS	CS, SS			
		HPCS	CS, SS			
		Relief & In-Line Valves				
		Valve Body	CS, CASS			
		Bonnet	CS, CASS			
		Nuts & Bolts	CS, SS			
		Recirculation Pump				
		Bowl	CASS			
		Cover	CASS			
		Heat Exchanger	SS			
		Nuts & Bolts	SS			
		Integral Support	CS, SS			
Wear	Attrition	Relief & In-Line Valves		G-1, G-3, S-VII-1	ASME Sect. XI, Table IWB-2500-1, ISI includes VT-1 of flange, nuts, bushing, & washer surfaces, & volumetric exam. of bolts & studs.	ASME Sect. XI, Subsect. IWB, ISI exam. categories B-G-1 & -2, & testing category B-P for system leakage,[16] are current & effective ISI & testing programs for detection & evaluation-repair-replacement of valve & pump components.
		Seal Flange	CS, SS			
		Recirculation Pump				
		Seal Flange	SS			

Table B6. Brief summary of technical information and NUMARC/NRC agreements from BWR primary coolant pressure boundary industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	Relief & In-Line Valves		G-2, G-5, S-II-1, S-II-3, S-II-4, S-II-7, S-II-8	Unresolved issue. *NUMARC proposal*: Non-significant for these components. *NRC proposal:* Fatigue issues are unresolved until an agreement is reached in the ongoing discussions on fatigue evaluation for license renewal between NUMARC & staff.	*NUMARC basis*: No operating experience of flaws induced by fatigue, &/or plant-specific or typical evaluations of components show fatigue usage <0.4 for SS & <0.25 for CS for 40-y operation.
		Valve Body	CS, CASS			
		Bonnet	CS, CASS			
		Seal Flange	CS, SS			
		Nuts & Bolts	CS, SS			
		Recirculation Pump				
		Bowl	CASS			
		Cover (Bingham)	CASS			
		Heat Exchanger (Bingham)	SS			
		Seal Flange	SS			
		Nuts & Bolts	SS			
		Integral Support	CS, SS			
Fatigue	Cumulative fatigue damage	Piping & Fittings		G-1, G-3, G-5, S-II-2 through S-II-7, S-II-9	Unresolved issue. *NUMARC proposal*: ASME Sect. III, Subsect. NB[18] reanalysis of usage factor; actual plant transient, cycle monitoring, & partial cycle counting; ASME Sect. XI, Subsect. IWB[16] inspection requirements of IWB-2500-1; & defect repair-replacement. *NRC proposal*: Same as above.	*NUMARC basis*: Verification of continued adequacy of fatigue design basis through reanalysis of fatigue usage factor; assurance provided by ASME Sect. XI ISI and system testing programs; & defect repair & component replacement.
		MS	CS			
		FW	CS			
		Recirc.	SS			
		RHR	CS, SS			
		RCIC	CS			
		HPCI	CS			
		LPCI	CS, SS			
		LPCS	CS, SS			
		HPCS	CS, SS			

Table B6. Brief summary of technical information and NUMARC/NRC agreements from BWR primary coolant pressure boundary industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	Recirculation Pump		G-1, G-3, G-5, S-II-2 through S-II-7, S-II-9, S-II-10	Unresolved issue. *NUMARC proposal*: ASME Sect. XI, Subsect. IWB inspection requirements IWB-2500-1; ASME Code Case N-481 flaw evaluation; & component repair & replacement.	*NUMARC basis*: Combination of inservice inspection and flaw evaluation programs, and component repair & replacement.
		Cover (Byron Jackson	CASS			
		Heat Exchanger (Byron Jackson)	SS			

[a] The following comments were not included in the table because they deal with scope of the IR and, therefore, have no relevance for NUMARC/NRC agreements but would be important for preparing the SRP document: G-6, S-I-1, S-I-2, S-I-3, and S-I-4.

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17. NUREG/CR-4513 Rev. 1, ANL-93/22, "Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems-Revision 1," O. K. Chopra, Argonne National Laboratory, August 1994; also in NUREG/CR-6177, ANL-94/2, "Assessment of Thermal Embrittlement of Cast Stainless Steels," O. K. Chopra and W. J. Shack, Argonne National Laboratory, May 1994.

18. ASME B & PV Code - "Boiler and Pressure Vessel Code of Design and Construction Practices," Section III: "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017.

LIST OF BWR PRIMARY COOLANT PRESSURE BOUNDARY COMPONENTS:

BWR Primary Coolant Pressure Boundary

Piping & Fittings
- Main Steam (MS)
- Feedwater (FW)
- Recirculation
- Residual Heat Removal (RHR)
- Low Pressure Coolant Injection (LPCI)
- High Pressure Coolant Injection (HPCI)
- Low Pressure Core Spray (LPCS)
- High Pressure Core Spray (HPCS)
- Reactor Core Isolation Cooling (RCIC)

Relief & In-Line Valves
- Valve Body
- Bonnet
- Seal Flange
- Nuts & Bolts

Recirculation Pump
- Bowl
- Cover
- Heat Exchanger
- Seal Flange
- Nuts & Bolts

Integral Support

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron Irradiation Embrittlement	Loss of fracture toughness	*Upper Internals Assembly*		G-2, G-6, G-11, G-22, S-13, S-14	Non-significant	Although neutron irradiation causes a decrease in fracture toughness for SS & Ni-alloy components, the fracture toughness levels remain adequate even at end-of-life fluence levels because the applied stresses are low.[1]
		W Upper Support Plate	SS			
		C Upper Guide Structure Support Plate	SS			
		B Plenum Cover & Cylinder	SS			
		W RCCA Guide Tube Assemblies				
		RCCA Guide Tube	See Irr. Emb.			
		RCCA Guide Tube Bolts	SS, Ni alloy			
		RCCA Guide Tube Sup. Pins	SS, Ni alloy			
		C CEA Shrouds Assemblies				
		CEA Shroud	SS, CASS			
		CEA Shroud Bolts	SS, Ni alloy			
		B CRA Guide Tube Assemblies				
		CRA Guide Tubes	See Irr. Emb.			
		CRA Guide Tube Bolts	SS, Ni alloy			
		W Upper Support Columns	SS, CASS			
		Upper Support Column Bolts	SS, Ni alloy			
		W Upper Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Fuel Alignment Plate	SS			
		B Upper Grid Assembly				
		Upper Grid Rib Section	SS			
		Upper Grid Assembly Bolts	SS, Ni alloy			
		Fuel Guide Pads	SS			
		Lower Internals Assembly				
		W Radial Keys and Clevis Insert	SS			

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron Irradiation Embrittle-ment	Loss of fracture toughness	*Core Support Assembly*		G-11, G-22, S-13, S-14	ASME Sect. XI[4] requires visual VT-3 examination.	ASME Sect. XI, Subsect. IWB,[4] exam. category B-N-3, is current & effective program for detection of cracks & evaluation-repair-replacement for vessel internal components that are accessible or can be rendered accessible by removal of the core &/or other internals.
		W Core Barrel	SS			
		Upper Core Barrel Flange	SS			
		Core Barrel Nozzles	SS			
		C Core Support Barrel	SS			
		Core Sup. Barrel Upper Flange	SS			
		B Core Support Shield	SS			
		Core Support Shield Flange	SS			
		B Vent Valve Assemblies	Ni alloy, SS, CASS, Stellite, Martensitic SS			
		W Baffle/Former Assembly				
		Baffle/Former Assm. Baffles	SS			
		Baffle/Former Assembly Bolts	SS, Ni alloy			
		C Core Shroud Assembly	SS			
		Core Shroud Assembly Bolts	SS, Ni alloy			
		Core Shroud Tie Rods	SS			
		B Core Barrel	SS			
		Core Barrel Bolts	SS, Ni alloy			
		Baffle/Former Plates	SS			
		Baffle/Former Bolts	SS, Ni alloy			
		W Upper Core Plate Align. Pins	SS, Ni alloy			
		C Fuel Align. Plate Guide Lugs	SS	Contd. on next page	Continued on next page	Continued on next page

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron Irradiation Embrittlement	Loss of fracture toughness	*Lower Internals Assembly*		See previous page	See previous page	See previous page
		W Lower Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Core Support Plate	SS			
		Fuel Alignment Pins	SS, Ni alloy			
		B Lower Grid Top Rib Section	SS			
		Fuel Guide Pads	SS			
		W Lower Support Plate	SS, CASS			
		C Lower Support Structure Beam Assemblies	SS			
		B Lower Grid Bottom Rib Weldment	SS			
		W Lower Support Columns	SS, CASS			
		Lower Support Column Bolts	SS, Ni alloy			
		C Core Support Columns	SS, CASS			
		Core Support Column Bolts	SS, Ni alloy			
		B Lower Grid Assembly Support Posts	SS			
		Lower Grid Assembly Bolts	SS, Ni alloy			
		C Core Support Barrel Snubber Assembly	SS			
		B Lower Grid Cylinder and Guide Blocks	SS			

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
SCC	Crack initiation & growth	*Upper Internals Assembly*		G-2, G-6, G-11, G-17, G-22, S-15, S-18, S-19 Open issues G-19, S-16	Unresolved issue *NUMARC proposal*: SCC is non-significant for these components that are fabricated from SS subjected to stress levels within design specs. and PWR water chemistry. *NRC proposal*: Crevices are known to promote SCC in SSs even in the absence of high stress, particularly for components that are near the core because radiation may produce aggressive conditions in crevices. Evaluate the potential of SCC of components with crevices or creviced geometry.	*NUMARC basis*: Fabricated of SS or SS with >5% ferrite; and subjected to stress levels within design specifications; and PWR operating chemistry limits the oxygen to <5 ppb & halogens to <150 ppb.[5]
		W Upper Support Plate	SS			
		C Upper Guide Structure Support Plate	SS			
		B Plenum Cover & Cylinder	SS			
		W RCCA Guide Tube Assemblies				
		RCCA Guide Tube	See Irr. Emb.			
		RCCA Guide Tube Bolts	SS, Ni alloy			
		C CEA Shrouds Assemblies				
		CEA Shroud	SS, CASS			
		B CRA Guide Tube Assemblies				
		CRA Guide Tubes	See Irr. Emb.			
		CRA Guide Tube Bolts	SS, Ni alloy			
		W Upper Support Columns	SS, CASS			
		Upper Support Column Bolts	SS, Ni alloy			
		W Upper Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Fuel Alignment Plate	SS			
		B Upper Grid Assembly				
		Upper Grid Rib Section	SS			
		Upper Grid Assembly Bolts	SS, Ni alloy			
		Fuel Guide Pads	SS			
		Core Support Assembly				
		W Core Barrel	SS			
		Upper Core Barrel Flange	SS			
		Core Barrel Nozzles	SS			
		C Core Support Barrel	SS			
		Core Sup. Barrel Upper Flange	SS			
		B Core Support Shield	SS	Contd.		
		Core Support Shield Flange	SS	on next		
		B Vent Valve Assemblies	See Irr. Emb.	page	Continued on next page	Continued on next page

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
SCC	Crack initiation & growth	*W* Baffle/Former Assembly		See previous page	Unresolved issue: Continued from previous page	Continued from previous page
		Baffle/Former Assm. Baffles	SS			
		Baffle/Former Assembly Bolts	SS, Ni alloy			
		C Core Shroud Assembly	SS			
		Core Shroud Assembly Bolts	SS, Ni alloy			
		Core Shroud Tie Rods	SS			
		B Core Barrel	SS			
		Baffle/Former Plates	SS			
		Baffle/Former Bolts	SS, Ni alloy			
		W Upper Core Plate Align. Pins	SS, Ni alloy			
		C Fuel Align. Plate Guide Lugs	SS			
		Lower Internals Assembly				
		W Lower Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Core Support Plate	SS			
		Fuel Alignment Pins	SS, Ni alloy			
		B Lower Grid Top Rib Section	SS			
		Fuel Guide Pads	SS			
		W Lower Support Plate	SS, CASS			
		C Lower Support Structure Beam Assemblies	SS			
		B Lower Grid Bot. Rib Weldment	SS			
		W Lower Support Columns	SS, CASS			
		Lower Support Column Bolts	SS, Ni alloy			
		C Core Support Columns	SS, CASS			
		Core Support Column Bolts	SS, Ni alloy			
		B Lower Grid Assm. Sup. Posts	SS			
		Lower Grid Assembly Bolts	SS, Ni alloy			
		W Radial Keys and Clevis Insert	SS			
		C Core Support Barrel Snubber Assembly	SS			
		B Lower Grid Cylinder & Guide Blocks	SS			

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
SCC	Crack initiation & growth	*Upper Internals Assembly*		G-1, G-6, G-7 to G-9, G-15, G-18, G-20, S-2, S-3, S-10, S-11, S-15, S-18, S-19 Open issue G-16	Unresolved issue *NUMARC proposal*: ASME Sect. XI[4] requires visual VT-3 exam. & replacement is with designs that reduce applied stress. *NRC proposal*: Augmented ISI of components when conditions, such as sensitized material, high residual stresses, crevices, stagnant flow, & history of coolant contamination, are present.	*NUMARC basis*: ASME Sect. XI, Subsect. IWB,[4] exam. category B-N-3 is current & effective program for detection & evaluation-repair-replacement.
		W RCCA Guide Tube Support Pins	SS, Ni alloy			
		C CEA Shroud Bolts	SS, Ni alloy			
		Core Support Assembly				
		B Core Barrel Bolts	SS, Ni alloy			

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IASCC	Crack initiation & growth	*Upper Internals Assembly*		G-1, G-6, G-7, G-8, G-9, G-11, G-15, G-18, G-20, G-22, S-1, S-2, S-3, S-10, S-11, S-17, S-20	ASME Sect. XI[4] requires visual VT-3 examination & replacement is with designs that reduce applied stress.	ASME Sect. XI, Subsect. IWB,[4] exam. category B-N-3 is current & effective program for detection & evaluation-repair-replacement for vessel internal components that are accessible or can be rendered accessible by removal of the core &/or other internals.
		W Upper Support Plate	SS			
		C Upper Guide Structure Support Plate	SS			
		B Plenum Cover & Cylinder	SS			
		W RCCA Guide Tube Assemblies				
		RCCA Guide Tube	See Irr. Emb.			
		RCCA Guide Tube Bolts	SS, Ni alloy			
		RCCA Guide Tube Sup. Pins	SS, Ni alloy			
		C CEA Shrouds Assemblies				
		CEA Shroud	SS, CASS			
		CEA Shroud Bolts	SS, Ni alloy			
		B CRA Guide Tube Assemblies				
		CRA Guide Tubes	See Irr. Emb.			
		CRA Guide Tube Bolts	SS, Ni alloy			
		W Upper Support Columns	SS, CASS			
		Upper Support Column Bolts	SS, Ni alloy			
		W Upper Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Fuel Alignment Plate	SS			
		B Upper Grid Assembly				
		Upper Grid Rib Section	SS			
		Upper Grid Assembly Bolts	SS, Ni alloy			
		Fuel Guide Pads	SS			
		Core Support Assembly				
		W Core Barrel	SS			
		Upper Core Barrel Flange	SS			
		Core Barrel Nozzles	SS			
		C Core Support Barrel	SS			
		Core Sup. Barrel Upper Flange	SS			
		B Core Support Shield	SS	Contd. on next page		
		Core Support Shield Flange	SS			
		B Vent Valve Assemblies	See Irr. Emb.		Continued on next page	Continued on next page

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IASCC	Crack initiation & growth	*W* Baffle/Former Assembly		See previous page	See previous page	See previous page
		Baffle/Former Assm. Baffles	SS			
		Baffle/Former Assembly Bolts	SS, Ni alloy			
		C Core Shroud Assembly	SS			
		Core Shroud Assembly Bolts	SS, Ni alloy			
		Core Shroud Tie Rods	SS			
		B Core Barrel	SS			
		Core Barrel Bolts	SS, Ni alloy			
		Baffle/Former Plates	SS			
		Baffle/Former Bolts	SS, Ni alloy			
		W Upper Core Plate Align. Pins	SS, Ni alloy			
		C Fuel Align. Plate Guide Lugs	SS			
		Lower Internals Assembly				
		W Lower Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Core Support Plate	SS			
		Fuel Alignment Pins	SS, Ni alloy			
		B Lower Grid Top Rib Section	SS			
		Fuel Guide Pads	SS			
		W Lower Support Plate	SS, CASS			
		C Lower Support Structure Beam Assemblies	SS			
		B Lower Grid Bot. Rib Weldment	SS			
		W Lower Support Columns	SS, CASS			
		Lower Support Column Bolts	SS, Ni alloy			
		C Core Support Columns	SS, CASS			
		Core Support Column Bolts	SS, Ni alloy			
		B Lower Grid Assm. Sup. Posts	SS			
		Lower Grid Assembly Bolts	SS, Ni alloy			
		W Radial Keys and Clevis Insert	SS			
		C Core Support Barrel Snubber Assembly	SS			
		B Lower Grid Cylinder & Guide Blocks	SS			

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Erosion/ Corrosion (E/C)	Wall thinning	*Upper Internals Assembly*		G-2, G-6, G-11, G-22, S-21	Non-significant	All components are constructed of SS or Ni-alloys that are considered resistant to E/C in PWR environment; relatively low fluid flow velocities; pH level in the bulk coolant minimize corrosion; operating pressures of PWR preclude cavitation erosion; & purity & particulate control of the coolant eliminate particulate erosion.[6]
		W Upper Support Plate	SS			
		C Upper Guide Structure Support Plate	SS			
		B Plenum Cover & Cylinder	SS			
		W RCCA Guide Tube Assemblies				
		RCCA Guide Tube	See Irr. Emb.			
		RCCA Guide Tube Bolts	SS, Ni alloy			
		RCCA Guide Tube Sup. Pins	SS, Ni alloy			
		C CEA Shrouds Assemblies				
		CEA Shroud	SS, CASS			
		CEA Shroud Bolts	SS, Ni alloy			
		B CRA Guide Tube Assemblies				
		CRA Guide Tubes	See Irr. Emb.			
		CRA Guide Tube Bolts	SS, Ni alloy			
		W Upper Support Columns	SS, CASS			
		Upper Support Column Bolts	SS, Ni alloy			
		W Upper Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Fuel Alignment Plate	SS			
		B Upper Grid Assembly				
		Upper Grid Rib Section	SS			
		Upper Grid Assembly Bolts	SS, Ni alloy			
		Fuel Guide Pads	SS			
		Core Support Assembly				
		W Core Barrel	SS			
		Upper Core Barrel Flange	SS			
		Core Barrel Nozzles	SS			
		C Core Support Barrel	SS			
		Core Sup. Barrel Upper Flange	SS			
		B Core Support Shield	SS	Contd. on next page		
		Core Support Shield Flange	SS			
		B Vent Valve Assemblies	See Irr. Emb.		Continued on next page	Continued on next page

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Erosion/ Corrosion (E/C)	Wall thinning	*W* Baffle/Former Assembly		See previous page	See previous page	See previous page
		Baffle/Former Assm. Baffles	SS			
		Baffle/Former Assembly Bolts	SS, Ni alloy			
		C Core Shroud Assembly	SS			
		Core Shroud Assembly Bolts	SS, Ni alloy			
		Core Shroud Tie Rods	SS			
		B Core Barrel	SS			
		Core Barrel Bolts	SS, Ni alloy			
		Baffle/Former Plates	SS			
		Baffle/Former Bolts	SS, Ni alloy			
		W Upper Core Plate Align. Pins	SS, Ni alloy			
		C Fuel Align. Plate Guide Lugs	SS			
		Lower Internals Assembly				
		W Lower Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Core Support Plate	SS			
		Fuel Alignment Pins	SS, Ni alloy			
		B Lower Grid Top Rib Section	SS			
		Fuel Guide Pads	SS			
		W Lower Support Plate	SS, CASS			
		C Lower Support Structure Beam Assemblies	SS			
		B Lower Grid Bot. Rib Weldment	SS			
		W Lower Support Columns	SS, CASS			
		Lower Support Column Bolts	SS, Ni alloy			
		C Core Support Columns	SS, CASS			
		Core Support Column Bolts	SS, Ni alloy			
		B Lower Grid Assm. Sup. Posts	SS			
		Lower Grid Assembly Bolts	SS, Ni alloy			
		W Radial Keys and Clevis Insert	SS			
		C Core Support Barrel Snubber Assembly	SS			
		B Lower Grid Cylinder & Guide Blocks	SS			

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging–Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Creep	Change in dimension	*Upper Internals Assembly*		G-2, G-6, G-10, G-11, G-22, S-22	Non-significant	The maximum temperatures of 427°C experienced by PWR vessel internals, including localized temperature excursions, are well below the temperatures at which creep is a concern for any of the SS or Ni-alloy PWR vessel internal components.[7]
		W Upper Support Plate	SS			
		C Upper Guide Structure Support Plate	SS			
		B Plenum Cover & Cylinder	SS			
		W RCCA Guide Tube Assemblies				
		RCCA Guide Tube	See Irr. Emb.			
		RCCA Guide Tube Bolts	SS, Ni alloy			
		RCCA Guide Tube Sup. Pins	SS, Ni alloy			
		C CEA Shrouds Assemblies				
		CEA Shroud	SS, CASS			
		CEA Shroud Bolts	SS, Ni alloy			
		B CRA Guide Tube Assemblies				
		CRA Guide Tubes	See Irr. Emb.			
		CRA Guide Tube Bolts	SS, Ni alloy			
		W Upper Support Columns	SS, CASS			
		Upper Support Column Bolts	SS, Ni alloy			
		W Upper Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Fuel Alignment Plate	SS			
		B Upper Grid Assembly				
		Upper Grid Rib Section	SS			
		Upper Grid Assembly Bolts	SS, Ni alloy			
		Fuel Guide Pads	SS			
		Core Support Assembly				
		W Core Barrel	SS			
		Upper Core Barrel Flange	SS			
		Core Barrel Nozzles	SS	Contd.		
		C Core Support Barrel	SS	on next		
		Core Sup. Barrel Upper Flange	SS	page	Continued on next page	Continued on next page

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Creep	Change in dimension	*B* Core Support Shield	SS	See previous page	See previous page	See previous page
		Core Support Shield Flange	SS			
		B Vent Valve Assemblies	See Irr. Emb.			
		W Baffle/Former Assembly				
		Baffle/Former Assm. Baffles*	SS			
		Baffle/Former Assembly Bolts	SS, Ni alloy			
		C Core Shroud Assembly*	SS			
		Core Shroud Assembly Bolts	SS, Ni alloy			
		Core Shroud Tie Rods	SS			
		B Core Barrel	SS			
		Core Barrel Bolts	SS, Ni alloy			
		Baffle/Former Plates*	SS			
		Baffle/Former Bolts	SS, Ni alloy			
		W Upper Core Plate Align. Pins	SS, Ni alloy			
		C Fuel Align. Plate Guide Lugs	SS			
		Lower Internals Assembly				
		W Lower Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Core Support Plate	SS			
		Fuel Alignment Pins	SS, Ni alloy			
		B Lower Grid Top Rib Section	SS			
		Fuel Guide Pads	SS			
		W Lower Support Plate	SS, CASS			
		C Lower Support Structure Beam Assemblies	SS			
		B Lower Grid Bot. Rib Weldment	SS	Contd. on next page		
		W Lower Support Columns	SS, CASS			
		Lower Support Column Bolts	SS, Ni alloy		Continued on next page	Continued on next page

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Creep	Change in dimension	*C* Core Support Columns	SS, CASS	See previous page	See previous page	See previous page
		Core Support Column Bolts	SS, Ni alloy			
		B Lower Grid Assm. Sup. Posts	SS			
		Lower Grid Assembly Bolts	SS, Ni alloy			
		W Radial Keys and Clevis Insert	SS			
		C Core Support Barrel Snubber Assembly	SS			
		B Lower Grid Cylinder & Guide Blocks	SS			

* Recent studies (Refs. 8 and 9) indicate that dimensional changes and swelling could occur in the baffle/former assemblies where the flux/fluences are very high.

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Stress Relaxation	Loss of preload	*Upper Internals Assembly*		G-2, G-6, G-11, G-22,	Non-significant	These components do not depend on preload for functionality.
		W Upper Support Plate	SS			
		C Upper Guide Structure Support Plate	SS			
		B Plenum Cover & Cylinder	SS			
		W RCCA Guide Tube Assemblies				
		RCCA Guide Tube	See Irr. Emb.			
		RCCA Guide Tube Sup. Pins	SS, Ni alloy			
		C CEA Shrouds Assemblies				
		CEA Shroud	SS, CASS			
		B CRA Guide Tube Assemblies				
		CRA Guide Tubes	See Irr. Emb.			
		W Upper Support Columns	SS, CASS			
		W Upper Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Fuel Alignment Plate	SS			
		B Upper Grid Assembly				
		Upper Grid Rib Section	SS			
		Fuel Guide Pads	SS			
		Core Support Assembly				
		W Core Barrel	SS			
		Upper Core Barrel Flange	SS			
		Core Barrel Nozzles	SS			
		C Core Support Barrel	SS			
		Core Sup. Barrel Upper Flange	SS			
		B Core Support Shield	SS	Contd. on next page		
		Core Support Shield Flange	SS			
		B Vent Valve Assemblies	See Irr. Emb.		Continued on next page	Continued on next page

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Stress Relaxation	Loss of preload	*W* Baffle/Former Assembly		See previous page	See previous page	See previous page
		Baffle/Former Assm. Baffles	SS			
		C Core Shroud Assembly	SS			
		B Core Barrel	SS			
		Baffle/Former Plates	SS			
		W Upper Core Plate Align. Pins	SS, Ni alloy			
		C Fuel Align. Plate Guide Lugs	SS			
		Lower Internals Assembly				
		W Lower Core Plate	SS			
		C Core Support Plate	SS			
		B Lower Grid Top Rib Section	SS			
		Fuel Guide Pads	SS			
		W Lower Support Plate	SS, CASS			
		C Lower Support Structure Beam Assemblies	SS			
		B Lower Grid Bot. Rib Weldment	SS			
		W Lower Support Columns	SS, CASS			
		C Core Support Columns	SS, CASS			
		B Lower Grid Assm. Sup. Posts	SS			
		W Radial Keys and Clevis Insert	SS			
		C Core Support Barrel Snubber Assembly	SS			
		B Lower Grid Cylinder & Guide Blocks	SS			

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Stress Relaxation	Loss of preload	*Upper Internals Assembly* *W* RCCA Guide Tube Bolts *C* CEA Shroud Bolts *B* CRA Guide Tube Bolts *W* Upper Support Column Bolts *B* Upper Grid Assembly Bolts *Core Support Assembly* *C* Core Shroud Tie Rods *B* Core Barrel Bolts *Lower Internals Assembly* *W* Fuel Pins *C* Fuel Alignment Pins *W* Lower Support Column Bolts *C* Core Support Column Bolts *B* Lower Grid Assembly Bolts	 SS, Ni alloy SS, Ni alloy SS, Ni alloy SS, Ni alloy SS, Ni alloy SS SS, Ni alloy SS, Ni alloy SS, Ni alloy SS, Ni alloy SS, Ni alloy SS, Ni alloy	G-1, G-6 to G-9, G-15, G-18, G-20, S-2, S-3, S-10, S-11, S-23, S-24, S-43	Unresolved issue *NUMARC proposal*: ASME Sect. XI[4] requires visual VT-3 exam. including root cause determination & evaluation. *NRC Comments S-37 & 38:* Some bolts & pins (page 5-18 of Report 90-05, Sept. 1990) were excluded because of redundancy and periodic inspection. This issue has been resolved because NUMARC has included all bolts & pins in ARD management in the revised IR.	*NUMARC proposal:* ASME Sect. XI, Subsect. IWB,[4] exam. category B-N-3 is current & effective program for detection & evaluation-repair-replacement for vessel internals that are or can be rendered accessible. NRC/industry has ongoing implementation under Unresolved Issue A-46 to verify bolt preload during service.
Stress Relaxation	Loss of preload	*Core Support Assembly* *W* Baffle/Former Assembly Bolts *C* Core Shroud Assembly Bolts *B* Baffle/Former Bolts	 SS, Ni alloy SS, Ni alloy SS, Ni alloy	open issue S-37, S-38	Current practices to be enhanced, and requires further plant-specific evaluation.	Management program is to be justified on a plant-specific basis.

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Wear	Attrition	*Upper Internals Assembly*		G-2,	Non-significant	Not subjected to relative motion associated with sliding, flow induced vibration, loss of clamping force, or from thermal effects.
		W Upper Support Plate	SS	G-5,		
		C Upper Guide Structure Support Plate	SS	G-6, G-11,		
		B Plenum Cover & Cylinder	SS	G-22		
		W RCCA Guide Tube Assemblies				
		RCCA Guide Tube Bolts	SS, Ni alloy			
		RCCA Guide Tube Sup. Pins	SS, Ni alloy			
		C CEA Shrouds Assemblies				
		CEA Shroud Bolts	SS, Ni alloy			
		B CRA Guide Tube Assemblies				
		CRA Guide Tube Bolts	SS, Ni alloy			
		W Upper Support Columns	SS, CASS			
		Upper Support Column Bolts	SS, Ni alloy			
		W Upper Core Plate	SS			
		B Upper Grid Assembly				
		Upper Grid Rib Section	SS			
		Upper Grid Assembly Bolts	SS, Ni alloy			
		Core Support Assembly				
		W Core Barrel	SS			
		Core Barrel Nozzles	SS			
		C Core Support Barrel	SS			
		B Core Support Shield	SS	on next		
		B Vent Valve Assemblies	See Irr. Emb.	page	Continued on next page	Continued on next page

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Wear	Attrition	*W* Baffle/Former Assembly		See previous page	Non-significant	See previous page
		Baffle/Former Assm. Baffles	SS			
		Baffle/Former Assembly Bolts	SS, Ni alloy			
		C Core Shroud Assembly	SS			
		Core Shroud Assembly Bolts	SS, Ni alloy			
		Core Shroud Tie Rods	SS			
		B Core Barrel	SS			
		Core Barrel Bolts	SS, Ni alloy			
		Baffle/Former Plates	SS			
		Baffle/Former Bolts	SS, Ni alloy			
		Lower Internals Assembly				
		W Lower Core Plate	SS			
		C Core Support Plate	SS			
		B Lower Grid Top Rib Section	SS			
		W Lower Support Plate	SS, CASS			
		C Lower Support Structure Beam Assemblies	SS			
		B Lower Grid Bot. Rib Weldment	SS			
		W Lower Support Columns	SS, CASS			
		Lower Support Column Bolts	SS, Ni alloy			
		C Core Support Columns	SS, CASS			
		Core Support Column Bolts	SS, Ni alloy			
		B Lower Grid Assm. Sup. Posts	SS			
		Lower Grid Assembly Bolts	SS, Ni alloy			

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Wear	Attrition	*Upper Internals Assembly*		G-1, G-5 to G-9, G-15, G-18, G-20, S-2, S-3, S-10, S-11, S-39	ASME Sect. XI[4] requires visual VT-3 examination.	ASME Sect. XI, Subsect. IWB,[4] exam. category B-N-3 is current & effective program for detection & evaluation-repair-replacement for vessel internal components that can be removed from the vessel.
		W RCCA Guide Tube	SS, Nicrobraze			
		C CEA Shrouds	SS, CASS			
		B CRA Guide Tubes	CASS, SS, Nicrobraze			
		W Fuel Pins	SS, Ni alloy			
		C Fuel Alignment Plate	SS			
		B Fuel Guide Pads	SS			
		Core Support Assembly				
		W Upper Core Barrel Flange	SS			
		C Core Support Barrel Upper Flange	SS			
		B Core Support Shield Flange	SS			
		W Upper Core Plate Alignment Pins	SS, Ni alloy			
		C Fuel Alignment Plate Guide Lugs	SS			
		Lower Internals Assembly				
		W Fuel Pins	SS, Ni alloy			
		C Fuel Alignment Pins	SS, Ni alloy			
		B Fuel Guide Pads	SS			
		W Radial Keys and Clevis Insert	SS			
		C Core Sup Barrel Snubber Assembly	SS			
		B Lower Grid Cylinder and Guide Blocks	SS			

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Thermal Embrittle-ment	Loss of fracture toughness	*Upper Internals Assembly*		G-2, G-6, G-11, G-22,	Non-significant	Wrought SS & Ni-alloys are not susceptible to thermal aging embrittlement.
		W Upper Support Plate	SS			
		C Upper Guide Structure Support Plate	SS			
		B Plenum Cover & Cylinder	SS			
		W RCCA Guide Tube Assemblies				
		RCCA Guide Tube	See Irr. Emb.			
		RCCA Guide Tube Bolts	SS, Ni alloy			
		RCCA Guide Tube Sup. Pins	SS, Ni alloy			
		C CEA Shrouds Assemblies				
		CEA Shroud	SS			
		CEA Shroud Bolts	SS, Ni alloy			
		B CRA Guide Tube Assemblies				
		CRA Guide Tubes	See Irr. Emb.			
		CRA Guide Tube Bolts	SS, Ni alloy			
		W Upper Support Columns	SS			
		Upper Support Column Bolts	SS, Ni alloy			
		W Upper Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Fuel Alignment Plate	SS			
		B Upper Grid Assembly				
		Upper Grid Rib Section	SS			
		Upper Grid Assembly Bolts	SS, Ni alloy			
		Fuel Guide Pads	SS			
		Core Support Assembly				
		W Core Barrel	SS			
		Upper Core Barrel Flange	SS			
		Core Barrel Nozzles	SS			
		C Core Support Barrel	SS			
		Core Sup. Barrel Upper Flange	SS			
		B Core Support Shield	SS	Contd. on next page		
		Core Support Shield Flange	SS			
		B Vent Valve Assemblies	See Irr. Emb.		Continued on next page	Continued on next page

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Thermal Embrittlement	Loss of fracture toughness	*W* Baffle/Former Assembly		See previous page	See previous page	See previous page
		Baffle/Former Assm. Baffles	SS			
		Baffle/Former Assembly Bolts	SS, Ni alloy			
		C Core Shroud Assembly	SS			
		Core Shroud Assembly Bolts	SS, Ni alloy			
		Core Shroud Tie Rods	SS			
		B Core Barrel	SS			
		Core Barrel Bolts	SS, Ni alloy			
		Baffle/Former Plates	SS			
		Baffle/Former Bolts	SS, Ni alloy			
		W Upper Core Plate Align. Pins	SS, Ni alloy			
		C Fuel Align. Plate Guide Lugs	SS			
		Lower Internals Assembly				
		W Lower Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Core Support Plate	SS			
		Fuel Alignment Pins	SS, Ni alloy			
		B Lower Grid Top Rib Section	SS			
		Fuel Guide Pads	SS			
		W Lower Support Plate	SS			
		C Lower Support Structure Beam Assemblies	SS			
		B Lower Grid Bot. Rib Weldment	SS			
		W Lower Support Columns	SS			
		Lower Support Column Bolts	SS, Ni alloy			
		C Core Support Columns	SS			
		Core Support Column Bolts	SS, Ni alloy			
		B Lower Grid Assm. Sup. Posts	SS			
		Lower Grid Assembly Bolts	SS, Ni alloy			
		W Radial Keys and Clevis Insert	SS			
		C Core Support Barrel Snubber Assembly	SS			
		B Lower Grid Cylinder & Guide Blocks	SS			

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Thermal Embrittle-ment	Loss of fracture toughness	*Upper Internals Assembly*		S-12, S-25, S-40, S-41	Unresolved issue. *NUMARC proposal*: ASME Sect. XI, Subsect. IWB,[4] requires visual VT-3 exam.	*NUMARC basis*: Components with ferrite content >20% for all centrifugally cast & static-cast CF-3 & CF-8, or >14% for static cast CF-3M & CF-8M (S-26), ASME Sect. XI, Subsect. IWB,[4] exam. category B-N-3 (S-42), is current & effective program for detection & evaluation-repair-replacement.
		C CEA Shrouds	CASS			
		B CRA Guide Tubes	CASS			
		W Upper Support Columns	CASS			
		Core Support Assembly				
		B Vent Valve Assemblies	CASS	Open issue S-26, S-42	*NRC proposal*: Ferrite criteria is inadequate tool for screening & VT-3 can not reliably detect tight cracks.	
		Lower Internals Assembly				
		W Lower Support Plate	CASS			
		W Lower Support Columns	CASS			
		C Core Support Columns	CASS			

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging–Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion	Loss of material/ corrosion product buildup	*Upper Internals Assembly*		G-2, G-6, G-11, G-22, Open issue S-16	Unresolved issue (S-16) *NUMARC proposal*: Corrosion, general or local, is non-significant. *NRC proposal*: There is no assurance that components made from SSs are not exposed to locally corrosive environments.	*NUMARC basis*: The materials used in the construction of PWR vessel internals, e.g., SSs & Ni-alloys, are not susceptible to general corrosion in benign PWR primary coolant environment; components containing crevices are fabricated from SSs, a material that is also resistant to pitting corrosion; & hydrogen overpressure present in the reactor vessel controls crevice corrosion by minimizing the adverse effects of oxygen by recombination.
		W Upper Support Plate	SS			
		C Upper Guide Structure Support Plate	SS			
		B Plenum Cover & Cylinder	SS			
		W RCCA Guide Tube Assemblies				
		RCCA Guide Tube	See Irr. Emb.			
		RCCA Guide Tube Bolts	SS, Ni alloy			
		RCCA Guide Tube Sup. Pins	SS, Ni alloy			
		C CEA Shrouds Assemblies				
		CEA Shroud	SS, CASS			
		CEA Shroud Bolts	SS, Ni alloy			
		B CRA Guide Tube Assemblies				
		CRA Guide Tubes	See Irr. Emb.			
		CRA Guide Tube Bolts	SS, Ni alloy			
		W Upper Support Columns	SS, CASS			
		Upper Support Column Bolts	SS, Ni alloy			
		W Upper Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Fuel Alignment Plate	SS			
		B Upper Grid Assembly				
		Upper Grid Rib Section	SS			
		Upper Grid Assembly Bolts	SS, Ni alloy			
		Fuel Guide Pads	SS			
		Core Support Assembly				
		W Core Barrel	SS			
		Upper Core Barrel Flange	SS			
		Core Barrel Nozzles	SS			
		C Core Support Barrel	SS			
		Core Sup. Barrel Upper Flange	SS			
		B Core Support Shield	SS	Contd. on next page		
		Core Support Shield Flange	SS			
		B Vent Valve Assemblies	See Irr. Emb.		Continued on next page	Continued on next page

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion	Loss of material/ corrosion product buildup	*W* Baffle/Former Assembly		See previous page	See previous page	See previous page
		Baffle/Former Assm. Baffles	SS			
		Baffle/Former Assembly Bolts	SS, Ni alloy			
		C Core Shroud Assembly	SS			
		Core Shroud Assembly Bolts	SS, Ni alloy			
		Core Shroud Tie Rods	SS			
		B Core Barrel	SS			
		Core Barrel Bolts	SS, Ni alloy			
		Baffle/Former Plates	SS			
		Baffle/Former Bolts	SS, Ni alloy			
		W Upper Core Plate Align. Pins	SS, Ni alloy			
		C Fuel Align. Plate Guide Lugs	SS			
		Lower Internals Assembly				
		W Lower Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Core Support Plate	SS			
		Fuel Alignment Pins	SS, Ni alloy			
		B Lower Grid Top Rib Section	SS			
		Fuel Guide Pads	SS			
		W Lower Support Plate	SS, CASS			
		C Lower Support Structure Beam Assemblies	SS			
		B Lower Grid Bot. Rib Weldment	SS			
		W Lower Support Columns	SS, CASS			
		Lower Support Column Bolts	SS, Ni alloy			
		C Core Support Columns	SS, CASS			
		Core Support Column Bolts	SS, Ni alloy			
		B Lower Grid Assm. Sup. Posts	SS			
		Lower Grid Assembly Bolts	SS, Ni alloy			
		W Radial Keys and Clevis Insert	SS			
		C Core Support Barrel Snubber Assembly	SS			
		B Lower Grid Cylinder & Guide Blocks	SS			

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	*Upper Internals Assembly*		G-2, G-6, G-13, S-4, S-28, S-31, S-32, S-33, S-34, S-36 Open issue S-27	Unresolved issue *NUMARC proposal*: Non-significant ARDM. *NRC proposal*: Until an agreement is reached in the ongoing generic discussion on fatigue evaluation for license renewal between NUMARC and staff, the fatigue issues are unresolved.	*NUMARC basis*: A review of calculated fatigue usage factors for components designed to ASME Sect. III; a review of plant design stress reports & hot functional test data; and a comparison of geometry & operating similarities.
		C Upper Guide Structure Support Plate	SS			
		B Plenum Cover & Cylinder	SS			
		W RCCA Guide Tube Assemblies				
		RCCA Guide Tube Bolts	SS, Ni alloy			
		RCCA Guide Tube Sup. Pins	SS, Ni alloy			
		C CEA Shrouds Assemblies				
		CEA Shroud	SS, CASS			
		B CRA Guide Tube Assemblies				
		CRA Guide Tubes	See Irr. Emb.			
		CRA Guide Tube Bolts	SS, Ni alloy			
		W Upper Support Columns	SS, CASS			
		Upper Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
		C Fuel Alignment Plate	SS			
		B Upper Grid Assembly				
		Upper Grid Rib Section	SS			
		Upper Grid Assembly Bolts	SS, Ni alloy			
		Fuel Guide Pads	SS			
		Core Support Assembly				
		W Core Barrel	SS			
		Upper Core Barrel Flange	SS			
		C Core Support Barrel	SS			
		Core Sup. Barrel Upper Flange	SS			
		B Core Support Shield	SS			
		Core Support Shield Flange	SS			
		B Vent Valve Assemblies	See Irr. Emb.			
		C Core Shroud Assembly Bolts	SS, Ni alloy			
		B Core Barrel	SS			
		Core Barrel Bolts	SS, Ni alloy			
		Baffle/Former Plates	SS			
		Baffle/Former Bolts	SS, Ni alloy	Contd. on next page	Continued on next page	Continued on next page

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	*W* Baffle/Former Assm. Baffles	SS	See previous page	Unresolved issue	See previous page
		C Fuel Align. Plate Guide Lugs	SS		See previous page	
		Lower Internals Assembly				
		W Fuel Pins	SS, Ni alloy			
		C Fuel Alignment Pins	SS, Ni alloy			
		B Lower Grid Top Rib Section	SS			
		Fuel Guide Pads	SS			
		W Lower Support Plate	SS, CASS			
		C Lower Support Structure Beam Assemblies	SS			
		B Lower Grid Bot. Rib Weldment	SS			
		W Lower Support Column Bolts	SS, Ni alloy			
		C Core Support Column Bolts	SS, Ni alloy			
		B Lower Grid Assm. Sup. Posts	SS			
		Lower Grid Assembly Bolts	SS, Ni alloy			
		W Radial Keys and Clevis Insert	SS			
		C Core Support Barrel Snubber Assembly	SS			
		B Lower Grid Cylinder and Guide Blocks	SS			

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components[a]	Materials	NRC Comment Number[b]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	*Upper Internals Assembly*		See previous page	Unresolved issue *NUMARC proposal*: ASME Sect. XI,[4] visual VT-3 examination; Sect. III,[10] Subsect. NG-5200, reanalysis of usage factor; & transient monitoring. *NRC proposal*: Until an agreement is reached in the ongoing generic discussion on fatigue evaluation for license renewal between NUMARC and staff, the fatigue issues are unresolved.	*NUMARC basis*: Visual inspection according to ASME Sect. XI, Subsect. IWB,[4] exam. category B-N-3; verification of continued adequacy of the fatigue design basis through reanalysis in accordance with ASME Sect. III[10] Subsect. NG-5200; transient monitoring; and component repair & replacement.
		W Upper Support Plate	SS			
		W RCCA Guide Tube	See Irr. Emb.			
		C CEA Shroud Bolts	SS, Ni alloy			
		W Upper Support Column Bolts	SS, Ni alloy			
		Core Support Assembly				
		W Core Barrel Nozzles	SS			
		C Core Shroud Assembly	SS			
		Core Shroud Tie Rods	SS			
		W Upper Core Plate Align. Pins	SS, Ni alloy			
		Lower Internals Assembly				
		W Lower Core Plate	SS			
		C Core Support Plate	SS			
		W Lower Support Columns	SS, CASS			
		C Core Support Columns	SS, CASS			
Fatigue	Cumulative fatigue damage	*Core Support Assembly*				
		W Baffle/Former Assembly Bolts	SS, Ni alloy		Current practices to be enhanced, select plant specific program.*	Select plant specific management plan that could include qualified visual & volumetric[11,12] inspection, evaluation, & repair & replacement.

* Items concerning chapter six were not the focus of the NRC staff review.

a PWR vessel internal components of vendors W = Westinghouse, C = Combustion Engineering, and B = Babcock & Wilcox. RCCA = Rod control cluster assemblies; CEA = Control element assemblies; and CRA = Control rod assemblies.

b The following comments were not included in the table because they deal with clarification or modification of contents of the IR: G-3, S-29, S-30, and S-35. The following comments were also excluded because they deal with scope of the IR: G-4, G-14, G-21, S-5, S-6, S-7, S-8, and S-9 and an open issue comment G-12.

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2. "Potential High Fluence Response of Pressure Vessel Internals Constructed from Austenitic Stainless Steels," F. A. Garner, L. R. Greenwood, and D. L. Harrod, *Proc. of the Sixth Intl. Symp. on Environmental Degradation of Materials in Nuclear Power Systems - Water Reactors*," R. E. Gold and E. P. Simonen, eds., The Metallurgical Society, Warrendale, PA, pp. 783- 790, 1994.

3. "Problems Anticipated in Austenitic Pressure Vessel Internals Arising from Void Swelling, Irradiation Creep, and Swelling-Related Embrittlement," F. A. Garner, presented at the *17th ASTM Symp. on Effects of Radiation on Materials*, Sun Valley, ID, June 21-23, 1994.

4. ASME B & PV Code - "Boiler and Pressure Vessel Code of Design and Construction Practices," Section XI: "Rules for In-Service Inspection (ISI) of Nuclear Power Plant Components," American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017.

5. EPRI NP-7077, "PWR Primary Water Chemistry Guidelines: Revision 2," Electric Power Research Institute, Palo Alto, CA, November 1990.

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7. "Physical Metallurgy for Engineers," D. S. Clark and W. R. Varney, D. Van Nostrand Co., New York, 1962.

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10. ASME B & PV Code - "Boiler and Pressure Vessel Code of Design and Construction Practices," Section III: "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017.

11. EPRI NP-5769, Vol. 1, Section 3, "Degradation and Failure of Bolting in Nuclear Power Plants," R. E. Nickell, Electric Power Research Institute, Palo Alto, CA, April 1988.

12. FRAMATOME 3N89, "Recent Developments in the Inspection of Primary System. Measurements Associated with Mechanical Component Replacement," F. Bodson, A. Martin, and A. Thomas, October 18, 1989.

LIST OF PWR VESSEL INTERNAL COMPONENTS:

Westinghouse	Combustion Engineering	Babcock & Wilcox
	UPPER INTERNALS ASSEMBLY	
Upper Internals Assembly	**Upper Guide Structure Assembly**	**Plenum Assembly**
1. Upper Support Plate	1. Upper Guide Structure Support Plate	1. Plenum Cover and Plenum Cylinder
2. RCCA Guide Tube Assemblies RCCA Guide Tubes RCCA Guide Tube Bolts RCCA Guide Tube Support Pins	2. CEA Shroud Assemblies CEA Shrouds CEA Shroud Bolts	2. CRA Guide Tube Assemblies CRA Guide Tubes CRA Guide Tube Bolts
3. Upper Support Column Upper Support Column Bolts	3. NA	3. NA
4. Upper Core Plate Fuel Pins	4. Fuel Alignment Plate	4. Upper Grid Assembly Upper Grid Rib Section Upper Grid Assembly Bolts Fuel Guide Pads
	CORE SUPPORT ASSEMBLY	
Core Barrel and Baffle/Former Assembly	**Core Support Barrel & Core Shroud Assembly**	**Core Support Assembly**
1. Core Barrel Upper Core Barrel Flange Core Barrel Nozzles	1. Core Support Barrel Core Support Barrel Upper Flange	1. Core Support Shield Core Support Shield Flange
2. NA	2. NA	2. Vent Valve Assemblies
3. Baffle/Former Assembly Baffle/Former Assembly Bolts Baffle/Former Assembly Baffles	3. Core Shroud Assembly Core Shroud Assembly Bolts Core Shroud Tie Rods	3. Core Barrel Assembly Core Barrel Bolts Baffle/Former Plates Baffle/Former Bolts
4. Upper Core Plate Alignment Pins	4. Fuel Alignment Plate Guide Lugs	4. NA
	CORE SUPPORT ASSEMBLY	
Lower Internals Assembly	**Core Support Assembly**	**Lower Grid Assembly**
1. Lower Core Plate	1. Core Support Plate	1. Lower Grid Top Rib Section
2. Fuel Pins	2. Fuel Alignment Pins	2. Fuel Guide Pads
3. Lower Support Plate	3. Lower Support Structure Beam Assemblies	3. Lower Grid Bottom Rib Weldment
4. Lower Support Column Lower Support Column Bolts	4. Core Support Column Core Support Column Bolts	4. Lower Grid Assembly Support Posts Lower Grid Assembly Bolts
5. Radial Keys and Clevis Inserts	5. Core Support Barrel Snubber Assemblies	5. Lower Grid Cylinder Guide Blocks

Table B8. Brief summary of technical information and NUMARC/NRC agreements from BWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IGSCC	Crack initiation & growth	Orificed Fuel Support	SS, CASS	G-19, G-20, G-32, S-9	Non-significant	Are made of CASS, and/or subjected to low stresses.
IGSCC	Crack initiation & growth	Low Power Range Monitor (LPRM)	SS	G-19, G-32 Open issue S-45	Unresolved issue *NUMARC proposal*: Non-significant *NRC proposal*: Demonstrate how LPRMs can be qualified in view of Millstone Unit 1 experience.	*NUMARC basis*: Are replaced due to limited operating life.
IGSCC	Crack initiation & growth	Jet Pump	SS, *(CASS)*, Alloy 600, X 750	G-2, G-5, G-9, G-11, G-22, G-24, G-27 to G-29, G-33, G-34, S-4, S-6, S-8, S-17, S-22, S-25 to S-27, S-32, S-48, S-54 Open issue G-21	Unresolved issue *NUMARC proposal*: Monitoring pump performance, inspection, evaluation, & replacement. *NRC proposal*: Evaluate possible adverse effect of hydrogen water chemistry (HWC) on Ni-Alloys.	*NUMARC basis*: Periodic inspection & continuous monitoring of performance by plant instrumentation are current & effective programs for detection & evaluation-replacement of jet pumps.

Table B8. Brief summary of technical information and NUMARC/NRC agreements from BWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IGSCC	Crack initiation & growth	Access Hole Cover	Alloy 600	G-2, G-5, G-9, G-11, G-22, G-24, G-27, G-28, G-29, G-33, G-34, S-3, S-4, S-6, S-17, S-22, S-25, S-26 to S-29, S-31, S-32, S-38, S-49, S-54, S-55	GESIL 462S1[1] recommends volumetric inspections, implementation is plant specific, & recommended repair is to attach reinforcement hardware.	Recommendations of GESIL 462S1[1] & safety analysis are current & effective inspection programs for detection & evaluation-repair of access hole covers.
		Core Shroud Head Bolts	SS, Alloy 600		GESIL 433[2] recommends UT examination during outages, implementation is plant specific, & replacement is with crevice-free design.	Recommendations of GESIL 433[2] & replacement with crevice-free design are current & effective inspection programs for detection & evaluation-replacement of core shroud head bolts.
		Control Blade	SS		GESIL 157 routine replacement,[3] operational parameter monitoring, inspection, evaluation, & replacement.[4]	Routine replacement & operational parameter monitoring are current & effective programs for detection & evaluation-replacement of control blades.
		Control Rod Drive (CRD) Housing	SS		ASME Sect. XI[5] requires volumetric exam. of welds & VT-2 of pressure retaining boundary & system leakage & hydro-static tests.	ASME Sect. XI, Subsect. IWB,[5] exam. categories B-O & B-P is current & effective program for detection & evaluation-repair-replacement of CRD housing.
		Core Spray Sparger	SS		NRC Bulletin 80-13[6] recommends visual inspection during refueling outages; analytical evaluation; & repair.	NRC Bulletin 80-13[6] & safety analysis are effective inspection programs for detection, evaluation-repair of core spray sparger.
		Intermediate Range Monitor/ Source Range Monitor (IRM/SRM) Dry Tubes	SS		GESIL 409, Rev. 1[7] recommends visual inspection; leakage monitoring; replacement is with crevice-free design; & resistant material.	Recommendations of GESIL 409,[7] leakage monitoring, & replacement with crevice-free design, are effective inspection programs for detection & evaluation-replacement of dry tubes.

Table B8. Brief summary of technical information and NUMARC/NRC agreements from BWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IGSCC	Crack initiation & growth	Core Plate	SS	G-12, G-14, G-30, S-11, S-30, S-33, S-44	Current practices to be enhanced. Select plant-specific aging management program.	Select plant specific management comprising of qualified inspection & monitoring; water chemistry control;[8] evaluation-repair-replacement.
		Core Shroud	SS			
		Core Spray Int. Piping	SS			
		Top Guide	SS	Open issue S-5*		

* Items concerning chapter six were not the focus of the NRC staff review. This includes comment number S-5.

Table B8. Brief summary of technical information and NUMARC/NRC agreements from BWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IASCC	Crack initiation & growth	Access Hole Cover	Alloy 600	G-19, G-20, G-32, S-9, S-23, S-34	Non-significant	Total fast neutron fluence within the license renewal term is non-significant for IASCC because it is less than 5×10^{20} n/cm^2 for low-stressed components & less than 1×10^{20} n/cm^2 for high-stressed components.[9,10]
		CRD Housing	SS			
		Core Plate*	SS			
		Core Shroud Head Bolts	SS, Alloy 600			
		Core Spray Int. Piping	SS			
		Core Spray Sparger	SS			
		Jet Pump	SS, *(CASS)*, Alloy 600, X 750			
		LPRM	SS			
		Orificed Fuel Support*	SS, CASS			
IASCC	Crack initiation & growth	Control Blade	SS	Same as the comments for IGSCC G-2, G-5, through S-32, S-54 & S-52, S-56	GESIL 157 routine replacement,[3] operational parameter monitoring, inspection, evaluation, & replacement.[4]	Routine replacement & operational parameter monitoring are current & effective programs for detection & evaluation-replacement of control blades.
		IRM/SRM Dry Tubes	SS		GESIL 409, Rev. 1[7] recommends visual inspection; leakage monitoring; replacement is with crevice-free design; & resistant material.	Recommendations of GESIL 409,[7] leakage monitoring, & replacement with crevice-free design, are current & effective inspection programs for detection & evaluation-replacement of dry tubes.

* For these components, the total fluence can not be accurately determined from the available information, but it is likely to exceed 5×10^{20} n/cm^2. Also, the effectiveness of HWC in the presence of high flux has not been established.

Table B8. Brief summary of technical information and NUMARC/NRC agreements from BWR vessel internals industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IASCC	Crack initiation & growth	Core Shroud	SS	G-12, G-14, G-30, S-11, S-30, S-33, S-44	Current practices are to be enhanced.	Select plant specific aging management plan that may include qualified visual and/or volumetric inspection of susceptible locations; SCC monitoring; analytical evaluation;[5] enhanced water chemistry control;[8] appropriate repair methods for irradiated materials; & replacement.
		Top Guide	SS	Open issue S-5*		

* Items concerning chapter six were not the focus of the NRC staff review. This includes comment number S-5.

Table B8. Brief summary of technical information and NUMARC/NRC agreements from BWR vessel internals industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron Irradiation Embrittle-ment	Loss of fracture toughness	Access Hole Cover	Alloy 600	G-19, G-20, G-32 S-9, S-18, S-50, S-51	Non-significant	Although neutron irradiation causes a decrease in fracture toughness for SS & Ni-alloy components, the fracture tough-ness levels remain adequate even at end-of-life fluence levels because the applied stresses are low.
		CRD Housing	SS			
		Core Plate	SS			
		Core Shroud Head Bolts	SS, Alloy 600			
		Core Spray Int. Piping	SS			
		Core Spray Sparger	SS			
		LPRM	SS			
		Control Blade	SS			
		Core Shroud	SS			
		IRM/SRM Dry Tubes	SS			
		Jet Pump	SS, *(CASS)*, Alloy 600, X 750			
		Orificed Fuel Support	SS, CASS			
		Top Guide	SS			

Table B8. Brief summary of technical information and NUMARC/NRC agreements from BWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Thermal Embrittlement	Loss of fracture toughness	Access Hole Cover	Alloy 600	G-19, G-20, G-32 S-9, S-10, S-19, S-20	Non-significant	Wrought SS & Ni-alloys are not susceptible to thermal aging embrittlement.
		Control Blade	SS			
		CRD Housing	SS			
		Core Plate	SS			
		Core Shroud	SS			
		Core Shroud Head Bolts	SS, Alloy 600			
		Core Spray Int. Piping	SS			
		Core Spray Sparger	SS			
		IRM/SRM Dry Tubes	SS			
		Jet Pump	SS, Alloy 600, X 750			
		LPRM	SS			
		Orificed Fuel Support	SS			
		Top Guide	SS			
Thermal Embrittlement	Loss of fracture toughness	Orificed Fuel Support	CASS	G-19, G-20, G-32 S-9, S-19, S-20	Non-significant	Not subjected to stress levels of sufficient magnitude.
		Jet Pump	*(CASS)*			

Table B8. Brief summary of technical information and NUMARC/NRC agreements from BWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion	Loss of matertial	Access Hole Cover	Alloy 600	G-19, G-20, G-32 S-9	Non-significant	Corrosion rates of all materials are very low.
		Control Blade	SS			
		CRD Housing	SS			
		Core Spray Int. Piping	SS			
		Core Spray Sparger	SS			
		IRM/SRM Dry Tubes	SS			
		Jet Pump	SS, *(CASS)*, Alloy 600, X 750			
		LPRM	SS			
		Top Guide	SS			
		Orificed Fuel Support	SS, CASS			
		Core Plate	SS			
		Core Shroud	SS			
		Core Shroud Head Bolts	SS, Alloy 600			

Table B8. Brief summary of technical information and NUMARC/NRC agreements from BWR vessel internals industry report

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Erosion/ Corrosion (E/C)	Wall thinning	Access Hole Cover	Alloy 600	G-19 to G-21, G-32, S-9,, S-47, S-53	Non-significant	SS and Ni-alloys are resistant to E/C, and/or low flow range, and implementation of HWC in which requires oxygen addition the feedwater line to limit E/C.
		Control Blade	SS			
		CRD Housing	SS			
		Core Plate	SS			
		Core Shroud	SS			
		Core Shroud Head Bolts	SS, Alloy 600			
		Core Spray Int. Piping	SS			
		Core Spray Sparger	SS			
		IRM/SRM Dry Tubes	SS			
		LPRM	SS			
		Orificed Fuel Support	SS, CASS			
		Top Guide	SS			
		Jet Pump	SS, *(CASS)*, Alloy 600, X 750			

Table B8. Brief summary of technical information and NUMARC/NRC agreements from BWR vessel internals industry report

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number[a]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	Access Hole Cover	Alloy 600	G-19, G-20, G-28, G-32, S-4, S-9, Open issues G-4, S-7	Unresolved issue. *NUMARC proposal:* Non-significant ARDM. *NRC proposal:* Not acceptable without detailed information. Plant specific or typical evaluations based on ASME Sect. III, of all safety-related internal components should show a usage of <0.7 for the 60-y lifetime.	*NUMARC basis:* Cyclic stresses are minimal or absent such that ASME Sect. III, NB-3200[11] analysis is not required; LPRM are replaced due to limited operating life.
		Core Plate	SS			
		Top Guide	SS			
		Control Blade	SS			
		Core Shroud	SS			
		Core Shroud Head Bolts	SS, Alloy 600			
		Core Spray Int. Piping	SS			
		Core Spray Sparger	SS			
		IRM/SRM Dry Tubes	SS			
		Orificed Fuel Support	SS, CASS			
		LPRM	SS			
Fatigue	Cumulative fatigue damage	CRD Housing	SS	G-22 G-24, G-33, G-34, S-3, S-26 S-27	Unresolved issue *NUMARC proposal:* ASME Sect. XI[5] requires volumetric exam. of welds & VT-2 of pressure retaining boundary & system leakage & hydrostatic tests. *NRC proposal:* Not acceptable without detailed information.	*NUMARC basis:* ASME Sect. XI, Subsect. IWB,[5] exam. categories B-O & B-P, & recommendations of NP-5181M & NP-5836M[12,13] are current & effective programs for detection & evaluation-repair-replacement of CRD housing.
		Jet Pump	SS, *(CASS)*, Alloy 600, X 750	Open issues G-4, S-7	Unresolved issue *NUMARC proposal:* Monitoring pump performance, inspection, evaluation, & replacement. *NRC proposal:* Not acceptable without detailed information.	*NUMARC basis:* Periodic inspection & continuous monitoring of performance by plant instrumentation are current & effective programs for detection & evaluation-replacement of jet pumps.

[a] The following comments were not included in the table because they deal with clarification or modification of contents of the IR: G-3, G-7, G-10, G-13, G-15, G-17, G-18, G-35, S-1, S-15, S-21, S-24, S-31, and S-39. The following comments were also excluded because they deal with scope of the IR: G-8, G-16, G-25, G-26, S-13, S-16, S-36, and S-41 to S-43.

REFERENCES:

1. GE SIL 462S1, "Shroud Support Access Hole Cover Cracks," GE Service Information Letter, GE Nuclear Energy, February 22, 1984.
2. GE SIL 433, "Shroud Head Bolt Cracks," GE Service Information Letter, GE Nuclear Energy, February 7, 1986.
3. GE SIL 157 Rev. 2, "Control Blade Lifetime," GE Service Information Letter, GE Nuclear Energy, September, 1981.
4. GE RDE-21-0986, "REPORT (REV 05) on Acceptance Criteria for Control Rod Cracks," GE Nuclear Energy, September, 1986.
5. ASME B & PV Code - "Boiler and Pressure Vessel Code of Design and Construction Practices," Section XI: "Rules for In-Service Inspection (ISI) of Nuclear Power Plant Components," American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017.
6. IE Bulletin 80-013, "Cracking in Core Spray Spargers," US NRC, Office of Inspection and Enforcement, May 12, 1980.
7. GE SIL 409 Rev. 1, "Inspection of SRM/IRM Dry Tubes," GE Service Information Letter, GE Nuclear Energy, July 31, 1986.
8. EPRI NP-4946SR, "BWR Normal Water Chemistry Guidelines (1986 Revision)," Electric Power Research Institute, Palo Alto, CA, 1986.
9. "Material Aspects of BWR Plant Life Extension," B. M. Gordon and G. M. Gordon, *Nuclear Engineering and Design*, Vol. 98, pp 109-121, 1987.
10. "Irradiation Assisted Stress Corrosion Cracking as a Factor in Nuclear Power Plant Aging," A. J. Jacobs and G. P. Wozadlo, *Proc. of the International Conference on Nuclear Power Plant Aging Availability Factor and Reliability Analysis*, American Society of Metals, August 1985.
11. ASME B & PV Code - "Boiler and Pressure Vessel Code of Design and Construction Practices," Section III: "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017.
12. EPRI NP-5181M, "BWR Pilot Plant Life Extension Study at the Monticello Plant: Phase I," Electric Power Research Institute, Palo Alto, CA, May 1987.
13. EPRI NP-5836M, "BWR Pilot Plant Life Extension Study at the Monticello Plant: Interim Phase II," Electric Power Research Institute, Palo Alto, CA, October 1988.

LIST OF BWR VESSEL INTERNAL COMPONENTS:

BWR VESSEL INTERNALS

Access Hole Cover
Control Blades
Control Rod Drive (CRD) Housing
Core Plate
Core Shroud
Core Shroud Head Bolt
Core Spray Internal Piping
Core Spray Sparger
Intermediate Range Monitor (IRM) Dry Tubes
Jet Pump
Low Power Range Monitor (LPRM)
Orificed Fuel Support
Source Range Monitor (SRM) Dry Tubes
Top Guide

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
General	Age related degradation effects	Structures	Concrete and steel	Open issues: G-3, G-4, G-5, S-73, S-76	Unresolved issue (One-time inspection) *NUMARC proposal*: Resolution of the effects of ARD mechanisms is based upon the review/evaluation of plant-specific features, including appropriate CLB documents/ information. Inspection of structures is not required if a set of acceptance criteria (including a review of plant performance history, to assure that contradictory evidence does not exist) are satisfied. *NRC proposal*: A one time focussed plant-specific inspection of structures is proposed as part of an applicant's activities for identification and resolution of potential ARDM.	*NUMARC basis*: The ARD mechanisms are evaluated for significance using the available research & industry data. If acceptance criteria (including a review of plant performance history, to assure that contradictory evidence does not exist) are satisfied, then the inspection for that mechanism/ component combination is not needed.

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Freeze-thaw	Scaling, cracking & spalling	1. BWR Reactor Building*	Concrete: F, ExA, & ExB	G-2, G-7, G-8, G-11, G-12, G-13, G-16, G-18, S-6, S-7, S-21, S-49, S-51, S-52, S-77	For Class 1 concrete structures that meet the basis requirements, freeze-thaw is non-significant ARDM.§	Freeze-thaw is non-significant[1] for Class 1 concrete structures located in a geographic regions of negligible weathering conditions (weathering index <100 day-inch/yr); and if located in severe (weathering index >500 day-inch/yr) or moderate (100-500 day-inch/yr) weathering conditions the concrete mix design meets the air content & water-to-cement ratio requirements of ACI 318-63[2] or ACI 349-85.[3]
		PWR Shield Building				
		Control Room/Building				
		2. BWR Reactor Building with Steel Superstructure**				
		3. Auxiliary Building				
		Diesel Generator Building				
		Radwaste Building				
		Turbine Building†				
		Aux FW Pump House				
		Utility/Piping Tunnels				
		5. Fuel Storage Facility				
		Refueling Canal				
		7. Concrete Tanks				
		9. BWR Unit Vent Stack	Concrete: F, GA, & GB			
		6. Intake Structure	Concrete: ExA, ExB, F, & InS			
		Cooling Tower				
		Spray Pond				
		8. Steel Tanks	Concrete: F			

* Reinforced concrete structures up to and including the roof.
**Upper portion (above the refueling floor elevation) being a steel-framed structure with metal siding and roof panels.
† Part of turbine building contains Class 1 components.
§ See also NUMARC/NRC agreement concerning Freeze-thaw page B-46 (Table B4) and comment S-10, page B-29 (Table B3).

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Leaching of calcium hydroxide	Increase of porosity & permeability	1. BWR Reactor Building	Concrete: F, ExA, & ExB	G-2, G-7, G-8, G-11, G-12, G-13, G-16, G-18, S-6, S-8, S-21, S-77	For Class 1 concrete structures that meet the basis requirements, leaching of calcium hydroxide is non-significant ARDM.	Leaching of calcium hydroxide is non-significant for Class 1 concrete structures not exposed to flowing water; and for structures that are exposed to flowing water but are constructed using the guidance of ACI 201.2R-67[4] to ensure dense, well-cured concrete with low permeability and control cracking through proper arrangement & distribution of reinforcement.
		PWR Shield Building				
		Control Room/Building				
		2. BWR Reactor Building with Steel Superstructure				
		3. Auxiliary Building				
		Diesel Generator Building				
		Radwaste Building				
		Turbine Building				
		Aux FW Pump House				
		Utility/Piping Tunnels				
		5. Fuel Storage Facility				
		Refueling Canal				
		6. Intake Structure				
		Cooling Tower				
		Spray Pond				
		7. Concrete Tanks				
		9. BWR Unit Vent Stack	Concrete: F, GA, & GB			
		8. Steel Tanks	Concrete: F			

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging–Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Aggressive chemical attack	Increase of porosity & permeability, cracking, & spalling	1. BWR Reactor Building	Concrete: ExA, InW, & InS; & Masonry block walls	G-2, G-7, G-8, G-11, G-12, G-13, G-16, G-18, S-6, S-21, S-53, S-54, S-77	For Class 1 concrete structures that meet the basis requirements, aggressive chemical attack is non-significant ARDM.	Degradation caused by aggressive chemical attack is non-significant for Class 1 concrete structures not exposed to aggressive environment (pH <5.5), or to chloride or sulfate solutions beyond defined limits (>500 ppm chloride,[2] and 1500 ppm sulfate);[5] or if exposed to ground water that exceeds the pH, chloride, sulfate limits the exposure is for intermittent periods only.
		PWR Shield Building				
		Control Room/Building				
		2. BWR Reactor Building with Steel Superstructure				
		3. Auxiliary Bldg				
		Diesel Gen Bldg				
		Radwaste Bldg				
		Turbine Bldg				
		Utility/Piping Tunnels				
		Aux FW Pump House				
		5. Fuel Storage Facility				
		Refueling Canal				
		3. Switchgear Room	Concrete: InW & InS; & Masonry block walls			
		4. Containment Internal Structures	Concrete: InW & InS			
		6. Intake Structure	Masonry block walls			
		Cooling Tower				
		7. Concrete Tanks	Concrete: Int & ExA			
		9. BWR Unit Vent Stack	Concrete: GA			

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging–Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Aggressive chemical attack	Increase of porosity, & permeability, cracking & spalling	1. BWR Reactor Building	Concrete: F & ExB	G-6, G-10, G-14, G-15, S-4, S-25, S-74, S-78, S-80 Open issue G-19*	Aggressive chemical attack can cause potentially significant degradation of below-grade portions of class 1 concrete structures, select plant-specific aging management program.*	Select plant-specific aging management program that could include, monitoring of ground water chemistry, inspection, & testing.[6]
		PWR Shield Building				
		Control Room/Building				
		2. BWR Reactor Building with Steel Superstructure				
		3. Auxiliary Building				
		Diesel Generator Building				
		Radwaste Building				
		Turbine Building				
		Aux FW Pump House				
		Utility/Piping Tunnels				
		5. Fuel Storage Facility				
		Refueling Canal				
		7. Concrete Tanks	Concrete: F & ExB			
		8. Steel Tanks	Concrete: F			
		9. BWR Unit Vent Stack	Concrete: F & GB			
		6. Intake Structure	Concrete: ExA, ExB F, & InS	G-6, G-10, G-14, G-15, S-4, S-25, S-74, S-78, S-80	Reg. Guide 1.127[7] requires inspection at periodic intervals not to exceed 5 yr & includes engineering data compilation & inspection & evaluation of concrete surfaces, structural cracking, settlement, & water passage.	The periodic inspections including engineering data compilation & onsite inspection program outlined in Regulatory Guide 1.127[7] are current and effective programs for managing degradation of Group 6 concrete structures from aggressive chemical attack.
		Cooling Tower				
		Spray Pond				

* Items concerning chapter six were not the focus of the NRC staff review. This includes comment number G–19.

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Reaction with aggregates	Expansion & cracking	1. BWR Reactor Building	Concrete: ExA, ExB, InW,* InS, & F; and Masonry block walls	G-2, G-7, G-8, G-11, G-12, G-13, G-16, G-18, S-6, S-9, S-21, S-54, S-77	For Class 1 concrete structures that meet the basis requirements, reaction with aggregates is non-significant ARDM.§	Reactions with aggregates are non-significant for Class 1 concrete structures constructed either from aggregates taken from geographic regions other than those known to yield aggregates suspected of or known to cause alkali-aggregate reactions;[8] or from aggregates that were investigated, tested, & subject to petrographic exam. conducted in accordance with ASTM C295[9] or ASTM C227,[10] which showed that the aggregates are non-reactive; or if the aggregate was examined & found potentially reactive, the provisions of ACI 201.2R-67[4] were followed.
		PWR Shield Building				
		Control Room/Building				
		2. BWR Reactor Building with Steel Superstructure				
		3. Auxiliary Building				
		Diesel Generator Building				
		Radwaste Building				
		Turbine Building				
		Aux FW Pump House				
		Utility/Piping Tunnels				
		5. Fuel Storage Facility				
		Refueling Canal				
		6. Intake Structure				
		Cooling Tower				
		Spray Pond				
		3. Switchgear Room	Concrete: InW & InS; & Masonry block walls			
		4. Containment Internal Structures	Concrete: InW & InS			
		7. Concrete Tanks	Concrete: F, Int, & Ext			
		8. Steel Tanks	Concrete: F			
		9. BWR Unit Vent Stack	Concrete: F, GA, & GB			

* No interior walls InW for Intake Structures.
§ See also NUMARC/NRC agreement concerning Reaction with Aggregates page B-49 (Table B4) and comment S-12, page B-31 (Table B3).

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging–Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion of Embedded Steel	Cracking, spalling, loss of bond, & loss of material	1. BWR Reactor Building	Concrete:* ExA, InW, & InS; & Masonry block walls	G-2, G-7, G-8, G-11, G-12, G-13, G-16, G-18, S-6, S-20, S-21, S-22, S-23, S-55, S-56, S-77	For Class 1 concrete structures that meet the basis requirements, corrosion of embedded steel or rebar is non-significant ARDM.	Degradation due to corrosion of embedded & reinforcing steel is non-significant[11] for Class 1 concrete structures (above or below grade) not exposed to aggressive environment (pH <11.5 or chlorides >500 ppm);[12] or for structures exposed to aggressive environment but have low water-to-cement ratio (0.35-0.45), adequate air entrainment (3-6%), low permeability, and are designed in accordance with ACI 318-63[2] or ACI 349-85.[3]
		PWR Shield Building				
		Control Room/Building				
		2. BWR Reactor Building with Steel Superstructure				
		3. Auxiliary Building				
		Diesel Generator Building				
		Radwaste Building				
		Turbine Building				
		Aux FW Pump House				
		Utility/Piping Tunnels				
		5. Fuel Storage Facility				
		Refueling Canal				
		3. Switchgear Room	Concrete: InW & InS; & Masonry block walls			
		4. Containment Internal Structures	Concrete:* InW & InS			
		6. Intake Structure	Masonry block walls*			
		Cooling Tower				
		7. Concrete Tanks	Concrete:* Int & ExA			
		9. BWR Unit Vent Stack	Concrete:* GA			

* Embedded CS & reinforcing CS (rebar) in concrete structures or masonry block walls

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging–Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion of Embedded Steel	Cracking, spalling, loss of bond, & loss of material	1. BWR Reactor Building	Concrete:* F & ExB	G-6, G-10, G-14, G-15, S-25, S-30, S-78, S-79, S-80 Open issue G-19†	Corrosion of rebar may cause potentially significant degradation of below-grade portions of class 1 concrete structures, select plant specific aging management program.§	Select plant-specific aging management program that could include, monitoring of ground water chemistry, inspection, & testing.[6]
		PWR Shield Building				
		Control Room/Building				
		2. BWR Reactor Building with Steel Superstructure				
		3. Auxiliary Building				
		Diesel Generator Building				
		Radwaste Building				
		Turbine Building				
		Utility/Piping Tunnels				
		Aux FW Pump House				
		5. Fuel Storage Facility				
		Refueling Canal				
		7. Concrete Tanks	Concrete:* F & Ext			
		8. Steel Tanks	Concrete:* F			
		9. BWR Unit Vent Stack	Concrete:* F & GB			
		6. Intake Structure	Concrete:* ExA, ExB F, & InS	G-6, -10 G-14, G-15, S-25, S-30, S-74, S-78 to S-80	Reg. Guide 1.127[7] requires inspection at periodic intervals not to exceed 5 yr & includes engineering data compilation & inspection & evaluation of concrete surfaces, structural cracking, settlement, & water passage.	Periodic inspections including engineering data compilation & onsite inspection program outlined in RG 1.127[7] are current & effective programs for managing degradation of Group 6 concrete structures from corrosion of embedded steel.
		Cooling Tower				
		Spray Pond				

* Embedded CS & reinforcing CS (rebar) in concrete structures or masonry block walls
§ See also NUMARC/NRC agreement concerning Corrosion of Embedded Steel page B-52 (Table B4) and comment S-42, page B-36 (Table B3).
† Items concerning chapter six were not the focus of the NRC staff review. This includes comment number G-19.

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion	Loss of material	1. BWR Reactor Building PWR Shield Building Control Room/Building 2. BWR Reactor Building with Steel Superstructure 3. Auxiliary Building Diesel Generator Building Radwaste Building Turbine Building Switchgear Room Aux FW Pump House Utility/Piping Tunnels 4. Containment Internal Structures	Structural steel, and Jet impingement barriers	G-6, G-10, G-14, G-15, S-24, S-25, S-74, S-78, S-80 Open issue G-19*	Periodic VT-3 inspection and maintenance practices of coated and uncoated surfaces, preventive measures, and coating repair & replacement, are effective programs to manage corrosion of accessible structural steel. Corrosion is potentially significant for inaccessible structural steel, select plant specific aging management program.	Select plant-specific aging management program that could include, monitoring of ground water chemistry, inspection,& testing.[13-15]
		5. Fuel Storage Facility Refueling Canal	Structural steel			
		7. Concrete Tanks 8. Steel Tanks	CS			
Corrosion	Loss of material	6. Intake Structure Cooling Tower Spray Pond	Structural steel	Same as above	Reg. Guide 1.127[7] requires inspection at periodic intervals not to exceed 5 yr & includes engineering data compilation & inspection & evaluation of concrete surfaces, structural cracking, settlement, & water passage.	Periodic inspections including engineering data compilation & onsite inspection program outlined in RG 1.127[7] are current & effective programs for managing degradation of Group 6 concrete structures from corrosion of embedded steel.

* Items concerning chapter six were not the focus of the NRC staff review. This includes comment number G-19.

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging–Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion	Loss of material	2. BWR Reactor Building with Steel Superstructure	Metal siding, metal roofing	G-6, G-10, G-14, G-15, S-5, S-24, S-25, S-28, S-74, S-78, S-80 Open issue: S–5	Unresolved issue *NUMARC proposal:* Pressure retaining capability testing, frequency, & acceptance criteria in accordance with plant's technical specification. *NRC proposal:* Demonstrate how building pressurization test is effective in timely detection of corrosion degradation.	*NUMARC basis:* Routine pressure retaining capability testing is effective in verifying the integrity & for timely detection, corrosion prevention/mitigation measures for metal siding & roof decking in BWR reactor buildings.
IGSCC & Crevice corrosion	Crack initiation & growth, loss of material	5. Fuel Storage Facility Refueling Canal	SS liner	G-6, G-10, G-14, G-15, S-25, S-74, S-78, S-80	Current leakage detection & inventory monitoring systems provide timely means of identifying, monitoring, & repair of liner degradation.	Periodic monitoring of the leak chase system drain lines &/or the leak detection sump are effective methods for early detection-repair of leaks in spent fuel pool/refueling canal liners.
IGSCC & Crevice corrosion	Crack initiation & growth, loss of material	4. Containment Internal Structures 7. Concrete Tanks 8. Steel Tanks	Wet well liner (Mark II & III BWRs) SS liner	Same as above & Open issue: G-19	IGSCC & crevice corrosion may affect the ability of SS liners to perform their safety function; select plant specific aging management program.*	Select plant specific aging management program.

* Some instances may require benchmarking and trending of degradation data taken periodically to evaluate extended life operability.

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion of Steel Piles	Loss of material	1. BWR Reactor Building	Steel Piles	S-15, S-66	Non-significant ARDM.	Steel piles driven in undisturbed soils have been unaffected by corrosion & those driven in disturbed soil experience minor to moderate corrosion to a small area of metal.[16,17]
		PWR Shield Building				
		Control Room/Building				
		2. BWR Reactor Building with Steel Superstructure				
		3. Auxiliary Building				
		Diesel Generator Building				
		Radwaste Building				
		Turbine Building				
		Switchgear Room				
		Aux FW Pump House				
		Utility/Piping Tunnels				
		5. Fuel Storage Facility				
		Refueling Canal				
		6. Intake Structure				
		Cooling Tower				
		Spray Pond				
		7. Concrete Tanks				
		8. Steel Tanks				
		9. BWR Unit Vent Stack				

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Elevated temperature	Loss of strength & modulus	1. BWR Reactor Building	Concrete including rebar: InW, InS; Masonry block walls	G-2, G-7, G-8, G-11, G-12, G-13, G-16, G-18, S-6, S-10, S-21, S-57, S-77	Non-significant ARDM.	Degradation from exposure to elevated temperatures is non-significant for Class 1 concrete structures maintained at operating temperatures <66°C (150°F) and local area temperatures <93°C (200°F);[18,19] or for structures that operate above these limits, plant-specific justification is provided in terms of concrete strength at elevated temperatures or from application of special provisions described in ACI 349-85[3] or justified equivalent. Degradation from exposure to elevated temperatures is non-significant for reinforcing steel (rebar) used in Class 1 concrete maintained at temperatures <316°C (600°F).[20]
		PWR Shield Building				
		Control Room/Building				
		2. BWR Reactor Building with Steel Superstructure				
		3. Auxiliary Building				
		Diesel Generator Building				
		Radwaste Building				
		Turbine Building				
		Switchgear Room				
		Utility/Piping Tunnels				
		Aux FW Pump House				
		5. Fuel Storage Facility				
		Refueling Canal				
		4. Containment Internal Structures	Concrete including rebar: InW & InS			
		6. Intake Structure	Masonry block walls			
		Cooling Tower				
		7. Concrete Tanks	Concrete including rebar: Int & Ext			
		8. Steel Tanks	Concrete: F			
		9. BWR Unit Vent Stack	Concrete including rebar: GA			

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Elevated temperature	Loss of strength & modulus	1. BWR Reactor Building PWR Shield Building Control Room/Building 3. Auxiliary Building Diesel Generator Building Radwaste Building Turbine Building Switchgear Room Aux FW Pump House Utility/Piping Tunnels 6. Intake Structure Cooling Tower Spray Pond	Structural steel	G-2, G-7, G-8, G-11, G-12, G-13, G-16, G-18, S-6, S-16, S-21, S-77	Non-significant ARDM.	*NUMARC basis:* Degradation from exposure to elevated temperatures is non-significant for Class 1 structural steel components, metal sidings, or liners maintained at temperatures <371°C (700°F).
		2. BWR Reactor Building with Steel Superstructure	Structural steel, metal siding			
		4. Containment Internal Structures 5. Fuel Storage Facility Refueling Canal	Structural steel, SS liner			
		7. Concrete Tanks	CS & SS liner			
		8. Steel Tanks	CS & SS			

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Irradiation of Concrete	Loss of strength & modulus	1. BWR Reactor Building PWR Shield Building Control Room/Building 2. BWR Reactor Building with Steel Superstructure 3. Auxiliary Building Diesel Generator Building Radwaste Building Turbine Building Switchgear Room Aux FW Pump House Utility/Piping Tunnels 5. Fuel Storage Facility Refueling Canal	Concrete including rebar: InW, InS; Masonry block walls	G-2, G-7, G-8, G-11, G-12, G-13, G-16, G-18, S-6, S-11, S-14, S-21, S-58, S-63, S-64, S-65, S-77	Non-significant ARDM.	The neutron fluence levels & maximum integrated gamma doses incurred by Class 1 concrete structures, including rebars, for both the current & license renewal period do not exceed the level at which measurable degradation of concrete strength properties occurs (5×10^{19} n/cm^2 & 10^{10} rads, respectively).[21,22]
		4. Containment Internal Structures	Concrete including rebar: InW & InS			
		7. Concrete Tanks	Concrete including rebar: Int & Ext			
		8. Steel Tanks	Concrete: F			
		9. BWR Unit Vent Stack	Concrete including rebar: GA			

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Irradiation of Steel	Loss of fracture toughness	1. BWR Reactor Building	Structural steel	G-2, G-7, G-8, G-11, G-12, G-13, G-16, G-18, S-6, S-17, S-21, S-77	Non-significant ARDM.	The total neutron fluence levels incurred by Class 1 structural steel, metal siding, & liners do not exceed the level at which measurable degradation in mechanical properties is observed.
		PWR Shield Building				
		Control Room/Building				
		3. Auxiliary Building				
		Diesel Generator Building				
		Radwaste Building				
		Turbine Building				
		Switchgear Room				
		Aux FW Pump House				
		Utility/Piping Tunnels				
		2. BWR Reactor Building with Steel Superstructure	Structural steel, metal siding			
		4. Containment Internal Structures	Structural steel, SS liner			
		5. Fuel Storage Facility				
		Refueling Canal				
		7. Concrete Tanks	CS & SS liner			
		8. Steel Tanks	CS & SS			

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Creep*	Deformation	1. BWR Reactor Building	Concrete: ExA, ExB, InW, InS, & F; and Masonry block walls	G-2, G-7, G-8, G-11, G-12, G-13, G-16, G-18, S-6, S-12, S-21, S-60, S-77	Non-significant ARDM.	Creep experienced by Class 1 reinforced concrete structures is insignificant because the actual compressive stresses experienced by the structures are generally low.[23]
		PWR Shield Building				
		Control Room/Building				
		2. BWR Reactor Building with Steel Superstructure				
		3. Auxiliary Building				
		Diesel Generator Building				
		Radwaste Building				
		Turbine Building				
		Switchgear Room				
		Aux FW Pump House				
		Utility/Piping Tunnels				
		5. Fuel Storage Facility				
		Refueling Canal				
		6. Intake Structure				
		Cooling Tower				
		Spray Pond				
		4. Containment Internal Structures	Concrete: InW & InS			
		7. Concrete Tanks	Concrete: F, Int, & Ext			
		8. Steel Tanks	Concrete: F			
		9. BWR Unit Vent Stack	Concrete: F, GA, & GB			

* This review applies to Class 1 reinforced concrete structures only. Prestressed concrete structures may be subjected to more pronounced creep and relaxation effects. Creep degradation of prestressed Class 1 concrete structures is outside the scope of the IR and must be evaluated on a plant-specific basis.

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging–Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Shrinkage	Cracking	1. BWR Reactor Building	Concrete: ExA, ExB, InW,* InS, & F; and Masonry block walls	G-2, G-7, G-8, G-11, G-12, G-13, G-16, G-18, S-6, S-13, S-21, S-61, S-77	Non-significant ARDM.	Most concrete shrinkage occurs in the first five years of a structure's life,[23] it is not a serious degradation mechanism after five years.
		PWR Shield Building				
		Control Room/Building				
		2. BWR Reactor Building with Steel Superstructure				
		3. Auxiliary Building				
		Diesel Generator Building				
		Radwaste Building				
		Turbine Building				
		Switchgear Room				
		Aux FW Pump House				
		Utility/Piping Tunnels				
		5. Fuel Storage Facility				
		Refueling Canal				
		6. Intake Structure				
		Cooling Tower				
		Spray Pond				
		4. Containment Internal Structures	Concrete: InW & InS			
		7. Concrete Tanks	Concrete: F, Int, & Ext			
		8. Steel Tanks	Concrete: F			
		9. BWR Unit Vent Stack	Concrete: F, GA, & GB			

* No interior walls InW for Intake Structures.

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Abrasion & cavitation	loss of material	6. Intake Structure* Cooling Tower Spray Pond	Concrete: ExA, ExB F, & InS	G-6, G-10, G-14, G-15, S-25, S-74, S-78, S-80	Reg. Guide 1.127[7] requires inspection at periodic intervals not to exceed 5 ys & includes engineering data compilation & inspection & evaluation of concrete surfaces, structural cracking, settlement, & water passage.	The periodic inspections including engineering data compilation & onsite inspection program outlined in Regulatory Guide 1.127[7] are current and effective programs for managing degradation of Group 6 concrete structures from aggressive chemicals.
Restraint, Shrinkage, Creep, & Aggressive Environment	Cracking of masonry block walls	1. BWR Reactor Building PWR Shield Building Control Room/Building 2. BWR Reactor Building with Steel Superstructure 3. Auxiliary Building Diesel Generator Building Radwaste Building Turbine Building Switchgear Room Aux FW Pump House Utility/Piping Tunnels 5. Fuel Storage Facility 6. Intake Structure Cooling Tower	Masonry block walls	G-6, G-10, G-14, G-15, S-4, S-25, S-74, S-78, S-80	Bulletin 80-11[24] requires identification of masonry walls in close proximity to or have attachments from safety-related piping or equipment & reevaluation of design adequacy & construction practices; & Info. Notice 87-67[25] proposed plant-specific corrective actions.	Inspection requirements imposed in I&E Bulletin No. 80-11[24] & plant-specific monitoring requirements proposed by Information Notice No. 87-67[25] are current & effective programs for cracking of masonry block walls.

* Non-significant ARD mechanism for other Class 1 structures.

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging–Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Cathodic Protection Current	Cathodic protection effect on bond strength	1. BWR Reactor Building	Concrete: ExA, ExB, InW,* InS, & F	S-19	Non-significant	Cathodic protection systems are designed to operate at ≈21.5 mA/m^2 (≈2 mA/ft^2) of steel surface, a level well below the value of 10,764 mA/m^2 (1000 mA/ft^2) at which concrete can soften at the reinforcing bar surface.[28]
		PWR Shield Building				
		Control Room/Building				
		2. BWR Reactor Building with Steel Superstructure				
		3. Auxiliary Building				
		Diesel Generator Building				
		Radwaste Building				
		Turbine Building				
		Switchgear Room				
		Aux FW Pump House				
		Utility/Piping Tunnels				
		5. Fuel Storage Facility				
		Refueling Canal				
		6. Intake Structure				
		Cooling Tower				
		Spray Pond				
		4. Containment Internal Structures	Concrete: InW & InS			
		7. Concrete Tanks	Concrete: F, Int, & Ext			
		8. Steel Tanks	Concrete: F			
		9. BWR Unit Vent Stack	Concrete: F, GA, & GB			

* No interior walls InW for Intake Structures.

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging–Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	1. BWR Reactor Building	Concrete: ExA, ExB, InW, InS, & F; Structural steel	S-18, S-70, S-71, S-72	Non-significant	Class 1 concrete structures subjected to repeated load are designed in accordance with ACI 318[2] or an equivalent code, which limits the max. design stress level to <50% of static strength (working stress design); concrete structures can resist $>10^7$ cycles of loading in this stress range.[26] Class 1 steel structures subjected to repeated loading are designed in accordance with AISC Code[27] or its equivalent, which limits the stress ranges in steel components & connections.
		PWR Shield Building				
		Control Room/Building				
		2. BWR Reactor Building with Steel Superstructure				
		3. Auxiliary Building				
		Diesel Generator Building				
		Radwaste Building				
		Turbine Building				
		Switchgear Room				
		Utility/Piping Tunnels				
		Aux FW Pump House				
		5. Fuel Storage Facility				
		Refueling Canal				
		4. Containment Internal Structures	Concrete: InW & InS; Structural steel & SS liner			
		6. Intake Structure	Concrete: ExA, ExB, InW, InS, & F			
		Cooling Tower				
		Spray Pond				
		7. Concrete Tanks	Concrete: F, Int, & Ext			
		8. Steel Tanks	Concrete: F			
		9. BWR Unit Vent Stack	Concrete: F, GA, & GB			

Table B9. Brief summary of technical information and NUMARC/NRC agreements from class 1 structures industry report

Aging-Related Degradation Mechanism	Aging Effects	Structures[a]	Materials:[b] Structural Component	NRC Comment Number[c]	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Settlement	Cracking, distortion, increase in component stress level	1. BWR Reactor Building PWR Shield Building Control Room/Building 2. BWR Reactor Building with Steel Superstructure 3. Auxiliary Building Diesel Generator Building Radwaste Building Turbine Building Switchgear Room Aux FW Pump House Utility/Piping Tunnels 5. Fuel Storage Facility Refueling Canal 6. Intake Structure Cooling Tower Spray Pond 7. Concrete Tanks 8. Steel Tanks 9. BWR Unit Vent Stack	Concrete: F	S-26, S-75	Structure settlement monitoring during construction, & continued monitoring during operation for sites with soft soil and/or significant changes in ground water conditions.§	Structure settlement monitoring initiated during construction phase to confirm that actual settlement is consistent with the allowances included in design basis, & continued settlement monitoring during operation for sites with soft soil and/or significant changes in ground water conditions are current & effective programs to assure structural integrity & functionality of Class 1 structures.

§ See also NUMARC/NRC agreement concerning Settlement page B-62 (Table B4) and comment S-63, page B-42 (Table B3).

[a] Typical Class 1 structures are listed in 9 groupings on the basis of structural components, environmental service condition, and functions.

[b] Designation for concrete components are as follows:
F = foundation including concrete piles, ExA = exterior concrete above grade, ExB = exterior concrete below grade, InW = interior concrete walls including columns, InS = interior concrete slabs including beams, Int = interior concrete structures, Ext = exterior concrete structures, GA = Concrete above grade, and GB = Concrete below grade.
Structural steel components include columns, baseplates, beams, girders, trusses, and bracings for Groups 1-6; and jet impingement barriers for Groups 1-4.
Designation for steel materials are as follows: SS = stainless steel, CS = carbon steel.

[c] The following comments were not included in the table because they deal with clarification or modification of contents of the IR: G-17, S-29, S-31, S-38, S-40, S-41, S-50, S-59, S-62, S-68, and S-69. The following comments were also excluded because they deal with scope of the IR: G-1, G-9, S-1 to S-3, S-27, S-32 to S-37, S-39, S-42 to S-48, and S-67.

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LIST OF BWR PRESSURE VESSEL COMPONENTS:

Class 1 Structures

1. BWR Reactor Building
   PWR Shield Building
   Control Room/Building
2. BWR Reactor Bldg with Steel Superstructure†
3. Auxiliary Building
   Diesel Generator Building
   Radwaste Building
   Turbine Building
   Switchgear Room
   Auxiliary Feedwater Pump House
   Utility/Piping Tunnels
4. Containment Internal Structures
5. Fuel Storage Facility
   Refueling Canal
6. Intake Structure
   Cooling Tower
   Spray Pond
7. Concrete Tanks
8. Steel Tanks
9. BWR Unit Vent Stack

Appendix C: Aging Mechanisms and Effects

The following "ARDMs" and their "effects" have been considered to affect Structures and Components in the reactor containment and Class 1 structures:

Aging Mechanism	Aging Effects
Concrete Structures:	
1. Freeze–Thaw	Scaling, cracking, and spalling
2. Leaching of Calcium Hydroxide	Increase of porosity and permeability
3. Aggressive Chemical Attack	Increase of porosity and permeability, cracking, and spalling
4. Reaction with Aggregates	Expansion and cracking
5. Elevated Temperature	Loss of strength and modulus
6. Iradiation of Concrete	Loss of strength and modulus
7. Creep	Deformation
8. Shrinkage	Cracking
9. Corrosion	Loss of material
10. Abrasion and Cavitation	Loss of material
11. Restrain, Shrinkage, Creep, and Aggressive Environment	Cracking of masonry walls
12. Concrete Interaction with Aluminum	Loss of strength
13. Cathodic Protection Current	Cathodic protection effect on bond strength
Structural Steel & Stainless Steel Liner	
1. Corrosion, Local Corrosion, Atmospheric Corrosion	Loss of material
2. Elevated Temperature	Loss of strength and modulus
3. Irradiation	Loss of fracture toughness
4. Stress Corrosion Cracking	Crack initiation and growth
Reinforcing Steel (Rebar)	
1. Corrosion of Embedded Steel	Cracking, spaling, loss of bond, & loss of material
2. Elevated Temperature	Loss of strength and modulus
3. Irradiation	Loss of strength and modulus
Miscellaneous	
1. Fatigue	Cumulative fatigue damage
2. Settlement	Cracking, distortion, increase in component stress level
3. Mechanical Wear	Lockup
4. Strain Aging (of Carbon Steel)	Loss of fracture toughness
5. Loss of Prestress	Reduction in design margin
6. Corrosion of Steel Piles	Loss of material
7. Corrosion of Tendons	Loss of material

The following "ARDMs" and their "effects" have been considered to affect Structures and Components in the reactor pressure vessel (RPV), reactor vessel internals, and primary coolant pressure boundary (PCPB) :

Aging Mechanism	Aging Effects
1. Corrosion, Boric Acid corrosion,+ Microbiologically induced corrosion	Loss of material++++
2. Creep	Change in dimension
3. Erosion/Corrosion (E/C)	Wall thinning
4. Fatigue	Cumulative fatigue damage
5. Stress Corrosion Cracking (SCC)++ (includes IGSCC, TGSCC, & IASCC)	Crack initiation and growth
6. Neutron Irradiation Embrittlement	Loss of fracture toughness
7. Stress Relaxation	Loss of preload
8. Wear	Attrition
9. Thermal Embrittlement+++	Loss of fracture toughness

+ Boric acid wastage of external surfaces

++ IGSCC: Intergranular SCC; TGSCC: Transgranular SCC; IASCC: Irradiation assisted SCC.

+++ Includes thermal embrittlement of cast austenitic stainless steel (CASS)

++++ "Corrosion product buildup "is also included as an effect in PWR vessel internals and BWR pressure vessel

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)	NUREG-1557
2. TITLE AND SUBTITLE	Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal
3. DATE REPORT PUBLISHED	MONTH: October; YEAR: 1996
4. FIN OR GRANT NUMBER	
5. AUTHOR(S)	C. Regan, S. Lee/NRC O. K. Chopra, D. C. Ma, W. J. Shack/ANL
6. TYPE OF REPORT	Technical
7. PERIOD COVERED (Inclusive Dates)	

8. PERFORMING ORGANIZATION - NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

Division of Reactor Program Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Argonne National Laboratory
Argonne, IL 60439

9. SPONSORING ORGANIZATION - NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Same as 8. above.

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

In about 1990, the Nuclear Management and Resources Council (NUMARC) submitted for NRC review ten industry reports (IRs) addressing aging issues associated with specific structures and components of nuclear power plants and one IR addressing the screening methodology for integrated plant assessment. The NRC staff had been reviewing the ten NUMARC IRs; their comments on each IR and NUMARC responses to the comments have been compiled as public documents. This report provides a brief summary of the technical information and NUMARC/NRC agreements from the ten IRs, except for the Cable License Renewal IR. The technical information and agreements documented herein represent the status of the NRC staffs review when the NRC staff and industry resources were redirected to address rule implementation issues. The NRC staff plans to incorporate appropriate technical information and agreements into the draft standard review plan for license renewal.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

Industry Reports
Reactor Vessel
Containment
Class I Structures
Reactor Vessel Internals
Aging
License Renewal
Degradation Mechanisms

13. AVAILABILITY STATEMENT	unlimited
14. SECURITY CLASSIFICATION (This Page)	unclassified
(This Report)	unclassified
15. NUMBER OF PAGES	
16. PRICE	