In-Service Inspection and Monitoring Requirements for Highly-Likely Degradation Scenarios from NUREG/CR-6923, “Expert Panel Report on Proactive Materials Degradation Assessment”

Technical Letter Report

Greg Oberson

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

July 2013

Table of Contents

Tables

Acronyms

AMP	Aging management program
ASME	American Society of Mechanical Engineers
B&W	Babcock and Wilcox
BMI	Bottom mounted instrumentation
BWR	Boiling water reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CVCS	Chemical and volume control system
CE	Combustion Engineering
CFR	Code of Federal Regulations
CRDM	Control rod drive mechanism
DP	Differential pressure
EPRI	Electric Power Research Institute
FAC	Flow assisted corrosion
GALL	Generic Aging Lessons Learned
HAZ	Heat affected zone
HPCS	High pressure cores spray
HPSI	High pressure safety injection
ICI	Incore instrumentation
ID	Inner diameter
IGSCC	Intergranular stress corrosion cracking
IMI	Incore monitoring instrumentation
LPCI	Low pressure coolant injection
MA	Mill annealed
MRP	Materials Reliability Program
MIC	Microbiologically influenced corrosion
NRC	Nuclear Regulatory Commission
PORV	Power operated relief valve
PWR	Pressurized water reactor
RCIC	Reactor core isolation cooling
RHR	Residual heat removal
RPV	Reactor pressure vessel
SCC	Stress corrosion cracking
SG	Steam generator
SI	Safety injection
SLC	Standby liquid control
SS	Stainless steel
TT	Thermally treated
U.S.	United States

Executive Summary

The potential for aging related materials degradation of reactor components is an important consideration for the long-term safe operation of nuclear power plants. In recent years, the U.S. Nuclear Regulatory Commission (NRC) engaged in activities to proactively identify potential degradation scenarios so that appropriate research or regulatory action could be undertaken before plant safety was compromised. One such activity was an expert panel convened to evaluate passive components in the primary, secondary, and some tertiary systems, the failure of which could lead to a release of radioactivity, or could adversely affect the functionality of the safety systems. The panel evaluated the current state of knowledge and attempted to qualify the likelihood for a range of degradation scenarios that could affect these components. The panel findings were reported in NUREG/CR-6923 “Expert Panel Report on Proactive Materials Degradation Assessment.” In NUREG/CR-6923, a degradation scenario was characterized as a highly-likely scenario if there was demonstrated, compelling evidence for occurrence, or multiple plant observations.

Effective management of materials degradation involves the implementation of in-service inspection and monitoring programs. The purpose of this report is to summarize these programs for the degradation scenarios characterized as highly likely in NUREG/CR-6923. Inspection and monitoring programs are incorporated into NRC’s regulatory framework in a variety of ways. Title 10 of the Code of Federal Regulations, Part 50.55a incorporates by reference the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” as well as selected ASME code cases. Industry groups, such as the Electric Power Research Institute (EPRI), may propose enhancements or alternatives to the Section XI requirements. NRC may review and endorse these alternative programs for generic or conditional use. For emerging safety issues that could affect multiple plants, NRC may also issue generic communications to request information concerning licensee programs for detecting and managing degradation. Moreover, the monitoring and inspection programs described above are included as part of aging management programs (AMPs) for plants with renewed licenses. Revision 2 of NUREG-1801, “Generic Aging Lessons Learned (GALL) Report,” provides the latest NRC guidance on AMPs that NRC staff has determined to be acceptable to meet the requirements for license renewal, including any subsequent guidance changes through the license renewal interim staff guidance (LR-ISG) process.

A total of 162 degradation scenarios were characterized as highly likely in NUREG/CR-6923, 65 for pressurized water reactor (PWR) components, and 97 for boiling water reactor (BWR) components. Nearly all of the components for BWRs and PWRs are subject to inspection under Section XI of the ASME Boiler and Pressure Vessel Code or NRC-approved code cases. For many components, NRC has also endorsed alternative inspection programs. For PWRs, these include EPRI Materials Reliability Program and Nuclear Energy Institute Steam Generator Program guidelines, among others. For BWRs, these are largely the EPRI Boiling Water Reactor Vessel and Internals Program guidelines.

Stress corrosion cracking of the steam generator divider plate in PWRs is the only scenario for which an inspection and monitoring program was not identified. Cracking in this component has been reported in French plants but inspections have not yet been performed in U.S. plants. The operating experience to date indicates that the cracking in the French plants is minor and does not appear to be growing through the divider plate quickly, if at all. In addition, EPRI Report 1016552, “Divider Plate Cracking in Steam Generators: Results of Phase II: Evaluation of the Impact of a Cracked Divider Plate on LOCA and Non-LOCA Analyses (2008), concluded that extensive cracking of the divider plate is not a safety concern. The staff has been requesting further information on this issue for applicants seeking plant license renewal, given the potential for cracks in the divider plate to grow into the pressure boundary. The industry is currently performing analyses to determine the long-term safety significance of divider plate cracking.

Acknowledgements

The author also gratefully acknowledges the review and suggestions provided by Kenneth Karwoski, Allen Hiser, Gene Carpenter, and Robert Tregoning.

1 Introduction

Materials degradation has been observed in a range of nuclear power plant primary pressure boundary components, including primary water stress corrosion cracking in the hot leg at V.C. Summer, circumferential cracking in the vessel head penetration nozzles at Oconee, Unit 3, wastage in the reactor pressure vessel head at Davis-Besse, Unit 1, and cracks in the lower head penetrations at South Texas, Unit 1. In some cases, the conditions that brought about degradation were not fully appreciated by the U.S. Nuclear Regulatory Commission (NRC) or industry until after these events occurred. As such, NRC and industry needed to undertake time-intensive and costly efforts to understand the scope of the issues and develop appropriate regulatory and corrective actions. A majority of the domestic reactor fleet is expected to apply for a license renewal to operate for 60 years and industry is actively considering a second license renewal to operate up to 80 years. Therefore, the potential for further aging related degradation is an important consideration for long-term safe operation of nuclear power plants.

In light of the materials degradation events described above, in recent years NRC has engaged in activities to proactively identify potential degradation scenarios so that appropriate research or regulatory action could be undertaken before plant safety was compromised. One such activity was an expert panel convened to evaluate passive components in the primary, secondary, and some tertiary systems, the failure of which could lead to a release of radioactivity, or could adversely affect the functionality of the safety systems. The panel evaluated the current state of knowledge and attempted to qualify the likelihood for a range of degradation scenarios that could affect these components. The panel findings were issued in NUREG/CR-6923 “Expert Panel Report on Proactive Materials Degradation Assessment.” NUREG/CR-6923 considered a potential degradation scenario to be a highly likely scenario if there was demonstrated, compelling evidence for occurrence, or multiple plant observations.

Effective management of potential degradation scenarios is likely to involve the implementation of in-service inspection and monitoring programs. Inspection and monitoring programs are incorporated into NRC’s regulatory framework in a variety of ways. Title 10 of the Code of Federal Regulations (CFR), Part 50.55a incorporates by reference the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” as well as selected ASME code cases. Industry groups, such as the Electric Power Research Institute (EPRI), may propose enhancements or alternatives to the Section XI requirements. NRC may review and endorse these alternative programs for generic or conditional use. For emerging safety issues that could affect multiple plants, NRC may also issue generic communications to request information concerning licensee programs for detecting and managing degradation.

For plants seeking renewal of their operating license beyond the initial 40 year period, the provisions of 10 CFR, Part 54, “Requirements for the Renewal of Operating Licenses for Nuclear Power Plants,” require that licensees demonstrate that the effects of aging will be adequately managed during the period of extended operation. The monitoring and inspection programs described above are included as part of aging management programs for plants with renewed licenses. NUREG-1801, “Generic Aging Lessons Learned (GALL) Report,” Revision 2

(2010) describes generic aging management programs (AMPs) that NRC staff determined to be acceptable to meet the requirements of 10 CFR, Part 54, including any subsequent guidance changes through the license renewal interim staff guidance (LR-ISG) process. One element of the GALL Report AMPs is "Detection of Aging Effects," which includes inspection and monitoring techniques to detect aging effects in components before there is a loss of intended function. Some of the AMPs provided in the GALL Report reference the provisions of the ASME Boiler and Pressure Vessel Code, or approved alternative requirements described above.

The purpose of this report is to list and summarize the inspection and monitoring programs for degradation scenarios characterized as highly likely in NUREG/CR-6923. Section 2 describes the approach for this report. Sections 3 and 4 describe the programs for pressurized water reactor (PWR) and boiling water reactor (BWR) components, respectively. Section 5 provides a summary of the findings.

2 Approach

To identify inspection and monitoring programs, sources reviewed include Section XI of the ASME Boiler and Pressure Vessel Code (2007 Edition, 2008 Addenda) and approved code cases, EPRI and other industry guidelines, NRC generic communications, and Revision 2 of the GALL Report, including changes implemented by LR-ISGs. In this report, reactor components for PWRs and BWRs, respectively, were placed into groups for which the inspection and monitoring programs are similar. For each group, the tables list the references for the inspection and monitoring programs which may be applicable. The inspection methods are summarized in the appendices. Different plants may have different classifications for certain component groups. Therefore, all programs may not apply at each plant. Plant safety analysis reports and in-service inspection reports should be referred to for specific applicability. For each component group, the AMP(s) from Revision 2 of the GALL Report (including changes implemented by LR-ISGs) which may be applicable are also listed.

3 Inspection and Monitoring Programs – PWR Components

3.1 PWR Alloy 600/82/182 Nozzles and Dissimilar Metal Welds

Table 3-1 lists the degradation scenarios for Alloy 600/82/182 nozzles and dissimilar metal welds in PWRs that NUREG/CR-6923 characterized as highly likely.

Table 3-1: Highly likely degradation scenarios for PWR Alloy 600/82/182 nozzles and dissimilar metal welds

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Pressurizer	Alloy 82/182 dissimilar metal welds	Primary water	SCC
	Forged Alloy 600 nozzles	Primary water	SCC
RPV	Alloy 82/182 dissimilar metal welds	Primary water	Fatigue
			SCC
	Forged alloy 600 nozzles	Primary water	SCC
Steam generator	Alloy 82/182 dissimilar metal welds	Primary water	SCC

The references for the inspection programs which may apply to Alloy 600/82/182 nozzles and dissimilar metal welds are listed in Table 3-2, along with the appendices where the inspection methods are summarized.

Table 3-2: Inspection programs for PWR Alloy 600/82/182 nozzles and dissimilar metal welds

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-F, "Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles"	C
ASME Code, Section XI, Category B-P, "All Pressure Retaining Components"	I
[1]ASME Code Case N-722, "Additional Examination Requirements for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials"	P
[1]ASME Code Case N-770, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities"	R

[1]See 10 CFR, Part 50.55a – Codes and standards

Alloy 600/82/182 nozzles and dissimilar metal welds may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M11B – Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components (PWRs Only)

3.2 Steam Generator Tubes

Table 3-3 lists the degradation scenarios for steam generator tubes in PWRs that NUREG/CR-6923 characterized as highly likely.

Table 3-3: Highly likely degradation scenarios for steam generator tubes

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Steam generator tubes	Alloy 600 mill annealed	Primary water	SCC
		Secondary water	Wear
			SCC
	Alloy 600 thermally treated	Primary water	SCC
		Secondary water	Wear
			SCC
	Alloy 600 sensitized	Primary water	SCC
		Secondary water	Wear
			SCC

The references for the inspection programs which may apply to steam generator tubes are listed in Table 3-4, along with the appendices where the inspection methods are summarized.

Table 3-4: Inspection programs for steam generator tubes

REFERENCE	APPENDIX
[1,2,3]NEI 97-06, "Steam Generator Program Guidelines"	AI
Steam Generator Management Program, "Pressurized Water Reactor Steam Generator Examination Guidelines"	AP
"Steam Generator Integrity Assessment Guidelines"	AQ

[1]See NUREG-1430, Standard Technical Specifications, Babcock and Wilcox Plants, Volume 1, Specification 3.4.17, "Steam Generator (SG) Tube Integrity"
[2]See NUREG-1431, Standard Technical Specifications, Westinghouse Plants, Volume 1, Specification 3.4.20, "Steam Generator (SG) Tube Integrity"
[3]See NUREG-1432, Standard Technical Specifications, Combustion Engineering Plants, Volume 1, Specification 3.4.18, "Steam Generator (SG) Tube Integrity"

Steam generator tubes may also be in the scope of the following GALL Report AMP:

- XI.M19 – Steam Generators

3.3 PWR Socket Welds

Table 3-5 lists the degradation scenarios for socket welds in PWRs that NUREG/CR-6923 characterized as highly likely.

Table 3-5: Highly likely degradation scenarios for PWR socket welds

SYSTEM/COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
• Cold leg piping • Crossover leg piping • Hot leg piping	304-308/316SS	Primary water	Fatigue
• Pressurizer spray piping • Pressurizer surge piping • Pressurizer piping to PORVs	304-308/316SS	Primary water	Fatigue
CVCS piping	304SS	Primary water	Fatigue

The references for the inspection programs which may apply to socket welds in PWRs are listed in Table 3-6, along with the appendices where the inspection methods are summarized.

Table 3-6: Inspection programs for PWR socket welds

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-J, “Pressure Retaining Welds in Piping”	F
ASME Code, Section XI, Category B-P, “All Pressure Retaining Components”	I
ASME Code, Section XI, Category C-F-1, “Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping”	L
ASME Code, Section XI,, Category C-H, “All Pressure Retaining Components”	N
MRP-146, “Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines”	AI

The PWR socket welds may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M35 – One-Time Inspection of ASME Code Class 1 Small-Bore Piping

3.4 PWR Reactor Pressure Vessel and Control Rod Drive Mechanism Housing

Table 3-7 lists the degradation scenarios for the RPV and CRDM housing in PWRs that NUREG/CR-6923 characterized as highly likely.

Table 3-7: Highly likely degradation scenarios for RPV and CRDM housing

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
RPV shell, plate, forging, welds	A533 or SA508	Primary water	Boric acid corrosion
CRDM housing	304/308SS	Primary water (stagnant)	SCC

The references for the inspection programs which may apply to the RPV and CRDM housing in PWRs are listed in Table 3-8, along with the appendices where the inspection methods are summarized.

Table 3-8: Inspection programs for PWR RPV and CRDM Housing

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-A, “Pressure Retaining Welds in Reactor Vessel”	A
ASME Code, Section XI, Category B-O, “Pressure Retaining Welds in Control Rod Drive and Instrument Nozzle Housings”	H
ASME Code, Section XI, Category B-P, “All Pressure Retaining Components”	I
[1]ASME Code Case N-729, “Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds”	Q
NRC Generic Letter 88-05 “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants”	AK
WCAP-15988, “Generic Guidance for an Effective Boric Acid Inspection Program for Pressurized Water Reactors.”	AR

[1]See Title 10 of the Code of Federal Regulations, Part 50.55a – Codes and standards

The PWR RPV and CRDM housing may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M10 – Boric Acid Corrosion

3.5 Service Water System Components

Table 3-9 lists the degradation scenarios for the service water system components in PWRs that NUREG/CR-6923 characterized as highly likely.

Table 3-9: Highly likely degradation scenarios for PWR service water system

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Service water suction piping from pond	Carbon steel components, welds, HAZ	Pond water	MIC
			Pitting
		Lake/sea salt water	MIC
			Pitting
			Crevice corrosion
Service water pump discharge piping	Carbon steel components, welds, HAZ	Pond water	MIC
			Pitting
	Closed cooling water heat exchanger tubes, Cu-Zn	Pond or closed cooling water	Pitting
			SCC
	Carbon steel closed cooling water heat exchanger shell, tubesheets, fittings	Pond water	MIC
			Pitting
Service water piping inside containment	Carbon steel components, welds, HAZ	Pond water	MIC
			Pitting

The references for the inspection programs which may apply to the PWR service water system components are listed in Table 3-10, along with the appendices where the inspection methods are summarized.

Table 3-10: Inspection programs for PWR service water system components

REFERENCE	APPENDIX
ASME Code, Section XI, Category C-A, "Pressure Retaining Welds in Pressure Vessels"	J
ASME Code, Section XI, Category C-F-2, "Pressure Retaining Welds in Carbon or Low Alloy Steel Piping"	M
ASME Code, Section XI, Category C-H, "All Pressure Retaining Components"	N
ASME Code, Section XI, Category D-B, "All Pressure Retaining Components"	O
NRC Generic Letter 89-13 "Service Water System Problems Affecting Safety-Related Equipment"	AM

The service water system components may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M20 – Open-Cycle Cooling Water System
- XI.M21A – Closed Treated Water Systems

3.6 PWR Internals

Table 3-11 lists the degradation scenarios for the internals in PWRs that NUREG/CR-6923 characterized as highly likely.

Table 3-11: Highly likely degradation scenarios for PWR internals

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Internals	304 stainless steel HAZ (>0.5 dpa)	Primary water	SCC
	308 stainless steel weld metal (>0.5 dpa)	Primary water	SCC
	316 cold worked stainless steel (>0.5 dpa)	Primary water	Irradiation creep
			Swelling
			SCC
	High strength bolts, A286/X750 (>0.5 dpa)	Primary water	Irradiation creep
			Swelling
			SCC
	High strength fasteners/springs, X750, 718	Primary water	Irradiation creep
	304 stainless steel plates/tubes (>0.5 dpa)	Primary water	SCC

The references for the inspection programs which may apply to PWR internals are listed in Table 3-12, along with the appendices where the inspection methods are summarized.

Table 3-12: Inspection programs for PWR internals

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-N-2, “Welded Core Support Structures and Internal Attachments to Reactor Vessels” and Category B-N-3, “Removable Core Support Structures”	G
MRP-227, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines”	AH

The PWR internals may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M16A – PWR Vessel Internals (see also LR-ISG-2011-04, ADAMS Accession No. ML12270A436)

3.7 PWR Piping Dissimilar Metal Welds

Table 3-13 lists the degradation scenarios for the piping dissimilar metal welds in PWRs that NUREG/CR-6923 characterized as highly likely.

Table 3-13: Highly likely degradation scenarios for PWR piping dissimilar metal welds

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Accumulator and CVCS piping to cold leg	308/309 stainless steel, Alloys 82/182 dissimilar metal weld (CE/B&W)	Primary water	SCC
SI/RHR piping to hot leg	308/309 stainless steel, Alloys 82/182 dissimilar metal weld (CE/B&W)	Primary water	SCC
• Cold leg piping • Crossover leg piping • Hot leg piping	308/309 stainless steel dissimilar metal weld	Primary water	SCC

The references for the inspection programs which may apply to piping dissimilar metal welds are listed in Table 3-14, along with the appendices where the inspection methods are summarized.

Table 3-14: Inspection programs for PWR piping dissimilar metal welds

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-J, “Pressure Retaining Welds in Piping”	F
ASME Code, Section XI, Category B-P, “All Pressure Retaining Components”	I
ASME Code, Section XI, Category C-F-1, “Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping”	L
ASME Code, Section XI, Category C-H, “All Pressure Retaining Components”	N
[1]ASME Code Case N-722, “Additional Examination Requirements for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials”	P

[1]See Title 10 of the Code of Federal Regulations, Part 50.55a – Codes and standards

PWR piping dissimilar metal welds may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M11B – Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components

3.8 PWR Carbon Steel Components

Table 3-15 lists the degradation scenarios for carbon steel components in PWRs that NUREG/CR-6923 characterized as highly likely.

Table 3-15: Highly likely degradation scenarios for PWR carbon steel components

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Main steam	Carbon steel components and welds	Saturated steam, 445-530^{o}F	FAC
Main feedwater	Carbon steel components and welds	Water, 450^{o}F	FAC
			Fatigue
Steam generator blowdown	Carbon steel components and welds	Saturated water, 550^{o}F	FAC
			Fatigue
Steam generator drilled hole tube sheet plate	Carbon steel	Secondary water, 544-620^{o}F	FAC

The references for the inspection programs which may apply to carbon steel components are listed in Table 3-16, along with the appendices where the inspection methods are summarized.

Table 3-16: Inspection programs for PWR carbon steel components

REFERENCE	APPENDIX
ASME Code, Section XI, Category C-F-2, “Pressure Retaining Welds in Carbon or Low Alloy Steel Piping”	M
ASME Code, Section XI, Category C-H, “All Pressure Retaining Components”	N
NRC Generic Letter 89-08 “Erosion/Corrosion Induced Pipe Wall Thinning”	AL
NSAC-202L, ”Recommendations for an Effective Flow-Accelerated Corrosion Program”	AN

The carbon steel components may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M17 – Flow-Accelerated Corrosion

3.9 CE Pressurizer Heater Sleeves

Table 3-17 lists the degradation scenario for CE pressurizer heater sleeves that NUREG/CR-6923 characterized as highly likely.

Table 3-17: Highly likely degradation scenarios for CE pressurizer heater sleeves

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Pressurizer heater sleeve (CE)	Alloy 600 cold worked	Primary water	SCC

The references for the inspection programs which may apply to CE pressurizer heater sleeves are listed in Table 3-18, along with the appendices where the inspection methods are summarized.

Table 3-18: Inspection programs for CE pressurizer heater sleeves

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-P, “All Pressure Retaining Components”	I
ASME Code Case N-722, “Additional Examination Requirements for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials”	P
WCAP-16180, “WOG CE Fleet Pressurizer Heater Sleeve Inspection Program”	AS

The CE pressurizer heater sleeves may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M11B – Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in Reactor Coolant Pressure Boundary Components

3.10 Reactor Coolant Pump Components

Table 3-19 lists the degradation scenario for reactor coolant pump components that NUREG/CR-6923 characterized as highly likely.

3-19: Highly likely degradation scenarios for reactor coolant pump components

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Reactor coolant pump high strength parts	A286, 17-4PH, 403, X750	Primary water, 558°F	SCC

The references for the inspection programs which may apply to reactor coolant pump components are listed in Table 3-20, along with the appendices where the programs are summarized.

Table 3-20: Inspection programs for reactor coolant pump components

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-G-1, “Pressure Retaining Bolting Greater than 2 in. in Diameter”	D
ASME Code, Section XI, Category B-G-2, “Pressure Retaining Bolting 2 in. and Less in Diameter”	E
ASME Code, Section XI, Category C-D, “Pressure Retaining Bolting Greater than 2 in. in Diameter”	K

The reactor coolant pump components may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M18 – Bolting Integrity

3.11 Steam Generator Divider Plate

Table 3-21 lists the degradation scenario for the steam generator divider plate that NUREG/CR-6923 characterized as highly likely.

3-21: Highly likely degradation scenario for the steam generator divider plate

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Steam generator divider plate	Alloy 600	Primary water, 556-620^{o}F	SCC

There is no specified inspection guidance for this component. Cracking in this component has been reported in French plants but inspections have not been performed in U.S. plants. The operating experience to date indicates that the cracking in the French plants is minor and does not appear to be growing through the divider plate quickly, if at all. In addition, EPRI Report 1016552, "Divider Plate Cracking in Steam Generators: Results of Phase II: Evaluation of the Impact of a Cracked Divider Plate on LOCA and Non-LOCA Analyses (2008), concluded that extensive cracking of the divider plate is not a safety concern. The staff has been requesting further information on this issue for applicants seeking plant license renewal, given the potential for cracks in the divider plate to grow into the pressure boundary. The industry is currently performing analyses to determine whether this concern is warranted.

4 Inspection and Monitoring Programs – BWR Components

4.1 Jet Pump Components

Table 4-1 lists the degradation scenarios for jet pumps in BWRs that NUREG/CR-6923 characterized as highly likely.

Table 4-1: Highly likely degradation scenarios for jet pump components

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Jet pump assembly	304 SS HAZ, pipes and elbows	Reactor water, 525^{o}F	SCC
	X750 holddown beam	Reactor water, 525^{o}F	SCC
	Alloy 600, access hole cover	Reactor water, 525^{o}F	SCC
	Alloy 182 weld metal, access hole cover	Reactor water, 525^{o}F	SCC
	304 SS riser brace weld HAZ	Reactor water, 525^{o}F	SCC
	SS HAZ on adapter and diffuser welds	Reactor water, 525^{o}F	SCC
	Alloy 182 adapter-to-ledge weld metal	Reactor water, 525^{o}F	SCC
	304 SS HAZ, riser bracket	Reactor water, 525^{o}F	SCC

The references for the inspection programs which may apply to jet pump components are listed in Table 4-2, along with the appendices where the inspection methods are summarized.

Table 4-2: Inspection programs for jet pump components

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-N-2, "Welded Core Support Structures and Internal Attachments to Reactor Vessels"	G
BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines"	X
BWRVIP-48, "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines"	Z

Jet pump components may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M4 – BWR Vessel ID Attachment Welds
- XI.M9 – BWR Vessel Internals

4.2 Core Shroud

Table 4-3 lists the degradation scenarios for the core shroud in BWRs that NUREG/CR-6923 characterized as highly likely.

Table 4-3: Highly likely degradation scenarios for core shroud

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Core shroud	304 SS HAZ, vertical and circumferential weld	Reactor water, 533°F	SCC
	Alloy 182, shell-shroud support ring weld metal	Reactor water, 533°F	SCC
	304 SS HAZ, shell-shroud support ring	Reactor water, 533°F	SCC
	304 SS (with 308L welds), miscellaneous brackets and pads	Reactor water, 533°F	SCC

The references for the inspection programs which may apply to core shrouds are listed in Table 4-4, along with the appendices where the inspection methods are summarized.

Table 4-4: Inspection programs for core shroud components

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-N-2, “Welded Core Support Structures and Internal Attachments to Reactor Vessels”	G
BWRVIP-38, “BWR Shroud Support Inspection and Flaw Evaluation Guidelines”	W
BWRVIP-76, “BWR Core Shroud Inspection and Flaw Evaluation Guidelines”	AC

The core shroud may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M9 – BWR Vessel Internals

4.3 Core Plate

Table 4-5 lists the degradation scenarios for the core plate in BWRs that NUREG/CR-6923 characterized as highly likely.

Table 4-5: Highly likely degradation scenarios for core plate

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Core plate	304 SS, core plate structure	Reactor water, 533^{o}F	SCC
	X750, core plate bypass flow plug spring	Reactor water, 533^{o}F	SCC

The references for the inspection programs which may apply to the core plate are listed in Table 4-6, along with the appendices where the inspection methods are summarized.

Table 4-6: Inspection programs for core plate

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-N-2, “Welded Core Support Structures and Internal Attachments to Reactor Vessels”	G
BWRVIP-25, “BWR Core Plate Inspection and Flaw Evaluation Guidelines”	T

The core plate may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M9 – BWR Vessel Internals

4.4 Top Guide

Table 4-7 lists the degradation scenarios for the top guide in BWRs that NUREG/CR-6923 characterized as highly likely.

Table 4-7: Highly likely degradation scenarios for top guide

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Top guide	304 SS, guide structure (moderate to high dose)	Reactor water, 533^{o}F	SCC

The references for the inspection programs which may apply to the top guide are listed in Table 4-8, along with the appendices where the inspection methods are summarized.

Table 4-8: Inspection programs for top guide

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-N-2, “Welded Core Support Structures and Internal Attachments to Reactor Vessels”	G
BWRVIP-26, “BWR Top Guide Inspection and Flaw Evaluation Guidelines”	U
BWRVIP-183, “Top Guide Grid Beam Inspection and Flaw Evaluation Guidelines”	AE

The top guide may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M9 – BWR Vessel Internals

4.5 Steam Separator and Dryer

Table 4-9 lists the degradation scenarios for the steam separator and dryer in BWRs that NUREG/CR-6923 characterized as highly likely.

Table 4-9: Highly likely degradation scenarios for steam separator and dryer

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Steam separator and dryer	304 SS, steam separator and dryer	Reactor water/wet steam, 550°F	Fatigue
	308L SS, steam dryer weld metal	Reactor water/wet steam, 550°F	Fatigue
	304 SS, steam dryer HAZ	Reactor water/wet steam, 550°F	Fatigue
			SCC
	Alloy 182, dryer hold down bracket weld	Reactor coolant steam, 547°F	SCC

The references for the inspection programs which may apply to the steam separator and dryer are listed in Table 4-10, along with the appendices where the inspection methods are summarized.

Table 4-10: Inspection programs for steam separator and dryer

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-N-2, "Welded Core Support Structures and Internal Attachments to Reactor Vessels"	G
BWRVIP-48: "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines"	Z
BWRVIP-139, "Steam Dryer Inspection and Flaw Evaluation Guidelines"	AD

The steam separator and dryer may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M4 – BWR Vessel ID Attachment Welds
- XI.M9 – BWR Vessel Internals

4.6 RPV Flange

Table 4-11 lists the degradation scenarios for the BWR RPV flange that NUREG/CR-6923 characterized as highly likely.

Table 4-11: Highly likely degradation scenarios for BWR RPV closure head flange

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
RPV flange	Alloy 182 weld, A508 nozzle to 304 SS flange	Reactor coolant steam, 547°F	SCC
	304 SS, flange HAZ	Reactor coolant steam, 547°F	SCC

The references for the inspection programs which may apply to the RPV closure head flange are listed in Table 4-12, along with the appendices where the inspection methods are summarized.

Table 4-12: Inspection programs for BWR RPV closure head flange

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-A, "Pressure Retaining Welds in Reactor Vessel"	A
ASME Code, Section XI, Category B-F, "Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles"	C
ASME Code, Section XI, Category B-P, "All Pressure Retaining Components"	I

The RPV flange may also be in the scope of the following GALL Report AMP:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

4.7 BWR Feedwater Nozzle

Table 4-13 lists the degradation scenarios for the BWR feedwater nozzle that NUREG/CR-6923 characterized as highly likely.

Table 4-13: Highly likely degradation scenarios for BWR feedwater nozzle and sparger

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Feedwater nozzle	Alloy 600, safe end	Reactor water, 427^{o}F	SCC
	Alloy 600, thermal sleeve	Reactor water, 427^{o}F	SCC
	Alloy 182 weld, thermal sleeve to A508 nozzle	Reactor water, 427^{o}F	SCC
	304 SS weld HAZ, pipe and header to T-box welds	Reactor water, 427^{o}F	SCC
	304 SS weld HAZ, sparger	Reactor water, 427^{o}F	SCC

The references for the inspection programs which may apply to the feedwater nozzle are listed in Table 4-14, along with the appendices where the inspection methods are summarized.

Table 4-14: Inspection programs for BWR feedwater nozzle and sparger

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-F, “Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles”	C
ASME Code, Section XI, Category B-P, “All Pressure Retaining Components”	I
GE-NE-523-A71-0594-A, “Alternate BWR Feedwater Nozzle Inspection Requirements”	AF
NUREG-0619, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking: Resolution of Generic Technical Activity A-10”	AO

The feedwater nozzle may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M5 – BWR Feedwater Nozzle

4.8 Differential Pressure/Standby Liquid Control Penetration

Table 4-15 lists the degradation scenarios for the differential pressure/standby liquid control penetrations that NUREG/CR-6923 characterized as highly likely.

Table 4-15: Highly likely degradation scenarios for BWR differential pressure/standby liquid control penetrations

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
DP and SLC control structures	Alloy 182 weld metal	Reactor water, 525°F	SCC
	304 SS HAZ	Reactor water, 525°F	SCC

The references for the inspection programs which may apply to the differential pressure/standby liquid control penetrations are listed in Table 4-16, along with the appendices where the programs are summarized.

Table 4-16: Inspection programs for BWR differential pressure/standby liquid control penetration

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-D, “Full Penetration Welded Nozzles in Vessels”	B
ASME Code, Section XI, Category B-F, “Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles”	C
ASME Code, Section XI, Category B-J, “Pressure Retaining Welds in Piping”	F
ASME Code, Section XI, Category B-P, “All Pressure Retaining Components”	I
BWRVIP-27, “BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines”	V

The differential pressure/standby liquid control penetrations may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M8 – BWR Penetrations

4.9 Core Spray Piping

Table 4-17 lists the degradation scenarios for the BWR core spray piping that NUREG/CR-6923 characterized as highly likely.

Table 4-17: Highly likely degradation scenarios for BWR core spray piping

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Core spray piping	304 SS HAZ	Reactor water, 533^{o}F	SCC

The references for the inspection programs which may apply to core spray piping are listed in Table 4-18, along with the appendices where the inspection methods are summarized.

Table 4-18: Inspection programs for BWR core spray piping

REFERENCE	APPENDIX
ASME Code, Section XI, Category C-F-1, "Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping"	L
ASME Code, Section XI, Category C-H, "All Pressure Retaining Components"	N
BWRVIP-18, "BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines"	S

The core spray piping may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M9 – BWR Vessel Internals

4.10 Austenitic SS Piping

Table 4-19 lists the degradation scenarios for the austenitic SS piping in BWRs that NUREG/CR-6923 characterized as highly likely.

Table 4-19: Highly likely degradation scenarios for BWR piping, nozzles, and penetrations

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Recirculation system	316 SS HAZ	Reactor water, 550°F	SCC
	304 SS HAZ	Reactor water, 550°F	SCC
	308 weld metal on 304/316SS, socket welds	Reactor water, 550°F	SCC
			Fatigue
RWCU system	304 SS HAZ, elbows, reducers, and pipe	BWR water, 535°F	SCC
	304 SS base metal, weld, and HAZ, weldolets and sockolets	BWR water, 535°F	SCC
RHR system	304 HAZ, RHR suction line to RHR pumps	BWR water, 549°F	SCC
LPCI system	304 SS HAZ	Reactor water, 533°F	SCC

The references for the inspection programs which may apply to the austenitic SS piping are listed in Table 4-20, along with the appendices where the inspection methods are summarized.

Table 4-20: Inspection programs for BWR piping, nozzles, and penetrations

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-J, “Pressure Retaining Welds in Piping”	F
ASME Code, Section XI, Category B-P, “All Pressure Retaining Components”	I
ASME Code, Section XI, Category C-F-1, “Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping”	L
ASME Code, Section X, Category C-H, “All Pressure Retaining Components”	N
BWRVIP-75, “Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedule”	AB
NRC Generic Letter 88-01, “NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping”	AJ

The austenitic SS piping may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M7 – BWR Stress Corrosion Cracking
- XI.M25 – Reactor Water Cleanup System

4.11 Bottom Head Penetrations

Table 4-21 lists the degradation scenarios for the bottom head penetrations in BWRs that NUREG/CR-6923 characterized as highly likely.

Table 4-21: Highly likely degradation scenarios for BWR bottom head penetrations

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
RPV bottom head	Alloy 82/182 J-groove weld metal, ICI and CRD penetrations	Reactor water, 533-547^{o}F	SCC
	316NG or 316L SS HAZ, safe end and thermal sleeve	Reactor water, 533-547^{o}F	SCC
	304 SS HAZ safe end	Reactor water, 533-547^{o}F	SCC
	Alloy 82/182 weld pad between thermal sleeve and nozzle	Reactor water, 533-547^{o}F	SCC
	Alloy 600 HAZ	Reactor water, 533-547^{o}F	SCC
Core controls	Alloy 182 weld metal, RPV stub tube to CRD housing	Reactor water, 525^{o}F	SCC
	304 SS HAZ, RPV stub tube to CRD housing	Reactor water, 525^{o}F	SCC
	Alloy 600 HAZ, RPV stub tube to CRD housing	Reactor water, 525^{o}F	SCC
	304 SS in core guide tube assembly	Reactor water, 525^{o}F	SCC

The references for the inspection programs which may apply to the bottom head penetrations are listed in Table 4-22, along with the appendices where the inspection methods are summarized.

Table 4-22: Inspection programs for BWR bottom head penetrations

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-J, “Pressure Retaining Welds in Piping”	F
ASME Code, Section XI, Category B-O, “Pressure Retaining Welds in Control Rod Drive and Instrument Nozzle Housings”	H
ASME Code, Section XI, Category B-P, “All Pressure Retaining Components”	I
BWRVIP-47, “BWR Lower Plenum Inspection and Flaw Evaluation Guidelines”	Y

The bottom head penetrations may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M7 – BWR Stress Corrosion Cracking
- XI.M8 – BWR Penetrations

4.12 Penetrations and Attachments

Table 4-23 lists the degradation scenarios for the penetrations and attachments in BWRs that NUREG/CR-6923 characterized as highly likely.

Table 4-23: Highly likely degradation scenarios for BWR penetrations and attachments

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
RPV shell	316 SS safe ends and thermal sleeves	Reactor water, 427°F	SCC
	Alloy 82/182 weld between carbon steel and Alloy 600	Reactor water, 427°F	SCC
	Alloy 82/182 weld pad between A508 and Alloy 600	Reactor water, 533°F	SCC
	Alloy 182 attachment pads	Reactor water or steam, 575°F	SCC

The references for the inspection program which may apply to penetrations and attachments are listed in Table 4-24, along with the appendices where the inspection methods are summarized.

Table 4-24: Inspection programs for BWR penetrations and attachments

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-D, “Full Penetration Welded Nozzles in Vessels”	B
ASME Code, Section XI, Category B-F, “Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles”	C
ASME Code, Section XI, Category B-P, “All Pressure Retaining Components”	I
BWRVIP-48: Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines”	Z
BWRVIP-49, “Instrument Penetration Inspection and Flaw Evaluation Guidelines”	AA

The penetrations and attachments may also be in the scope of the following GALL Report AMPs:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M4 – BWR Vessel ID Attachment Welds
- XI.M8 – BWR Penetrations

4.13 BWR Carbon Steel Components

Table 4-25 lists the degradation scenarios for the BWR carbon steel components that NUREG/CR-6923 characterized as highly likely.

Table 4-25: Highly likely degradation scenarios for BWR carbon steel components

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
RCIC	Carbon steel base metal and weld, weldolets and sockolets	Stagnant wet steam, 547°F	Fatigue
RHR system	Carbon steel-brass joint, RHR spray piping, sockolets	Suppression pool water, <100°F	Crevice corrosion
			Galvanic corrosion
Condensate storage tank	Carbon steel SA105, 106, 216, 234 elbows, flanges, and pipes	Condensate storage water, <100°F	Crevice corrosion
			MIC
			Pitting
	Carbon steel SA105, 106, 216, 234 weld/HAZ	Condensate storage water, <100°F	Crevice corrosion
			MIC
			Pitting

The references for the inspection programs which may apply to carbon steel components are listed in Table 4-26, along with the appendices where the inspection methods are summarized.

Table 4-26: Inspection programs for BWR carbon steel components

REFERENCE	APPENDIX
ASME Code, Section XI, Category C-F-2, “Pressure Retaining Welds in Carbon or Low Alloy Steel Piping”	M
ASME Code, Section XI, Category C-H, “All Pressure Retaining Components”	N
ASME Code, Section XI, Category D-B, “All Pressure Retaining Components”	O

The carbon steel components may also be in the scope of the following GALL Report AMP:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

4.14 BWR Bolting

Table 4-27 lists the degradation scenarios for bolting that NUREG/CR-6923 characterized as highly likely.

Table 4-27: Highly likely degradation scenarios for BWR bolting

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Low pressure core spray	Carbon steel and higher strength bolts at strainer	Suppression pool water, <100°F	Crevice corrosion
Main steam	Low alloy steel bolts for t-quencher/sparger	Suppression pool water, <90°F	Pitting
			SCC
HPCS pump	Carbon steel/low alloy steel A106, A516 parts	Condensate storage water, <100°F	Crevice corrosion
			Pitting
			MIC

The references for the inspection programs which may apply to BWR bolting are listed in Table 4-28, along with the appendices where the inspection methods are summarized.

Table 4-28: Inspection programs for BWR bolting

REFERENCE	APPENDIX
ASME Code, Section XI, Category C-D, “Pressure Retaining Bolting Greater than 2 in. in Diameter”	K
ASME Code, Section XI, Category C-H, “All Pressure Retaining Components”	N

The BWR bolts may also be in the scope of the following GALL Report AMP:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD
- XI.M18 – Bolting Integrity

4.15 Heat Exchanger Components

Table 4-29 lists the degradation scenarios for the BWR heat exchanger components that NUREG/CR-6923 characterized as highly likely.

Table 4-29: Highly likely degradation scenarios for BWR heat exchanger components

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Main condenser	Titanium tubes, exterior	Wet steam	Erosion corrosion
RHR system	Carbon steel heat exchanger fitting, RHR pump discharge piping to RHR heat exchanger	Suppression pool water, 125-334°F	Pitting

The references for the inspection programs which may apply to heat exchanger components are listed in Table 4-30, along with the appendices where the inspection methods are summarized.

Table 4-30: Inspection programs for BWR heat exchanger components

REFERENCE	APPENDIX
ASME Code, Section XI, Category B-P, “All Pressure Retaining Components”	I
ASME Code, Section XI, Category C-H, “All Pressure Retaining Components”	N

The heat exchanger components may also be in the scope of the following GALL Report AMP:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

4.16 BWR Valves

Table 4-31 lists the degradation scenarios for the BWR valves that NUREG/CR-6923 characterized as highly likely.

Table 4-31: Highly likely degradation scenarios for BWR valves

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
RCIC	Carbon steel SA216 valves	Suppression pool water, <100°F	Crevice corrosion
Condensate storage tank	Carbon steel SA105, 106, 216, 234 valves	Condensate storage water, <100°F	Crevice corrosion

The references for the inspection programs which may apply to valves are listed in Table 4-32, along with the appendices where the inspection methods are summarized. Valves may also be tested as part of the requirements in the ASME Code for the Operation and Maintenance of Nuclear Power Plants, particularly Subsection ITSC, "Inservice Testing of Valves in Light-Water Reactor Nuclear Power Plants."

Table 4-32: Inspection programs for BWR valves

REFERENCE	APPENDIX
ASME Code, Section XI, Category C-H, "All Pressure Retaining Components"	N
ASME Code, Section XI, Category D-B, "All Pressure Retaining Components"	O

The BWR valves may also be in the scope of the following GALL Report AMP:

- XI.M1 – ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

4.17 Control Rod Blade

Table 4-33 lists the degradation scenario for the control rod blade that NUREG/CR-6923 characterized as highly likely.

4-33: Highly likely degradation scenario for the control rod blade

SYSTEM/ COMPONENT	MATERIAL	ENVIRONMENT	DEGRADATION MECHANISM
Control rod blade	304/316SS (4-6 dpa)	Reactor water, 525°F	SCC

Section 4.2 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," states that surveillance of control rods containing B_4C should be performed to preclude reactivity loss. A proposed design life and surveillance program are aspects of the safety review, specific to each design.

5 Summary

This report summarized the in-service inspection and monitoring programs for degradation scenarios identified as highly-likely in NUREG/CR-6923. A total of 162 degradation scenarios were identified as highly-likely, 65 for PWR components, and 97 for BWR components. Nearly all of the components for BWRs and PWRs are subject to inspection under Section XI of the ASME Boiler and Pressure Vessel Code or NRC-approved code cases. For many components, NRC has also endorsed alternative inspection programs. For PWRs, these include EPRI Materials Reliability Program and Nuclear Energy Institute Steam Generator Program guidelines, among others. For BWRs, these are largely the EPRI Boiling Water Reactor Vessel and Internals Program guidelines. Stress corrosion cracking of the steam generator divider plate in PWRs is the only component for which an inspection and monitoring program was not identified. Cracking in this component has been reported in French plants but inspections have not yet been performed in U.S. plants. The operating experience to date indicates that the cracking in the French plants is minor and does not appear to be growing through the divider plate quickly, if at all. In addition, EPRI Report 1016552, "Divider Plate Cracking in Steam Generators: Results of Phase II: Evaluation of the Impact of a Cracked Divider Plate on LOCA and Non-LOCA Analyses (2008), concluded that extensive cracking of the divider plate is not a safety concern. The staff has been requesting further information on this issue for applicants seeking plant license renewal, given the potential for cracks in the divider plate to grow into the pressure boundary. The industry is currently performing analyses to determine whether this concern is warranted.

APPENDICES

Appendix A – Inspection Program Summary
ASME Boiler and Pressure Vessel Code, Section XI, Category B-A, "Pressure Retaining Welds in Reactor Vessel"

SYSTEM/COMPONENT	CATEGORIES	METHODS
Reactor vessel	Shell welds	Volumetric examination
	Head welds	
	Shell-to-flange weld	
	Repair welds, beltline	
	Head-to-flange weld	Volumetric and surface examination

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Subsection XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," ASME International, New York, NY (2008).

Sections Referenced:

- PWR Reactor Pressure Vessel and Control Rod Drive Mechanism Housing
- BWR Reactor Vessel Flange

Appendix B – Inspection Program Summary
ASME Boiler and Pressure Vessel Code, Section XI, Category B-D, "Full Penetration Welded Nozzles in Vessels"

SYSTEM/COMPONENT	CATEGORIES	METHODS
• Reactor vessel • Pressurizer • Steam generators (primary side) • Heat exchangers (primary side)	Nozzle to vessel welds and nozzle inside radius	Volumetric examination

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Subsection XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," ASME International, New York, NY (2008).

Sections Referenced:

- BWR Penetrations and Attachments
- BWR Differential Pressure/Standby Liquid Control Penetration

Appendix C – Inspection Program Summary
ASME Boiler and Pressure Vessel Code, Section XI, Category B-F, “Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles”

SYSTEM/COMPONENT	CATEGORIES	METHODS
• Reactor vessel • Pressurizer • Steam generator • Heat exchangers	Large nozzle to safe end butt welds	Volumetric and surface examination
	Small nozzle to safe end butt welds and socket welds	Surface examination

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Subsection XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” ASME International, New York, NY (2008).

Sections Referenced:

- PWR Alloy 600/82/182 Nozzles and Dissimilar Metal Welds
- BWR Feedwater Nozzle
- BWR Differential Pressure/Standby Liquid Control Penetration
- BWR Penetrations and Attachments

Appendix D – Inspection Program Summary
ASME Boiler and Pressure Vessel Code, Section XI, Category B-G-1, "Pressure Retaining Bolting Greater than 2 in. in Diameter"

SYSTEM/COMPONENT	CATEGORY/ DESCRIPTION	METHODS
• Pressurizer • Steam generator • Heat exchangers • Piping • Pumps • Valves	Bolts and studs	Volumetric examination
	Flange surfaces, nuts, bushings, and washers	Visual examination

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Subsection XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," ASME International, New York, NY (2008).

Section Referenced:

- Reactor Coolant Pump Components

Appendix E – Inspection Program Summary
ASME Boiler and Pressure Vessel Code, Section XI, Category B-G-2, “Pressure Retaining Bolting 2 in. and Less in Diameter”

SYSTEM/COMPONENT	CATEGORY/DESCRIPTION	METHODS
• Reactor vessel • Pressurizer • Steam generator • Heat exchangers • Piping • Pumps • Valves	Bolts, studs, and nuts	Visual examination

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Subsection XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” ASME International, New York, NY (2008).

Section Referenced:

- Reactor Coolant Pump Components

Appendix F – Inspection Program Summary
ASME Boiler and Pressure Vessel Code, Section XI, Category B-J, "Pressure Retaining Welds in Piping"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Piping welds	Large circumferential welds and large branch pipe connection welds	Surface and volumetric examination
	Circumferential welds of PWR HPSI systems	Volumetric examination
	Small circumferential welds, small branch pipe connection welds, and socket welds	Surface examination

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Subsection XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," ASME International, New York, NY (2008).

Sections Referenced:

- PWR Socket Welds
- PWR Piping Dissimilar Metal Welds
- BWR Differential Pressure/Standby Liquid Control Penetration
- BWR Austenitic SS Piping
- BWR Bottom Head Penetrations

Appendix G – Inspection Program Summary
ASME Boiler and Pressure Vessel Code, Section XI, Category B-N-2, "Welded Core Support Structures and Internal Attachments to Reactor Vessels" and Category B-N-3, "Removable Core Support Structures"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Reactor vessel	Vessel interior	Visual examination
	Interior attachments	Visual examination, level depending on whether inside or outside beltline
	Core support structures	Visual examination

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Subsection XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," ASME International, New York, NY (2008).

Sections Referenced:

- PWR Internals
- BWR Jet Pump Components
- BWR Core Shroud
- BWR Core Plate
- BWR Top Guide
- BWR Steam Separator and Dryer

Appendix H – Inspection Program Summary
ASME Boiler and Pressure Vessel Code, Section XI, Category B-O, "Pressure Retaining Welds in Control Rod Drive and Instrument Nozzle Housings"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Reactor vessel	Welds in CRD and ICI nozzle housings	Volumetric or surface examination

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Subsection XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," ASME International, New York, NY (2008).

Sections Referenced:

- PWR Reactor Pressure Vessel and Control Rod Drive Mechanism Housing
- BWR Bottom Head Penetrations

Appendix I – Inspection Program Summary
ASME Boiler and Pressure Vessel Code, Section XI, Category B-P, "All Pressure Retaining Components"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Pressure retaining components	Class I pressure boundary	System leakage test

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Subsection XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," ASME International, New York, NY (2008).

Sections Referenced:

- PWR Alloy 600/82/182 Nozzles and Dissimilar Metal Welds
- PWR Socket Welds
- PWR Reactor Pressure Vessel and Control Rod Drive Mechanism Housing
- PWR Piping Dissimilar Metal Welds
- CE Pressurizer Heater Sleeves
- BWR Reactor Vessel Flange
- BWR Feedwater Nozzle
- BWR Differential Pressure/Standby Liquid Control Penetration
- BWR Austenitic SS Piping
- BWR Bottom Head Penetrations
- BWR Penetrations and Attachments

Appendix J – Inspection Program Summary
ASME Boiler and Pressure Vessel Code, Section XI, Category C-A, "Pressure Retaining Welds in Pressure Vessels"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Pressure vessels	Shell and head circumferential welds, tubesheet-to-shell weld	Volumetric examination

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Subsection XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," ASME International, New York, NY (2008).

Section Referenced:

- Service Water System

Appendix K – Inspection Program Summary
ASME Boiler and Pressure Vessel Code, Section XI, Category C-D, "Pressure Retaining Bolting Greater than 2 in. in Diameter"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
• Pressure vessels • Piping • Pumps • Valves	Bolts and studs	Volumetric examination

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Subsection XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," ASME International, New York, NY (2008).

Sections Referenced:

- Reactor Coolant Pump Components
- BWR Bolting

Appendix L – Inspection Program Summary
ASME Boiler and Pressure Vessel Code, Section XI, Category C-F-1, “Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping”

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Piping welds	Large circumferential welds	Surface and volumetric examination
	Socket welds and pipe branch connection welds	Surface examination

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Subsection XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” ASME International, New York, NY (2008).

Sections Referenced:

- PWR Socket Welds
- PWR Piping Dissimilar Metal Welds
- BWR Core Spray Piping
- BWR Austenitic SS Piping

Appendix M – Inspection Program Summary
ASME Boiler and Pressure Vessel Code, Section XI, Category C-F-2, "Pressure Retaining Welds in Carbon or Low Alloy Steel Piping"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Piping welds	Large circumferential welds	Surface and volumetric examination
	Socket welds and pipe branch connection welds	Surface examination

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Subsection XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," ASME International, New York, NY (2008).

Sections Referenced:

- Service Water System
- PWR Carbon Steel Components
- BWR Carbon Steel Components

Appendix N – Inspection Program Summary
ASME Boiler and Pressure Vessel Code, Section XI, Category C-H, “All Pressure Retaining Components”

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Pressure retaining components	Portions of the system required to operate or support the safety function	System leakage test

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Subsection XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” ASME International, New York, NY (2008).

Sections Referenced:

- PWR Socket Welds
- Service Water System
- PWR Piping Dissimilar Metal Welds
- PWR Carbon Steel Components
- BWR Core Spray Piping
- BWR Austenitic SS Piping
- BWR Carbon Steel Components
- BWR Bolting
- BWR Heat Exchanger Components

Appendix O – Inspection Program Summary
ASME Boiler and Pressure Vessel Code, Section XI, Category D-B, "All Pressure Retaining Components"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Pressure retaining components	Portions of the system required to operate or support the safety-related function up	System leakage test

American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Subsection XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," ASME International, New York, NY (2008).

Sections Referenced:

- Service Water System
- BWR Carbon Steel Components

Appendix P – Inspection Program Summary
ASME Boiler and Pressure Vessel Code Case N-722, "Additional Examination Requirements for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
• Reactor vessel • Steam generators • Pressurizer • Piping	Partial or full penetration welds in Class 1 components fabricated with Alloy 600/82/182 material.	Visual examination

American Society of Mechanical Engineers (ASME), Code Case N-722, "Additional Examination Requirements for PWR Pressure Retaining Welds in Class 1 Components Fabricated with Alloy 600/82/182 Materials," ASME International, New York, NY (2009).

Sections Referenced:

- PWR Alloy 600/82/182 Nozzles and Dissimilar Metal Welds
- PWR Piping Dissimilar Metal Welds

Appendix Q – Inspection Program Summary
ASME Boiler and Pressure Vessel Code Case N-729, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
RPV head	Dissimilar metal welds	Visual examination
Nozzles and partial penetration welds	Dissimilar metal welds	Volumetric and surface examination

American Society of Mechanical Engineers (ASME), Code Case N-729, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds," ASME International, New York, NY (2006).

Section Referenced:

- PWR Reactor Pressure Vessel and Control Rod Drive Mechanism Housing

Appendix R – Inspection Program Summary
ASME Boiler and Pressure Vessel Code Case N-770, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Piping and vessel nozzle butt welds	Unmitigated hot leg and cold leg butt welds	Visual and volumetric examination
	Mitigated hot leg and cold leg butt welds	Volumetric and possible surface examination, depending on mitigation type

American Society of Mechanical Engineers (ASME), Code Case N-770, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities," ASME International, New York, NY (2009).

Section Referenced:

- PWR Alloy 600/82/182 Nozzles and Dissimilar Metal Welds

Appendix S – Inspection Program Summary
BWRVIP-18: “BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines”

SYSTEM/COMPONENT	CATEGORY/DESCRIPTION	METHODS
Core spray system	Piping and sparger welds	Visual and/or volumetric examination

Electric Power Research Institute (EPRI), BWRVIP-18: BWR Vessel and Internals Project, “BWR Core Spray Internals Inspection and Flaw Evaluation Guidelines,” EPRI, Palo Alto, CA (2005), Report Number 1011469.

Section Referenced:

- BWR Core Spray Piping

Appendix T – Inspection Program Summary
BWRVIP-25, “BWR Core Plate Inspection and Flaw Evaluation Guidelines”

SYSTEM/COMPONENT	CATEGORY/DESCRIPTION	METHODS
Core plate	Rim hold down bolts	Visual or volumetric examination

Electric Power Research Institute (EPRI), BWRVIP-25: BWR Vessel and Internals Project, “BWR Core Plate Inspection and Flaw Evaluation Guidelines,” EPRI, Palo Alto, CA (1996), Report Number 107284.

Section Referenced:

- BWR Core Plate

Appendix U – Inspection Program Summary
BWRVIP-26, "BWR Top Guide Inspection and Flaw Evaluation Guidelines"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Top guide	Aligner pins and sockets	Visual examination
	Hold-down assemblies	
	C-clamps	
	Studs	
	Rim pins and rim welds	

Electric Power Research Institute (EPRI), BWRVIP-26: BWR Vessel and Internals Project, "BWR Top Guide Inspection and Flaw Evaluation Guidelines," EPRI, Palo Alto, CA (2004), Report Number 1009946.

Section Referenced:
- BWR Top Guide

Appendix V – Inspection Program Summary
BWRVIP-27, "BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Core plate ΔP penetration	Alloy 600 partial penetration welds	System leakage test
	Low alloy steel nozzle to vessel welds	Volumetric examination
Nozzle to safe end weld	Dissimilar metal welds and carbon or low alloy steel safe end	Volumetric and surface examination
	Stainless steel safe end	System leakage test
Safe end extension	Dissimilar metal welds	Volumetric and surface examination
	Bored safe end extension	Volumetric examination
	Inaccessible safe end extension and weld	System leakage test

Electric Power Research Institute (EPRI), BWRVIP-27: BWR Vessel and Internals Project, "BWR Standby Liquid Control System/Core Plate ΔP Inspection and Flaw Evaluation Guidelines," EPRI, Palo Alto, CA (2003), Report Number 1007279.

Section Referenced:
- BWR Differential Pressure/Standby Liquid Control Penetration

Appendix W – Inspection Program Summary
BWRVIP-38, "BWR Shroud Support Inspection and Flaw Evaluation Guidelines"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Core shroud supports	Alloy 82/182 gusset welds and support plate welds	Volumetric or visual examination

Electric Power Research Institute (EPRI), BWRVIP-38: BWR Vessel and Internals Project, "BWR Shroud Support Inspection and Flaw Evaluation Guidelines," EPRI, Palo Alto, CA (1997), Report Number 108823.

Section Referenced:

- BWR Core Shroud

Appendix X – Inspection Program Summary
BWRVIP-41, “BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines”

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Jet pump assembly	Various parts including riser brace, thermal sleeve, riser pipe, inlet, mixer, restrainer bracket assembly, diffuser, tailpipe, and adapter/lower ring	Visual examination
	Beam bolt assembly	Volumetric examination

Electric Power Research Institute (EPRI), BWRVIP-41: BWR Vessel and Internals Project, “BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines,” EPRI, Palo Alto, CA (2005), Report Number 1012137.

Sections Referenced:

- BWR Jet Pump Components

Appendix Y – Inspection Program Summary
BWRVIP-47, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Guide tube	CRD guide tube sleeve to alignment lug weld	Visual examination
	CRD guide tube body to sleeve weld	
	CRD tube base to body weld	
	Guide tube and fuel support alignment pin to core plate weld and pin	

Electric Power Research Institute (EPRI), BWRVIP-47: BWR Vessel and Internals Project, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines," EPRI, Palo Alto, CA (2004), Report Number 1009947.

Section Referenced:

- BWR Bottom Head Penetrations

Appendix Z – Inspection Program Summary
BWRVIP-48, "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines"

SYSTEM/COMPONENT	CATEGORY/ DESCRIPTION	METHODS
• Jet pump riser brace • Core spray piping • Steam dryer support and feedwater bracket	Attachment welds	Visual examination

Electric Power Research Institute (EPRI), BWRVIP-47: BWR Vessel and Internals Project, "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines," EPRI, Palo Alto, CA (2004), Report Number 1009948.

Section Referenced:

- BWR Jet Pump Components
- BWR Steam Separator and Dryer
- BWR Penetrations and Attachments

Appendix AA – Inspection Program Summary
BWRVIP-49, "Instrument Penetration Inspection and Flaw Evaluation Guidelines"

SYSTEM/ COMPONENT	CATEGORY/ DESCRIPTION	METHODS
Instrument penetrations	Alloy 600	System leakage test
Nozzles	Low alloy steel	Volumetric examination
Instrument nozzle to safe end weld or penetration to extension weld	Dissimilar metal weld	Volumetric and/or surface examination

Electric Power Research Institute (EPRI), BWRVIP-49: BWR Vessel and Internals Project, "Instrument Penetration Inspection and Flaw Evaluation Guidelines," EPRI, Palo Alto, CA (1998), Report Number 108695.

Section Referenced:

- BWR Penetrations and Attachments

Appendix AB – Inspection Program Summary
BWRVIP-75, “Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedule”

SYSTEM/COMPONENT	CATEGORY/DESCRIPTION	METHODS
Austenitic materials	Piping systems	Volumetric examination

Electric Power Research Institute (EPRI), BWRVIP-75: BWR Vessel and Internals Project, “Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedule,” EPRI, Palo Alto, CA (2008), Report Number 1012621.

Section Referenced:

- BWR Austenitic SS Piping

Appendix AC – Inspection Program Summary
BWRVIP-76, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Core shroud	Horizontal and vertical shroud welds	Volumetric, surface, or visual examination
	Radial ring welds	Plant specific

Electric Power Research Institute (EPRI), BWRVIP-76: BWR Vessel and Internals Project, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines," EPRI, Palo Alto, CA (2009), Report Number 1019057.

Section Referenced:

- BWR Core Shroud

Appendix AD – Inspection Program Summary
BWRVIP-139, "Steam Dryer Inspection and Flaw Evaluation Guidelines"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Steam dryer	Various parts such as outer bank hoods, hood end plates, cover plate, tie bars, drain channels, lifting rods/attachment, guide channels, guide rod followers, and upper ring support	Visual examination

Electric Power Research Institute (EPRI), BWRVIP-139: BWR Vessel and Internals Project, "Steam Dryer Inspection and Flaw Evaluation Guidelines," EPRI, Palo Alto, CA (2005), Report Number 1011463.

Section Referenced:

- BWR Steam Separator and Dryer

Appendix AE – Inspection Program Summary
BWRVIP-183, “Top Guide Grid Beam Inspection and Flaw Evaluation Guidelines”

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Top guide grid beams	Grid beam cells containing CRD blades	Volumetric or visual examination
	Rim areas containing weld and HAZ	

Electric Power Research Institute (EPRI), BWRVIP-183: BWR Vessel and Internals Project, “Top Guide Grid Beam Inspection and Flaw Evaluation Guidelines,” EPRI, Palo Alto, CA (2007), Report Number 1013401.

Section Referenced:

- BWR Top Guide

Appendix AF – Inspection Program Summary
GE-NE-523-A71-0594-A, “Alternate BWR Feedwater Nozzle Inspection Requirements”

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Feedwater nozzle	Nozzle	Volumetric and visual examination
	Sparger	Visual examination

GE Nuclear Energy, GE-NE-523-A71-0594-A, “Alternate BWR Feedwater Nozzle Inspection Requirements,” General Electric Company, San Jose, CA (2000).

Section Referenced:

- BWR Feedwater Nozzle

Appendix AG – Inspection Program Summary
MRP-146, "Materials Reliability Program: Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines"

SYSTEM/ COMPONENT	CATEGORY/ DESCRIPTION	METHODS
• Safety injection lines • Charging lines not in service • Drain lines • Excess letdown lines • RHR suction lines • Other normally stagnant lines	Piping diameters greater than one inch	Visual or volumetric examination

Electric Power Research Institute (EPRI), Materials Reliability Program, "Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines (MRP-146)," EPRI, Palo Alto, CA (2005), Report Number 1011955.

Section Referenced:

- PWR Socket Welds

Appendix AH – Inspection Program Summary MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Core internals	Various internal components such as core support shield assembly, core barrel assembly, lower grid assembly, upper grid assembly, IMI guide tube assembly, core shroud assembly, lower support structure, control rod guide tube assembly, baffle-former assembly, thermal shield assembly, control element assembly	Visual examination
	Core barrel bolts, baffle-former bolts, core shroud bolts, thermal shield bolts	Volumetric examination

Electric Power Research Institute (EPRI), Materials Reliability Program, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227)," EPRI, Palo Alto, CA (2008), Report Number 1016596.

Section Referenced:

- PWR Internals

Appendix AI – Inspection Program Summary
NEI 97-06, "Steam Generator Program Guidelines"

SYSTEM/ COMPONENT	REQUIREMENT
Steam generator	Follow the inspection guidelines contained in EPRI *PWR Steam Generator Examination Guidelines*.
	Use EPRI *Steam Generator Integrity Assessment Guidelines* to determine the evaluation methods, margins, and uncertainty considerations used to evaluate tube integrity.

Nuclear Energy Institute (NEI), NEI 97-06, Revision 2, "Steam Generator Program Guidelines," NEI, Washington, DC (2005).

Section Referenced:

- Steam Generator Tubes

Appendix AJ – Inspection Program Summary NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Austenitic materials	Piping larger than 4 inches in diameter, as well as reactor vessel attachments and appurtenances	Volumetric examination

U.S. Nuclear Regulatory Commission (NRC), "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping (Generic Letter 88-01)," NRC, Washington, DC (1988), ADAMS ML031130463.

Sections Referenced:

- BWR Austenitic SS Piping
- BWR Bottom Head Penetrations

Appendix AK – Inspection Program Summary
NRC Generic Letter 88-05 "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants"

SYSTEM/ COMPONENT	REQUIREMENT
Reactor coolant pressure boundary components	Plant specific program, see document page 2: "…provide assurances that a program has been implemented consisting of systematic measures to ensure that boric acid corrosion does not lead to degradation of the assurance that the reactor coolant pressure boundary will have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture."

U.S. Nuclear Regulatory Commission (NRC), "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants (Generic Letter 88-05)," NRC, Washington, DC (1988), ADAMS ML031130424.

Section Referenced:

- PWR Reactor Pressure Vessel and Control Rod Drive Mechanism Housing

Appendix AL – Inspection Program Summary
NRC Generic Letter 89-08 “Erosion/Corrosion Induced Pipe Wall Thinning”

SYSTEM/ COMPONENT	REQUIREMENT
Carbon steel piping systems	Plant specific program, see page 3 of GL 89-08: “…implement a long term erosion/corrosion monitoring program that provides assurances that procedures or administrative controls are in place to assure that the NUMARC program [NUREG-1344, Appendix A] or another equally effective program is implemented and the structural integrity of all high- energy (two phase as well as single phase) carbon steel systems is maintained.”

U.S. Nuclear Regulatory Commission (NRC), “Erosion/Corrosion Induced Pipe Wall Thinning (Generic Letter 89-08),” NRC, Washington, DC (1989), ADAMS ML031470660.

Section Referenced:

- PWR Carbon Steel Components

Appendix AM – Inspection Program Summary
NRC Generic Letter 89-13 "Service Water System Problems Affecting Safety-Related Equipment"

SYSTEM/ COMPONENT	REQUIREMENT
Service water system	Plant specific program. See recommended action III: "Ensure by establishing a routine inspection and maintenance program for open-cycle service water system piping and components that corrosion, erosion, protective coating failure, silting, and biofouling cannot degrade the performance of the safety-related systems supplied by service water."

U.S. Nuclear Regulatory Commission (NRC), "Service Water System Problems Affecting Safety-Related Equipment (Generic Letter 89-13)," NRC, Washington, DC (1989), ADAMS ML031150348.

Section Referenced:

- Service Water System

Appendix AN – Inspection Program Summary NSAC-202L, "Recommendations for an Effective Flow-Accelerated Corrosion Program"

SYSTEM/COMPONENT	METHODS
Piping systems made of carbon or low-alloy steel	Plant specific program. See Section 4 of NSAC-202L for recommended FAC tasks.

Electric Power Research Institute (EPRI), "Recommendations for an Effective Flow-Accelerated Corrosion Program (NSAC-202L)," EPRI, Palo Alto, CA (2006), Report Number 1011838.

Section Referenced:

- PWR Carbon Steel Components

Appendix AO – Inspection Program Summary NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking: Resolution of Generic Technical Activity A-10"

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Feedwater nozzle	Nozzle	Volumetric and surface examination
	Sparger	Visual examination

U.S. Nuclear Regulatory Commission (NRC), NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking: Resolution of Generic Technical Activity A-10," NRC, Washington, DC (1980), ADAMS ML031600712.

Section Referenced:

- BWR Feedwater Nozzle

Appendix AP – Inspection Program Summary
Steam Generator Management Program, “Pressurized Water Reactor Steam Generator Examination Guidelines”

SYSTEM/ COMPONENT	CATEGORY/DESCRIPTION	METHODS
Steam generator tubes, primary side	Alloy 600 tubing	Volumetric examination

Electric Power Research Institute (EPRI), “Steam Generator Management Program: Pressurized Water Reactor Steam Generator Examination Guidelines: Revision 7,” EPRI, Palo Alto, CA (2007), Report Number 1013706.

Section Referenced:

- Steam Generator Tubes

Appendix AQ – Inspection Program Summary, “Steam Generator Integrity Assessment Guidelines”

SYSTEM/COMPONENT	METHODS
Steam generator tubes, secondary side	Visual examination

Electric Power Research Institute (EPRI), “Steam Generator Integrity Assessment Guidelines: Revision 2,” EPRI, Palo Alto, CA (2006), Report Number 1012987

Section Referenced:

- Steam Generator Tubes

Appendix AR – Inspection Program Summary
WCAP-15988, “Generic Guidance for an Effective Boric Acid Inspection Program for Pressurized Water Reactors”

SYSTEM/ COMPONENT	REQUIREMENT
Carbon steel components	Development of plant specific program

Westinghouse Electric Company, WCAP-15988, “Generic Guidance for an Effective Boric Acid Inspection Program for Pressurized Water Reactors,” Westinghouse Electric Company, Pittsburgh, PA (2005).

Section Referenced:

- PWR Reactor Pressure Vessel and Control Rod Drive Mechanism Housing

Appendix AS – Inspection Program Summary
WCAP-16180, “WOG CE Fleet Pressurizer Heater Sleeve Inspection Program”

SYSTEM/ COMPONENT	CATEGORY/ DESCRIPTION	METHODS
Pressurizer heater sleeve (CE)	Alloy 600	Visual examination

Westinghouse Owners Group, WCAP-16180, “WOG CE Fleet Pressurizer Heater Sleeve Inspection Program,” (2004).

Section Referenced:

- CE Pressurizer Heater Sleeves