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

RECEIVED OCT 2 8 1996

351

# **U.S. Nuclear Regulatory Commission**

# **Office of Nuclear Reactor Regulation**

C. Regan, S. Lee/NRC O. K. Chopra, D. C. Ma, and W. J. Shack/ANL



MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

#### AVAILABILITY NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

- The NRC Public Document Room, 2120 L Street, NW., Lower Level, Washington, DC 20555-0001
- 2. The Superintendent of Documents, U.S. Government Printing Office, P. O. Box 37082, Washington, DC 20402-9328
- 3. The National Technical Information Service, Springfield, VA 22161-0002

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC bulletins, circulars, information notices, inspection and investigation notices; licensee event reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the Government Printing Office: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, international agreement reports, grantee reports, and NRC booklets and brochures. Also available are regulatory guides, NRC regulations in the Code of Federal Regulations, and Nuclear Regulatory Commission Issuances.

Documents available from the National Technical Information Service include NUREG-series reports and technical reports prepared by other Federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal articles, and transactions. *Federal Register* notices, Federal and State legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Office of Administration, Distribution and Mail Services Section, U.S. Nuclear Regulatory Commission, Washington DC 20555–0001.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, Two White Flint North, 11545 Rockville Pike, Rockville, MD 20852–2738, for use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018–3308.

## NUREG-1557

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

Manuscript Completed: June 1996 Date Published: October 1996

C. Regan, S. Lee O. K. Chopra,\* D. C. Ma,\* and W. J. Shack\*

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

#### DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

# DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

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

#### 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.

# **Contents**

Executive Summary							
Abbreviations, Acronyms, and Designations							
1	Intro	duction	1				
2	Technical Information and Agreements						
3	Conclusions						
Refe	erence	S	4				
Арр	endix	A: Public Documents	A-1				
	A1	Pressure Water Reactor Vessel License Renewal Industry Report	A-1				
	A2	Boiling Water Reactor Vessel License Renewal Industry Report	A-2				
	<b>A3</b>	Pressurized Water Reactor Containment Structures	A-3				
	A4	BWR Containments License Renewal Industry Report	<u>A</u> –5				
	A5	PWR Reactor Coolant System License Renewal Industry Report	A-5				
	A6	BWR Primary Coolant Pressure Boundary	A6				
	A7	Pressurized Water Reactor Vessel Internals	A-7				
	A8	Boiling Water Reactor Vessel Internals	A-8				
	A9	Class I Structures License Renewal Industry Report	A-9				
	A10	Low-Voltage In-Containment Environmentally Qualified Cable	A10				
Арр	endix	B: NUMARC/NRC Agreements	B-1				
App	endix	C: Aging Mechanisms and Effects	C-1				

# List of Tables

Bl	Brief Summary of Technical Information and NUMARC/NRC Agreements from Pressure Water Reactor Vessel License Renewal Industry Report
B2	Brief Summary of Technical Information and NUMARC/NRC Agreements from Boiling Water Reactor Vessel License Renewal Industry Report B-16
B3	Brief Summary of Technical Information and NUMARC/NRC Agreements from Pressurized Water Reactor Containment Structures
B4	Brief Summary of Technical Information and NUMARC/NRC Agreements from BWR Containments License Renewal Industry Report B–46
B5	Brief Summary of Technical Information and NUMARC/NRC Agreements from PWR Reactor Coolant System License Renewal Industry Report B–66
B6	Brief Summary of Technical Information and NUMARC/NRC Agreements from BWR Primary Coolant Pressure Boundary
B7	Brief Summary of Technical Information and NUMARC/NRC Agreements from Pressurized Water Reactor Vessel Internals
B8	Brief Summary of Technical Information and NUMARC/NRC Agreements from Boiling Water Reactor Vessel InternalsB-123
В9	Brief Summary of Technical Information and NUMARC/NRC Agreements from Class I Structures License Renewal Industry Report

# **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 SRP-LR.

NUREG-1557

viii

# 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
B-E	ASME Section XI, Table IWB-2500-1, Examination Category for Pressure Retaining
	Partial Penetration Welds in Vessels
B-F	ASME Section XI, Table IWB-2500-1, Examination Category for Pressure Retaining
	Dissimilar Metal Welds
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
DOF	United States Department of Energy
F_A	ASME Section XI Table IWE-2500-1 Examination Category for Containment
L-A	Surfaces
F_B	ASME Section XI Table IWE-2500-1 Examination Category for Containment
<u>B-D</u>	Surfaces
F_C	ASME Section XI Table IWE-2500-1 Examination Category for Pressure Petaining
<b>D</b> -C	Welde
F. D	ASME Section XI Table IWE-2500 1 Evamination Category for Seals Caskets
<u>D</u> –D	and Mojeture Barriers
F_F	ASME Section XI Table IWE_2500_1 Examination Category for Pressure Retaining
L-r	Dissimilar Metal Welds
FC	ASME Section XI Table IWE_2500_1 Examination Category for Pressure Retaining
E-U	Rolting
Бр	ASME Section VI Table IWE-2500-1 Examination Category for all Pressure
Iv−r	Pataining Components
F/C	Fresion /Corresion
ECCS	Enosion Core Cooling System
FCP	Electrochemical Potential
FFDV	Effective Full Dower Vears
EFF1 FDA	United States Environmental Protection Agency
FDDI	Fleatric Power Desearch Institute
FO	Environmental Qualification
FSAR	Final Safety Analysis Report
FW	Freedwater
CDS	Ceneral Design Criteria
GDS Cr	Crode
	Glaue
HDCI	High Pressure Coolent Injection
HDCS	High Pressure Core Spray
HEW	Heat Sink Walding
INVO	Heating Ventilation and Air Conditioning
HVAC	nearing, venuiation, and Air Conditioning

х

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
2000	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"
IN/T	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 Schanglin and Becker
	ASME Section XI. Table IVII _2500_1. Examination Category for Concrete
	Licensee Event Deport
LER	Loss of Coolant Accident
LUCA	Loss of Coolant Accident
	Low Pressure Core Sprey
LPCS	Low Power Pange Monitor
	Low-Fower Kange Monitor
	Light Water Depater
LWIN	Main Steam
MOID	Main Steam
MON	Mein Steem Jeeletien Velve
ND	Subsection of ASME Code, Section III "Dules for Construction of Nuclear Power
ND	Blant Components " dealing with Class 1 Components
NUM	Nuclear Energy Institute
NC	Subsection of ASME Code, Section III "Dules for Construction of Nuclear Power
NG	Subsection of ASME Code, Section in Rules for Construction of Nuclear Power
NDAD	Nuclear Dept Aging Research
NPAR	Nuclear Plant Aging Research
NDCA	Nuclear Regulatory Commission
NRCA	Nuclear Steem Supply System
NUMADO	Nuclear Steam Supply System
NUMARC	Nuclear Management and Resources Council
PCM	Pipe Crack Monitor (System)
POPU	Primary Coolant Pressure Doundary
PORV	Power-Operated Kellel Valve
ppp	Parts per Dillion
ppm	Parts per million
PT PT	Pressure reinperature (Limits)
r15	riessunzeu inerniai Snock

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	ş
	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	1
	system pressure or functional tests	
VT-3	Visual examination specified in ASME Section XI inservice inspection conducted to	כ
	determine the mechanical and structural integrity, such as the examination o	f
	conditions that could affect functionality of components and their supports	

xii

# 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,<sup>1-10</sup> and one IR addressing the screening methodology for integrated plant assessment (IPA).<sup>11</sup> 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.<sup>12,13</sup> In SECY-93-049,<sup>12</sup> 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.<sup>\*</sup> NUMARC agreed with the NRC staff's approach in a letter dated March 3, 1994.<sup>\*\*</sup>

This report provides a brief summary of the technical information and NUMARC/NRC agreements from the ten IRs, except for the cable IR.<sup>10</sup> The cable IR addresses the issue of environmental qualification (EQ) of electric equipment, which was superseded by the EQ action plan.<sup>†,††</sup> 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.<sup>#</sup> The NEI submitted comments

<sup>\*</sup> Staff Requirements Memorandum from Samuel J. Chilk, dated June 28, 1993.

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

<sup>&</sup>lt;sup>†</sup> Letter from William T. Russel of NRC to William H. Rasin of the Nuclear Management and Resources Council, dated July 30, 1993.

<sup>&</sup>lt;sup>††</sup> "Environmental Qualification of Electric Equipment," Memorandum for the Commissioners from James M. Taylor, dated April 8, 1994.

<sup>&</sup>lt;sup>#</sup> December 12, 1994, Meeting Summary "License Renewal Industry Report," dated December 22, 1994.

dated December 20, 1995,<sup>##</sup> 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,<sup>\*</sup> 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.<sup>\*\*</sup>

# **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 effects 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

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

<sup>\* &</sup>quot;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.

<sup>\*\* &</sup>quot;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.

- For a specific ARDM or ARDM/component combination, if the effects of aging are not po-1. tentially 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 freezethaw 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

- "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.
- "Boiling Water Reactor Containments License Renewal Industry Report," NUMARC Report Number 90–10, Nuclear Management and Resource Council, July 1990; Revision 1, Dec. 1991.
- "PWR Reactor Coolant System License Renewal Industry Report," NUMARC Report Number 90–07, Nuclear Management and Resource Council, Oct. 1990; Revision 1, May 1992.

- "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.
- "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.
- "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.

## 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.

A-3

- 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.

A-4

## 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).

. .

Aging–Related Degradation Mechanism	Aging Effects	Components <sup>a</sup>	Materials <sup>b</sup>	NRC Comment Number <sup>c</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron	Loss of	Closure Head Dome	SA302-Gr B,	None	Non-significant	Total fast neutron fluence level
Irradiation	fracture		SA533-Gr B			is low & is less than
Embrittle-	toughness	CRD Mechanism Housing	SB-166,	1		$1 \ge 10^{17} \text{ n/cm}^2$ , above which a
ment			SA182 Type			surveillance program is
			304 or 316			required under Appendix H of
		Refueling Seal Ledge	SA212-Gr B,		· · · · · · · · · · · · · · · · · · ·	10CFR Part 50.
			SA516-Gr 70,			
	н. Г		SA533-Gr B			
		Closure Head Lifting Lugs	SA302-Gr B,	1		
			SA533-Gr B			
		Shroud Support Ring	SA212-Gr B,			
		11 0	SA516-Gr 70		· · · · · · · · · · · · · · · · · · ·	
		Closure Head Flange	SA336, SA508	3		
		Closure Stud Assembly	SA-540-B23	1		
			or B24,			
			SA-320-L43			
		Vessel Flange	SA336, SA508	3		a second s
		Leakage Monitoring Tubes	SB-166,	1		
			SB-167,			
			SA-312			
			Type 316			
		Core Support Pads (Lugs)	SB-166,			
			SB-168			
		Bottom Head Dome	SA302-Gr B,			
	]		SA533-Gr B			
		Instrumentation	SB-166,			
		Tubes/Penetrations*	SB-167			

\* Includes the vent pipe on the closure head dome.

NUREG-1557

Aging–Related Degradation	Aging			NRC Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials <sup>b</sup>	Number <sup>c</sup>	Agreement or Proposal	Agreement or Proposal
Neutron	Loss of	Intermediate & Lower Shell	SA302-Gr B	1.	Unresolved issue	NUMARC basis: Reactor vessel
Irradiation	fracture		SA533-Gr B	-, 11 to 13.		material surveillance program in
Embrittle-	toughness		SA336, SA508	(S1)S-15.	NUMARC proposal:	accordance with 10CFR50
ment				(S2) 2.	Verification of pressure vessel	Appendix $H^1$ implementing rec-
				(S2) 4.	integrity by requirements of	ommendations of ASTM STD
		· · · ·		(S2) 5.	fracture toughness & materials	$E185-82.^2$ & guidelines of
				(S2) 7.	surveillance program delin-	RG 1.99. Rev. 2. <sup>3</sup> ensure that
				(S3)S-1.	eated in Appendices G & H of	the beltline material is not ex-
				(S3)S-4	10CFR50: <sup>1</sup> guidelines of RG	cessively irradiated during the
				(S3)S-6,	1.99 <sup>3</sup> to estimate embrittle-	license renewal term; pressur-
				(S3)S-7,	ment & $1.154^4$ for PTS safety	ized thermal shock (PTS) rule
				(S3)S-8	analysis. If requirements of	10CFR 50.61 & RG 1.154 <sup>4</sup> en-
	1. 1.				upper-shelf toughness can not	sure that vessel integrity is
				Open	be met, then 100% volumetric	maintained; Appendix G of
	-			issue	inspection of beltline; supple-	10CFR50, <sup>1</sup> using guidelines of
				3, 4,	mental fracture toughness	ASME Sect. III, <sup>5</sup> Appendix G, as-
				(S1)S-1,	testing; fracture mechanics	sures that beltline materials will
				(S2) 3,	analysis showing equivalent	maintain adequate upper-shelf
				(S2) 6	margins of safety, & methodol-	toughness; & 10CFR50,
					ogy of NUREG 0744 for opera-	Appendix G, <sup>1</sup> with recommen-
					tion; NUREG 0244, <sup>6</sup> SRP Sect.	dations of NUREG 0244, <sup>6</sup> SRP
			6		5.2.2, <sup>7</sup> & BTP RSB 5-2 <sup>8</sup> pro-	Sect. 5.2.2, <sup>7</sup> & BTP RSB 5-2, <sup>8</sup>
					vide guidance on low temp over	assure that PT limits will not ad-
					pressure protection.	versely affect vessel integrity. If
						upper-shelf requirements cannot
					If above acceptance criteria can	be met, then 100% volumetric
					not be met then select plant-	inspection of beltline region in
					specific aging management	accordance with ASME Sect. XI,
					program that may include	Table IWB-2500-1; <sup>9</sup> supplemen-
					mitigation by flux reduction or	tal fracture toughness tests;
					thermal annealing.	fracture mechanics analysis that
						conservatively demonstrate
					(Contd. next page)	(Contd. next page)

V

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

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

V

\* Includes safety injection nozzles on some vessels.

NUREG-1557

Aging-Related Degradation Mechanism IGSCC	Aging Effects Crack initiation & growth	Components <sup>a</sup> Closure Stud Assembly	Materials <sup>b</sup> Alloy 4340	NRC Comment Number <sup>c</sup> None	NUMARC/NRC Agreement or Proposal ASME Sect. XI, <sup>9</sup> Subsect. IWB, exam. category B-G-1 visual VT-1, surface, & vol- umetric exam. of closure stud assemblies, in accordance with RG 1.65; <sup>11</sup>	Basis for Agreement or Proposal ASME Sect. XI, <sup>9</sup> exam. category B-G-1 & guidelines of RG 1.65 <sup>11</sup> for closure stud assemblies is current & effective program.
IGSCC	Crack initiation & growth	Closure Head Dome CRD Mechanism Housing Refueling Seal Ledge Closure Head Lifting Lugs Shroud Support Ring Closure Head Flange Vessel Flange Upper (Nozzle) Shell	SA302-Gr B,   SA533-Gr B   SA182 Type   304 or 316   SA212-Gr B,   SA516-Gr 70,   SA533-Gr B   SA302-Gr B,   SA516-Gr 70   SA533-Gr B   SA533-Gr B,   SA533-Gr B,   SA516-Gr 70   SA533-Gr B,   SA516-Gr 70   SA536,   SA508   SA336,   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 sensitiza- tion of SS cladding is possi- ble. <sup>12-14</sup> 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 condi- tions; consider the informa-	NUMARC basis: Low alloy steels & SS cladding with >5% ferrite content are not susceptible to SCC in PWR environment; implementation of RG 1.43 <sup>16</sup> to prevent underclad cracking & guidelines of RG 1.44 <sup>17</sup> to avoid sensitization of SS; control of halogens & oxy- gen in the primary water to <5 & <0.01 ppm, respectively; <sup>18</sup> & monitor & control of water chemistry during shutdown to mitigate the potential of SCC; or the components are not sub- jected to corrosive environment.
		Primary Coolant Nozzles Leakage Monitoring Tubes Intermediate & Lower Shell Bottom Head Dome	SA336, SA508     SA336, SA508     SA508     SA-312     Type 316     SA502-Gr B,     SA336, SA508     SA302-Gr B,     SA336, SA508     SA302-Gr B,		tion in NUREG/CR-5020.40	

clated			NRC		
n Aging		•	Comment	NUMARC/NRC	Basis for
n Effects	Components <sup>a</sup>	Materials <sup>D</sup>	Numberc	Agreement or Proposal	Agreement or Proposal
Crack initiation &	CRD Mechanism Housing	SB-166	2, 7a. (S1)S-2,	Unresolved issue	NUMARC basis: ASME Sect. XI, <sup>9</sup> Subsect. IWB,
growth	Leakage Monitoring Tubes	SB-166,	(S1)S-11,	NUMARC proposal:	inspection & testing programs
		SB-167,	(S2) 1a	ASME Sect. XI, <sup>9</sup> Subsect. IWB,	that include exam. category B-O for welds in CRD housing:
		221 GS		citiza valumatria & curface	B.F for nartial nenetration
	Core Support Pags (Lugs)	SB-160, SB-168	Upen issue	quires volumetric & surface exam. of CRD housings; B-E	welds; B-N-2 for integrally-
			Inconel	calls for visual VT-2 of the	welded interior attachment
	Instrumentation	SB-166,	$182^{*}$	external surfaces of partial	welds for core support pads; &
	Tubes/Penetrations	SB-167		penetration welds for leakage	plant-specific review of compo-
				during hydrotests; B-N-2 cov-	nent material, e.g., Alloy 600
				ers visual VT-3 exam. of inte-	applications, & fabrication
				rior attachment welds for core	records, component stress re-
				support pads; plant specific	ports, & component service
				review of component materials;	temperature; & evaluation,
				& evaluation, defect repair , &	defect repair, & replacement are
		-		replacement.	current & effective to manage
					SCC damage.
				NRC proposal:	
				Alloy 600 should be further	
				evaluated; evaluate the poten-	
				tial 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 ad-	
		-		dressed & should incorporate	
				requirements of ASME Sect.	
				XI Anendices VII & VIII.	

	I	I	I	-		
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Componentsa	Materials <sup>D</sup>	Number <sup>c</sup>	Agreement or Proposal	Agreement or Proposal
Corrosion/	Lose of	Refueling Seal Ledge	SA212-Gr B	(53)5-2	Non-significant	Internal surfaces of reactor
Borio Acid	material	Actucing Scar Leuge	SA516 Cr 70	(63)6-2,	Non-significant	vessel components whether
Wostore of	material		SA510-01 70,	0000-0		olod or fabricated of SS or Allow
wastage of			SA555-GFB	9		COO are not subject to correction
External		Closure Head Linning Lugs	SASUZ-GFB,			obb, are not subject to conosive
Surfaces			SA533-GF B			attack in a PwR environment;
		Shroud Support Ring	SA212-Gr B,			& corrosion of reactor vessel
			SA516-Gr 70			base metal due to removal of
		Upper (Nozzle) Shell	SA302-Gr B,			protective cladding results in
		1	SA533-Gr B,			very low corrosion rates; or
			SA336, SA508			components are not exposed to
-		Primary Coolant Nozzles	SA336, SA508			corrosive environment or are not
	1	Leakage Monitoring Tubes	SB-166 & 167			susceptible to potential boric
			SA-312-316			acid leak.
		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	SB-166,			
		Tubes/Penetrations	SB-167			
Corrosion/	Loss of	Closure Head Dome	SA302-Gr B,	(S3)S-4,	Implementation of Generic	Recommendations of Generic
Boric Acid	material		SA533-Gr B	(S3)S-6,	Letter 88-05. <sup>19</sup>	Letter 88-05 <sup>19</sup> are current &
Wastage of		CRD Mechanism Housing	SB-166,	(S3)S-7	Exam. category B-P, Subsect.	effective program to monitor &
External			SA182 Type		IWB <sup>9</sup> requires VT-2 inspection	control primary coolant leakage.
Surfaces			304 or 316		during leakage & hydrostatic	
		Closure Head Flange	SA336, SA508		tests & VT-3 inspection of bolt	of bolt
		Closure Stud Assembly	SA-540-B23		in case of leakage; exam. cate-	
	1		or B24,		gory B-E provides visual VT-2	and the second
			SA-320-L43		of partial penetration welds	
1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 19	N	Vessel Flange	SA336, SA508		during system hydrotesting, &	
-					visual VT-1 of closure nuts.	
					washers, & bushings.	
					· · · · · · · · · · · · · · · · · · ·	
---------------	----------	----------------------------	------------------------	---------------------	---------------------------------------	------------------------------------------
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Componentsa	Materials <sup>b</sup>	Number <sup>c</sup>	Agreement or Proposal	Agreement or Proposal
Erosion/	Wall	Closure Head Dome	SA302-Gr B,	(S3)S-2,	Non-significant	Austenitic SS & Ni alloys on
Corrosion	thinning		SA533-Gr B	(\$3)\$-3		inside reactor vessel surfaces
(E/C)		CRD Mechanism Housing	SB-166,			are resistant to E/C; moderate
			SA182 Type			fluid velocities & single phase
			304 or 316			flow; and controls on reactor
		Refueling Seal Ledge	SA212-Gr B.			coolant system chemistry.
			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			
ł		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	SB-166,			
а. С		Tubes/Penetrations	SB-167			
Erosion/	Wall	Closure Head Flange	SA336, SA508	(S3)S-4,	ASME Sect. XI exam. category	ASME Sect. XI, <sup>9</sup> Subsect. IWB
Corrosion	thinning			(S3)S-6,	B-P requires VT-2 visual	exam. category B-P is current &
(E/C)		Vessel Flange	SA336, SA508	(S3)S-7	system leakage & hydrotests.	effective program to manage the
						effects of E/C.

· · · · · · · · · · · · · · · · · · ·	r <sup></sup>	T	T		r	r
Aging-Related				NRC	· · · ·	
Degradation	Aging		1	Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials <sup>b</sup>	Number <sup>c</sup>	Agreement or Proposal	Agreement or Proposal
Wear	Attrition	Closure Head Doma	SA202 C+ P	10	Non cignificant	Not subjected to motion relative
wear	AUTION	Closure nead Dome	SASU2-GI D,	10,	Non-significant	to other components
			SASSS-GF D	(51)5-14		to other components.
		CRD Mechanism Housing	SB-166,	(\$3)\$-2,		
· · · · ·			SA182 Type	(53)5-3		
			304 or 316			
		Refueling Seal Ledge	SA212-Gr B,			
		a second s	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	SB-166,			
		Tubes/Penetrations	SB-167			
Wear	Attrition	Closure Head Flange	SA336, SA508	(S3)S-4,	ASME Sect. XI exam. category	ASME Sect. XI, <sup>9</sup> Subsect. IWB
				(S3)S-6,	B-P requires VT-2 during	exam. category B-P for closure
		Closure Stud Assembly	SA-540-B23	(S3)S-7	system leakage & hydrotests,	head & vessel flanges, B-G-1 for
			or B24,		B-G-1 requires visual, surface,	closure studs, & B-N-1 for core
			SA-320-L43		& volumetric exam. of closure	support pads are current
		Vessel Flange	SA336, SA508		stud assemblies, & B-N-1	& effective programs to manage
		Core Support Pads (Lugs)	SB-166 & 168		requires VT-3 of support pads.	the effects of wear.

Table B1.	Brief summary e	of technical information	and NUMARC/NRC ag	greements from PWR	pressure vessel industry report
-----------	-----------------	--------------------------	-------------------	--------------------	---------------------------------

Aging-Related Degradation Mechanism	Aging Effects	Components <sup>a</sup>	Materials <sup>b</sup>	NRC Comment Number <sup>c</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Creep	Change in dimension	Closure Head Dome	SA302-Gr B, SA533-Gr B SB-166, SA182 Type 304 or 316	(S3)S-2, (S3)S-3	Non-significant	The PWR reactor vessel operat- ing temperatures are <343°C (<650°F) which is well below the creep range; creep is not a con- cern below 427°C (800°F) for
		Refueling Seal Ledge	SA212-Gr B, SA516-Gr 70, SA533-Gr B			low alloy steels & below 538°C (1000°F) for SS. No significant effect of irradiation on creep of
		Closure Head Lifting Lugs	SA302-Gr B, SA533-Gr B			vessel materials has been identified. <sup>20</sup>
		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	1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -		
		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			

Aging-Related	· .		and the second se	NRC	l	
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Componentsa	Materialsb	Numberc	Agreement or Proposal	Agreement or Proposal
		· · · · · · · · · · · · · · · · · · ·				
Thermal	Loss of	Closure Head Dome	SA302-Gr B,	8,	Non-significant	None of the reactor vessel
Embrittle-	fracture		SA533-Gr B	(S1)S-7,		components are susceptible to
ment	toughness	CRD Mechanism Housing	SB-166,	(S1)S-8,		the various forms of thermal
			SA182 Type	(S3)S-2,		aging, e.g., thermal embrittle-
			304 or 316	(S3)S-3		ment of CASS, temper embrit-
		Refueling Seal Ledge	SA212-Gr B,			tlement, or strain aging em-
	÷	en Antonio Maria de Caractería de Caractería	SA516-Gr 70,		5 7	brittlement, because there is no
			SA533-Gr B			CASS component (CRD housing
	÷	Closure Head Lifting Lugs	SA302-Gr B,	4 2		made of CASS is outside the
			SA533-Gr B			scope of this industry Report,
		Shroud Support Ring	SA212-Gr B,			the relatively low operating
			SA516-Gr 70			DUD- 8 the meterial controls
ł		Closure Head Flange	SA336, SA508			PWRS & the material controls
	5	Closure Stud Assembly	SA-540-B23			used during manufacturing.
:	-		01 D24,			
		Vessel Flange	SA-320-L45			
		Upper (Neggle) Shell	SA330, SA308			
e e		opper (Nozzie) Shen	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			
X.		Core Support Pads (Lugs)	SB-166 & 168			
		Bottom Head Dome	SA302-Gr B,	a .		
			SA533-Gr B			
A	1	Instrumentation	SB-166 & 167			and the second second second second
		Tubes/Penetrations		1.5		

Tuble D1. Drog bannang of technetic information and month (C) mile agreements from 1 mile pressure besser i autorig	chnical information and NUMARC/NRC agreements from PWR pressure vessel industry report
---------------------------------------------------------------------------------------------------------------------	----------------------------------------------------------------------------------------

		<b>I</b>			· · · · · · · · · · · · · · · · · · ·	· · · · · · · · · · · · · · · · · · ·
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis Ior
Mechanism	Effects	Components <sup>a</sup>	Materials <sup>D</sup>	Number <sup>c</sup>	Agreement or Proposal	Agreement or Proposal
Fatigue	Cumulative	Closure Head Dome	SA302-Gr B	5(a)-(c).	Unresolved issue	NUMARC basis:
1 utigue	fatigue		SA533-Gr B	6,		Fatigue usage factors are <0.4
	damage	Refueling Seal Ledge	SA212-Gr B,	(S3)S-2,	NUMARC proposal:	for the current license term, &
			SA516-Gr 70,	(S3)S-3	Non-significant	are anticipated to be signifi-
			SA533-Gr B			cantly less than ASME Code
		Closure Head Lifting Lugs	SA302-Gr B,	Open	NRC proposal:	limit of 1.0 for the entire license
			SA533-Gr B	issues	Until an agreement is	renewal term.
		Shroud Support Ring	SA212-Gr B,	5(d),	reached on the draft staff dis-	
			SA516-Gr 70	14,	cussion paper on fatigue, the	
		Closure Head Flange	SA336, SA508	(S1)S-3,	issue is unresolved.	
				(S1)S-12,		
		Vessel Flange	SA336, SA508	(S1)S-16,		
		Upper (Nozzle) Shell	SA302-Gr B,	(S1)S-17,		
			SA533-Gr B,	(S1)S-18		
			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			

.

			· · · · · · · · · · · · · · · · · · ·		······································	
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials <sup>D</sup>	Numberc	Agreement or Proposal	Agreement or Proposal
	0 1.11		on 100	5(2) (2)		NUMADC hapin. Varification of
raugue	Cumulative	CRD Mechanism Housing	SB-100,	5(a)-(c),	Unresolved issue	WOWARC Dusts. Verification of
	laugue		SA182 Type	6		continued adequacy of fatigue
	damage		304 or 316	(53)5-4,	NUMARC proposal:	design basis through reanalysis
		Closure Stud Assembly	SA-540-B23	(S3)S-6,	ASME Sect. III <sup>5</sup> Subsect. NB	of usage factor.
			or B24,	(S3)S-7	reanalysis of usage factor,	ASME Sect. XI, <sup>9</sup> Subsect. IWB,
			SA-320-L43		regrouping design-basis tran-	inspection & testing programs
		Primary Coolant Nozzles	SA336, SA508	Open	sients, actual plant transients,	that include, exam. category B-A
				issues	cycle monitoring, & partial	& Supplement 6, Appendix VIII
		Instrumentation Tubes	SB-166,	5(d),	cycle counting; ASME Sect.	for volumetric inspection of
		(Penetrations)	SB-167	14,	XI, <sup>9</sup> Subsect. IWB, inspection	pressure retaining welds; exam.
				(S1)S-3,	requirements of IWB-2500-1; &	category B-D & Supplement 7,
				(S1)S-12,	evaluation, defect repair &	Appendix VIII for volumetric in-
				(S1)S-16,	replacement.	spection of full penetration noz-
[ ]			· ·	(\$1)\$-17,	-	zle welds; exam. category B-E for
				(S1)S-18	NRC proposal:	visual VT-2 inspection of ex-
					Until an agreement is reached	ternal surfaces of partial pene-
					on the draft staff discussion	tration welds; & exam. category
			{	ł	paper on fatigue, the issue is	B-G-1, Appendix VI, &
			•		unresolved.	Supplement 8 of Appendix VIII
				1		for volumetric & surface inspec-
						tion of closure studs, replace-
. · · ·						ment in accordance with RG
						1.65:11 & evaluation, defect
						repair, & replacement are
						current & effective to manage
						fatigue degradation.

Table B1. Brief summary of technical information and NOMARC/NRC agreements from PWR pressure besset indusu
------------------------------------------------------------------------------------------------------------

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.
 <sup>b</sup> 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.
 <sup>c</sup> 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.

NUREG-1557

- 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

/	·····	·····			······	
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materialsa	Numberb	Agreement or Proposal	Agreement or Proposal
NT.	T		1			The total fact montron fluones
Neutron	LOSS OI	Attachment weids	55,	G-7,	Non-significant	within the license renewed term
Irradiation	Iracture		Alloy 182	S-23,		within the incense renewal term
Embrittle-	toughness	Bottom Head	SA302-Gr B,	(S1)G-2,		is less than 1 x 10 <sup>17</sup> n/cm <sup>2</sup> , the
ment			SA533-Gr B	(SI)G-5,		level identified in TOCFRSO
1		Closure Studs	SA-193,	(S1)S-1,		Appendix H <sup>1</sup> requiring a mate-
			SA-540	1(51)5-9		rials surveillance program; or
		Nozzles		4		the components are made of SS
		Feedwater	SA508-Cl 2	ł		or Ni-Cr-Fe alloys that are not
		BWR/2 CRD return line (RL)	SA508-Cl 2	4	(1,1,2,2,2,2,2,2,2,2,2,2,2,2,2,2,2,2,2,2	susceptible to neutron embrit-
		All Other	SA508-C12	ł		tlement at fluences less than
		Penetrations		1		$1 \ge 10^{20} \text{ n/cm}^2$ .
1		CRD Stub Tubes	SS, SB-167	]		
		All Other	SB-167			
		Safe Ends		•		
		BWR/5 LPCI	SS, SB-166			44 -
		Feedwater	CS, SB-166	]		
·		BWR/2 CRDRL	CS, SB-166	}		
		All Other	CS, SB-166	]		
		Vessel Flange	SA336	1		
	:		SA508-Cl 2			
		Vessel Shell		]		
		Other than beltline	SA302-Gr B,			
			SA533-Gr B	l		
		Top Head	SA302-Gr B,	1		
		-	SA533-Gr B	1		
Neutron	Loss of	Support Skirt	SA533-Gr B	G-7,	Non-significant	During the license renewal term,
Irradiation	fracture			(S1)G-2,		shift in reference temp. due to
embrittle-	toughness			(S1)G-5,		neutron exposure is <11°C
ment	-			(S1)S-1,		(<20°F) & irradiation
				S-60		embrittlement due to thermal
			·	{		neutrons is not significant.

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials <sup>a</sup>	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
NT. 4	<b>X</b>	AT 1		0.1	A	ACME Section XI Annondices
Neutron	Loss of	Nozzles		G-4,	Agreement that inspection will	ASIME Section AI, Appendices
Irradiation	fracture	BWR/5 LPCI	SA508-Cl 2	G-5,	be performed in accordance	VII and VIII provide an adequate
Embrittle-	toughness	Vessel Shell		G-18,	with ASME Sect. XI,	process for qualifying inspection
ment		Beltline Weld	Low-alloy steel	S-5,	Appendices VII and VIII.	personnel, equipment, and
		, a, e,	(LAS)	S-7,	Unresolved issue related to the	procedures.
1			weldment		extent of inspection.	
		Beltline Shell	SA302-Gr B,	S-9,		NUMARC basis:
	:		SA533-Gr B	S-48 to	NUMARC proposal:	Pressure vessel integrity is as-
	2			S-52.	Verification of pressure vessel	sured by operating require-
				S-54.	integrity by operating &	ments of Appendix G of
				S-55	surveillance requirements of	10CFR50. <sup>1</sup> using guidelines of
				S-60	Appendices G & H of	ASME Sect. III. <sup>2</sup> Appendix G.
				S-63	10CFR50.1	MTEB 5-2. <sup>3</sup> & Reg. Guide 1.99.
				S-67		Rev 2.4 & surveillance re-
				(SUG-10	In the event that fracture	quirements of Appendix H of
				(\$1)G-11	toughness requirements reach	10CFR50 implementing guide-
				(61)6-11,	acceptance oriteria, then our-	lines of ASTM STD F185-82 5
				(51)5-2,	rent practices to be enhanced	miles of ASTM STD E105 02.
				(51)5-12,	rent practices to be enhanced,	If requirements of 100FP50
				(51)5-13	select plant-specific manage-	in requirements of TOCFROO
					ment that may include com-	cannot be met, then select a
				Open	plete volumetric inspection of	plant-specific management plan
				issue	beltline, supplemental fracture	that may include a complete
				S-53	toughness testing, & fracture	volumetric inspection of beitline
					mechanics analysis showing	region in accordance with ASME
				1. A.	equivalent margins of safety for	Sect. XI, Table IWB-2500-1;°
					operation.	supplemental tests to obtain
						additional evidence of change in
				1	NRC proposal:	fracture toughness; fracture
					A 100% volumetric inspection	mechanics analysis that
					of all beltline & all other	conservatively demonstrate
					accessible welds required by	adequate margins of safety; &
					ASME Sect. XI. <sup>6</sup> Exemptions	methodology of NUREG-07447
					for license renewal will be	for operation.
					reviewed on a case by case	-
					basis.	

т

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials <sup>a</sup>	NRC Comment Number <sup>b</sup>	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, <sup>6</sup> Subsect. IWB, inspection & testing requirements of IWB-2500-1,	ASME Sect. XI, <sup>6</sup> Subsect. IWB, inspection & testing programs; exam. category B-G-1 for clo-
		Nozzles BWR/5 LPCI	SA508-Cl 2	S-10, S-28,	augmented by NUREG 0313 <sup>10</sup> & Generic letter 88-01 <sup>11</sup> for	sure studs, recommendations of RICSIL-055, <sup>8</sup> & replacement
		Feedwater	SA508-Cl 2	S-29,	nozzles & safe ends, RICSIL	in accordance with RG 1.65; <sup>9</sup>
		BWR/2 CRDRL	SA508-Cl 2	S-36	055 <sup>8</sup> & RG 1.65 <sup>9</sup> for closure	categories B-D & B-F for noz-
		All Other	SA508-Cl 2	S-37	studs; & evaluation, defect	zles & safe ends, additional re-
		Penetrations		S-6,	repair, & replacement.	quirements of NUREG 0313 <sup>10</sup>
		CRD Stub Tubes	SS, SB-167	S-11,		implemented by Generic letter
		All Other	SB-167	S-12,		88-01; <sup>11</sup> category B-E for pene-
				S-19,		trations; & analytical evalua-
		Safa Enda		(51)5-14		mont are current & effective
		BUD /5 LDOL	SS SD 166	C 12		ment are current & chective
		Bwk/5 LFCI Feedwater	CS SP 166	S-13,		programs.
		RWR /2 CPDPI	CS, SB-166	5-14		
		All Other	CS_SB-166			
		Am Other	<u>CO, OD-100</u>	Common		
				to above		
-				S-25,		
				S-26,		
				S-31,		
				S-32,		
				S-42 to		
				S-45,		
				S-66		

Table B2.	Brief summary o	of technical informati	on and NUMARC/NRC	agreements from BWR	pressure vessel industry report
			•		

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials <sup>a</sup>	NRC Comment Number <sup>b</sup>	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 & moni- toring; <sup>12</sup> water chemistry con- trol; <sup>13</sup> & analytical evaluation- repair-replacement.
		· · · · · · · · · · · · · · · · · · ·		1	T	
IGSCC	Crack initiation &	Bottom Head	SA302-Gr B, SA533-Gr B	G-5, G-7,	Non-significant	Low-alloy steel & SS clad with >5% ferrite are not susceptible
	growth	Vessel Flange	SA336, SA508-Cl 2	G-18, (S1)G-2,		to SCC, <sup>14</sup> and/or applied & residual stresses are low, or
		Vessel Shell		(S1)G-5.		are not subjected to corrosive
· · · ·		Beltline Shell	SA302-Gr B, SA533-Gr B	(S1)S-1, S-41		environment.
		Other than beltline shell & weld	SA302-Gr B, SA533-Gr B	S-27, S-33 to		
		Top Head	SA302-Gr B, SA533-Gr B	S-35, S-24		
		Support Skirt	SA533-Gr B	(\$1)\$-3		
IGSCC	Crack	Vessel Shell		See	Unresolved issue	NUMARC basis:
	initiation &	Beltline Weld	LAS weldment	above &	NUMARC proposal:	Weld metal with at least 5%
	growth			Open	Non-significant	ferrite is not susceptible to SCC,
				issue	NRC proposal: See neutron	& control of water chemistry
				S-53	irradiation embrittlement of	such that oxygen is <10 ppb
					beltline welds.	& halogen level <5 ppm.

1.1

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

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

Aging-Related	[			NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materialsa	Numberb	Agreement or Proposal	Agreement or Proposal
IASCC	Crack initiation &	Attachment Welds	SS, Alloy 182	G-7, S-39,	Non-significant	IASCC is non-significant in low- alloy or CS components
	growth	Bottom Head	SA302-Gr B, SA533-Gr B	(S1)G-2, (S1)G-5,		subjected to neutron fluences typical of BWR vessel service;
		Closure Studs	SA-193, SA-540	(S1)S-1		IASCC is non-significant for SS and Ni-Cr-Fe alloy components
		Nozzles				because the total fast neutron
		BWR/5 LPCI	SA508-Cl 2	· ·		fluence within the license re-
		Feedwater	SA508-Cl 2			newal term is $<1 \ge 10^{20} \text{ n/cm}^2$
· .		BWR/2 CRDRL	SA508-Cl 2			for highly stressed components
		All Other	SA508-Cl 2			$\& <5 \ge 10^{20} \text{ n/cm}^2$ for compo-
		Penetrations				nents that are subjected to
		CRD Stub Tubes	SS, SB-167			stresses <68 MPa (<10 ksi).
		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,			
	ĺ		SA533-Gr B			
		All Other	SA302-Gr B,			
			SA533-Gr B			
		Top Head	SA302-Gr B,			
	4. (C)		SA533-Gr B			
		Support Skirt	SA533-Gr B			

Aging Peloted	<b>.</b>	T	T	MDC		······································
Aging-Actated	Aging			Comment	NUMARC /NRC	Basis for
Mechanism	Effects	Components	Materialea	Numberb	Agreement or Proposal	Adreement or Proposal
Meenamom	Difecto	Components	Matchais	Mumber	Agreement of Troposal	
Corrosion	Loss of	Attachment Welds	SS,	G-7,	Non-significant	The pressure vessel components
	material/		Alloy 182	(S1)G-2,		are internally clad with SS or
	corrosion	Bottom Head	SA302-Gr B,	(S1)G-5,		fabricated of SS or Ni-Cr-Fe
	product		SA533-Gr B	(S1)S-1		alloy which are very resistant to
	buildup	Closure Studs	SA-193,	:		corrosion; for unclad regions
		· · · · · · · · · · · · · · · · · · ·	SA-540			corrosion rates in typical BWR
· · · · · ·		Nozzles				environments are very low.
		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			$(x_1,y_2,\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{n-1},\ldots,y_{$
		BWR/2 CRDRL	CS, SB-166		1	
		All Other	CS, SB-166			
		Vessel Flange	SA336			
	· · · ·	······································	SA508-Cl 2			
		Vessel Shell				
		Beltline Weld	LAS weldment			
		Beltline Shell	SA302-Gr B,		and the second second second	
-		· · · ·	SA533-Gr B			
		All Other	SA302-Gr B,			
			SA533-Gr B			
		Top Head	SA302-Gr B,			
			SA533-Gr B			
		Support Skirt	SA533-Gr B	1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -		

Materialsa           SS,           SS,           SS,           SS,           SS,           Alloy I82           SA302-Gr B,           SA533-Gr B,           SA-193,           SA-193,           SA-540           SA-5608-Cl 2           SA508-Cl 2           SS, SB-167	Number <sup>b</sup> G-7, S-40, (S1)G-2, (S1)G-5, (S1)S-1, (S1)S-10	Agreement or Proposal Non-significant	Agreement or Proposal 7 Most of the carbon & low-alloy
SS, Alloy 182 SA302-Gr B, SA533-Gr B, SA533-Gr B, SA533-Gr B, SA540 SA-193, SA-193, SA508-Cl 2 SA508-Cl 2 SA50	G-7, S-40, (S1)G-2, (S1)S-1, (S1)S-10 (S1)S-10	Non-significant	Most of the carbon & low-alloy
SA302-Gr B, SA533-Gr B, SA-193, SA-540 SA-540 SA508-Cl 2 SA508-Cl	(S1)G-2, (S1)G-5, (S1)S-1, (S1)S-10 (S1)S-10		steel components are clad with
SA-193, SA-540 SA-540 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA508-C12 SA50	(S1)S-10 (S1)S-10		austenitic SS. Austenitic SSs & Ni-Cr-Fe allovs are resistant
SA-540 SA508-Cl 2 SA508-Cl 2 SA50	01-5(15)		to E/C and/or relatively low
SA508-CI 2 SA508-CI 2 SA508-CI 2 SA508-CI 2 SA508-CI 2 SS, SB-167 SS, SB-167 SB-167			CT.WOIL
SA508-CI 2           SA508-CI 2           SA508-CI 2           SA508-CI 2           SA508-CI 2           SA508-I           SA508-I           SS, SB-167           SB-167			
SA508-CI 2 SA508-CI 2 SS, SB-167 SB-167 SB-167			
SA508-CI 2 SS, SB-167 SB-167 SB-167			
SS, SB-167 SB-167 SB-167			
SS, SB-167 SB-167			
SB-167			
SS, SB-166			
CS, SB-166		х. х	
CS, SB-166	-		
CS, SB-166			
SA336			
SA508-CI 2			
LAS weldment			
SA302-Gr B,			
SA533-Gr B			
SA302-Gr B,			
SA533-Gr B			
SA302-Gr B,			
SA533-Gr B			
SA533-Gr B	Same as	Non-significant	Not exposed to flowing liquid.
	above		
	SB-167           SS, SB-166           SS, SB-166           CS, SB-166           CS, SB-166           CS, SB-166           SA336           SA508-C12           SA533-Gr B,           SA533-Gr B,	SA508-CI 2         SA508-CI 2         SS, SB-167         SB-167         SB-167         SS, SB-166         SS, SB-166         CS, SB-166         CS, SB-166         CS, SB-166         CS, SB-166         SA336         SA332         SA533         SA533 <td>SA508-C12         SA508-C12         SS, SB-167         SB-167         SB-167         SB-166         CS, SB-166         CS, SB-166         CS, SB-166         CS, SB-166         CS, SB-166         SA336         SA503-Gr B,         SA533-Gr B,         <t< td=""></t<></td>	SA508-C12         SA508-C12         SS, SB-167         SB-167         SB-167         SB-166         CS, SB-166         CS, SB-166         CS, SB-166         CS, SB-166         CS, SB-166         SA336         SA503-Gr B,         SA533-Gr B, <t< td=""></t<>

NUREG-1557

Aging-Related			<b> </b>	NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials <sup>a</sup>	Numberb	Agreement or Proposal	Agreement or Proposal
Fatigue	Cumulative	Nozzles	4	G-7,	Unresolved issue	NUMARC basis:
	fatigue	Other than Feedwater &		S-15 to	NUMARC proposal:	No fatigue cracking under
	damage	Uncapped		S-17,	Non-significant	expected operating conditions; &
		BWR/2 CRDRL	SA508-Cl 2	S-22,		design-basis or plant-specific
		Penetrations	4	S-46,	NRC proposal:	fatigue usage factor is <0.25 for
		Other than CRD Stub Tubes	SB-167	(S1)G-2,	Licensee verifies that plant-	CS in high stress & high oxygen
				(S1)G-5,	specific analyses, based on a	service, & is <0.4 for all other
		Safe Ends		(S1)S-1	conservative extrapolation of	material & service conditions.
		Other than Feedwater &		(S1)S-4	an enveloping set of actual	
		BWR/2 CRDRL	SS, SB-166		plant transients, demonstrate	
		Vessel Shell		Open	that fatigue cumulative usage	
		Beltline Weld	LAS weldment	issues:	factor will be <1; & analysis	
				G-8 to	should consider the effects of	
		Beltline Shell	SA302-Gr B,	G-11, &	BWR environment, e.g.,	
			SA533-Gr B	S-47	coolant chemistry, loading	
		All Other	SA302-Gr B,		frequency, & temp.	
		· · · · · · · · · · · · · · · · · · ·	SA533-Gr B			
		Top Head	SA302-Gr B,			
		-	SA533-Gr B			
		Attachment Welds	SS, Alloy 182			
Fatigue	Cumulative	Bottom Head	SA302-Gr B,	Same as	Unresolved issue	NUMARC basis:
-	fatigue		SA533-Gr B	above &	NUMARC proposal:	Same as above
	damage			G-12,	Non-significant for all BWRs	
	_			S-17	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	

<b></b>		······	I		l	
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materialsa	Numberb	Agreement or Proposal	Agreement or Proposal
Fotigue	Cumulative	Vessel Flange	54336	Same as	Unresolved issue	NUMARC basis
Faugue	fatigue	vesser Flange	SA508 C1 2	for Vessel	NUMARC proposal:	Same as for Vessel shell
	damada		57500-012	chell &	Non-significant	balle us for vesser shen
	uamage			SHELL Q	NDC proposal: See Vessel	
				5-17	shell & ASME Sect XI task	
				Onen	group has identified fatigue as	
				lisque	a significant APDM for BWR	· · ·
				S-21	vessel flange.	
Fatigue	Cumulative	Closure Studs	SA-193,	G-18,	Unresolved issue	NUMARC basis: Verification of
	fatigue		SA-540	S-41,		continued adequacy of fatigue
	damage	Nozzles		(S1)S-4,	NUMARC proposal:	design basis through reanalysis
		Feedwater	SA508-C1 2	(S1)S-7,	ASME Sect. III, <sup>2</sup> Subsect. NB	of usage factor.
		Safe Ends		(S1)G-12,	reanalysis of usage factor, re-	ASME Sect. XI, <sup>6</sup> Subsect. IWB,
		Feedwater	SA508-Cl 2	G-13,	grouping design-basis tran-	inspection & testing programs
		Uncapped		G-14,	sients, actual plant tran-sients,	that include, exam. category B-
		BWR/2 CRDRL	SA508-Cl 2	S-10,	cycle monitoring, & partial	G-1 for closure studs, rec-
		· ·		S-57 to	cycle counting; ASME	ommendations of RICSIL-055,8
				S-59,	Sect. XI, <sup>6</sup> Subsect. IWB, in-	& replacement in accordance
				(S1)S-15,	spection requirements of IWB-	with Reg. Guide 1.65; <sup>9</sup> cate-
				(S1)S-16,	2500-1, augmented by RICSIL	gories B-D & B-F for nozzles &
				(S1)S-8,	055 & RG 1.65 for closure	safe ends, additional require-
			5	(S1)S-20	studs, NUREG 0313, Generic	ments of NUREG 0313 <sup>10</sup> im-
					letter 88-01, & NUREG 0619	plemented by Generic letter 88-
				Open	for safe ends & nozzles; &	01, <sup>11</sup> leakage monitoring,
				issues	evaluation, defect repair, &	establishing plant-specific re-
				G-8 to	replacement.	furbishment period for feed-
				G-11, &		water sparger, & conformity
				S-47	NRC proposal:	with guidelines of NUREG
					Same as for vessel shell	0619; <sup>16</sup> & evaluation, defect
1. A.						repair, and replacement.

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

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials <sup>a</sup>	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	Nozzles Uncapped BWR/2 CRDRL	SA508-Cl 2	G-13, G-14, S-10, S-57 to S-59, (S1)S-15, (S1)S-16	Current practices to be en- hanced, 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; <sup>6</sup> water chemistry control; flaw evaluation-repair; & capping the nozzle.

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

<sup>a</sup> 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-7, (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

Aging-Related				NRC	•	
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materialsa	Numberb	Agreement or Proposal	Agreement or Proposal
					· ·	
General	Age related	Concrete- & steel- containment	Concrete &	G-2,	Unresolved issue	NUMARC basis:
	degradation	components	steel	S-8,	(One-time inspection)	The ARD mechanisms are eval-
	effects			S-31,		uated for significance using the
				S-32,	NUMARC proposal:	available research & industry
				S-34,	Resolution of the effects of ARD	data. If acceptance criteria
				S-71	mechanisms is based upon the	(including a review of plant per-
					review/evaluation of plant-	formance history, to assure that
				Open	specific features, including	contradictory evidence does not
				issues:	appropriate CLB documents/	exist) are satisfied, then the in-
				G-7,	information. General baseline	spection for that mechanism/
				S-7	inspection are not warranted if	component combination is not
				S-42,	the criteria used in the	needed.
				S-67,	evaluations are validated.	
				S-70		
					NRC proposal:	
					A one time focused inspection	
					of containment is proposed to	
					provide a reasonable level of	
1					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	
			1		the license renewal term.	

Aging-Related				NRC		Decis for
Degradation	Aging			Comment	NUMARC/NRC	Dasis Ior
Mechanism	Effects	Components	Materialsa	Numberb	Agreement or Proposal	Agreement or Proposal
	<u> </u>			L		
Freeze-	Scaling,	Concrete Containments	Concrete	G-12,	For concrete containment	Freeze-thaw is non-significant
thaw	cracking, &	Reinforced/Prestressed		S-5,	structures that meet the basis	for concrete containment
	spalling	Concrete Dome		S-38 to	requirements, freeze-thaw is	structures located in a geo-
		•Concrete Containment Wall		S-40	non-significant ARDM.	graphic regions of negligible
· · · · · · · · · · · · · · · · · · ·		Above Grade				weathering conditions
		•Concrete Containment Wall		Open	Open issue: NRC considers	(weathering index <100 day-
		Below Grade		issue	that potential freeze-thaw	inch/yr); <sup>1</sup> and if located in se-
		Concrete Basemat		S-10*	damage of the dome of the	vere (weathering index >500
		Free-Standing Steel Containment			concrete containments should	day-inch/yr) or moderate (100-
		with Flat Bottom & an Ice			be addressed.	500 day-inch/yr) weathering
		Condenser			and the second	conditions the concrete mix
		•Concrete Basemat			A second second second second second	design meets the air content &
				-	and the second second	water-to-cement ratio require-
						ments of ASTM C260 <sup>2</sup> or equiva-
						lently, the ASME Sect. III,
						Division 2, <sup>3</sup> paragraph
						CC 2231.7.1. <sup>3</sup>

Table B3. Bri	ef summaru o	of technical information	and NUMARC/NRC a	areements from PWR	l containment structu	res industry report
		J				

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

Leaching of	Increase of	Concrete Containments	Concrete	G-12,	For concrete containment	Leaching of calcium hydroxide is
Calcium	porosity &	Reinforced/Prestressed		S-5,	structures that meet the ba-	non-significant for concrete
Hydroxide	permeability	•Concrete Dome		S-11,	sis requirements, leaching	containment structures not ex-
		•Concrete Containment Wall		S-38 to	of calcium hydroxide is non-	posed to flowing water; and for
		Above Grade		S-40	significant ARDM.	structures that are exposed to
		•Concrete Containment Wall				flowing water but are con-
		Below Grade				structed using the guidance of
		•Concrete Basemat				ACI 201.2R-77 <sup>4</sup> to ensure
		Free-Standing Steel Containment				dense, well-cured concrete with
		with Flat Bottom & an Ice				low permeability and control
		Condenser				cracking through proper
		•Concrete Basemat				arrangement & distribution of
					1	reinforcement.

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials <sup>a</sup>	NRC Comment Number <sup>b</sup>	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 ba- sis requirements, aggressive chemical attack is non- significant ARDM.	Degradation caused by aggres- sive chemical attack is non- significant for concrete contain- ment structures not exposed to aggressive environment (pH <5.5), or to chloride or sulfate solutions beyond defined limits (>500 ppm chloride, <sup>5</sup> and 1500 ppm sulfate); <sup>6</sup> or if ex- posed to ground water that ex- ceeds the pH, chloride, sulfate limits the exposure is for inter- mittent periods only.
Aggressive Chemical Attack	Increase of porosity & permeability, cracking, & spalling	Concrete Containments <u>Reinforced/Prestressed</u> •Concrete Containment Wall <u>Below Grade</u> •Concrete Basemat Free-Standing Steel Containment with Flat Bottom & an Ice <u>Condenser</u> •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 proce- dures of Type A <sup>7</sup> integrated leak rate test, or in accordance with ASME Sect. XI, Subsect. IWL. <sup>8</sup> 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 aggres- sive groundwater (pH <5.5, chloride >500 ppm, & sulfate >1500 ppm), periodic inspec- tion of accessible concrete sur- faces as part of Type A inte- grated leak rate test performed under Appendix J, 10CFR50, <sup>7</sup> or in accordance with ASME Sect. XI, Subsect. IWL, <sup>8</sup> exam. category L-A, & guidelines of ACI 201.1. <sup>9</sup> Further evaluation for management of inaccessible areas is to be justified on a plant-specific basis.

Arting Dalatad				NDO		
Aging-Related				NRC		Dente for
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materialsa	Numberb	Agreement or Proposal	Agreement or Proposal
D	ъ ·		0	0.10	TT 1 1.	
Reaction with	Expansion	Concrete Containments	Concrete	G-12,	Unresolved issue	NUMARC basis:
Aggregates	& cracking	Reinforced/Prestressed		S-5,		Reactions with aggregates are
		Concrete Dome		S-38 to	NUMARC proposal:	non-significant for concrete
н. С		•Concrete Containment Wall		S-40	For concrete containment	containment structures con-
		Above Grade			structures that meet the basis	structed either from aggregate
		•Concrete Containment Wall		Open	requirements, reaction with	taken from geographic regions
		Below Grade		issue	aggregates is non-significant	other than those known to yield
		•Concrete Basemat		S-12*	ARDM.	aggregates suspected of or
		Free-Standing Steel Containment				known to cause alkali-aggregate
		with Flat Bottom & an Ice	1. T		NRC proposal:	reactions; <sup>4,10</sup> or from aggregate
		Condenser			Alkaline-aggregate reactions	that was investigated, tested,
		•Concrete Basemat			can not be ruled out. Tests	& subject to petrographic exam.
					involving aggregates alone are	conducted in accordance with
					not satisfactory in predicting	ASTM C295,11 or ASTM C227,12
			1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1. 1	and the second	aggregate performance.	which showed that the aggregate
					Alkaline-aggregate reaction	is non-reactive; or if the aggre-
					may occur after 25 or more	gate was examined & found
					years. Use of pozzolans & low	potentially reactive, the provi-
					alkali content cement may not	sions of ACI 201.2R-77 <sup>4</sup> were
					control reactions for concrete	followed.
					fabricated using sand-gravel	
					aggregates.	

)

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

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

			·			
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials <sup>a</sup>	Numberb	Agreement or Proposal	Agreement or Proposal
Eleveted	T and of	Companya Comtainmenta	Comenate	0.10	New significant if it mosts that	Degradation from exposure to
Elevated	LOSS OI	Concrete Containments	Concrete	G-12,	Non-significant if it meets the	Degradation from exposure to
1emp.	strength a	Reinforced/Prestressed		5-5,	basis requirements.	elevated temperatures is non-
	modulus	•Concrete Dome	embedded	S-19,		significant for concrete con-
		•Concrete Containment Wall	CS &	S-38 to		tainment structures maintained
		Above Grade	reinforcing	S-40		at operating temperatures<66°C
		•Concrete Containment Wall	CS (rebar) in	S-44,		150°F) and local area tempera-
		Below Grade	concrete	S-45		tures $<93^{\circ}$ C (200°F); <sup>3,13</sup> or for
		Concrete Basemat				structures that operate above
		Free-Standing Steel Containment				these limits, plant-specific
		with Flat Bottom & an Ice				justification is provided in
		Condenser		1	· · · ·	accordance with ACI 349-85.13
		Concrete Basemat	1			
Elevated	LOSS OI	Concrete Containments	00	Same as	Non-significant	Normal operating temperatures
remp.	strength &	Reinforced/Prestressed	CS I	above		within PWR containment
	modulus	•Dome Reinforcing Steel				structures are 49-66°C (120-
		•Cont. Wall Reinforcing Steel				150°F) which are well below the
		Above Grade				371°C (700°F) level at which the
		•Cont. Wall Reinforcing Steel				structural integrity of rebar/
		Below Grade				concrete combination begins to
	n	Basemat Reinforcing Steel				be significantly affected. <sup>14</sup>
		Free-Standing Steel Containment				
		with Flat Bottom & an Ice				
		Condenser				·
		Basemat Reinforcing Steel				
Disected	T			0	Numerican Constant	
Lievated	LOSS OI	Concrete Containments		Same as	Non-significant	Pwk containment presuressing
Temp.	strength &	Prestressed		above		tendons are subjected to
	modulus	Prestressing Tendons	CS			temperatures <60°C (140°F).

B-32

Elevated Loss of Concrete Containments CS Same as for rebar for rebar for rebar modulus C-Ontainment Liner Int. Surface - COntainment Liner Int. Surface - Containment Liner Surface - Containment Liner Surface - Containment Liner Surface - Containment Liner Surface - Basemat Liner Interior Surface - Basemat Liner Containment streter Surface - Basemat Liner Containment Shell Int. Surface - Cylindrical Shell Int. Surface - Cylin	Aging–Related Degradation Mechanism	Aging Effects	Components	Materials <sup>a</sup>	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
modulus       •Containment Liner Int. Surface       itures are 49-66°C (120-150°F)         •Containment Liner Above       Grade Exterior Surface       (700°F) level at which the structure         •Containment Liner Below       Grade Exterior Surface       (700°F) level at which the structure         •Basemat Liner Interior Surface       •Basemat Liner Interior Surface       combination begins to be signif         •Basemat Liner Interior Surface       •Liner Anchors Above Gr.       •Liner Anchors Below Gr.         •Liner Anchors Below Gr.       •Liner Anchors Below Gr.       •Liner Anchors Below Gr.         •Containment Shell Int. Surface       •Containment Shell Int. Surface       •Containment Shell Int. Surface         •Containment Shell Int. Surface       •Containment Shell Int. Surface       •Containment Shell Int. Surface         •Containment Shell Int. Surface       •Containment Shell Int. Surface       •Containment Shell Int. Surface         •Containment Shell Int. Surface       •Containment Shell Int. Surface       •Containment Shell Int. Surface         •Containment Shell Int. Surface       •Containment Surface       •Containment Surface         •Containment Shell Interior Surface       •Containment Surface       •Containment Surface         •Containment Shell Interior Surface       •Dome Shell Interior Surface       •Containment         •Dome Shell Interior Surface       •Containment       •Li	Elevated Temp.	Loss of strength &	Concrete Containments Reinforced/Prestressed	cs	Same as for rebar	Non-significant	Normal operating temperatures within PWR containment struc-
Containment Liner Above Grade Exterior Surface Crotainment Liner Below Containment Liner Below Grade Exterior Surface Basemat Liner Interior Surface Basemat Liner Interior Surface eliner Anchors Above Gr. eliner Anchors Above Gr. eliner Anchors Below Gr. Free-Standing Cylindrical & Spherical Steel Containment with Elliptical Bottom •Containment Shell Int. Surface •Containment Shell Ext. Surf. •Embedded Shell Region Free-Standing Steel Cont. with Flat Bottom & an lee Condenser •Dome Shell Interior Surface •Cylindrical Shell Cont. with Flat Bottom & an lee Condenser •Dome Shell Exterior Surface •Cylindrical Shell Ext. Surface •Cylindrical Shell Region •Basemat Liner •Liner Anchors Common Components •Penetration Sleeves		modulus	•Containment Liner Int. Surface				tures are 49-66°C (120-150°F)
Grade Exterior Surface       [700°F] level at which the struct         • Containment Liner Below       tural integrity of rebar/concret         Grade Exterior Surface       ebasemat Liner Interior Surface         • Basemat Liner Exterior Surface       cantly affected.14         • Uiner Anchors Above Gr.       eliner Anchors Below Gr.         • Liner Anchors Below Gr.       entite Structure         • Uiner Anchors Below Gr.       entity affected.14         • Containment Shell Containment       with Elliptical Bottom         • Containment Shell Int. Surface       • Containment Shell Ext. Surf.         • Dimedded Shell Region       • Saad Pocket Region         • Free-Standing Steel Cont. with       Flat Bottom & an lee Condenser         • Dome Shell Interior Surface       • Cylindrical Shell Int. Surface         • Cylindrical Shell Int. Surface       • Cylindrical Shell Int. Surface         • Dome Shell Exterior Surface       • Dome Shell Interior Surface         • Dome Shell Interior Surface       • Cylindrical Shell Int. Surface         • Cylindrical Shell Region       • Basemat Liner         • Liner Anchors       • Dome Common Components         • Penetration Sleeves       • Anter Surface			•Containment Liner Above				which are well below the 371°C
<ul> <li>Containment Liner Below Grade Exterior Surface</li> <li>Basemat Liner Interior Surface</li> <li>Basemat Liner Exterior Surface</li> <li>Liner Anchors Above Gr.</li> <li>Liner Anchors Below Gr.</li> <li>Free-Standing Cylindrical &amp; Spherical Steel Containment with Elliptical Bottom</li> <li>Containment Shell Int. Surface</li> <li>Containment Shell Ext. Surf.</li> <li>Embedded Shell Region</li> <li>Free-Standing Steel Cont. with Flat Bottom &amp; an lee Condenser</li> <li>Dome Shell Interior Surface</li> <li>Cylindrical Shell Ext. Surface</li> <li>Cylindrical Shell Ext. Surface</li> <li>Cylindrical Shell Ext. Surface</li> <li>Some Shell Ext. Surface</li> <li>Cylindrical Shell Ext. Surface</li> <li>Cylindrical Shell Ext. Surface</li> <li>Cylindrical Shell Ext. Surface</li> <li>Some Shell Exterior Surface</li> <li>Cylindrical Shell Ext. Surface</li> <li>Cylindrical Shell Ext. Surface</li> <li>Embedded Shell Region</li> <li>Basemat Liner</li> <li>Liner Anchors</li> <li>Common Components</li> <li>Penetration Sleeves</li> </ul>			Grade Exterior Surface				(700°F) level at which the struc-
Grade Exterior Surface       combination begins to be signil         •Basemat Liner Interior Surface       cantly affected.14         •Basemat Liner Exterior Surface       cantly affected.14         •Liner Anchors Below Gr.          •Liner Anchors Below 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         •Cylindrical Shell Int. Surface       •Cylindrical Shell Int. Surface         •Cylindrical Shell Ext. Surface       •Cylindrical Shell Int. Surface         •Cylindrical Shell Interior Surface       •Cylindrical Shell Int. Surface         •Cylindrical Shell Ext. Surface       •Cylindrical Shell Ext. Surface         •Cylindrical Shell Ext. Surface       •Cylindrical Shell Region         •Basemat Liner       •Liner Anchors         Common Components       •Penetration Sleeves			•Containment Liner Below				tural integrity of rebar/concrete
•Basemat Liner Interior Surface       cantly affected. <sup>14</sup> •Basemat Liner Exterior Surface       cantly affected. <sup>14</sup> •Liner Anchors Below Gr.       Free-Standing Cylindrical &         Spherical Steel Containment       spherical Steel Containment         with Elliptical Bottom       •Containment Shell Int. Surface         •Containment Shell Int. Surface       •Containment Shell Region         •Sand Pocket Region       •Sand Pocket Region         •Free-Standing Steel Cont. with       Flat Bottom & an Ice Condenser         •Dome Shell Interior Surface       •Cylindrical Shell Int. Surface         •Cylindrical Shell Int. Surface       •Cylindrical Shell Int. Surface         •Cylindrical Shell Region       •Basemat Liner         •Liner Anchors       •Liner Anchors         •Dome Shell Ext. Surface       •Cylindrical Shell Region			Grade Exterior Surface				combination begins to be signifi-
•Basemat Liner Exterior Surface         •Liner Anchors Above Gr.         •Liner Anchors Below Gr.         •Liner Anchors Below 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 lee Condenser         •Dome Shell Exterior Surface         •Optimized Shell Int. Surface         •Cylindrical Shell Int. Surface         •Cylindrical Shell Ext. Surface         •Cylindrical Shell Ext. Surface         •Embedded Shell Region         •Basemat Liner         •Liner Anchors         Common Components         •Penetration Sleeves			•Basemat Liner Interior Surface				cantly affected. <sup>14</sup>
Liner Anchors Above Gr.     I.Iner Anchors Below Gr.     Free-Standing Cylindrical &     Spherical Steel Containment     with Elliptical Bottom     Containment Shell Int. Surface     Containment Shell Int. Surface     Containment Shell Ext. Surf.     Embedded Shell Region     Free-Standing Steel Cont. with     Flat Bottom & an lee Condenser     Dome Shell Interior Surface     Cylindrical Shell Int. Surface     Embedded Shell Region     Basemat Liner     Liner Anchors     Common Components     Penetration Sleeves			•Basemat Liner Exterior Surface				
•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 Ext. Surface         •Cylindrical Shell Ext. Surface         •Embedded Shell Region         •Basemat Liner         •Liner Anchors         Common Components         •Penetration Sleeves			•Liner Anchors Above Gr.				
Free-Standing Cylindrical & Spherical Steel Containment with Elliptical Bottom •Containment Shell Int. Surface •Containment Shell Int. Surface •Embedded Shell Region •Sand Pocket Region Free-Standing Steel Cont. with Flat Bottom & an Ice Condenser •Dome Shell Interior Surface •Dome Shell Interior Surface •Opimerical Shell Int. Surface •Cylindrical Shell Int. Surface •Cylindrical Shell Ext. Surface •Embedded Shell Region •Basemat Liner •Liner Anchors Common Components •Penetration Sleeves			•Liner Anchors Below Gr.				
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 Interior Surface         •Cylindrical Shell Int. Surface         •Cylindrical Shell Ext. Surface         •Embedded Shell Region         •Basemat Liner         •Liner Anchors         Common Components         •Penetration Sleeves			Free-Standing Cylindrical &				
with Elliptical Bottom         •Containment Shell Int. Surface         •Containment Shell Ext. Surf.         •Embedded Shell Region         •Sand Pocket Region         •Sand Pocket Region         •Free-Standing Steel Cont. with         Flat Bottom & an Ice Condenser         •Dome Shell Interior Surface         •Dome Shell Interior Surface         •Cylindrical Shell Int. Surface         •Cylindrical Shell Int. Surface         •Embedded Shell Region         •Basemat Liner         •Liner Anchors         Common Components         •Penetration Sleeves			Spherical Steel Containment				
Containment Shell Int. Surface     Containment Shell Ext. Surf.     Embedded Shell Region     Sand Pocket Region     Free-Standing Steel Cont. with     Flat Bottom & an Ice Condenser     Obme Shell Interior Surface     Obme Shell Interior Surface     Obme Shell Interior Surface     Cylindrical Shell Int. Surface     Cylindrical Shell Ext. Surface     Embedded Shell Region     Basemat Liner     Liner Anchors     Common Components     Penetration Sleeves			with Elliptical Bottom				
•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			<ul> <li>Containment Shell Int. Surface</li> </ul>				
•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			•Containment Shell Ext. Surf.				
•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			•Embedded Shell Region				
Free-Standing Steel Cont. with         Flat Bottom & an Ice Condenser         •Dome Shell Interior Surface         •Dome Shell Interior Surface         •Cylindrical Shell Int. Surface         •Cylindrical Shell Ext. Surface         •Embedded Shell Region         •Basemat Liner         •Liner Anchors         Common Components         •Penetration Sleeves			•Sand Pocket Region				
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			Free-Standing Steel Cont. with				
•Dome Shell Interior Surface•Dome Shell Exterior Surface•Cylindrical Shell Int. Surface•Cylindrical Shell Ext. Surface•Embedded Shell Region•Basemat Liner•Liner AnchorsCommon Components•Penetration Sleeves			Flat Bottom & an Ice Condenser				
•Dome Shell Exterior Surface     •Cylindrical Shell Int. Surface     •Cylindrical Shell Ext. Surface     •Embedded Shell Region     •Basemat Liner     •Liner Anchors Common Components     •Penetration Sleeves			•Dome Shell Interior Surface				
•Cylindrical Shell Int. Surface         •Cylindrical Shell Ext. Surface         •Embedded Shell Region         •Basemat Liner         •Liner Anchors         Common Components         •Penetration Sleeves			•Dome Shell Exterior Surface				
•Cylindrical Shell Ext. Surface         •Embedded Shell Region         •Basemat Liner         •Liner Anchors         Common Components         •Penetration Sleeves			•Cylindrical Shell Int. Surface		l .		
•Embedded Shell Region     •Basemat Liner     •Liner Anchors Common Components     •Penetration Sleeves			•Cylindrical Shell Ext. Surface				
•Basemat Liner     •Liner Anchors     Common Components     •Penetration Sleeves			•Embedded Shell Region				
•Liner Anchors Common Components      •Penetration Sleeves			•Basemat Liner	]			
Common Components  •Penetration Sleeves			•Liner Anchors	1			
Penetration Sleeves	n an an an Araba. An Araba		Common Components				$\frac{1}{2} \mathbf{A}_{i} = \frac{1}{2} \left[ \frac{1}{2} \mathbf{A}_{i} + \frac{1}{2} \mathbf{A}_{i} \right] + \frac{1}{2} \left[ \frac{1}{2} \mathbf{A}_{i} + \frac{1}{2} \mathbf{A}_{i} \right] + \frac{1}{2} \left[ \frac{1}{2} \mathbf{A}_{i} + \frac{1}{2} \mathbf{A}_{i} \right]$
	1. 1		Penetration Sleeves	and the second	1. A.	and the second	
•Penetration Bellows SS, CS		an a	•Penetration Bellows	SS, CS			
Personnel Airlock     CS			Personnel Airlock	CS	1.00		
•Equipment Hatches	1. S.		•Equipment Hatches	· ·			

Aging–Related Degradation Mechanism	Aging Effects	Components	Materialsa	NRC Comment Number <sup>b</sup>	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 $(10^{19} \text{ n/cm}^2 \text{ & } 10^{10} \text{ rads, respectively}).^{5,15}$
			:			
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 expo- sure experienced by reinforced concrete PWR containment structures during the license renewal term is far below the level of 10 <sup>19</sup> n/cm <sup>2</sup> for degra- dation of reinforcing steel. <sup>16</sup>
Irradiation of	Loss of	Concrete Containments		Same as	Non-significant	PWR containment tendons &
Steel	fracture toughness	Prestressed  Prestressing Tendons	cs	above		corrosion inhibitors will not re- ceive enough radiation exposure during the license renewal term to incur age related degradation (<4 x $10^{16}$ n/cm <sup>2</sup> , & $10^{10}$ rads, respectively). <sup>13</sup>

A rive of Dalata d			<u> </u>		l	T
Aging-Related	A stim st			Commont		Basis for
Degradation	Aging	Componenta	Motoriolal	Numberb	Advectment or Proposal	Agreement or Proposal
Mechanishi	Effects	Components	Materials	ivuilibei~	Agreement of Proposal	Agreement of Troposal
Irradiation of	Loss of	Concrete Containments	<u></u>	G-12,	Non-significant	The cumulative radiation expo-
Steel	fracture	Reinforced/Prestressed	CS	S-5,		sure experienced by concrete
	toughness	•Containment Liner Int. Surface		S-38 to		PWR containment liners or free-
		•Containment Liner Above		S-40,	· · ·	standing steel containment
		Grade Exterior Surface	· · ·	S-46,		shells throughout the license re-
		•Containment Liner Below		S-51,		newal term is far below the level
		Grade Exterior Surface		S-55		of 2 x 10 <sup>17</sup> n/cm <sup>2</sup> (>1 MeV)
2.	$  _{\mathcal{L}_{p}} =   _{\mathcal{L}_{p}$	•Basemat Liner Interior Surface				which could cause a change in
		Basemat Liner Exterior Surface				mechanical or physical proper-
		•Liner Anchors Above Gr.				ties. <sup>17</sup>
		•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	and the second second	· ·		
		Liner Anchors	and the second	ļ		
		Common Components	and the second	Î.		
n an	е. с.	Penetration Sleeves	· · · · ·			
		•Penetration Bellows	SS, CS			
	-	•Personnel Airlock	CS			
		•Equipment Hatches				

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials <sup>a</sup>	NRC Comment Number <sup>b</sup>	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 envi- ronment, pH <11.5 or chlorides >500 ppm; <sup>18</sup> 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), ade- quate air entrainment (3-6%), low permeability, and designed in accordance with ACI 318 <sup>5</sup> or ASME Sect. III, Div. 2. <sup>3</sup>
Corrosion of Embedded Steel	Cracking. spalling, loss of bond, & loss of material	Concrete Containments <u>Reinforced/Prestressed</u> •Concrete Containment Wall <u>Below Grade</u> •Concrete Basemat •Containment Wall Reinforcing <u>Steel Below Grade</u> •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	Accessible concrete surfaces are periodically examined in accordance with procedures of Type A <sup>7</sup> integrated leak rate test, or ASME Sect. XI, Subsect. IWL. <sup>8</sup> 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, <sup>7</sup> or in accordance with ASME Sect. XI, Subsect. IWL, <sup>8</sup> exam. category L-A, & guidelines of ACI 201.1. <sup>9</sup> 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).

# NUREG-1557

B-36

		T	l			· · · · · · · · · · · · · · · · · · ·
Aging-Related	A			NRC		De sie fan
Degradation	Aging			Comment	NUMARC/NRC	Basis ior
Mechanism	Effects	Components	Materialsa	Number	Agreement or Proposal	Agreement or Proposal
Corrosion of	Loss of	Concrete Containments		G-5,	For PWR containment liners	Galvanic corrosion & corrosion
Structural	material	Reinforced/Prestressed	CS	G-12,	that meet the basis require-	due to aggressive aqueous so-
Steel & Liner		•Containment Liner Interior		S-5,	ments, corrosion is non-	lutions will not occur if dissimi-
		Surface		S-16,	significant ARDM.	lar metals are not used in con-
		•Containment Liner Above		S-38 to		struction & if aggressive ground
		Grade Exterior Surface		S-40,		water (chlorides >500 ppm) is
		•Basemat Liner Interior Surface		S-62		not present. SCC is not signifi-
		•Liner Anchors Above Grade				cant because PWR containment
		Common Components				liners only experience compres-
		Penetration Sleeves		· .		sive stresses due to dead load &
		•Dissimilar Metal Welds				prestress.
		Personnel Airlock				
		•Equipment Hatches	· · · ·			· ·
Occurring of	T and of					Calvania compation & SCC are
Corrosion of	LOSS OI	Free-Standing Cylindrical &		G-5,	For PWR free standing steel	Galvanic corrosion & SCC are
Structural	materiai	spherical Steel Containment	00	G-12,	containment that meet the	nioma if diagimilar metals are
Sleer a		Containment Shall Interior	65	5-5,	basis requirements, corrosion	net used in the construction of
Liner		•Containment Shell Interior		5-0, 5-16	is non-significant ARDM.	PWR free-standing steel con-
		Containment Shell Exterior		S-38 to		tainment: & in the case of SS
		Surface		S-30 10		bellows assemblies for CS vent
		Free-Standing Steel Containment		5 40		lines or nine sleeves if the mat-
		with Flat Bottom & an Ice				erials are protected by shields
		Condenser				from corrosive environment.
		•Dome Shell Interior Surface				
		•Dome Shell Exterior Surface				
		•Cvlindrical Shell Interior				
		Surface				
		•Cylindrical Shell Exterior				
		Surface				
		Common Components		1		
		Penetration Bellows	SS			

,

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

Aging-Related	Aging			NRC		Bosis for
Mechanism	Fffecte	Components	Materialea	Numberb	Adreement or Proposal	Agreement or Proposal
Weenamsin	Ellects	components	Matchals-	Number~	Agreement of Proposal	Agreement of Troposal
Corrosion of	Loss of	Concrete Containments		G-8,	Unresolved issue	NUMARC basis:
Tendons	material	Prestressed		G-9,	NUMARC proposal:	Examination of tendon anchor-
		•Prestressing Tendons	cs	G-19,	RG 1.35 & ASME Sect XI,	age hardware in accordance with
				G-11,	Subsect. IWL require testing &	the provisions of RG 1.35 <sup>21</sup> or
		· · · ·		G-16,	examination of tendons & leak-	the requirements of ASME
				S-9,	age of corrosion protection	Sect. XI, <sup>8</sup> Subsect. IWL, includ-
				S-18,	medium; VT-1 includes anchor	ing visual examination of ten-
		,		S-50,	head, bearing plates, wedges,	don anchorage hardware, evalu-
				S-64	buttonheads, shims, & con-	ation of corrosion protection
			· · ·		crete; acceptance criteria IWL-	medium, & identification &
	•		1. Sec. 1. Sec	Open	3221.2 include absence of	testing of any free water; repair
				issue	physical damage, corrosion	& replacement; are effective in
				S-7,	limits; & minimum specified	managing degradation by corro-
]		and the second		S-61	material properties; IWL-2525-	sion of prestressing tendons &
					1 examines corrosion protec-	anchor heads.
		and the second			tion 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.	

<u>.</u>		1	I	NDO		T
Aging-Related	A			NRC		Rosis for
Degradation	Aging	Componente	Motorialaa	Numberb	Agreement or Proposal	Agreement or Proposal
Mechanishi	Effects	Components	Materials	Taninner~	Agreement of 110posal	Agreement of Troposal
Fatigue	Cumulative	Concrete Containments		G-12,	Non-significant	Containment concrete, rein-
	fatigue	Reinforced/Prestressed	Concrete	S-5,		forcing steel, prestressing sys-
	damage	Concrete Dome	including	S-21,		tems, steel liners, & free-
		•Concrete Containment Wall	embedded CS	S-38 to		standing steel containments are
		Above Grade	&	S-40	•	designed to have good fatigue
		•Concrete Containment Wall	reinforcing			strength properties (10 <sup>5</sup> cycles)
		Below Grade	CS (rebar)			of below yield load in accordance
-		Concrete Basemat	in concrete			with ASME Sect. III, Division $2,^3$
		•Dome Reinforcing Steel				& ACI 215R-74 <sup>23</sup> codes.
		•Containment Wall Reinforcing				Potential low-cycle fatigue due to
		Steel Above Grade				localized elevated temperatures
		•Containment Wall Reinforcing				are not anticipated to be signifi-
		Steel Below Grade				cant.
		•Basemat Reinforcing Steel				
		Free-Standing Cylindrical &				
		Spherical Steel Containment	~~			
		with Elliptical Bottom	cs			
		•Containment Shell Int. Surface				1
		•Containment Shell Ext. Surf.				
		Free-Standing Steel Cont. with				
		Flat Bottom & an Ice Condenser				×
		•Dome Shell Interior Surface				
		•Dome Shell Exterior Surface				
1		•Cylindrical Shell Int. Surface				
		•Cylindrical Shell Ext. Surface				
		•Concrete Basemat				
		•Basemat Reinforcing Steel				
		Common Components				
		Personnel Airlock				
		•Equipment Hatches				

		<b>I</b>	T	100	I	
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materialsa	Numberb	Agreement or Proposal	Agreement or Proposal
Fatigue	Oumulativa	Common Components	1	0.11	I Immer hand to such	NUMADC haste
raugue	Cumulative	Common Components		G-11,	Ulliesowed issue	NUMARC DUSIS.
	latigue	•Penetration Sleeves	CS	G-16,		Fatigue reanalysis conducted
	damage	•Penetration Bellows	SS, CS	S-14,	NUMARC proposal:	in accordance with ASME Sect.
				S-64,	Fatigue reanalysis of penetra-	III, <sup>20</sup> Subsect. NB, to show that
					tions in accordance with ASME	fatigue usage factors are main-
				Open	Sect. III, Subsect. NB; <sup>20</sup> & ISI	tained below unity throughout
				issue	in accordance with ASME Sect.	the license renewal term, moni-
			]	G-3,	XI, Subsect. IWE, <sup>8</sup> exam. cate-	toring of penetration tempera-
				G-4,	gory E-B requires visual VT-1	tures may be required to estab-
	1			S-68	of containment penetration	lish the magnitude & frequency
1					welds, including bellows seal	of transients; ISI in accordance
					circumferential weld.	with ASME Sect. XI, <sup>8</sup> Subsect.
	1	and the second	I			IWE, to ensure that component
1					NRC proposal:	integrity is maintained in the
					Sensitivity evaluations & ap-	presence of known or suspected
	ļ				propriate references should be	fatigue damage, including a flaw;
					included for fatigue of bellows	are effective to manage the ef-
					assemblies. Fatigue of pene-	fects of fatigue damage
					tration sleeve anchors can be	accumulation or fatigue crack
					induced by thermal cyclic	growth.
					loading & may not be de-	
					tectable by the leak rate tests.	
					tootable by the roan rate tootbi	
Loss of	Reduction	Concrete Containments		G-9,	Inspection & load monitoring	Periodic monitoring of prestress-
Prestress	of design	Prestressed		G-14,	to detect progressive reduc-	ing losses in accordance with
	margin	•Prestressing Tendons*	cs	S-18,	tions in the levels of prestress;	tendon lift-off test of RG 1.35; <sup>21</sup>
	Ű	Ŭ,		S-33.	evaluation for the license	validation with predictions of
				S-45.	renewal term using RG 1.35:21	prestressing loss; identification
	]			S-47.	& corrective action.	of reportable conditions of
			· · ·	S-48.		RG1.35: documentation of RG
				S-52	and the second	1.16: <sup>22</sup> & plant-specific
				S-53		evaluation & corrective actions
						are effective in managing the
						affects of pre-stressing loss
L		L		L	l	enects of pre-stressing loss.

\* Rock anchors are outside the scope of this review.

B-41

NUREG-1557

Aging-Related	Artina			NRC		Basis for
Degradation	Aging	Common to	Madanda 1a9	Normhand	NUMARC/NRC	Adreement or Proposal
Mechanism	Ellects	Components	Materials«	Number	Agreement of Proposal	Agreement of Proposal
Concrete	Loss of	Concrete Containments	Concrete	S-59	For concrete containment	Adverse effects of concrete
Interaction	strength	Reinforced/Prestressed		1	structures that meet the ba-	interactions with aluminum
with		Concrete Dome		ļ	sis requirements, concrete	would have been identified
Aluminum		Concrete Containment Wall			interaction with aluminum is	during the initial acceptance
1	1	Above Grade		1	non-significant ARDM.	test prior to initial operation.
	]	•Concrete Containment Wall				If no degradation of concrete
		Below Grade				strength was noted during
and the second second		Concrete Basemat				initial structural testing, or if
						aluminum piping were not
		Free-Standing Steel Containment				used for concrete placement,
		with Flat Bottom & an Ice				then concrete interaction with
and the second second		Condenser				aluminum is not significant.
	ļ	•Concrete Basemat	<u> </u>			
Settlement	Cracks,	Concrete Containments	Concrete		Unresolved issue	NUMARC basis:
	distortion,	Reinforced/Prestressed		Open		Structure settlement monitor-
	increase in	Concrete Basemat		issue	NUMARC proposal:	ing initiated during construc-
	component	Free-Standing Steel Containment		S-63*	Structure settlement monitor-	tion phase to confirm that actual
	stress level	with Flat Bottom & an Ice			ing during construction, &	settlement is consistent with the
		Condenser			continued monitoring during	allowances included in design
		•Concrete Basemat			operation for sites with soft soil	basis, & continued settlement
					and/or significant changes in	monitoring during operation for
					ground water conditions.	sites with soft soil and/or
						significant changes in ground
						water conditions.
-					NRC proposal:	
		<ul> <li>A state of the sta</li></ul>			Effect of settlement of the PWR	
					containments need to be	
					evaluated.	

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

Aging–Related Degradation Mechanism	Aging Effects	Components	Materialsa	NRC Comment Number <sup>b</sup>	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 •Containment Shell Int. Surf. •Containment Shell Ext. Surf.	CS	None	For containment structures that meet the basis require- ments, 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
		•Embedded Shell Region •Sand Pocket Region Free-Standing Steel Containment with Flat Bottom & an Ice Condenser •Dome Shell Interior Surface				strain aging is non-significant for free standing steel contain- ment structures that were not cold worked; or if cold worked during the forming process, the plates were normalized or stress
		•Dome Shell Exterior Surface     •Operation Shell Exterior Surface     •Cylindrical Shell Int. Surface     •Cylindrical Shell Ext. Surface     Common Components     •Pagetration Slogues				relieved or both after forming with minimal (<5%) subsequent cold working.
		Penetration Bellows     Personnel Airlock     Equipment Hatches	CS CS			

<sup>a</sup> Designation for steel materials are as follows: SS = stainless steel, CS = carbon steel.

<sup>b</sup> 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:**

1. ASTM C33-82, "Standard Specification for Concrete Aggregates," American Society for Testing and Materials, Philadelphia, PA, 1982.

- 2. ASTM C260-77, "Specification for Air Entraining Admixture for Concrete," American Society for Testing and Materials, Philadelphia, PA, 1977.
- 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.
- 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.
- 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, 1992 Edition with 1992 Addenda.

Subsection IWE: "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants." Subsection IWL: "Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants."

- 9. ACI 201.1R-68, "Guide for Making a Condition Survey of Concrete in Service," American Concrete Institute, Detroit, MI, Revised 1984.
- 10. "Petrographic Identification of Reactive Constituents in Concrete Aggregate," B. Mather, ASTM Proc. Vol. 48, American Society of Testing and Materials, Philadelphia, PA, pp. 1120-1125, 1948.
- 11. ASTM C295-85, "Practice for Petrographic Examination of Aggregates for Concrete," American Society of Testing and Materials, Philadelphia, PA.
- 12. ASTM C227-87, "Test Method for Potential Alkali Reactivity of Cement-Aggregate Combination," American Society of Testing and Materials, Philadelphia, PA.
- 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.
- 15. ACI Publication SP-55, "The Effects of Nuclear Radiation on the Mechanical Properties of Concrete," H. R. Hilsdorf, J. Kropp, and H. J. Koch, Douglas McHenry Intl. Symp. on Concrete and Concrete Structures, American Concrete Institute, 1978.
- 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.
- 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 Interior 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			[	NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials <sup>a</sup>	Numberb	Agreement or Proposal	Agreement or Proposal
Freeze-thaw	Scaling,	Mark III Concrete Containments*		G-1,	For Mark III concrete con-	Freeze-thaw is non-significant
	cracking, &		Concrete	G-2,	tainment components that	for Mark III concrete contain-
	spalling	•Concrete Containment Walls		G-3,	meet the basis requirements,	ment components located in a
1		Above Grade		G-4,	freeze-thaw is a non-signifi-	geographic region of negligible
		•Concrete Containment Walls		G-11,	cant ARDM.§	weathering conditions (weath-
		Below Grade		G-16,		ering index <100 day-inch/yr);
		•Concrete Dome		G-17,		and if located in severe (weath-
				G-19,		ering index >500 day-inch/yr) or
			1. Contract (1997)	G-21,		moderate (100-500 day-inch/yr)
				S-74,		weathering conditions the
1.1				S-81,		concrete mix design meets the
						air content and water-to-cement
						ratio requirements of ASTM-
		· · · · · · · · · · · · · · · · · · ·		1. Contract (1. Contract)		C260-77 <sup>1</sup> or ASME Sect. III,
						Division 2, <sup>2</sup> CC-2231.7.1; or the
						susceptible surfaces are
		and the second				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				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials <sup>a</sup>	Number <sup>b</sup>	Agreement or Proposal	Agreement or Proposal
Leaching of	Increase of	Mark I Concrete Containments		G-1,	For containment components	Leaching of calcium hydroxide is
Calcium	porosity &		Concrete	G-2,	that meet the basis require-	non-significant for con-
Hydroxide	permeability	Drywell Concrete		G-3,	ments, leaching of calcium	tainment concrete components
		•Torus Concrete		G-4,	hydroxide is a non-significant	not exposed to flowing water;
		Mark II Concrete Containments		G-11,	ARDM.	and for structures that are ex-
				G-16,		posed to flowing water but are
		Containment Concrete		G-17,		constructed using the guidance
	5.0	•Concrete Basemat		G-19,		of ACI 201.2R-77 <sup>3</sup> to ensure
		Mark III Concrete Containments		G-21,		dense, well-cured concrete with
				S-74,		low permeability and con-
		•Concrete Containment Wall		S-81		trol cracking through proper
		Above Grade				arrangement and distribution
		Concrete Containment Wall				of reinforcement.
		Below Grade				
		Concrete Dome				$(A_{i},A_{i}) = (A_{i},A_{i}) + (A_{i},A_{i}$
		Concrete Basemat				
		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 <sup>a</sup>	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Aggressive	Increase of	Mark I Concrete Containment		G-1.	For containment concrete	Degradation caused by ag-
Chemical	porosity &	· · · · · · · · · · · · · · · · · · ·	Concrete	G-2,	components that meet the	gressive chemical attack is
Attack	permeability,	Drywell Concrete		G-3,	basis requirements, aggres-	non-significant for contain-
	cracking, &	•Torus Concrete		G-4,	sive chemical attack is a non-	ment components not exposed
	spalling	Mark III Concrete Containments		G-11,	significant ARDM.	to aggressive environment (pH
				G-16,		<5.5), or to chloride or sulfate
		•Concrete Containment Walls		G-17,		solutions beyond defined limits
		Above Grade		G-19,		(>500 ppm chloride, <sup>4</sup> and
		•Concrete Dome		G-21,		>1500 ppm sulfate); <sup>5</sup> or if ex-
				S-71,		posed to groundwater that ex-
	-			S-74,		ceeds the pH, chloride, sulfate
				S-81,		limits the exposure is for in-
		· · · · · · · · · · · · · · · · · · ·		5-98	· · · · · · · · · · · · · · · · · · ·	termitient periods only.
Aggressive	Increase of	Mark II Concrete Containments		G-1 to G-	Management for the effects of	Plant-specific program is to be
Chemical	porosity &		Concrete	4,	aggressive chemical of con-	justified for the inaccessible
Attack	permeability,	•Concrete Basemat		G-11,	crete surfaces that are not	areas.
	cracking, &	Mark III Concrete Containments		G-16,	periodically examined due to	
	spalling			G-17,	inaccessibility requires plant-	
		•Concrete Containment Walls		G-19 to	specific evaluation.	
		Below Grade	4.	G-21,		
		Concrete Basemat	*	S-71,		
		Mark III Steel Containments		S-98		
		•Concrete Basemat				

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

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

§ 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 <sup>a</sup>	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Atmospheric	Loss of	Common Components		G-1.	For containment steel com-	Atmospheric corrosion is not a
Corrosion	material	Penetration Sleeves	cs	G-2.	ponents that meet basis re-	significant ARD for steel con-
		Penetration Bellows	SS	G-3.	quirements, atmospheric	tainment components fabri-
		Personnel Airlock	CS	G-4.	corrosion is a non-significant	cated from stainless steel, or
		•Equipment Hatches		G-7.	ARDM.	for components having intact
		•CRD Hatch		G-11.		protective coatings, or for
		Mark I Steel Containment		G-16.		components having a corro-
		Orvwell Interior Surface	cs	G-17.		sion allowance $\geq 1/32$ inch.
		•Drywell Head		G-19.		Austenitic SS is corrosion re-
		•Torus Interior Surface		G-21.		sistant. The atmospheric cor-
		•Torus Exterior Surface		S-3.		rosion for carbon and low alloy
l		•Vent Lines		S-9.		steels without protective
-		•Ring Girder		S-12 to		coatings is less than 0.5 mils
		•Vent Line Bellows	SS	S-15.		per vear or $<1/32$ inches
		•Vent Header	CS	S-21.		(0.03125  inches) for a 60-year
		•Downcomer & Bracing		S-32 to		period.9,10
		•Vent System Supports		S-35.		1
		Torus Seismic Restraints		S-38.		
		•Torus Support	CS. graphite	S-43.		
		Columns/Saddles	,, 8 <u>-</u>	S-46.		
		Mark II Steel Containments		S-48.		
		Drywell Interior Surface	CS	S-61,		
		•Drywell Head		S-63		
		•Suppression Chamber Interior				
		Surface				
		•Downcomer Pipes & Bracing				
		Mark III Steel Containments	CS	1		
		•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	Aging	·		NRC		Basis for
Mechanism	Fffects	Components	Materialea	Numberb	Adreement or Proposal	Agreement or Proposal
meenamoni	Directo		Materiais	Humber	Agreement of Troposal	Agreement of Troposal
Atmospheric	Loss of	Mark III Steel Containments		Same as	Same as Mark I Steel	Same as Mark I Steel
Corrosion	material	•Basemat Liner	SS	Mark I	Containment	Containment
		Mark I Concrete Containments		Cont.		
		•Vent Lines	cs			
		•Vent Headers				
		•Vent Line Bellows	SS			
		•Vent System Supports	CS			
		•Drywell Head				
		Mark II Concrete Containments				
1		•Drywell Head				
Atmospheric	Loss of	Mark I Steel Containments		G-5,	The Examination Categories	Appendix J of 10CFR50 requires
Corrosion	material			G-10,	E-A, E-P, & E-C of ASME	general inspection of contain-
		•Drywell Exterior Surface	cs	G-12,	Sect. XI, Subsect. IWE <sup>11</sup> are	ment & component surfaces
		•ECCS Suction Header"		G-19,	required to be performed in	prior to Type A leak rate test. If
				S-6,	conjunction with IOCFR50,	there is any evidence of degra-
		•Ocean Plants with Uncoated		S-18,	Appendix J, Type A leak rate	dation, Type A tests shall not be
		CS Component Surfaces		5-37,	test. <sup>12</sup>	formed until corrective action is
		•Uncoated Submerged CS		5-85		taken. Exam. Category E-P of
		Components				ASME Sect. IX, Subsect. IWE
		Deres li Esterior Soufra				provides v1-3 examination on
		•Drywell Exterior Surface				accessible containment pressure
		•Ocean Plants with Uncoated				boundary & E-A provides v1-3
	н. 1917 - С.	CS Surfaces				exam. for the containment shell
		• Oncoated Submerged CS				weids. Exam. E-C provides VI-1
		Components				hispection on surface areas
		Surface				approximation of the areas are found
		Surrace				to be defective volumetric
						examination is required.

\* 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.

B-51

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

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

§ 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				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials <sup>a</sup>	Number <sup>b</sup>	Agreement or Proposal	Agreement or Proposal
Local	Loss of	Common Components		G-5,	Periodically examined by the	IWE-1240 of ASME Sect. XI,
Corrosion	material	•Dissimilar Metal Welds*	CS welded	G-10,	Exam. Category E-C under the	Subsect. IWE provides for the
			with SS	G-12,	provisions of IWE-1240	identification of accessible sur-
		Mark I Steel Containments		S-6,	of ASME Sect. XI, Subsect.	face areas likely to experience is
		•Torus Interior Surface at	CS	S-10,	IWE. <sup>11</sup>	accelerated corrosion. These
		Waterline		S-15,		areas are included in the inspec-
		Downcomers and Bracing		S-44,		tion plan, subject to VT-1 visual
		•Drywell Exterior Shell with	CS, poly-	S-47,		examination and ultrasonic
		Compressible Material	urethane	S-80,		thickness measurements. If ab-
· · ·		Mark II Steel Containments	generation of the	S-85,		normal conditions are identified,
	1	•Suppression Chamber Interior	CS	S-88	e de la companya de l	the area should be repaired, re-
·		Surface at Waterline		ł		placed, or justified by engineer-
		•Downcomer Pipes & Bracing	л.			ing evaluation. Reexamination
		•Drywell Exterior Shell with	CS, poly-			required for 100% of the sus-
		Compressible Material	urethane			pect area at every 10 yrs of in-
			,			spection interval.
Less	T		<u> </u>	0.10		
Local	LOSS OI	Free added Chall Destant		G-12,	A plant-specific aging program	The evaluation for management
Corrosion	material	•Embedded Shell Region	cs	G-19,	is required to manage the local	of inaccessible areas is to be
		•Drywell Support Skirt		G-20,	corrosion of these inaccessible	justified on a plant-specific
		•Sand Pocket Region		S-20,	and/or embedded carbon steel	basis.
		Mark II Steel Containment		S-87,	containment components.	
		•Embedded Shell Region		S-97		
		Support Skirt				
		Sand Pocket Region	$(1,1,2,\dots,2,n) \in \mathbb{N}$			
		•Region Shielded by Diaphragm				
		Floor	1.5			
		Mark III Steel Containment				
		<ul> <li>Embedded Shell Region</li> </ul>	and the second sec	1		

\* 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.

Aging–Related Degradation Mechanism	Aging Effects	Components	Materialsa	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Local Corrosion	Loss of material	Common Components •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 Interior Surface •Drywell Liner Interior Surface •Basemat Liner •Liner Anchors Mark III Concrete Containments •Containment Liner Interior Surface •Containment Liner Exterior Surface •Suppression Chamber Liner Interior Surface or Cladding Surface •Basemat Liner	CS CS or SS CS SS in pool region, CS rest SS SS SS SS	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 con- structed of mild carbon steel, with a coating applied to the surface to protect it from corro- sion effects. Corrosion of the liner plate is mitigated by pro- tective coatings on the interior surface, and the alkaline envi- ronment on the exterior surface. SS is corrosion resistant.
		•Suppression Chamber Liner Exterior Surface	SS			

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

e 10.

.e.; 64

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials <sup>a</sup>	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Local	Loss of	Common Components	· · · · · · · · · · · · · · · · · · ·	0.10	Deviation low amination under	Lindomuston surfaces are con
Correction	LUSS UI	Dissimilar Matal Wolds	00.00	0-10,	the musiciant of WE 1940	olderwater surfaces are coll-
Corrosion	material	•Dissimiar Metar weids	05, 55	G-12,	the provisions of IWE-1240	sidered as accessible by the
		Mark I Concrete Containments		G-19, :	(Exam. Category E-C) of ASME	rules of IWE-1240 of ASME
		•Torus Liner Interior Surface at	cs	S-6,	Sect. XI, Subsect. IWE. <sup>11</sup>	Sect. XI; Subsect. IWE requires
:		Waterline	- 10 - 10 - 10 - 10 - 10 - 10 - 10 - 10	S-15,	4.	the identification, assessment, &
1		•Downcomers & bracing		S-85		mitigation of local corrosion of
		Mark II Concrete Containments				containment liner components.
		•Suppression Chamber Liner		Ì.		Visual VT-1 and ultrasonic
		Interior Surface at Waterline				thickness measurements fol-
		•Downcomers & bracing	1994) 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -			lowed by engineering evaluation
						are required.
Local	Loss of	Mark II Concrete Containment	<u>CS</u>	C-5	Plant encoific management	Evaluation is to be justified on a
Corrosion	moterial	Pagion Shielded by Dianhrogm		G 10	Plant-specific management	plant anacific basis
COTTOSION	material	Floor		G-12,	program is required for	plant-specific basis.
		F 1001		<u>G-19</u>	maccessible areas.	
Local	Loss of	Mark II Concrete Containment	CS &	G-10,	Prestressing tendons and	Corrosion of prestressed ten-
Corrosion	material		Concrete	G-13,	tendon anchorage hardware	dons can be identified and
		Prestressed Tendons		G-14,	should be examined in ac-	managed by established pro-
				G-19,	cordance with the provisions of	grams for periodic visual exami-
		n an		S-2,	RG 1.35. <sup>14</sup>	nation of the tendon anchor
				S-6,		heads as well as frequent exam-
				S-16,		ination of the corrosion protec-
	•			S-17,		tion medium to ensure absence
	•			S-75,		of corrosive fluids, as prescribed
				S-78,		in RG 1.35.
				S-85		

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

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 <sup>a</sup>	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Elevated Temp-	Loss of strength &	Common Components •Penetration Sleeves	cs	G-1, G-2, G-3	Non-significant ARDM if it meets the basis requirements.	Degradation from exposure to elevated temperatures is non- significant for containment
crature	mouuus	Mark I Concrete Containments  •Drywell & Torus Concrete  •Drywell & Torus Reinforcing Steel  Mark II Concrete Containments  •Containment Concrete  •Containment Concrete Reinforcing Steel  •Concrete Fill in Annulus	CS, concrete	G-3, G-4, G-11, G-16 to G-18, G-21, S-1, S-65		components maintained at op- erating temperatures <66°C (150°F) and local area tem- peratures <93°C (200°F) <sup>15</sup> or for structures that operate above these limits, plant-spe- cific justification is provided in accordance with ACI 349-85, <sup>16</sup> or ASME Sect III Division 2
		•Concrete Pin 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 &				Class CC. <sup>2</sup> Degradation from exposure to elevated temperatures is non- significant for reinforcing steel (rebar) maintained at temp- eratures <316°C (600°F). <sup>15</sup>
		Dome Reinforcing Steel  Concrete Basemat & Reinforcing Steel Mark III Steel Containments  Concrete Basemat  Concrete Fill in Annulus				

Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials <sup>a</sup>	Number <sup>b</sup>	Agreement or Proposal	Agreement or Proposal
Irradiation	Loss of	All Components of Mark I, II,	Concrete, CS,	G-1,	Non-significant ARDM.	The neutron fluence levels &
	fracture	and III Steel and Concrete	SS	G-2,		maximum integrated gamma
	toughness	Containments		G-3,		doses incurred by containment
	for steel &			G-4,		components, including rebars &
	loss of		1 1. 	G-11,		prestressed tendons for both the
	strength &			G-16,	Contraction of the second sec second second sec	current and license renewal
	modulus for	and the state of the second second second	н. - С С С С С С С С	G-17,		period do not exceed the level at
	concrete			G-18,	and the second	which measurable degradation
- -				G-21,		occurs. (4 x $10^{16}$ n/cm <sup>2</sup> for
				S-72		prestressed tendons; 2 x
						$10^{17}$ n/cm <sup>2</sup> for all components
						made of CS, SS including rebar,
						linear plate; $1 \ge 10^{19} \text{ n/cm}^2$
			1			neutron radiation & $1 \ge 10^{10}$
			1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -			rads gamma radiation for
			100 B			concrete), 17,18

Table B4.	Brief summary of	technical inf	formation an	nd NUMARC/I	NRC agreements	s from BWR	containments licen	se renewal i	ndustry
	report								

r	eport					
Aging-Related	Aging			NRC Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials <sup>a</sup>	Number <sup>b</sup>	Agreement or Proposal	Agreement or Proposal
Fatigue	Cumulative	Common Components	CS	S-5	Unresolved issue	NUMARC basis:
	Fatigue	Penetration Sleeves		S-7,	NUMARC proposal:	Existing component fatigue ana-
	Damage			S-8,	Perform ASME Code Sect. III	lyses may be used to prorate
	-	Mark I Steel Containment		S-51,	fatigue reanalysis to ensure	the calculated usage factors for
		•Vent Header <sup>†</sup>		S-52A,	that the fatigue usage factors	the license renewal period. For
		Downcomers & Bracing	1	S-53,	can be maintained <1 through-	components having no fatigue
		Mark II Steel Containment		S-56,	out the license renewal term;	design basis, ASME Code Sect.
		•Unbraced Downcomers		S-67,	or ASME Sect. XI, Subsect.	III provides guidance for
		Mark I and Mark II Concrete		S-68,	IWE, inspection to ensure that	supplemental plant-unique

S-70.

S-90, S-91

Open

issue

G-6. G-15.

S-50,

S-52B,

S-91

component integrity is main-

tained throughout the license

renewal term, including the

continued service of a comp-

rejectable flaw, as justified by

Damage of bellows by low-cycle

when environmentally assisted,

fatigue is credible, especially

& 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

an engineering evaluation.

onent with an otherwise

NRC proposal:

of operation.

fatigue assessment. Exam.

Category E-B provides a VT-1

examination for containment

penetration welds, including

penetration sleeve.

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

At vent header and downcomer intersection.

Containment

•Vent Header\*

Downcomers & Bracing

B-58

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

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

3 10

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 <sup>a</sup>	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Mechanical Wear	Lockup	Common Components  Personnel Airlock  Equipment Hatches  CRD Hatch Mark I Steel Containments  Drywell Head  Downcomers & Bracing  Vent System Supports  Torus Seismic Restraints 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  Downcomers & Bracing  Vent System Supports  Mark II Concrete Containments  Drywell Head  Downcomers & Bracing  Vent System Supports  Mark II Concrete Containments  Drywell Head  Downcomers & Bracing  Vent System Supports  Mark II Concrete Containments  Drywell Head  Downcomers & Bracing  Vent System Supports  Mark II Concrete Containments  Drywell Head	CS CS Saddle: CS lubron plate: graphite 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 <sup>11</sup> & IWF, <sup>20</sup> as applicable, to ensure that component integrity is main- tained throughout the license renewal term.	The pressure retaining compo- nents such as airlock, equip- ment hatches, CRD hatch and drywell head are required to be examined once every inspection interval (10 yrs) according to ASME XI, Subsect. IWE <sup>11</sup> 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, defor- mation and other degradation, in accordance with ASME Sect. XI, Subsect. IWF. <sup>20</sup>
1. A.		•Downcomer Pipes & Bracing				

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials <sup>a</sup>	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Strain Aging	Loss of fracture toughness	Common Components  •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 downco- mer pipes and bracing  Mark III Steel Containments  •Containment Shell Interior Surface  •Containment Shell Exterior Surface	CS	G-1 to G-4, G-11, G-16 to G-18, G-21	Strain aging is a non-signifi- cant ARDM for containment steel components that meet the basis requirements.	Strain aging is a non-signifi- cant ARDM for containment steel components having ser- vice stress in the elastic region and without severely cold working in the forming pro- cess. <sup>21</sup> If severe cold working was used in the forming pro- cess, but the plates are normal- ized, or stress relieved or both after forming with minimal (<5%) subsequent cold working, then strain aging is not significant.
		<ul> <li>Suppression Chamber Shell Interior Surface</li> <li>Suppression Chamber Shell Exterior Surface</li> <li>Embedded Shell Region</li> <li>Mark I Concrete Containments</li> <li>Vent Line Bellows</li> <li>Drywell Head</li> <li>Mark II Concrete Containments</li> <li>Drywell Head</li> </ul>				

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

Table B4.	Brief summary of technical	information and NUM	IARC/NRC	agreements j	from BWR containment	s license renewal industry
	report					

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

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials <sup>a</sup>	NRC Comment Number <sup>b</sup>	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 require- ments.	For austenitic SS containment components, SCC is not a sig- nificant ARDM if they are only exposed to the containment or reactor building environment or their normal operational stress levels are less than mate- rials yield strength or fracture mechanics analysis has estab- lished that cracks do not propa- gate. 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.

<sup>a</sup> Designation for steel materials are as follows: SS = stainless steel, CS = carbon steel.
 <sup>b</sup> 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.

**REFERENCES:** 

- 1. Specification for Air Entraining Admixture for Concrete, ASTM C260-77, American Society for Testing and Materials, Philadelphia, PA, 1977.
- 2. <u>Code for Concrete Reactor Vessels and Containments</u>, 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.
- 4. ACI 318-63, Building Code Requirements for Reinforced Concrete, American Concrete Institute, Detroit, Michigan.
- 5. Prasad, N., et al., "Concrete Degradation Monitoring and Evaluation," contained in <u>Proceedings of the International Nuclear Power Plant Aging</u> <u>Symposium</u>, Aug. 30-Sept. 1, 1988, NUREG CP-100, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC.
- 6. "Petrographic Identification of Reactive Constituents in Concrete Aggregate," B. Mather, ASTM Proc. Vol. 48, American Society of Testing and Materials, Philadelphia, PA, pp. 1120-1125, 1948.
- 7. ASTM C295-54, "Practice for Petrographic Examination of Aggregates for Concrete," American Society of Testing and Materials, Philadelphia, PA.
- 8. ASTM C227-50, "Potential Alkali Reactivity of Cement Aggregate Combination," American Society of Testing and Materials, Philadelphia, PA.
- 9. Metals Handbook, Ninth Edition, Volume 13, Corrosion, ASME International, 1987.
- 10. Handbook of Corrosion Data, ASME International, 1989.
- 11. ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE, Rules for Inservice Inspection of Nuclear Plant Components, 1992 Edition with 1992 Addenda.
- 12. Code of Federal Regulations, 10CFR50, Appendix J, January 1989.
- 13. Troxell, G. E., David, H. E., and Kelly, J. W., Composition and Properties of Concrete, Second Edition, McGraw-Hill, 1968.
- 14. Regulatory Guide 1.35, Revision 3, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures," U.S. Nuclear Regulatory Commission, July 1990.
- 15. "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.
- 16. ACI 349-85, "Code Requirements for Nuclear Safety Related Concrete Structures," American Concrete Institute.
- 17. Glasstone, S. and Sesonske, A., "Nuclear Reactor Engineering," Von Nostrand, Inc., 1967.
- 18. Hilsdorf, H. R., Kroop, J., and Koch, H. J., "The Effects of Nuclear Radiation on the Mechanical Properties of Concrete," Douglas McHenry International Symposium on Concrete and Concrete Structures, American Concrete Publication SP-55, 1978.
- 19. "Considerations for Design of Concrete Structures Subjected to Fatigue Loading," ACI 215R-74, American Concrete Institute, 1981.
- 20. "Inservice Inspection of Nuclear Power Plant Components," ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWF-Component Supports.
- 21. Wulpi, D. J., "Understanding How Components Fails," American Society of Metals, 1985.

B-64

22. "Classification, Standards of Accuracy, and General Specification of Geodetic Control Surveys," Federal Geodetic Survey, National Oceanic and Atmospheric Administration, May 1974.

#### 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 Containment Wall Reinforcing Steel 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 **Dome Reinforcing Steel Basemat Reinforcing Steel** COMMON COMPONENTS Penetration Sleeves **Dissimilar Metal Welds** Penetration Bellows Personnel Airlock Equipment Hatches **CRD** Hatch

· · · ·	·		· · · · · · · · · · · · · · · · · · ·		· · · · · · · · · · · · · · · · · · ·	
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials	Numbera	Agreement or Proposal	Agreement or Proposal
800	Orreals	Deceter Ocelant Dump		0.1	I lange always include	NI MARC basis:
SCC	Crack	Reactor Coolant Pump	0400	G-1,	Unresolved issue	Components fobricated of
	initiation &	Casing	CASS	G-2,		Components labricated of
	growth	Cover	CASS	G-3,	NUMARC proposal:	CASS or CS internally clad
	-	Casing Flange	CASS	G-7,	Non-significant ARDM	with SS (>5% ferrite) have
		Cover Flange	CASS	S-V-36,		reduced susceptibility to SCC
		Nozzles	SS	S-V-37	NRC proposal:	(see S1 S-1), <sup>1</sup> underlying CS
		Pressurizer			IGSCC can occur under the	base metal is not susceptible to
	-	Shell/Heads	CS	Open	operating conditions (water	decohesion; <sup>2</sup> & concentrations
		Spray Line Nozzle	CS	issues	chemistry) during shutdown	of oxygen, halogens, & sulfates
		Valve Nozzle	CS	S-V-38,	because oxygen is introduced	are monitored & controlled in
		Manway	CS	S1 S-1	to primary coolant during cool	the coolant (see S-V-38); <sup>3</sup> and/
	11 - 11 - 11 - 11 - 11 - 11 - 11 - 11	Instrument Nozzle	CS		down to control CRUD-bursts,	or not subjected to corrosive
-		Surge Line Nozzle	CS, CASS		& coolant is exposed to air	environment.
		Support Skirt	CS		during many shutdowns.	
		Safety & Relief Valves			The potential of cracking in	
		Valve Body	SS, CASS		cladding remote from welds	
	ж.	Bonnet	SS, CASS		should be addressed. SS	
		Body Flange	SS, CASS		cladding may have regions	
		Bonnet Flange	SS, CASS		of low delta ferrite that have	
		Nozzles	SS		been sensitized during PWHT	
		Seats & Disks	Stellite, SS		& thus susceptible to IGSCC;	
		Piping & Fittings			ASME Sect. XI requires	
		Cold-Leg	CS, SS, CASS		inspection of weld & weld	
	:	Hot-Leg	CS, SS, CASS		regions.	
		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		1	
		Seats & Disks Piping & Fittings Cold-Leg Hot-Leg Surge Line Spray Line Nozzles & Safe Ends Auxiliary Piping DHRS CFS Fittings, Nozzles, & Safe Ends Integral Support	Stellite, SS CS, SS, CASS SS, CASS SS, CASS SS CS, SS, CASS SS SS SS, CASS CS, SS		& thus susceptible to IGSCC; ASME Sect. XI requires inspection of weld & weld regions.	

<sup>†</sup> 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.

	r					
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials	Numbera	Agreement or Proposal	Agreement or Proposal
SCC	Crack	Pressurizer		S-V-39	Evaluate Ni-alloy applications	Review of use of Ni-alloys &
	initiation &	Instrument Nozzle	Ni-Alloy		in pressurizers; ASME	ASME Sect. XI, Subsect. IWB <sup>4</sup>
	growth	Heater Sleeves	Ni-Alloy		Sect. XI Subsect. IWB <sup>4</sup> re-	exam. category B-E supple-
		and the second			quirements of VT-2 supple-	mented by augmented inspec-
					mented by augmented in-	tion programs based on NRC
					spection programs based on	Inf. notice No. 90-10 <sup>5</sup> are
					NRC Inf. notice No. 90-10. <sup>5</sup>	current & effective programs for
						detection, sizing, evaluation, &
						remediation.
ISCC	Crack	Pressurizer		None	Surface &/or volumetric	ASME Sect. XI, Subsect. IWB <sup>4</sup>
	initiation &	Safe Ends	SS		exam. of ASME Sect. XI,	exam. category B-F, are
	growth				Subsect. IWB. <sup>4</sup>	current & effective programs
	· · ·					for detection, sizing,
					· ·	evaluation, & remediation.
				l		
scc	Crack	Reactor Coolant Pump		None	ASME Sect. XI, Subsect. IWB,	ASME Sect. XI, Subsect. IWB <sup>4</sup>
	initiation &	Closure Bolting	HSLAS <sup>†</sup>		includes VT-1 of flange,	exam. category B-G-1 & -2,
	growth	Pressurizer			nuts, bushing, & washer	are current & effective
		Manway Bolting	HSLAS <sup>†</sup>		surfaces, & volumetric exam.	programs for detection, sizing,
	4	Safety & Relief Valves			of bolts & studs. <sup>4</sup>	evaluation, & remediation.
		Closure Bolting	HSLAS <sup>†</sup>		· · · · · · · · · · · · · · · · · · ·	

<sup>†</sup> High-strength low-alloy steel.

		·			· · · · · · · · · · · · · · · · · · ·	
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials	Numbera	Agreement or Proposal	Agreement or Proposal
						F
Thermal	Loss of	Reactor Coolant Pump		G-4,	Unresolved issue.	NUMARC basis:
Embrittle-	fracture	Casing	CASS	S-I-12,		Components with ferrite con-
ment	toughness	Cover	CASS	S-III-32,	NUMARC proposal:	tent >20% for all centrifugally
		Casing Flange	CASS	S-III-34	ASME Sect. XI, Subsect. IWB	cast & static-cast CF-3 & CF-
-		Cover Flange	CASS		requires visual VT-3 exam.	8, or >14% for static -cast CF-
		Pressurizer		Open	& analytical evaluation	3M & CF-8M (S-III-28), ASME
		Surge Line Nozzle	CASS	issues	procedures for flaw tolerance.	Sect. XI, Subsect. IWB, <sup>4</sup> exam
		Safety & Relief Valves		S-III-28,		categories B-L-1 & -2, B-M-1
		Valve Body	CASS	S-III-33,		& -2, & B-P with flaw accep-
		Bonnet	CASS	S1 S-2	NRC proposal:	tance standard IWB-3500, and
		Body Flange	CASS		Ferrite criteria is inadequate	UT procedures of supplements
		Bonnet Flange	CASS		tool for screening & VT-3 as	to mandatory Appendix VIII; &
		Piping & Fittings			referenced in Code Case N-481	flaw tolerance evaluation of
		Cold-Leg	CASS		is not intended for detection of	ASME Code Case N-481 (S-III-
		Hot-Leg	CASS		cracks. Fracture toughness	33 & S1 S-2); are current & ef-
		Surge Line	CASS		may be estimated based on	fective programs for detection &
		Nozzles & Safe Ends	CASS		NUREG/CR-4513 Rev. 1.6	evaluation-repair-replacement.
		Auxiliary Piping				
		Fittings, Nozzles, & Safe Ends	CASS			

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Thermal	Loss of	Reactor Coolant Pump		G-1.	Non-significant	Proper material selection and
Embrittle-	fracture	Nozzles	SS	G-2.		relatively low PWR operating
ment	toughness	Closure Bolting	HSLAS	G-3.		temperatures.
	U U	Pressurizer		G-7.		•
		Shell/Heads	CS	S-III-29,		
		Spray Line Nozzle	CS	S-III-30,		
		Valve Nozzle	CS	S-111-31		
		Manway	CS	1		
		Instrument Nozzle	CS, Ni-Alloy	1		
		Surge Line Nozzle	CS			
		Heater Sleeves	Ni-Alloy	1		
		Safe Ends	SS			
		Support Skirt	CS			
		Manway Bolting	HSLAS	1		
		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			3
		Spray Line	SS			
		Nozzles & Safe Ends	CS, SS			
		Auxiliary Piping				
		DHRS	SS			
		CFS	SS			
		Fittings, Nozzles, & Safe Ends	SS			
		Integral Support	CS, SS			

Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials	Numbera	Agreement or Proposal	Agreement or Proposal
Corrosion	Loss of	Reactor Coolant Pump		G-1,	Non-significant	Components are fabricated of
	material	Nozzles	SS	G-2,		or are internally clad with SS or
		Pressurizer		G-3,		Ni-alloys; hydrogen overpres-
		Shell/Heads	CS	G-7,		sure provides protection
		Spray Line Nozzle	CS	G-9 c,		against crevice corrosion; and/
		Valve Nozzle	CS	S-VI-41,		or components not in contact
		Manway	CS	S-VI-42,		with primary coolant.
	-	Instrument Nozzle	CS, Ni-Alloy	S-VI-44		
	1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -	Surge Line Nozzle	CS, CASS			
		Heater Sleeves	Ni-Alloy			
		Safe Ends	SS			
1997) 1997 - 1997 1997 - 1997	1	Support Skirt	CS			
		Safety & Relief Valves				
		Nozzles	SS			
1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -		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		1		
		DHRS	SS			
		CFS	SS			
		Fittings, Nozzles, & Safe Ends	SS, CASS			· · · · · · · · · · · · · · · · · · ·

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion/ Boric Acid Wastage of external surfaces	Loss of material	Reactor Coolant Pump Casing Cover Casing Flange Cover Flange Closure Bolting Pressurizer Top head Manway Bolting Safety & Relief Valves Valve Body Bonnet Body Flange Bonnet Flange	CASS CASS CASS CASS HSLAS CS HSLAS SS, CASS SS, CASS SS, CASS SS, CASS SS, CASS	S-VI-43	Program developed & implemented based on Generic Letter 88-05.7	Recommendations of Generic Letter 88-05 <sup>7</sup> are current & effective program to monitor & control primary coolant leakage.

Aging–Related Degradation	Aging		.   .	NRC Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials	Number <sup>a</sup>	Agreement or Proposal	Agreement or Proposal
Erosion/	Wall	Safety & Relief Valves		G-1,	Non-significant	High-alloy steels, nickel-base
Corrosion	thinning	Valve Body & Body Flange	SS, CASS	G-2, G-3,		alloys, & SSs are resistant to
(E/C)		Bonnet & Bonnet Flange	SS, CASS			E/C, and/or relatively low flow
	-	Nozzles	SS	G-7,		& pH control in PWR environ-
		Seats & Disks	Stellite, SS	S-VI-40		ments.
		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			1
		Valve Nozzle	CS			
		Manway	CS			
		Instrument Nozzle	CS, Ni-Alloy			
		Surge Line Nozzle	CS, CASS			
		Heater Sleeves	Ni-Alloy	1		
		Safe Ends	SS			
		Support Skirt	CS			
		Manway Bolting	HSLAS	1		

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron	Loss of	Reactor Coolant Pump		G-1.	Non-significant	Neutron irradiation embrittle-
Irradiation	fracture	Casing & Casing Flange	CASS	G-2.	3	ment is non-significant be-
Embrittle-	toughness	Cover & Cover Flange	CASS	G-3,		cause of low fluence level. <sup>9,10</sup>
ment	0	Nozzles	SS	G-7		
		Closure Bolting	HSLAS	S-IV-35		1
		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			

· · · · · · · · · · · · · · · · · · ·	r		<b>.</b>	r	r	· · · · · · · · · · · · · · · · · · ·
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials	Numbera	Agreement or Proposal	Agreement or Proposal
Creen	Change in	Reactor Coolant Pump	<u></u>	G.I	Non-significant	Operating temperatures are
	dimension	Casing & Casing Flange	CASS	G.2	from significant	<371°C (<700°F) for CS
	Gimension	Cover & Cover Flande	CASS	G_3		<538°C (<1000°F) for SS
		Nozzles	CASS SS	G_7		
ļ		Closure Bolting		10-7		
		Dressurizer	HOLAO	4		
		Shell/Heads		-		
		Shell/ Heads		4		
		Volte Nogelo		-		
		Valve Nozzie		4	•	
		Manway		-		
		Instrument Nozzle	CS, NI-Alloy	4		
		Surge Line Nozzle	CS, CASS	-		
		Heater Sleeves	Ni-Alloy	-		
		Sate Ends	SS			
		Support Skirt	CS	4		
		Manway Bolting	HSLAS			
		Safety & Relief Valves	<u></u>	l .'		
		Valve Body & Body Flange	SS, CASS	4		
		Bonnet & Bonnet Flange	SS, CASS			
		Nozzles	SS	]		
	2	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		<b>]</b>		
		DHRS	SS			
· · · · · · · · · · ·		CFS	SS			
		Fittings, Nozzles, & Safe Ends	SS, CASS	1		
		Integral Support	CS, SS			

Aging Dalatad			-		I	
Aging-Related	Artina			NRC Onment		Decis for
Degradation	Aging Effects	Componente	Motoriolo	Numbera	NUMARC/NRC	Basis for
Weenamsm	Effects	Components	Materials	Number~	Agreement of Proposal	Agreement of Proposal
Stress	Loss of	Reactor Coolant Pump		G-1,	Non-significant	These components do not
Relaxation	preload	Casing & Casing Flange	CASS	G-2,		depend on preload for
		Cover & Cover Flange	CASS	G-3,		functionality.
		Nozzles	SS	G-7		
	-	Pressurizer				
		Shell/Heads	CS			
		Spray Line Nozzle	CS			
		Valve Nozzle	CS			
		Manway	CS			
		Instrument Nozzle	CS, Ni-Alloy			₹ S
		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			$(1,1,2,\dots,n_{n-1}) \in \mathbb{R}^{n-1}$
		Seats & Disks	Stellite, SS			
		Piping & Fittings				
		All Components	CS, SS, CASS			
		Auxiliary Piping				
		All Components	SS, CASS			
-		Integral Support	CS, SS			
Stress	Loss of	Reactor Coolant Pump		S-II-17	ASME Sect. XI Table IWB-	ASME Sect. XI, Subsect. IWB,
Relaxation	preload	Closure Bolting	HSLAS		2500-1 includes VT-1 of nuts	exam. categories B-G-1 & -2, &
	4 - 4 - 1	Pressurizer			bushings, & washer surfaces,	B-P for leakage, <sup>4</sup> & corrective
	• .	Manway Bolting	HSLAS		& volumetric exam. of bolts &	measure IWA-5250, acceptance
		Safety & Relief Valves		A Sec.	studs; <sup>4</sup> corrective measure	criteria IWA-3142; are current
	1.4	Closure Bolting	HSLAS		IWA-5250; acceptance	& effective for detection &
					criteria IWA-3142.	correction of preload.

r		· · · · · · · · · · · · · · · · · · ·	r			
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials	Numbera	Agreement or Proposal	Agreement or Proposal
Fatigue	Cumulative	Reactor Coolant Pump		G-4	Unresolved issue	NUMARC basis:
0	fatigue	Casing & Casing Flange	CASS	G-9 b,d		ASME Sect. III, Subsect. NB <sup>11</sup>
	damage	* Cover & Cover Flange	CASS	S-I-14	NUMARC proposal:	reanalysis of usage factor (see
	0	* Nozzles	SS	S-I-15	Verification of continued ade-	G-9 a & G-11); actual plant
		* Closure Bolting	HSLAS	S-II-16	quacy of the fatigue design	transient, cycle monitoring, &
		Pressurizer		through	basis through reanalysis of	partial cycle counting; ASME
		* Shell/Heads	CS	S-II-27	fatigue usage factor, actual	Sect. XI, Subsect. IWB <sup>4</sup>
		Spray Line Nozzle	CS	]	plant transients, & partial	inspection requirements of
		Valve Nozzle	CS	Open	cycle counting; assurance	IWB-2500-1 (see G-10) & UT
	i	Manway	CS	issues	provided by ASME Sect. XI	procedures of supplements to
		Instrument Nozzle	CS, Ni-Alloy	G-8	ISI and system testing pro-	mandatory Appendix VIII (see
		* Surge Line Nozzle	CS, CASS	G-9 a	grams, including commit-	G-8); continuation of NRC
		Heater Sleeves	Ni-Alloy	G-10	ments to enhance or augment	Bulletin 88-08 <sup>12</sup> program for
		Safe Ends	SS	G-11	inspection as a result of plant	unisolable piping & NRC
		Support Skirt	CS	S-I-13	experience or regulatory	Bulletin 88-11 <sup>13</sup> program for
	Í	Manway Bolting	HSLAS	]	action; and defect repair &	pressurizer surge line thermal
		Safety & Relief Valves			component replacement.	stratification in accordance
		* Valve Body & Body Flange	SS, CASS			with NSSS-specific operational
		Bonnet & Bonnet Flange	SS, CASS		NRC proposal:	procedures (see S-I-13); are
		Nozzles	SS		Fatigue issues are unresolved	current & effective programs
		Seats & Disks	Stellite, SS		until an agreement is reached	for detection & evaluation-
		* Closure Bolting	HSLAS		in the ongoing discussions	repair-replacement.
	-	Piping & Fittings			on fatigue evaluation for	
		* Cold- & Hot-Leg	CS, SS, CASS		license renewal between	
		* Surge Line	SS, CASS	. I	NUMARC and staff	
		Spray Line	SS		(G-8, G-9a, G-10, & G-11).	
		* Nozzles & Safe Ends	CS, SS, CASS		Potentially significant fatigue	
		Auxiliary Piping			damage due to cold water	
		* DHRS	SS	$(x_{i}, y_{i}) \in [0, \infty]$	spraying onto the pressurizer	
		* CFS	SS	1	shell should be evaluated	
		* Fittings, Nozzles, & Safe Ends	SS, CASS		(S-I-13).	
		Integral Support	CS, SS			

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

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Wear	Attrition	Reactor Coolant PumpCasingCoverNozzlesPressurizerShell/HeadsSpray Line NozzleValve NozzleManwayInstrument NozzleSurge Line NozzleHeater SleevesSafe EndsSupport SkirtManway BoltingSafety & Relief ValvesValve BodyBonnetNozzlesPiping & FittingsAll ComponentsAuxiliary PipingAll Components	CASS CASS SS CS CS CS CS CS CS CS CS CS CS CS C	G-1, G-2, G-3, G-7, S-VII-45	Non-significant	Not subjected to relative motion or does not incorporate clamped joints.
		Integral Support	CS, SS	].		

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

<sup>a</sup> 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.

#### **REFERENCES:**

- 1. Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," U.S. Nuclear Regulatory Commission, Washington, DC, May 1973.
- 2. WRC Bulletin No. 197, "A Review of Unclad Cracking in Pressure-Vessel Components," A. G. Vickier and A. W. Pense, Welding Research Council, New York, August 1974.
- 3. NP-7077, "PWR Primary Water Chemistry Guidelines: Revision 2," EPRI, Final REPORT (REV 05), November 1990.
- 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. NRC Information Notice No. 90-10: "Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600," U.S. Nuclear Regulatory Commission, Washington, DC, February 23, 1990.
- 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.
- NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," March 17, 1988.

- NUREG/CR-4731, Vol. 2, "Residual Life Assessment of Major Light Water Reactor Component-Overview," V. N. Shah and P. E. MacDonald, November 1989.
- 9. "Pressure Vessel Steel Irradiation Embrittlement Formulas Derived from Surveillance Data," G. L. Guthrie, *Trans. ANS*, Vol. 44, p. 222, 1983.
- 10. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," U.S. Nuclear Regulatory Commission, Washington, DC, May 1988.
- 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. NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," U.S. Nuclear Regulatory Commission, June 22, 1988.
- 13. NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," U.S. Nuclear Regulatory Commission, December 1988.
- 14. ASME/ANSI OM, "Operation and Maintenance of Nuclear Power Plants," Parts 1 and 6, American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017, 1990.

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
r	1		-1		r	1
Aging-Related				NRC		Pogio for
Degradation	Aging			Comment	NUMARC/NRC	Agreement or Propage1
Mechanism	Effects	Components	Materials	Numbera	Agreement or Proposal	Agreement of Proposal
IGSCC	Crack	Piping & Fittings		G-2,	Non-significant	Wrought & cast CS are resistant
	initiation &	MS	CS	S-III-1,		to sensitization, and/or applied
	growth	FW	CS	S-III-2		& residual stresses are low, &/
		RHR	CS	1		or are not subjected to corrosive
		RCIC	CS			environment. Bolting degrada-
		НРСІ	CS	7		tion or failure has been ad-
		LPCI	CS	7		dressed in Generic Safety Issue
		LPCS	CS			29. <sup>1</sup>
		HPCS	CS			
		Relief & In-Line Valves		1		
		Valve Body	CS	7		
		Bonnet	CS	7		
		Seal Flange	CS			
		Nuts & Bolts	CS			
		Integral Support	CS	·	·	
IGSCC	Crack	Relief & In-Line Valves		- G-2	Non-significant	Applied & residual stresses are
	initiation &	Seal Flange	SS	_		low &/or not subjected to cor-
	growth	Nuts & Bolts	SS	-		rosive environment. Bolting
		Recirculation Pump				degradation or failure has been
		Heat Exchanger	SS			addressed in Generic Safety
		Seal Flange	SS			Issue 29. <sup>1</sup>
		Nuts & Bolts	SS			
		Integral Support	SS			

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IGSCC	Crack initiation & growth	Relief & In-Line Valves         Valve Body         Bonnet         Recirculation Pump         Bowl         Cover	CASS CASS CASS CASS CASS	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. <sup>7</sup> NRC proposal: CASS materials that meet the NUREG-0313, Rev. 2 <sup>5</sup> guidelines of $\leq 0.035\%$ C & $\geq 7.5\%$ ferrite have reduced susceptibility to IGSCC.	NUMARC basis: CASS materials that meet the C & ferrite content criteria of Hughes et al <sup>7</sup> are resistant to IGSCC.
IGSCC	Crack initiation & growth	Piping & Fittings Recirc. RHR LPCI LPCS HPCS	SS SS SS SS SS SS	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, <sup>5</sup> & implemented through NRC Generic letter 88-01. <sup>8</sup>	Implementation of effective inspection, mitigation, & repair techniques are adequate programs.

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

NUREG-1557

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IASCC	Crack initiation &	Piping & Fittings MS	CS	G-2	Non-significant	IASCC is non-significant because the total fast neutron
	growth	FW	CS	]		fluence within the license
	-	Recirc.	SS			renewal term is less than
		RHR	CS, SS			$1 \ge 10^{20} \text{ n/m}^2$ .
		RCIC	CS	1		
	-	HPCI	CS			
		LPCI	CS, SS	1		
	-	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				
	ļ	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 <sup>a</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Degradation Mechanism Corrosion	Aging Effects Loss of material	Components Piping & Fittings MS FW Recirc. RHR RCIC HPCI LPCI LPCI LPCS HPCS Relief & In-Line Valves Valve Body Bonnet Seal Flange Nuts & Bolts Recirculation Pump Bowl Cover	Materials CS CS SS CS, SS CS, SS CS, SS CS, SS CS, SS CS, CASS CS, CASS CS, CASS CS, SS CS, SS CS, SS CS, SS CS, SS CS, SS	Comment Number <sup>a</sup> G-2, S-IV-1, S-IV-2, S-IV-3	NUMARC/NRC Agreement or Proposal Non-significant	Basis for Agreement or Proposal Water quality & chemistry are controlled according to techni- cal specifications requirements and corrosion allowances are defined according to pressure integrity requirements.
		Seal Flange Nuts & Bolts	SS SS CS_SS	-		

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Erosion/ Corrosion (E/C)	Wall thinning	Piping & FittingsRecirc.RHRRCICHPCILPCILPCSHPCSRelief & In-Line ValvesValve BodyBonnetSeal FlangeNuts & BoltsRecirculation PumpBowlCoverHeat ExchangerSeal Flange	SS         CS, SS         CS         CS, SS         CASS         CS, SS         CS, SS         CS, SS         CASS         CS, SS         SS         SS	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.
		Nuts & Bolts Integral Support	SS CS, SS			
E/C	Wall	Piping & Fittings		G-1,	Appendix A of NUREG-1344	NUREG-1344 <sup>13</sup> recommends
2,0	thinning	MS FW	CS CS	G-3, S-VI-1	for single-phase lines, <sup>13</sup> CHECMATE Code for two- phase lines. <sup>14</sup>	industry program for control of $E/C$ in single-phase systems & CHECMATE <sup>14</sup> predicts $E/C$ in two-phase systems.

Table B6.	Brief summary of technical information and NUMARC/NRC agreements fi	rom BWR primary coolant	pressure bounda	ry_
	industry report			

. .

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
				0.1	AGME Seet VI mentions VT 2	ASME Soot VI Subsect IWB
E/C	Wall	Relief & In-Line Valves		[G-1,	ASME Sect. XI requires VI-3	ASIME Sect AI, Subsect. IWB,
	thinning	Valve Body	cs	G-3,	of valve body internal	exam. categories $B-M-1 \approx -2$ ,
				S-VI-2	surfaces & VT-2 of pressure	& B-P; and guidelines of
	the second second	and the second	· · · ·	4	retaining boundary & system	NEDC-31743; <sup>15</sup> are current &
1	£	the second s	1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1 1		leakage & hydrostatic tests.	effective ISI and testing
1					Also, compliance with NEDC	programs for detection and
	a ser esta			j	31743 <sup>15</sup> is necessary.	evaluation-repair-replacement
4.1.1						of valve components.
4	·					
Thermal	Loss of	Relief & In-Line Valves		_G-1,	Unresolved issue.	NUMARC basis:
Embrittle-	fracture	Valve Body	CASS	G-3,		Components with ferrite content
ment of	toughness	Bonnet	CASS	S-V-2,	NUMARC position:	>20% for all centrifugally cast
CASS	:	Recirculation Pump		S-V-3,	ASME Sect. XI, Subsect. IWB,	& static-cast CF-3 & CF-8, or
		Bowl	CASS	S-V-4	requires visual VT-3 exam.	>14% for static -cast CF-3M &
	· ·	Cover	CASS		& analytical evaluation	CF-8M, ASME Sect. XI,
	1			Open	procedures for flaw tolerance.	Subsect. IWB, <sup>16</sup> exam categories
			in the second second	issues		B-L-1 & -2, B-M-1 & -2, & B-P;
		1. A set of the set	and the second sec	S-V-1,	NRC position:	& flaw tolerance evalution of
		<ul> <li>A second sec second second sec</li></ul>	and the second sec	S-V-5,	Ferrite criteria is inadequate	ASME Code Case 481 is current
			1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -	S-V-6	tool for screening & VT-3	& effective program for
		and the second sec			can not reliably detect tight	detection & evaluation-repair-
х.		and the second			cracks. Fracture toughness	replacement.
	and the second				may be estimated based on	
:		and the second			NUREG/CR-4513 Rev. 1.17	

and the second secon

Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials	Numbera	Agreement or Proposal	Agreement or Proposal
Neutron	Loss of	Piping & Fittings		G-2	Non-significant	Total fast neutron fluence
Irradiation	fracture	MS	CS	1	<u> </u>	within the license renewal term
Embrittle-	toughness	FW	CS	1	4.	is $<1 \times 10^{17} \text{ n/m}^2$ for the PCPB
ment		Recirc.	SS	1		components.
		RHR	CS, SS	1		-
		RCIC	CS	1		
	-	HPCI	CS	1		
		LPCI	CS, SS			
		LPCS	CS, SS			
		HPCS	CS, SS			
		Relief & In-Line Valves		1		
		Valve Body	CS, CASS			
		Bonnet	CS, CASS			
		Seal Flange	CS, SS			
		Nuts & Bolts	CS, SS			
		Recirculation Pump		1		
		Bowl	CASS			
		Cover	CASS	]		
		Heat Exchanger	SS	]		
		Seal Flange	SS	]		
		Nuts & Bolts	SS			
		Integral Support	CS, SS			

Aging-Related Degradation	Aging			NRC Comment	NUMARC/NRC	Basis for Arreement or Proneed
Mechanism	Ellects	Components	Materiais	INUITIDE	Agreement or Froposar	USICILIATION ALLANDAR
Creep	Change in	Piping & Fittings		G-2	Non-significant	Operating temperatures are
1	dimension	MS	cs			<371°C (<700°F) for CS,
		FW	cs			<538°C (<1000°F) for SS.
		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			

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Stress Relaxation	Loss of preload	Piping & FittingsMSFWRecirc.RHRRCICHPCILPCILPCSHPCSRelief & In-Line ValvesValve BodyBonnetSeal FlangeRecirculation PumpBowlCoverHeat ExchangerSeal FlangeIntegral Support	CS CS SS CS, SS CS, SS CS, SS CS, SS CS, CASS CS, CASS CS, CASS CS, CASS CS, CASS CS, SS CS, SS CS, SS CS, SS CASS SS SS SS SS SS CS, SS	G-2	Non-significant	These components do not depend on preload for functionality.
Stress Relaxation	Loss of preload	Relief & In-Line Valves Nuts & Bolts Recirculation Pump Nuts & Bolts	CS, SS SS	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, <sup>16</sup> cor- rective measure IWA-5250, & acceptance criteria IWA-3142	ASME Sect. XI, Subesct. IWB, exam. categories B-G-1 & -2, & testing category B-P for sys- tem leakage, <sup>16</sup> are current & ef- fective ISI and testing programs for detection & correction of loss of bolting preload.

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

NUREG-1557

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative fatigue damage	Relief & In-Line Valves Valve Body Bonnet Seal Flange Nuts & Bolts Recirculation Pump Bowl Cover (Bingham) Heat Exchanger (Bingham) Seal Flange Nuts & Bolts Integral Support	CS, CASS CS, CASS CS, SS CS, SS CS, SS CASS CASS SS SS SS SS CS, SS	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 evalua- tions of components show fatigue usage <0.4 for SS & <0.25 for CS for 40-y operation.
Fatigue	Cumulative fatigue damage	Piping & Fittings         MS         FW         Recirc.         RHR         RCIC         HPCI         LPCI         LPCS         HPCS	CS CS SS CS, SS CS CS CS, SS CS, SS CS, SS CS, SS	G-1, G-3, G-5, S-II-2 through S-II-7, S-II-9	Unresolved issue. <i>NUMARC proposal</i> : ASME Sect. III, Subsect. NB <sup>18</sup> reanalysis of usage factor; actual plant transient, cycle monitoring, & partial cycle counting; ASME Sect. XI, Subsect. IWB <sup>16</sup> inspection requirements of IWB-2500-1; & defect repair-replacement. <i>NRC proposal</i> : Some as above	NUMARC basis: Verification of continued adequacy of fatigue design basis through reanalysis of fatigue usage factor; assur- ance provided by ASME Sect. XI ISI and system testing programs; & defect repair & component replacement.

and we want to the second provide the second state of the second state of the second state of the second second

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

<sup>a</sup> 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.

REFERENCES:

State of the second second

- 1. NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," R. E. Johnson, June 1990.
- 2. "Investigation of Corrosion and Stress Corrosion Cracking in Bolting Materials on Light Water Reactors," C. Czajkowski, *Int. J. Pres. Ves. & Piping* 22, 1986.
- 3. NUREG-0943, "Thread-Fastener Experience in Nuclear Power Plant," W. H. Koo, January 1983.
- 4. NUREG/CR-2993, "Examination of Failed Studs from No. 2 Steam Generator at the Maine Yankee Nuclear Power Station," C. Czajkowski, Brookhaven National Laboratory, Upton, NY, February 1983; also NUREG/CR-3776, "Testing of Safety-Related Nuclear Power Plant Equipment at the Central Receiver Test Facility," V. J. Dandini and J. J. Aragon, Sandia National Laboratory, Albuquerque, NM, July 1984.
- 5. NUREG-0313, Rev. 2, "Technical REPORT (REV 03) on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," W. S. Hazelton and W. H. Koo, January 1988.
- 6. V. Pasupathi, Battelle Columbus Laboratory, Letter REPORT (REV 03) to K. Trosen, Northern States Power Co., May 1984.
- 7. "Intergranular Stress Corrosion Resistance of Austenitic Stainless Steel Castings," N. R. Hughes, W. L. Clarke, and D. E. Delwiche, ASTM STP 756, 1982.
- 8. NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," January 1988.
- 9. "Stress Corrosion Cracking Behavior of Carbon Steel in High-Purity Water at 100 to 288°C," S. Pednekar et al, paper #244 presented at *Corrosion* 82, Houston, March 22-26, 1982.
- 10. "Stress Corrosion Cracking of ASTM A508 Cl 2 Steel in Oxygenated Water at Elevated Temperatures," H. Choi et al, Corrosion Vol. 38, No. 3, March 1982.

B-92

- 11. "The Effect of Chloride and Oxygen on the Stress Corrosion Cracking of Stainless Steel: Review of Literature," B. M. Gordon, *Materials Performance*, Vol. 19, No. 4, April 1980.
- 12. NUREG/CR-4490, Vol. 1, ANL REPORT (REV 03) 85-75, Vol. 1, "Evaluation of Non environmental Corrective Actions," P. S. Maiya and W. J. Shack, in Light-Water-Reactor Safety Materials Engineering Research Programs: Quarterly REPORT (REV 03), January-March 1985È, March 1986.
- 13. NUREG-1344, "Erosion-Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants," April, 1989.
- 14. "CHECMATE," Electric Power Research Institute (EPRI), Users Manual NSAC-145L, April 1989.
- 15. NEDC-31743, GE REPORT (REV 03), "BWROG REPORT (REV 03) on Cavitation and Erosion Assessment of Selected Safety Related Valves," November 1990.
- 16. 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.
- 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 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

Aging Dalated	I			NIPC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Componentsa	Materials	Numberb	Agreement or Proposal	Agreement or Proposal
Meenamism	Directo		Materialo			
Neutron	Loss of	Upper Internals Assembly		G-2,	Non-significant	Although neutron irradiation
Irradiation	fracture	W Upper Support Plate	SS	G-6,		causes a decrease in fracture
Embrittle-	toughness	C Upper Guide Structure	SS	G-11,		toughness for SS & Ni-alloy
ment		Support Plate		G-22,		components, the fracture tough-
1. A		B Plenum Cover & Cylinder	SS	S-13,		ness levels remain adequate
l	Į	W RCCA Guide Tube Assemblies		S-14		even at end-of-life fluence levels
		RCCA Guide Tube	See Irr. Emb.			because the applied stresses are
		RCCA Guide Tube Bolts	SS, Ni alloy			low. <sup>1</sup>
		RCCA Guide Tube Sup. Pins	SS, Ni alloy			
		C CEA Shrouds Assemblies				
and the		CEA Shroud	SS, CASS			• • • •
		CEA Shroud Bolts	SS, Ni alloy			
		B CRA Guide Tube Assemblies		-		
	and the second second	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			
л. С		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			

Aging–Related Degradation Mechanism	Aging Effects	Components <sup>a</sup>	Materials	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron	Loss of	Core Support Assembly		G-11,	ASME Sect. XI <sup>4</sup> requires	ASME Sect. XI, Subsect. IWB,4
Irradiation	fracture	W Core Barrel	SS	G-22,	visual VT-3 examination.	exam. category B-N-3, is current
Embrittle-	toughness	Upper Core Barrel Flange	SS	S-13,		& effective program for detection
ment	Ū	Core Barrel Nozzles	SS	S-14		of cracks & evaluation-repair-
		C Core Support Barrel	SS			replacement for vessel internal
		Core Sup. Barrel Upper Flange	SS			components that are accessible
		B Core Support Shield	SS			or can be rendered accessible by
		Core Support Shield Flange	SS			removal of the core &/or other
		B Vent Valve Assemblies	Ni alloy, SS, CASS, Stellite,			internals.
		W Boffle (Former Assembly	Martenshie 55			
		Baffle/Former Assm. Baffles	22			
		Baffle /Former Assembly Bolte	SS Ni allov			
		C Core Shroud Assembly	SS, INI alloy			
		Core Shroud Assembly Bolts	SS Ni allov			
		Core Shroud Tie Rods	SS 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	Contd. on		
		C Fuel Align. Plate Guide Lugs	SS	next page	Continued on next page	Continued on next page

Aging–Related Degradation Mechanism	Aging Effects	Components <sup>a</sup>	Materials	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron	Loss of	Lower Internals Assemblu		See	See previous page	See previous page
Irradiation	fracture	W Lower Core Plate	SS	previous		
Embrittle-	toughness	Fuel Pins	SS, Ni alloy	page		
ment	-	C Core Support Plate	SS			
		Fuel Alignment Pins	SS, Ni alloy			
		B Lower Grid Top Rib Section	SS			
ĺ		Fuel Guide Pads	SS			
		W Lower Support Plate	SS, CASS			
		C Lower Support Structure	SS			
		Beam Assemblies		ļ		
		B Lower Grid Bottom Rib Weldment	SS			
		W Lower Support Columns	SS, CASS			
		Lower Support Column Bolts	SS, Ni alloy	ļ		
		C Core Support Columns	SS, CASS			
		Core Support Column Bolts	SS, Ni alloy			
		B Lower Grid Assembly Support	SS			
		Posts		4		
		Lower Grid Assembly Bolts	SS, Ni alloy	ļ		
		C Core Support Barrel Snubber	SS			
		Assembly		1		]
		B Lower Grid Cylinder and	SS			
		Guide Blocks				

2

Aging-Related	[		[	NPC	[	
Degradation	Aging	i		Comment		Basis for
Mechanism	Effects	Componentsa	Materials	Numberb	Agreement or Proposal	Agreement or Proposal
	200000		materialo	Ivanioer	Agreement of Troposal	Agreement of Hoposai
SCC	Crack	Upper Internals Assembly		G-2,	Unresolved issue	NUMARC basis:
	initiation &	W Upper Support Plate	SS	G-6,		Fabricated of SS or SS with
	growth	C Upper Guide Structure	SS	G-11,	NUMARC proposal: SCC is	>5% ferrite; and subjected to
		Support Plate		G-17,	non-significant for these	stress levels within design
		B Plenum Cover & Cylinder	SS	G-22,	components that are fabri-	specifications; and PWR operat-
		W RCCA Guide Tube Assemblies		S-15,	cated from SS subjected to	ing chemistry limits the oxygen
		RCCA Guide Tube	See Irr. Emb.	S-18,	stress levels within design	to <5 ppb & halogens to
		RCCA Guide Tube Bolts	SS, Ni alloy	S-19	specs. and PWR water	<150 ppb. <sup>5</sup>
		C CEA Shrouds Assemblies			chemistry.	
		CEA Shroud	SS, CASS	Open		
		B CRA Guide Tube Assemblies		issues	NRC proposal: Crevices are	
		CRA Guide Tubes	See Irr. Emb.	G-19,	known to promote SCC in SSs	
	-	CRA Guide Tube Bolts	SS, Ni alloy	S-16	even in the absence of high	
		W Upper Support Columns	SS, CASS		stress, particularly for compo-	
		Upper Support Column Bolts	SS, Ni alloy		nents that are near the core	
		W Upper Core Plate	SS		because radiation may pro-	
		Fuel Pins	SS, Ni alloy		duce aggressive conditions in	
		C Fuel Alignment Plate	SS		crevices. Evaluate the poten-	
		B Upper Grid Assembly			tial of SCC of components with	
		Upper Grid Rib Section	SS	i i i	crevices or creviced geometry.	
		Upper Grid Assembly Bolts	SS, Ni alloy			
		Fuel Guide Pads	SS			
		Core Support Assembly				
		W Core Barrel	SS	i		
		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

Aging-Related		······································		NRC	·····	
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials	Numberb	Agreement or Proposal	Agreement or Proposal
SCC	Crack	W Baffle/Former Assembly		See	Unresolved issue:	Continued from previous page
	initiation &	Baffle/Former Assm. Baffles	ss	previous	Continued from previous page	
	growth	Baffle/Former Assembly Bolts	SS. Ni allov	page		
	0	C Core Shroud Assembly	SS		{	
		Core Shroud Assembly Bolts	SS. Ni alloy			
		Core Shroud Tie Rods	SS	1		
		B Core Barrel	SS			
		Baffle/Former Plates	SS	1	1	
		Baffle/Former Bolts	SS, Ni alloy			
		W Upper Core Plate Align. Pins	SS, Ni alloy			
	1	C Fuel Align. Plate Guide Lugs	SS			
		Lower Internals Assembly				
		W Lower Core Plate	SS			
		Fuel Pins	SS, Ni alloy			
1		C Core Support Plate	SS			
		Fuel Alignment Pins	SS, Ni alloy		a para series de la composición de la c	
		<b>B</b> Lower Grid Top Rib Section	SS			
		Fuel Guide Pads	SS			n.
		W Lower Support Plate	SS, CASS	1. 		
		C Lower Support Structure	SS			
ŀ		Beam Assemblies				
r	:	B Lower Grid Bot. Rib Weldment	SS			1
		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, Ní alloy			
		W Radial Keys and Clevis Insert	SS			
		C Core Support Barrel Snubber Assembly	SS			
		B Lower Grid Cylinder & Guide	SS			
		DIUCKS		l		

Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials	Numberb	Agreement or Proposal	Agreement or Proposal
SCC	Crack	Upper Internals Assembly		G-1,	Unresolved issue	NUMARC basis:
	initiation &	W RCCA Guide Tube Support	SS, Ni alloy	G-6,		ASME Sect. XI, Subsect. IWB, <sup>4</sup>
	growth	Pins		G-7 to	NUMARC proposal: ASME	exam. category B-N-3 is current
	Ũ	C CEA Shroud Bolts	SS, Ni alloy	G-9,	Sect. XI <sup>4</sup> requires visual VT-3	& effective program for detection
				G-15,	exam. & replacement is with	& evaluation-repair-replace-
		Core Support Assembly		G-18,	designs that reduce applied	ment.
		B Core Barrel Bolts	SS, Ni alloy	G-20,	stress.	
				S-2,		
				S-3,	NRC proposal: Augmented ISI	
				S-10,	of components when condi-	
				S-11,	tions, such as sensitized ma-	
				S-15,	terial, high residual stresses,	
				S-18,	crevices, stagnant flow, & his-	
				S-19	tory of coolant contamination,	
1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -				Open	are present.	
				issue		
				G-16		

r				r	r	
Aging-Related				NRC		Det Ca
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials	Number <sup>D</sup>	Agreement or Proposal	Agreement or Proposal
TASCC	Crack	Upper Internals Assembly		G-1	ASME Sect XI4 requires	ASME Sect. XI. Subsect. IWB. <sup>4</sup>
INDEC	initiation &	W Upper Support Plate	88	G-6	visual VT-3 examination &	exam, category B-N-3 is
	grouth	C Upper Guide Structure	99	G_7	replacement is with designs	current & effective program for
	growin	Support Plate		G-8	that reduce applied stress.	detection & evaluation-repair-
		B Plenum Cover & Cylinder	SS	G-9	inter routes appress second	replacement for vessel internal
		W RCCA Guide Tube Assemblies	~~	G-11		components that are accessible
		RCCA Guide Tube	See Irr Emb	G-15		or can be rendered accessible by
		RCCA Guide Tube Bolts	SS Ni allov	G-18		removal of the core &/or other
		RCCA Guide Tube Sup Pins	SS Ni alloy	G-20		internals.
		C CEA Shrouds Assemblies		G-22		
		CEA Shroud	SS CASS	S-1		
		CEA Shroud Bolts	SS Ni allov	S-2		
		B CBA Guide Tube Assemblies	bb, manoy	S-3		
		CRA Guide Tubes	See Irr Emb	S-10		
		CRA Guide Tube Bolts	SS. Ni allov	S-11		
		W Upper Support Columns	SS. CASS	S-17.		
		Upper Support Column Bolts	SS. Ni allov	S-20		
		W Upper Core Plate	SS			
		Fuel Pins	SS. Ni allov	1		
		C Fuel Alignment Plate	SS	1		
		B Upper Grid Assembly		1		
		Upper Grid Rib Section	SS			
		Upper Grid Assembly Bolts	SS, Ni allov	1		
		Fuel Guide Pads	SS	1		
		Core Support Assembly		1		
		W Core Barrel	SS			
		Upper Core Barrel Flange	SS	1		
		Core Barrel Nozzles	SS	1		
		C Core Support Barrel	SS			
		Core Sup. Barrel Upper Flange	SS	1		
-		B Core Support Shield	SS	Contd.		
		Core Support Shield Flange	SS	on next		
de la companya de la		B Vent Valve Assemblies	See Irr. Emb.	page	Continued on next page	Continued on next page

Aging-Related		·····	· · · · · · · · · · · · · · · · · · ·	NRC		
Degradation	Aging		-	Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials	Numberb	Agreement or Proposal	Agreement or Proposal
IASCC	Crack	W Baffle/Former Assembly		See	See previous page	See previous page
	initiation &	Baffle/Former Assm. Baffles	SS	previous		
	growth	Baffle/Former Assembly Bolts	SS, Ni alloy	page		
		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	SS			
		Beam Assemblies				
		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	SS			
		Assembly		-		
		B Lower Grid Cylinder & Guide	SS			
		Blocks				

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

Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Componentsa	Materials	Numberb	Agreement or Proposal	Agreement or Proposal
						9
Erosion/	Wall	Upper Internals Assembly		G-2,	Non-significant	All components are constructed
Corrosion	thinning	W Upper Support Plate	SS	G-6,		of SS or Ni-alloys that are
(E/C)		C Upper Guide Structure	SS	G-11,		considered resistant to E/C in
		Support Plate		G-22,		PWR environment; relatively low
		B Plenum Cover & Cylinder	SS	S-21	· · · ·	fluid flow velocities; pH level in
		W RCCA Guide Tube Assemblies				the bulk coolant minimize
		RCCA Guide Tube	See Irr. Emb.			corrosion; operating pressures
		RCCA Guide Tube Bolts	SS, Ni alloy			of PWR preclude cavitation
		RCCA Guide Tube Sup. Pins	SS, Ni alloy			erosion; & purity & particulate
		C CEA Shrouds Assemblies				control of the coolant eliminate
		CEA Shroud	SS, CASS			particulate erosion. <sup>6</sup>
		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			
3		Fuel Pins	SS, Ni alloy			
		C Fuel Alignment Plate	SS	1		
		B Upper Grid Assembly	· ·	]		
		Upper Grid Rib Section	SS	·		
·		Upper Grid Assembly Bolts	SS, Ni alloy	]		
		Fuel Guide Pads	SS			
		Core Support Assembly		5		
		W Core Barrel	SS			
(1,1,2,2,2,2,2,2,2,2,2,2,2,2,2,2,2,2,2,2		Upper Core Barrel Flange	SS	ļ		l l
		Core Barrel Nozzles	SS	1		
		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

Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials	Numberb	Agreement or Proposal	Agreement or Proposal
Erosion/	Wall	W Baffle/Former Assembly	1.	See	See previous page	See previous page
Corrosion	thinning	Baffle/Former Assm. Baffles	SS	previous		
(E/C)	_	Baffle/Former Assembly Bolts	SS, Ni alloy	page		
		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	ļ		
		W Lower Support Plate	SS, CASS			
		C Lower Support Structure	SS			
		Beam Assemblies	· · ·			
		B Lower Grid Bot. Rib Weldment	SS			
		W Lower Support Columns	SS, CASS	1		
:		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			

Aring Dalatad		r	[	NIDO	· · · · · · · · · · · · · · · · · · ·	
Aging-Related	Aging		Į	Comment		Basis for
Mechanism	Fifects	Componentsa	Materials	Numberb	Agreement or Proposal	Agreement or Proposal
Mechanism	Effects	Components-	Materials	Number*	Agreement of Troposal	Agreement of Proposal
Creep	Change in	Upper Internals Assembly		G-2,	Non-significant	The maximum temperatures
	dimension	W Upper Support Plate	SS	G-6,		of 427°C experienced by PWR
		C Upper Guide Structure	SS	G-10,		vessel internals, including
		Support Plate		G-11,		localized temperature excur-
		B Plenum Cover & Cylinder	SS	G-22,		sions, are well below the tem-
		W RCCA Guide Tube Assemblies		S-22		peratures at which creep is a
		RCCA Guide Tube	See Irr. Emb.			concern for any of the SS
		RCCA Guide Tube Bolts	SS, Ni alloy			or Ni-alloy PWR vessel internal
		RCCA Guide Tube Sup. Pins	SS, Ni alloy			components. <sup>7</sup>
		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			н. - С
		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

Aging-Related Degradation Mechanism	Aging Effects	Components <sup>a</sup>	Matorials	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Creep	Change in	B Core Support Shield	SS	See	See previous page	See previous page
*	dimension	Core Support Shield Flange	SS	previous		
		B Vent Valve Assemblies	See Irr. Emb.	page		
		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	SS			
		Beam Assemblies				
		B Lower Grid Bot. Rib Weldment	SS	Contd.		
		W Lower Support Columns	SS, CASS	on next		
		Lower Support Column Bolts	SS, Ni alloy	page	Continued on next page	Continued on next page

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

\* 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.

Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials	Number <sup>D</sup>	Agreement or Proposal	Agreement or Proposal
Stress	Loss of	Upper Internals Assembly	······	G-2,	Non-significant	These components do not
Relaxation	preload	W Upper Support Plate	SS	G-6,		depend on preload for
		C Upper Guide Structure	SS	G-11,		functionality.
(		Support Plate		G-22,		
		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			ŕ
		B CRA Guide Tube Assemblies		1		
		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		e de la construcción de la constru	
		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

Aging–Related Degradation	Aging			NRC Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials	Numberb	Agreement or Proposal	Agreement or Proposal
Stress	Loss of	W Baffle/Former Assembly		See	See previous page	See previous page
Relaxation	preload	Baffle/Former Assm. Baffles	SS	previous		
	-	C Core Shroud Assembly	SS	page		
		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	j		
		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			
		<i>B</i> Lower Grid Cylinder & Guide Blocks	SS	· · · ·		

Aging-Related Degradation Mechanism	Aging Effects	Components <sup>a</sup>	Materials	NRC Comment Number <sup>b</sup>	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 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 <i>NUMARC proposal:</i> ASME Sect. XI <sup>4</sup> requires visual VT-3 exam. including root cause determi- nation & evaluation. <i>NRC Comments S-37 &amp; 38:</i> Some bolts & pins (page 5-18 of Report 90-05, Sept. 1990) were excluded because of re- dundancy and periodic in- spection. 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, <sup>4</sup> 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 im- plementation 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	issue S-37, S-38	Current practices to be en- hanced, and requires further plant-specific evaluation.	Management program is to be justified on a plant-specific basis.

Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials	Numberb	Agreement or Proposal	Agreement of Proposal
Wear	Attrition	Upper Internals Assembly		G-2,	Non-significant	Not subjected to relative motion
		W Upper Support Plate	SS	G-5,		associated with sliding, flow
		C Upper Guide Structure	SS	G-6,		induced vibration, loss of
		Support Plate		G-11,		clamping force, or from thermal
		B Plenum Cover & Cylinder	SS	G-22		effects.
		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		
1		B Vent Valve Assemblies	See Irr. Emb.	page	Continued on next page	Continued on next page

				L NDO		
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials	Number	Agreement or Proposal	Agreement or Proposal
Wear	Attrition	W Baffle/Former Assembly		See	Non-significant	See previous page
		Baffle/Former Assm. Baffles	SS	previous		
		Baffle/Former Assembly Bolts	SS, Ni alloy	page		
		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>B</b> Lower Grid Top Rib Section	SS	(		
		W Lower Support Plate	SS, CASS			
		C Lower Support Structure	SS			
		Beam Assemblies				
		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			

Aging–Related Degradation Mechanism	Aging Effects	Components <sup>a</sup>	Materials	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Wear	Aging Effects Attrition	Components <sup>a</sup> Upper Internals Assembly W RCCA Guide Tube C CEA Shrouds B CRA Guide Tubes W Fuel Pins C Fuel Alignment Plate B Fuel Guide Pads Core Support Assembly W Upper Core Barrel Flange C Core Support Barrel Upper Flange B Core Support Shield Flange W Upper Core Plate Alignment Pins C Fuel Alignment Plate Guide Lugs Lower Internals Assembly W Fuel Pins C Fuel Alignment Pins B Fuel Guide Pads	Materials SS, Nicrobraze SS, CASS CASS, SS, Nicrobraze SS, Ni alloy SS SS SS SS SS SS SS SS SS, Ni alloy SS SS, Ni alloy	G-1, G-5 to G-9, G-15, G-18, G-20, S-2, S-3, S-10, S-11, S-39	Agreement or Proposal ASME Sect. XI <sup>4</sup> requires visual VT-3 examination.	Agreement or Proposal ASME Sect. XI, Subsect. IWB, <sup>4</sup> 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 Radial Keys and Clevis Insert C Core Sup Barrel Snubber Assembly	SS SS			
		B Lower Grid Cylinder and Guide Blocks	SS			

Aging–Related Degradation Mechanism	Aging Effects	Components <sup>a</sup>	Materials	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Thermal	Loss of	Upper Internals Assembly		G-2,	Non-significant	Wrought SS & Ni-alloys are not
Embrittle-	fracture	W Upper Support Plate	SS	G-6,		susceptible to thermal aging
ment	toughness	C Upper Guide Structure Support Plate	SS	G-11, G-22,		embrittlement.
		B Plenum Cover & Cylinder	SS	· ·		
		W RCCA Guide Tube Assemblies				
		RCCA Guide Tube	See Irr. Emb.			
		RCCA Guide Tube Bolts	SS. Ni allov			
		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			
т.		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

Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials	Number <sup>b</sup>	Agreement or Proposal	Agreement or Proposal
Thermal	Loss of	W Baffle/Former Assembly		See	See previous page	See previous page
Embrittle-	fracture	Baffle/Former Assm. Baffles	SS	previous		
ment	toughness	Baffle/Former Assembly Bolts	SS, Ni al <del>l</del> oy	page		
		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	ļ		
		B Lower Grid Top Rib Section	SS			
		Fuel Guide Pads	SS			
		W Lower Support Plate	SS			
		C Lower Support Structure	SS			
		Beam Assemblies				
	-	B Lower Grid Bot. Rib Weldment	SS			
		W Lower Support Columns	SS.	_		
		Lower Support Column Bolts	SS, Ni alloy			
		C Core Support Columns	SS		1	
		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	SS			
		Assembly				
		B Lower Grid Cylinder & Guide	SS			
		Blocks				

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

Table B7. Brief summary of technical information and NUMARC/NRC agreements from PWR vessel internals industry report
Aging–Related Degradation Mechanism	Aging Effects	Components <sup>a</sup>	Materials	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion	Loss of	Upper Internals Assembly		G-2,	Unresolved issue (S-16)	NUMARC basis:
	material/	W Upper Support Plate	SS	G-6,		The materials used in the
	corrosion	C Upper Guide Structure	SS	G-11,	NUMARC proposal:	construction of PWR vessel
	product	Support Plate		G-22,	Corrosion, general or local,	internals, e.g., SSs & Ni-alloys,
	buildup	B Plenum Cover & Cylinder	SS	Į	is non-significant.	are not susceptible to general
		W RCCA Guide Tube Assemblies		Open		corrosion in benign PWR
		RCCA Guide Tube	See Irr. Emb.	issue	NRC proposal:	primary coolant environment;
		RCCA Guide Tube Bolts	SS, Ni alloy	S-16	There is no assurance that	components containing crevices
		RCCA Guide Tube Sup. Pins	SS, Ni alloy	Į	components made from SSs	are fabricated from SSs, a
		C CEA Shrouds Assemblies			are not exposed to locally	material that is also resistant to
		CEA Shroud	SS, CASS		corrosive environments.	pitting corrosion; & hydrogen
		CEA Shroud Bolts	SS, Ni alloy			overpressure present in the
		<b>B</b> CRA Guide Tube Assemblies				reactor vessel controls crevice
		CRA Guide Tubes	See Irr. Emb.	1		corrosion by minimizing the
		CRA Guide Tube Bolts	SS, Ni alloy			adverse effects of oxygen by
		W Upper Support Columns	SS, CASS			recombination.
		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	1		
		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

Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials	Numberb	Agreement or Proposal	Agreement or Proposal
Corrosion	Loss of	W Baffle/Former Assembly		See	See previous page	See previous page
	material/	Baffle/Former Assm. Baffles	SS	previous		
	corrosion	Baffle/Former Assembly Bolts	SS, Ni alloy	page		
	product	C Core Shroud Assembly	SS			
	buildup	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, <u>Ni</u> alloy			
		C Fuel Align. Plate Guide Lugs	SS			
		Lower Internals Assembly				
		W Lower Core Plate	SS			
r F		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	SS			
		Beam Assemblies				
		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, <u>Ni</u> 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			

Aging–Related Degradation Mechanism	Aging Effects	Components <sup>a</sup>	Materials	NRC Comment Number <sup>b</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Fatigue	Cumulative	Upper Internals Assembly		G-2,	Unresolved issue	NUMARC basis:
	fatigue	C Upper Guide Structure	SS	G-6,		A review of calculated fatigue
	damage	Support Plate		G-13,	NUMARC proposal:	usage factors for components
		B Plenum Cover & Cylinder	SS	S-4,	Non-significant ARDM.	designed to ASME Sect. III; a
		W RCCA Guide Tube Assemblies		S-28,		review of plant design stress re-
		RCCA Guide Tube Bolts	SS, Ni alloy	S-31,	NRC proposal:	ports & hot functional test data;
		RCCA Guide Tube Sup. Pins	SS, Ni alloy	S-32,	Until an agreement is	and a comparison of geometry &
		C CEA Shrouds Assemblies		S-33,	reached in the ongoing	operating similarities.
		CEA Shroud	SS, CASS	S-34,	generic discussion on fatigue	
•		B CRA Guide Tube Assemblies		S-36	evaluation for license	
		CRA Guide Tubes	See Irr. Emb.		renewal between NUMARC	
		CRA Guide Tube Bolts	SS, Ni alloy	Open	and staff, the fatigue issues	
		W Upper Support Columns	SS, CASS	issue	are unresolved.	
1		Upper Core Plate	SS	S-27		
		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			
1		B Core Support Shield	SS	1		
		Core Support Shield Flange	SS			
		B Vent Valve Assemblies	See Irr. Emb.			
		C Core Shroud Assembly Bolts	SS, Ni alloy	1		
		B Core Barrel	SS	1		
- 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 1999 - 199		Core Barrel Bolts	SS, Ni alloy	Contd.		
		Baffle/Former Plates	SS	on next		
		Baffle/Former Bolts	SS, Ni alloy	page	Continued on next page	Continued on next page

Aging–Related Degradation	Aging		-	NRC Comment	NUMARC/N	VRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials	Number <sup>D</sup>	Agreement or P	roposal	Agreement or Proposal
Fatigue	Cumulative	W Baffle/Former Assm. Baffles	SS	See	Unresolved issue		See previous page
	fatigue	C Fuel Align. Plate Guide Lugs	SS	previous	See previous page		
	damage	Lower Internals Assembly		page			
		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	SS		3 •		
		Beam Assemblies					
		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	ľ	·		
		W Radial Keys and Clevis Insert	SS				
		C Core Support Barrel Snubber	SS		1		
		Assembly					
		B Lower Grid Cylinder and	SS			·	
		Guide Blocks	· · · · · · · · · · · · · · · · · · ·				

	y		T	1	J	J
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components <sup>a</sup>	Materials	Numberb	Agreement or Proposal	Agreement or Proposal
				<u></u>		
Fatigue	Cumulative	Upper Internals Assembly		See	Unresolved issue	<b>_</b>
	fatigue	W Upper Support Plate	SS	previous		NUMARC basis:
	damage	W RCCA Guide Tube	See Irr. Emb.	page	NUMARC proposal:	Visual inspection according to
		C CEA Shroud Bolts	SS, Ni alloy		ASME Sect. XI, <sup>4</sup> visual VT-3	ASME Sect. XI, Subsect. IWB, <sup>4</sup>
		W Upper Support Column Bolts	SS, Ni alloy		examination; Sect. III, <sup>10</sup>	exam. category B-N-3; verifica-
		Core Support Assembly			Subsect. NG-5200, reanalysis	tion of continued adequacy of
		W Core Barrel Nozzles	SS		of usage factor; & transient	the fatigue design basis through
		C Core Shroud Assembly	SS		monitoring.	reanalysis in accordance with
		Core Shroud Tie Rods	SS	]		ASME Sect. III <sup>10</sup> Subsect. NG-
		W Upper Core Plate Align. Pins	SS, Ni alloy		NRC proposal:	5200; transient monitoring; and
		Lower Internals Assembly			Until an agreement is	component repair & replace-
		W Lower Core Plate	SS		reached in the ongoing	ment.
1		C Core Support Plate	SS	· ·	generic discussion on fatigue	
		W Lower Support Columns	SS, CASS		evaluation for license	
		C Core Support Columns	SS, CASS		renewal between NUMARC	
1.1			e de la companya de l La companya de la comp		and staff, the fatigue issues	
					are unresolved.	·
Fatigue	Cumulative	Core Support Assembly				
	fatigue	W Baffle/Former Assembly Bolts	SS, Ni alloy		Current practices to be	Select plant specific manage-
	damage		·		enhanced, select plant specific	ment plan that could include
1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -					program.*	qualified visual & volumet-
			• 			ric <sup>11,12</sup> inspection, evaluation,
			1. Sec. 1. Sec			& 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.

#### **REFERENCES:**

- 1. "Post Irradiation Tensile Properties of Annealed and Cold-Worked AISI-304 Stainless Steel," R. E. Robbins, J. J. Holmes, and J. E. Irvin, *Trans. Am. Nuclear Society*, pp. 488-489, November 1967.
- "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.
- 6. "Relative Erosion Resistance of Several Materials," J. S. Hansen, ASTM-STP-664, Philadelphia, PA, pp. 148-162, 1979.
- 7. "Physical Metallurgy for Engineers," D. S. Clark and W. R. Varney, D. Van Nostrand Co., New York, 1962.
- Potential High Fluence Response of Pressure Vessel Internals Constructed from Austenitic Stainless Steels," F. A. Garner, L. R. Greenwood, and D. L. Harrod, in *Proc. of the Sixth Intl. Symp. on Environmental Degradation of Materials in Nuclear Power Systems Water Reactors*, TMS, Warrendale, PA, pp. 783-790, 1993.
- 9. "Problems Anticipated in Austenitic Pressure Vessel Internals Arising from Void Swelling, Irradiation Creep and Swelling-Related Embrittlement," F. A. Garner, paper presented at the 17th ASTM Symp. on Effects of Radiation on Materials, June 21-23, 1993.
- 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.

B-121

### 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	2. CEA Shroud Assemblies	2. CRA Guide Tube Assemblies		
RCCA Guide Tubes	CEA Shrouds	CRA Guide Tubes		
RCCA Guide Tube Bolts	CEA Shroud Bolts	CRA Guide Tube Bolts		
RCCA Guide Tube Support Pins	• ,			
3. Upper Support Column	3. NA	3. NA		
Upper Support Column Bolts				
4. Upper Core Plate	4. Fuel Alignment Plate	4. Upper Grid Assembly		
Fuel Pins		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	1. Core Support Barrel	1. Core Support Shield		
Upper Core Barrel Flange	Core Support Barrel Upper Flange	Core Support Shield Flange		
Core Barrel Nozzles				
2. NA	2. NA	2. Vent Valve Assemblies		
3. Baffle/Former Assembly	3. Core Shroud Assembly	3. Core Barrel Assembly		
Baffle/Former Assembly Bolts	Core Shroud Assembly Bolts	Core Barrel Bolts		
Baffle/Former Assembly Baffles	Core Shroud Tie Rods	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	3. Lower Grid Bottom Rib Weldment		
	Beam Assemblies			
4. Lower Support Column	4. Core Support Column	4. Lower Grid Assembly Support Posts		
Lower Support Column Bolts	Core Support Column Bolts	Lower Grid Assembly Bolts		
5. Radial Keys and Clevis Inserts	5. Core Support Barrel Snubber Assemblies	5. Lower Grid Cylinder Guide Blocks		

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	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	Unresolved issue NUMARC proposal: Monitoring pump perfor- mance, inspection, evaluation, & replacement.	NUMARC basis: Periodic inspection & continu- ous monitoring of performance by plant instrumentation are current & effective programs for detection & evaluation-replace-
				G-27 to G-29, G-33, G-34, S-4,	NRC proposal: Evaluate possible adverse effect of hydrogen water chemistry (HWC) on Ni-Alloys.	ment of jet pumps.
				S-6, S-8, S-17, S-22, S-25 to		
				S-27, S-32, S-48, S-54 Open issue		

·····							
Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	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,	GESIL 462S1 <sup>1</sup> recommends volumetric inspections, im- plementation is plant specific, & recommended repair is to attach reinforcement hardware.	Recommendations of GESIL 462S1 <sup>1</sup> & safety analysis are current & effective inspection programs for detection & eval- uation-repair of access hole covers.	
		Core Shroud Head Bolts	SS, Alloy 600	G-24, G-27, G-28, G-29, G-33, G-34, S-3,	GESIL 433 <sup>2</sup> recommends UT examination during outages, implementation is plant specific, & replacement is with crevice-free design.	Recommendations of GESIL 433 <sup>2</sup> & replacement with crevice -free design are current & effec- tive inspection programs for de- tection & evaluation-replace- ment of core shroud head bolts.	
		Control Blade	SS	S-4, S-6, S-17, S-22, S-25,	GESIL 157 routine replace- ment, <sup>3</sup> operational parameter monitoring, inspection, evaluation, & replacement. <sup>4</sup>	Routine replacement & opera- tional parameter monitoring are current & effective programs for detection & evaluation- replacement of control blades.	
		Control Rod Drive (CRD) Housing	SS	S-26 to S-29, S-31, S-32, S-38,	ASME Sect. XI <sup>5</sup> requires vol- umetric exam. of welds & VT-2 of pressure retaining boundary & system leakage & hydro-static tests.	ASME Sect. XI, Subsect. IWB, <sup>5</sup> exam. categories B-O & B-P is current & effective program for detection & evaluation-repair- replacement of CRD housing.	
		Core Spray Sparger	SS	S-49, S-54, S-55	S-49, S-54, S-55	NRC Bulletin 80-13 <sup>6</sup> recom- mends visual inspection during refueling outages; analytical evaluation; & repair.	NRC Bulletin 80-13 <sup>6</sup> & safety analysis are effective inspection programs for detection, evalua- tion-repair of core spray sparger.
		Intermediate Range Monitor/ Source Range Monitor (IRM/SRM) Dry Tubes	SS		GESIL 409, Rev. 1 <sup>7</sup> recom- mends visual inspection; leakage monitoring; re- placement is with crevice-free design; & resistant material.	Recommendations of GESIL 409,7 leakage monitoring, & re- placement with crevice-free de- sign, are effective inspection programs for detection & evalua- tion-replacement of dry tubes.	

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
IGSCC	Crack initiation & growth	Core Plate	SS	G-12, G-14, G-30,	Current practices to be enhanced. Select plant- specific aging management	Select plant specific manage- ment comprising of qualified inspection & monitoring; water
		Core Shroud	SS	S-11, S-30, S-33, S-44	program.	chemistry control; <sup>8</sup> evaluation- repair-replacement.
		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.

·						
Aging-Related			1	NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials	Numbera	Agreement or Proposal	Agreement or Proposal
TACOO	Com alla	Assess Usis Correct	Aller C00	0.10	Non cignificant	Total fact neutron fluence
IASCC	Crack	Access Hole Cover	Alloy 600	G-19,	Non-significant	within the license renewal term
	initiation &	OPD Hausing	00	G-20,		is pop-significant for IASCC be-
	growin	CRD Housing	00	G-32,		is non-significant for insect be
		Core Plate	55			cause it is less than 5 x 10 <sup>-1</sup>
		Core Shroud Head Bolts	SS, Alloy 600	S-9,		n/cm <sup>2</sup> for low-stressed compo-
	}	Core Spray Int. Piping	SS	S-23,		nents & less than 1 x 10 <sup>20</sup>
		Core Spray Sparger	SS	S-34		n/cm <sup>2</sup> for high-stressed com-
		Jet Pump	SS, (CASS),			ponents. <sup>9,10</sup>
			Alloy 600,			
			X 750			
		LPRM	SS			
		Orificed Fuel Support*	SS, CASS			
11000				0		Denting works opposed & opposed
IASCC	Crack	Control Blade	SS	Same as	GESIL 157 routine replace-	Routine replacement & opera-
	initiation &			the com-	ment, <sup>3</sup> operational parameter	tional parameter monitoring are
	growth			ments for	monitoring, inspection,	current & effective programs for
				IGSCC	evaluation, & replacement.4	detection & evaluation-
		· · · · · · · · · · · · · · · · · · ·		G-2,		replacement of control blades.
		IRM/SRM Dry Tubes	SS	G-5,	GESIL 409, Rev. 17 recom-	Recommendations of GESIL
				through	mends visual inspection;	409, <sup>7</sup> leakage monitoring, &
				S-32,	leakage monitoring; re-	replacement with crevice-free
				S-54	placement is with crevice-free	design, are current & effective
				&	design; & resistant material.	inspection programs for detec-
				S-52,		tion & evaluation-replacement
				S-56		of dry tubes.

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

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	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,	Current practices are to be enhanced.	Select plant specific aging man- agement plan that may include qualified visual and/or volu- metric inspection of susceptible locations; SCC monitoring; ana- lytical evaluation; <sup>5</sup> enhanced
				S-44		water chemistry control; <sup>8</sup> ap-
		Top Guide	SS	Open		propriate repair methods for ir- radiated materials; & replace-
				issue S-5*		ment.

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

Aging–Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Neutron Irradiation	Loss of fracture	Access Hole Cover	Alloy 600	G-19, G-20, G-32 S-9, S-18,	Non-significant	Although neutron irradiation causes a decrease in fracture toughness for SS & Ni-alloy components, the fracture tough- ness levels remain adequate
Embrittle- toughness ment	toughness	CRD Housing	SS			
		Core Plate	SS	S-50,		even at end-of-life fluence
		Core Shroud Head Bolts	SS, Alloy 600	S-51		levels because the applied
		Core Spray Int. Piping	SS			stresses are low.
		Core Spray Sparger	SS			· · · · · · · · · · · · · · · · · · ·
		LPRM	SS			
		Control Blade	SS			
		Core Shroud	SS			
		IRM/SRM Dry Tubes	SS		1	
		Jet Pump	SS, <i>(CASS)</i> , Alloy 600, X 750			
		Orificed Fuel Support	SS, CASS			
		Top Guide	SS			

Aging Palatad	1	Τ	1	NDO	n	
Aging-Related	Aging			Comment		Pasis for
Mochaniam	- Aging Efforto	Componento	Motoriala	Comment Number	NUMARC/NRC	Basis Ior
Mechanism	Effects	Components	Materials	Number«	Agreement of Proposal	Agreement of Proposal
Thermal	Loss of	Access Hole Cover	Alloy 600	G-19,	Non-significant	Wrought SS & Ni-alloys are not
Embrittle-	fracture		-	G-20,		susceptible to thermal
ment	toughness	Control Blade	SS	G-32		aging embrittlement.
				S-9,		
				S-10,		
		CRD Housing	SS	S-19,		
				S-20		
		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,	1		
			Alloy 600,			
			X 750			
		LPRM	SS			
		Orificed Fuel Support	SS			
		Top Guide	SS			
(T)	7 6					
Inermal	Loss of	Ormced Fuel Support	CASS	G-19,	Non-significant	Not subjected to stress levels of
Embrittle-	Iracture			G-20,		sufficient magnitude.
ment	toughness	Jet Pump	(CASS)	G-32		
				S-9,		
				S-19,		
				S-20		

Aging-Related Degradation Mechanism	Aging Effects	Components	Materials	NRC Comment Number <sup>a</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Corrosion	Loss of matertial	Access Hole Cover	Alloy 600	G-19, G-20,	Non-significant	Corrosion rates of all materials are very low.
		Control Blade	SS	G-32 S-9		
		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		1	

······						
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials	Numbera	Agreement or Proposal	Agreement or Proposal
Erosion/	Wall	Access Hole Cover	Alloy 600	G-19 to	Non-significant	SS and Ni-alloys are resistant
Corrosion	thinning	Control Blade	SS	G-21,		to E/C, and/or low flow range,
(E/C)	1	CRD Housing	SS	G-32,		and implementation of HWC in
		Core Plate	SS	S-9,,		which requires oxygen addition
		Core Shroud	SS	S-47,		the feedwater line to limit E/C.
		Core Shroud Head Bolts	SS, Alloy 600	S-53		
	4	Core Spray Int. Piping	SS	].		
		Core Spray Sparger	SS	]		
		IRM/SRM Dry Tubes	SS			
		LPRM	SS			· · · ·
		Orificed Fuel Support	SS,			
	с		CASS			
		Top Guide	SS	]		
		Jet Pump	SS, (CASS),			
			Alloy 600,			
			X 750			· · · · · · · · · · · · · · · · · · ·

			r			
Aging-Related				NRC		
Degradation	Aging			Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Components	Materials	Number <sup>a</sup>	Agreement or Proposal	Agreement or Proposal
Fatione	Cumulative	Access Hole Cover	Allov 600		Unresolved issue	NUMARC basis: Cyclic stresses
raugue	fatione		7 moy 000	G-19	omesowed issue.	are minimal or absent such
	damade	Core Plate	99	G-20	NUMARC proposal:	that ASME Sect III NB-320011
	uamage	Top Guide	89	G-28	Non-significant ARDM	analysis is not required. LPRM
		Control Blade	89	G 20,	NPC proposal: Not	are replaced due to limited
		Core Shroud	00 66	G-32, G-4	when proposition in the second	operating life
		Core Shroud Head Balts	55 55 Aller COO	5-4, C 0	information Diant apositio on	operating inc.
		Core Shroud Head Bolts	55, Alloy 600	5-9,	mormation. Plant specific or	
		Core Spray Int. Piping	55	Open	typical evaluations based on	
	:	Core Spray Sparger	55	issues	ASME Sect. III, of all	
		IRM/SRM Dry Tubes	SS	G-4,	safety-related internal com-	
		Orificed Fuel Support	SS, CASS	S-7	ponents should show a usage	
		LPRM	SS		of <0.7 for the 60-y lifetime.	
Fatigue	Cumulative	CPD Housing	22	G-22	Linresolved issue	NUMARC basis
raugue	fatique	CIAD Housing	55	G-24	NUMARC proposal: ASME	ASME Sect XI Subsect IWB 5
	damade			G-24, G-33	Sect XI5 requires volumetric	exam categories B-O & B-P &
	uamage			G-34	evam of welds & VT-2 of	recommendations of NP-5181M
				S-3	pressure retaining boundary &	& NP-5836M <sup>12,13</sup> are current
				S-26	system leakage & hydrostatic	& effective programs for
				S-27	tests.	detection & evaluation-repair-
				~	NRC proposal: Not acceptable	replacement of CRD housing.
					without detailed information.	- · F · · · · · · · · · · · · · · · · ·
		Jet Pump	SS.	Open	Unresolved issue	NUMARC basis:
	÷	000	(CASS).	issues	NUMARC proposal: Monitoring	Periodic inspection & continu-
			Allov 600.	G-4.	nump performance, inspection.	ous monitoring of performance
			X 750	S-7	evaluation. & replacement.	by plant instrumentation are
					NRC proposal: Not acceptable	current & effective programs for
			· · · ·		without detailed information.	detection & evaluation-replace-
						ment of jet pumps.

<sup>a</sup> 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.

, state

#### **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

Aging Rolated		· · · · · · · · · · · · · · · · · · ·	Motorioloph	NIPC	· · · · · · · · · · · · · · · · · · ·	
Aging-Related	Artimo		Structurel	Commont		Basis for
Machaniam	Aging Defense	Structure al	Structural	Number	A recent or Dropoon	Agreement or Proposal
Mechanism	Lilects	Structures	Component	Number	Agreement of Proposal	Agreement of Proposal
General	Age related	Structures	Concrete and	Open	Unresolved issue	NUMARC basis:
uonona	degradation		steel	issues.	(One-time inspection)	The ARD mechanisms are eval-
	effects			G-3.		uated for significance using the
	Chrotis			G-4	NUMARC proposal:	available research & industry
				G-5	Resolution of the effects of ARD	data. If acceptance criteria
1	1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 19			S-73	mechanisms is based upon the	(including a review of plant per-
· · · · ·				S-76	review/evaluation of plant-	formance history to assure that
					specific features including	contradictory evidence does not
					appropriate CLB documents/	exist) are satisfied then the in-
		n an			information Inspection of	spection for that mechanism/
	•		: · · · ·		structures is not required if a	component combination is not
					set of acceptance criteria	needed.
] ]					lincluding a review of plant	
					performance history to assure	
			1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -		that contradictory evidence	
					does not exist) are satisfied	
					does not endy die ballshed.	
					NRC proposal	
					A one time focussed plant-	
· · · ·					specific inspection of struc-	
					tures is proposed as part of an	
					applicant's activities for	
					identification and resolution of	
					notential ARDM	

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

\* 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).

Aging-Related			Materials: <sup>b</sup>	NRC		
Degradation	Aging		Structural	Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Structures <sup>a</sup>	Component	Number <sup>c</sup>	Agreement or Proposal	Agreement or Proposal
Leaching of	Increase of	1. BWR Reactor Building	Concrete:	G-2,	For Class 1 concrete struc-	Leaching of calcium hydroxide
calcium	porosity &	PWR Shield Building	F, ExA, & ExB	G-7,	tures that meet the basis	is non-significant for Class 1
hydroxide	permeability	Control Room/Building		G-8,	requirements, leaching of	concrete structures not exposed
		2. BWR Reactor Building with		G-11,	calcium hydroxide is non-	to flowing water; and for struc-
		Steel Superstructure		G-12,	significant ARDM.	tures that are exposed to flow-
		3. Auxiliary Building		G-13,		ing water but are constructed
		Diesel Generator Building		G-16,		using the guidance of ACI
		Radwaste Building		G-18,		201.2R-67 <sup>4</sup> to ensure dense,
		Turbine Building	·	S-6,		well-cured concrete with low
		Aux FW Pump House		S-8.		permeability and control
		Ittility/Pining Tunnels		S-21.		cracking through proper
		5 Fuel Storage Facility		S-77		arrangement & distribution of
1		D. Fuci Otorage Facility		0.11		reinforcement.
		Refueining Carlai				
		6. Intake Structure	•	-		
		Cooling Tower	· ·			
		Spray Pond		· · ·		
		7. Concrete Tanks				
		9. BWR Unit Vent Stack	Concrete: F,			
			GA, & GB			
		8. Steel Tanks	Concrete: F	]		

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

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

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

NUREG-1557

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

\* 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).

B-139

NUREG-1557

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

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

Aging-Related			Materials:b	NRC		
Degradation	Aging		Structural	Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Structures <sup>a</sup>	Component	Number <sup>c</sup>	Agreement or Proposal	Agreement or Proposal
Corrosion of	Cracking,	1. BWR Reactor Building	Concrete:*	G-6,	Corrosion of rebar may cause	Select plant-specific aging man-
Embedded	spalling,	PWR Shield Building	F & ExB	G-10,	potentially significant degrada-	agement program that could in-
Steel	loss of bond,	Control Room/Building		G-14,	tion of below-grade portions of	clude, monitoring of ground
	& loss of	2. BWR Reactor Building with		G-15,	class 1 concrete structures,	water chemistry, inspection, &
	material	Steel Superstructure	1	S-25,	select plant specific aging	testing. <sup>6</sup>
		3. Auxiliary Building		S-30,	management program.§	
		Diesel Generator Building		S-78,		
		Radwaste Building		S-79,		
		Turbine Building		S-80		
		Utility/Piping Tunnels	1991 - Sec. 1997 - Sec. 199			
		Aux FW Pump House		Open		
		5. Fuel Storage Facility		issue		
		Refueling Canal		G-19†		
		7. Concrete Tanks	Concrete:* F			
			& Ext			
		8. Steel Tanks	Concrete:* F			
		9. BWR Unit Vent Stack	Concrete:* F			
		n de la construcción de la constru La construcción de la construcción d	& GB			
		6. Intake Structure	Concrete:*	G-6, -10	Reg. Guide 1.127 <sup>7</sup> requires	Periodic inspections including
		Cooling Tower	ExA, ExB	G-14,	inspection at periodic inter-	engineering data compilation &
	[	Spray Pond	F, & InS	G-15,	vals not to exceed 5 yr & in-	onsite inspection program out-
				S-25,	cludes engineering data	lined in RG 1.127 <sup>7</sup> are current &
				S-30,	compilation & inspection &	effective programs for managing
				S-74,	evaluation of concrete sur-	degradation of Group 6 concrete
				S-78 to	faces, structural cracking,	structures from corrosion of
				S-80	settlement, & water passage.	embedded steel.

\* 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.

			I		I	
Aging-Related			Materials: <sup>D</sup>	NRC		
Degradation	Aging		Structural	Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Structures <sup>a</sup>	Component	Numberc	Agreement or Proposal	Agreement or Proposal
Corrosion	Loss of	1 BWR Reactor Building	Structural	G-6.	Periodic VT-3 inspection and	Select plant-specific aging man-
Controbion	material	PWR Shield Building	steel and	G-10	maintenance practices of	agement program that could
		Control Room /Building	Jet	G-14	coated and uncoated surfaces	include monitoring of ground
		2 BWR Reactor Building with	impingement	G-15	preventive measures and	water chemistry inspection &
		Steel Superstructure	harriers	S-24	coating repair & replacement	testing 13-15
1.0		3 Auviliary Building	Darriers	S-27,	are effective programs to	testing.
		Diesel Cenerator Building		S-74	manage correction of	
		Padwasta Building		G 79	accessible structural steel	
		Turbing Building	· · · ·	S-70,	Correction is potentially	
		Turbine Building		5-80	Corrosion is potentially	
		Switchgear Room		0	significant for maccessible	
	1	Aux FW Pump House		Open	structural steel, select plant	
		Utility/Piping Tunnels		issue	specific aging management	
	and the second second	4. Containment Internal		G-19*	program.	
		Structures		4		
		5. Fuel Storage Facility	Structural			
		Refueling Canal	steel		2 	
		7. Concrete Tanks	cs			
		8. Steel Tanks		·		
	T				D	
Corrosion	LOSS OI	6. Intake Structure	Structural	Same as	Reg. Guide 1.1277 requires in-	Periodic inspections including
	material	Cooling Tower	steel	above	spection at periodic intervals	engineering data compilation &
		Spray Pond			not to exceed 5 yr & includes	onsite inspection program out-
					engineering data compilation &	lined in RG 1.127 <sup>7</sup> are current &
					inspection & evaluation of	effective programs for managing
					concrete surfaces, structural	degradation of Group 6 concrete
					cracking, settlement, & water	structures from corrosion of
					passage.	embedded steel.

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

Aging–Related Degradation Mechanism	Aging Effects	Structures <sup>a</sup>	Materials: <sup>b</sup> Structural Component	NRC Comment Number <sup>c</sup>	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:	Unresolved issue NUMARC proposal: Pressure retaining capability testing, frequency, & accep- tance criteria in accordance with plant's technical specifi- cation. NRC proposal: Demonstrate how building pressurization test is effective in timely detection of corrosion degradation.	<i>NUMARC basis</i> : Routine pressure retaining ca- pability testing is effective in verifying the integrity & for timely detection, corrosion pre- vention/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	S-5 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 de- tection-repair of leaks in spent fuel pool/refueling canal liners.
IGSCC & Crevice corrosion	Crack initiation & growth, loss of material	<ol> <li>Containment Internal Structures</li> <li>Concrete Tanks</li> <li>Steel Tanks</li> </ol>	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.

B-143

NUREG-1557

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

Aging–Related Degradation Mechanism	Aging Effects	Structures <sup>a</sup>	Materials: <sup>b</sup> Structural Component	NRC Comment Number <sup>c</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
Elevated	Loss of	1. BWR Reactor Building	Concrete	G-2,	Non-significant ARDM.	Degradation from exposure to
temperature	strength &	PWR Shield Building	including	G-7,		elevated temperatures is non-
	modulus		rebar:	G-8,		significant for Class 1 concrete
			InW, InS;	G-11,		structures maintained at oper-
		Control Room/Building	Masonry	G-12,		ating temperatures <66°C
			block walls	G-13,		(150°F) and local area tempera-
		2. BWR Reactor Building with		G-16,		tures <93°C (200°F); <sup>18,19</sup> or for
		Steel Superstructure		G-18,		structures that operate above
	•	3. Auxiliary Building		S-6,		these limits, plant-specific justi-
алан (1997) Сайтар		Diesel Generator Building		S-10,		fication is provided in terms of
		Radwaste Building		S-21,	•	concrete strength at elevated
		Turbine Building		S-57.		temperatures or from applica-
		Switchgear Room	,	S-77		tion of special provisions de-
• •		Utility/Piping Tunnels				scribed in ACI 349-85 <sup>3</sup> or justi-
		Aux FW Pump House	-			fied equivalent.
		5. Fuel Storage Facility				Degradation from exposure to
		Refueling Canal	1			elevated temperatures is non-
		4 Containment Internal	Concrete			significant for reinforcing steel
		Structures	including			(rebar) used in Class 1 concrete
			rebar:			maintained at temperatures
			InW & InS			<316°C (600°F). <sup>20</sup>
		6. Intake Structure	Masonry			
		Cooling Tower	block walls			
		7. Concrete Tanks	Concrete in-			
			cluding rebar:			
	$(h_{i},h_{i}) \in [h_{i},h_{i}]$		Int & Ext			
		8. Steel Tanks	Concrete: F			
		9. BWR Unit Vent Stack	Concrete			
			including			
			rebar: GA			

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

	is for	or Proposal	ence levels &	rated gamma	by Class 1 con-	, including re-	ie current & li-	eriod do not ex-	which measur-	n of concrete	ties occurs	2 & 10 <sup>10</sup> rads,	22																
	Basi	Agreement	The neutron flue	maximum integ	doses incurred l	crete structures	bars, for both th	cense renewal p	ceed the level at	able degradation	strength proper	$(5 \times 10^{19} \text{ n/cm}^{2})$	respectively). <sup>21.</sup>											-					
	NUMARC/NRC	Agreement or Proposal	Non-significant ARDM.																										
NRC	Comment	Number <sup>c</sup>	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										
Materials: <sup>b</sup>	Structural	Component	Concrete	including	rebar:	InW, InS;	Masonry	block walls			:									Concrete in-	cluding rebar:	InW & InS	Concrete in-	cluding rebar:	Int & Ext	Concrete: F	Concrete in-	cluding rebar:	
		Structures <sup>a</sup>	1. BWR Reactor Building	<b>PWR Shield Building</b>			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	4. Containment Internal	Structures		7. Concrete Tanks			8. Steel Tanks	9. BWR Unit Vent Stack		
	Aging	Effects	Loss of	strength &	modulus										·														
Aging-Related	Degradation	Mechanism	Irradiation	of Concrete																									

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

NUREG-1557

		r			· · · · · · · · · · · · · · · · · · ·	r
Aging-Related			Materials: <sup>D</sup>	NRC		
Degradation	Aging		Structural	Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Structures <sup>a</sup>	Component	Number <sup>c</sup>	Agreement or Proposal	Agreement or Proposal
Creen*	Deformation	1 BWR Reactor Building	Concrete:	G-2	Non-significant ARDM	Creen experienced by Class 1
unoop	20101111111011	PWR Shield Building	FYA FYB	G-7		reinforced concrete structures is
		I wit billet building	InW InS &	G-8		insignificant because the actual
			F: and	G-11.		compressive stresses
		Control Room/Building	Masonry	G-12.		experienced by the structures
			block walls	G-13.		are generally low.23
		2. BWR Reactor Building with		G-16,		Ŭ
		Steel Superstructure		G-18,		
		3. Auxiliary Building		S-6,		
		Diesel Generator Building		S-12,		
		Radwaste Building		S-21,		
		Turbine Building		S-60,		
		Switchgear Room		S-77		
		Aux FW Pump House	]			
		Utility/Piping Tunnels				
		5. Fuel Storage Facility				
		Refueling Canal				
		6. Intake Structure				
		Cooling Tower	] 			
		Spray Pond				
		4. Containment Internal	Concrete: InW			
		Structures	& 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.

•

Aging–Related			Materials:b	NRC		
Degradation	Aging		Structural	Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Structures <sup>a</sup>	Component	Numberc	Agreement or Proposal	Agreement or Proposal
Shrinkage	Cracking	1. BWR Reactor Building	Concrete:	G-2,	Non-significant ARDM.	Most concrete shrinkage oc-
		PWR Shield Building	ExA, ExB,	G-7,		curs in the first five years of a
			InW,* InS, &	G-8,	· · · ·	structure's life, <sup>20</sup> it is not a
			F; and	G-11,		serious degradation mechanism
		Control Room/Building	Masonry	G-12,		after five years.
			block walls	G-13,		
		2. BWR Reactor Building with		G-16,		
		Steel Superstructure	-	G-18,	· · · · ·	
		3. Auxiliary Building	4	S-6,		
	. · · ·	Diesel Generator Building		S-13,		
		Radwaste Building		S-21,	and the second	
		Turbine Building		S-61,		
		Switchgear Room		S-77		
		Aux FW Pump House	4			
		Utility/Piping Tunnels				
		5. Fuel Storage Facility	ļ			
		Refueling Canal		and the second second		
		6. Intake Structure				
		Cooling Tower	3.			
		Spray Pond				
		4. Containment Internal	Concrete: InW			
		Structures	& 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.

۰Þ

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

Aging-Related			Materials:b	NRC		
Degradation	Aging		Structural	Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Structures <sup>a</sup>	Component	Numberc	Agreement or Proposal	Agreement or Proposal
Abrasion &	loss of	6. Intake Structure*	Concrete:	G-6,	Reg. Guide 1.127 <sup>7</sup> requires	The periodic inspections in-
cavitation	material	Cooling Tower	ExA, ExB	G-10,	inspection at periodic intervals	cluding engineering data
· · ·		Spray Pond	F, & InS	G-14,	not to exceed 5 ys & includes	compilation & onsite inspection
			and the second second	G-15,	engineering data compilation &	program outlined in Regulatory
				S-25,	inspection & evaluation of	Guide 1.127 <sup>7</sup> are current and
			and the second	S-74,	concrete surfaces, structural	effective programs for managing
				S-78,	cracking, settlement, & water	degradation of Group 6 concrete
	,			S-80	passage.	structures from aggressive
						chemicals.
* Non-significan	t ARD mechanis	m for other Class 1 structures.				
Restraint,	Cracking of	1. BWR Reactor Building	Masonry	G-6,	Bulletin 80-11 <sup>24</sup> requires	Inspection requirements im-
Shrinkage,	masonry	PWR Shield Building	block	G-10,	identification of masonry	posed in I&E Bulletin No. 80-
Creep, &	block walls	Control Room/Building	walls	G-14,	walls in close proximity to or	11 <sup>24</sup> & plant-specific monitor-
Aggressive		2. BWR Reactor Building with		G-15,	have attachments from safety	ing requirements proposed by
Environment	· · · ·	Steel Superstructure		S-4,	related piping or equipment &	Information Notice No. 87-6725
		3. Auxiliary Building		S-25,	reevaluation of design ade-	are current & effective programs
		Diesel Generator Building		S-74,	quacy & construction prac-	for cracking of masonry block
		Radwaste Building		S-78,	tices; & Info. Notice 87-67 <sup>25</sup>	walls.
		Turbine Building		S-80	proposed plant-specific cor-	
		Switchgear Room			rective actions.	
		Aux FW Pump House				
		Utility/Piping Tunnels				
		5. Fuel Storage Facility				
		6. Intake Structure				
	· · · · ·	Cooling Tower			· · · · · · · · · · · · · · · · · · ·	
Aging–Related Degradation Mechanism	Aging Effects	Structures <sup>a</sup>	Materials: <sup>b</sup> Structural Component	NRC Comment Number <sup>c</sup>	NUMARC/NRC Agreement or Proposal	Basis for Agreement or Proposal
-------------------------------------------	---------------------------------------------------------	------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------	-------------------------------------------------------------------------------------------------	---------------------------------------	-------------------------------------	---------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------
Cathodic Protection Current	Cathodic protection effect on bond strength	<ol> <li>BWR Reactor Building         <ul> <li>PWR Shield Building</li> <li>Control Room/Building</li> </ul> </li> <li>BWR Reactor Building with         <ul> <li>Steel Superstructure</li> <li>Auxiliary Building</li> <li>Diesel Generator Building</li> <li>Radwaste Building</li> <li>Turbine Building</li> <li>Switchgear Room</li> <li>Aux FW Pump House</li> <li>Utility/Piping Tunnels</li> </ul> </li> <li>Fuel Storage Facility         <ul> <li>Refueling Canal</li> <li>Intake Structure</li> <li>Cooling Tower</li> <li>Spray Pond</li> </ul> </li> </ol>	Concrete: ExA, ExB, InW,* InS, & F	S-19	Non-significant	Cathodic protection systems are designed to operate at ≈21.5 mA/m <sup>2</sup> (≈2 mA/ft <sup>2</sup> ) of steel surface, a level well below the value of 10,764 mA/m <sup>2</sup> (1000 mA/ft <sup>2</sup> ) at which concrete can soften at the reinforcing bar surface. <sup>28</sup>
		<ol> <li>Containment Internal Structures</li> <li>Concrete Tanks</li> <li>Steel Tanks</li> <li>BWR Unit Vent Stack</li> </ol>	Concrete: InW & InS Concrete: F, Int, & Ext Concrete: F Concrete: F, GA, & GB			

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

\* No interior walls InW for Intake Structures.

Aging-Related		r	Materials:b	NPC	Г	1
Degradation	Aging		Structural	Comment		Basis for
Mechanism	Effects	Structuresa	Component	NumberC	Agreement or Proposal	Agreement or Proposal
	Breets			Number	Agreement of Troposal	rigiteement of Proposal
Fatigue	Cumulative	1. BWR Reactor Building	Concrete:	S-18,	Non-significant	Class 1 concrete structures
÷	fatigue	PWR Shield Building	ExA, ExB,	S-70,		subjected to repeated load are
	damage	Control Room/Building	InW, InS, &	S-71,		designed in accordance with
		2. BWR Reactor Building with	F; Structural	S-72		ACI 318 <sup>2</sup> or an equivalent code,
		Steel Superstructure	steel			which limits the max. design
		3. Auxiliary Building				stress level to <50% of static
		Diesel Generator Building				strength (working stress design);
		Radwaste Building				concrete structures can resist
		Turbine Building				$>10^7$ cycles of loading in this
		Switchgear Room				stress range. <sup>26</sup> Class 1 steel
		Utility/Piping Tunnels				structures subjected to repeated
		Aux FW Pump House				loading are designed in accor-
		5. Fuel Storage Facility				dance with AISC Code <sup>27</sup> or its
		Refueling Canal				equivalent, which limits the
a.		4. Containment Internal	Concrete:	]		stress ranges in steel compo-
		Structures	InW & InS;			nents & connections.
			Structural			
			steel &			
			SS liner			
		6. Intake Structure	Concrete:			
т		Cooling Tower	ExA, ExB,			
		Spray Pond	InW, InS, & F			
		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			Materials:b	NRC		
Degradation	Aging		Structural	Comment	NUMARC/NRC	Basis for
Mechanism	Effects	Structures <sup>a</sup>	Component	Number <sup>c</sup>	Agreement or Proposal	Agreement or Proposal
Settlement	Cracking,	1. BWR Reactor Building	Concrete: F	S-26,	Structure settlement	Structure settlement monitor-
1	distortion,	PWR Shield Building		S-75	monitoring during construc-	ing initiated during construc-
	increase in	Control Room/Building			tion, & continued monitoring	tion phase to confirm that ac-
	component	2. BWR Reactor Building with	1		during operation for sites with	tual settlement is consistent
	stress level	Steel Superstructure			soft soil and/or significant	with the allowances included in
		3. Auxiliary Building		ł	changes in ground water	design basis, & continued
		Diesel Generator Building	1	ļ	conditions.§	settlement monitoring during
an an an an an a	· .	Radwaste Building				operation for sites with soft soil
		Turbine Building	]			and/or significant changes in
		Switchgear Room				ground water conditions are
		Aux FW Pump House				current & effective programs to
	$(\alpha_{i})_{i \in \mathbb{N}} = (\alpha_{i})_{i \in \mathbb{N}} = (\alpha_{i})$	Utility/Piping Tunnels	1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -			assure structural integrity &
	1.1.1.1	5. Fuel Storage Facility	a	(		functionality of Class 1 struc-
	· ·	Refueling Canal				tures.
	1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 - 1997 -	6. Intake Structure				
	- * , ,	Cooling Tower			and the second	
		Spray Pond	1			
and the second se		7. Concrete Tanks				
1. A.	1 d. %	8. Steel Tanks			l	
and the second		9. BWR Unit Vent Stack				

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

§ 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.

<sup>b</sup> 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.

<sup>c</sup> 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.

## **REFERENCES:**

- 1. ACI SP100-59, Vol. 2, "Evalution and Prediction of Concrete Durability Ontario Hydro's Experience," V. Sturrup, R. Horton, P. Mukherjee, and T. Carmichael, American Concrete Institute, 1987.
- 2. ACI 318-63, "Building Code Requirements for Reinforced Concrete," American Concrete Institute.
- 3. ACI 349-85, "Code Requirements for Nuclear Safety Related Concrete Structures," American Concrete Institute.
- 4. ACI 201.2R-67, "Guide to Durable Concrete," American Concrete Institute.
- 5. "Concrete Degradation Monitoring and Evaluation," N. Prasad et al., NUREG/CP-0100, Proc. Intl. Nuclear Power Plant Aging Symposium,
- 6. ACI 207 3R-79, "Practices for Evaluation of Concrete in Existing Massive Structures for Service Condition," American Concrete Institute, Revised 1985.
- 7. Regulatory Guide 1.127, Revision 1, "Inspection of Water-Control Structures Associated with Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1978.
- 8. "Petrographic Identification of Reactive Constituents in Concrete Aggregate," B. Mather, ASTM Proc. Vol. 48, American Society of Testing and Materials, Philadelphia, PA, pp. 1120-1125, 1948.
- 9. ASTM C295-54, "Practice for Petrographic Examination of Aggregates for Concrete," American Society of Testing and Materials, Philadelphia, PA.
- 10. ASTM C227-50, "Potential Alkali Reactivity of Cement Aggregate Combination," American Society of Testing and Materials, Philadelphia, PA.
- 11. ACI 222R-85, "Corrosion of Metals in Concrete," American Concrete Institute.
- 12. "Concrete Structure, Properties and Materials," P. K. Kumar, Prentice-Hall, Inc., 1986.
- 13. ASTM E94-77, "Recommended Practice for Radiographic Testing," American Society of Testing and Materials, Philadelphia, PA.
- 14. ASTM E709-80, "Magnetic Particle Examination," American Society of Testing and Materials, Philadelphia, PA.
- 15. ASTM E165-80, "Liquid Penetrant Inspection Method," American Society of Testing and Materials, Philadelphia, PA.
- 16. "Corrosion of Steel Pilings in Soils," M. Ramanoff, National Bureau of Standards, Monograph 58, October 1962.
- 17. "Performance of Steel Pilings in Soils," M. Ramanoff, Proc. 25th Conf. National Association of Corrosion Engineers, March 1969.
- 18. "Composition and Properties of Concrete," Second Edition, G. E Troxell, H. E. Davis, and J. W. Kelly, McGraw-Hill, 1968.
- 19. ORNL/TM-7632, "Task 2: Concrete Properties in Nuclear Environment A Review of Concrete Material Systems for Applications to Prestressed Concrete Pressure Vessels," Oak Ridge National Laboratory, May 1981.
- 20. "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.
- 21. NUREG/CR-4652, ORNL/TM-10059, "Concrete Component Aging and its Significance Relative to Life Extension of Nuclear Power Plants," D. J. Naus, Oak Ridge National Laboaratory, September 1986.

NUREG-1557

- 22. ACI Publication SP-55, "The Effects of Nuclear Radiation on the Mechanical Properties of Concrete," H. R. Hilsdorf, J. Kropp, and H. J. Koch, Douglas McHenry Intl. Symp. on Concrete and Concrete Structures, American Concrete Institute, 1978.
- 23. ACI 209 R-82, "Prediction of Creep, Shrinkage and Temperature Effects in Concrete Structures," American Concrete Institute.
- 24. Inspection and Enforcement Bulletin No. 80-11, "Masonry Wall Design," U.S. Nuclear Regulatory Commission, May 9, 1980.
- 25. Information Notice 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to I & E Bulletin 80-11," U.S. Nuclear Regulatory Commission, December 31, 1987.
- 26. ACI 215 R-74, "Consideration for Design of Concrete Structures Subjected to Fatigue Loading," American Concrete Institute, 1986.
- 27. "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," American Institute of Steel Construction, Revision 1970.
- 28. "Corrosion of Steel in Portland Cement Concrete: Fundamental Studies," C. E. Locke, in Corrosion Effect of Stray Currents and the Techniques for Evaluation Corrosion of Rebars in Concrete, American Society for Testing and Materials, STP 906, Philadelphia, PA.

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<sup>†</sup>
- 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
Con	crete 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	Cracking of masonry walls
	Aggressive Environment	
12.	Concrete Interaction with Aluminum	Loss of strength
13.	Cathodic Protection Current	Cathodic protection effect on bond strength
Stru	ctural Steel & Stainless Steel Liner	
1.	Corrosion, Local Corrosion,	Loss of material
	Atmospheric Corrosion	
2.	Elevated Temperature	Loss of strength and modulus
3.	Irradiation	Loss of fracture toughness
4.	Stress Corrosion Cracking	Crack initiation and growth
Reir	nforcing 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
Mise	cellaneous	
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

C-1

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,+	Loss of material++++
	Microbiologically induced corrosion	
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
+ Bo	ric 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

U.S. NUCLEAR REGULATORY COMMISSION  I. REPORT NUCLEAR REGULATORY COMMISSION  I. REPORT NUCLEAR SUBJECT  Son, so  IDELLOGRAPHIC DATA SHEET  Commission  Commissin  Commissin	INC FORM 335       U.S. NUCLEAR REGULATORY COMMISSION (2-89)       1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.)         3201, 3202       BIBLIOGRAPHIC DATA SHEET (See instructions on the reverse)       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       YEAR         5. AUTHOR(S)       C. Regan, S. Lee/NRC       6. TYPE OF REPORT       6. TYPE OF REPORT         C. Regan, S. Lee/NRC       Technical       7. PERIOD COVERED (inclusive Dates)         8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, end mailing address; if contractor, provide name and mailing address.)       Division of Reactor Program Management       Argonne National Laboratory
BIBLIOGRAPHIC DATA SHEET  Comments of the means Data Sheet	NRCM 1102, 3201, 3202       BIBLIOGRAPHIC DATA SHEET (see instructions on the reverse)       Image: Comparison of the reverse)         2. TITLE AND SUBTITLE       NUREG-1557         Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal       3. DATE REPORT PUBLISHED MONTH         S. AUTHOR(S)       0. TYPE OF REPORT         C. Regan, S. Lee/NRC       6. TYPE OF REPORT         O. K. Chopra, D. C. Ma, W. J. Shack/ANL       Technical         8. PERFORMING ORGANIZATION - NAME AND ADDRESS (# NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)       Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, Argonne National Laboratory
(December 2)     (	3201, 3202       (See instructions on the reverse)       NUREG-1557         2. TITLE AND SUBTITLE       3. DATE REPORT PUBLISHED       3. DATE REPORT PUBLISHED         Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal       3. DATE REPORT PUBLISHED         MONTH       YEAR       October       1996         4. FIN OR GRANT NUMBER       4. FIN OR GRANT NUMBER       4. FIN OR GRANT NUMBER         5. AUTHOR(S)       6. TYPE OF REPORT       Technical         C. Regan, S. Lee/NRC       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         Argonne National Laboratory       Argonne National Laboratory       Division
2. TITLE AND SUBTITLE Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal Summary of Technical Information and Agreements form Nuclear Management and Resources Council Industry Reports Regar, S. Lee/NRC O. K. Chopra, D. C. Ma, W. J. Shack/ANL Summary of Regarded and addressing the Regarded and Agreement and Resources Council Industry Regards Summary of Regarded and addressing the Regard and Agreement Argonne National Laboratory Office of Nuclear Resource Regulatory Commission Washington, DC 20555-0001 9. EPCRORENC ORGANZATION - NAME AND ADDRESS (FINRC, type Same as above? Footback: provide NRC Divide, Otice of Regard U.S. Nuclear Regulatory Commission Washington, DC 20555-0001 9. EPCRORENC ORGANZATION - NAME AND ADDRESS (FINRC, type Same as above? Footback: provide NRC Divide, Otice of Regord, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001 9. EPCRORENC ORGANZATION - NAME AND ADDRESS (FINRC, type Same as above? Footback: provide NRC Divide, Otice of Regord, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001 9. EPCRORENC ORGANZATION - NAME AND ADDRESS (FINRC, type Same as above? Footback: provide NRC Divide, Otice of Regord, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001 9. EPCRORENC ORGANZATION - NAME AND ADDRESS (FINRC, type Same as above? Footback: provide NRC Divide, Otice of Regord, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001 9. EPCRORENC ORGANZATION - NAME AND ADDRESS (FINRC, type Same as above? Footback: Divide NRC Divide, Otice of Regord, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001 9. EPCRORENC ORGANZATION - NAME AND ADDRESS (FINRC, type Same as above? Footback: Divide NRC Divid	2. TITLE AND SUBTITLE       NUREG-1557         Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal       3. DATE REPORT PUBLISHED         MONTH       YEAR         October       1996         4. FIN OR GRANT NUMBER       6. TYPE OF REPORT         5. AUTHOR(S)       6. TYPE OF REPORT         C. Regan, S. Lee/NRC       Technical         O. K. Chopra, D. C. Ma, W. J. Shack/ANL       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       Argonne National Laboratory
Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal       3. DATE REPORT PUBLISHED MOTIVI YEAR CCtober 1998         S. AUTHOR(6)       8. TYPE OF REPORT         C. Regan, S. Lee/NRC       9. TYPE OF REPORT         O. K. Chopra, D. C. Ma, W. J. Shack/ANL       7. PEROCOVERED meansure demay         S. PERFORMING ORGANIZATION - NAME AND ADDRESS (#VMC, provide Division, Office or Region, U.S. Nuclear Regulator, and mailing address.)       9. TYPE OF REPORT         S. PERFORMING ORGANIZATION - NAME AND ADDRESS (#VMC, provide Division, Office or Region, U.S. Nuclear Regulator, and mailing address.)       9. Technical         Office of Nuclear Rescut Regulatory Commission       Argonne, IL 60439       10. SUPLEMENTARY NOTES         10. SUPPLEMENTARY NOTES       11. ASSTRACT (20 avoide a feasible seasement. The NC factor for nuclear Resources Council (NUMARC) submitted for NRC review ten industry reports (RE) addressing aging issues associated with specific structures and components of nuclear power plants and one IR addressing the screening metantion and agreements documents have been compiled as public documents. This report provides the Resources the status of the NRC staffs arow when the NRC staffs and hours on and agreements documents have been compiled as public documents. The NRC staff for and nuclear Management and Resources the status of the NRC staffs for and one IR addressing the screening from the ten Resources to the comments in ave been compiled as public documents. This report provides a brief summary of the technical information and agreements documented berein represent the status of the NRC staffs for and one IR addressing the s	Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal
Control industry Reports Addressing License Renewal     C. Regan, S. LeefNRC     C. Regan, S. LeefNRC     C. Regan, S. LeefNRC     C. K. Chopra, D. C. Ma, W. J. Shack/ANL     C. Regan, S. LeefNRC     C. K. Chopra, D. C. Ma, W. J. Shack/ANL     C. Regan, S. LeefNRC     C. K. Chopra, D. C. Ma, W. J. Shack/ANL     C. Technical     T. PERIOD COVERED (numeric terms and terms gradients)     Division of Reactor Program Management     Argonne National Laboratory     Office of Nuclear Resource Regulatory     Commission     Washington, DC 20555-0001     Sonceare Regulatory Commission     Washington, DC 20555-0001     Sonceare Regulatory Commission     Washington, DC 2055-0001     Sonceare Regulatory Commission     Washington Regulatory Commission     Washington	3. DATE REPORT PUBLISHED         Resources Council Industry Reports Addressing License Renewal         3. DATE REPORT PUBLISHED         MONTH       YEAR         October       1996         4. FIN OR GRANT NUMBER         5. AUTHOR(S)       6. TYPE OF REPORT         C. Regan, S. Lee/NRC       Technical         O. K. Chopra, D. C. Ma, W. J. Shack/ANL       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       Argonne National Laboratory
AUTHOR(S)      AUTHOR(S)      AUTHOR(S)      C. Regan, S. Lee/NRC      O. K. Chopra, D. C. Ma, W. J. Shack/ANL       Deleteronamic and approximate approximate and approximate approxis approximate approximate approxim	MONTH       YEAR         October       1996         4. FIN OR GRANT NUMBER         5. AUTHOR(S)         C. Regan, S. Lee/NRC         O. K. Chopra, D. C. Ma, W. J. Shack/ANL         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.)         B. 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
Cobber 1996     A FN OR GRANT NUMBER      S. AUTHOR(S)     C. Regan, S. Lee/NRC     O. K. Chopra, D. C. Ma, W. J. Shack/ANL       C. Regan, S. Lee/NRC     O. K. Chopra, D. C. Ma, W. J. Shack/ANL       C. Technical     Technical     T. PERIOD COVERED (inclusive Dates)       Cover and the set of the se	October       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
A UTHOR(s)     A FIN OR GRANT NUMBER	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
S. AUTHOR(S) C. Regan, S. Lee/NRC O. K. Chopra, D. C. Ma, W. J. Shack/ANL C. Performance and making address, J. Contract Control of the cont	5. AUTHOR(S)       6. TYPE OF REPORT         C. Regan, S. Lee/NRC       Technical         O. K. Chopra, D. C. Ma, W. J. Shack/ANL       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.)       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       Argonne National Laboratory
C. Regan, S. Lee/NRC O. K. Chopra, D. C. Ma, W. J. Shack/ANL Technical T. PEROD COVERED (industre Dates)  B. PERFORMING ORGANIZATION - NAME AND ADDRESS (INRC, provide Division. Office or Rights, U.S. Nuclear Regulatory Commission, and mailing address, if contracts, provide and and mailing address) Division of Reactor Program Management U.S. Nuclear Regulatory Commission Washington, DC 20555-0001 B. SPORDENG ORGANIZATION - NAME AND ADDRESS (INRC, type "Same as above"; if contracts, provide MRC Division, Office or Region, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001 B. SPORDENG ORGANIZATION - NAME AND ADDRESS (INRC, type "Same as above"; if contracts, provide MRC Division, Office or Region, U.S. Nuclear Regulatory Commission Washington, DC 20555-0001 B. SPORDENG ORGANIZATION - NAME AND ADDRESS (INRC, type "Same as above"; if contracts, provide MRC Division, Office or Region, U.S. Nuclear Regulatory Commission washington, DC 20555-0001 B. SPORDENG ORGANIZATION - NAME AND ADDRESS (INRC, type "Same as above"; if contracts, provide MRC Division, Office or Region, U.S. Nuclear Regulatory Commission washington, DC 20555-0001 B. SPORDENG ORGANIZATION - NAME AND ADDRESS (INRC, type "Same as above"; if contracts, provide MRC Division, Office or Region, U.S. Nuclear Regulatory Commission washington, DC 20555-0001 B. SPORDENG ORGANIZATION - NAME AND ADDRESS (INRC, type "Same as above"; if contracts, provide MRC Division, Office or Region, U.S. Nuclear Regulatory Commission some as 8, above. B. SUPPLEMENTARY NOTES II. ADSTRACT (200 words or face) II. ADSTRACT contexts to the comments that be been compiled as public documents. This report provides a brief summary of the tochnical information and NUMARC/INRC agreements from the ten INS, except for the Cable License Renewal IR. The tochnical information and AUMARC/INRC agreements from the ten INS, except for the Cable License Renewal IR. The tochnical information	C. Regan, S. Lee/NRC O. K. Chopra, D. C. Ma, W. J. Shack/ANL 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 Argonne National Laboratory
O. K. Chopra, D. C. Ma, W. J. Shack/ANL  T. PERIOD COVERED (inclusive Deeley)  PERFORMING ORGANIZATION - NAME AND ADDRESS (f WIC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address, if contractor, provide name and mailing address, and mailing address, if contractor, provide name and mailing address, or contraction, and mailing address, if contractor, PONSION of Reactor Program Management Argonne National Laboratory Office of Nuclear Regulatory Commission Washington, DC 20555-0001 9. SPONSONING ORGANIZATION - NAME AND ADDRESS (if WIC, type "Same as above", if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission I. I. ADVILABULY NOTES  11. ADSTRACT (200 words or leave	O. K. Chopra, D. C. Ma, W. J. Shack/ANL       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       Argonne National Laboratory
PERFORMING ORGANIZATION - NAME AND ADDRESS (#NRC, provide Division, Office of Region, U.S. Nuclear Regulatory Commission, and mailing address; (# contracts, provide name and mailing address)      Division of Reactor Program Management Argonne National Laboratory      Office of Nuclear Reagulation Argonne, IL 60439      U.S. Nuclear Regulatory Commission      Washington, D.C 20555-0001      SPENIOSTING DEGARIZATION - NAME AND ADDRESS (# NRC, type "Same as above"; # contracts, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)      Sento SGNK DEGARIZATION - NAME AND ADDRESS (# NRC, type "Same as above"; # contracts, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)      Same as 8, above.       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/CR cagements have been compiled as public documents. This report provides a brief summary of the technical information and agreements documented herain 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 renewall.      If AVALABLITY STATEMENT     unlimited     Itactory 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 renewall.	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         Argonne National Laboratory
B. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Divisor, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address, If contractor, provide name and mailing address, If contractor, Office of Nuclear Regulatory Commission Argonne, IL 60439 U.S. Nuclear Regulatory Commission Washington, DC 20555-0001 9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type Same as above; If contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address, J Contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission 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 bes) 11. ADDIT 1990, the Nuclear Management and Resources Council (NUMARC) submitted for NRC review ten industry reports (IRS) addressing gaing 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 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 plants to incorporate appropriate technical information and agreements into the draft standard review plan for license renewal.  12. KEY WORDS/DESCRIPTORS (List words or physee that will assist researching in for license renewal.  13. AVALABLITY STATEMENT Unlimited 14. SECURT CLASSIFICATION Class I Structures Reactor Vessel Containment Class I Structures Reactor Vessel Containment Class I Structures Reactor Vessel Containment Class I Structures Reactory Vessel Containment Class I Structures Reactory Vessel Containment Class I St	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 Argonne National Laboratory
	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 Argonne National Laboratory
Division of Reactor Program Management         Argonne National Laboratory           Office of Nuclear Regulatory Commission         Argonne, IL 60439           U.S. Nuclear Regulatory Commission         Washington, DC 20555-0001           9 SPINDSRINE ORGANIZATION - NAME AND ADDRESS (# NRC, type "Same as above"; # contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and making address.           9 SPINDSRINE ORGANIZATION - NAME AND ADDRESS (# NRC, type "Same as above"; # contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and making address.           10. SUPPLEMENTARY NOTES           11. ABSTRACT (200 words or bass)           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 birle summary of the technical information and NUMARC/NRC agreements from the ten IRs, except for the Cable License Renewal IR.           12. KEY WORDS/DESCRIPTORS (Lat words or phrases that will easted neseworkers in localing the report)         11. AVAILABILITY STATEMENT           Industry Reports         12. KEY WORDS/DESCRIPTORS (Lat words or phrases that will easted neseworkers in localing the report)         13. AVAILABILITY STATEMENT           Industry Reports         Reactor Vessel         1	provide name and mailing address.) Division of Reactor Program Management Argonne National Laboratory
Chice of Nuclear Rescutor Frugulation analogement Argonne National Laboratory Office of Nuclear Rescutor Regulatory Commission Washington, DC 20555-0001 Second Provide NRC Division, DC 20555-0001 Second Provide Pro	Division of Reactor Program Management Argonne National Laboratory
U.S. Nuclear Regulatory Commission Washington, DC 20555-0001 9. SPCNSQRING ORGANIZATION - NAME AND ADDRESS (# NRC, type "Same as above"; # 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 best) 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/NRC agreements documents have been compiled as public documents. This report provides a brief summary of the technical information and AUMARC/NRC/NRC agreements form the ten IRs, except for the Cable License Renewal IR. The technical information and agreements documented herein represent the status of the NRC staff plants to incorporate appropriate technical information and agreements documented herein represent the status of the NRC staff plants to incorporate appropriate technical information and agreements documented herein represent the status of the NRC staff plants to incorporate appropriate technical information and agreements documented herein represent the status of the NRC staff plants to incorporate appropriate technical information and agreements into the draft standard review plan for license renewal.	Office of Nuclear Deputer Deputeries
0. 3. Nuclear Regulatory Commission Washington, DC 2055-0001         9. SPONSORING ORGANIZATION - NAME AND ADDRESS (#NRC, type "Same as above"; # contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and making 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 (Rs; 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 plans to incorporate appropriate technical information and agreements into the draft standard review plan for license renewal.         12. KEY WORDS/DESCRIPTORS (List words or phreses thet will essist researchers in locating the report)       13. AVALABILITY STATEMENT Unlimited         14. SECURITY CLASSIFICATION Class I Structures Reactor Vessel       13. AVALABILITY STATEMENT Unlimited         14. SECURITY CLASSIFICATION (The Review Lessel Internation       13. AVALABILITY STATEMENT Unlimited         14. SECURITY CLASSIFICATION (The Review Lessel Internation       13. AVALABILITY STATEMENT Unclassified The Secure Vessel	Office of Nuclear Reactor Regulation Argonne, iL 60439
12. KEY WORDS/DESCRIPTORS (Let words or phrases that will assist researchers in locating the report)       13. AVAILABILITY STATEMENT         12. KEY WORDS/DESCRIPTORS (Let words or phrases that will assist researchers in locating the report)       13. AVAILABILITY STATEMENT         11. Addition and agreements into the draft standard review plan for license renewal.       13. AVAILABILITY STATEMENT	Washington DC 20555-0001
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 agreements documented herein represent the status of the NRC staff plans to incorporate Renewal IR. The technical information and agreements documented herein represent the NRC staff plans to incorporate appropriate technical information and agreements into the draft standard review plan for license renewal.         12. KEY WORDS/DESCRIPTORS (Lat words or phrases that will assist researchers in locating the report) <ul> <li>Industry Reports</li> <li>Reactor Vessel</li> <li>Containment</li> <li>Class I Structures</li> <li>Parent</li> <li>Vessel Internals</li> </ul> 13. AVAILABLITY STATEMENT <ul> <li>unclassified</li> <li>The vessel Structures</li> <li>unclassified</li> <li>The vessel</li> <li>Containment</li> <li>Containment</li> </ul> 13. AVAILABLITY STATEMENT <ul> <li>unclassified</li> <li>The vessel</li> <li>unclassified</li> <li>The vessel</li> <li>the secource</li> </ul> <	9. SPONSORING ORGANIZATION - NAME AND ADDRESS (# NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission,
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 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 localing the report)       13. AVALABILITY STATEMENT unlimited         14. MSCLENTY Reports       13. AVALABILITY STATEMENT       14. ASSIGNATION         Containment       Containments       14. SSIGNATION         Class I Structures       0.       14. SSIGNATION         Cheered       0.       14. SSIGNATION	and mailing address.)
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/IRC 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 plans to incorporate appropriate technical information and agreements into the draft standard review plan for license renewal.         12. KEY WORDS/DESCRIPTORS (Lat words or phrases that will essist researchers in localing the report)       13. AVAILABILITY STATEMENT unlimited         12. KEY WORDS/DESCRIPTORS (Lat words or phrases that will essist researchers in localing the report)       14. SECURITY CASSINGLATION (This Pege)         12. KEY WORDS/DESCRIPTORS (Lat words or phrases that will essist researchers in localing the report)       14. SECURITY CASSINGLATION (This Pege)         12. KEY WORDS/DESCRIPTORS (Lat words or phrases that will essist researchers in localing the report)       14. SECURITY CASSINGLATION (This Pege)         12. KEY WORDS/DESCRIPTORS (Lat words or phrases that will essist researchers in localing the report)       14. SECURITY CASSINGLATION (This Pege)         14. SECURITY CLASSINGLATION       14. SECURTY CLASSINGLATION (This Pege)	Same as 8. above.
10. SUPPLEMENTARY NOTES         11. ABSTRACT (200 words or less)         11. ADDET (200 words or less)         12. KEY WORDS/DESCRIPTORS (Lat words or phrases that will essist researchers in locating the report)         12. KEY WORDS/DESCRIPTORS (Lat words or phrases that will essist researchers in locating the report)         13. AVAILABILITY STATEMENT         14. SECURITY CAUSE         15. AVAILABILITY STATEMENT         16. ADDE (Lat words or phrases that will essist researchers in locating the report)         11. Industry Reports	
10. SUPPLEMENTARY NOTES         11. ABSTRACT (200 words or less)         11. about 1990, the Nuclear Management and Resources Council (NUMARC) submitted for NRC review ten industry reports (IRs) 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/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 plans to incorporate appropriate technical information and agreements into the draft standard review plan for license renewal.         12. KEY WORDS/DESCRIPTORS (Lat words or phrases that will essist researchers in locating the report)       13. AVAILABILITY STATEMENT unlimited         14. SECURITY CASSIFICATION       14. SECURITY CLASSIFICATION       14. SECURITY CLASSIFICATION         (This Page)       Unclassified       14. SECURITY CLASSIFICATION       14. SECURITY CLASSIFICATION	
11. ABSTRACT (200 words or less)         11. ABSTRACT (200 words or less)         11. 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 essist researchers in loceling the report)       13. AVAILABILITY STATEMENT         Undustry Reports       14. SECURITY CLASSIFICATION         Containment       Class I Structures         Class I Structures       14. SECURITY CLASSIFICATION	
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 (Lat words or phrases that will assist researchers in localing the report.)       13. AVAILABILITY STATEMENT unlimited         14. SECURITY CLASSIFICATION       Class I Structures       14. SECURITY CLASSIFICATION         Containment       Class I Structures       unclassified	
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 Agreements documented herein represent the status of the NRC staff 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.	11. ABSTRACT (200 words or less)
12. KEY WORDS/DESCRIPTORS (List words or phreses thet will essist researchers in locating the grant stand by the draft standard review plan for license renewal.       13. AVAILABILITY STATEMENT         12. KEY WORDS/DESCRIPTORS (List words or phreses thet will essist researchers in locating the report.)       13. AVAILABILITY STATEMENT         13. AVAILABILITY STATEMENT       unlimited         14. SECURITY CLASSIFICATION       14. SECURITY CLASSIFICATION         (The technical information and agreements into the draft standard review plan for license renewal.       13. AVAILABILITY STATEMENT         12. KEY WORDS/DESCRIPTORS (List words or phreses thet will essist researchers in locating the report.)       13. AVAILABILITY STATEMENT         13. AVAILABILITY STATEMENT       unlimited         14. SECURITY CLASSIFICATION       14. SECURITY CLASSIFICATION	In sheet 4000, the Nuclear Management and Descurres Ocurall (NII MADO) submitted for NDO regions for industry reports (IDs)
12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report)       13. AVAILABILITY STATEMENT unlimited         12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report)       13. AVAILABILITY STATEMENT unlimited         14. SECURITY CLASSIFICATION (This Page)       13. AVAILABILITY STATEMENT unlimited         14. SECURITY CLASSIFICATION (This Page)       13. AVAILABILITY STATEMENT unclassified	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 puclear power plants and one IR addressing the
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)       13. AVAILABILITY STATEMENT unlimited         14. SECURITY CLASSIFICATION       14. SECURITY CLASSIFICATION         Containment       Containment         Class I Structures       Beactor Vessel         Reactor Vessel       Industry responses	screening methodology for integrated plant assessment. The NRC staff had been reviewing the ten NUMARC IRs; their comments
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 essist researchers in locating the report.) Industry Reports Reactor Vessel Containment Class I Structures Reactor Vessel Internals	on each IR and NUMARC responses to the comments have been compiled as public documents. This report provides a brief
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 essist researchers in locating the report.)       13. AVAILABILITY STATEMENT         Industry Reports       unlimited         Reactor Vessel       14. SECURITY CLASSIFICATION         Class I Structures       unclassified         The Staff plans to incorporate appropriate technical information and agreements into the draft standard review plan for license renewal.	summary of the technical information and NUMARC/NRC agreements from the ten IRs, except for the Cable License Renewal IR.
12. KEY WORDS/DESCRIPTORS (List words or phreses that will assist researchers in locating the report.) <ul> <li>Industry Reports</li> <li>Reactor Vessel</li> <li>Containment</li> <li>Class I Structures</li> <li>Report Vessel</li> <li>Inclustry resources</li> <li>Inclustry Reports</li> <li>Inclus</li></ul>	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
12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)  13. AVAILABILITY STATEMENT  14. SECURITY CLASSIFICATION  14. SECURITY CLASSIFICATION  14. SECURITY CLASSIFICATION  15. Class I Structures  Reactor Vessel Internals  Chip Securit	technical information and agreements into the draft standard review plan for license renewal.
12. KEY WORDS/DESCRIPTORS (List words or phreses that will assist researchers in locating the report.)       13. AVAILABILITY STATEMENT         Industry Reports       unlimited         Reactor Vessel       14. SECURITY CLASSIFICATION         Containment       (This Pege)         Class I Structures       unclassified         Reactor Vessel       (This rege)         Unclassified       (This pege)	
12. KEY WORDS/DESCRIPTORS (List words or phreses that will essist researchers in locating the report.)  13. AVAILABILITY STATEMENT  14. SECURITY CLASSIFICATION  14. SECURITY CLASSIFICATION  14. SECURITY CLASSIFICATION  15. Containment  Class I Structures  Reactor Vessel Internals  Chin Percent	
12. KEY WORDS/DESCRIPTORS (List words or phreses that will assist researchers in locating the report.)  13. AVAILABILITY STATEMENT  14. SECURITY CLASSIFICATION  14. SECURITY CLASSIFICATION  14. SECURITY CLASSIFICATION  15. Containment  Class I Structures  Reactor Vessel Internals  Chic Security  Class I Structures  Class I Structures  Comparison  Compa	
12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)       13. AVAILABILITY STATEMENT         Industry Reports       unlimited         Reactor Vessel       14. SECURITY CLASSIFICATION         Containment       (This Pege)         Class I Structures       unclassified         Reactor Vessel       (This Pege)	
12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)  13. AVAILABILITY STATEMENT  14. SECURITY CLASSIFICATION  14. SECURITY CLASSIFICATION  (This Pege)  (This Pege	
12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)  13. AVAILABILITY STATEMENT  14. SECURITY CLASSIFICATION  14. SECURITY CLASSIFICATION  (This Page)  (This Page)  Unclassified  (This Page)  (This Pag	
12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)  13. AVAILABILITY STATEMENT  14. SECURITY CLASSIFICATION  (This Page)  Class I Structures  Reactor Vessel Internals  (This Securit	
12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)  13. AVAILABILITY STATEMENT  14. SECURITY CLASSIFICATION  14. SECURITY CLASSIFICATION  (This Page)  Unclassified  (This Page)  (This Pag	
Industry Reports Reactor Vessel Containment Class I Structures Reactor Vessel Internals (This Pege) (This Pegee) (This Pege) (This Pege) (This Pegee) (This Peg	
Industry Reports Reactor Vessel Containment Class I Structures Reactor Vessel Internals	12. KET WORDSIDESCHIP TORS (List words or privases that will assist researchers in locating the report.)
Containment Class I Structures Reactor Vessel Internals	Unlimited
Class I Structures unclassified	Industry Reports unlimited Exactor Vessel 14. SECURITY CLASSIFICATION
Reactor Vessel Internals	Industry Reports Reactor Vessel Containment
A nine (Inis Report)	Industry Reports       Industry Reports     unlimited       Reactor Vessel     14. SECURITY CLASSIFICATION       Containment     (This Page)       Class I Structures     unclassified
Aging unclassified	Industry Reports       unlimited         Reactor Vessel       14. SECURITY CLASSIFICATION         Containment       (This Pege)         Class I Structures       unclassified         Reactor Vessel Internals       (This Report)
Degradation Mechanisms 15. NUMBER OF PAGES	Industry Reports Reactor Vessel Containment Class I Structures Reactor Vessel Internals Aging License Renewal
	Industry Reports       unlimited         Reactor Vessel       14. SECURITY CLASSIFICATION         Containment       (This Pege)         Class I Structures       unclassified         Reactor Vessel Internals       (This Report)         Aging       unclassified         License Renewal       0         Degradation Mechanisms       15. NUMBER OF PAGES
16. PRICE	Industry Reports       unlimited         Reactor Vessel       14. SECURITY CLASSIFICATION         Containment       (This Pege)         Class I Structures       unclassified         Reactor Vessel Internals       (This Report)         Aging       unclassified         License Renewal       15. NUMBER OF PAGES         Degradation Mechanisms       16. PRICE
	Industry Reports       unlimited         Reactor Vessel       14. SECURITY CLASSIFICATION         Containment       (This Pege)         Class I Structures       unclassified         Reactor Vessel Internals       (This Report)         Aging       unclassified         License Renewal       15. NUMBER OF PAGES