LICENSE RENEWAL APPLICATION

DRESDEN NUCLEAR POWER STATION
DOCKET Nos. 50-237 AND 50-249
Facility Operating License Nos.
DPR-19 and DPR-25

QUAD CITIES NUCLEAR POWER STATION
DOCKET Nos. 50-254 AND 50-265
Facility Operating License Nos.
DPR-29 and DPR-30

January 2003

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CHAPTER 1
ADMINISTRATIVE INFORMATION

1.0 ADMINISTRATIVE INFORMATION

1.1 GENERAL INFORMATION - 10 CFR 54.19

1.1.1 Names of Applicant and Co-Owners

Exelon Generation Company (EGC), LLC, (formerly Commonwealth Edison Company) hereby applies for renewed operating licenses for Dresden Nuclear Power Station (DNPS), Units 2 and 3 and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. EGC submits this application individually for DNPS, Units 2 and 3 and as agent for the co-owner licensees named on the operating license for QCNPS, Units 1 and 2. The co-owner licensee for QCNPS, Units 1 and 2 is:

- MidAmerican Energy Company

1.1.2 Addresses of Applicant and Co-Owners

- Exelon Generation Company, LLC
  4300 Winfield Road
  Warrenville, Illinois 60555

- MidAmerican Energy Company
  666 Grand Avenue, Suite 2900 , P.O. Box 657
  Des Moines, Iowa 50303

1.1.3 Descriptions of Business or Occupation of Applicant and Co-Owners

Exelon Generation Company (EGC), LLC is a limited liability company formed to own, operate, and acquire nuclear and other electric generating stations; to engage in the sale of electrical energy; and to perform other business activities. EGC is a wholly-owned corporate subsidiary of Exelon Ventures Company, which in turn is wholly owned by Exelon Corporation, a corporation formed under the laws of the Commonwealth of Pennsylvania. EGC is the exclusive licensed operator of DNPS, Units 2 and 3, and QCNPS, Units 1 and 2 which are the subject of this application. The current DNPS Unit 2 License (Facility Operating License No. DPR-19) expires at midnight on December 22, 2009, and the Unit 3 License (Facility Operating License No. DPR-25) expires at midnight on January 12, 2011. The current QCNPS Unit 1 License (Facility Operating License No. DPR-29) expires at midnight on December 14, 2012. The current QCNPS Unit 2 License (Facility Operating License No. DPR-30) expires at midnight on December 14, 2012. EGC will be named as the exclusive licensed operator on the renewed operating licenses.

MidAmerican Energy Company is engaged in the generation and transmission of electricity and the distribution and sale of such electricity within the States of Iowa, Illinois, and South Dakota. MidAmerican provides electric service to 673,000 customers and natural gas service to 653,000 customers in Iowa, Illinois, Nebraska, and South

Dakota. MidAmerican has rated capability in excess of 4400 MW and currently provides retail electric service in Iowa, Illinois, and South Dakota. MidAmerican Energy Company is a co-owner and licensee of Quad Cities Nuclear Power Station Units 1 and 2.

1.1.4 Descriptions of Organization and Management of Applicant and Co-Owners

Exelon Generation Company, LLC

Exelon Generation Company (EGC), LLC is organized under the laws of the Commonwealth of Pennsylvania. EGC's principal place of business will be in the Commonwealth of Pennsylvania. Exelon Ventures Company and Exelon Corporation are corporations organized under the laws of the Commonwealth of Pennsylvania with their headquarters and principal places of business in Chicago. Exelon Corporation is a publicly-traded corporation whose shares are widely traded on the New York Stock Exchange. Exelon Ventures Company is a wholly-owned subsidiary of Exelon Corporation. All of the directors, management committee members, and principal officers of Exelon Generation Company, LLC, Exelon Ventures Company, and Exelon Corporation are U.S. citizens. Neither Exelon Generation Company, LLC nor its parents, Exelon Ventures Company or Exelon Corporation, are owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. The principal officers of EGC and their addresses are presented below:

Principal Officers (Exelon Generation Company, LLC)		
Name	**Title**	**Address**
Oliver D. Kingsley, Jr.	Chief Executive Officer and President, Exelon Generation	4300 Winfield Road Warrenville, Il 60555
Jack Skolds	President and Chief Nuclear Officer, Exelon Nuclear	4300 Winfield Road Warrenville, IL 60555
Ian P. McLean	President, Exelon Power Team	300 Exelon Way Kennett Square, PA 19348
William H. Bohlke	Sr VP, Nuclear Services, Exelon Nuclear	4300 Winfield Road Warrenville, IL 60555
Christopher M. Crane	Sr VP, MidWest Regional Operating Group, Exelon Nuclear	4300 Winfield Road Warrenville, IL 60555
Charles G. Pardee	Sr VP, MidAtlantic Regional Operating Group, Exelon Nuclear	200 Exelon Way Kennett Square, PA 19348
Preston D. Swafford	Sr VP, Exelon Generation	200 Exelon Way Kennett Square, PA 19348
David W. Woods	Sr VP, Communications, Govermental & Public Affairs, Exelon Generation	300 Exelon Way Kennett Square, PA 19348
James S. Jablonski	VP, Financial Trading, Exelon Power Team	300 Exelon Way Kennett Square, PA 19348
Edward Fedorchak	VP, Fuels, Exelon Power Team	300 Exelon Way Kennett Square, PA 19348
Carol Anderson	VP, Information Technology, Exelon Nuclear	300 Exelon Way Kennett Square, PA 19348

Principal Officers (Exelon Generation Company, LLC)		
Name	**Title**	**Address**
Jeffrey A. Benjamin	VP, Licensing and Regulatory Affairs, Exelon Nuclear	4300 Winfield Road Warrenville, IL 60555
Kenneth W. Cornew	VP, Long-term Transactions, Exelon Power Team	200 Exelon Way Kennett Square, PA 19348
Edward J. Cullen, Jr.	VP, Legal and Secretary	300 Exelon Way Kennett Square, PA 19348
Michael Erdlen	VP, Information Technology, Exelon Power Team	300 Exelon Way Kennett Square, PA 19348
Jan H. Freeman	VP, Public Affairs, Exelon Generation	300 Exelon Way Kennett Square, PA 19348
Martin Coveney	VP, Finance	4300 Winfield Road Warrenville, IL 60555
Joel Dimmette	VP, Nuclear Oversight, Exelon Nuclear	4300 Winfield Road Warrenville, IL 60555
Dorothy M. Hawkins	VP, Business Operations, Exelon Nuclear	4300 Winfield Road Warrenville, IL 60555
Susan O. Ivey	VP, Short-term Transactions, Exelon Power Team	300 Exelon Way Kennett Square, PA 19348
Theodore E. Jennings	VP, Engineering and Operation Support , Exelon Generation	4300 Winfield Road Warrenville, IL 60555
Richard J. Landy	VP, Human Resources & Administration, Exelon Generation	4300 Winfield Road Warrenville, IL 60555
Rod Krich	VP, Licensing Projects	4300 Winfield Road Warrenville, IL 60555
Charles P. Lewis	VP, Strategy & Development, Exelon Generation	300 Exelon Way Kennett Square, PA 19348
James P. Malone	VP, Fuels Management, Exelon Nuclear	4300 Winfield Road Warrenville, IL 60555
James R. Meister	VP, Nuclear Engineering, Exelon Nuclear	4300 Winfield Road Warrenville, IL 60555
Michael Metzner	VP, Finance/Analytics/Risk, Exelon Power Team	300 Exelon Way Kennett Square, PA 19348
Marilyn C. Kray	VP, Special Projects	200 Exelon Way Kennett Square, PA 19348
John L. Settelen, Jr.	VP & Controller, Exelon Generation	300 Exelon Way Kennett Square, PA 19348
Ronald DeGregorio	Site VP – Oyster Creek	Oyster Creek Nuclear Power Station Route 9 South P.O. Box 388 Forked River, NJ 08731
Rusty West	Site VP – Peach Bottom Atomic Power Station	Peach Bottom Atomic Power Station 1848 Lay Road Delta, PA 17314

Principal Officers (Exelon Generation Company, LLC)		
Name	**Title**	**Address**
William Levis	Site VP – Limerick Generating Station	Limerick Generating Station Evergreen & Sanatoga Roads Pottstown, PA 19464
Richard Lopriore	Site VP – Byron Station	Byron Nuclear Station 4450 North German Church Road Byron, IL 61010
George P. Barnes, Jr.	Site VP – LaSalle County Station	LaSalle County Station 2601 North 21st Road Marseilles, IL 61341
Robert Hovey	Site VP – Dresden Nuclear Power Station	6500 North Dresden Road Morris, IL 60450
Timothy J. Tulon	Site VP – Quad Cities Nuclear Power Station	22710 206th Avenue North Cordova, IL 61242
Michael J. Pacilio	Site VP – Clinton Nuclear Power Station	Clinton Power Station P.O. Box 678 Clinton, Il 61726
James von Suskil	Site VP – Braidwood Generating Station	Braidwood Generating Station Rural Route 1, Box 84 Braceville, IL 60407
Bruce C. Williams	Site VP – Three Mile Island	Three Mile Island Route 441 South P.O. Box 480 Middletown, PA 17057
Robert K. McDonald	VP, Generation	Bank One Building 10 South Dearborn Street Chicago, IL 60603
Edward F. Sproat, III	VP, International Projects	200 Exelon Way Kennett Square, PA 19348
Dave B. Wozniak	VP, Midwest Regional Operating Group Support, Exelon Nuclear	4300 Winfield Road Warrenville, IL 60555
Michael McMahan	VP, Outage Planning, Project Management & Services, Exelon Nuclear	4300 Winfield Road Warrenville, IL 60555
Ron DeGregorio	VP, MidAtlantic Regional Operating Support, Exelon Nuclear	200 Exelon Way Kennett Square, PA 19348
Kevin Yessian	VP Supply, Exelon Nuclear	4300 Winfield Road Warrenville, IL 60555
J. Barry Mitchell	VP and Treasurer	Bank One Building 10 South Dearborn Street Chicago, IL 60603
George R. Shicora	Assistant Treasurer	Main Office Building 2301 Market Street Philadelphia, PA 19101
Charles S. Walls	Assistant Treasurer	Bank One Building 10 South Dearborn Street Chicago, IL 60603

Principal Officers (Exelon Generation Company, LLC)		
Name	**Title**	**Address**
Todd D. Cutler	Assistant Secretary	Main Office Building 2301 Market Street Philadelphia, PA 19101
Scott N. Peters	Assistant Secretary	Bank One Building 10 South Dearborn Street Chicago, IL 60603

MidAmerican Energy Company

MidAmerican Energy Company is incorporated in the state of Iowa. MidAmerican's principal place of business is in the state of Iowa. MidAmerican Energy Company is a wholly owned subsidiary of MHC Inc. MidAmerican Energy Company is headquartered in Des Moines, Iowa. All the directors, management committee members, and principal officers of MidAmerican Energy Company are United States citizens with the exception of Mr. Gregory E. Abel, president, who is a citizen of the country of Canada and a resident of the United States.

Neither MidAmerican Energy Company nor its parent companies are owned, controlled, or dominated by an alien, foreign corporation, or a foreign government. The principal officers of MidAmerican Energy Company, and their addresses are presented below:

Principal Officers (MidAmerican Energy Company)		
Name	**Title**	**Address**
David L. Sokol	Chairman of the Board of Directors	302 South 36th Street, Omaha, NE 68131-3845
Gregory E. Abel	President	666 Grand Avenue, 29th Floor Des Moines, IA 50309
Jack L. Alexander	Senior Vice President	666 Grand Avenue, 29th Floor Des Moines, IA 50309
Patrick J. Goodman	Senior Vice President and Chief Financial Officer	666 Grand Avenue, 29th Floor Des Moines, IA 50309

1.1.5 Class of License, Use of the Facility, and Period of Time for Which the License Is Sought

Exelon Generation Company (EGC), LLC requests renewal of the Class 104 operating licenses for Dresden Nuclear Power Station, Units 2 and 3, (License Nos. DPR-19 and DPR-25) for a period of 20 years beyond the expiration of the current licenses, midnight on December 22, 2009 for Unit 2 and midnight on January 12, 2011 for Unit 3 and Quad Cities Nuclear Power Station, Units 1 and 2, (License Nos. DPR-29 and DPR-30) for a period of 20 years beyond the expiration of the current licenses, midnight on December 14, 2012 for Unit 1 and midnight on December 14, 2012 for Unit 2.

Because the current licensing basis is carried forward with the possible exception of some aging issues, EGC expects the form and content of the licenses to be generally the same as they now exist. EGC, thus, also requires similar extensions of specific licenses under 10 CFR Parts 30, 40, and 70 that are contained in the current operating licenses.

1.1.6 Earliest and Latest Dates for Alterations, If Proposed

No physical plant alterations or modifications have been identified as necessary in order to implement the provisions of the application.

1.1.7 Restricted Data

With regard to the requirements of 10 CFR 54.17(f), this application does not contain any "Restricted Data," as that term is defined in the Atomic Energy Act of 1954, as amended, or other defense information, and it is not expected that any such information will become involved in these licensed activities.

In accordance with the requirements of 10 CFR 54.17(g), the applicants will not permit any individual to have access to, or any facility to possess restricted data or classified national security information until the individual and/or facility has been approved for such access under the provisions of 10 CFR Parts 25 and/or 95.

1.1.8 Regulatory Agencies

EGC recovers its share of the costs incurred from operating Dresden, Units 2 and 3 and Quad Cities, Units 1 and 2, in its own wholesale rates, and recovers the remaining costs from MidAmerican Energy Company in relation to their ownership interests in each of the units. The rates charged and services provided by EGC are subject to regulation by the Federal Energy Regulatory Commission under the Federal Power Act. EGC is also subject to regulation as a public utility company by the Securities and Exchange Commission under the Public Utility Holding Company Act of 1935, as amended.

Securities and Exchange Commission
450 Fifth Street, NW
Washington, DC 20549

Federal Energy Regulatory Commission
888 First St. N.E.
Washington, DC 20426

1.1.9 Local News Publications

News publications in circulation near Dresden Station that are considered appropriate to give reasonable notice of the application are as follows:

Joliet Herald News

300 Caterpillar Drive
Joliet, Illinois 60436
Tel. (815) 729-6046
Fax. (815) 729-6059

The Free Press Advocate/Braidwood Journal/Coal City Courant

111 South Water Street
Wilmington, Illinois 60481
Tel. (815) 4767966
Fax. (815) 476-7002

Morris Daily Herald

1804 North Division
Morris, Illinois 60450
Tel. (815) 942-3221
Fax. (815) 942-0988

News publications in circulation near Quad Cities Station that are considered appropriate to give reasonable notice of the application are as follows:

Quad-City Times

500 East Third Street
Davenport, Iowa 52801
Tel. (563) 383-2318
Fax. (563) 383-2370

Rock Island Argus/Moline Dispatch

1724 4th Avenue
Rock Island, Illinois 61201
Tel. (309) 786-6441
Fax. (309) 786-7639

The Clinton Herald

221 6th Avenue
Clinton, Iowa 52732
Tel. (800) 729-7101
Fax. (563) 242-7147

Fulton Journal

Post Office Box 30
Fulton, Illinois 61252
Tel. (815) 589-2424

Whiteside News Publication

Post Office Box 31
Morrison, Illinois 61270
Tel. (815) 772-7244
Fax. (815) 772-4105

1.1.10 Conforming Changes To Standard Indemnity Agreement

10 CFR 54.19(b) requires that "each application must include conforming changes to the standard indemnity agreement, 10 CFR 140.92, Appendix B, to account for the expiration term of the proposed renewed license." The current indemnity agreement for Dresden and Quad Cities state in Article VII that the agreement shall terminate at the time of expiration of the licenses specified in Item 3 of the Attachment to the agreement. Item 3 of the Attachment to the indemnity agreement, lists license numbers, DPR-19, DPR-25, DPR-29 and DPR-30. Applicant requests that any necessary conforming changes be made to Article VII and Item 3 of the Attachment, and any other sections of the indemnity agreement as appropriate to ensure that the indemnity agreement continues to apply during both the terms of the current licenses and the terms of the renewed licenses. Applicant understands that no changes may be necessary for this purpose if the current license numbers for each of the units are retained.

1.2 GENERAL LICENSE INFORMATION

1.2.1 Application Updates, Renewed Licenses, and Renewal Term Operation

In accordance with 10 CFR 54.21(b), during NRC review of this application, an annual update to the application to reflect any change to the current licensing basis that materially affects the contents of the license renewal application will be provided.

In accordance with 10 CFR 54.37(b), EGC will maintain a summary list in the DNPS, Units 2 and 3 and the QCNPS, Units 1 and 2, Updated Final Safety Analysis Reports (UFSAR) of activities that are required to manage the effects of aging for the systems, structures or components in the scope of license renewal during the period of extended operation and summaries of the time-limited aging analyses evaluations.

1.2.2 Incorporation By Reference

There are no documents incorporated by reference as part of the application. Any document references, either in text or in Section 1.7 are listed for information only.

1.2.3 Contact Information

Any notices, questions, or correspondence in connection with this filing should be directed to:

Mr. Keith R. Jury
Director – Licensing
Midwest Regional Operating Group
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

with copies to:

Mr. Frederick W. Polaski
Manager License Renewal
Exelon Nuclear
200 Exelon Way
Kennett Square, PA 19348

Mr. Thomas O'Neill
Associate General Counsel
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

Mr. Robert Stachniak
D/QC License Renewal Project Engineer
Midwest Regional Operating Group
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

1.3 PURPOSE

This document provides information required by 10 CFR 54 to support the application for renewed licenses for DNPS, Units 2 and 3 and QCNPS, Units 1 and 2. The application contains technical information required by 10 CFR 54.21 and environmental information required by 10 CFR 54.23. The information contained herein is intended to provide the NRC with an adequate basis to make the finding required by 10 CFR 54.29.

1.4 DESCRIPTION OF THE PLANT

The Dresden Nuclear Power Station site consists of approximately 953 acres located in Grundy County, Illinois. The site boundaries generally follow the Illinois River to the north and the Kankakee River to the east. A cooling lake, which was formed by constructing an impervious earth-fill dike, encompasses a storage area of approximately 1275 acres. The lake is connected to the intake and discharge flumes of Units 2 and 3 by two canals (one intake and one discharge); each canal is about 11,000 feet long. Dresden Units 2 and 3 are BWR/3's designed and supplied by GE Nuclear Energy with 251 inch vessels. The primary containment of each unit is of the Mark 1 design that consists of a drywell, a suppression chamber in the shape of a torus and a connecting vent system between the drywell and the suppression chamber. Each unit is authorized to operate at a steady state reactor power level not to exceed 2957 megawatts thermal.

The Quad Cities Nuclear Power Station site consists of approximately 784 acres in Rock Island County, Illinois, on the east bank of the Mississippi River opposite the mouth of the Wapsipinicon River, and about 3 miles north of Cordova, Illinois. The site is about 20 miles northeast of the Quad Cities (Davenport and Bettendorf Iowa; Rock Island, Moline, and East Moline, Illinois). QCNPS Units 1 and 2 are BWR/3's designed and supplied by GE Nuclear Energy with 251 inch reactor vessels. The primary containment of each unit is of the Mark 1 design that consists of a drywell, a suppression chamber in the shape of a torus and a connecting vent system between the drywell and the suppression chamber. Each unit is authorized to operate at a steady state reactor power level not to exceed 2957 megawatts thermal.

The Dresden Nuclear Power Station Unit 1, (Facility Operating License No. DPR-2) shares the site and surrounding area with Units 2 and 3. DNPS Unit 1 is dual cycle, boiling water reactor owned by EGC. EGC has placed DNPS Unit 1 in a safe storage condition (SAFSTOR) until Units 2 and 3 are ready for decommissioning.

EGC is providing information regarding DNPS Unit 1 due to SSCs installed in the Unit 1 side that meet the criteria of 10 CFR 54.4 (a)(1-3) for support of operation of Units 2 and 3. The SSCs are listed in the scoping tables of Section 2 in this application. The aging effects of the Unit 1 SSCs will also be adequately managed so that the intended functions will be maintained consistent with the current licensing basis throughout the period of extended operation.

1.5 APPLICATION STRUCTURE

This license renewal application is structured in accordance with Regulatory Guide 1.188, “Standard Format and Content for Applications to Renew Nuclear Plant Operating Licenses,” and NEI 95-10, "Industry Guideline on Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule,” Revision 3. In addition, Section 3, Aging Management Review Results and Appendix B, Aging management programs and activities are structured to address the guidance provided in NUREG-1800, “Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants,” April 2001. NUREG-1800 references NUREG-1801, “Generic Aging Lessons Learned (GALL) Report.” The GALL report was used to determine the adequacy of existing aging management programs and which existing programs should be augmented for license renewal. The results of the aging management review, using the GALL, has been documented and is illustrated in table format in Section 3 of this application.

As an aid to the reviewer, electronic versions of the application contain marked hypertext which provide links to the referenced sections.

The application is divided into the following major sections:

Section 1 – Administrative Information

This section provides the administrative information required by 10 CFR 54.17 and 10 CFR 54.19. It describes the plants and states the purpose for this application. Included in this section are the names, addresses, business descriptions, and organization and management descriptions of the applicant, as well as other administrative information. This section also provides an overview of the structure of the application, general references, and a listing of acronyms used throughout the application.

Section 2 – Scoping and Screening Methodology for Identifying Structures and Components Subject to Aging Management Review and Implementation Results

This section describes and justifies the methods used in the integrated plant assessment to identify those structures and components subject to an aging management review in accordance with the requirements of 10 CFR 54.21(a)(2). These methods consist of: 1) scoping, which identifies the systems, structures, and components that are within the scope of 10 CFR 54.4(a), and 2) screening under 10 CFR 54.21(a)(1), which identifies those in-scope structures and components that perform their intended function without moving parts or a change in configuration or properties, and that are not subject to replacement based on a qualified life or specified time period.

Additionally, the results for systems and structures are described in this section. Scoping results are presented in Table 2.2-1, "Mechanical Systems Scoping Results," Table 2.2-2, "Structures Scoping Results," and Table 2.2-3, "Electrical and Instrumentation and Control Systems Scoping Results." Screening results are presented in Sections 2.3, 2.4, and 2.5.

The screening results consist of lists of components or component groups that require aging management review. Brief descriptions of mechanical systems and structures within the scope of license renewal are provided as background information. Mechanical system and structure intended functions are provided for in-scope systems and structures. For each in-scope system and structure, components requiring an aging management review are identified, associated component intended functions are identified, and appropriate reference to the Section 3 Table reference providing the aging management review results is made.

Selected structural and electrical component groups, such as component supports and cables, were evaluated as commodities. Under the commodity approach, selected structural and electrical component groups were evaluated based upon common environments and materials. For each of these commodities, the components or component groups requiring aging management are presented in Sections 2.4 and 2.5.

Section 3 – Aging Management Review Results

10 CFR 54.21 (a)(3) requires a demonstration that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis throughout the period of extended operation. Section 3 presents the results of the aging management reviews. Section 3 is the link between the scoping and screening results provided in Section 2 and the aging management activities provided in Appendix B.

Aging management review results are presented in tabular form, in a format in accordance with NUREG-1800, "Standard Review Plan for Review of License Renewal Applications," April 2001. For mechanical systems, aging management review results are provided in Sections 3.1 through 3.4 for the reactor vessel, internals, and reactor coolant system, engineered safety features, auxiliary systems, and steam and power conversion systems. Aging management review results for containments, structures, and component supports are provided in Section 3.5. Aging management review results for electrical and instrumentation and controls are provided in Section 3.6.

Tables are provided in each of these sections in accordance with NUREG-1800 which provide aging management review results for components, materials, environments, and aging effects and the aging management review results for components, materials, environments, and aging effects which are not addressed in NUREG-1801.

Section 4 – Time-Limited Aging Analyses

Time-limited aging analyses (TLAAs), as defined by 10 CFR 54.3, are listed in this section. This section includes each of the TLAAs identified in the NRC Standard Review Plan for License Renewal Applications and in plant-specific analyses. This section includes a summary of the time-dependent aspects of the analyses. A demonstration is provided to show that the analyses remain valid for the period of extended operation, the analyses have been projected to the end of the period of extended operation, or the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Appendix A – Updated Final Safety Analysis Report Supplement

As required by 10 CFR 54.21(d), the Updated Final Safety Analysis Report (UFSAR) supplement contains a summary of activities credited for managing the effects of aging for the period of extended operation. In addition, summary descriptions of time-limited aging analyses evaluations are provided. A separate UFSAR Supplement is provided for both Dresden, Units 2 and 3 and Quad Cities, Units 1 and 2.

Appendix B – Aging Management Programs

Appendix B describes the activities that are credited for managing aging effects for components or structures during the period of extended operation based upon the aging management review results provided in Section 3 and the time-limited aging analyses results provided in Section 4.

The first section of Appendix B discusses those activities which are discussed in Section X and Section XI of GALL. A description of the aging management activity is provided and a conclusion based upon the results of an evaluation to each of the ten elements provided in GALL is made. In some cases, acceptable alternatives for managing aging are provided for specific elements. Additional operating experience is provided.

The second section of Appendix B addresses each of the ten program elements for activities that are credited for managing aging and are not evaluated in GALL.

Appendix C – Commodity Groups (Optional)

Appendix C is not used.

Appendix D – Technical Specification Changes

No technical specification changes or additions necessary to manage the effects of aging during the period of extended operation are submitted with this application. Appendix D is not used.

Appendix E – Environmental Information

This Appendix satisfies the requirements of 10 CFR 54.23 to provide a supplement to the environmental report that complies with the requirements of subpart A of 10 CFR Part 51 for Dresden, Units 2 and 3.

Appendix F – Environmental Information

This Appendix satisfies the requirements of 10 CFR 54.23 to provide a supplement to the environmental report that complies with the requirements of subpart A of 10 CFR Part 51 for Quad Cities, Units 1 and 2.

1.6 ACRONYMS

Acronym	Meaning
ACAD	Atmospheric containment air dilution system
ADS	Automatic depressurization system
AFU	Air filtration unit
AHU	Air handling unit
AMR	Aging management review
ARI	Alternate rod insertion
ASME	American Society of Mechanical Engineers
ATWS	Anticipated transients without scram
BWR	Boiling water reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CAM	Containment atmospheric monitoring (system)
CASS	Cast austenitic stainless steel
CCST	Contaminated condensate storage tank
CCSW	Containment cooling service water
CFR	Code of Federal Regulations
CLB	Current licensing basis
CRD (H)	Control rod drive (hydraulic)
CS	Core spray (system)
CUF	Cumulative usage factor
DBD	Design baseline document
DBE	Design-basis event
DG	Diesel generator
DNI	Drywell nitrogen inerting
ECCS	Emergency core cooling systems
EDG	Emergency diesel generator
EFPY	Effective full-power years
EPRI	Electric Power Research Institute
EQ	Environmental qualification
ESF	Engineered safety feature
FAC	Flow accelerated corrosion
FSSD	Fire safe shutdown
FWRV	Feedwater regulating valve
GL	Generic Letter
HCU	Hydraulic control unit

Acronym	Meaning
HELB	High energy line break
HEPA	High efficiency particulate air
HPCI	High pressure coolant injection (system)
HRSS	High radiation sampling system
HVAC	Heating, ventilation, and air conditioning
HX	Heat exchanger
I & C	Instrumentation and controls
IGSCC	Intergranular stress corrosion cracking
IN	Information Notice
INPO	Institute of Nuclear Power Operations
IPA	Integrated plant assessment
ISI	Inservice inspection
IST	Inservice testing
LER	Licensee event report
LLRT	Local leak rate test
LOCA	Loss of coolant accident
LPCI	Low pressure coolant injection (system)
LPRM	Local power range monitor
LRA	License renewal application
MCC	Motor control center
MG	Motor generator
MIC	Microbiologically influenced corrosion
MOV	Motor-operated valve
MSV	Main stop valve
MSIV	Main steam isolation valve
NCAD	Nitrogen containment atmospheric dilution
NDE	Nondestructive examination
NBI	Nuclear boiler instrumentation
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
OE	Operating experience
P&ID	Piping and instrumentation diagram
PCIS	Primary containment isolation system
PM	Preventive maintenance
P-T curves	Pressure-temperature limit curves
PUA	Plant-unique analyses
RBCCW	Reactor building closed cooling water
RCIC	Reactor core isolation cooling (system)

Acronym	Meaning
RCS	Reactor coolant system
RFP	Reactor feed pump
RHR	Residual heat removal (system)
RHRSW	Residual heat removal service water
RMS	Radiation monitoring system
RPS	Reactor protection system
RPV	Reactor pressure vessel
RT_{NDT}	nil-ductility transition reference temperature
RVWLIS	Reactor vessel water level instrumentation system
RWM	Rod worth minimizer
RWCU	Reactor water cleanup system
SBLC	Standby liquid control (system)
SBO	Station blackout
SCC	Stress corrosion cracking
SBGTS	Standby gas treatment system
SJAE	Steam jet air ejector
SRM	Source range monitor
SRV	Safety relief valve
SSCs	Systems, structures, and components
SSMP	Safe shutdown makeup pump (system)
SV	Safety valve
SW	Service water
TBCCW	Turbine building closed cooling water
TCV	Turbine control valve
TID	Total integrated dose
TIP	Transverse incore probe
TLAAs	Time-limited aging analyses
UFSAR	Updated final safety analysis report
USE	Upper-shelf energy

1.7 GENERAL REFERENCES

1. 10 CFR 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

2. NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," Revision 3, April 16, 2001.

3. Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses."

4. NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," United States Nuclear Regulatory Commission, April 2001.

5. NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," United States Nuclear Regulatory Commission, April 2001.

6. 10 CFR 50.48, "Fire Protection."

7. 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants."

8. 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

9. 10 CFR 50.63, "Loss of All Alternating Current Power."

10. 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants."

11. 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

12. 10 CFR 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions."

CHAPTER 2
SCOPING AND SCREENING METHODOLOGY FOR IDENTIFYING STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW AND IMPLEMENTATION RESULTS

List of Tables

2.0 SCOPING AND SCREENING METHODOLOGY FOR IDENTIFYING STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW AND IMPLEMENTATION RESULTS

Section 2.0 of this application provides the following information that is required by 10 CFR Part 54, the license renewal rule (the Rule) and described in NUREG-1800, the Standard Review Plan (SRP) for Review of License Renewal Applications for Nuclear Power Plants:

- Scoping and Screening Methodology
- Plant Level Scoping Results
- Scoping and Screening Results: Mechanical Systems
- Scoping and Screening Results: Structures
- Scoping and Screening Results: Electrical and Instrumentation and Controls Systems

2.1 SCOPING AND SCREENING METHODOLOGY

For systems, structures and components (SSCs) within the scope of license renewal, 10 CFR 54.21(a)(1) requires the license renewal applicant to identify and list the structures and components subject to an aging management review (AMR). 10 CFR 54.21(a)(2) further requires that the methods used to implement the requirements of 10 CFR 54.21(a)(1) be described and justified.

As required by 10 CFR 54.21(a)(2), this section of the application provides a description and justification of the methodology used to identify and list structures and components subject to an AMR.

2.1.1 Introduction

The first step in the integrated plant assessment (IPA) process identified the plant systems, structures and components (SSCs) within the scope of 10 CFR Part 54. For those SSCs identified to be within the scope of the Rule, the IPA process then identified and listed the structures and components that are subject to an aging management review (AMR).

The process of identifying plant SSCs within the scope of 10 CFR Part 54 is called scoping. The process of identifying and listing the structures and components that are subject to an AMR is called screening. Scoping and screening have been performed consistent with the guidelines presented in NEI-95-10 (Reference 2.1).

Section 2.1.2 discusses the application of the 10 CFR 54.4(a) scoping criteria. Section 2.1.3 provides a discussion of the documentation used to perform scoping and screening. Section 2.1.4 describes the scoping methodology, Section 2.1.5 describes the screening methodology, and Section 2.1.6 provides a discussion of additional considerations incorporated into the methodology.

2.1.2 Application of Scoping Criteria in 10 CFR 54.4(a)

10 CFR 54.4(a)(1), (a)(2) and (a)(3) provide criteria for including systems, structures, and components (SSCs) within the scope of the license renewal rule (the Rule). These criteria are briefly identified as follows:

1. Title 10 CFR 54.4(a)(1) – Safety-related

2. Title 10 CFR 54.4(a)(2) – Non-safety related affecting safety-related

3. Title 10 CFR 54.4(a)(3) – The five regulated events

    - Fire Protection (10 CFR 50.48)

    - Environmental Qualification, EQ (10 CFR 50.49)

- Pressurized Thermal Shock (10 CFR 50.61) (*PWRs only*)
- Anticipated Transient Without Scram, ATWS (10 CFR 50.62)
- Station Blackout, SBO (10 CFR 50.63)

The application of each of these criteria to plant SSCs is discussed in Section 2.1.2.1, Section 2.1.2.2, and Section 2.1.2.3, respectively.

2.1.2.1 Title10 CFR 54.4(a)(1) – Safety-related

10 CFR 54.4(a)(1) requires that plant SSCs within the scope of license renewal include –

> Safety related SSCs which are those relied upon to remain functional during and following design-basis events (as defined in 10 CFR 50.49(b)(1)) to ensure the following functions –
>
> (i) The integrity of the reactor coolant pressure boundary
>
> (ii) The capability to shutdown the reactor and maintain it in a safe shutdown condition; or
>
> (iii) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11, as applicable.

Safety related classifications for systems and structures at Dresden and Quad Cities are reported in their respective UFSARs or in design basis documents of record such as engineering drawings, evaluations, or calculations. At Dresden and Quad Cities, safety related classifications for components are documented on engineering drawings and in a controlled plant component database. The safety related classification as reported in these source documents has been relied upon to identify SSCs satisfying criteria of 10 CFR 54.4(a)(1) and have been identified as within the scope of license renewal.

2.1.2.2 Title 10 CFR 54.4(a)(2) – Non-safety related affecting safety-related

10 CFR 54.4(a)(2) requires that plant SSCs within the scope of license renewal include –

> All non-safety related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified for 10 CFR 54.4(a)(1)(i), (ii) or (iii), as listed in preceding Section 2.1.2.1.

Initial determination and identification of SSCs satisfying criterion 10 CFR 54.4(a)(2) was completed as described below based on review of applicable CLB documents, the Maintenance Rule system/structure functional reports, and by system and structure functional evaluations. However, following publication of the NRC's staff interim guidance (Reference 2.2) on the identification and treatment of SSCs which meet 10 CFR 54.4(a)(2), the initial methodology for scoping to criterion 10 CFR 54.4(a)(2) was reviewed and revised to incorporate the NRC interim staff guidance (Reference 2.2). The revised methodology includes identification of non-safety related SSCs that are connected to safety related SSCs and non-safety related SSCs that could spatially interact with safety related SSCs. The description of the scoping methodology for non-safety related SSCs is presented here as it actually occurred in preparation of this application.

Initial Non-safety Related Scoping

For every non-safety related plant system or structure, applicable sections of the Dresden and Quad Cities UFSARs and other current licensing basis (CLB) documents were reviewed to determine whether the system or structure was credited with supporting satisfactory accomplishment of a safety related function. Non-safety related systems or structures explicitly credited in CLB documents with supporting accomplishment of a safety related function were classified as satisfying criterion 10 CFR 54.4(a)(2) and were included within the scope of license renewal.

To support implementation of the Maintenance Rule (10 CFR 50.65), Dresden and Quad Cities developed a Maintenance Rule Database that documents functional evaluations performed on the plant systems and structures at each site. Maintenance Rule functional evaluation reports include an evaluation of system and structure functions against criteria similar to 10 CFR 54.4(a)(1) and 10 CFR 54.4(a)(2). The Dresden and Quad Cities Maintenance Rule system/structure functional evaluation reports, were reviewed to identify any additional non-safety related system or structure functions categorized therein as supporting satisfactory accomplishment of a safety related function. Where such functions were identified in the Maintenance Rule Database and were confirmed to be part of the CLB, the system or structure was classified as satisfying criterion 10 CFR 54.4(a)(2) and was identified as within the scope of license renewal. The Maintenance Rule Database is further discussed in Section 2.1.3.2.

For each of the non-safety related systems and structures identified to be in-scope per 10 CFR 54.4(a)(2), as discussed above, a evaluation was performed. This evaluation identified those non-safety related components that satisfied criterion 10 CFR 54.4(a)(2). These components were identified as within the scope of license renewal.

Revised Non-safety Related Scoping

For non-safety related SSCs that are connected to a safety related SSC, the non-safety related SSC was included within the scope of the Rule up to the first equivalent anchor past the safety/non-safety interface.

Non-safety related SSC’s which are not connected to safety related piping or which are beyond the first equivalent anchor past the safety/non-safety interface, and have a spatial relationship such that their failure could adversely impact the performance of a safety related SSC intended function, were included in the scope of license renewal. The following spatial interactions were considered when determining if a non-safety related SSC should be included in-scope:

- pipe whip
- jet impingement
- general flooding

- spray
- displacement/falling.

Plant walkdowns were performed to identify those areas containing safety related SSCs. In those instances where a plant walkdown could not be performed, plant drawings were used to identify those areas containing safety related SSCs. Once those areas containing safety related SSCs were identified, each area was walked down once again to identify non-safety related SSCs that could spatially interact with safety related SSCs located in the same area. Plant drawings were used to identify those component interactions where plant walkdowns could not be performed. Each of the spatial interactions between non-safety related SSCs and safety related SSCs were documented in each area. Pipe whip, jet impingement, general flooding, or spray of a gas were not considered credible interactions for gas systems to adversely affect safety related SSCs. As such, those systems containing gas were excluded from the scope of the spatial interaction review and plant walkdowns. Displacement/falling was considered a credible interaction and as such, all supports for gas systems were included in the scope of the Rule.

2.1.2.3 Title 10 CFR 54.4(a)(3) – The Five Regulated Events

10 CFR 54.4(a)(3) requires that plant SSCs within the scope of license renewal include –

> All SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.61), and station blackout (10 CFR 50.63).

Because Dresden and Quad Cities are boiling water reactors (BWRs), 10 CFR 50.61, the regulation for pressurized thermal shock, is not applicable to them. However, for each of the other four applicable regulations, technical position papers were prepared (See Section 2.1.3.5) to provide input to the scoping process. The purpose of these position papers was (1) to identify the systems and structures that are relied upon to demonstrate compliance with these regulations, (2) to identify functional requirements for each system or structure, and (3) to identify additional documentation that may be used for scoping of components credited to demonstrate compliance with each of the applicable regulated events. Guidance provided by the technical position papers was used to determine components credited in the regulated events. SSCs credited in the regulated events have been classified as satisfying criterion 10 CFR 54.4(a)(3), and have been identified as within the scope of license renewal. The regulated event technical position papers are discussed in Section 2.1.3.5, Technical Position Papers.

2.1.3 Documentation Sources Used for Scoping and Screening

Various documentation sources were used during the scoping and screening process. These documentation sources are listed below and described in the following sections.

- Updated Final Safety Analysis Reports
- Maintenance Rule Database
- Current Licensing Basis and Design Basis Documents
- System and Structure Operational Description Documents
- Technical Position Papers
- Controlled Plant Component Database
- Systems and Structures List
- License Renewal Database

2.1.3.1 Updated Final Safety Analysis Reports

The Dresden and Quad Cities Updated Final Safety Analysis Reports (UFSAR) were used as a source of CLB information for each plant. One UFSAR applies for Dresden Units 2 and 3. A separate UFSAR applies for Quad Cities Units 1 and 2.

2.1.3.2 Maintenance Rule Database

The Dresden and the Quad Cities Maintenance Rule Databases were used as sources to identify functions for systems and structures. Each plant maintains a separate Maintenance Rule Database. For system and structures listed therein, the Maintenance Rule Databases provided descriptions of the functions for each system and structure, and evaluates those functions against the safety related criteria listed in 10 CFR 50.65(b)(1), which are identical to the safety related criteria of 10 CFR 54.4(a)(1).

The Maintenance Rule Database also identifies system and structure functions that may fall into the category of non-safety related affecting safety related. The criterion related to non-safety related affecting safety related for license renewal (10 CFR 54.4(a)(2)) and for the maintenance rule (10 CFR 50.65(b)(2)(ii)) are similar, and the evaluations documented in the Maintenance Rule Databases were used as inputs for license renewal evaluations. The functional evaluation results documented in the Maintenance Rule Databases were confirmed by additional review of applicable CLB documents.

2.1.3.3 Current Licensing Basis and Design Basis Documents

In addition to the UFSAR, a variety of current licensing basis and design basis documents were used to confirm or to determine additional system, structure, or component functions and evaluate them against the criteria of 10 CFR54.4(a). These documents are:

<u>Safety evaluation reports (SERs)</u> reflect the NRC Staff understanding of plant SSC functional requirements, performance characteristics, and related regulatory commitments. Dresden and Quad Cities SERs were reviewed to obtain information relevant to scoping and screening.

Technical Specifications provide safety limits, limiting conditions for operation, and surveillance requirements applicable to plant SSCs whose functions are critical to nuclear safety. Technical Specification bases provide discussions of SSC functional characteristics that underlie the limits and requirements. The Dresden and Quad Cities Technical Specification requirements and bases were reviewed to obtain additional information supporting the functional evaluation of SSCs.

Licensing correspondence reflecting Dresden or Quad Cities agreements, commitments or relief requests related to various SSCs and programs were reviewed.

Engineering drawings that provide configuration details were reviewed for every Dresden and Quad Cities mechanical system. This included electrical, mechanical, and structural drawings. Building lay-out drawings were reviewed for Dresden and Quad Cities structures. These drawing reviews supported evaluations related to component scoping.

Engineering evaluations and calculations provide detailed information about the requirements and characteristics of their subject SSCs. Engineering evaluations and calculations have been used to provide additional clarification of SSC requirements information obtained from other scoping and screening source documents.

2.1.3.4 System and Structure Operational Description Documents

System description documents and licensed-operator lesson plans are not classified as engineering or CLB documents at Dresden or Quad Cities. These descriptive documents were used in some instances to obtain clearer understanding of SSC functions, interfaces, and applicable requirements. The Dresden and Quad Cities scoping and screening evaluation process required that information obtained from these source documents be substantiated by design basis or licensing basis documents.

2.1.3.5 Technical Position Papers

Technical position papers were prepared as part of the preparation of the license renewal project to support scoping evaluations of the regulated events identified in 10 CFR 54.4(a)(3). Each technical position paper is further described in the following paragraphs.

Fire Protection

Criterion 10 CFR 54.4(a)(3) requires that plant SSCs within the scope of license renewal include all SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the regulations for fire protection (10 CFR 50.48). Title 10 CFR 50.48 requires each operating nuclear power plant to have a fire protection plan that satisfies the requirement of Criterion 3 of 10 CFR 50 Appendix A, and further requires all nuclear power plants licensed to operate prior to January 1, 1979, to comply with Sections III.G, III.J and III.O of Appendix R to 10 CFR 50.

These regulations may be broadly summarized as requiring the Dresden and Quad Cities fire protection plans to meet the following objectives for SSCs important to safety:

- Reduce the likelihood of fires;
- Promptly detect and extinguish fires that do occur;
- Maintain safe shutdown capability if a fire does occur; and
- Prevent release of significant amounts of radioactive material if a fire does occur.

A fire protection technical position paper was created for each plant. These position papers summarize the results of a detailed review performed on the fire protection program documents demonstrating compliance with the requirements of 10 CFR 50.48 for each plant. The position papers provide a list of systems and structures credited in the fire protection program documents for each plant. For the listed systems and structures, the position papers also identify appropriate CLB references used in developing the summary information. The system descriptions identified in the fire protection technical position papers were used in scoping evaluations to identify SSCs that demonstrate compliance with 10 CFR 50.48. All SSCs classified as satisfying criterion 10 CFR 54.4(a)(3) were identified as within the scope of license renewal.

<u>Environmental Qualification</u>

Criterion 10 CFR 54.4(a)(3) requires that plant SSCs within the scope of license renewal include all SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the regulations for environmental qualification (10 CFR 50.49). The Dresden and Quad Cities environmental qualification (EQ) programs apply to electrical equipment important to safety that are located in a harsh environment. The EQ programs also include electrical equipment important to safety that is subject to significant known degradation due to aging where a qualified life has been established. Detailed qualification information related to EQ components at Dresden and Quad Cities is maintained in EQ binders.

Technical position papers were created for each plant. The environmental qualification position papers summarize the results of a review of the Dresden and Quad Cities EQ program documents and binders. The EQ position papers provide lists of systems that include EQ components and identify EQ binders applicable for components in those systems. All components within the scope of the Dresden and Quad Cities EQ programs which demonstrate compliance with 10 CFR 49 and the systems containing those components were classified as satisfying criterion 10 CFR 54.4(a)(3) and were identified as within the scope of license renewal.

<u>Pressurized Thermal Shock</u>

Criterion 10 CFR 54.4(a)(3) requires that plant SSCs within the scope of license renewal include all SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the regulations for pressurized thermal shock (10 CFR 50.61). This requirement is applicable only to pressurized water reactors

(PWRs). Dresden and Quad Cities are boiling water reactors (BWRs). No technical position paper was prepared for pressurized thermal shock.

Anticipated Transients Without Scram

Criterion 10 CFR 54.4(a)(3) requires that plant SSCs within the scope of license renewal include all SSCs relied upon in safety analyses or plant evaluations to perform a function that demonstrates compliance with the regulations for anticipated transients without scram (10 CFR 50.61). An anticipated transient without scram (ATWS) is a postulated operational transient that generates an automatic scram signal accompanied by a failure of the reactor protection system to shutdown the reactor. 10 CFR 50.62 requires improvements in the design and operation of BWRs to reduce the likelihood of failure to shutdown the reactor following anticipated transients and to mitigate the consequences of an ATWS event. The requirements for a BWR are to install an alternate rod insertion (ARI) system, a reactor coolant recirculation pump trip (RPT) system to be actuated for conditions indicative of an ATWS, and an adequately sized standby liquid control (SLC) system.

Both the Dresden and the Quad Cities plants have ATWS mitigation systems that provide instrumentation and logic necessary for ARI and RPT to mitigate the consequences of an ATWS event. Both Dresden and Quad Cities plants also have SLC systems credited to mitigate ATWS event consequences. All components of the ATWS mitigation systems and those components of the SLC system that function to mitigate the consequences of an ATWS event satisfy criterion 10 CFR 54.4(a)(3), and were identified as within the scope of the Rule. In addition to the systems designed explicitly to mitigate an ATWS event, other plant systems are included in the analysis for ATWS events as described in the Dresden and Quad Cities UFSARs. A technical position paper was created for each plant. The ATWS technical position papers summarize the results of a detailed review of the ATWS evaluation documents for each plant. The position papers provide lists of systems and structures credited in the Dresden and Quad Cities ATWS evaluations. For the listed systems and structures, the position papers also identify appropriate CLB references used in developing the summary information. The system descriptions identified in the ATWS technical position papers were used in scoping evaluations to identify SSCs that demonstrate compliance with 10 CFR 50.61. All SSCs classified as satisfying criterion 10 CFR 54.4(a)(3) were identified as within the scope of license renewal.

Station Blackout

Criterion 10 CFR 54.4(a)(3) requires that plant SSCs within the scope of license renewal include all SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the regulations for station blackout (10 CFR 50.63). A station blackout (SBO) event is a complete loss of alternating current (AC) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of the offsite electric power system concurrent with generator trip and unavailability of the onsite emergency AC power sources). SBO does not include the loss of available AC power to buses fed by station batteries through inverters or by alternate AC sources, nor does it assume a concurrent single failure or design basis accident.

Both Dresden and Quad Cities implemented plant modifications and procedures in response to 10 CFR 50.63 to enable each plant to withstand and recover from a SBO of a specified duration without sustaining reactor damage. Recovery includes the ability to achieve and maintain hot shutdown of the reactors. The Dresden and Quad Cities capabilities, commitments, and requirements that demonstrate compliance with 10 CFR 50.63 are documented in the UFSARs, NRC SERs, and supporting correspondence related to the SBO rule. A technical position paper was created for each plant. The SBO technical position papers summarize the results of a detailed review of the SBO documentation for each plant. The position papers provide lists of systems and structures and their functions credited in the Dresden and Quad Cities SBO evaluations. For the listed systems and structures, the position papers also identify appropriate CLB references used in developing the summary information. The systems descriptions and required functions identified in the SBO technical position papers were used in scoping evaluations to identify SSCs that demonstrate compliance with 10 CFR 50.63. All SSCs classified as satisfying criterion 10 CFR 54.4(a)(3) were identified as within the scope of license renewal.

The SBO technical papers identify the Dresden and Quad Cities SSCs credited with coping and recovering from a station blackout. Recovery from SBO, as described in the Dresden and Quad Cities SBO evaluations, is based on restoration of adequate AC power by restoration of AC power from on-site emergency diesel generators. The SBO technical position papers do not include all of the SSCs required to restore an off-site power supply to the plant. However, the NRC interim staff guidance on scoping of equipment relied on to meet the requirements of the SBO rule (10 CFR 50.63) for license renewal (Reference 2.3) was incorporated into the Dresden and Quad Cities scoping methodology. The NRC interim staff guidance states that the plant system portion of the offsite power system should be included within the scope of license renewal and describes this as plant "offsite power system … structures and components within the switchyard and down to the safety related buses." As a result of the NRC interim staff guidance, the SBO technical position papers were augmented to include additional structures and components of the offsite power system for each plant required to restore power from the on-site switchyard down to the safety related buses in the plant. The plant offsite power system and these additional structures and components were classified as satisfying criterion 10 CFR 54.4(a)(3), and were identified as within the scope of the Rule.

2.1.3.6 Controlled Plant Component Database

Dresden and Quad Cities each maintain a controlled plant component database containing integrated design and maintenance record management information. Each plant component database lists plant components at the level of detail for which discrete maintenance or modification activities typically are performed. This is a finer level of detail than is normally shown on piping and instrument diagrams. At Dresden and Quad Cities, the controlled plant databases provide a comprehensive list of components.

Each component in the controlled plant database is uniquely identified by the combination of the following fields: station, unit, system, equipment part number (EPN) and component type. The plant component database is used, in part, to track modification and maintenance history for components.

A license renewal project database was created to support scoping, screening, and aging analysis, as discussed in Section 2.1.3.8. This license renewal (LR) database includes a component table which was populated by down loading the component records from the Dresden and Quad Cities controlled plant component databases.

2.1.3.7 Systems and Structures List

The list of systems and structures evaluated for license renewal scoping was created from lists contained in the plant controlled database and the Maintenance Rule databases. This list of systems was also reviewed against the Dresden and Quad Cities UFSARs, plant design drawings, maintenance rule databases, and other plant design documents. A site walkdown was conducted to ensure that all site structures were appropriately listed for evaluation. A complete list of systems and structures evaluated is provided in Section 2.2 of this application.

Certain structures and equipment were excluded at the outset because they do not meet criteria 10 CFR 54.4(a)(1), (a)(2), or (a)(3). These include: driveways and parking lots, temporary equipment, health physics equipment, portable radios, portable measuring and testing equipment, tools, spare parts, and motor vehicles. None of the miscellaneous items excluded were credited for use in the CLB or needed to support site response to any of the regulated events. In addition, structures and equipment for emergency preparedness and security were excluded from the scope of license renewal.

2.1.3.8 License Renewal Database

The Dresden/Quad Cities license renewal (LR) database was developed as a project tool to support scoping and screening evaluation activities, aging management evaluation activities, and to document references and other information used in preparing the license renewal application. A file was created in the LR database for each system and structure at Dresden and Quad Cities. These systems and structures were those as described in Section 2.1.3.7. The LR database files for each system and structure were initially populated by down loading component records from the Dresden and Quad Cities controlled plant component databases. The following data fields were down loaded for each component in the plant component database: 1) station; 2) unit; 3) system number, 4) equipment part number (EPN); 5) component type code; 6) component description; and 7) safety classification. The LR database also contains references used in the scoping and screening evaluations and provides a repository for scoping and screening evaluation results and references. This information includes: 1) classification of the component as in-scope or not-in-scope; 2) classification of the component as active or passive; 3) component intended function(s); 4) classification of the component as short-lived or long-lived; and 5) a field used to identify the component as subject to an aging management review. Additional information, such as the component material, internal and external environments, were also populated to enhance efficiency of the aging management evaluation. Such information was validated during the aging management review process.

2.1.4 Scoping Methodology

Scoping of the Dresden and Quad Cities systems, structures and components (SSCs) was performed to the criteria of 10 CFR 54.4(a) to identify those (SSCs) within the scope of the license renewal rule. Scoping evaluations were performed separately for Dresden and Quad Cities. The Dresden and Quad Cities scoping evaluation results have been retained in the shared Dresden/Quad Cities license renewal (LR) database that is described in Section 2.1.3.8. The following sections describe the methodology used for scoping. Separate discussions of mechanical system scoping methodology, electrical and instrumentation and control (I&C) system scoping methodology, and structure scoping methodology are provided. The description of scoping methodology provided for mechanical systems in Section 2.1.4.1 is applicable for scoping of electrical/I&C systems and structures, except where differences are identified. The scoping and screening process was performed using controlled written instructions.

2.1.4.1 Mechanical System Scoping Methodology

Dresden and Quad Cities are boiling water reactors, classified as General Electric BWR/3 designs with Mark I primary containments. A list of mechanical systems was developed as described in Section 2.1.3.7 and these mechanical systems were scoped against the criteria of 10 CFR 54.4(a). The list of mechanical systems and the results of the scoping are provided in Section 2.2, Table 2.2-1, of this application. Most of the systems listed in Table 2.2-1 exist at both plants and have many common design features, functions, and component types.

For every mechanical system listed in Table 2.2-1, the following scoping process was applied.

- Identification of the System Purpose and Functions
- Determination of the System Evaluation Boundary
- Comparison of System Functions Against 10 CFR 54.4 (a)(1-3)
- Identification of Supporting Systems
- Creation of License Renewal Boundary Diagrams
- Component Level Scoping
- Document Scoping Results and References

The major process steps are described in this methodology in a discrete sequence. During actual implementation, this sequence was generally used, but at times steps were performed jointly to improve efficiency. Also, as information was obtained and conclusions reached during the system and component scoping, the need was identified to provide feedback and revise interim results in each of the scoping process steps.

Identification of the System Purpose and Functions

A description was prepared for each mechanical system that included the purpose and identified all functions that the system was designed to perform. This summary description was prepared using information obtained from the UFSAR system descriptions, Maintenance Rule database records, current licensing basis documents, design basis documents (including P&IDs), and system operating descriptions. The system functions identified at this point in the scoping process included all functions (intended and non-intended) of the system.

Determination of the System Evaluation Boundary

After the system purpose and functions were identified, the system evaluation boundary was determined and marked–up on P&IDs as the start of the LR boundary diagram. Included in the evaluation boundary are all of the components needed for the system to perform all of its functions, including those functions determined to not be system intended functions. All of the components within the system evaluation boundary were reviewed and scoped against criteria of 10 CFR 54.4(a) during evaluation of the system. The evaluation boundary for each system was based on the information sources listed above and from the component information found in the plant component database. Mechanical system P&IDs that show the system configuration, including component equipment part numbers, were used to define the evaluation boundary of a system and to define the level of detail that was appropriate to support the scoping and screening evaluations of mechanical components. The system scoping summaries included in Section 2.3 of this application provide a description of the evaluation boundary for each mechanical system in the scope of the Rule.

The process to determine the system evaluation boundary required close examination of interfaces with other systems. System interfaces were closely examined for two reasons. The first reason was to ensure that all components were included in the evaluation boundary of one of the interfacing systems. The potential concern was that the two interfacing systems could be identified in such a way that one or more components at the interface between systems would not be in one of the two systems' evaluation boundaries.

The second reason was to ensure that the components were included in the proper system based on the system intended functions. Some components were initially included in the evaluation boundary of a system that had no intended functions, but whose components were needed to support the intended function of an interfacing system. When this occurred, the evaluation boundaries of the two interfacing systems were changed so the component needed to support the intended function was included in the appropriate system. This change of system evaluation boundaries allowed all components that are required to support system intended functions to be included in the system with the associated intended function. For example, out of scope systems containing primary containment isolation valves were evaluated with the primary containment system.

Comparison of System Functions Against 10 CFR 54.4.(a)(1-3)

All of the system functions, as described previously, were compared against the criteria of 10 CFR 54.4(a)(1), (a)(2) and (a)(3). Each of the system functions satisfying the scoping criteria in 10 CFR 54.4(a) was identified as a system intended function. Functions performed by safety related parts of the evaluated system were identified as satisfying criterion (a)(1) and were classified as intended functions. Functions performed by non-safety related systems or parts of systems that are required to ensure success of a safety related function were identified as satisfying criterion (a)(2) and classified as intended functions. Functions that were credited in one of the regulated events were identified as satisfying criterion (a)(3) and classified as intended functions. A function may have been classified as an intended function under more than one of the three criteria in 10 CFR 54.4.

Any system that contained one or more intended functions (i.e. satisfying criterion (a)(1), (a)(2), or (a)(3)) was classified as a system within the scope of the Rule. Those systems for which no functions were identified as satisfying any of the three scoping criteria were classified as systems outside the scope of the Rule. For systems classified as outside the scope of the Rule, no further evaluation was performed, and the system description documented earlier in the LR database was augmented to state that the system was determined to be outside the scope of the Rule. When a system was determined to be outside the scope of the Rule, all of the components for that system listed in the LR database were identified as outside the scope of the Rule and were excluded from further scoping or screening evaluations. However, a review was performed on all components before they were excluded from further review to further ensure that there were no safety related or environmentally qualified components in the population. Component scoping for in-scope systems is discussed later in this section.

<u>Identification of Supporting Systems</u>

After a system was determined to be in the scope of the Rule, an evaluation was performed to identify all of the in-scope system's supporting systems. Each of the supporting systems was then reviewed to determine if its failure could prevent satisfactory accomplishment of any intended functions of the in-scope system. When it was identified that a supporting system was needed to maintain an intended function of the in-scope system, the supporting system was determined to be in-scope. When a supporting system was identified as being in-scope, the scoping evaluation for the supporting system was reviewed and revised as necessary. This step in the scoping process ensured that supporting systems were correctly classified with respect to criteria 10 CFR 54.4(a)(2).

<u>Creation of License Renewal Boundary Diagrams</u>

A license renewal (LR) boundary diagram was created for each mechanical system determined to be in-scope of license renewal. The LR boundary diagrams were created in conjunction with the component scoping. LR boundary diagrams reflect the system evaluation boundary, as previously discussed in this section. The diagrams were created by marking-up the plant piping and instrumentation diagrams (P&IDs) associated with the mechanical system being evaluated. LR boundary diagrams include: 1) the system evaluation boundary; 2) the in-scope components whose function is required to ensure success of the system intended functions; and 3) the out of scope components whose function is not required to ensure success of the system-level intended functions. In-scope components are designated on the LR boundary diagrams by highlighting them with color. Out of scope components are shown in black. Those non-safety related SSCs included in the scope of the Rule solely for 10 CFR 54.4(a)(2) are shown on the LR boundary diagrams in red. All other SSCs in the scope of the Rule are shown in green on the LR boundary diagrams.

Component Level Scoping

The system component list obtained from the plant component database was down loaded into the LR database component table as described in Section 2.1.3.8. System components are uniquely identified by the combination of station, unit, system number, equipment part number (EPN), and component type code, that are included in each record of the LR database.

Components shown on the LR boundary diagrams for a particular station, unit, and system include the EPN for each component. The EPN is used to identify the components in both the plant component database and the LR database. Component records in the LR database component table also include a safety-classification field, which was populated from the controlled plant component database.

Every safety related component was included within the scope of the Rule. All other components in the LR database for a given system were reviewed to determine if they supported any of the intended functions for a given system. This was done by reviewing the system functions, drawings, and other information sources to determine if failure of the component would result in failure of a system intended function.

A component was determined to be in-scope if it was safety related meeting the criteria of 10 CFR 54.4a(1), if it was determined that the component was needed to fulfill a system intended function, if the component met the criteria of 10 CFR 54.4a(2), or if the component was needed to support the intended function of the system needed to meet the regulation for regulated events. The results of the component scoping are documented in the LR database.

The LR boundary drawing for each in-scope system was reviewed to identify those components within the system required to support the system intended functions. Not all components on the P&IDs are included in the plant component and LR databases. Examples are piping and pipe fittings. Each system LR boundary diagram was reviewed and all commodities (such as piping and pipe fittings) indicated on the LR boundary diagrams to be in-scope for license renewal were added to the component list in the LR database or confirmed to already be in the LR database. These components were identified as in-scope of the Rule.

The LR database includes uniquely identified components that are not explicitly shown on the P&IDs or on the LR boundary diagrams. Each of these components was evaluated individually based on the EPN, component type, and component description. Based on this evaluation, it was determined whether the component supports a system level intended function, meets the criteria of 10 CFR 54.4a(2), or is credited for a regulated event. Components meeting one of these three criteria were identified in the LR database as within the scope of the Rule. Components not meeting one of these three criteria were identified in the LR database as out-of-scope for the Rule.

The component scoping methodology described above was performed for every mechanical component found within an in-scope system. Electrical and instrument and control (I&C) components were treated differently. All electrical and I&C components within the evaluation boundary of in-scope mechanical systems were included within the scope of the Rule and evaluated using the spaces approach described in Section 2.1.4.2 of this application. The electrical and I&C components from all plant systems (mechanical / electrical and I&C) and structures were scoped collectively under one electrical and I&C component group. This conservative scoping approach was applied to all electrical and I&C components provided the component did not have a mechanical component function such as a pressure boundary. In those instances, the components were evaluated individually for aging along with other components in the mechanical system.

All mechanical system components in the LR database that were identified as in-scope for license renewal were screened against the criteria of 10 CFR 54.21(a)(1) to determine whether they were subject to an aging management review. The screening methodology is discussed in Section 2.1.5.

Document Scoping Results and References

Throughout the scoping process described above, scoping results were documented in the LR database for each mechanical system. The current licensing basis and design basis documents reviewed in support of the scoping activities were also documented in the LR database for each system.

2.1.4.2 Electrical and Instrumentation and Control System Scoping Methodology

A list of electrical and instrumentation and control (I&C) systems was developed as described in Section 2.1.3.7. These electrical and I&C systems were scoped against the criteria of 10 CFR 54.4(a). The list of electrical and I&C systems and the results of the scoping are provided in Section 2.2, Table 2.2-3, of this application. Most of the systems listed in Table 2.2-3 exist at both stations, are similarly named, and have many common design features, functions, and component types.

Electrical and I&C systems include commodity-type components, such as cables and connectors, which are not listed in the controlled plant component database. These commodity-type components were identified by reviewing documents such as plant drawings and the site controlled cable database. Industry documents, such as NEI 95-10 and SAND96-0344, (References 2.1 and 2.5 respectively), provide an inclusive list of typical electrical components found in nuclear power plants. These lists were reviewed against engineering information for the plants to determine which electrical components are installed at Dresden and Quad Cities. These typical electrical components were included in the license renewal database for evaluation.

System Level Scoping

At the system level, the scoping methodology utilized for electrical and I&C systems was identical to the mechanical system-level scoping described in Section 2.1.4.1. The UFSAR descriptions, Maintenance Rule database records, CLB and design basis documents, and system description documents applicable to the system were reviewed to determine the system safety classification and to identify all of the system functions. All system level functions were evaluated against the criteria of 10 CFR 54.4(a)(1), (a)(2) and (a)(3). The supporting systems needed to maintain the in-scope system intended functions were identified and evaluated against the criteria in 10 CFR 54.4(a)(2). The results of the system level scoping along with a list of references supporting the evaluation of each electrical and I&C system were documented in the LR database.

System Evaluation Boundary

A few electrical and I&C systems include mechanical components that support the intended function of in-scope mechanical systems. For example, the traversing incore probe (TIP) system penetrates the primary containment structure and includes safety related mechanical isolation valves and associated tubing that support the intended functions of the primary containment structure. For these electrical and I&C systems, the evaluation boundary of the interfacing in-scope mechanical system was expanded to encompass the supporting mechanical components provided by the electrical and I&C system. The mechanical components were evaluated as part of the evaluation boundary for the interfacing mechanical system. The results of these evaluations were documented in the LR database. Because all electrical components, whether in mechanical systems or electrical systems, were included in-scope, LR Boundary Diagrams were not created for electrical systems. However, boundary diagrams showing the basic electrical distribution throughout the plant and the associated switchyards were created for each site. (See boundary diagrams LR-DRE-E-1, LR-DRE-E-2, LR-QDC-E-1, and LR-QDC-E-2.)

Component Level Scoping

Unlike mechanical systems, the electrical and I&C components for in-scope systems that are in the scope of the rule were evaluated using the “spaces” approach. The “spaces” approach for electrical and I&C components is an approach to both scoping and aging management. The aging management part of the “spaces” approach performs evaluations of all passive electrical and I&C components in an area of the plant, regardless of whether the components are part of in-scope systems or not. Therefore, the scoping part of the “spaces” approach does not include a detailed scoping review to identify specific electrical and I&C components in each in-scope system.

Scoping for electrical and I&C components using the “spaces” approach identifies the electrical and I&C commodity groups that are installed in the plant. The identification of these commodity groups was described previously in this section. The aging management review using the “spaces” approach determined in which area of the plant the components were located and reviewed how the components age based on the environment in each of the areas. See Section 2.1.6 of this application for more information on the “spaces” approach used for aging management.

2.1.4.3 Structure Scoping Methodology

A list of structures was developed as described in Section 2.1.3.7. Structures as discussed in this section include free-standing buildings, separately evaluated rooms that are contiguous with free-standing buildings, the primary containment shell, tank foundations, the station chimney, and commodity-like groupings of cranes and hoists. Several plant buildings or structures that have similar structural characteristics and functions were grouped together. These structures and structure groupings are listed in Section 2.2, Table 2.2-2, of this application. The list of structures used for scoping was developed through review of site plot drawings in conjunction with a walkdown of the property at each site as described in Section 2.1.3.7. The UFSARs were relied upon to identify safety classification of structures and structural components. Class I structures and structural components were considered safety related.

Individual structures are not listed in the controlled plant component databases at Dresden or Quad Cities. Unlike mechanical and electrical systems, structures at Dresden and Quad Cities stations are not assigned unique identifiers and are not listed in the controlled plant component database. Similarly named structures at the separate plants perform similar functions and have similar design requirements. However, they may differ in structural features and details.

The scoping methodology utilized for structures was identical to the mechanical system-level scoping described in Section 2.1.4.1. Structure descriptions were prepared, including the structure purpose and all functions. Structure evaluation boundaries were determined, including examination of structure interfaces. This information was included in the LR database. All structure functions were evaluated against the criteria of 10 CFR 54.4(a)(1), (a)(2) and (a)(3) and the results of this evaluation were documented in the LR database. In those instances where the structure intended functions required support from other structures or systems, the supporting systems or structures were identified and evaluated against the criteria in 10 CFR 54.4(a)(2). A list of references supporting the evaluation of each structure was documented in the LR database.

Structural Boundary Drawings

Unlike mechanical systems, individual LR boundary diagrams were not created for structures. However, a single boundary diagram based on site plot or equipment lay-out drawings was created for each site. The LR boundary diagram for each site displays all of the structures in relation to one another. Those structures within the scope of the rule are clearly identified on the drawing and in Table 2.2-2.

Structural Component Scoping

Although the controlled plant component database does include some structural components such as pipe supports, equipment anchors, ladders, and doors, it does not include most of the structural components that comprise a structure, and that are evaluated in an AMR. For structures determined to be within the scope of the Rule, more detailed structural drawings were reviewed and, where needed, walkdowns were performed, to identify structural elements (such as structural steel, foundations, floors, walls, ceilings, penetrations, stairways or curbs). For in-scope structures, all structural components that are required to support the intended functions of the structure were entered into the LR database and were identified as in-scope of the Rule. These structural components were entered in the LR database as generic structural components, not as individual components. For example, each structural beam and girder was not entered, but structural steel was entered as a generic structural component. For each in-scope structure, all of the structural components listed in the LR database were evaluated and a determination was made whether the structural component was required to support the intended functions of the structure. Structural components that support the intended functions of the structure were identified in the LR database as within the scope of the Rule. Structural components that were determined not to support the intended functions of the structure were identified within the LR database as outside the scope of the Rule.

2.1.5 Screening Methodology

Screening is the process of determining, identifying, and listing the structures and components that are subject to an aging management review (AMR). This section, and the accompanying subsections for mechanical systems, electrical and I&C systems, and structures, describes the process used to perform screening for the Dresden and Quad Cities license renewal application.

All systems, structures and components (SSCs) listed in the Dresden and Quad Cities LR database were scoped to the criteria of 10 CFR 54.4(a) as described in Section 2.1.4. All of the Dresden and Quad Cities structures and components categorized as within the scope of the Rule were screened against the criteria of 10 CFR 54.21(a)(1)(i) and (1)(ii) to determine whether they are subject to an AMR. The screening methodology utilized is described in this section of the application.

Title 10 CFR 54.21 states that the structures and components subject to an AMR shall encompass those structures and components within the scope of the license renewal rule –

(i) That perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties; and

(ii) That are not subject to replacement based on a qualified life or specified time period.

For simplicity, the word “passive” is used in the screening process for all components that perform intended functions without moving parts or a change in configuration or properties. All components that are not “passive” are known as “active”. Also for simplicity, the word “long-lived” is used in the screening process for all components that are not subject to replacement based on qualified life of specific time period. Components that are not “long-lived” are known as “short-lived”.

NEI-95-10 (Reference 2.1), Appendix B, “Typical Structure, Component and Commodity Groupings and Active/Passive Determinations for the Integrated Plant Assessment,” provides industry guidance for screening structures and components against criterion of 10 CFR54.21(a)(1)(i). The guidance provided in NEI-95-10, Appendix B, has been incorporated into the Dresden and Quad Cities license renewal screening process. Slightly differing screening methodologies have been applied for mechanical systems, electrical and I&C systems, and structures. The screening methodology applied for each category of system and for structures is described in the following paragraphs.

2.1.5.1 Mechanical System Component Screening Methodology

In mechanical systems, component screening was a continuation of the component scoping activity. Mechanical systems were scoped at the system level and scoping continued to the component level as described in Section 2.1.4.1. After a mechanical system component was categorized in the LR database as within the scope of the Rule, the classification as an active or passive component was determined based on evaluation of the component description and type. The active/passive component determinations documented in NEI-95-10 (Reference 2.1), Appendix B, provided guidance for this activity. In-scope components that were determined to be passive were identified in the LR database as subject to an AMR.

Each component that was identified as subject to an AMR was evaluated to determine its component intended function(s). The component intended function(s) for each in-scope passive component was identified based on an evaluation of the component type and the way(s) in which the component supports the system intended functions. Most in-scope passive components perform only one intended function. However, a few in-scope component types may perform more than one function (e.g., a flow orifice may have the component intended functions of pressure boundary and throttle). The results of the component screening were recorded in the LR database. The list of component intended functions utilized in the screening of mechanical system components can be found in Table 2.1-1, Component Intended Functions.

During the screening process, a few in-scope passive components were recognized in the screening process as short-lived components. Examples of such components are the reactor head-seal o-rings and the control blades. Components that were recognized during screening as short-lived were eliminated from the AMR process and the basis for the classification as short-lived was documented in the LR database. All other in-scope passive components were identified in the LR database as subject to an AMR. During the AMR process, if detailed review of maintenance procedures and requirements determined that a component previously categorized as long-lived was subject to replacement based on a qualified life or specified time period, the component was re-categorized as short-lived and eliminated from the AMR evaluation process.

2.1.5.2 Electrical and I&C System Component Screening Methodology

In electrical and I&C systems, component screening was a continuation of the component scoping activity. Electrical and I&C systems were initially scoped at the system level, and scoping continued to the component level as described in Section 2.1.4.2. Based on the spaces approach to aging management review for electrical components, all electrical and I&C components classified as within the scope of the Rule by the methodology described in Section 2.1.4.2 were evaluated as a consolidated electrical and I&C component group. Components were categorized as “active” or “passive” based on the determinations documented in NEI-95-10, Appendix B. In-scope components that were determined to be passive were identified in the LR database as subject to an AMR. The component-level intended function(s) were determined for each in-scope passive component and recorded in the LR database. All passive electrical and I&C commodity components, such as cables, are subject to an AMR unless they were specifically evaluated and determined not to perform an intended function as described in 10 CFR 54.4. See Section 2.1.6 of this application for more information on evaluation of electrical and I&C components using the spaces approach.

As stated in Section 2.1.4.1, electrical and I&C components from mechanical systems were screened collectively using the spaces approach along with similar components from electrical and I&C systems. This also applied to any electrical and I&C components associated with structures. Any mechanical or structural components in electrical and I&C systems that were determined to be within the scope of the Rule were categorized as “active” or “passive” based on the determinations documented in NEI-95-10. In-scope components that were determined to be passive were identified in the LR database as subject to an AMR. The component-level intended function(s) was also determined and recorded in the LR database. The list of component intended functions utilized in the screening of electrical and I&C system components are found in Table 2.1-1, Component Intended Functions.

2.1.5.3 Structural Component Screening Methodology

Structures and structural components typically perform their functions without moving parts and without a change in configuration or properties. When a structure or structural component, was determined to be within-scope of the Rule by the scoping process described in Section 2.1.4.3, the structure screening methodology classified the component as passive. This is consistent with guidance found in NEI-9510, Appendix B. During the structural screening process, the intended function(s) of structural components were determined and recorded in the LR database. In the structure screening process, an evaluation was made to determine whether in-scope structural components were subject to replacement based on a qualified time period. For example, an elastomer seal may have been determined to be replaced on a specified time period. If such a determination was made for an in-scope structural component, the component was identified as short-lived and was excluded from an AMR. In such a case, the basis for determining that the structural component was short-lived was documented in the LR database. Except for a very limited number of structural components that were excluded on the basis of criterion 10 CFR 54.21(a)(1)(ii), all structures and structural components that are in-scope are subject to an AMR. The list of component intended functions utilized in the screening of structural components are found in Table 2.1-1, Component Intended Functions.

2.1.6 Additional Considerations Incorporated into the Methodology

This section describes additional considerations incorporated into the Dresden and Quad Cities scoping and screening methodology.

Integration of Scoping with Screening

Scoping and screening were not treated as independent processes. Scoping initially was performed at the system/structure level. Scoping was continued at the component level and identified the components that are required to ensure successful completion of the intended function(s) of a system or structure as defined in 10 CFR 54.4. Components that are required to ensure success of the intended functions of a system or structure were identified as within the scope of the Rule. The component level evaluation also determined the active/passive classification of each in-scope component, and all results have been stored in the LR database, described in Section 2.1.3.8. During this evaluation, the scoping and screening processes were intimately linked and information flowed between scoping and screening.

Integration of Scoping and Screening with Aging Management Review

Part of the scoping and screening activity was carried forward into the AMR process. During the scoping and screening process, relatively few in-scope, passive components were evaluated in detail with regard to whether they were long-lived or short-lived. For most in-scope, passive components the determination of whether the component was short-lived or long-lived was made during the AMR process because it was during the AMR process that the procedures for maintaining and replacing plant equipment were reviewed in detail. The AMR process also included a detailed evaluation of component functions and interfaces. During AMR evaluations, revision of initial scoping and screening classifications were identified. The integrated process permitted initial scoping and screening evaluation results to be revised based on new information determined in the AMR evaluations.

Use of NEI-95-10 for Active/Passive Classification and Intended Function Determination of Structures and Components

Title 10 CFR 54.21(a)(1) states, in part, that structures and components subject to an AMR are those (i) that perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties, and (ii) that are not subject to replacement based on a qualified life or specified time period. A component or structure that performs an intended function without moving parts or without a change in configuration or properties is called "passive". A component or structure that performs all intended function(s) with moving parts and with a change in configuration or properties is called "active". NEI-95-10 (Reference 2.1) provides guidelines for classification of active and passive components during the license renewal screening activity. Appendix B of NEI-95-10 provides a listing of typical nuclear plant component types and a determination of whether each component is active or passive.

The Dresden and Quad Cities controlled plant component databases provide a component type identification field for each listed component. The component types identified in the plant component equipment databases and engineering drawings were compared with the component types listed in Appendix B of NEI-95-10. Results of that comparison were used during the screening of in-scope structures and components to determine the active/passive classification of components in-scope of license renewal.

Title 10 CFR 54.21(a)(1) requires that all long-lived passive structures and components that perform an intended function as described in 10 CFR 54.4 be subject to an AMR. For each such structure or component, the intended function(s) of the component or structure were documented for use during the AMR. A list of functions for passive structures and components was developed based on a similar listing of typical intended function for passive structures and components provided in NEI-95-10. The list of functions for passive structures and components was used during the structure and component screening activity to support identification of the intended functions of components that are subject to an AMR.

Scoping and Screening of Electrical Components Based on Electrical Spaces Approach for Aging Management Review

The spaces approach to aging management review (AMR) is based on areas where bounding environmental conditions are identified. The bounding environmental conditions are applied during AMR to evaluate the aging effects on all generic component groups that are located within the bounding area. Use of the spaces approach for AMR of electrical and I&C components eliminates the need to associate electrical and I&C components with specific systems that are within the scope of license renewal. Anticipated use of the spaces approach for AMR of electrical components was incorporated into the scoping and screening of Dresden and Quad Cities electrical and I&C components. As stated in Section 2.1.5.2, all electrical and I&C components classified as within the scope of the Rule, were evaluated as a consolidated component group.

10 CFR 54.21(a)(1) requires only that those in-scope structures and components which are passive and long lived are subject to an AMR. Based on this requirement, any in-scope component that is determined to be active can be categorized as a component not requiring an AMR. All electrical components assigned to this consolidated electrical and I&C component group were classified as in the scope of the Rule and were screened to determine whether they were subject to an AMR. Electrical and I&C components that were determined to be active were classified as not subject to an AMR. Electrical and I&C components that were determined to be passive were classified as subject to an AMR, and are listed in Section 2.5.1.

Treatment of Piping and Equipment Insulation During Scoping and Screening

Dresden and Quad Cities use thermal insulation and jacketing on piping and equipment for a variety of purposes. The insulation on hot piping in a system that is functionally out-of-scope for license renewal may provide the function of protecting nearby, in-scope SSCs from overheating. The insulation on hot piping in a system that is functionally in-scope for license renewal may serve only to improve process efficiency by retaining heat in the process fluid and may not be credited to perform any function that would bring it into scope of the Rule. Based on these observations, it is recognized that thermal insulation and jacketing on piping and equipment cannot readily be scoped against requirements of 10 CFR 54.4(a)(2) based simply on the plant system where it is used.

For each plant piping system, the plant specification that lists insulated pipe was reviewed to determine whether the system has insulation installed. For each piping system determined to have insulation installed, an insulation commodity component representing all of the insulation of that type within the system was identified and was initially classified as in-scope for license renewal. During aging management review of the insulation commodity group, further evaluation of the design purpose for the insulation requirement for each system confirmed or revised the in-scope classification of the system insulation components.

Treatment of External Environments During Aging Management Review

Passive, long-lived in-scope mechanical components were reviewed to determine if aging management was required for the component external surfaces. The system boundary drawings identify the systems within the scope of license renewal for the

Dresden and Quad Cities stations. Component external surfaces were evaluated using a 'commodity group' approach where all components from various station systems constructed of a like material in a like environment were evaluated as a commodity group. The like aging effects were determined and evaluated, and appropriate aging management activities identified.

The external surface aging mechanisms for passive, long-lived piping and mechanical system components were evaluated based on the various like materials of construction in like external environments. The aging management review results for those commodity groups whose material of construction and environment are evaluated in NUREG-1801 are presented in Tables 3.X.1 in each section in Chapter 3 of this application. The aging management review results for those commodity groups whose material of construction and environment are not evaluated in NUREG-1801 are presented in Tables 3.X.2 in each section in Chapter 3 of this application.

Commodity Groups

The scoping methodology utilized for Dresden and Quad Cities incorporated the use of commodity groups for some types of components:

- Mechanical system piping components that are uniquely identified in the controlled plant component database and on piping and instrument diagrams (P&IDs) (such as valves, heat changers, pumps, etc.) were evaluated as individual components. However, mechanical system piping components that are not uniquely identified were evaluated as part of a system-specific commodity group. For example, P&IDs typically depict but do not provide equipment part numbers (EPNs) for piping and fittings. When a system was determined to include in-scope commodity items, the commodity items for the system were evaluated and were entered as a representative commodity component into the LR database for the system. The representative commodity component was then screened to determine if it was subject to an AMR. During the AMR process, the system may be determined to have in-scope commodity components made of more than one material. For example, a single system may include carbon steel piping and stainless steel piping, both of which are in-scope. When such a determination was made, the LR database was modified to include multiple commodity component records corresponding to the different piping materials.

- Electrical and I&C components in electrical and I&C systems and some electrical and I&C components in mechanical systems and structures were evaluated as a consolidated electrical & I&C component group. Electrical components were identified and assigned to this consolidated electrical and I&C component group for scoping and screening evaluation. All components assigned to the consolidated electrical and I&C component group were initially identified as in-scope of the Rule. After their assignment to the consolidated electrical and I&C component group, some individual components were re-evaluated on the basis of their specific design function. If it was determined that the component did not perform an intended function as described in 10 CFR 54.4, the component was classified as outside the scope of the Rule. All components in the consolidated electrical and I&C component group that are in-scope for license renewal were then screened with respect to criterion 10 CFR 54.21(a)(1)(i) and classified as “active” or “passive.” In-scope electrical and I&C components categorized as “active” were excluded from AMR. In-scope electrical and I&C components categorized as “passive” were identified in the LR database as subject to an AMR.

- Structures and systems (mechanical, electrical and I&C) contain structural components. These structural elements and components are evaluated in commodity groupings. The commodity groupings were determined based on similarity of materials and component functions. For example, an in-scope building is comprised of several wall elements and may include multiple, but similar structural components such as equipment supports or doors. The scoping and screening evaluation has not identified and evaluated the multiple structural components on an individual basis. Rather, the evaluation grouped like-kind structural components as generic components for scoping and screening.

Complex Assemblies

No in-scope SSCs were evaluated as complex assemblies. Attendant passive components (e.g., cooling water piping, instrument lines and valves) of active components (such as the emergency diesel generators) are shown on the LR boundary diagrams for that system. Attendant passive components of active components were scoped and screened separately from the active component. Attendant passive components that support a system intended function were identified as in-scope of the Rule and were identified in the LR database as subject to an AMR.

Hypothetical Failures and Cascading

As described in Section 2.1.4, the Dresden and Quad Cities scoping methodology has required a review of applicable UFSAR sections and other CLB and design basis documents to identify the current licensing basis and design functions of each system or structure being evaluated. This review included only hypothetical failures described in the current licensing basis. The hypothethical failures for the system or structure being evaluated, were reviewed to determine if other SSCs were affected by the hypothetical failure.

During the system/structure level scoping process, systems or structures that provide support to the intended functions of the system/structure being evaluated were identified. The required supporting function was documented in the evaluation package for the supporting system or structure. For safety related intended functions, this process identified support systems down to a level necessary to provide for the satisfactory accomplishment of the safety related functions identified in 10 CFR 54.4(a)(1). This could result in “cascading” through several layers of support systems. However, at Dresden and Quad Cities, this review typically identified only support systems that were already in-scope. When a supporting system or structure was identified for an intended function that satisfies only criterion 10 CFR 54.4(a)(3), the scoping process did not require that the supporting function be classified as an intended function unless a requirement in a current licensing basis document explicitly identifies a requirement for the supporting function.

Multiple Functions and Heat Exchanger Intended Functions

The potential for multiple functions was considered in identifying the intended functions of structures and components. The LR database accommodates the documentation of multiple functions. For in-scope heat exchangers, both the pressure boundary and the heat transfer functions were considered, and where applicable, the heat transfer function, in addition to the pressure boundary function, was identified as a component intended function for consideration during the AMR process.

Table 2.1-1 Component Intended Functions

The tables below are the complete list of the component intended functions used in component screening.

Passive Intended Functions of Mechanical Components	
Component Intended Function	**Description**
Spray	Convert fluid flow into spray
Filter	Provide filtration
Fire Barrier	Provide fire rated barrier to confine or retard a fire from spreading to or from adjacent areas of the plant
Heat Transfer	Provide Heat Transfer
Leakage Boundary (Spatial)	Non-safety related piping component maintains mechanical and structural integrity to prevent spatial interactions that could cause failure of safety related SSCs
Pressure Boundary	Provide pressure boundary so that sufficient flow at adequate pressure is delivered, provide fission product barrier for reactor containment pressure boundary piping and components, or provide containment isolation for fission product retention
Containment, Holdup and Plateout	Provide post accident containment, plateout of iodine and hold-up (for radioactive decay) of iodine and noncondensable gases before release
Structural Integrity (Attached)	Non-safety related piping component maintains mechanical and structural integrity to provide structural support to attached safety related piping and components
Structural Support	Provide structural and or functional support to safety related equipment
Throttle	Provide flow restriction

Passive Intended Functions of Electrical and I&C Components	
Component Intended Function	**Description**
Electrical Continuity	Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current, or signals
Insulate	Insulate and support an electrical conductor

Table 2.1-1 Component Intended Functions (Continued)

Passive Intended Functions of Structures, Structural Components and Thermal Insulation	
Component Intended Function	**Description**
Absorb Neutrons	Absorb Neutrons
Direct Flow	Provide spray shield or curbs for directing flow
Expansion / Separation	Provide for thermal expansion and/or seismic separation
Fire Barrier	Provide fire rated barrier to confine or retard a fire from spreading to or from adjacent areas of the plant
Fission Product Barrier	Provide fission product barrier
Flood Barrier	Provide flood protection barrier for an internal and external event
Gaseous Release Path	Provide path for release of filtered and unfiltered gaseous discharge
Heat Sink	Provide heat sink during SBO or design basis accidents
HELB Shielding	Provide shielding against high energy line breaks
Insulating Characteristics	Control heat loss
Insulation Jacket Integrity	Prevent moisture absorption of insulation
Missile Barrier	Provide missile barrier for internally or externally generated events
Non-S/R Structural Support	Provide structural support to non-safety related components whose failure could prevent satisfactory accomplishment of any of the required safety related functions
Pipe Whip Restraint	Provide pipe whip restraint
Pressure Relief	Provide over-pressure protection
Shelter, Protection, Shielding	Provide shelter, protection, and radiation shielding
Shutdown Cooling Water	Provide source of cooling water for plant shutdown
Structural Pressure Barrier	Provide pressure boundary or essentially leak tight barrier to protect public health and safety in the event of any postulated design basis events
Structural Support	Provide structural and/or functional support to safety related equipment

Section 2.1 References (Scoping and Screening Methodology)

2.1 NEI-95-10, Industry Guideline for Implementing the Requirements of 10 CFR Part 54 – the License Renewal Rule.

2.2 Letter of March 15, 2002, from Christopher I. Grimes of the NRC to Alan Nelson or NEI, Subject: License Renewal Issue: Guidance on the Identification and Treatment of Structures, Systems, and Components Which Meet 10 CFR 54.4(a)(2).

2.3 Letter of March 1, 2002, from Christopher I. Grimes of the NRC to Alan Nelson of NEI, Subject: Proposed Staff Guidance on Scoping of Equipment Relied on to Meet the Requirements of the Station Blackout (SBO) Rule (10 CFR 50.63) for License Renewal (10 CFR 54.4(a)(3)).

2.4 NRC letter of February 19, 1987, from Harold R. Denton, Subject: Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46 (Generic Letter 87-02).

2.5 SAND96-0344, "Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations," Sandia National Laboratories, September 1996.

2.2 PLANT-LEVEL SCOPING RESULTS

Table 2.2-1, "Mechanical Systems Scoping Results," provides the results of the plant-level scoping for each of the mechanical systems at Dresden, Units 2 and 3 and Quad Cities, Units 1 and 2. Table 2.2-2, "Structures Scoping Results," provides the results of the plant-level scoping for each of the structures at Dresden and Quad Cities. Table 2.2-3, "Electrical and Instrumentation and Controls Systems Scoping Results," provides the results of the plant-level scoping for each of the electrical systems.

Systems and structures that are installed at both stations are included and marked appropriately as to whether they are in the scope of license renewal. Systems and structures which exist at both stations and are in the scope of license renewal for only one station are marked as in-scope “Yes” for one site and “No” for the other. Systems and structures, which only exist at one station, are marked in the tables as Dresden only or Quad Cities only, as appropriate.

Included in each of the tables with the mechanical system and structure names are references to the appropriate sections in the application that provide system and structure descriptions, system and structure intended functions, and identification of the component groups requiring aging management review. For electrical systems, these references are not provided because aging management review of electrical components and commodities was performed using a ”spaces” approach.

Table 2.2-1 Mechanical Systems Scoping Results

REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM			
Description	**Dresden In-Scope?**	**Quad Cities In-Scope?**	**Comments**
REACTOR VESSEL			
Reactor vessel (2.3.1.1)	Yes	Yes	
INTERNALS			
Reactor internals (2.3.1.2)	Yes	Yes	
Fuel assemblies (2.3.1.2.2)	Yes	Yes	
Control blades (2.3.1.2.3)	Yes	Yes	
REACTOR COOLANT SYSTEM			
Reactor recirculation system, recirculation flow control and MG sets (2.3.1.3.1)	Yes	Yes	
Reactor vessel head vent system (2.3.1.3.2)	Yes	Yes	
Nuclear boiler instrumentation system (2.3.1.3.3)	Yes	Yes	
Head spray system (Dresden only) (2.3.1.3.4)	Yes	N/A	The head spray system is not installed at Quad Cities.
Reactor coolant pressure boundary components in other systems (2.3.1.3.5)	Yes	Yes	The reactor coolant pressure boundary components of the following systems are evaluated with their assigned systems: • High pressure coolant injection system (2.3.2.1) • Core spray system (2.3.2.2) • Reactor core isolation cooling system (Quad Cities only) 2.3.2.4) • Isolation condenser (Dresden only) (2.3.2.5) • Residual heat removal system (Quad Cities only) (2.3.2.6) • Low pressure coolant injection system (Dresden only) (2.3.2.7) • Standby liquid control system (2.3.2.8) • Shutdown cooling system (Dresden only) (2.3.3.2) • Control rod drive hydraulic system (2.3.3.3) • Reactor water cleanup system (2.3.3.4) • Main steam system (2.3.4.1) • Feedwater system (2.3.4.2)

Table 2.2-1 Mechanical Systems Scoping Results (Continued)

ENGINEERED SAFETY FEATURES			
Description	**Dresden In-Scope?**	**Quad Cities In-Scope?**	**Comments**
High pressure coolant injection system (2.3.2.1)	Yes	Yes	
Core spray system (2.3.2.2)	Yes	Yes	
Containment isolation components and primary containment piping system (2.3.2.3)	Yes	Yes	The containment isolation components and primary containment piping system evaluation boundary consists of: • primary containment pressure instruments • suppression chamber to reactor building vacuum breaker lines • purge supply and exhaust penetrations (HVAC – Primary containment) • suppression chamber level instrumentation penetrations • LLRT test penetrations • containment isolation barriers from the following systems: o transverse incore probe system o floor and equipment drain collection system and liquid radwaste system o atmospheric containment air dilution (ACAD) system o service air system o instrument air system
Reactor core isolation cooling system (Quad Cities only) (2.3.2.4)	N/A	Yes	The RCIC system is not installed at Dresden.
Isolation condenser (Dresden only) (2.3.2.5)	Yes	N/A	The isolation condenser is not installed at Quad Cities.
Residual heat removal system (Quad Cities only) (2.3.2.6)	N/A	Yes	The RHR system is not installed at Dresden. The functions of this system are performed by a combination of LPCI (2.3.2.7) and shutdown cooling (2.3.3.2) at Dresden.
Low pressure coolant injection system (Dresden only) (2.3.2.7)	Yes	N/A	The LPCI system is not installed at Quad Cities. The functions of the LPCI system are performed by one of the RHR (2.3.2.6) modes at Quad Cities.
Standby liquid control system (2.3.2.8)	Yes	Yes	The standby liquid control system is included as an ESF in Section 6.0 of the UFSAR.
Standby gas treatment system (2.3.2.9)	Yes	Yes	
Automatic depressurization system (2.3.2.10)	Yes	Yes	The relief valves that perform the ADS function are included as components of the main steam system (2.3.4.1).
Anticipated transients without scram system (2.3.2.11)	Yes	Yes	The ATWS system is grouped under this heading because of similarities with other systems that are characterized as ESF systems. However, the ATWS system is not classified in the Dresden or the Quad Cities UFSARs as an ESF system.

Table 2.2-1 Mechanical Systems Scoping Results (Continued)

AUXILIARY SYSTEMS			
Description	**Dresden In-Scope?**	**Quad Cities In-Scope?**	**Comments**
Refueling equipment (2.3.3.1)	Yes	Yes	
Shutdown cooling system (Dresden only) (2.3.3.2)	Yes	N/A	Shutdown cooling system is not installed at Quad Cities. The functions of this system are performed by the RHR system (2.3.2.6) at Quad Cities.
Control rod drive hydraulic system (2.3.3.3)	Yes	Yes	
Reactor water cleanup system (2.3.3.4)	Yes	Yes	
Fire protection system (2.3.3.5)	Yes	Yes	The fire protection system contains fire dampers from the following systems: • HVAC-Recirc pump, MG set (Quad Cities only) • Battery room ventilation • HVAC – Auxiliary electric room and auxiliary computer room • UPS battery and computer room HVAC
Emergency diesel generator and auxiliaries (2.3.3.6)	Yes	Yes	
HVAC – Main control room (2.3.3.7)	Yes	Yes	
HVAC – Reactor building (2.3.3.8)	Yes	Yes	
ECCS corner room HVAC (2.3.3.9)	Yes	Yes	
Station blackout building HVAC (2.3.3.10)	Yes	Yes	
Station blackout system (diesels and auxiliaries) (2.3.3.11)	Yes	Yes	
Diesel generator cooling water system (2.3.3.12)	Yes	Yes	
Diesel fuel oil system (2.3.3.13)	Yes	Yes	
Process sampling system (2.3.3.14)	Yes	Yes	
Carbon dioxide system (2.3.3.15)	Yes	Yes	
Service water system (2.3.3.16)	Yes	Yes	
Reactor building closed cooling water system (2.3.3.17)	Yes	Yes	.
Turbine building closed cooling water system (2.3.3.18)	Yes	No	At Dresden only, the TBCCW system is credited in Appendix R evaluations by providing a flow path for other required equipment.
Demineralized water make-up system (2.3.3.19)	Yes	Yes	
Residual heat removal service water system (Quad Cities only) (2.3.3.20)	N/A	Yes	The functions of this system are performed by the containment cooling service water system (2.3.3.21) at Dresden.
Containment cooling service water system (Dresden only) (2.3.3.21)	Yes	N/A	The functions of this system are performed by the RHR service water system (2.3.3.20) at Quad Cities.

Table 2.2-1 Mechanical Systems Scoping Results (Continued)

AUXILIARY SYSTEMS			
Description	**Dresden In-Scope?**	**Quad Cities In-Scope?**	**Comments**
Ultimate heat sink (2.3.3.22)	Yes	Yes	
Fuel pool cooling and filter demineralizers system (2.3.3.23)	Yes	No	Because of differences in plant equipment lay out, some of the fuel pool cooling system piping at Dresden can potentially fail in a way to cause failure of near-by safety related equipment. A similar equipment layout does not exist at the Quad Cities plant.
Plant heating system (2.3.3.24)	Yes	Yes	
Containment atmosphere monitoring system (2.3.3.25)	Yes	Yes	
Nitrogen containment atmosphere dilution system (2.3.3.26)	Yes	Yes	
Drywell nitrogen inerting system (2.3.3.27)	Yes	Yes	
Safe shutdown makeup pump system (Quad Cities only) (2.3.3.28)	N/A	Yes	This system is not installed at Dresden.
Drywell and suppression chamber DP pumpback air system	No	No	
Atmospheric containment atmosphere dilution system	N/A	N/A	System is retired in place. Primary containment isolation valves and ESF Division I and II primary containment pressure and level indication instrumentation and associated manual isolation valves are included in the evaluation of the containment isolation components and primary containment piping system (2.3.2.3).
Floor and equipment drain collection system and liquid radwaste system	No	No	Primary containment isolation valves and components associated with drywell floor drain and equipment drain sump pump discharges are evaluated with the containment isolation components and primary containment piping system (2.3.2.3).
Solid radwaste	No	No	
Waste water treatment system	No	No	
Reactor building floor and equipment drain system	No	No	
Turbine building equipment drains	No	No	
Station hydrogen supply system	No	No	
Oxygen injection – hydrogen addition system	No	No	For Quad Cities only, the electro-chemical potential monitoring probes are evaluated with the nuclear boiler instrumentation system (2.3.1.3.3).
Sewage treatment	No	No	
Screen wash	No	No	At Dresden only, portions of the screen wash system are evaluated with the ultimate heat sink system (2.3.3.22).
Well water	No	No	
Laundry	No	No	
Cooling lake level control (Lift pumps and weir gates) (Dresden only)	No	N/A	This system is not installed at Quad Cities.
Circulating water system	No	No	At Quad Cities only, portions of the circulating water system ice melting line are evaluated with the ultimate heat sink system (2.3.3.22).
Spray canal lift station and blowdown piping (Quad Cities only)	N/A	No	This system is not installed at Dresden.

Table 2.2-1 Mechanical Systems Scoping Results (Continued)

AUXILIARY SYSTEMS			
Description	**Dresden In Scope?**	**Quad Cities In Scope?**	**Comments**
Hypochlorite (Chemical addition)	No	No	
Service air and emergency breathing air	No	No	Class I service air components associated with drywell penetrations are evaluated with the containment isolation components and primary containment piping system (2.3.2.3).
Instrument air and drywell pneumatic supply system	No	No	Primary containment isolation valves and associated test and drain valves are evaluated with the containment isolation components and primary containment piping system (2.3.2.3).
Corrosion test loop (Dresden only)	No	N/A	The system is retired at Dresden and was not installed at Quad Cities.
Off gas system	No	No	
Condensate demineralizer	No	No	
Acid and caustic supply	No	N/A	This system is not installed at Quad Cities.
HVAC – Primary containment	No	No	Primary containment isolation valves and associated piping are evaluated with the containment isolation components and primary containment piping system (2.3.2.3).
HVAC – Radwaste building	No	No	
HVAC – Tech support center	No	No	
HVAC – West turbine building	No	No	
HVAC – Recirc pump, MG set	No	No	For Quad Cities only, fire dampers are evaluated with the fire protection system (2.3.3.5).
HVAC – Feedwater pump	No	No	
HVAC – East turbine building	No	No	
Battery room ventilation	No	No	Fire dampers are evaluated with the fire protection system (2.3.3.5).
HVAC – Auxiliary electric room and auxiliary computer room	No	No	Fire dampers are evaluated with the fire protection system (2.3.3.5)`.
Off Gas recombiner room ventilation	No	No	
HVAC – High radiation sampling system	No	No	
Safety valve test system (Dresden only)	No	N/A	This system is not installed at Quad Cities.
Counting room HVAC	N/A	No	HVAC for the counting room at Dresden is supplied by the Chemistry Building HVAC covered under Outbuilding HVAC.
Guardhouse HVAC	No	No	
Dresden Unit 1 gaseous monitoring (Dresden only)	No	N/A	This system is not installed at Quad Cities.
Outbuilding HVAC	No	No	
Service building HVAC	No	No	
UPS battery and computer room HVAC	No	No	Fire dampers are evaluated with the fire protection system (2.3.3.5).
Seismic monitoring	No	No	

Table 2.2-1 Mechanical Systems Scoping Results (Continued)

AUXILIARY SYSTEMS			
Description	**Dresden In-Scope?**	**Quad Cities In-Scope?**	**Comments**
Security diesel, distribution and auxiliaries	No	No	
High radiation sampling system	No	No	Isolation valves for isolating standby gas treatment system from the high radiation sampling system are evaluated with the standby gas treatment system (2.3.2.9). For Dresden only, isolation valves for isolating low pressure coolant injection system from the high radiation sampling system are evaluated with the low pressure coolant injection system (2.3.2.7). For Quad Cities only, isolation valves for isolating core spray system from the high radiation sampling system are evaluated with the core spray system (2.3.2.2). Also, isolation valves for isolating drywell nitrogen inerting system from the high radiation sampling system are evaluated with the drywell nitrogen inerting system (2.3.3.27).
Zinc injection system	No	No	
Fish hatchery system (Quad Cities only)	N/A	No	This system is not installed at Dresden.
Natural gas (Quad Cities only)	N/A	No	This system is not installed at Dresden.

Table 2.2-1 Mechanical Systems Scoping Results (Continued)

STEAM AND POWER CONVERSION			
Description	**Dresden In-Scope?**	**Quad Cities In-Scope?**	**Comments**
Main steam system (2.3.4.1)	Yes	Yes	
Feedwater system (2.3.4.2)	Yes	Yes	
Condensate and condensate storage system (2.3.4.3)	Yes	Yes	
Main condenser (2.3.4.4)	Yes	Yes	
Main turbine and auxiliary systems (2.3.4.5)	Yes	Yes	
Turbine oil system (2.3.4.6)	No	Yes	At Quad Cities only, portions of the turbine oil system can spatially interact with safety related components located in the same area.
Main generator and auxiliaries (2.3.4.7)	No	Yes	At Quad Cities only, portions of the stator cooling water system can spatially interact with safety related components in the same area.
Extraction steam system	No	No	
Feedwater heater drains and vents	No	No	

Table 2.2-2 Structures Scoping Results

Structures Description	Dresden In-Scope?	Quad Cities In-Scope?	Comments
Primary containment (2.4.1)	Yes	Yes	
Reactor building (secondary containment) (2.4.2)	Yes	Yes	
Main control room and auxiliary electric equipment room (2.4.3)	Yes	Yes	
Turbine buildings (2.4.4)	Yes	Yes	
Dresden Unit 2/3 Diesel generator and HPCI building (Dresden only) (2.4.5)	Yes	N/A	This structure exists only at Dresden. The Quad Cities diesel generator room performs these structure functions at Quad Cities.
Quad Cities diesel generator room (2.4.5)	N/A	Yes	This structure exists only at Quad Cities. The Dresden diesel generator room and HPCI building performs these structure functions at Dresden.
Station blackout building (2.4.6)	Yes	Yes	
Isolation condenser pump house (Dresden only) (2.4.7)	Yes	N/A	This structure does not exist at Quad Cities.
Dresden Units 2 and 3 make up demineralizer building (2.4.8)	Yes	N/A	This structure does not exist at Quad Cities.
Radwaste floor drain surge tank and foundation (2.4.9)	Yes	Yes	
Diesel oil storage tanks foundation (2.4.10)	Yes	Yes	
Contaminated condensate storage tanks foundation (2.4.10)	Yes	Yes	
Crib house (2.4.11)	Yes	Yes	
Dresden Unit 1 crib house (Dresden only) (2.4.12)	Yes	N/A	This structure does not exist at Quad Cities.
Equipment access building (Quad Cities only) (2.4.2)	N/A	Yes	This structure does not exist at Dresden.
Station chimney (2.4.13)	Yes	Yes	
Cranes and hoists (2.4.14)	Yes	Yes	
Fuel oil pump house and oil storage tank foundation (Dresden only)	No	N/A	This structure does not exist at Quad Cities.
Miscellaneous administrative buildings	No	No	
Miscellaneous yard structures	No	No	
Miscellaneous radwaste buildings	No	No	
Meteorological tower	No	No	
Miscellaneous river water structures	No	No	
Miscellaneous Dresden Unit 1 structures (Dresden only)	No	N/A	These structures do not exist at Quad Cities.
Miscellaneous transmission and distribution structures	No	No	

Table 2.2-3 Electrical and Instrumentation and Controls Systems Scoping Results

Description	Dresden In-Scope?	Quad Cities In-Scope?	Comments
Reactor protection system	Yes	Yes	
Reactor manual control system	Yes	Yes	
Alternate rod insertion	Yes	Yes	
Rod worth minimizer	Yes	Yes	
Intermediate range monitors	Yes	Yes	
Source range monitors	Yes	Yes	
Average power range monitors	Yes	Yes	
Local power range monitors	Yes	Yes	
Control room panels and annunciators	Yes	Yes	
Local panels and racks for multiple systems (Includes panels, fuses, relays, breakers, etc)	Yes	Yes	
Process radiation monitoring	Yes	Yes	
Drywell area temperature monitoring	Yes	Yes	
4160 V switchgear and distribution	Yes	Yes	
480 V motor control centers, transformer switchgear and distribution	Yes	Yes	
DC emergency lighting	Yes	Yes	
Essential service bus	Yes	Yes	
120/240 VAC instrument bus	Yes	Yes	
250 VDC system	Yes	Yes	
125 VDC system	Yes	Yes	
Communications and evacuation alarm	Yes	Yes	
Transformers (Main, Unit, Reserve and Standby)	Yes	Yes	
138 KV & 345 KV distribution	Yes	Yes	
Area leak detection system (Quad Cities only)	N/A	No	This system is not installed at Dresden.
SRM and IRM drive control system	No	No	
Oscillation power range monitor	No	No	Scram trips generated by the oscillation power range monitor have not been enabled.
Rod position information system	No	No	
Rod block monitoring system	No	No	
Feedwater level control	No	No	
Traversing incore probes	No	No	Primary containment isolation valves and associated tubing assigned to the traversing incore probe system are evaluated with the primary containment and suppression pool piping system (2.3.2.3).
Area radiation monitoring	No	No	
24/48 VDC system	No	No	
12 & 34 KV distribution (Dresden only)	No	N/A	This system does not exist at Quad Cities.
13.8 KV distribution	N/A	No	This system does not exist at Dresden.
120 VAC distribution, lighting and receptacles	No	No	

Table 2.2-3 Electrical and Instrumentation and Controls Systems Scoping Results (Continued)

Description	Dresden In-Scope?	Quad Cities In-Scope?	Comments
Automatic dispatch (EGC)	No	No	
Cathodic protection	No	No	
Plant computers	No	No	
Grounding	No	No	

2.3 SCOPING AND SCREENING RESULTS: MECHANICAL SYSTEMS

The scoping and screening results for mechanical systems consist of lists of components and component groups that require aging management review, arranged by system. Brief descriptions of mechanical systems within the scope of license renewal are provided as background information. Mechanical system intended functions are provided for in-scope systems. For each in-scope system, components or component groups requiring an aging management review are provided.

Specifically, this section provides the results of the scoping and screening process for mechanical systems including:

- A general description of the system, its purpose, and the system evaluation boundary,
- A reference to the applicable UFSAR section(s),
- A reference to the applicable license renewal boundary diagrams,
- The system intended functions,
- A listing of mechanical components that are subject to an aging management review, associated component intended functions, and a reference to the Section 3 Table line-item containing the aging management review results.

The License Renewal Application is a joint application and discussions are applicable to both Dresden, Units 2 and 3 and Quad Cities, Units 1 and 2 unless otherwise noted. Clear statements (such as Dresden only or Quad Cities only) are made if discussions are applicable to only one station.

The mechanical scoping and screening results are provided in four subsections:

- Reactor vessel, internals, and reactor coolant system
- Engineered safety features
- Auxiliary systems
- Steam and power conversion systems

2.3.1 Reactor Vessel, Internals, and Reactor Coolant System

This section of the application addresses scoping and screening results for the following:

- Reactor Vessel
- Internals
- Reactor Coolant System

The reactor coolant system, as evaluated for license renewal, is comprised of the reactor recirculation system, the reactor vessel head vents, the nuclear boiler instrumentation system, the head spray system (Dresden only), and reactor coolant pressure boundary components for other systems connected to the reactor vessel.

2.3.1.1 Reactor Vessel

System Purpose

The reactor vessel contains the reactor core, the reactor internals, and the reactor core coolant-moderator. It serves as a high-integrity barrier against leakage of radioactive materials to the drywell.

System Operation

The reactor vessel is a vertical, cylindrical pressure vessel with hemispherical heads. The cylindrical shell and bottom hemispherical head of the reactor vessel are of welded construction and are fabricated of low alloy steel plate. The removable top head is attached to the cylindrical shell flange by bolting. The major safety function for the reactor vessel is to provide a radioactive material barrier. The vessel also provides a floodable core volume, contains the moderator, and provides support for the reactor vessel internals.

System Evaluation Boundary

The evaluation boundary for the reactor vessel includes the vessel shell and flange, top head and flange, bottom head, vessel closure studs and nuts, vessel nozzles (recirculation, main steam, feedwater and others), nozzle safe ends, vessel penetrations (CRD stub tubes, in-core instrument housings and others), vessel skirt, vessel shell course welds and vessel attachment welds. At both Dresden and Quad Cities the control rod drive return line nozzle has been capped. At Dresden and Quad Cities the vessel flange leak detection line is part of the Nuclear Boiler Instrumentation System and has been evaluated with that system.

UFSAR References

Dresden Station UFSAR Section(s): 5.3

Quad Cities Station UFSAR Section(s): 5.3

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-FSAR-3.9

Quad Cities Station:
LR-QDC-FSAR-3.9

System Intended Functions

Pressure boundary – maintains the integrity of the reactor coolant pressure boundary.

Containment – provides a fission product containment barrier.

Physical support – provides vertical and horizontal support for the core and other reactor vessel internals.

Core cooling – The reactor vessel, together with the reactor vessel internals, provides a means to distribute coolant to the fuel assemblies located in the core and provides a floodable volume to at least two-thirds core height following design basis accidents.

Component Groups Requiring Aging Management Review

Table 2.3.1-1 Component Groups Requiring Aging Management Review - Reactor Vessel

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.1.1.12
Nozzle Safe Ends	Pressure Boundary	3.1.1.1, 3.1.1.15, 3.1.2.60
Nozzles	Pressure Boundary	3.1.1.1, 3.1.1.3, 3.1.1.13, 3.1.2.4, 3.1.2.16, 3.1.2.17,
Penetrations • Bottom Head Drain ▪ Control Rod Drive Stub Tubes ▪ Incore Instrument Housings ▪ Instrumentation and Jet Pump Instrumentation ▪ Standby Liquid Control	Pressure Boundary	3.1.1.1, 3.1.1.16, 3.1.2.58, 3.1.2.62, 3.1.2.63
Penetrations (Control Rod Drive Stub Tubes)	Structural Support	3.1.1.16

Table 2.3.1-1 Component Groups Requiring Aging Management Review - Reactor Vessel (Continued)

Component Group	Component Intended Function	Aging Management Ref
Support Skirts and Attachment Welds	Structural Support	3.1.1.1, 3.1.2.33
Top Head Enclosure (Closure Studs and Nuts)	Pressure Boundary	3.1.1.8
Top Head Enclosure (Head Flanges)	Pressure Boundary	3.1.1.1, 3.1.2.36
Top Head Enclosure (Top Heads and Nozzles)	Pressure Boundary	3.1.2.4, 3.1.2.37, 3.1.2.59
Vessel Bottom Heads	Pressure Boundary	3.1.1.1, 3.1.2.54
Vessel Shell Attachment Welds	Structural Support	3.1.1.14
Vessel Shells ▪ Belt Line Welds ▪ Flange ▪ Intermediate Beltline Shell ▪ Intermediate Nozzle Shell ▪ Lower Shell ▪ Upper Shell	Pressure Boundary	3.1.1.1, 3.1.1.3, 3.1.1.4, 3.1.2.4, 3.1.2.55, 3.1.2.56, 3.1.2.57, 3.1.2.61

Aging management review results for the reactor vessel are provided in Section 3.1.

2.3.1.2 Internals

The discussion of Internals is divided into three parts, reactor internals, fuel assemblies, and control blades. Scoping and Screening results are provided separately for each part.

2.3.1.2.1 Reactor Internals

System Purpose

Reactor internals are installed to properly distribute the flow of coolant delivered to the vessel, to locate and support the fuel assemblies (evaluated separately) and control blades (evaluated separately), and to provide an inner volume containing the core that can be flooded following a break in the nuclear system process barrier external to the reactor vessel.

System Operation

The shroud is a stainless steel cylinder which surrounds the reactor core and provides a barrier to separate the upward flow of the coolant through the reactor core from the downward recirculation flow. Bolted on top of the shroud is the steam separator assembly which forms the top of the core discharge plenum. This provides a mixing chamber before the steam-water mixture enters the steam separator. The recirculation outlet and inlet plenum are separated by the baffle plate (part of the shroud support structure) joining the bottom of the shroud to the vessel wall. The jet pump diffuser sits on and is welded to the baffle plate, making the jet pump diffuser section an integral part of the baffle plate. The baffle plate supports carry all the vertical weight of the shroud, steam separator and dryer assembly, top guide and bottom core plate (core grids), peripheral fuel assemblies, and jet pump components carried on the shroud. The control rod guide tubes extend up from the control rod drive housing through holes in the core plate. Each tube is designed as a lateral guide for the control rod and as the vertical support for the fuel support piece which holds the four fuel assemblies surrounding the control rod.

System Evaluation Boundary

The evaluation boundary for the reactor internals includes all components that are inside the reactor vessel except the fuel assemblies and the control blades, both of which are short-lived components and evaluated separately. This includes the major components listed above plus additional components such as the feedwater spargers, the core spray spargers, the standby liquid control system sparger, the jet pumps and jet pump flow sensing lines, incore nuclear instrumentation tubes, and reactor internals modification/repair hardware.

UFSAR References

Dresden Station UFSAR Section(s): 3.9.5

Quad Cities Station UFSAR Section(s): 3.9.5

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-FSAR-3.9

Quad Cities Station:
LR-QDC-FSAR-3.9

System Intended Functions

Reactivity control – The control rod drive mechanisms insert negative reactivity for normal shutdown and for mitigation of operational transients and accidents. Reactor vessel internals, not directly involved with reactivity insertion, support reactivity insertion by maintaining appropriate geometry to permit proper functioning of the control rod drive mechanism. Standby liquid control system flow supports an alternate method for reactivity control.

Core cooling – distribute emergency core cooling system flow to the core and maintain coolable core geometry.

Support safety-related function(s) – Reactor vessel internals which do not perform a safety related function are required not to fail in a way that would cause a safety related function to fail.

Physical support – provides vertical and horizontal support for the core and other reactor vessel internals.

Component Groups Requiring Aging Management Review

Table 2.3.1-2 Component Groups Requiring Aging Management Review - Reactor Internals

Component Group	Component Intended Function	Aging Management Ref
Access Hole Covers (Mechanical)	Pressure Boundary	3.1.1.18
Access Hole Covers (Welded) (Dresden only)	Pressure Boundary	3.1.1.18
Control Rod Drive Housings	Pressure Boundary	3.1.1.17
Control Rod Drive Housings	Structural Support	3.1.1.17
Control Rod Guide Tubes	Structural Support	3.1.1.17
Core Plates	Structural Support	3.1.1.1
Core Plates and Bolts	Structural Support	3.1.1.17
Core Shrouds (Upper, Central, Lower)	Structural Support	3.1.1.17

Table 2.3.1-2 Component Groups Requiring Aging Management Review - Reactor Internals (Continued)

Component Group	Component Intended Function	Aging Management Ref
Core Spray Lines and Spargers	Pressure Boundary	3.1.1.1, 3.1.1.17
Core Spray Lines and Spargers	Spray	3.1.1.1, 3.1.1.17
Core Spray Lines and Spargers	Structural Support	3.1.1.1, 3.1.1.17
Incore Instrumentation Dry Tubes and Guide Tubes	Pressure Boundary	3.1.1.1, 3.1.1.17
Jet Pump Assemblies (Does not include Sensing Lines)	Pressure Boundary	3.1.1.1, 3.1.1.17, 3.1.1.19
Jet Pump Assemblies (Does not include Sensing Lines)	Structural Support	3.1.1.1, 3.1.1.17
Orificed Fuel Support Pieces	Structural Support	3.1.1.17, 3.1.1.19
Orificed Fuel Supports	Structural Support	3.1.1.1
Reactor Internals Modification/Repair Hardware ▪ Core Spray Clamp ▪ Jet Pump Riser Clamp (Quad Cities only) ▪ Jet Pump Riser Brace Clamp (Quad Cities only) ▪ Shroud Repair	Structural Support	3.1.1.17
Shroud Support Structures	Structural Support	3.1.1.17
Top Guides	Structural Support	3.1.1.1, 3.1.1.17

Aging management review results for the reactor internals are provided in Section 3.1.

2.3.1.2.2 Fuel Assemblies

System Purpose

The fuel assemblies provide a high integrity assembly of fissionable material that can be arranged in a critical array. The fuel assembly allows efficient heat transfer to the coolant, and maintains structural integrity and fission product barrier.

System Operation

Fuel assemblies contain the fissionable material that sustains a nuclear reaction when the reactor core is made critical. Each fuel assembly consist of fuel rods that contain uranium dioxide pellets, an upper and a lower tie plate, fuel rod spacers, a removable fuel channel, and mechanical fasteners. It may include design features such as water rods, for increased neutron moderation, and fuel pellets that contain burnable neutron absorbers.

System Evaluation Boundary

The evaluation boundary is the entire fuel assembly.

UFSAR References

Dresden Station UFSAR Section(s): 4.2

Quad Cities Station UFSAR Section(s): 4.2

License Renewal Boundary Diagram References

Dresden Station: None

Quad Cities Station: None

System Intended Functions

Fission product barrier - provides a cladding barrier to contain the fission gas released from the fuel throughout the design life of the fuel rod.

Component Groups Requiring Aging Management Review

Table 2.3.1-3 Component Groups Requiring Aging Management Review - Fuel Assemblies

Component Group	**Component Intended Function**	**Aging Management Ref**
None (Note 1)	Not Applicable	Not Applicable

Note 1: Fuel assemblies do not require aging management review because they are short-lived.

2.3.1.2.3 Control Blades

System Purpose

The control blades are used for flux shaping and for coarse reactivity control, such as startup, shutdown, and power changes. They are also used to compensate for reactivity changes caused by fuel and burnable poison depletion.

System Operation

The control blades are moveable cruciform-shaped neutron absorbers which can be inserted into the moderator region between fuel bundles to control neutron flux distribution and to shutdown the reactor.

System Evaluation Boundary

The evaluation boundary is the entire control blade assembly, including the individual rods containing neutron absorber, the cruciform control rod sheath, rollers, castings and the control rod velocity limiter.

UFSAR References

Dresden Station UFSAR Section(s): 4.6.2

Quad Cities Station UFSAR Section(s): 4.6.2

License Renewal Boundary Diagram References

Dresden Station: None

Quad Cities Station: None

System Intended Functions

Reactivity control – provides sufficient negative reactivity to shutdown the reactor from any normal or accident condition.

Component Groups Requiring Aging Management Review

Table 2.3.1-4 Component Groups Requiring Aging Management Review - Control Blades

Component Group	Component Intended Function	Aging Management Ref
None (Note 1)	Not Applicable	Not Applicable

Note 1: Control blades do not require aging management review because they are short-lived.

2.3.1.3 Reactor Coolant System

The reactor coolant system for BWRs as described in NUREG-1800 includes the reactor coolant recirculation system and portions of other systems connected to the pressure vessel extending to the first isolation valve outside of containment or to the first anchor point. For Dresden and Quad Cities, the reactor coolant system is comprised of the following plant systems:

- Reactor recirculation system, recirculation flow control and M/G sets
- Reactor vessel head vent system
- Nuclear boiler instrumentation system
- Head spray system (Dresden only)
- Reactor coolant pressure boundary components in other systems
    - High pressure coolant injection system
    - Core spray system
    - Reactor core isolation cooling system (Quad Cities only)
    - Isolation condenser (Dresden only)
    - Residual heat removal system (Quad Cities only)
    - Low pressure coolant injection system (Dresden only)
    - Standby liquid control system
    - Shutdown cooling system (Dresden only)
    - Control rod drive hydraulic system
    - Reactor water cleanup system
    - Main steam system
    - Feedwater system

2.3.1.3.1 Reactor Recirculation System, Recirculation Flow Control, and MG Sets

<u>System Purpose</u>

The reactor recirculation system provides a variable forced circulation of water through the reactor core, thereby achieving a higher specific power and load following capability. The recirculation flow control subsystem and the recirculation motor/generator (MG) sets provide variable frequency power and control to the recirculation pumps' variable speed motors.

<u>System Operation</u>

Cooling water is forced through the reactor core by the reactor recirculation system, which consists of two recirculation pump loops external to the reactor vessel and 20 jet pumps internal to the vessel (evaluated with reactor internals). Each external loop consists of a variable-speed, motor-driven recirculation pump, two motor-operated gate valves for pump isolation, piping, and required recirculation flow measurement and control devices. The two external recirculation loops supply high-pressure flow to piping systems, which connect ultimately to the jet pump nozzles. The two recirculation pumps are centrifugal units with mechanical shaft seals and are driven by variable-speed induction motors, which receive electrical power from variable frequency motor-generator sets. Pump speed control is through control of the variable speed generator of the associated MG set.

System Evaluation Boundary

The evaluation boundaries for this system include all recirculation system piping connected to the reactor vessel with associated valves, branch lines and instrumentation. The primary flow path for each recirculation loop begins at the suction piping interface with the vessel nozzle. It continues through the suction isolation valve, the pump, the discharge isolation valve and back through a ring header and risers to vessel penetrations. It does not include the jet pumps or associated piping and instrument sensing lines that are inside the reactor vessel. Also included in the evaluation boundary are the recirculation flow control instrumentation and the recirculation MG sets, fluid drive couplers and the MG set lube oil piping subsystem and oil coolers and their associated electrical controls.

UFSAR References

Dresden Station UFSAR Section(s): 5.4.1

Quad Cities Station UFSAR Section(s): 5.4.1

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-26-2, LR-DRE-M-357-2, and LR-DRE-M-174-2

Quad Cities Station:

LR-QDC-M-35-2, LR-QDC-M-35-4, LR-QDC-M-43, LR-QDC-M-77-2, LR-QDC-M-77-4, LR-QDC-M-77-6, LR-QDC-M-85, LR-QDC-M-1056-1, and LR-QDC-M-1061-1

System Intended Functions

Pressure boundary -- maintains integrity for the reactor coolant pressure boundary.

Flow path – provides an integral flow path for low pressure core injection (LPCI) flow into the reactor vessel. It also provides a flow path for establishing the shutdown cooling mode of operation.

Support ESF function(s) – provides signals and performs actions during a design basis loss of coolant accident for correct selection of the unbroken recirculation loop and closure of the recirculation system valves.

Credited in regulated event(s) – required to enable hot shutdown and cold shutdown during an Appendix R fire event and to provide trips of recirculation pumps to mitigate the ATWS event. The system also contains components that are relied upon for compliance with 10 CFR 50.49 (EQ).

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.1-5 Component Groups Requiring Aging Management Review - Reactor Recirculation System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting (includes flanges)	Pressure Boundary	3.1.1.1, 3.1.1.12, 3.1.2.1, 3.1.2.2
Dampeners (Quad Cities only)	Pressure Boundary	3.1.1.15, 3.1.2.7
Dampeners (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.1.2.7, 3.1.2.9
Filters/Strainers (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.1.2.3, 3.1.2.8, 3.1.2.12
Flow Elements	Pressure Boundary	3.1.1.15, 3.1.2.8
NSR Vents or Drains, Piping and Valves (attached support) (Dresden only)	Structural Integrity (attached)	3.1.2.18
Piping and Fittings	Pressure Boundary	3.1.1.1, 3.1.1.15
Piping and Fittings (spatial interaction)	Leakage Boundary (spatial)	3.1.2.3, 3.1.2.4, 3.1.2.7, 3.1.2.8, 3.1.2.21, 3.1.2.22, 3.1.2.23, 3.1.2.25, 3.1.2.26
Piping and Fittings (attached support)	Structural Integrity (attached)	3.1.2.3, 3.1.2.4, 3.1.2.7, 3.1.2.8, 3.1.2.22, 3.1.2.23, 3.1.2.24, 3.1.2.25, 3.1.2.26
Piping and Fittings (small bore)	Pressure Boundary	3.1.1.5, 3.1.2.8
Pumps	Pressure Boundary	3.1.1.1, 3.1.1.9, 3.1.1.15, 3.1.2.8
Restricting Orifices (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.1.2.3, 3.1.2.8, 3.1.2.27, 3.1.2.28, 3.1.2.29
Sight Glasses (attached support)	Structural Integrity (attached)	3.1.2.5, 3.1.2.6, 3.1.2.30, 3.1.2.32
Sight Glasses (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.1.2.5, 3.1.2.31, 3.1.2.32
Thermowells	Pressure Boundary	3.1.1.15, 3.1.2.8
Tubing	Pressure Boundary	3.1.2.41, 3.1.2.7, 3.1.2.8
Tubing (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.1.2.7, 3.1.2.8, 3.1.2.39, 3.1.2.40

Table 2.3.1-5 Component Groups Requiring Aging Management Review - Reactor Recirculation System (Continued)

Component Group	Component Intended Function	Aging Management Ref
Valves	Pressure Boundary	3.1.1.1, 3.1.1.9, 3.1.1.15, 3.1.2.7, 3.1.2.8, 3.1.2.45, 3.1.2.50
Valves (attached support)	Structural Integrity (attached)	3.1.2.48, 3.1.2.49, 3.1.2.51, 3.1.2.52, 3.1.2.53
Valves (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.1.2.3, 3.1.2.8, 3.1.2.45, 3.1.2.46, 3.1.2.47, 3.1.2.52, 3.1.2.53

Aging management review results for the reactor recirculation system are provided in Section 3.1.

2.3.1.3.2 Reactor Vessel Head Vent System

System Purpose

The reactor vessel head vent system removes non-condensable gases from the steam dome of the reactor vessel.

System Operation

A vent line penetrates the reactor vessel head (evaluated with the reactor vessel) to provide a flow path for steam and non-condensable gases during power and shutdown operations. During power operation, the reactor head vents from the reactor head to a main steam line (evaluated with main steam system) through the vent line and normally open in-line valve. The head vent line penetrates the steam line downstream of the venturi. The differential pressure across the venturi is the driving force for the flow of steam and non-condensable gases from the reactor head to the main steam line. During shutdown operations, manual valves or solenoid operated air valves vent the dome to the drywell equipment drain sump. The shutdown vent valves are in series requiring both valves to be open to provide a flow path. The vents are used after the vessel is depressurized. All vent valves are located near the reactor vessel head.

System Evaluation Boundary

The reactor vessel head vent system begins at the reactor vessel head flange. The main branch of reactor vessel head vent piping goes to the normally open reactor head vent valve and continues from there to the point where it ties back into a main steam line downstream of the main steam line venturi, but upstream of the inboard main steam isolation valve. All of this main branch of piping is part of the reactor coolant pressure boundary. A second branch of piping starts upstream of the reactor head vent valve and goes to normally closed dual isolation valves that are part of the reactor coolant pressure boundary and from there to the drywell equipment drain sump. The evaluation boundary includes the non-reactor coolant boundary piping and components between the normally closed isolation valves and the drywell equipment drain sump.

UFSAR References

Dresden Station UFSAR Section(s): None

Quad Cities Station UFSAR Section(s): None

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-26-1, LR-DRE-M-26-2, LR-DRE-M-357-1, LR-DRE-M-357-2

Quad Cities Station:

LR-QDC-M-35-1, LR-QDC-M-77-1

System Intended Functions

Pressure boundary - maintains the integrity of the reactor coolant pressure boundary.

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.1-6 Component Groups Requiring Aging Management Review - Reactor Vessel Head Vents

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting (includes flanges)	Pressure Boundary	3.1.1.12, 3.1.2.1, 3.1.2.2
NSR Vents or Drains, Piping and Valves (attached support)	Structural Integrity (attached)	3.1.2.18
Piping and Fittings	Pressure Boundary	3.1.1.11, 3.1.2.4
Piping and Fittings (small bore)	Pressure Boundary	3.1.1.5, 3.1.2.4
Sight Glasses (attached support) (Dresden only)	Structural Integrity (attached)	3.1.2.32
Tubing	Pressure Boundary	3.1.1.15, 3.1.2.4, 3.1.2.8, 3.1.2.41
Valves	Pressure Boundary	3.1.1.1 3.1.1.11, 3.1.1.15, 3.1.2.4, 3.1.2.8, 3.1.2.50

Aging management review results for the reactor vessel head vents are provided in Section 3.1.

2.3.1.3.3 Nuclear Boiler Instrumentation System

<u>System Purpose</u>

The nuclear boiler instrumentation (NBI) system provides trip signals to the reactor protection system, emergency core cooling systems, primary containment isolation logic, recirculation pump trip, and alternate rod insertion. It also provides the operator with indications of the reactor level, pressure, and temperature during normal and transient operation and for guidance in following emergency procedures and supporting post-accident operation. Parameters monitored by NBI are reactor vessel temperature, reactor vessel pressure, reactor vessel water level, reactor internal differential pressure, and reactor vessel flange leakage.

<u>System Operation</u>

Several reactor vessel penetrations (evaluated with reactor vessel) support the operation of the NBI system. Reactor vessel water level is measured by comparing the pressure from the variable level of water in the reactor vessel (“variable leg”) against the pressure from a reference column of water of a known height (“reference leg”). Both the variable leg and the reference leg experience the same steam overpressure. Steam from the reactor vessel enters the upper instrument tap of the reference leg and condenses in a condensing pot to keep the reference leg of the water instrument filled. Some water level instrument reference legs also receive continuous backfill from the reactor vessel water level instrument system (RVWLIS) backfill subsystem, which is also a part of the nuclear boiler instrumentation system. Reactor pressure is measured through the same piping that is used to measure the pressure in water level instrument reference legs. Thermocouples placed in important locations on the reactor vessel provide indication of reactor metal temperature. Reactor internal differential pressure is measured by instrumentation that compares pressure below the core in the standby liquid control injection line (evaluated with reactor internals) with the pressure above the core. Reactor vessel flange leakage is measured by using a pipe, valve and level switch arrangement that allows any leakage past the reactor head inner seal ring (evaluated with reactor vessel) to be monitored.

<u>System Evaluation Boundary</u>

The NBI system evaluation boundaries begin immediately downstream of associated reactor pressure vessel nozzles and terminate with the instrument(s) to which the sensing lines are piped. The RWCU bottom drain suction line is also included in the evaluation boundaries. These boundaries include all associated piping, condensing chambers, inline manual isolation globe valves, flow limiting check valves, local instrument racks, and mounted instruments for parameters monitored. All mechanical, electrical, and instrumentation and control components within evaluation boundaries are included. In addition, the NBI evaluation boundaries include all piping associated with the RVWLIS backfill subsystem, valves, filters, flow regulators and flow indicators in piping connecting with the control rod drive water header (evaluated with the control rod drive hydraulic system). The NBI system evaluation boundary also includes the thermocouples monitoring vessel temperature and the piping, valves, instruments and controls used for monitoring reactor vessel head leakage. For Quad Cities only, components associated with the electro-chemical potential probe in the hydrogen addition system are evaluated with the NBI system. Instruments associated with the main steam system, the ATWS system, and the feedwater control system are evaluated with their respective systems and not included in the boundaries of NBI system.

UFSAR References

Dresden Station UFSAR Section(s): 7.6.2

Quad Cities Station UFSAR Section(s): 7.6.2

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-26-1, LR-DRE-M-26-2, LR-DRE-M-26-3, LR-DRE-M-27, LR-DRE-M-34-1, LR-DRE-M-39, LR-DRE-M-357-1, LR-DRE-M-357-2, LR-DRE-M-357-3, LR-DRE-M-358, LR-DRE-M-365-1, and LR-DRE-M-369

Quad Cities Station:

LR-QDC-CID-50, LR-QDC-CID-89, LR-QDC-M-35-1, LR-QDC-M-35-5, LR- QDC -M-43, LR-QDC-M-77-1, LR-QDC-M-77-5, LR- QDC -M-85

System Intended Functions

Pressure boundary - maintains the integrity of the reactor coolant pressure boundary.

Support ESF function(s) – provides trip and initiation signals to the reactor protection system, emergency core cooling systems, containment isolation logic and to non-safety systems.

Credited in regulated event(s) – provides trip and initiation signals and process information and indications credited in mitigation of the Appendix R fire, ATWS, and SBO events. The system also contains components that are relied upon for compliance with 10 CFR 50.49, (EQ).

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.1-7 Component Groups Requiring Aging Management Review - Nuclear Boiler Instrumentation

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting (includes flanges)	Pressure Boundary	3.1.1.12, 3.1.2.1, 3.1.2.2
Dampeners (Quad Cities only)	Pressure Boundary	3.1.1.15, 3.1.2.7
Filters/Strainers (spatial interaction) (Dresden only)	Leakage Boundary (spatial)	3.1.2.13, 3.1.2.7, 3.1.2.8
NSR Vents or Drains, Piping and Valves (attached support) (Dresden only)	Structural Integrity (attached)	3.1.2.18
Pipes	Pressure Boundary	3.1.1.6, 3.1.2.3, 3.1.2.4, 3.1.2.7, 3.1.2.8, 3.1.2.19
Piping and Fittings (attached support)	Structural Integrity (attached)	3.1.2.3, 3.1.2.4, 3.1.2.7, 3.1.2.8, 3.1.2.23, 3.1.2.24
Piping and Fittings (spatial interaction) (Dresden only)	Leakage Boundary (spatial)	3.1.2.7, 3.1.2.8, 3.1.2.24
Piping and Fittings (Quad Cities only)	Pressure Boundary	3.1.1.11, 3.1.1.15, 3.1.2.3, 3.1.2.4, 3.1.2.7, 3.1.2.8, 3.1.2.25, 3.1.2.26
Piping and Fittings (small bore)	Pressure Boundary	3.1.1.5, 3.1.2.3, 3.1.2.4, 3.1.2.7, 3.1.2.8
Tanks	Pressure Boundary	3.1.1.15, 3.1.2.8
Thermowells	Pressure Boundary	3.1.1.15, 3.1.2.8
Tubing	Pressure Boundary	3.1.1.15, 3.1.2.7, 3.1.2.8
Tubing (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.1.2.7, 3.1.2.8, 3.1.2.38
Tubing (attached support) (Quad Cities only)	Structural Integrity (attached)	3.1.2.7, 3.1.2.8, 3.1.2.38, 3.1.2.42
Valves	Pressure Boundary	3.1.1.6, 3.1.1.15, 3.1.2.3, 3.1.2.4, 3.1.2.7, 3.1.2.8, 3.1.2.44, 3.1.2.52, 3.1.2.53
Valves (spatial interaction)	Leakage Boundary (spatial)	3.1.2.8, 3.1.2.49, 3.1.2.52, 3.1.2.53
Valves (attached support)	Structural Integrity (attached)	3.1.2.7, 3.1.2.49

Aging management review results for the nuclear boiler instrumentation are provided in Section 3.1.

2.3.1.3.4 Head Spray System (Dresden only)

System Purpose

The head spray system is used to collapse the steam bubble during vessel flooding, to cool the reactor vessel head, and to condense the steam in the vessel while the reactor is in the shutdown mode of operation. The head spray system may also be used for cold hydrostatic testing.

System Operation

The head spray system consists of the head spray line which interfaces with control rod drive (CRD) hydraulic piping (evaluated with the CRD hydraulic system) and associated valves. The head spray system uses the control rod drive pumps (evaluated with the CRD hydraulic system) to deliver condensate to the area inside the vessel head, through the head spray nozzle (evaluated with reactor vessel) and is discharged through the head spray element (evaluated with reactor internals). One or both CRD pumps must be operating and a normally locked closed manual valve in the CRD hydraulic system must be open. Condensate flows from the CRD pump(s) through a head spray flow element, the head spray flow control valve, a motor-operated outboard containment isolation valve, a drywell penetration and an inboard containment isolation check valve, to the head spray nozzle, and into the vessel.

System Evaluation Boundary

The head spray system begins downstream of a normally locked closed manual valve in the CRD hydraulic system. It includes a flow element that monitors flow from the CRD hydraulic system to the head spray line, a motor operated outboard containment isolation valve, an inboard containment isolation check valve, piping connection flanges and terminates at the piping interface with the vessel head spray nozzle. Branching vent and drain lines with dual manual isolation valves are also included in the evaluation boundary. The drywell mechanical penetration for the head spray line is evaluated with the primary containment and suppression chamber piping system.

UFSAR References

Dresden Station UFSAR Section(s): 5.4.15

Quad Cities Station UFSAR Section(s): Not Applicable

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-26-1 and LR-DRE-M-357-1

Quad Cities Station: Not Applicable

System Intended Functions

Pressure boundary - maintains the integrity of the reactor coolant pressure boundary.

Primary containment isolation – provides containment isolation for those portions of the system that interface with the primary containment.

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.1-8 Component Groups Requiring Aging Management Review - Head Spray System (Dresden only)

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting (Dresden only)(includes flanges)	Pressure Boundary	3.1.1.12, 3.1.2.1, 3.1.2.2
Flow Elements (attached support) (Dresden only)	Structural Integrity (attached)	3.1.2.3, 3.1.2.14
NSR Vents or Drains, Piping and Valves (attached support) (Dresden only)	Structural Integrity (attached)	3.1.2.18
Piping and Fittings (Dresden only)	Pressure Boundary	3.1.1.15, 3.1.2.7, 3.1.2.8
Piping and Fittings (attached support) (Dresden only)	Structural Integrity (attached)	3.1.2.3, 3.1.2.20
Piping and Fittings (small bore) (Dresden only)	Pressure Boundary	3.1.1.5, 3.1.2.7, 3.1.2.8
Valves (Dresden only)	Pressure Boundary	3.1.1.15, 3.1.2.7, 3.1.2.8
Valves (attached support) (Dresden only)	Structural Integrity (attached)	3.1.2.3, 3.1.2.43

Aging management review results for the head spray system are provided in Section 3.1.

2.3.1.3.5 Reactor Coolant Pressure Boundary Components in Other Systems

In addition to the systems and components described in preceding subsections, reactor coolant pressure boundary components included with other plant systems are evaluated in NUREG-1800 as part of the reactor vessel, internals and reactor coolant system. In Section 2.3, the components requiring aging management review, that have reactor coolant pressure boundary functions have been maintained in the plant system to which they are normally assigned, rather than grouped with other reactor coolant pressure boundary components in the reactor vessel, internals and reactor coolant system. Table 2.3.1-9 "Application sections where additional reactor coolant pressure boundary components are evaluated" presents a list of plant systems having reactor coolant pressure boundary components evaluated in NUREG-1800 as part of the reactor vessel, internals and reactor coolant system.

For each of these systems, applicable system descriptions, UFSAR references, license renewal boundary diagram references, system intended functions, and complete listings of component groups requiring aging management review are presented in the application section indicated in the Table 2.3.1-9.

Aging management review results for these reactor coolant pressure boundary components are presented in Section 3.1 in a similar manner to the presentation in NUREG-1801.

Table 2.3.1-9 Application sections where additional reactor coolant pressure boundary components are evaluated.

System Name	Other Application Section That Contain Reactor Coolant Pressure Boundary Components
High Pressure Coolant Injection System	2.3.2.1
Core Spray System	2.3.2.2
Reactor Core Isolation Cooling System (Quad Cities only)	2.3.2.4
Isolation Condenser (Dresden only)	2.3.2.5
Residual Heat Removal System (Quad Cities only)	2.3.2.6
Low Pressure Coolant Injection System (Dresden only)	2.3.2.7
Standby Liquid Control System	2.3.2.8
Shutdown Cooling System (Dresden only)	2.3.3.2
Control Rod Drive Hydraulic System	2.3.3.3
Reactor Water Cleanup System	2.3.3.4
Main Steam System	2.3.4.1
Feedwater System	2.3.4.2

2.3.2 Engineered Safety Features Systems

This section of the application addresses scoping and screening results for the following systems:

- High pressure coolant injection (HPCI) system
- Core spray (CS) system
- Containment isolation components and primary containment piping system
- Reactor core isolation cooling (RCIC) system (Quad Cities only)
- Isolation condenser (Dresden only)
- Residual heat removal (RHR) system (Quad Cities only)
- Low pressure coolant injection (LPCI) system (Dresden only)
- Standby liquid control (SLC) system
- Standby gas treatment (SBGT) system
- Automatic depressurization system (ADS)
- Anticipated transient without scram (ATWS) system

Note: The ATWS system is not classified in the Dresden or Quad Cities UFSAR as an ESF. However, the ATWS system is evaluated in this section because of similarities with other systems that are characterized as ESF systems.

2.3.2.1 High Pressure Coolant Injection System

System Purpose

The high pressure coolant injection (HPCI) system ensures that adequate core cooling takes place for all break sizes less than those sizes for which the low pressure coolant injection or core spray subsystems can adequately protect the core. Operation of the HPCI system in the emergency mode is completely independent of ac power.

System Operation

The HPCI system consists of a steam turbine driving a multi-stage high-pressure main pump and a gear driven single-stage booster pump, piping, auxiliary support systems, and instrumentation. The turbine is driven by nuclear steam and exhausts to the suppression chamber (evaluated with the primary containment structure). The preferred water source to the HPCI booster pump suction is supplied from the condensate storage system (evaluated with the condensate and condensate storage system), with a backup source from the suppression chamber. Water from the HPCI main pump is delivered to the reactor vessel (evaluated with the reactor vessel) through the “B” feedwater line (evaluated with the feedwater system) and distributed within the reactor vessel through the feedwater sparger (evaluated with reactor internals). The system is equipped with a test line to the condensate storage system to permit functional testing and a minimum flow bypass line to the suppression chamber for pump protection.

System Evaluation Boundary

The HPCI system evaluation boundary for water injection begins with the HPCI suction lines from the condensate storage system and the suppression chamber and continues through the HPCI booster pump and HPCI main pump. The discharge path runs from the output side of the HPCI main pump to the “B” feedwater line connection outside the primary containment. Included are all piping and components that feed the HPCI booster pump and the HPCI main pump. The discharge path also includes a minimum flow line and a flow test return line. The HPCI and RCIC systems at Quad Cities share a common flow test return line. This common line is evaluated for both systems with HPCI. The HPCI steam supply to the turbine runs directly from the reactor vessel at Dresden. At Quad Cities, the steam supply is provided from “B” main steam line on unit 1 and the “C” main steam line on unit 2. The HPCI turbine exhausts to the suppression chamber. Auxiliary subsystems include gland seal, drain pots, turbine oil, and turbine cooling water. All associated piping, components and instrumentation, contained within flow paths and subsystems described above are included in the HPCI system evaluation boundary. The HPCI room cooler maintains the HPCI room below the maximum equipment temperature limits. The HPCI room cooler has been evaluated with the ECCS corner room HVAC system.

UFSAR References

Dresden Station UFSAR Section(s): 6.3.1, 6.3.2

Quad Cities Station UFSAR Section(s): 6.3.1, 6.3.2

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-51 LR-DRE-M-35-1, LR-DRE-M-374, LR-DRE-M-369, LR-DRE-M-370, LR–DRE-M-39, and LR-DRE-M-40

Quad Cities Station:

LR-QDC-M-16-5, LR-QDC-M-46-1, LR-QDC-M-46-2, LR-QDC-M-46-3, LR-QDC-M-87-1, LR-QDC-M-87-2, LR-QDC-M-87-3, and LR-QDC-M-50-1.

System Intended Functions

Core cooling – provides cooling water to the reactor vessel during LOCA conditions that do not result in rapid depressurization of the reactor pressure vessel and provides coolant inventory make-up in non-LOCA events.

Pressure control – provides pressure control in events where the main steam isolation valves are closed.

Pressure boundary – maintains the integrity of the reactor coolant pressure boundary.

Primary containment isolation – provides containment isolation for those portions of the system that interface with the primary containment.

Credited in regulated event(s) – provides core make-up, cooling and pressure control credited in mitigation of the Appendix R fire, ATWS, and SBO events. The system also contains components that are relied upon for compliance with 10 CFR 50.49, (EQ).

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.2-1 Component Groups Requiring Aging Management Review - High Pressure Coolant Injection System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.1.1.1, 3.2.1.14, 3.2.2.1, 3.2.2.4, 3.2.2.6
Dampeners (Quad Cities only)	Pressure Boundary	3.2.1.13, 3.2.2.22, 3.2.2.125
Diffusers	Pressure Boundary	3.2.1.13
Filters/Strainers (includes separators)	Pressure Boundary	3.1.1.11, 3.2.1.2, 3.2.1.4, 3.2.1.13, 3.2.2.13, 3.2.2.22, 3.2.2.32, 3.2.2.137

Table 2.3.2-1 Component Groups Requiring Aging Management Review - High Pressure Coolant Injection System (Continued)

Component Group	Component Intended Function	Aging Management Ref
Filters/Strainers (includes separators)	Filter	3.2.1.2, 3.2.1.4, 3.2.1.13, 3.2.2.32
Flexible Hoses	Pressure Boundary	3.2.2.33, 3.2.2.34
Flow Orifices	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.2.137
Heat Exchangers (includes condensers)	Pressure Boundary	3.2.2.40, 3.2.2.41, 3.2.2.42, 3.2.2.43, 3.2.2.137
Heat Exchangers (includes condensers)	Heat Transfer	3.2.2.38, 3.2.2.39
NSR Vents or Drains, Piping and Valves (attached support)	Structural Integrity (attached)	3.2.2.55
Piping and Fittings (Includes thermowells)	Pressure Boundary	3.1.1.1, 3.1.1.11, 3.2.1.1, 3.2.1.2, 3.2.1.4, 3.2.1.13, 3.2.2.2, 3.2.2.9, 3.2.2.10, 3.2.2.13, 3.2.2.14, 3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.25, 3.2.2.26, 3.2.2.27, 3.2.2.28, 3.2.2.56, 3.2.2.58, 3.2.2.59, 3.2.2.64, 3.2.2.65, 3.2.2.68, 3.2.2.126, 3.2.2.137
Piping and Fittings (attached support)	Structural Integrity (attached)	3.2.2.14, 3.2.2.57, 3.2.2.137
Piping and Fittings (small bore)	Pressure Boundary	3.1.1.5, 3.2.2.9, 3.2.2.10, 3.2.2.13, 3.2.2.14, 3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.25, 3.2.2.26, 3.2.2.27, 3.2.2.28, 3.2.2.137,
Pumps	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.2.17, 3.2.2.69, 3.2.2.70, 3.2.2.71, 3.2.2.137
Restricting Orifices	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.2.13, 3.2.2.128, 3.2.2.137
Restricting Orifices	Throttle	3.2.1.2, 3.2.1.4, 3.2.2.128
Restricting Orifices (attached support)	Structural Integrity (attached)	3.2.2.72, 3.2.2.13, 3.2.2.128, 3.2.2.137
Rupture Discs	Pressure Boundary	3.2.2.22, 3.2.2.129
Sight Glasses (attached support)	Structural Integrity (attached)	3.2.2.20, 3.2.2.75, 3.2.2.76, 3.2.2.137
Sight Glasses (Quad Cities only)	Pressure Boundary	3.2.2.5, 3.2.2.20, 3.2.2.76
Tanks	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.2.13, 3.2.2.83, 3.2.2.84, 3.2.2.130, 3.2.2.137

Table 2.3.2-1 Component Groups Requiring Aging Management Review - High Pressure Coolant Injection System (Continued)

Component Group	Component Intended Function	Aging Management Ref
Thermowells	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.2.86, 3.2.2.137
Traps	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.2.13
Tubing	Pressure Boundary	3.1.1.15, 3.2.1.13, 3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.94, 3.2.2.96, 3.2.2.97, 3.2.2.132
Tubing (attached support)	Structural Integrity (attached)	3.2.2.13, 3.2.2.14, 3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.89, 3.2.2.92, 3.2.2.137
Turbine Casings	Pressure Boundary	3.2.2.8, 3.2.2.99,
Valves	Pressure Boundary	3.1.1.1, 3.1.1.11, 3.1.1.15, 3.2.1.2, 3.2.1.4, 3.2.1.12, 3.2.1.13, 3.2.2.7, 3.2.2.10, 3.2.2.11, 3.2.2.13, 3.2.2.14, 3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.100, 3.2.2.102, 3.2.2.104, 3.2.2.105, 3.2.2.106, 3.2.2.107, 3.2.2.109, 3.2.2.119, 3.2.2.121, 3.2.2.122, 3.2.2.133, 3.2.2.135, 3.2.2.137
Valves (attached support)	Structural Integrity (attached)	3.2.2.104, 3.2.2.107, 3.2.2.137

Aging management review results for the high pressure coolant injection system are provided in Section 3.1 for the reactor coolant pressure boundary functions and Section 3.2 for the additional high pressure coolant injection system functions.

2.3.2.2 Core Spray System

System Purpose

The core spray system provides core cooling for intermediate and large line break sizes. Two independent core spray loops are provided to ensure adequate core cooling. Each core spray loop is designed to operate in conjunction with low pressure coolant injection and either the automatic depressurization system or high pressure coolant injection system to provide adequate core cooling over the entire spectrum of liquid or steam break sizes.

System Operation

The core spray system consists of two independent loops, each with a motor driven pump, associated piping, valves, and instrumentation. The normal water source is supplied from the suppression chamber (evaluated with the primary containment structure). An alternate water source is the condensate storage system (evaluated with the condensate and condensate storage system). The core spray system delivers water directly to the reactor vessel (evaluated with the reactor vessel) onto the top of the fuel assemblies through the core spray spargers (evaluated with reactor internals). Each core spray loop is equipped with a test return line to the suppression chamber to permit functional testing and a minimum flow bypass line to the suppression chamber for pump protection.

System Evaluation Boundary

The core spray system evaluation boundary begins with the core spray suction lines from the suppression chamber and condensate storage system and continues through the core spray pump, and discharge lines from each pump to the point where it interfaces with the reactor vessel nozzles. All associated piping, components and instrumentation, contained within flow paths and subsystems described above are included in the core spray system evaluation boundary. The discharge path includes a minimum flow bypass line and test return line for each loop. The ECCS keep fill system is also included and is comprised of an ECCS keep fill pump with its suction and discharge lines. At Dresden only, the ECCS keep fill boundary includes a safety related isolation valve that separates the ECCS keep fill system from the condensate and condensate storage back up supply line. At Quad Cities only, the core spray boundary includes a safety related isolation valve from the high radiation sampling system. A room cooler maintains the equipment below the maximum equipment temperature limits. This room cooler has been evaluated with the ECCS Corner Room HVAC system.

UFSAR References

Dresden Station UFSAR Section(s): 6.3.1, 6.3.2

Quad Cities Station UFSAR Section(s): 6.3.1, 6.3.2

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-27, LR-DRE-M-35-1, LR-DRE-M-358, and LR-DRE-M-366

Quad Cities Station:

LR-QDC-M-36, LR-QDC-M-1056-1, LR-QDC-M-78, and LR-QDC-M-1061-1

System Intended functions

Pressure boundary - maintains the integrity of the reactor coolant pressure boundary.

Core cooling – in conjunction with low pressure coolant injection and either automatic depressurization or high pressure coolant injection, provides emergency core cooling for the entire spectrum of postulated design basis LOCAs.

Primary containment isolation – provides containment isolation for those portions of the system that interface with the primary containment.

Supports ESF function(s)– provides an ECCS keep fill subsystem which maintains core spray and low pressure coolant injection piping full of water to support a condition of stand-by readiness.

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Credited in regulated event(s) – The system contains components that are relied upon for compliance with 10 CFR 50.49, (EQ).

Component Groups Requiring Aging Management Review

Table 2.3.2-2 Component Groups Requiring Aging Management Review - Core Spray System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.1.1.1, 3.2.1.14, 3.2.2.1, 3.2.2.4
Flow Elements	Pressure Boundary	3.2.1.13, 3.2.2.22
Flow Elements	Throttle	3.2.1.13
NSR Vents or Drains, Piping and Valves (attached support)	Structural Integrity (attached)	3.2.2.55
Piping and Fittings	Pressure Boundary	3.1.1.1, 3.1.1.11, 3.1.1.15, 3.2.1.2, 3.2.1.4, 3.2.2.14, 3.2.2.22, 3.2.2.24, 3.2.2.137

Table 2.3.2-2 Component Groups Requiring Aging Management Review - Core Spray System (Continued)

Component Group	Component Intended Function	Aging Management Ref
Piping and Fittings (attached support)	Structural Integrity (attached)	3.2.2.57, 3.2.2.137
Pumps	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.2.137
Restricting Orifices	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.1.13, 3.2.2.22, 3.2.2.137
Restricting Orifices	Throttle	3.2.1.2, 3.2.1.4, 3.2.1.13
Sight Glasses (attached support)	Structural Integrity (attached)	3.2.2.75, 3.2.2.137
Thermowells	Pressure Boundary	3.2.1.13, 3.2.2.22
Tubing	Pressure Boundary	3.2.1.13, 3.2.2.22
Tubing (attached support) (Quad Cities only)	Structural Integrity (attached)	3.2.2.22, 3.2.2.92
Valves	Pressure Boundary	3.1.1.1, 3.1.1.11, 3.1.1.15, 3.2.1.2, 3.2.1.4, 3.2.1.12, 3.2.1.13, 3.2.2.14, 3.2.2.22, 3.2.2.24, 3.2.2.137, 3.2.2.122
Valves (attached support)	Structural Integrity (attached)	3.2.2.22, 3.2.2.104, 3.2.2.113, 3.2.2.115, 3.2.2.116, 3.2.2.124, 3.2.2.137

Aging management review results for the core spray system are provided in Section 3.1 for reactor coolant pressure boundary functions and Section 3.2 for additional core spray functions.

2.3.2.3 Containment Isolation Components and Primary Containment Piping System

System Purpose

The containment isolation components and primary containment piping system is a composite support system for the primary containment structure. The containment isolation components and primary containment piping system is comprised of primary containment isolation valves, penetrations and piping from non-safety related systems that perform no intended function except primary containment isolation. It also includes safety related piping, components and instrumentation that directly support intended functions of the primary containment structure and that are not assigned to other systems in-scope of license renewal. The containment isolation components and primary containment piping system ensures that the primary containment structure is able to perform its intended functions.

System Operation

In the event of a nuclear steam supply system piping failure within the drywell (evaluated with the primary containment structure) reactor water and/or steam would be released into the drywell. The resulting increased drywell pressure would force a mixture of radioactive materials, noncondensable gases, steam, and water through the connecting vent lines into the chamber of water in the suppression chamber, which is also called the torus (evaluated with primary containment structure). The steam would condense rapidly and completely in the suppression chamber resulting in suppression of the pressure increase in the drywell. During this period, the primary containment and suppression chamber piping isolation valves are relied upon to ensure the containment of these gases and liquids. Vacuum breakers between the suppression chamber and the reactor building and between the suppression chamber and the drywell ensure that venting of non-condensable gases to the suppression chamber together with condensation of the released steam does not result in exceeding drywell or suppression chamber external design differential pressure. Instrument air to the TIP tubing ensures a clean dry environment for the TIP probe while ensuring isolation capability. TIP ball and shear valves ensure tubing isolation, including situations where isolation is required concurrent with TIP probes transversing the core or an invessel stuck probe. Components associated with ACAD (indication instruments only) and drywell pneumatics provide the supply for air operated valves and for maintaining an atmosphere that inhibits the formation of a combustible gas mixture. Floor and equipment drains provide for the measurement and removal of leakage while maintaining offsite radiological consequences to within acceptable limits, by isolating and maintaining containment integrity when required.

System Evaluation Boundary

The containment isolation components and primary containment piping system evaluation boundary consists of:

- primary containment pressure instruments
- suppression chamber to reactor building vacuum breaker lines
- purge supply and exhaust penetrations (HVAC – primary containment)

- suppression chamber level instrumentation penetrations
- LLRT test penetrations
- containment isolation barriers from the following systems:
  - traversing incore probe
  - drywell equipment and floor drain sumps
  - atmospheric containment air dilution (ACAD)
  - service air
  - instrument air

All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the primary containment and suppression chamber piping system evaluation boundary. The components that provide primary containment isolation for other in-scope systems are included in the overall evaluations for those systems.

UFSAR References

Dresden Station UFSAR Section(s): 6.2.1

Quad Cities Station UFSAR Section(s): 6.2.1

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE-M-25, LR-DRE-M-26-1, LR-DRE-M-272, LR-DRE-M-37-2, LR-DRE-M-38-2, LR-DRE-M-39, LR-DRE-M-51, LR-DRE-M-356, LR-DRE-M-357-1, LR-DRE-M-367-2, LR-DRE-M-368, LR-DRE-M-369, LR-DRE-M-707-1, and LR-DRE-M-707-2.

Quad Cities Station:
LR-QDC-M-24-12, LR-QDC-M-24-13, LR-QDC-M-25-1, LR-QDC-M-34-1, LR-QDC-M-43, LR-QDC-M-71-7, LR-QDC-M-71-8, LR-QDC-M-72-1, LR-QDC-M-76-1, LR-QDC-M-85, LR-QDC-M-584-1, LR-QDC-M-584-2, LR-QDC-M-642-1, LR-QDC-M-642-2, LR-QDC-M-1056-1 and LR-QDC-M-1061-1

System Intended Functions

Primary containment isolation - provides functions that support isolation of primary containment, which in the event of a LOCA, will provide a barrier to control the release of fission products to the secondary containment.

Pressure suppression - provides functions that support absorption of energy by containment air and water volumes so that containment pressure is maintained within acceptable limits during and following design basis events.

Containment integrity - provides vacuum relief between drywell and suppression chamber, and suppression chamber and reactor building as required to maintain containment integrity.

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Credited in regulated event(s) – provides containment isolation, integrity and pressure suppression credited in mitigation of the Appendix R fire, ATWS and SBO events. The system also contains components that are relied upon for compliance with 10 CFR 50.49, (EQ)

Component Groups Requiring Aging Management Review

Table 2.3.2-3 Component Groups Requiring Aging Management Review - Containment Isolation Components and Primary Containment Piping System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.2.1.14, 3.2.2.1, 3.2.2.4
Flexible Hoses (Quad Cities only)	Pressure Boundary	3.2.2.35, 3.2.2.36
Flow Elements (Quad Cities only)	Pressure Boundary	3.2.2.22, 3.2.2.37
Isolation Barriers (includes piping, tubing, valves, and vacuum breakers)	Pressure Boundary	3.2.1.3, 3.2.1.5, 3.2.1.6, 3.2.2.14, 3.2.2.22, 3.2.2.24, 3.2.2.49, 3.2.2.50, 3.2.2.51, 3.2.2.52, 3.2.2.53, 3.2.2.137
Isolation Barriers (attached support) (includes piping and valves)	Structural Integrity (attached)	3.2.2.49, 3.2.2.52
NSR Vents or Drains, Piping and Valves (attached support) (includes tubing)	Structural Integrity (attached)	3.2.2.55
Piping and Fittings	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.2.14, 3.2.2.22, 3.2.2.24, 3.2.2.62, 3.2.2.67, 3.2.2.137
Piping and Fittings (attached support)	Structural Integrity (attached)	3.2.2.14, 3.2.2.22, 3.2.2.24, 3.2.2.57, 3.2.2.60, 3.2.2.62, 3.2.2.66, 3.2.2.137
Restricting Orifices (Dresden only)	Pressure Boundary	3.2.2.74
Tanks (includes drain pots)	Pressure Boundary	3.2.1.13, 3.2.2.3, 3.2.2.22
Thermowells (Quad Cities only)	Pressure Boundary	3.2.2.87, 3.2.2.137
Tubing	Pressure Boundary	3.2.1.13, 3.2.2.14, 3.2.2.18, 3.2.2.19, 3.2.2.22, 3.2.2.24, 3.2.2.90, 3.2.2.91, 3.2.2.98, 3.2.2.137

Table 2.3.2-3 Component Groups Requiring Aging Management Review - Containment Isolation Components and Primary Containment Piping System (Continued)

Component Group	Component Intended Function	Aging Management Ref
Valves	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.1.12, 3.2.1.13, 3.2.2.11, 3.2.2.12, 3.2.2.14, 3.2.2.22, 3.2.2.24, 3.2.2.103, 3.2.2.108, 3.2.2.110, 3.2.2.112, 3.2.2.113, 3.2.2.122, 3.2.2.123, 3.2.2.137
Valves (attached support)	Structural Integrity (attached)	3.2.2.11, 3.2.2.12, 3.2.2.14, 3.2.2.22, 3.2.2.24, 3.2.2.103, 3.2.2.104, 3.2.2.110, 3.2.2.112, 3.2.2.113, 3.2.2.122, 3.2.2.123, 3.2.2.137

Aging management review results for the containment isolation and primary containment piping system are provided in Section 3.2.

2.3.2.4 Reactor Core Isolation Cooling System (Quad Cities Only)

System Purpose

The reactor core isolation cooling (RCIC) system at Quad Cities provides cooling water to the reactor core in the event of a postulated isolation of the reactor from the main condenser with a loss of reactor feedwater.

System Operation

The RCIC system consists of a steam turbine-pump unit, piping, associated valves, auxiliary support systems and instrumentation. The turbine is driven by nuclear steam supplied from the “A” main steam line on Unit 1 and from the “D” main steam line on Unit 2, (evaluated with main steam), and exhausts to the suppression chamber (evaluated with primary containment structure), below the water line. All steam leakage from valve packing and the turbine shaft seals is routed to and condensed in the barometric condenser. The preferred water source to the RCIC pump suction is supplied from the condensate storage tank (evaluated under the condensate and condensate storage system), with a backup source from the suppression chamber (evaluated with primary containment structure). The pump discharge is delivered into the reactor vessel through a connection to the “A” feedwater line (evaluated with the feedwater system) and is distributed within the vessel through the feedwater spargers (evaluated with reactor internals). A minimum flow bypass line from the pump discharge line to the suppression chamber is provided for pump protection. The RCIC system is equipped with a test line used for functional testing that returns condensate to the condensate storage tank. The RCIC test return line is tied to the HPCI system test return line (evaluated under the HPCI system). RCIC auxiliaries include: the drain pot subsystem, the barometric condenser and vacuum subsystem, and the turbine oil subsystem. The RCIC turbine and pump are located in a room with a core spray pump, and the area is cooled by the core spray room cooler (evaluated with ECCS corner room HVAC).

System Evaluation Boundary

The RCIC system evaluation boundary for water injection begins with the RCIC suction line from the condensate storage tank and the RCIC suction line from the suppression chamber. Included are all piping and components that feed the RCIC pump. The discharge path from the RCIC pump is also included, along with supporting components. This includes the drain pot subsystem, the barometric condenser condensate and vacuum subsystems, and the turbine oil subsystem. Within the evaluation boundary is the RCIC injection path, test return line (a portion of the RCIC flow test return line was evaluated with the HPCI system) and all supporting components. This runs from the output side of the RCIC pump to the “A” feedwater line outside of the primary containment. The boundary continues with the steam supply path from a main steam line with all associated piping and components. The RCIC turbine exhaust to the suppression chamber, along with all supporting piping and components are also included in the RCIC system evaluation boundary.

UFSAR References

Dresden Station UFSAR Section(s): Not Applicable

Quad Cities Station UFSAR Section(s): 5.4.6

License Renewal Boundary Diagram References

Dresden Station: Not Applicable

Quad Cities Station:

LR-QDC-M-16-5, LR-QDC-M-50-1, LR-QDC-M-50-2, LR-QDC-M-89-1,

and LR-QDC-M-89-2.

System Intended Functions

Pressure boundary - maintains the integrity of the reactor coolant pressure boundary.

Primary containment isolation – provides containment isolation for those portions of the system that interface with the primary containment.

Core cooling – provides cooling water to the core and provides capability for level and pressure control during normal reactor isolation conditions.

Credited in regulated event(s) – provides core cooling, including capability for level and pressure control, credited in mitigation of the Appendix R fire, ATWS and SBO events. The system also contains components relied upon for compliance with 10 CFR 50.49, (EQ).

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.2-4 Component Groups Requiring Aging Management Review - Reactor Core Isolation Cooling System (Quad Cities only)

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting (Quad Cities only)	Pressure Boundary	3.1.1.1, 3.2.1.14, 3.2.2.1, 3.2.2.4
Dampeners (Quad Cities only)	Pressure Boundary	3.2.1.13, 3.2.2.22, 3.2.2.23, 3.2.2.125
Filters/Strainers (Quad Cities only)	Pressure Boundary	3.2.1.13, 3.2.2.32, 3.2.2.137
Filters/Strainers (Quad Cities only)	Filter	3.2.1.13
Flexible Hoses (Quad Cities only)	Pressure Boundary	3.2.2.35

Table 2.3.2-4 Component Groups Requiring Aging Management Review - Reactor Core Isolation Cooling System (Quad Cities only)

Component Group	Component Intended Function	Aging Management Ref
NSR Vents or Drains, Piping and Valves (attached support) (Quad Cities only)	Structural Integrity (attached)	3.2.2.55
Piping and Fittings (Quad Cities only) (includes rupture discs)	Pressure Boundary	3.1.1.1, 3.1.1.11, 3.2.1.2, 3.2.1.4, 3.2.1.13, 3.2.2.2, 3.2.2.13, 3.2.2.14, 3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.27, 3.2.2.58, 3.2.2.59, 3.2.2.64, 3.2.2.68, 3.2.2.126, 3.2.2.137
Piping and Fittings (small bore) (Quad Cities only)	Pressure Boundary	3.1.1.5, 3.2.2.137
Pumps (Quad Cities only)	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.2.17, 3.2.2.69, 3.2.2.70, 3.2.2.71, 3.2.2.137
Restricting Orifices (Quad Cities only)	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.1.13, 3.2.2.22, 3.2.2.73, 3.2.2.128, 3.2.2.137
Restricting Orifices (Quad Cities only)	Throttle	3.2.1.2, 3.2.1.4, 3.2.2.73, 3.2.2.128
Sight Glasses (Quad Cities only)	Pressure Boundary	3.2.2.20, 3.2.2.77
Tanks (Quad Cities only) (includes drain pots, actuators, and condensers)	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.2.13, 3.2.2.17, 3.2.2.83, 3.2.2.84, 3.2.2.85, 3.2.2.130, 3.2.2.137
Traps (Quad Cities only)	Pressure Boundary	3.2.2.13, 3.2.2.88, 3.2.2.131, 3.2.2.137
Tubing (Quad Cities only)	Pressure Boundary	3.2.1.13, 3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.96
Turbine Casings (Quad Cities only)	Pressure Boundary	3.2.2.8, 3.2.2.99
Valves (Quad Cities only)	Pressure Boundary	3.1.1.1, 3.1.1.11, 3.1.1.15, 3.2.1.2, 3.2.1.4, 3.2.1.12, 3.2.1.13, 3.2.2.13, 3.2.2.14, 3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.106, 3.2.2.107, 3.2.2.109, 3.2.2.119, 3.2.2.121, 3.2.2.133, 3.2.2.135, 3.2.2.137
Valves (small bore) (Quad Cities only)	Pressure Boundary	3.1.1.5, 3.2.2.13, 3.2.2.14, 3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.137

Aging management review results for the reactor core isolation cooling system are provided in Section 3.1 for the reactor coolant pressure boundary functions and Section 3.2 for the additional RCIC functions.

2.3.2.5 Isolation Condenser (Dresden Only)

System Purpose

The isolation condenser system at Dresden provides reactor core cooling in the event that the reactor becomes isolated from the turbine and main condenser by closure of the main steam isolation valves.

System Operation

The isolation condenser is a heat exchanger, which consists of two tube bundles immersed in a large water storage tank. The isolation condenser system operates by natural circulation without the need for power other than dc power to open the condensate return valve to initiate system operation. During isolation condenser system operation, steam flows through the isolation condenser steam supply line directly from the reactor vessel (evaluated with the reactor vessel), condenses in the tubes of the heat exchanger, and returns by gravity through the isolation condenser return line to the reactor via the “A” recirculation loop (evaluated with the reactor recirculation, recirculation flow control, and MG sets system). Isolation valves are provided on the lines that penetrate the primary containment. The differential water head, created when the steam is condensed, serves as the driving force. The water on the shell side of the condenser boils and vents to atmosphere. The tube side of the isolation condenser system is equipped with a high point vent which is used during normal operation to prevent the long term buildup of noncondensable gases. These gases are vented to the “A” main steam line, downstream of the main steam line flow restrictor (venturi) (evaluated with the main steam system). The differential pressure across the venturi provides the driving force for the flow of steam and non-condensable gases from the tube side of the isolation condenser system to the main steam line. The preferred makeup water source is the clean demineralized water storage tank via two diesel driven isolation condenser makeup water pumps. Alternate makeup water sources are the fire protection system (evaluated with the fire protection system) and the condensate storage system (evaluated with the condensate and condensate storage system). Two radiation monitors (evaluated with the process radiation monitoring system) are provided on the shell vent. In the event of excessive radiation levels, the tube side of the heat exchanger can be isolated from the reactor.

System Evaluation Boundary

The tube side boundary of the isolation condenser system begins with the steam supply line from the reactor vessel to the tube bundles located inside the isolation condenser. Included is the high point vent to a main steam line. The tube side flow continues with the condensate return line back to the “A” reactor recirculation piping. The shell side boundary includes an atmospheric vent and makeup piping from the isolation condenser makeup subsystem (diesel driven pumps), the clean demineralized water storage tank, and the clean demineralizer water fill valve (all are part of the demineralized water makeup system). Also included in the evaluation boundary are shell side vent and drain valves that are part of the reactor building equipment and floor drain system. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the isolation condenser evaluation boundary.

UFSAR References

Dresden Station UFSAR Section(s): 5.4.6

Quad Cities Station UFSAR Section(s): Not Applicable

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-12-2, LR-DRE-M-28, LR-DRE-M-32, LR-DRE-M-345-2, LR-DRE-M-35-1, LR-DRE-M-363, LR-DRE-M-39, LR-DRE-M-359, LR-DRE-M-366, LR-DRE-M-369, and LR-DRE-M-4203.

Quad Cities Station: Not Applicable

System Intended Functions

Pressure boundary - maintains the integrity of the reactor coolant pressure boundary.

Primary containment isolation – provides containment isolation for those portions of the system that interface with the primary containment.

Credited in regulated event(s) – provides reactor pressure control and core cooling functions (in vessel isolation conditions) credited in mitigation of the Appendix R fire, ATWS and SBO events. The system also contains components that are relied upon for compliance with 10 CFR 50.49, (EQ).

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.2-5 Component Groups Requiring Aging Management Review - Isolation Condenser (Dresden only)

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting (Dresden only)	Pressure Boundary	3.1.1.1, 3.2.1.14, 3.2.2.1, 3.2.2.4, 3.2.2.6
Isolation Condensers (Dresden only)	Pressure Boundary	3.1.1.2, 3.1.1.7, 3.2.2.137
Isolation Condensers (Dresden only)	Heat Transfer	3.1.2.15
NSR Vents or Drains, Piping and Valves (attached support) (Dresden only)	Structural Integrity (attached)	3.2.2.55

Table 2.3.2-5 Component Groups Requiring Aging Management Review - Isolation Condenser (Dresden only) (Continued)

Component Group	Component Intended Function	Aging Management Ref
Piping and Fittings (Dresden only)	Pressure Boundary	3.1.1.1, 3.1.1.15, 3.2.1.2, 3.2.1.4, 3.2.1.13, 3.2.2.10, 3.2.2.13, 3.2.2.14, 3.2.2.15, 3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.25, 3.2.2.26, 3.2.2.56, 3.2.2.137
Piping and Fittings (attached support) (Dresden only)	Structural Integrity (attached)	3.2.2.13, 3.2.2.14, 3.2.2.15, 3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.25, 3.2.2.61, 3.2.2.137
Piping and Fittings (small bore) (Dresden only)	Pressure Boundary	3.1.1.5, 3.2.2.13, 3.2.2.14, 3.2.2.15, 3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.25, 3.2.2.137
Pumps (Dresden only)	Pressure Boundary	3.2.1.13, 3.2.2.22
Flow Elements (Dresden only)	Pressure Boundary	3.1.1.15, 3.2.2.24, 3.2.1.13, 3.2.2.22
Sight Glasses (Dresden only)	Pressure Boundary	3.2.2.20, 3.2.2.76
Tanks (Dresden only)	Pressure Boundary	3.2.2.10, 3.2.2.82
Thermowells (Dresden only)	Pressure Boundary	3.1.1.15, 3.2.2.22
Tubing (Dresden only)	Pressure Boundary	3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.97
Valves (Dresden only)	Pressure Boundary	3.1.1.1, 3.1.1.15, 3.2.1.2, 3.2.1.4, 3.2.1.12, 3.2.2.13, 3.2.2.14, 3.2.2.15, 3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.25, 3.2.2.122, 3.2.2.137
Valves (attached support) (Dresden only)	Structural Integrity (attached)	3.2.2.22, 3.2.2.111, 3.2.2.116, 3.2.2.137

Aging management review results for the isolation condenser system are provided in Section 3.1 for the reactor coolant pressure boundary functions and Section 3.2 for the additional isolation condenser functions.

2.3.2.6 Residual Heat Removal System (Quad Cities Only)

System Purpose

The residual heat removal (RHR) system at Quad Cities has three modes of operation. The low pressure coolant injection (LPCI) mode of RHR is the only ESF function of the system and operates to restore water level in the reactor vessel. The containment cooling mode furnishes spray to the drywell and suppression chamber to aid in reducing containment pressure following a loss of coolant accident (LOCA). This mode also provides suppression chamber cooling to reduce water temperatures during operations that add heat to the suppression chamber and minimizes the amount of heat that the containment will need to accommodate during a LOCA. The shutdown cooling mode removes reactor residual and decay heat for shutdown, refueling and servicing operations.

System Operation

The RHR system consists of two loops, each loop containing two RHR pumps, one RHR heat exchanger (evaluated with the residual heat removal service water system) and the necessary valves and piping to connect these components to the reactor vessel via the recirculation system piping, the suppression chamber for spray / cooling and the drywell for spray. The RHR system piping is maintained full by the ECCS keep fill system (evaluated with the core spray system). Each loop of the system is equipped a minimum flow bypass line to the suppression chamber for RHR pump protection. During normal plant operation the RHR system is maintained in a lineup to be ready to inject water into either recirculation loop with all RHR pumps. Process lines that penetrate the primary containment structure contain isolation valves. The RHR room coolers are evaluated with the ECCS corner room HVAC system.

For the LPCI mode of operation, the primary source of water to the RHR system is supplied from the suppression chamber (evaluated with the primary containment structure). The backup source of water is the condensate storage tank (evaluated with the condensate and condensate storage system). For each loop, water is pumped from the suppression chamber, through the pumps to the heat exchanger (HX) and the HX bypass valve. Upon automatic initiation of the RHR system, the LPCI loop select logic will select the recirculation loop (evaluated with recirculation, recirc flow control & MG sets system) that appears most likely intact and, provided reactor pressure is sufficiently low, will inject to the intact recirculation loop.

For the containment cooling mode of operation, there are 3 different uses.

Drywell spray takes suction from the suppression chamber and pumps water to two spray nozzle headers in the drywell. These spray headers may be used during a LOCA to reduce drywell pressure.

Suppression chamber spray takes suction from the suppression chamber and pumps water to spray nozzles in the suppression chamber. This reduces suppression chamber pressure following a LOCA.

Suppression chamber cooling takes suction from the suppression chamber and pumps through a RHR heat exchanger, (which rejects heat to the RHR service water system) and pumps the water back to the suppression chamber. This mode provides a heat sink, external to the containment, which will limit suppression chamber water temperature during conditions such as RCIC operation and minimize the amount of heat that the suppression chamber will need to accommodate during a LOCA (for pressure suppression and ECCS pump required suction head).

For the shutdown cooling mode of operation, the RHR pumps take suction from the “B” reactor recirculation system suction piping, pumps water through a RHR heat exchanger (for heat removal via RHR service water system) and returns the water to the reactor vessel via the recirculation system pump discharge line.

System Evaluation Boundary

The RHR system evaluation boundary begins with the RHR suction lines from the suppression chamber, condensate storage tank and reactor recirculation piping. Included are the four RHR pumps and discharge piping up to the two RHR heat exchangers and minimum flow lines for each loop. Also included are the return lines from each heat exchanger to the point where the RHR piping interfaces with the reactor vessel via the reactor recirculation piping, spray piping to the suppression chamber and drywell for each loop, and cooling water return to the suppression chamber. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the RHR system evaluation boundary.

UFSAR References

Dresden Station UFSAR Section(s): Not Applicable

Quad Cities Station UFSAR Section(s): 5.4.7, 6.3.1, 6.3.2

License Renewal Boundary Diagram References

Dresden Station: Not Applicable

Quad Cities Station:
LR-QDC-CID-39, LR-QDC-CID-81, LR-QDC-M-16-5, LR-QDC-M-34-1, LR-QDC-M-35-2, LR-QDC-M-37, LR-QDC-M-38, LR-QDC-M-39-1, LR-QDC-M-39-2, LR-QDC-M-39-3, LR-QDC-M-39-4, LR-QDC-M-76-1, LR-QDC-M-77-2, LR-QDC-M-79, LR-QDC-M-80, LR-QDC-M-81-1, LR-QDC-M-81-2, LR-QDC-M-81-3, LR-QDC-M-85, LR-QDC-M-1056-2, and LR-QDC-M-1061-2.

System Intended Functions

Pressure boundary - maintains the integrity of the reactor coolant pressure boundary.

Core cooling – provides emergency core cooling for various postulated loss of coolant accidents for a range of failure sizes from those for which the core is adequately cooled by HPCI up to and including the design basis accident. In addition, provides heat removal sufficient to achieve and maintain cold shutdown conditions during normal operation.

Containment cooling – provides emergency containment cooling by recirculating suppression chamber water through the system heat exchangers and by spraying water into the drywell and the suppression chamber chamber.

Primary containment isolation – provides containment isolation for those portions of the system that interface with the primary containment.

Credited in regulated event(s) – provides containment cooling and decay heat removal credited in mitigation of the Appendix R fire and ATWS events. The system also contains components that are relied upon for compliance with 10 CFR 50.49, (EQ).

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.2-6 Component Groups Requiring Aging Management Review - Residual Heat Removal System (Quad Cities only)

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting (Quad Cities only)	Pressure Boundary	3.1.1.1, 3.2.1.14, 3.2.2.1, 3.2.2.4
Dampeners (Quad Cities only)	Pressure Boundary	3.2.2.22, 3.2.2.23, 3.2.2.30
Dampeners (attached support) (Quad Cities only)	Structural Integrity (attached)	3.2.2.22, 3.2.2.29
ECCS Suction Headers (Quad Cities only)	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.2.137
Filters/Strainers (Quad Cities only)	Pressure Boundary	3.2.1.13, 3.2.2.22
Filters/Strainers (Quad Cities only)	Filter	3.2.1.13
Flow Elements (Quad Cities only)	Pressure Boundary	3.2.1.13, 3.2.2.22
Flow Elements (Quad Cities only)	Throttle	3.2.1.13
NSR Vents or Drains, Piping and Valves (attached support) (Quad Cities only) (includes flow glasses)	Structural Integrity (attached)	3.2.2.55
Piping and Fittings (Quad Cities only)	Pressure Boundary	3.1.1.1, 3.1.1.15, 3.2.1.2, 3.2.1.3, 3.2.1.4, 3.2.1.12, 3.2.1.13, 3.2.2.14, 3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.137
Piping and Fittings (attached support) (Quad Cities only)	Structural Integrity (attached)	3.2.2.14, 3.2.2.22, 3.2.2.24, 3.2.2.57, 3.2.2.63, 3.2.2.137
Pumps (Quad Cities only)	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.2.137

Table 2.3.2-6 Component Groups Requiring Aging Management Review - Residual Heat Removal System (Quad Cities only) (Continued)

Component Group	Component Intended Function	Aging Management Ref
Restricting Orifices (Quad Cities only) (includes dampeners)	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.1.13, 3.2.2.22, 3.2.2.137
Restricting Orifices (Quad Cities only)	Throttle	3.2.1.2, 3.2.1.4, 3.2.1.13
Sight Glasses (attached support) (Quad Cities only)	Structural Integrity (attached)	3.2.2.20, 3.2.2.75, 3.2.2.76, 3.2.2.137
Spray Nozzles (Quad Cities only)	Pressure Boundary	3.2.2.12, 3.2.2.78
Spray Nozzles (Quad Cities only)	Spray	3.2.2.78
Thermowells (Quad Cities only)	Pressure Boundary	3.2.1.13, 3.2.2.22
Tubing (Quad Cities only)	Pressure Boundary	3.2.1.13, 3.2.2.22, 3.2.2.24, 3.2.2.97
Tubing (attached support) (Quad Cities only)	Structural Integrity (attached)	3.2.2.22, 3.2.2.24, 3.2.2.93
Valves (Quad Cities only)	Pressure Boundary	3.1.1.1, 3.1.1.15, 3.2.1.2, 3.2.1.4, 3.2.1.12, 3.2.1.13, 3.2.2.14, 3.2.2.22, 3.2.2.23, 3.2.2.24, 3.2.2.105, 3.2.2.110, 3.2.2.117, 3.2.2.137
Valves (attached support) (Quad Cities only)	Structural Integrity (attached)	3.2.2.14, 3.2.2.22, 3.2.2.24, 3.2.2.104, 3.2.2.115, 3.2.2.116, 3.2.2.137

Aging management review results for the residual heat removal system are provided in Section 3.1 for the reactor coolant pressure boundary functions and Section 3.2 for the additional RHR functions.

2.3.2.7 Low Pressure Coolant Injection System (Dresden Only)

System Purpose

The LPCI system is comprised of two independent loops, each with two pumps and a heat exchanger that supply water to the reactor core via the reactor recirculation system. The LPCI system provides core cooling during a loss of coolant accident (LOCA) for break sizes ranging from those for which the core is adequately cooled by the high pressure coolant injection (HPCI) system alone, up to and including a design basis accident (DBA). LPCI is capable of injecting large quantities of water into the reactor pressure vessel and provides core cooling by submerging the core in water. The LPCI system is also designed to supply cooling / spray water to the primary containment (drywell and suppression chamber) during accident conditions to maintain containment temperature and pressure below design limits. The LPCI system is also the normal means of removing water from the suppression chamber to maintain the water level in the normal band.

System Operation

The LPCI system consists of two independent loops, each with two motor driven pumps, a LPCI heat exchanger (evaluated with the containment cooling service water system (CCSW)), associated piping, valves, and instrumentation. The normal water source is supplied from the suppression chamber via an ECCS suction header (evaluated with the primary containment structure). An alternate source of water to the LPCI pumps is supplied from the condensate storage tank (evaluated with the condensate and condensate storage system). The LPCI pumps can route water to several discharge paths. The LPCI system supplies water to the reactor vessel through the LPCI heat exchanger and into the reactor recirculation system (evaluated with the recirculation, recirculation flow control and MG set system) downstream of the reactor recirculation pumps. A motor operated valve allows LPCI flow to bypass the heat exchanger. Each loop can deliver water to the reactor vessel through its own injection line or through the other LPCI loop injection line via a cross tie line. Each LPCI loop is equipped with a test return line to the suppression chamber to permit functional testing and a minimum flow bypass line to the suppression chamber for pump protection.

Each LPCI loop also has the capability to deliver cooling / spray water to the primary containment during accident conditions. The containment cooling mode of operation consists of: (a) drywell spray where LPCI pumps are aligned to pump water from the suppression chamber to headers equipped with spray nozzles in the drywell (evaluated with the primary containment structure) to reduce containment pressure following a LOCA, (b) suppression chamber spray where LPCI pumps are aligned to pump water from the suppression chamber to a header equipped with spray nozzles (evaluated with the primary containment structure) in the suppression chamber to reduce containment pressure following a LOCA, and (c) suppression chamber cooling where LPCI pumps are aligned to recirculate water from the suppression chamber, through the LPCI heat exchangers and back to the suppression chamber.

The LPCI system is also the normal means of removing water from the suppression chamber to maintain normal operational level band. Taking suction from the suppression chamber, the LPCI pumps can transfer water from the suppression chamber to the following locations: the suppression chamber of the other unit, the main condenser (evaluated with main condenser) of either unit, or to the floor drain collector tank (evaluated with radwaste and equipment drains).

System Evaluation Boundary

The LPCI system evaluation boundary begins with the LPCI suction lines from the suppression chamber and condensate storage system. Included are the four LPCI pumps and discharge piping, minimum flow lines for each loop, and the cross tie line connecting the LCPI loops. The LPCI heat exchangers have been excluded from the LPCI evaluation boundary because they are evaluated with the CCSW system. Also included in the LPCI evaluation boundary are return lines from each LPCI heat exchanger to the point where the LPCI piping interfaces with the reactor vessel via the reactor recirculation piping, spray piping to the suppression chamber and drywell for each loop, and cooling water return lines to the suppression chamber. The LPCI system evaluation boundary includes isolation valves from the high radiation sampling system (HRSS) and their associated instrumentation and manual isolation valves. These valves isolate the safety related LPCI system piping from the non-safety related HRSS. Additionally, HRSS grab sample manual isolation valves to the HRSS panel sample cooler are included. A room cooler maintains the equipment below the maximum equipment temperature limits. The LCPI room cooler has been evaluated with the ECCS corner room HVAC system.

UFSAR References

Dresden Station UFSAR Section(s): 6.2.2, 6.3.1.2, 6.3.2.2

Quad Cities Station UFSAR Section(s): Not Applicable

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE-M-29-1, LR-DRE-M-35-1, LR-DRE-M-360-1, LR-DRE-M-1234-1, and LR-DRE-M-1239-1.

Quad Cities Station: Not Applicable

System Intended Functions

Pressure boundary - maintains the integrity of the reactor coolant pressure boundary.

Core cooling – provides emergency core cooling for various postulated loss of coolant accidents for a range of failure sizes from those for which the core is adequately cooled by HPCI up to and including the design basis accident.

Containment cooling – provides emergency containment cooling by recirculating suppression chamber water through the system heat exchangers and by spraying water into the drywell and the suppression chamber chamber.

Primary containment isolation – provides containment isolation for those portions of the system that interface with the primary containment.

Credited in regulated event(s) – provides containment cooling functions credited in mitigation of the Appendix R fire protection and ATWS events. The system also contains components that are relied upon for compliance with 10 CFR 50.49, (EQ).

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.2-7 Component Groups Requiring Aging Management Review - Low Pressure Coolant Injection System (Dresden only)

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting (Dresden only)	Pressure Boundary	3.1.1.1, 3.2.1.14, 3.2.2.1, 3.2.2.4
ECCS Suction Headers (Dresden only)	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.2.137
Filters/Strainers (Dresden only)	Filter	3.2.1.13
Flow Elements (Dresden only)	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.1.13, 3.2.2.22, 3.2.2.137
Flow Elements (Dresden only)	Throttle	3.2.1.13
NSR Vents or Drains, Piping and Valves (attached support) (Dresden only) (includes flow glasses)	Structural Integrity (attached)	3.2.2.55,
Piping and Fittings (Dresden only)	Pressure Boundary	3.1.1.1, 3.1.1.15, 3.2.1.2, 3.2.1.3, 3.2.1.4, 3.2.1.12, 3.2.2.22, 3.2.2.24, 3.2.2.137
Piping and Fittings (attached support) (Dresden only)	Structural Integrity (attached)	3.2.2.57, 3.2.2.62, 3.2.2.137
Pumps (Dresden only)	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.2.137
Restricting Orifices (Dresden only)	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.2.137
Restricting Orifices (Dresden only)	Throttle	3.2.1.2, 3.2.1.4
Sight Glasses (attached support) (Dresden only)	Structural Integrity (attached)	3.2.2.75, 3.2.2.137
Spray Nozzles (Dresden only)	Pressure Boundary	3.2.2.12, 3.2.2.78
Spray Nozzles (Dresden only)	Spray	3.2.2.78
Thermowells (Dresden only)	Pressure Boundary	3.2.1.13, 3.2.2.22
Tubing (Dresden only)	Pressure Boundary	3.2.1.13, 3.2.2.22, 3.2.2.97
Tubing (attached support) (Dresden only)	Structural Integrity (attached)	3.2.2.22, 3.2.2.92

Table 2.3.2-7 Component Groups Requiring Aging Management Review - Low Pressure Coolant Injection System (Dresden only) (Continued)

Component Group	Component Intended Function	Aging Management Ref
Valves (Dresden only)	Pressure Boundary	3.1.1.1, 3.1.1.15, 3.2.1.2, 3.2.1.4, 3.2.1.12, 3.2.1.13, 3.2.2.22, 3.2.2.24, 3.2.2.122, 3.2.2.123, 3.2.2.137
Valves (attached support) (Dresden only)	Structural Integrity (attached)	3.2.2.22, 3.2.2.24, 3.2.2.104, 3.2.2.113, 3.2.2.115, 3.2.2.122, 3.2.2.137

Aging management review results for the low pressure coolant injection system are provided in Section 3.1 for the reactor coolant pressure boundary functions and Section 3.2 for the additional LPCI functions.

2.3.2.8 Standby Liquid Control System

System Purpose

The standby liquid control system is able to bring the reactor from full power to a cold, xenon-free sub critical condition, assuming that none of the withdrawn control rods can be inserted, by injecting sodium pentaborate solution into the reactor core.

System Operation

The standby liquid control system consists of a tank for sodium pentaborate solution storage, two parallel suction lines from the tank with normally-opened suction valves feeding a common pump suction header, two positive displacement pumps with normally-opened suction valves that discharge into a common header; two explosion-actuated shear plug valves arranged in parallel discharge lines from the common discharge header, and other piping, valves and instrumentation. The explosive valves are actuated to provide a flow path, and the sodium pentaborate solution is delivered to the reactor vessel (evaluated with the reactor vessel) by one or both of the positive displacement pumps. The pumps and piping are protected from overpressure by two relief valves which discharge back to the standby liquid control tank. Heaters are installed in the standby liquid control storage tank to ensure the solution is maintained at sufficient temperature to keep the sodium pentaborate in solution. System piping normally filled with the sodium pentaborate solution is heat traced to ensure that the sodium pentaborate does not precipitate in the piping. The system also includes a test tank and associated piping used to measure pump performance.

System Evaluation Boundary

The standby liquid control system evaluation boundary begins with the storage tank clean demineralized water supply line from the clean demineralized water and make up system (evaluated with the clean demineralized water and makeup system). Included on the intake side of the storage tank is the service air intake for the tank air sparger that is used in the mixing of the boron solution and the instrument air lines and associated instruments for the storage tank. The evaluation boundary continues with the storage tank, piping from the storage tank to the positive displacement pumps, relief valves, accumulators, pump discharge check valves, piping to the two explosive actuated valves and the piping to the reactor vessel. The evaluation boundary includes the test tank, drain lines, catch drums, heat tracing and all system controls. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the standby liquid control system evaluation boundary.

UFSAR References

Dresden Station UFSAR Section(s): 9.3.5

Quad Cities Station UFSAR Section(s): 9.3.5

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-33 and LR-DRE-M-364

Quad Cities Station:

LR-QDC-M-40 and LR-QDC-M-82

System Intended Functions

Reactivity control - provides the capability for bringing the reactor from full power to a cold, xenon-free shutdown assuming that none of the withdrawn control rods can be inserted.

Pressure boundary - maintains the integrity of the reactor coolant pressure boundary.

Containment isolation - provides containment isolation for those portions of the system that interface with the primary containment.

Credited in regulated event(s) – provides reactivity control credited in mitigation of the ATWS event.

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.2-8 Component Groups Requiring Aging Management Review - Standby Liquid Control System

Component Group	Component Intended Function	Aging Management Ref
Accumulators	Pressure Boundary	3.3.1.5, 3.3.2.3, 3.3.2.40
Closure Bolting	Pressure Boundary	3.1.1.1, 3.1.2.1, 3.1.2.2, 3.3.1.22
Dampeners (Quad Cities only)	Pressure Boundary	3.3.1.23, 3.3.2.40, 3.3.2.47
NSR Vents or Drains, Piping and Valves (attached support)	Structural Integrity (attached)	3.3.2.130
Piping and Fittings	Pressure Boundary	3.1.1.1, 3.1.1.15, 3.3.1.5, 3.3.1.23, 3.3.2.40, 3.3.2.42, 3.3.2.164, 3.3.2.165
Piping and Fittings (attached support)	Structural Integrity (attached)	3.3.1.5, 3.3.1.8, 3.3.1.23, 3.3.1.25, 3.3.2.40, 3.3.2.42, 3.3.2.165
Pumps	Pressure Boundary	3.3.1.23, 3.3.2.40, 3.3.2.185
Sight Glasses	Pressure Boundary	3.3.2.36, 3.3.2.203
Tanks	Pressure Boundary	3.3.1.23, 3.3.2.40, 3.3.2.219

Table 2.3.2-8 Component Groups Requiring Aging Management Review - Standby Liquid Control System (Continued)

Component Group	Component Intended Function	Aging Management Ref
Thermowells	Pressure Boundary	3.3.1.23, 3.3.2.40, 3.3.2.227
Tubing	Pressure Boundary	3.1.1.15, 3.3.1.23, 3.3.2.40, 3.3.2.253
Tubing (attached support)	Structural Integrity (attached)	3.3.1.23, 3.3.2.40, 3.3.2.253
Valves	Pressure Boundary	3.1.1.1, 3.1.1.15, 3.3.1.23, 3.3.2.40, 3.3.2.42, 3.3.2.293, 3.3.2.294
Valves (attached support)	Structural Integrity (attached)	3.3.1.5, 3.3.1.23, 3.3.1.25, 3.3.2.40, 3.3.2.272, 3.3.2.294

Aging management review results for the standby liquid control system are provided in Section 3.1 for the reactor coolant pressure boundary functions and Section 3.3 for the additional standby liquid control system functions.

2.3.2.9 Standby Gas Treatment System

System Purpose

The standby gas treatment system (SBGTS) processes and controls the intentional exhaust of radioactive material from the reactor building spaces to the environment during a design basis accident. This ensures that the requirements of 10 CFR Part 100 are met. The system is sized to maintain a small negative pressure in the reactor building relative to the atmosphere outside of the building. The SBGTS also processes radioactive effluent from the high pressure coolant injection (HPCI) gland seal exhaust subsystem (Dresden only) during HPCI operation. The SBGTS also receives effluent from the primary containment (drywell and suppression chamber) during vent and purge operation when necessary.

System Operation

The SBGTS consists of two, 100% capacity treatment trains. During normal operation one train is selected as the primary and the other is placed in standby. The SBGTS receives effluent from the following three sources; the reactor building, primary containment (drywell and suppression chamber), and the HPCI gland exhauster (Dresden only). Each train consists of piping that routes effluent flow to a demister, electrical heater, roughing prefilter, high efficiency filter, carbon iodine adsorber, high efficiency afterfilter, fan, associated valves, and instrumentation. Filters remove radioactive particles and charcoal adsorbers remove radioactive halogens (noble gases not included). Each SBGT train is capable of maintaining reactor building pressure under isolation conditions to prevent ground level escape of airborne activity. Exhaust from each SBGTS train is routed through piping to the reactor building ventilation chimney (evaluated separately). Process moisture removed by the demister is drained to the reactor building equipment drain tank (evaluated with reactor building equipment and floor drains). Both SBGTS trains are connected by a cross-tie line containing a restricting orifice and isolation damper. During operation, the primary train provides cooling flow to the standby train through the cross-tie line with air from the reactor building atmosphere at Dresden and from the Turbine Building at Quad Cities. The primary SBGTS train fan provides the motive force for both the treated flow through the primary train and the cooling flow through the standby train.

System Evaluation Boundary

The SBGTS evaluation boundary begins with intake lines from the reactor building, primary containment (drywell and suppression chamber) and the exhaust line from the HPCI gland exhauster (Dresden only). The evaluation boundary continues with the following components from each SBGT train: inlet damper to the demister, electric heater, rough prefilter, high efficiency filter, carbon iodine adsorbers, test orifice, high efficiency afterfilter, to the flow control damper, SBGTS fan, backdraft damper, outlet damper, through the total flow transmitter flow element. Also included in the SBGTS evaluation boundary are the demister drain lines and loop seal to the reactor building equipment drain tank and the cross-tie line connecting both SBGTS trains. The boundary concludes with the exhaust line to the ventilation chimney. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the SBGT boundary. The system evaluation boundary also includes the SBGTS to high radiation sampling system (HRSS) isolation valves, interconnecting piping and their associated components. These valves isolate the safety related SBGTS system piping from the non-safety related HRSS.

UFSAR References

Dresden Station UFSAR Section(s): 6.5.3

Quad Cities Station UFSAR Section(s): 6.5.3

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-49, LR-DRE-M-1235, and LR-DRE-M-1240

Quad Cities Station:

LR-QDC-M-44, LR-QDC-M-1057, and LR-QDC-M-1062

System Intended Functions

Filtration - filters the exhaust air to remove radioactive gases and particulates that may be present in the secondary containment prior to discharge to the environment following a design basis accident.

Containment - maintains a small negative pressure in the reactor building under isolation conditions.

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Credited in regulated event(s) – The system contains components that are relied upon for compliance with 10 CFR 50.49, (EQ).

Component Groups Requiring Aging Management Review

Table 2.3.2-9 Component Groups Requiring Aging Management Review – Standby Gas Treatment System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.2.2.6, 3.2.2.118
Duct (includes piping and fittings)	Pressure Boundary	3.2.1.3
Doors, Closure Bolts, Equip Frames (includes inlet bells, restricting orifices, and tubing)	Pressure Boundary	3.2.1.3
Doors, Closure Bolts, Equip Frames (includes restricting orifices and exhaust headers)	Throttle	3.2.1.3, 3.2.2.127

Table 2.3.2-9 Component Groups Requiring Aging Management Review – Standby Gas Treatment System (Continued)

Component Group	Component Intended Function	Aging Management Ref
Fan Housings	Pressure Boundary	3.2.1.3
Filters/Strainers (Dresden only) (includes demisters)	Pressure Boundary	3.2.2.31
Flex Collars, Doors and Damper Seals	Pressure Boundary	3.2.1.7
Housings and Supports (includes filters)	Pressure Boundary	3.2.1.3
Manifolds	Pressure Boundary	3.2.2.54
NSR Vents or Drains, Piping and Valves (attached support) (includes tubing)	Structural Integrity (attached)	3.2.2.55
Seals	Pressure Boundary	3.2.1.7
Tubing	Pressure Boundary	3.2.2.95
Tubing (attached support)	Structural Integrity (attached)	3.2.2.95
Valves	Pressure Boundary	3.2.2.101, 3.2.2.103, 3.2.2.108, 3.2.2.114, 3.2.2.120
Valves (attached support)	Structural Integrity (attached)	3.2.2.108, 3.2.2.120

Aging management review results for the standby gas treatment system are provided in Section 3.2.

2.3.2.10 Automatic Depressurization System

System Purpose

The automatic depressurization system (ADS) is designed to act as a backup to the high pressure coolant injection (HPCI) system and perform the function of vessel depressurization for all "small breaks" inside the primary containment or "small unisolable breaks" outside the containment. ADS is one of the emergency core cooling systems (ECCS) designed to operate with low pressure coolant injection (LPCI) and core spray (CS) to protect the reactor vessel/fuel in situations where the vessel is losing coolant. At Quad Cities, LPCI is an operational mode of the RHR system. For small breaks the vessel is depressurized in sufficient time to allow CS or LPCI to provide adequate core cooling. For large breaks the vessel depressurizes through the break without assistance.

System Operation

ADS will automatically and rapidly depressurize the reactor pressure vessel (evaluated with reactor pressure vessel) under certain accident conditions. When the logic circuitry detects an accident condition along with a LPCI pump (evaluated with LPCI system for Dresden and RHR system for Quad Cities) or CS pump (evaluated with CS) running, the circuit sends signals that actuate the reactor vessel's five relief valves to perform a rapid vessel depressurization. One of the five relief valves is a safety / relief valve. Each relief valve (evaluated with main steam) is connected to a main steam line (evaluated with main steam). When a relief valve opens, it discharges steam to a tail pipe (evaluated with main steam), which directs the steam below the surface of the water in the suppression chamber (evaluated with primary containment structure), through a tee quencher (evaluated with primary containment structure).

System Evaluation Boundary

The system evaluation boundary is comprised of the logic relays, timers and instrumentation that received process signal input and provide actuation signals to the relief valves actuated by ADS. The relief valves and the safety / relief valve, their tail pipes and vacuum breaker valves, related solenoids, pressure controllers, position switches, and pneumatic air components associated with the safety / relief valve are evaluated with the main steam system.

UFSAR References

Dresden Station UFSAR Section(s): 6.3.1, 6.3.2

Quad Cities UFSAR Section(s): 6.3.1, 6.3.2

License Renewal Boundary Diagram References

Dresden Station: None

Quad Cities Station: None

System Intended Functions

Core cooling – provides emergency core cooling by receiving process signal inputs and providing, through appropriate relay logic, actuation signal outputs to relief valves assigned to the main steam system. Opening of the relief valves actuated by the ADS depressurizes the reactor pressure vessel to support LPCI and low pressure core spray operation.

Credited in regulated event(s) – provides emergency core cooling credited in mitigation of the Appendix R fire protection and SBO events.

Component Groups Requiring Aging Management Review

Table 2.3.2-10 Component Groups Requiring Aging Management Review – Automatic Depressurization System

Dresden and Quad Cities design basis documents treat the ADS relief valves and associated piping, solenoids, pressure controllers and position switches as components of the main steam system. These mechanical components of the ADS subject to an aging management review are included as components of the main steam system in this license renewal application.

2.3.2.11 Anticipated Transient Without Scram System

System Purpose

The anticipated transient without scram (ATWS) system provides the instrumentation and logic necessary for control rod insertion and recirculation pump trips to mitigate the effects of an ATWS situation.

System Operation

ATWS events are beyond design basis accidents. They are those low probability events in which an anticipated transient occurs and is not followed by an automatic reactor shutdown (scram) when required. The failure of the reactor to scram quickly during these transients can lead to unacceptable reactor coolant system pressures and to fuel damage. The ATWS system logic, when energized, will reposition alternate rod insertion (ARI) solenoid valves to depressurize the scram air header for control rod insertion and, in the event of an automatic initiation, trip the recirculation pump motor generator field breakers. The ATWS system is divided into two separate divisions. The trip logic circuitry of each ATWS system division is capable of performing the required mitigating action (tripping both recirculation pump motor generator field breakers and actuating three of the six ARI valves). The ATWS system will automatically initiate upon signals of high reactor pressure or low-low reactor water level. The ATWS system can also be initiated manually.

System Evaluation Boundary

The ATWS system evaluation boundary encompasses those process lines and valves which lead to the ATWS sensors, including the instrument drain valves. It includes the pressure and level sensors that actuate the system, the system logic circuitry up to the ARI solenoid valves, and the recirculation pump generator field breakers and trip coils. The ARI solenoid valves and their related components are evaluated as part of the control rod drive system. The recirculation pump generator field breakers and field breaker trip coils are evaluated as part of the recirculation, recirc. flow control and MG set system.

UFSAR References

Dresden Station UFSAR Section(s): 7.8

Quad Cities Station UFSAR Section(s) 7.8

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-26-1 and LR-DRE-M-357-1

Quad Cities Station:

LR-QDC-M-35-1, LR-QDC-M-35-3, and LR-QDC-M-77-3

System Intended Functions

Reactivity control – provides an alternate means of control rod insertion and trips reactor recirculation pump M-G set field breakers.

Credited in regulated event(s) – provides reactivity control credited in mitigation of ATWS events.

Component Groups Requiring Aging Management Review

Table 2.3.2-11 Component Groups Requiring Aging Management Review – Anticipated Transients Without Scram System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.2.1.14
Piping and Fittings	Pressure Boundary	3.2.1.13, 3.2.2.22
Valves	Pressure Boundary	3.2.1.2, 3.2.1.4, 3.2.1.12, 3.2.1.13, 3.2.2.22, 3.2.2.137

Aging management review results for the anticipated transients without scram system are provided in Section 3.2.

2.3.3 Auxiliary Systems

This section of the application addresses scoping and screening results for the following systems:

- Refueling equipment
- Shutdown cooling system (Dresden only)
- Control rod drive hydraulic (CRDH) system
- Reactor water cleanup (RWCU) system
- Fire protection system
- Emergency diesel generator (EDG) and auxiliaries
- HVAC - Main control room
- HVAC – Reactor building
- ECCS corner room HVAC
- Station blackout building HVAC
- Station blackout (SBO) system (diesel and auxiliaries)
- Diesel generator (DG) cooling water system
- Diesel fuel oil system
- Process sampling system
- Carbon dioxide system
- Service water (SW) system
- Reactor building closed cooling water (RBCCW) system
- Turbine building closed cooling water (TBCCW) system
- Demineralizer water makeup system
- Residual heat removal service water (RHRSW) system (Quad Cities only)
- Containment cooling service water (CCSW) system (Dresden only)
- Ultimate heat sink (UHS)
- Fuel pool cooling (FPC) system and filter demineralizer system
- Plant heating system
- Containment atmosphere monitoring (CAM) system
- Nitrogen containment atmosphere dilution (NCAD) system
- Drywell nitrogen inerting (DNI) system
- Safe shutdown makeup pump (SSMP) system (Quad Cities only)

2.3.3.1 Refueling Equipment

System Purpose

The purpose of the fuel handling system is to receive and transfer nuclear fuel in a manner that precludes inadvertent criticality and to provide equipment for handling both new and irradiated fuel. To achieve this purpose the fuel handling equipment is designed to handle fuel assemblies and other reactor components.

System Operation

The major components of the system are the refueling platform and the reactor building overhead crane (evaluated with cranes and hoists). The refueling platform consists of a track mounted bridge which includes an electric motor drive and the controls, instrumentation, and service facilities required to support the operation of the main fuel grapple or the handling equipment used with the auxiliary hoists. The main fuel grapple consists of a mast and head assembly, and accessories. During refueling operations, the reactor cavity and the dryer-separator pit are flooded to the normal level of the fuel storage pool and the fuel pool storage pool gates are removed. Spent fuel is removed from the reactor and placed in racks (evaluated with the reactor building structure) in the fuel storage pool using the main grapple. The same equipment is used to transfer the new fuel from the fuel storage pool to the reactor. Channeling and dechanneling of irradiated fuel is performed using the fuel prep machines. The racks in which spent fuel assemblies are placed are designed and arranged to ensure sub-criticality in the pool.

System Evaluation Boundary

The refueling equipment system boundary consists of the refueling equipment required to transfer fuel assemblies and reactor components within the fuel storage pool, and between the pool and the reactor. The major component of the system is the refueling platform assembly, consisting of the refueling platform, fuel grapple, and associated equipment. The refueling platform bridge includes a walkway, railings and a trolley mounted control cab, a main grapple hoist, the adjacent frame mounted auxiliary hoist, a reverse mounted monorail auxiliary hoist, a hinged jib arm power winch, and the reels, drives, pulleys, and sheaves required for the hoist cables and the service air lines from the self contained, refueling platform mounted air compressor. The bridge air system includes the compressor, air receiver, shutoff valves, solenoid valves, air hose retrieval assist drives, and quick disconnect fittings. The system also includes the fuel prep machines and the fuel pool gates. Cranes and hoists used to transport fuel assemblies and reactor components, other than those associated with the refueling platform, are evaluated with the reactor building structure, as are the new and spent fuel racks. The inboard main steam line plugs, vents, and regulators associated with the reactor vessel system are evaluated with the refueling equipment system.

UFSAR References

Dresden Station UFSAR Section: 9.1.4

Quad Cities Station UFSAR Section: 9.1.4

License Renewal Boundary Diagram References

There are no boundary diagrams associated with this system.

System Intended Functions

Maintain structural integrity - maintains structural integrity to prevent collapse of the platform onto the spent fuel storage racks or the reactor core.

Preclude inadvertent criticality – provides interlocks to preclude inadvertent criticality.

Component Groups Requiring Aging Management Review

Table 2.3.3-1 Component Groups Requiring Aging Management Review – Refueling Equipment System

Component Group	Component Intended Function	Aging Management Ref
Cranes	Structural Support	3.3.1.14
Fuel Grapples	Structural Support	3.3.2.74
Fuel Pool Gates	Pressure Boundary	3.3.2.75
Fuel Preparation Machines	Structural Support	3.3.2.76

Aging management review results for the refuel equipment system are provided in Section 3.3.

2.3.3.2 Shutdown Cooling System (Dresden Only)

System Purpose

The shutdown cooling system at Dresden Station provides cooling of the reactor water when the temperature and pressure in the reactor fall below the point at which the main condenser can no longer be used as a heat sink following reactor shutdown. The system can also be used to help cool the fuel pool during refueling outages and to heat reactor water with steam from the plant heating system during startup from cold shutdown.

System Operation

The shutdown cooling system consists of three partial capacity cooling loops, each containing a pump, a heat exchanger, and associated piping, valves and instrumentation. The system takes suction from either reactor recirculation loop (evaluated with the reactor recirculation system), delivers the flow through each of the three separate cooling loops, and then directs it into either of the LPCI injection lines (evaluated with the low pressure coolant injection system). Capability also exists to permit flow from both reactor recirculation loops to both LPCI injection lines simultaneously. When used to augment fuel pool cooling (evaluated with the fuel pool cooling and demineralizer system), only one of the cooling loops is required. Each cooling loop is provided with a minimum flow valve to return pump discharge flow to the pump suction. The system heat exchangers are cooled by water from the reactor building closed cooling water system (evaluated with the reactor building closed cooling water system) in the cooling mode and heated by steam from the plant heating system (evaluated with the plant heating system) in the heating mode. Provision is also made for chemical sampling, clean-up via the RWCU system (evaluated with the reactor water cleanup system), and system drainage to the reactor building equipment drain system.

System Evaluation Boundary

The shutdown cooling system evaluation boundary begins with the system inlet MOV's that receive primary coolant from the reactor recirculation piping. The boundary continues through each of the system pumps, heat exchangers, and discharge lines to the outlet MOV's that direct the cooled primary coolant back to the reactor via the LPCI injection lines. All associated piping, components and instrumentation contained within flow paths and subsystems described above are included in the shutdown cooling system evaluation boundary. Each cooling loop includes a minimum flow line.

UFSAR References

Dresden Station UFSAR Section(s): 5.4.7

Quad Cities Station UFSAR Section(s): Not Applicable

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-29-1, LR-DRE-M-32, LR-DRE-M-39, LR-DRE-M-357-2, LR-DRE-M-369, and LR-DRE-M-363

Quad Cities Station: Not Applicable

System Intended Functions

Pressure boundary - maintains the integrity of the reactor coolant pressure boundary.

Primary containment isolation – provides containment isolation for those portions of the system that interface with the primary containment.

Credited in regulated event(s) – provides heat removal sufficient to achieve and maintain cold shutdown conditions during normal operation. This core cooling function is credited in mitigation of the Appendix R fire event. The system contains components that are relied upon for compliance with 10 CFR 50.49, (EQ).

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.3-2 Component Groups Requiring Aging Management Review - Shutdown Cooling System (Dresden only)

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting (Dresden only)	Pressure Boundary	3.3.1.22
Dampeners (Dresden only)	Pressure Boundary	3.3.1.8, 3.3.1.25, 3.3.2.40
Filters/Strainers (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.8
Filters/Strainers (Dresden only)	Filter	3.3.1.8
Heat Exchangers (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.26, 3.3.2.86, 3.3.2.87, 3.3.2.303
Heat Exchangers (Dresden only)	Heat Transfer	3.3.2.83, 3.3.2.114
NSR Vents or Drains, Piping and Valves (attached support) (Dresden only)	Structural Integrity (attached)	3.3.2.130
Piping and Fittings (Dresden only)	Pressure Boundary	3.1.1.11, 3.1.1.15, 3.3.1.3, 3.3.1.5, 3.3.1.8, 3.3.1.13, 3.3.1.25, 3.3.2.27, 3.3.2.40, 3.3.2.42
Pumps (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.8

Table 2.3.3-2 Component Groups Requiring Aging Management Review - Shutdown Cooling System (Dresden only) (Continued)

Component Group	Component Intended Function	Aging Management Ref
Restricting Orifices (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.8, 3.3.1.25, 3.3.2.40
Restricting Orifices (Dresden only)	Throttle	3.3.1.8
Sight Glasses (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.8
Sight Glasses (attached support) (Dresden only)	Structural Integrity (attached)	3.3.2.204
Thermowells (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.8
Valves (Dresden only)	Pressure Boundary	3.1.1.11, 3.1.1.15, 3.3.1.5, 3.3.1.8, 3.3.1.13, 3.3.1.25, 3.3.2.27, 3.3.2.40, 3.3.2.293

Aging management review results for the shutdown cooling system are provided in Section 3.1 for the reactor coolant pressure boundary functions and Section 3.3 for the additional shutdown cooling functions.

2.3.3.3 Control Rod Drive Hydraulic System

System Purpose

The control rod drive hydraulic (CRDH) system controls changes in reactivity by incrementally positioning the control rods in response to signals from the reactor manual control system. The system is also used to shut down the reactor quickly by rapidly inserting control rods into the core in response to a manual or automatic signal.

System Operation

The CRDH system is made up of supply pumps, filters, strainers, control valves, and associated instrumentation and controllers. It provides water at the required pressures to the hydraulic control units (HCUs) for cooling and all types of required control rod (evaluated with control blades) motion. The CRDH system allows control rod withdrawal or insertion at a limited rate, one rod at a time, for power level control and flux shaping during reactor operation. Stored energy available from gas (nitrogen) charged accumulators and from reactor pressure provides hydraulic power for rapid simultaneous insertion (scram) of all control rods for reactor shutdown. The hydraulic system is arranged so that the equipment common to each control rod drive (CRD) is packaged in modular form into HCUs, one HCU module to each drive. The HCUs are arranged into two banks, each of which has its own scram discharge volume (SDV), which consists of a scram discharge header and an instrument volume. The SDV is used to limit the loss of and contain the reactor vessel water from all the drives during a scram. Each CRD has its own separate control and scram devices. Alternate rod insertion (ARI) valves are supplied to provide an alternate means of initiating control rod insertion during an ATWS scram event (evaluated with the anticipated transients without scram system). Under normal operation, the CRDH system supplies unheated condensate by one of two system pumps to hydraulically position the control rods. These pumps take their suction from the condensate system (evaluated with the condensate and condensate storage system) at low pressure and direct it through the drive water filters. The discharge from the filters supply the reactor recirculation pump seal water supply header, the accumulators via the charging header, the HCUs via the drive water header, and the CRDs via the cooling water header. The pumps are provided with a minimum flow line to the contaminated condensate storage tank (CCST). The reactor recirculation pump seal water supply header supplies filtered and cooled water to the purge the reactor recirculation pump seals. The reactor recirculation pump seal water supply isolation check valves provide a pressure retaining boundary for the reactor coolant system. The charging header supplies pressurized water to maintain the accumulators charged and ready for service in the event of a scram. The drive water header provides the CRDs with motive force for moving the control rods. This header also supplies a continuous low flow to the reactor vessel water level instrumentation system (RVWLIS) backfill system (evaluated with the nuclear boiler instrumentation system). Filtered control air is provided to CRDH system air operated valves from the instrument air system. Provision is also made (at Dresden) to supply system flow to the reactor recirculation pump seals for hydro purposes.

System Evaluation Boundary

The CRDH system suction side evaluation boundary begins with the connection to the condensate reject line to the CCST. The boundary continues through the CRD pumps, the drive water filters, and branches to the reactor recirculation pump seal supply header, to the accumulators via the charging header, to the HCUs via the drive water header, and to the CRDs via the cooling water header. The evaluation boundary includes the suction piping, discharge piping, minimum flow line, filters, CRD pumps, HCUs, HCU manifolds, and the CRDs. Included within the boundary are the reactor recirculation pump seal water supply isolation check valves, the scram discharge header and instrument volume and the supply lines to the RVWLIS backfill system. The air supply evaluation boundary starts at the parallel instrument air supply lines to the CRD air filters and includes the filters and all piping, instrumentation, and valves necessary to supply air to the CRD air operated valves. Included within the evaluation boundary is the accumulator nitrogen charging system, which includes the nitrogen cylinders, header piping, piping, instrumentation, and valves from the cylinders to the HCU accumulators. At Dresden, the RVLIS backfill system supply header isolation valves, and the system valves on the CRD hydro supply line to the reactor recirculation pump seals are included in the evaluation boundary. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the control rod drive hydraulic system boundary.

UFSAR References

Dresden Station UFSAR Section(s): 4.6.3

Quad Cities Station UFSAR Section(s): 4.6.3

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE-M-26-2, LR-DRE-M-26-3, LR-DRE-M-34-1, LR-DRE-M-34-2, LR-DRE-M-35-1, LR-DRE-M-39, LR-DRE-M-177-4, LR-DRE-M-357-2, LR-DRE-M-357-3, LR-DRE-M-365-1, LR-DRE-M-365-2, LR-DRE-M-366, LR-DRE-M-369, and LR-DRE-M-419-4.

Quad Cities Station:

LR-QDC-M-41-1, LR-QDC-M-41-2, LR-QDC-M-41-3, LR-QDC-M-41-4, LR-QDC-M-83-1, LR-QDC-M-83-2, LR-QDC-M-83-3, and LR-QDC-M-83-4.

System Intended Functions

Reactivity control – provides a rapid shutdown (scram) of the reactor under appropriate conditions.

Pressure boundary - maintains pressure boundary to support integrity of the reactor coolant pressure boundary and to support in-scope pressure boundaries at interfaces with other in-scope systems.

Credited in regulated event(s) – provides scram discharge volume vent and drain isolation valves which are credited to remain closed in the Appendix R fire event and provides alternate rod insertion capability which is credited in the anticipated transient without scram event. At Dresden only, CRDH system water supply to the vessel is credited in the Appendix R fire event. Also at Dresden only, the system contains components that are relied upon for compliance with 10 CFR 50.49, (EQ).

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.3-3 Component Groups Requiring Aging Management Review - Control Rod Drive Hydraulic System

Component Group	Component Intended Function	Aging Management Ref
Accumulators	Pressure Boundary	3.3.1.5, 3.3.2.1, 3.3.2.5, 3.3.2.40
Closure Bolting	Pressure Boundary	3.3.1.22
Dampeners (Quad Cities only)	Pressure Boundary	3.3.1.5, 3.3.1.8, 3.3.1.25, 3.3.2.40
Dampeners (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.3.1.5, 3.3.1.8, 3.3.1.25, 3.3.2.40
Filters/Strainers	Pressure Boundary	3.3.1.25, 3.1.2.10, 3.1.2.11, 3.3.2.40
Filters/Strainers	Filter	3.3.1.25, 3.1.2.10, 3.1.2.11
Flow Elements (spatial interaction)	Leakage Boundary (spatial)	3.3.1.5, 3.3.1.8, 3.3.1.25, 3.3.2.40
Flow Elements (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.8, 3.3.1.25, 3.3.2.40
NSR Vents or Drains, Piping and Valves (attached support)	Structural Integrity (attached)	3.3.2.130
Piping and Fittings (includes dampeners and tubing)	Pressure Boundary	3.3.1.5, 3.3.1.8, 3.3.1.17, 3.3.1.25, 3.1.2.25, 3.1.2.26, 3.3.2.40, 3.3.2.42, 3.3.2.146, 3.3.2.164, 3.3.2.167
Piping and Fittings (spatial interaction)	Leakage Boundary (spatial)	3.3.1.5, 3.3.1.8, 3.3.1.25, 3.3.2.40
Piping and Fittings (attached support)	Structural Integrity (attached)	3.3.1.5, 3.3.1.8, 3.3.1.25, 3.3.2.40
Pumps (Dresden only)	Pressure Boundary	3.3.1.25, 3.3.2.40
Restricting Orifices (Dresden only)	Pressure Boundary	3.3.1.8, 3.3.1.25, 3.3.2.40
Rupture Discs	Pressure Boundary	3.3.2.40, 3.3.2.194
Tanks (includes accumulators)	Pressure Boundary	3.3.1.5, 3.3.1.8, 3.1.2.34, 3.1.2.35, 3.3.2.40, 3.3.2.216
Tubing	Pressure Boundary	3.3.1.5, 3.3.1.8, 3.3.1.17, 3.3.1.25, 3.3.2.40, 3.3.2.242, 3.3.2.248

Table 2.3.3-3 Component Groups Requiring Aging Management Review - Control Rod Drive Hydraulic System (Continued)

Component Group	Component Intended Function	Aging Management Ref
Valves	Pressure Boundary	3.1.2.52, 3.1.2.53, 3.3.1.5, 3.3.1.8, 3.3.1.17, 3.3.1.25, 3.3.2.23, 3.3.2.40, 3.3.2.42, 3.3.2.260, 3.3.2.262, 3.3.2.274, 3.3.2.293, 3.3.2.296
Valves (spatial interaction)	Leakage Boundary (spatial)	3.3.1.8, 3.3.1.25, 3.3.2.40
Valves (attached support)	Structural Integrity (attached)	3.3.1.25, 3.3.2.40, 3.3.2.296

Aging management review results for the control rod drive hydraulic system are provided in Section 3.1 for the reactor coolant pressure boundary functions and Section 3.3 for the additional control rod drive hydraulic system functions.

2.3.3.4 Reactor Water Cleanup System

System Purpose

The purpose of the reactor water cleanup system is to: remove insoluble, waterborne activation products from reactor coolant; prevent soluble inorganic impurities (i.e., chlorides) from concentrating in the reactor coolant and exceeding specified water quality limits; reduce beta and gamma radiation sources in the reactor coolant resulting from the presence of corrosion and fission products; and remove water from the reactor coolant system at reduced activity levels during startup and shutdown.

System Operation

The RWCU system consists of pumps, regenerative and non-regenerative heat exchangers, demineralizers, filters (Quad Cities only), containment isolation valves, and associated piping, valves, instrumentation and controls. The system takes suction from the reactor recirculation system at the shutdown cooling system connection at Dresden (evaluated with the shutdown cooling system) and the reactor recirculation pump suction connection at Quad Cities (evaluated with the reactor recirculation system) and the reactor vessel bottom drain connection (evaluated with the nuclear boiler instrumentation system). From these two suction sources, reactor water impurities are removed by directing the flow through the system's major components, piping and supporting components back to the reactor vessel via the feedwater system at Dresden (evaluated with the feedwater system) and RCIC system at Quad Cities (evaluated with the reactor core isolation cooling system) connections. The regenerative heat exchangers transfer heat from the water leaving the reactor to the water returning to the reactor. The non-regenerative heat exchangers are cooled by water from the reactor building closed cooling water system (evaluated with reactor building closed cooling water system). At Dresden, the portion of the system downstream of the pressure reducing valve is at low pressure. Flow through the entire system at Quad Cities is at reactor pressure. Isolation valves are provided for system process lines that penetrate primary containment. The system has automatic isolation features that respond to indications of a system high energy line break (HELB), actuation of the standby liquid control system (evaluated with the standby liquid control system), upon receipt of a low reactor water level signal, or high temperature downstream of the non-regenerative heat exchangers (to prevent resin damage). At Dresden, the system will also automatically isolate on a system high pressure condition, or an auxiliary pump high cooling water outlet temperature condition.

System Evaluation Boundary

The RWCU system evaluation boundary begins with suction from the shutdown cooling system connection (at Dresden) and the reactor recirculation pump suction connection (at Quad Cities) and the reactor vessel bottom drain connection. The boundary continues through containment isolation valves, the regenerative and non-regenerative heat exchangers, the pressure reducing valve (Dresden only), the demineralizer pumps, filters (Quad Cities only), and demineralizers, then back through the regenerative heat exchanger and through the return to reactor check valve to the feedwater system (at Dresden) and RCIC system (at Quad Cities) connections. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the RWCU system boundary.

UFSAR References

Dresden Station UFSAR Section(s): 5.4.8

Quad Cities Station UFSAR Section(s): 5.4.8

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE-M-14, LR-DRE-M-20, LR-DRE-M-26-2, LR-DRE-M-30, LR-DRE-M-32, LR-DRE-M-48, LR-DRE-M-347, LR-DRE-M-353, LR-DRE-M-357-2, LR-DRE-M-361, LR-DRE-M-363, and LR-DRE-M-372.

Quad Cities Station:
LR-QDC-M-35-1, LR-QDC-M-35-2, LR-QDC-M-47-1, LR-QDC-M-47-2, LR-QDC-M-77-1, LR-QDC-M-77-2, LR-QDC-M-88-1, and LR-QDC-M-88-2.

System Intended Functions

Pressure boundary - maintains the integrity of the reactor coolant pressure boundary.

Primary containment isolation – provides containment isolation for those portions of the system that interface with the Primary Containment.

Supports ESF function(s) – supports the ESF function of the standby liquid control system by shutdown of RWCU pumps and closure of RWCU valves to prevent dilution or removal of the injected boron.

Credited in regulated event(s) – credited in evaluation of the Appendix R fire and in the anticipated transient without scram events. The system also contains components relied upon for compliance with 10CFR50.49 (EQ).

Preclude adverse effects on safety-related SSCs – non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.3-4 Component Groups Requiring Aging Management Review - Reactor Water Cleanup System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.1.1.1, 3.3.1.22
NSR Vents or Drains, Piping and Valves (attached support)	Structural Integrity (attached)	3.3.2.130
Piping and Fittings	Pressure Boundary	3.1.1.1, 3.1.1.15, 3.3.1.3, 3.3.1.5, 3.3.1.24, 3.3.2.40, 3.3.2.42, 3.3.2.140
Piping and Fittings (attached support)	Structural Integrity (attached)	3.3.1.5, 3.3.1.8, 3.3.2.40, 3.3.1.25
Piping and Fittings (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.3.1.5, 3.3.1.8, 3.3.1.25, 3.3.2.40
Piping and Fittings (small bore)	Pressure Boundary	3.1.1.5, 3.3.2.40, 3.3.2.42
Sight Glasses (attached support) (Dresden only)	Structural Integrity (attached)	3.3.2.204
Valves	Pressure Boundary	3.1.1.1, 3.1.1.9, 3.1.1.15, 3.3.1.5, 3.3.2.40, 3.3.2.42, 3.3.2.270
Valves (attached support)	Structural Integrity (attached)	3.3.2.292, 3.3.2.40
Valves (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.3.2.292, 3.3.2.40

Aging management review results for the reactor water cleanup system are provided in Section 3.1 for the reactor coolant pressure boundary functions and 3.3 for the additional RWCU functions.

2.3.3.5 Fire Protection System

System Purpose

The fire protection system detects and suppresses fires and provides an alternate source of makeup water to various critical locations. Fire barriers prevent the spread of fire from one space to another.

System Operation

The fire protection system includes the diesel driven fire pumps, the fire protection headers to the various wet pipe, preaction and deluge systems as well as associated valves, instruments, hose reels and hydrants. Two fire pumps are provided in the 1/2 intake structure (cribhouse) at Quad Cities. One fire pump each is provided in the 2/3 cribhouse and the Unit 1 cribhouse at Dresden. The system takes suction from the associated cribhouse suction bays and provides water to its various components for fire suppression. Provisions are also made for the system to serve as an alternate makeup water source to the isolation condenser (at Dresden, evaluated with the isolation condenser) and the safe shutdown makeup pump system (at Quad Cities, evaluated with the safe shutdown makeup pump system). The system also includes fire dampers, the halon suppression system and the fire computer system, which includes smoke detectors, heat sensors, pressure/flow sensors, and actuation devices for preaction systems, and fire doors (evaluated with the associated structures). These systems detect, suppress, and prevent the spread of fires in various critical locations.

System Evaluation Boundary

The fire protection system evaluation boundary begins with the suction of the diesel driven fire pumps from the associated cribhouse suction bays. It continues through the common fire protection header to the various wet pipe, preaction and deluge systems as well as hose reels, hydrants, and sprinkler heads. The evaluation boundary extends to include its interface with the service water system, the isolation condenser system (at Dresden) and the safe shutdown makeup pump system (at Quad Cities). The evaluation boundary also includes fire dampers, the halon suppression system and the fire computer system, which includes smoke detectors, heat sensors, pressure/flow sensors, and actuation devices for preaction systems. The fire dampers of the in-scope HVAC systems are evaluated with those systems. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the fire protection system boundary..

UFSAR References

Dresden Station UFSAR Section(s): 9.5.1

Quad Cities Station UFSAR Section(s): 9.5.1

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE-M-23-1, LR-DRE-M-23-2, LR-DRE-M-23-3, LR-DRE-M-23-4, LR-DRE-M-23-5, LR-DRE-M-28, LR-DRE-M-359, LR-DRE-M-375-2, LR-DRE-M-375-3, LR-DRE-M-787, LR-DRE-M-1305, and LR-DRE-M-4204.

Quad Cities Station:
LR-QDC-M-27-1, LR-QDC-M-27-2, LR-QDC-M-27-3, LR-QDC-M-27-4, LR-QDC-M-27-5, LR-QDC-M-29-1, and LR-QDC-M-3030.

System Intended Functions

Credited in regulated events - provides methods to detect, suppress, contain, and monitor fire events in compliance with the requirements of 10CFR50.48. For Dresden only, the fire protection system provides a backup source of makeup water to the isolation condenser. For Quad Cities only, the fire protection system provides a backup source of makeup water for reactor pressure vessel level control and drywell flooding and a backup cooling water supply to the safe shutdown makeup pump room coolers in event of a fire.

Component Groups Requiring Aging Management Review

Table 2.3.3-5 Component Groups Requiring Aging Management Review - Fire Protection System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.3.1.22, 3.3.2.17, 3.3.2.18, 3.3.2.19, 3.3.2.20
Filters/Strainers	Pressure Boundary	3.3.1.5, 3.3.1.17, 3.3.1.19, 3.3.2.23, 3.3.2.56, 3.3.2.300
Filters/Strainers	Filter	3.3.1.5, 3.3.1.17, 3.3.1.19, 3.3.2.56
Fire Dampers	Pressure Boundary	3.3.1.18
Fire Dampers	Fire Barrier	3.3.1.18
Fire Hydrants	Pressure Boundary	3.3.1.19, 3.3.2.61, 3.3.2.32
Fire Wrap	Fire Barrier	3.3.2.63
Mufflers	Pressure Boundary	3.3.1.5
Penetration Seals	Fire Barrier	3.3.1.18
Penetration Seals	Flood Barrier	3.3.1.18

Table 2.3.3-5 Component Groups Requiring Aging Management Review - Fire Protection System (Continued)

Component Group	Component Intended Function	Aging Management Ref
Piping and Fittings (includes flex hoses, hose reels, hoses, nozzles, tubing, sprinklers, and gaskets of buried fire mains)	Pressure Boundary	3.3.1.5, 3.3.1.16, 3.3.1.19, 3.3.1.20, 3.3.2.26, 3.3.2.30, 3.3.2.33, 3.3.2.34, 3.3.2.38, 3.3.2.39, 3.3.2.40, 3.3.2.131, 3.3.2.138, 3.3.2.144, 3.3.2.150, 3.3.2.153, 3.3.2.154, 3.3.2.157, 3.3.2.158, 3.3.2.300
Piping and Fittings (Quad Cities only) (includes orifices)	Throttle	3.3.1.19
Pumps	Pressure Boundary	3.3.1.19, 3.3.2.170, 3.3.2.171, 3.3.2.178
Sprinklers (includes nozzles)	Pressure Boundary	3.3.1.5, 3.3.1.19, 3.3.2.26, 3.3.2.205, 3.3.2.206
Tanks (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.20
Valves (Fire Protection) (includes hydrants, and nozzles)	Pressure Boundary	3.3.1.19, 3.3.2.301
Valves (includes nozzles)	Pressure Boundary	3.3.1.5, 3.3.1.19, 3.3.1.20, 3.3.2.23, 3.3.2.31, 3.3.2.32, 3.3.2.300, 3.3.2.260, 3.3.2.262, 3.3.2.265, 3.3.2.283, 3.3.2.301

Aging management review results for the fire protection system are provided in Section 3.3.

2.3.3.6 Emergency Diesel Generator and Auxiliaries

System Purpose

The emergency diesel generator (EDG) system provides an emergency source of AC power to the emergency core cooling system or safe shutdown equipment for each unit in the event off-site power supply is not available. The system consists of three emergency diesel generators per site: one unit per reactor and a shared unit. The emergency diesel generator produces power at a voltage and frequency compatible with normal bus requirements. The auxiliaries associated with this system include the jacket water system, the starting air system, the lubrication system, the ventilation system and the combustion air intake and exhaust.

System Operation

Each EDG is an assembly consisting of a diesel engine and a generator. Each assembly is located in a separate diesel generator room. Included in this assembly are directly attached piping, valves, manifolds, starters, turbochargers, internal coolers, instrumentation, and other equipment. The diesel engine is provided with fuel oil from the diesel oil system (evaluated with the diesel fuel oil system).

Each diesel generator room has an independent ventilation system, which maintains the associated room within equipment temperature limits. Each ventilation system has a supply fan which auto starts when the EDG starts. The fan draws in either outside air or turbine building air through temperature-controlled modulation dampers, isolation dampers, and fire dampers. The air exhausts the room through pneumatically operated dampers. Each ventilation system is used only when its respective diesel generator is operating.

The EDG jacket water system provides cooling to the lube oil cooler, and other diesel subcomponents, and maintains the diesel at an acceptable starting temperature during standby. The system is a closed loop system that starts with the EDG jacket water heat exchangers, passes through the diesel engine and lube oil cooler, and returns to the heat exchanger. It includes engine-driven pumps, an immersion heater, expansion tank, piping, valves, controls and instrumentation. The EDG jacket water heat exchangers are provided with cooling water from the diesel generator cooling water system (evaluated with the diesel generator cooling water system).

The EDG starting air system provides a motive force to start the EDG. The system starts with the EDG starting air compressors, then passes through the receiver tanks, and moisture separators, to the EDG start motors. It includes piping, valves, controls and instrumentation.

The lubrication system provides lubrication to the EDG and subcomponents and maintains the diesel at an acceptable starting temperature during standby. The system includes five pumping circuits for each EDG. Each circuit starts from the engine sump, then passes through strainers, pumps, filters, and coolers, to the supplied service. It includes piping, valves, controls and instrumentation. Also included is the oil pumping circuit for the governor drive assembly.

The EDG combustion air intake and exhaust system provides filtered air for engine combustion and provides a flow path to drive the turbocharger, and to remove exhaust gases to the outside. The system starts from the air intake hood, then passes through intake filters, to the engine, to the exhaust manifold, and through the exhaust silencer and the outlet screen.

System Evaluation Boundary

The EDG assembly system evaluation boundary includes directly attached piping, valves, manifolds, starters, turbochargers, internal coolers, instrumentation, and other equipment. The EDG ventilation system evaluation boundary begins with the room's air intake and continues through the associated temperature-controlled modulation dampers, isolation dampers, and fire dampers and supply fan, and the air return through the pneumatically operated damper. The EDG jacket water system evaluation boundary begins with the EDG cooling water heat exchangers, continues through the diesel engine and lube oil cooler, and returns to the heat exchanger. The EDG starting air system evaluation boundary begins with the EDG starting air compressors, continues through the receiver tanks and moisture separators, to the EDG start motors. The EDG lubrication system evaluation boundary for each EDG begins with the engine sump, continues through the strainers, pumps, filters, and coolers, to the supplied service for each of the five circuits. Also included is the oil pumping circuit for the governor drive assembly. The EDG combustion air intake and exhaust system evaluation boundary begins with the air intake hood, continues through the intake filters, the engine, the exhaust manifold, through the exhaust silencer and to the outlet screen. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the emergency diesel generator system boundary.

UFSAR References

Dresden Station UFSAR Section(s): 8.3.1 and 9.4.7

Quad Cities Station UFSAR Section(s): 8.3.1 and 9.4.5

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE-M-173, LR-DRE-M-40, LR-DRE-M-478-1, LR-DRE-M-478-2, LR-DRE-M-478-3, LR-DRE-M-517-1, LR-DRE-M-517-2, LR-DRE-M-517-3, and LR-DRE-M-974.

Quad Cities Station:
LR-QDC-M-22-3, LR-QDC-M-25-2, LR-QDC-M-69-3, LR-QDC-M-72-2, LR-QDC-M-813-1, LR-QDC-M-813-2, LR-QDC-M-813-3, LR-QDC-M-943-1, LR-QDC-M-943-2, and LR-QDC-M-943-3.

System Intended Functions

Provide emergency AC power - provides independent power source to assure safe reactor shutdown under emergency conditions on a total loss of offsite power concurrent with a design basis accident.

Credited in regulated event(s) – credited in support of fire protection (10CFR50.48). The system contains components that are relied upon for compliance with 10 CFR 50.49, (EQ).

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.3-6 Component Groups Requiring Aging Management Review - Emergency Diesel Generator and Auxiliaries

Component Group	Component Intended Function	Aging Management Ref
Air Accumulator Vessels (includes tanks)	Pressure Boundary	3.3.1.5, 3.3.2.6
Closure Bolting	Pressure Boundary	3.3.1.22, 3.3.2.18
Debris Screens (Quad Cities only)	Filter	3.3.2.59
Doors, Closure Bolts, Equip Frames (includes dampers, duct, and housings)	Pressure Boundary	3.3.1.5
Duct Fittings, Hinges, Latches (includes anchors, bolts, and fasteners)	Pressure Boundary	3.3.2.49
Filters/Strainers	Pressure Boundary	3.3.1.5, 3.3.2.29, 3.3.2.55, 3.3.2.58, 3.3.2.60
Filters/Strainers	Filter	3.3.1.5, 3.3.2.55, 3.3.2.58, 3.3.2.60
Flex Collars, Doors and Damper Seals	Pressure Boundary	3.3.1.2
Flexible Hoses	Pressure Boundary	3.3.2.65, 3.3.2.66
Heat Exchangers (includes coolers)	Pressure Boundary	3.3.1.5, 3.3.2.34, 3.3.2.94, 3.3.2.95, 3.3.2.100, 3.3.2.101, 3.3.2.108, 3.3.2.109, 3.3.2.110, 3.3.2.111
Heat Exchangers (includes coolers)	Heat Transfer	3.3.2.93, 3.3.2.96, 3.3.2.99
Lubricators	Pressure Boundary	3.3.1.5

Table 2.3.3-6 Component Groups Requiring Aging Management Review - Emergency Diesel Generator and Auxiliaries (Continued)

Component Group	Component Intended Function	Aging Management Ref
Mufflers	Pressure Boundary	3.3.1.5, 3.3.2.29
NSR Vents or Drains, Piping and Valves (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.2.130
Piping and Fittings (Includes dryers, heaters and tubing)	Pressure Boundary	3.3.1.5, 3.3.1.13, 3.3.2.21, 3.3.2.23, 3.3.2.29, 3.3.2.132, 3.3.2.133, 3.3.2.134, 3.3.2.137, 3.3.2.139, 3.3.2.147, 3.3.2.159, 3.3.2.300
Piping and Fittings (attached support)	Structural Integrity (attached)	3.3.1.5, 3.3.1.7, 3.3.1.13, 3.3.2.23, 3.3.2.139
Pumps (includes governors)	Pressure Boundary	3.3.1.5, 3.3.1.13, 3.3.2.174, 3.3.2.175, 3.3.2.177, 3.3.2.300
Restricting Orifices	Pressure Boundary	3.3.1.5, 3.3.2.40, 3.3.2.187
Restricting Orifices	Throttle	3.3.2.187
Sight Glasses	Pressure Boundary	3.3.2.36, 3.3.2.200, 3.3.2.202
Tanks	Pressure Boundary	3.3.1.5, 3.3.1.13, 3.3.2.211
Thermowells	Pressure Boundary	3.3.1.5, 3.3.2.23, 3.3.2.40, 3.3.2.220, 3.3.2.221, 3.3.2.224, 3.3.2.225
Tubes (Quad Cities only) (includes heat exchangers)	Heat Transfer	3.3.1.15
Tubing	Pressure Boundary	3.3.1.5, 3.3.1.15, 3.3.2.34, 3.3.2.40, 3.3.2.232, 3.3.2.233, 3.3.2.235, 3.3.2.236, 3.3.2.237, 3.3.2.240, 3.3.2.241, 3.3.2.245, 3.3.2.249, 3.3.2.250
Tubing (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.2.40, 3.3.2.245, 3.3.2.250
Turbochargers	Pressure Boundary	3.3.2.256, 3.3.2.300
Valves	Pressure Boundary	3.3.1.5, 3.3.1.13, 3.3.2.23, 3.3.2.40, 3.3.2.258, 3.3.2.260, 3.3.2.261, 3.3.2.262, 3.3.2.267, 3.3.2.269, 3.3.2.275, 3.3.2.286, 3.3.2.290, 3.3.2.291, 3.3.2.300
Valves (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.1.5, 3.3.1.7, 3.3.2.269

Aging management review results for the emergency diesel generator and auxiliaries are provided in Section 3.3.

2.3.3.7 HVAC - Main Control Room

System Purpose

The control room HVAC system provides: (1) a suitable environment during normal operation for the control room operators and equipment, (2) a habitable environment after a design basis accident in which the operators can safely shutdown and maintain the plant for the duration of the accident, (3) an environment from which the operators can safely occupy and operate the plant during an onsite or offsite toxic chemical accident, and (4) fire protection to the operators with fire dampers, for fires outside the control room, and a smoke purge function mode for fires inside the control room.

System Operation

The control room HVAC system consists of the train A HVAC system (a multizone unit), the train B HVAC system (a single zone unit), the emergency air filtration unit (AFU), the toxic gas analyzer system, and the smoke detection system. Each system train includes manual and air-operated dampers, an air handling unit (AHU), and distribution air ducts to and from the control room (including the cable spreading room and the auxiliary electric equipment room for Quad Cities) and the train B HVAC equipment room. The train A HVAC system also includes an exhaust fan and the smoke detection system. The train B HVAC system also includes the AFU, associated booster fans, a refrigeration condensing unit (RCU), the toxic gas analyzer system, and associated valves, instrumentation and controls. A sprinkler system is also provided to the AFU from the fire protection system (evaluated with the fire protection system). The train A AHU is the primary unit to provide the temperature control and air distribution for the control room. The train B AHU serves as a backup to the train A AHU and provides the source of cooling for the control room in the event the train A AHU is lost. During normal operation, outside air is mixed with return air to maintain the control room emergency zone at a positive pressure. In the event of a design basis accident, the normal outside air intakes are isolated, and the AFU provides filtered makeup air to maintain pressurization of the control room emergency zone. In the event smoke is detected in the intake air ducts, the train A HVAC system outdoor air intake is automatically isolated and the system air is recirculated. In the event smoke is detected in the return air ducts, the train A HVAC system is automatically switched to the purge mode and the system is supplied with 100% outdoor air. The train B RCU is normally cooled with service water (evaluated with the service water system). However, upon loss of service water, the RCU may be cooled by containment cooling service water (evaluated with the containment cooling service water system) at Dresden and by residual heat removal service water (evaluated with the residual heat removal service water system) at Quad Cities. The toxic gas analyzer (at Quad Cities) continuously monitors the outside air intake of the operating AHU, and automatically isolates outdoor air intakes in the event specified toxic gas limits are approached. The toxic gas analyzer at Dresden has been determined to not be needed, and has been abandoned in place. The outdoor air intakes can still be manually isolated, if required.

System Evaluation Boundary

The train A control room HVAC system evaluation boundary starts with the outside air intake isolation damper, includes the train A AHU, distribution air ducts to the control room and the train B HVAC equipment room, and associated dampers. The boundary continues through the rooms' exhaust ducts, dampers, and the exhaust fan, to the exhaust isolation dampers, and the train A AHU return ducts and isolation damper. The boundary also includes the train B control room HVAC system which starts with the outside air intake isolation damper, continues through the AFU, associated booster fans, the train B AHU, and the distribution ducts to the control room and the train B HVAC equipment room. The evaluation boundary also includes the RCU, the AFU sprinkler system and toxic gas analyzer for the train B AHU and associated valves, instrumentation and controls. The toxic gas analyzer at Dresden has been removed from service, but its components remain in place. Thus it is included in the evaluation boundary. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the control room HVAC system boundary.

UFSAR References

Dresden Station UFSAR Section(s): 6.4 and 9.4.1

Quad Cities Station UFSAR Section(s): 6.4 and 9.4.1

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-273-1, LR-DRE-M-273-2, LR-DRE-M-1013-2, and LR-DRE-M-3121.

Quad Cities Station:

LR-QDC-M-725-1, LR-QDC-M-725-2, and LR-QDC-M-725-3.

System Intended Functions

Isolation and filtration - provides isolation and filtration for the control room during accident conditions.

Environmental control - provides habitable environment for the control room during normal, abnormal, accident, and post-accident conditions.

Credited in regulated event(s) – demonstrates compliance with NRC regulations regarding SBO (10CFR50.63) and fire protection (10CFR50.48).

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.3-7 Component Groups Requiring Aging Management Review – HVAC – Main Control Room

Component Group	Component Intended Function	Aging Management Ref
Air Handlers Heating/Cooling (CR HVAC)	Pressure Boundary	3.3.1.5, 3.3.2.11
Air Handlers Heating/Cooling (CR HVAC)	Heat Transfer	3.3.2.10
Dampeners (Quad Cities only)	Pressure Boundary	3.3.2.23, 3.3.2.46
Debris Screens	Filter	3.3.2.59
Diffusers	Pressure Boundary	3.3.1.5, 3.3.2.48
Doors, Closure Bolts, Equip Frames (includes dampers, duct, housings, and silencers)	Pressure Boundary	3.3.1.5
Duct Fittings, Hinges, Latches	Pressure Boundary	3.3.2.49
Filters/Strainers	Pressure Boundary	3.3.1.5, 3.3.2.53
Filters/Strainers	Filter	3.3.2.53
Flex Collars, Doors and Damper Seals (includes duct)	Pressure Boundary	3.3.1.2
Flow Elements (Dresden only)	Throttle	3.3.2.73
Heat Exchangers	Pressure Boundary	3.3.1.5, 3.3.2.84, 3.3.2.85
Heat Exchangers	Heat Transfer	3.3.2.82
Housings and Supports	Pressure Boundary	3.3.1.5, 3.3.2.40
NSR Vents or Drains, Piping and Valves (attached support) (Dresden only)	Structural Integrity (attached)	3.3.2.130
Piping and Fittings	Pressure Boundary	3.3.1.5, 3.3.2.34, 3.3.2.145, 3.3.2.152
Piping and Fittings (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.1.5, 3.3.1.15, 3.3.2.34, 3.3.2.152
Seals	Pressure Boundary	3.3.1.2
Sight Glasses	Pressure Boundary	3.3.2.34, 3.3.2.196, 3.3.2.198
Tubing (Dresden only)	Pressure Boundary	3.3.2.244

Table 2.3.3-7 Component Groups Requiring Aging Management Review – HVAC – Main Control Room (Continued)

Component Group	Component Intended Function	Aging Management Ref
Valves (includes dampers)	Pressure Boundary	3.3.1.5, 3.3.1.15, 3.3.2.23, 3.3.2.37, 3.3.2.40, 3.3.2.260, 3.3.2.264, 3.3.2.279, 3.3.2.280, 3.3.2.284, 3.3.2.293, 3.3.2.300

Aging management review results for the HVAC – main control room system are provided in Section 3.3.

2.3.3.8 HVAC – Reactor Building

System Purpose

The reactor building ventilation system provides conditioned air to the reactor building and primary containment structures to remove the heat remaining from the primary process and operating equipment, minimize the level of airborne contaminants, make the plant atmosphere adequate to support the presence of personnel and maintain the reactor building at a negative pressure to minimize the release of radioactive contaminants to the environment. Emergency isolation dampers are provided to isolate the reactor building in the event of high radiation. The reactor building ventilation system also removes exhaust air from the drywell and suppression chamber purge system when the reactor is shutdown and/or whenever primary containment access is required.

System Operation

Using supply fans, outside air is drawn into the reactor building ventilation system, filtered, tempered and discharged into the supply system ducts. The supply air ducts distribute air throughout the building via air registers. The system also uses two emergency isolation dampers in series in the main supply duct upstream of all branch ducts. The exhaust fans draw building air into exhaust vents located throughout the building and discharge it through the reactor building vent stack. The normal ventilation exhaust duct for the spent fuel, reactor cavity and dryer/separator pool area is arranged to take suction through multiple inlets around the periphery of the pools above the water line. Two emergency isolation dampers are installed in series in the main exhaust duct upstream of the exhaust fan air intake and downstream of any branch connections to the exhaust duct. The reactor building ventilation system also removes exhaust air from the drywell and suppression chamber purge system when the reactor is shutdown for maintenance or whenever primary containment access is required.

System Evaluation Boundary

The reactor building ventilation system evaluation boundary begins with an air filter, steam heating coil, air wash/evaporative cooler (Quad Cities only, but no longer in use), chilled water coil (Dresden only), and three supply fans. Outside air is drawn into the reactor building ventilation system through the aforementioned components, filtered, temperature controlled and discharged into the supply system ducts. The supply air ducts distribute the air through the building to the air registers. In addition, the system uses two emergency isolation dampers, installed in series, in the main supply duct. Three exhaust fans draw building air into exhaust vents located throughout the building and discharge it through the reactor building vent stack. The normal ventilation exhaust duct for the spent fuel, reactor cavity and dryer/separator pool area is arranged to take suction through multiple inlets around the periphery of the pools above the water line. Two more emergency isolation dampers are installed in series in the main exhaust duct upstream of the exhaust fan air intake and downstream of any branch connections to the exhaust duct. The drywell and suppression chamber purge system is used for purging, deinerting and ventilating the primary containment. This system consists of two prefilters, two absolute filters and two exhaust fans. Air is drawn through the system through associated piping connected to the drywell and suppression chamber and discharged to the reactor building main exhaust duct. At Quad Cities, the parts of the RB ventilation system that are within the scope of the rule include (a) the emergency isolation dampers in the main supply duct and main exhaust duct, and (b) the fire damper in the exhaust duct from the main steam pipe area in the turbine building. The remaining parts of the system are not within the scope of the Rule.

At Dresden, all associated piping, components, and instrumentation contained within the flow paths and systems described above are a requirement for design basis as described in Section 15.7.3 of the Dresden UFSAR (Design basis fuel handling accident during refueling) and are included in the RB ventilation system evaluation boundary.

UFSAR References

Dresden Station UFSAR Section(s): 6.2.3, 9.4.2, 9.4.5, 9.4.6

Quad Cities Station UFSAR Section(s): 6.2.3, 9.4.2, 9.4.7

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-269-1, LR-DRE-M-269-2, LR-DRE-M-269-3, LR-DRE-M-422-1, LR-DRE-M-529-1, LR-DRE-M-529-2, LR-DRE-M-529-3 and LR-DRE-M-973

Quad Cities Station:

LR-QDC-M-371, LR-QDC-M-373-1 and LR-QDC-M-457

System Intended Functions

Support ESF function(s) – provides instrumentation to trip automatically on a secondary containment isolation signal.

Containment isolation – isolation dampers help ensure that adequate secondary containment is maintained during and after an accident by minimizing potential paths to the environment.

Credited in regulated event(s) – fire dampers provide isolation to prevent spread of a fire credited in mitigation of the Appendix R fire.

Flow path – (Dresden only) conducts effluent directly to the reactor building stack or the standby gas treatment system.

Pressure control – (Dresden only) maintains a negative pressure of at least ¼” H_2O between the reactor building and the environment.

Radioactivity control – (Dresden only) collects radioactivity released from a fuel handling accident in the openings located on the periphery of the refueling pools, and provide timely removal to the reactor building exhaust plenum providing sufficient level to isolate secondary containment.

Component Groups Requiring Aging Management Review

Table 2.3.3-8 Component Groups Requiring Aging Management Review – HVAC – Reactor Building

Component Group	Component Intended Function	Aging Management Ref
Debris Screens (Dresden only)	Filter	3.3.2.59
Doors, Closure Bolts, Equip Frames (includes dampers, duct, housings, piping, and valves)	Pressure Boundary	3.3.1.5
Duct Fittings, Hinges, Latches	Pressure Boundary	3.3.2.49
Filters/Strainers (Dresden only)	Pressure Boundary	3.3.2.23, 3.3.2.52
Flex Collars, Doors, Duct, and Damper Seals (Dresden only)	Pressure Boundary	3.3.1.2
Housings and Supports (Dresden only)	Pressure Boundary	3.3.1.5
Piping and Fittings (Dresden only)	Pressure Boundary	3.3.2.166
Seals (Dresden only)	Pressure Boundary	3.3.1.2
Tubing	Pressure Boundary	3.3.2.34, 3.3.2.40, 3.3.2.242, 3.3.2.251
Valves (Dresden only) (includes dampers)	Pressure Boundary	3.3.1.5, 3.3.2.23, 3.3.2.40, 3.3.2.262, 3.3.2.273, 3.3.2.295

Aging management review results for the HVAC – Reactor Building is provided in Section 3.3.

2.3.3.9 ECCS Corner Room HVAC

System Purpose

The ECCS corner room HVAC room coolers maintain the compartment temperature below the qualification temperature of the components that are required for safe shutdown of the plant. Each of the rooms has a water-cooled heat exchanger and fan unit that functions as a room cooler.

System Operation

At Dresden, the ECCS corner room HVAC includes the HPCI and LPCI room coolers. The core spray pumps are located in the LPCI rooms. There are two LPCI room coolers and one HPCI room cooler per unit. During normal plant operating conditions, the cooling water for the room coolers is provided by the service water system. For Dresden, the containment cooling service water system provides backup cooling water to all three room coolers. At Quad Cities, the ECCS corner room HVAC includes one HPCI, two core spray, and two RHR room coolers per unit. The Quad Cities Unit 1 and Unit 2 DG cooling water pumps (evaluated with the diesel generator service water system) provide cooling water to the room coolers in their respective units. The service water system can also provide a non-safety related alternate supply of cooling water to the HPCI room emergency coolers.

System Evaluation Boundary

The ECCS corner room HVAC evaluation boundary for room cooling begins at the cooling water inlet and ends at the discharge nozzle for each individual room cooler. The ECCS corner room coolers include the following: At Dresden, the LPCI room coolers and HPCI room cooler. At Quad Cities, the core spray room coolers, RHR room coolers and HPCI room cooler. Each room cooler is fin and coil type and is provided with a dedicated fan, enclosure and cooling coil. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the ECCS corner room HVAC system evaluation boundary.

UFSAR References

Dresden Station UFSAR Section(s): 9.2.1, 9.2.2, 9.4.6

Quad Cities Station UFSAR Section(s): 6.3.2, 9.2.2, 9.5.5

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-22, LR-DRE-M-269-2, LR-DRE-M-355, LR-DRE-M-529-2 and LR-DRE-M-973

Quad Cities Station:

LR-QDC-M-22-5 and LR-QDC-M-69-5

System Intended Functions

Environmental control- provides ventilation to maintain an acceptable environment to support proper ECCS pump operation during normal plant operating conditions and following design basis events.

Credited in regulated event(s) – (Dresden) The system contains components that are relied upon to demonstrate compliance with 10 CFR Part 50.49 (EQ).

Credited in regulated event(s) – (Quad Cities) provides ventilation and cooling credited in mitigation of the Appendix R fire. The system also contains components that are relied upon to demonstrate compliance with 10 CFR Part 50.49 (EQ).

Component Groups Requiring Aging Management Review

Table 2.3.3-9 Component Groups Requiring Aging Management Review – ECCS Corner Room HVAC

Component Group	Component Intended Function	Aging Management Ref
Air Handlers Heating/Cooling (Aux&RW HVAC)	Pressure Boundary	3.3.2.8, 3.3.2.9, 3.3.2.21
Air Handlers Heating/Cooling (Aux&RW HVAC)	Heat Transfer	3.3.2.7
Ducts & Fittings, Access Doors, Closure Bolts, Equip Frames	Pressure Boundary	3.3.1.5

Aging management review results for the ECCS corner room HVAC is provided in Section 3.3.

2.3.3.10 Station Blackout Building HVAC

System Purpose

The SBO HVAC system maintains the SBO diesel generator and support equipment ambient temperatures within an acceptable range for diesel readiness, provides annunciation of temperature abnormalities, reacts to fire alarm actuation, and removes hydrogen gas and diesel fumes from the building.

System Operation

The station blackout building HVAC system provides heating and ventilation for the SBO diesel generator rooms, the electrical equipment rooms, and the battery rooms. The ventilation system for each diesel generator is capable of maintaining the design room conditions, with the diesel generators running at full load and maintaining design room conditions in winter with the SBO diesel generators in the standby mode. The ventilation system for each electrical equipment room is capable of maintaining the design room conditions with the diesel generators running at full load. The heating systems for these rooms will maintain room conditions in winter. The ventilation systems for the battery rooms have an air-cooled condensing unit, an air handling unit, and an electric heater capable of maintaining the battery room at nominal design conditions. The exhaust fans in the battery rooms and the day tank rooms are interlocked with the fire system to shut down on fire alarm actuation.

System Evaluation Boundary

Quad Cities

The SBO building HVAC system starts at the outside air intake dampers to the electrical equipment and diesel generator rooms. It includes all the HVAC components in the electrical equipment, diesel generator, battery and day tank rooms, plus a limited amount of ductwork associated with the battery and day tank rooms. Equipment and diesel generator room exhaust fans equipped with discharge dampers draw air out of the rooms, allowing outside air to enter through the intake dampers and filters. Separate exhaust fans draw air from the day tank and battery rooms, and discharge through ductwork to the outside. Intake louvers between the day tank room and the diesel generator room, and between the battery room and the electrical equipment room, allow air to enter the rooms. Additionally, electric heaters heat the air spaces within the electrical equipment and diesel generator rooms, and an air handling unit and associated air-cooled condenser cool the battery room when needed.

Dresden

The SBO building HVAC system starts at the outside air intake dampers and intake ductwork to the electrical equipment and diesel generator rooms. It includes all the ductwork and HVAC components in the electrical equipment, diesel generator, and day tank rooms. The system continues through fans that supply the building. The day tank rooms have independent exhaust fans and ductwork within the evaluation boundary. The mechanical/electrical equipment room HVAC supply and exhaust fans, ductwork, dampers and components are also included within the evaluation boundary. The system ends at the exhaust dampers/ductwork to the outside air. The SBO building structure forms an integral part of the HVAC system (evaluated with the Unit 2/3 SBO Building).

UFSAR References

Dresden Station UFSAR Section(s): 9.5.9

Quad Cities Station UFSAR Section(s): 8.3.1.9

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-4356-1, LR-DRE-M-4356-2, LR-DRE-M-4356-3, LR-DRE-M-4356-4 and LR-DRE-M-4356-5

Quad Cities Station:

LR-QDC-M-3033-1 and LR-QDC-M-3033-2

System Intended Functions

Environmental control – supports SBO diesel generators in providing AC power during a loss of offsite power and station blackout by supplying the necessary HVAC to required SBO diesel generators and auxiliaries, i.e., equipment rooms, battery rooms, and day tank rooms.

Credited in regulated event(s) – (Quad Cities only) system functions to prevent spread of a fire credited in mitigation of the Appendix R fire.

Component Groups Requiring Aging Management Review

Table 2.3.3-10 Component Groups Requiring Aging Management Review – SBO Building HVAC

Component Group	Component Intended Function	Aging Management Ref
Air Handlers Heating/Cooling (DGB HVAC) (Quad Cities only)	Pressure Boundary	3.3.2.14, 3.3.2.21
Air Handlers Heating/Cooling (DGB HVAC) (Quad Cities only)	Heat Transfer	3.3.2.13
Debris Screens	Filter	3.3.2.59
Doors, Closure Bolts, Equip Frames	Pressure Boundary	3.3.1.5
Duct Fittings, Hinges, Latches	Pressure Boundary	3.3.2.49
Flex Collars, Doors and Damper Seals	Pressure Boundary	3.3.1.2
Flow Elements (Dresden only)	Pressure Boundary	3.3.2.40, 3.3.2.73
Tubing	Pressure Boundary	3.3.2.34, 3.3.2.40, 3.3.2.244, 3.3.2.254

Aging management review results for the SBO Building HVAC are provided in Section 3.3.

2.3.3.11 Station Blackout System (diesels and auxiliaries)

System Purpose

Station blackout (SBO) diesel generators and auxiliaries provide an additional, independent AC power source as a backup to the emergency diesel generators.

System Operation

The system consists of two diesel generator (DG) sets. Each SBO DG set is an assembly consisting of two diesel engines and a generator arranged in tandem. Each assembly is located in a separate diesel generator room. Included in this assembly are directly attached piping, valves, manifolds, starters, turbochargers, internal coolers, instrumentation, and other equipment. The diesel engines are provided with fuel oil from the SBO diesel oil system. Each diesel generator room has an independent ventilation system (evaluated with the station blackout building HVAC system), which maintains the associated room at or below equipment temperature limits. Each generator is connectable to the safe shutdown equipment on one nuclear unit and can also be connected to the opposite unit via the 4 kV cross-tie. The SBO DGs must be manually started and manually connected to the appropriate safe shutdown loads. Under conditions of total or partial loss of offsite power, the SBO DG can be remotely or locally started, and dead bus connections can be made to allow the SBO DGs to provide power to safe shutdown plant loads. The auxiliary systems associated with the station blackout system include the SBO DG engine jacket water system, the SBO engine exhaust/combustion air system, the SBO DG engine air start system, the SBO DG engine lube oil system, and the SBO DG fuel oil system.

System Evaluation Boundary

SBO diesel generators and auxiliaries system evaluation boundary starts with the two diesel generator sets and continues with the auxiliary systems. The auxiliary systems included in the station blackout system evaluation boundary are the SBO DG engine jacket water system, SBO engine exhaust/combustion air system, SBO DG engine air start system, SBO DG engine lube oil system, and SBO DG fuel oil system . All associated cabling, piping, components and instrumentation contained within the flow paths and systems described above are included in the station blackout (diesels and auxiliaries) evaluation boundary.

UFSAR References

Dresden Station UFSAR Section(s): 9.5.9

Quad Cities Station UFSAR Section(s): 8.3.1.9

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE-M-4305, LR-DRE-M-4305A, LR-DRE-M-4305B, LR-DRE-M-4306-1, LR-DRE-M-4306-2, LR-DRE-M-4307, LR-DRE-M-4308,
LR-DRE-M-4308A, LR-DRE-M-4308B, LR-DRE-M-4308C, LR-DRE-M-4308D,

LR-DRE-M-4308E, LR-DRE-M-4308F, LR-DRE-M-4359-1, LR-DRE-M-4359-2, LR-DRE-M-4359-3, LR-DRE-M-4359-4, LR-DRE-M-4360-1, LR-DRE-M-4360-2, LR-DRE-M-4360-3, LR-DRE-M-4360-4, LR-DRE-M-4361-1, LR-DRE-M-4361-2, LR-DRE-M-4361-3 and LR-DRE-M-4361-4

Quad Cities Station:
LR-QDC-M-3028-1, LR-QDC-M-3028-2, LR-QDC-M-3029-1, LR-QDC-M-3029-2, LR-QDC-M-3029-3, LR-QDC-M-3029-4, LR-QDC-M-3031-1, LR-QDC-M-3031-2, LR-QDC-M-3032-1, LR-QDC-M-3032-2, LR-QDC-M-3034-1 and LR-QDC-M-3034-2

System Intended Functions

Credited in regulated event(s) – provides an alternate source of AC electrical power to plant equipment in the event of a station blackout (10 CFR Part 50.63).

Credited in regulated event(s) – (Quad Cities only) provides an alternate power source credited in mitigation of the Appendix R fire.

Component Groups Requiring Aging Management Review

Table 2.3.3-11 Component Groups Requiring Aging Management Review – Station Blackout (diesels and auxiliaries)

Component Group	Component Intended Function	Aging Management Ref
Air Accumulator Vessels	Pressure Boundary	3.3.1.5
Closure Bolting	Pressure Boundary	3.3.1.22, 3.3.2.18
Filters/Strainers	Pressure Boundary	3.3.1.5, 3.3.1.7, 3.3.2.21, 3.3.2.50, 3.3.2.54, 3.3.2.55, 3.3.2.60, 3.3.2.300
Filters/Strainers (includes separators)	Filter	3.3.1.5, 3.3.1.7, 3.3.2.55, 3.3.2.60
Flexible Hoses	Pressure Boundary	3.3.2.64, 3.3.2.66
Flow Elements	Pressure Boundary	3.3.2.40, 3.3.2.70
Heat Exchangers (includes coolers and heat exchangers)	Pressure Boundary	3.3.1.5, 3.3.2.21, 3.3.2.29, 3.3.2.102, 3.3.2.103, 3.3.2.104, 3.3.2.105, 3.3.2.106, 3.3.2.107
Heat Exchangers (includes coolers and heat exchangers)	Heat Transfer	3.3.2.97, 3.3.2.98
Lubricators	Pressure Boundary	3.3.1.5, 3.3.2.21, 3.3.2.126, 3.3.2.127
Mufflers	Pressure Boundary	3.3.1.5, 3.3.2.29

Table 2.3.3-11 Component Groups Requiring Aging Management Review – Station Blackout (diesels and auxiliaries) (Continued)

Component Group	Component Intended Function	Aging Management Ref
Piping and Fittings (includes heaters, orifices, and thermowells)	Pressure Boundary	3.3.1.5, 3.3.1.7, 3.3.1.13, 3.3.2.23, 3.3.2.29, 3.3.2.34, 3.3.2.38, 3.3.2.40, 3.3.2.135, 3.3.2.136, 3.3.2.139, 3.3.2.148, 3.3.2.149, 3.3.2.151, 3.3.2.155, 3.3.2.156, 3.3.2.160, 3.3.2.162, 3.3.2.163, 3.3.2.300
Piping and Fittings (includes restricting orifices)	Throttle	3.3.1.5
Pumps	Pressure Boundary	3.3.1.5, 3.3.1.13, 3.3.2.175, 3.3.2.176, 3.3.2.177, 3.3.2.300
Pumps (Dresden only)	Throttle	3.3.1.13
Restricting Orifices	Pressure Boundary	3.3.1.5, 3.3.2.40, 3.3.2.190, 3.3.2.191, 3.3.2.192
Restricting Orifices	Throttle	3.3.2.190, 3.3.2.192
Sight Glasses	Pressure Boundary	3.3.2.36, 3.3.2.200
Tanks	Pressure Boundary	3.3.1.5, 3.3.1.7, 3.3.1.13
Thermowells	Pressure Boundary	3.3.1.5, 3.3.1.7, 3.3.2.23, 3.3.2.40, 3.3.2.222, 3.3.2.224, 3.3.2.226
Tubing	Pressure Boundary	3.3.2.34, 3.3.2.40, 3.3.2.238, 3.3.2.246, 3.3.2.247, 3.3.2.249
Turbochargers	Pressure Boundary	3.3.2.256, 3.3.2.300
Valves	Pressure Boundary	3.3.1.5, 3.3.1.7, 3.3.1.13, 3.3.2.23, 3.3.2.40, 3.3.2.261, 3.3.2.269, 3.3.2.287, 3.3.2.288, 3.3.2.290, 3.3.2.291

Aging management review results for the station blackout system (diesels and auxiliaries) are provided in Section 3.3.

2.3.3.12 Diesel Generator Cooling Water System

System Purpose

The diesel generator cooling water (DGCW) system provides cooling water to each of the three emergency diesel generators. In addition, the system provides cooling water for the room coolers of the high pressure coolant injection (HPCI) room, residual heat removal (RHR) rooms, core spray rooms and diesel generator cooling water pump cubicle at Quad Cities, and provides an alternate water supply for the containment cooling service water (CCSW) keep fill system at Dresden.

System Operation

At Dresden, three motor driven submersible pumps take suction in the crib-house and provide cooling water to the DGCW heat exchangers (evaluated with emergency diesel generator and Auxiliaries system). The DGCW pumps may be used as the alternate safety related water supply to the CCSW keep fill system (evaluated with the service water system) which is normally supplied by service water (evaluated with the service water system). A connection is provided with cross connection piping between Unit 2 and Unit 3 as well. The DGCW return water is then routed back to the service water discharge pipe (evaluated with the service water system).

At Quad Cities, the DGCW pumps, which are located in separate RHR service water pump vaults, take suction from the RHR service water inlet header (evaluated with the RHR service water system). These pumps provide cooling water to the DG engine heat exchangers (evaluated with emergency diesel generator and auxiliaries system), to the HPCI, RHR and core spray room coolers (evaluated with ECCS corner room HVAC system), to the diesel generator cooling water pump cubicle cooler, and ultimately discharge into the service water discharge pipe (evaluated with service water system).

System Evaluation Boundary

At Dresden, the DGCW system starts at the pump suction piping in the crib house. Three motor driven submersible pumps provide cooling water to the DGCW heat exchangers. A connection is provided to the CCSW keep fill system by the emergency core cooling system (ECCS) room cooler cross-tie header to backup the normal supply of service water and act as an alternate safety related water supply. Cross connection piping between Unit 2 and Unit 3 exists. The return water is then routed to the service water system where the system ends.

At Quad Cities, the DGCW system starts at the DGCW pumps which are located in a separate RHR service water pump vault and take suction from the RHR service water inlet header. These pumps provide cooling water to the heat exchangers of DG engines and to the room coolers of the HPCI room, the RHR rooms, the core spray rooms, and the diesel generator cooling water pump cubicle, and ultimately discharge into the service water discharge pipe.

UFSAR References

Dresden Station UFSAR Section(s): 7.4.2, 9.5.5

Quad Cities Station UFSAR Section(s): 3.4.1.2, 6.3.2, 9.5.5

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE-M-22, LR-DRE-M-29-2, LR-DRE-M-36, LR-DRE-M-355, LR-DRE-M-360-2, LR-DRE-M-517-1, LR-DRE-M-517-2, LR-DRE-M-517-3

Quad Cities Station:
LR-QDC-M-22-1, LR-QDC-M-22-3, LR-QDC-M-22-5, LR-QDC-M-69-1, LR-QDC-M-69-3, LR-QDC-M-69-5

System Intended Functions

Supports emergency supply of AC power – provides cooling water to emergency diesel generator heat exchangers and engine jacket cooling.

Support ESF function(s) – (Quad Cities only) provides cooling water to ECCS room coolers to ensure proper environment for ECCS pump operation.

Structural support – non-safety related portions of this system provide structural support to attached safety related piping.

Credited in regulated event(s) – provides cooling water to safe shutdown equipment credited in mitigation of the Appendix R fire.

Component Groups Requiring Aging Management Review

Table 2.3.3-12 Component Groups Requiring Aging Management Review –Diesel Generator Cooling Water System

Component Group	Component Intended Function	Aging Management Ref
Air Handlers Heating/Cooling (DGB HVAC) (Quad Cities only) (includes DGCW pump cubicle coolers)	Pressure Boundary	3.3.2.15, 3.3.2.16
Air Handlers Heating/Cooling (DGB HVAC) (Quad Cities only) (includes DGCW pump cubicle coolers)	Heat Transfer	3.3.2.12
Closure Bolting	Pressure Boundary	3.3.1.22, 3.3.2.18
Doors, Closure Bolts, Equip Frames (Quad Cities only)	Pressure Boundary	3.3.1.5
NSR Vents or Drains, Piping and Valves (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.2.130
Orifice Bodies	Pressure Boundary	3.3.1.15, 3.3.2.40, 3.3.2.41
Orifice Bodies	Throttle	3.3.1.15
Orifice Bodies (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.1.15, 3.3.2.40

Table 2.3.3-12 Component Groups Requiring Aging Management Review –Diesel Generator Cooling Water System (Continued)

Component Group	Component Intended Function	Aging Management Ref
Piping and Fittings (includes flow elements)	Pressure Boundary	3.3.1.5, 3.3.1.15, 3.3.2.26, 3.3.2.45, 3.3.2.141, 3.3.2.169
Piping and Fittings (attached support)	Structural Integrity (attached)	3.3.1.5, 3.3.1.15
Pulsation Dampeners (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.1.15, 3.3.2.41
Pumps	Pressure Boundary	3.3.1.15, 3.3.2.26, 3.3.2.173, 3.3.2.183, 3.3.2.184
Strainer Bodies (Dresden only)	Pressure Boundary	3.3.2.31, 3.3.2.208
Strainer Screens (Dresden only)	Filter	3.3.1.15
Thermowells	Pressure Boundary	3.3.1.15, 3.3.2.26
Tubing	Pressure Boundary	3.3.1.15, 3.3.2.40, 3.3.2.41
Valves	Pressure Boundary	3.3.1.5, 3.3.1.15, 3.3.1.27, 3.3.2.23, 3.3.2.24, 3.3.2.26, 3.3.2.31, 3.3.2.40, 3.3.2.41, 3.3.2.45, 3.3.2.279, 3.3.2.280, 3.3.2.298, 3.3.2.300
Valves (attached support)	Structural Integrity (attached)	3.3.1.15, 3.3.1.27, 3.3.2.23, 3.3.2.24, 3.3.2.40, 3.3.2.41, 3.3.2.279, 3.3.2.280, 3.3.2.300

Aging management review results for the diesel generator cooling water system are provided in Section 3.3.

2.3.3.13 Diesel Fuel Oil System

System Purpose

The diesel fuel oil system stores and transfers diesel fuel oil for the emergency diesel generators (EDG's), the station blackout (SBO) diesel generators (DG's), the diesel fire pumps, and (at Dresden only) the isolation condenser makeup pump diesels.

System Operation

The diesel fuel oil system consists of tanks, pumps, filters, strainers, and associated piping, valves, instrumentation and controls necessary to provide fuel oil to the EDG's (evaluated with the emergency diesel generator system), SBO DG's (evaluated with the station blackout diesels and auxiliaries), fire pump diesels (evaluated with the fire protection system), and the isolation condenser makeup pump diesels at Dresden (evaluated with the isolation condenser system). A separate fuel oil storage and transfer system is provided for each EDG. Each storage and transfer system includes a fuel oil storage tank and a fuel oil day tank. Fuel is transferred from the fuel oil storage tank to the fuel oil day tank by the diesel oil transfer pump. Transfer is accomplished automatically by level switches on the day tank. During EDG start up operations, fuel from the fuel oil day tank is drawn through the strainer via the fuel oil priming pump. Once started and running, fuel from the fuel oil day tank is drawn through the engine-driven fuel oil pump and discharged through the duplex fuel oil filter and on to the diesel engine injectors. Any excess fuel is returned to the fuel oil day tank. At Dresden, the Unit 2 EDG fuel oil transfer system also supplies fuel oil to the isolation condenser makeup pump fuel oil day tanks, and the Unit 3 EDG fuel oil transfer system also supplies fuel oil to the Unit 2/3 diesel fire pump day tank. Each operation requires opening a manual isolation valve. At Quad Cities, the Unit 1 and Unit 2 EDG fuel oil transfer systems also supply fuel oil for the diesel fire pump day tanks. Either the Unit 1 or Unit 2 EDG fuel oil transfer pump can be aligned to provide manual fill capability to the diesel fire pump day tanks.

System Evaluation Boundary

The evaluation boundary for each EDG fuel oil system begins with the diesel fuel oil storage tank, passes through the transfer pump to the associated day tank, through the diesel fuel oil priming and supply pumps, to the piping connections on the diesels and back to the day tank. The boundary also includes the day tanks for the diesel fire pump diesels and the isolation condenser makeup pump diesels (at Dresden) and all associated fuel oil system piping. Also included is the 150,000 gallon storage tank for the heating boiler system at Dresden (evaluated with the plant heating system) and supporting components. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the diesel fuel oil system boundary.

UFSAR References

Dresden Station UFSAR Section(s): 9.5.4

Quad Cities UFSAR Section(s): 9.5.4

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE-M-41-1, LR-DRE-M-41-2, LR-DRE-M-518-1, LR-DRE-M-518-2, LR-DRE-M-518-3, LR-DRE-M-4204.

Quad Cities Station:
LR-QDC-M-29-1 and LR-QDC-M-29-2.

System Intended Functions

Support ESF function(s) - stores and provides a source of clean diesel fuel oil to the emergency diesel generators which supply on-site AC power to ESF systems.

Credited in regulated events - stores and provides oil to the diesel fire pump (10CFR50.48) and the isolation condenser system (at Dresden only) diesel.

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.3-13 Component Groups Requiring Aging Management Review - Diesel Fuel Oil System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.3.1.22, 3.3.2.18
Filters/Strainers	Pressure Boundary	3.3.1.5, 3.3.1.7, 3.3.1.20, 3.3.2.21, 3.3.2.50, 3.3.2.54, 3.3.2.300
Filters/Strainers	Filter	3.3.1.7, 3.3.1.20
Flame Arrestors	Fire Barrier	3.3.1.5
Piping and Fittings	Pressure Boundary	3.3.1.5, 3.3.1.7, 3.3.2.29, 3.3.2.139
Piping and Fittings (attached support)	Structural Integrity (attached)	3.3.1.5, 3.3.1.7
Pumps	Pressure Boundary	3.3.1.5, 3.3.1.7, 3.3.2.176, 3.3.2.300
Restricting Orifices (Quad Cities only)	Pressure Boundary	3.3.2.40, 3.3.2.191
Sight Glasses	Pressure Boundary	3.3.2.36, 3.3.2.201
Tanks	Pressure Boundary	3.3.1.5, 3.3.1.7, 3.3.1.20, 3.3.2.215, 3.3.2.217, 3.3.2.218
Tubing	Pressure Boundary	3.3.2.40, 3.3.2.247
Valves	Pressure Boundary	3.3.1.5, 3.3.1.7, 3.3.1.20, 3.3.2.40, 3.3.2.259, 3.3.2.288, 3.3.2.300
Valves (attached support)	Structural Integrity (attached)	3.3.1.5, 3.3.1.7, 3.3.1.20

Aging management review results for the diesel fuel oil system are provided in Section 3.3.

2.3.3.14 Process Sampling System

System Purpose

The process sampling systems/subsystems are used to monitor process parameters from various systems.

System Operation

Samples are taken and analyzed on a continuous and/or laboratory basis. Selected parameters are recorded and/or alarmed. At Dresden, the process sampling systems include nitrogen inerting and drywell oxygen sampling, turbine building and radwaste air particulate sampling, drywell air particulate sampling, and off-gas building air particulate sampling. At Quad Cities, the process sampling systems include the drywell oxygen analysis, the drywell air particulate, and the turbine building particulate sampling. Sample lines that penetrate the primary containment are provided with isolation valves.

System Evaluation Boundary

At Dresden, the evaluation boundary includes the process sampling systems for nitrogen inerting and drywell oxygen sampling, drywell air particulate sampling, turbine building and radwaste air particulate sampling, and off-gas building air particulate sampling. These systems are either unitized or common and include the following subsystems:

- Air sample manifold system (unitized)
- Containment oxygen sampling system (unitized)
- Drywell (primary containment) air particulate sampling system (unitized)
- Turbine building air particulate sampling system (unitized)
- Off-gas filter building air particulate sampling system (common)
- Off-gas sampling system (unitized)
- Turbine building sample panel (unitized)
- Reactor building sampling system (unitized)
- Feedwater particulate sampling system (unitized)
- Radwaste building sampling system (common)
- Off-gas filter building constant air monitoring (common)
- Reactor building crane monitor (common).

At Quad Cities, the evaluation boundary includes the process sampling systems for the drywell oxygen analysis, the drywell air particulate, turbine building particulate sampling and off-gas filter building continuous air monitor. These systems are either unitized or common and include the following subsystems:

- Primary containment sampling systems (unitized)
- Turbine building air particulate sampling (unitized)
- Reactor building vent particulate sampling (unitized)
- Turbine building sample panel (unitized)
- Reactor building sample panel (unitized)
- Off gas sampling (unitized)
- Off gas filter building sample racks (unitized)
- Radwaste building sample system (common)
- Off gas filter building continuous air monitor (common)

UFSAR References

Dresden Station UFSAR Section(s): 1.2.2.4, 5.2.5.6.1, 6.2.1, 7.3.2, 9.3.2

Quad Cities Station UFSAR Section(s): 1.2, 5.2, 6.2, 7.3, 9.3.2,

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-25, LR-DRE-M-178, LR-DRE-M-356, LR-DRE-M-421

Quad Cities Station: LR-QDC-M-461-1, LR-QDC-M-463-1

System Intended Functions

Primary containment isolation – provides isolation of air sampling system piping penetrating primary containment.

Component Groups Requiring Aging Management Review

Table 2.3.3-14 Component Groups Requiring Aging Management Review –Process Sampling System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.3.1.22
NSR Vents or Drains, Piping and Valves (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.2.130
Piping and Fittings (Quad Cities only)	Pressure Boundary	3.3.1.5, 3.3.2.40, 3.3.2.145, 3.3.2.166
Piping and Fittings (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.2.145
Tubing	Pressure Boundary	3.3.2.34, 3.3.2.40, 3.3.2.42, 3.3.2.244, 3.3.2.254
Valves	Pressure Boundary	3.3.1.5, 3.3.2.23, 3.3.2.40, 3.3.2.264, 3.3.2.273, 3.3.2.293, 3.3.2.295
Valves (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.3.1.5, 3.3.2.40, 3.3.2.266, 3.3.2.285
Valves (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.2.273

Aging management review results for the process sampling system are provided in Section 3.3.

2.3.3.15 Carbon Dioxide System

System Purpose

The carbon dioxide system supplies carbon dioxide for fire suppression and generator purge functions. Carbon dioxide fire suppression systems are installed to quickly detect and suppress fires.

System Operation

The carbon dioxide system consists of a liquid carbon dioxide storage tank with a refrigeration compressor (the Cardox Unit), a steam driven carbon dioxide vaporizer unit, and two headers for fire suppression. The vaporizer unit is supplied with steam from the heating steam system (evaluated with the heating steam system) and provides purge gas to the main generators. One of the fire suppression headers supplies carbon dioxide to manually operated hose reels located throughout the plant. The other supplies the total flooding distribution manifold that services the three emergency diesel generator rooms, the two alternator exciters in the main generator housings and, at Dresden only, the auxiliary equipment room. The system also includes automatic control valves and sensors to provide the automatic initiation of the carbon dioxide fire suppression systems.

System Evaluation Boundary

The carbon dioxide system evaluation boundary begins with the carbon dioxide storage tank and continues through the two headers and automatic control valves to the hose reels and the emergency diesel generator rooms, alternator exciters, and auxiliary equipment room (at Dresden only.) via the flooding distribution manifold. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the carbon dioxide system boundary.

UFSAR References

Dresden Station UFSAR Section(s): None

Quad Cities Station UFSAR Section(s): None

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-42.

Quad Cities Station:

LR-QDC-M-30-1, LR-QDC-M-30-2, and LR-QDC-M-30-3.

System Intended Functions

Credited in regulated event(s) - functions to demonstrate compliance with fire protection requirements of 10 CFR 50.48.

Component Groups Requiring Aging Management Review

Table 2.3.3-15 Component Groups Requiring Aging Management Review - Carbon Dioxide System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.3.1.22
Piping and Fittings (includes thermowells and nozzles)	Pressure Boundary	3.3.1.5, 3.3.2.138
Tanks	Pressure Boundary	3.3.1.5, 3.3.2.212
Tubing	Pressure Boundary	3.3.1.5, 3.3.2.234
Valves (includes nozzles)	Pressure Boundary	3.3.1.5, 3.3.2.23, 3.3.2.260, 3.3.2.268

Aging management review results for carbon dioxide systems are provided in Section 3.3.

2.3.3.16 Service Water System

System Purpose

The service water system provides strained river water to cool various loads in the reactor building, turbine building, and auxiliary building. The service water system is a non-safety related system that provides cooling water to components required for normal operation.

System Operation

The service water system is an open loop system. The system has five pumps, three strainers, and a common distribution header. Two service water pumps are provided per unit and the fifth shared pump is used as a backup. During normal plant operation, four pumps typically operate with the fifth in standby. The system is cross connected between the units. All five pumps take suction from bays within the crib house filled with river water and discharge into a common header. The header supplies the three strainers in parallel, with the strainer output flowing into a common header with branches that supply the loads listed below. The majority of the loads are heat exchangers, coolers, and condensers. Each service water load discharges into one of two standpipes on each unit, which connect to the plant discharge flume. Heat exchangers that can potentially leak radioactively contaminated water into the service water system discharge into the largest of the two discharge standpipes, which is monitored for radioactive contamination by a radiation monitor which will alarm in the control room if high radiation is sensed. The service water loads include the following systems/components :

- Reactor building closed cooling water (RBCCW) heat exchangers
- Turbine building closed cooling water (TBCCW) heat exchangers
- Traveling screen wash spray
- Fire protection system
- Turbine oil coolers
- Reactor recirculation pump M-G set oil coolers
- Generator hydrogen coolers
- Generator stator water coolers
- Standby coolant supply
- Control room air conditioning condensers
- Auxiliary electric equipment room air conditioning condenser (Dresden only)
- Off-gas glycol chillers
- X-area (Dresden)/MSIV room (Quad Cities) coolers (steam tunnel coolers)
- Off-gas filter building sample system heat exchanger
- Control rod drive pump coolers (Dresden only, and only as a backup to TBCCW)

- Radwaste max recycle condensate holding tanks (Dresden only)
- Maximum recycle concentrator condensers (Dresden only)
- Containment cooling service water (CCSW) keep fill (Dresden only)
- Low pressure coolant injection (LPCI) room coolers (normal, non-emergency cooling water supply at Dresden only)
- High pressure coolant injection (HPCI) room coolers (normal, non-emergency cooling water supply)
- 1B instrument air compressor heat exchanger (Quad Cities)
- Station heating system condensate cooler (Quad Cities)
- Service building air conditioning compressors (Quad Cities)
- Safe shutdown makeup pump room cooler (Quad Cities)

System Evaluation Boundary

The service water system evaluation boundary begins with the suction of the service water pumps from the common intake structure (crib house). The boundary continues through the common header through the service water strainers to the distribution header for the plant equipment. After cooling various systems and components, the service water is discharged through one of two discharge headers in each unit. The system evaluation boundary includes the components and piping associated with these flow paths as well as the discharge piping.

UFSAR References

Dresden Station UFSAR Section(s): 9.2.2, Table 9.2-2, 9.2.8, 9.5.5

Quad Cities Station UFSAR Section(s): 6.0, 6.0.1.10, 6.4.2.2, 9.2.2, Table 9.2-2, 9.2.8, 9.3.1, 9.3.5, 9.5.1, 9.5.5

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE-M-21, LR-DRE-M-22, LR-DRE-M-23-1, LR-DRE-M-29-2, LR-DRE-M-354-2, LR-DRE-M-355, LR-DRE-M-360-2, LR-DRE-M-626, LR-DRE-M-722, and LR-DRE-M-3121.

Quad Cities Station:
LR-QDC-M-22-1, LR-QDC-M-22-4, LR-QDC-M-22-5, LR-QDC-M-69-1, LR-QDC-M-69-4, and LR-QDC-M-69-5.

System Intended Functions

Preclude adverse effects on safety-related SSC's – Non-safety related components that could spatially interact and be a hazard to safety related SSC's maintain sufficient integrity so that the intended function of safety related SSC's is not adversely affected.

Structural support – non-safety related portions of this system provide structural support to attached safety related piping. (Quad Cities only)

Pressure boundary –maintain pressure boundary of the CCSW keep fill line. (Dresden only)

Support ESF function(s) –provides cooling water to ECCS room coolers to ensure proper environment for ECCS pump operation. (Dresden only)

Credited in regulated event(s) –provides cooling water to safe shutdown equipment credited in mitigation of the Appendix R fire. (Dresden only)

Emergency makeup – provides an alternate supply of water for makeup to the isolation condenser. (Dresden only)

Plant component cooling – provide strained cooling water RBCCW. (Dresden only)

Component Groups Requiring Aging Management Review

Table 2.3.3-16 Component Groups Requiring Aging Management Review – Service Water System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.3.1.22, 3.3.2.19
Flow Orifices (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.1.5, 3.3.1.15
Orifice Bodies (Dresden only)	Pressure Boundary	3.3.1.15, 3.3.2.40
Orifice Bodies (Dresden only)	Throttle	3.3.1.15
Piping and Fittings (spatial interaction)	Leakage Boundary (spatial)	3.3.1.5, 3.3.1.15
Piping and Fittings (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.15, 3.3.2.26, 3.3.2.29, 3.3.2.40, 3.3.2.45, 3.3.2.169
Piping and Fittings (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.1.5, 3.3.1.15
Pumps (Dresden only)	Pressure Boundary	3.3.2.170, 3.3.2.171, 3.3.2.179, 3.3.2.180, 3.3.2.300
Strainer Bodies (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.15, 3.3.2.208, 3.3.2.300
Strainer Screens (Dresden only)	Filter	3.3.1.15
Thermowells (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.15
Tubing (Dresden only)	Pressure Boundary	3.3.2.34, 3.3.2.242
Valves (spatial interaction)	Leakage Boundary (spatial)	3.3.1.15, 3.3.1.27, 3.3.2.23, 3.3.2.279, 3.3.2.280, 3.3.2.281, 3.3.2.300

Table 2.3.3-16 Component Groups Requiring Aging Management Review – Service Water System (Continued)

Component Group	Component Intended Function	Aging Management Ref
Valves (attached support)	Structural Integrity (attached)	3.3.1.15, 3.3.1.27, 3.3.2.23, 3.3.2.279, 3.3.2.280, 3.3.2.300
Valves (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.13, 3.3.1.15, 3.3.1.27, 3.3.2.23, 3.3.2.31, 3.3.2.40, 3.3.2.45, 3.3.2.262, 3.3.2.276, 3.3.2.277, 3.3.2.279, 3.3.2.280, 3.3.2.298, 3.3.2,300

Aging management review results for the service water system are provided in Section 3.3.

2.3.3.17 Reactor Building Closed Cooling Water

System Purpose

The purpose of the reactor building closed cooling water (RBCCW) system is to provide cooling for equipment and systems in the reactor building while also minimizing the potential for release of radioactive material from the reactor equipment to the service water. For Dresden only, the system also provides cooling water to the shutdown cooling heat exchangers.

System Operation

The RBCCW system is a closed loop system. An expansion tank is connected to the common RBCCW pump suction header to ensure adequate net positive suction head for the pumps (two for each unit, with one additional shared spare). A level control valve (evaluated with the clean demineralized water and makeup system) provides automatic makeup to the expansion tank. The pumps discharge into a common header from which cooling water is provided to loads arranged in several loops. One loop provides cooling to reactor building auxiliary loads such as the reactor water clean up non-regenerative heat exchangers, the fuel pool heat exchangers, and the reactor building equipment drain tank heat exchanger. A loop inside the primary containment provides cooling to the primary containment coolers, primary containment equipment drain sump heat exchanger, and the reactor recirculation pump seals and motor oil coolers. Primary containment isolation valves are provided for the RBCCW lines penetrating the primary containment. For Dresden only, a loop also provides cooling to the shutdown cooling heat exchangers. The loops all discharge into a common header at the inlet to the RBCCW heat exchangers (also two for each unit with one additional shared spare). The RBCCW discharge from the heat exchangers flows back into the RBCCW pump suction header. The service water system cools the RBCCW heat exchangers. The pressure in the RBCCW system is maintained lower than that of the service water system, ensuring that any heat exchanger leakage would be into the RBCCW and not into the service water. Additionally, the RBCCW discharge radiation monitor (evaluated with the process radiation monitoring system) alarms upon sensing a high radiation condition in the closed cooling water.

System Evaluation Boundary

The RBCCW system evaluation boundary includes loop components in the closed cooling water system such as the RBCCW expansion tank, RBCCW pumps, RBCCW heat exchanger, primary containment penetration, primary containment isolation valves and other equipment isolation and check valves. The cooled loads, such as heat exchangers, pump coolers, cooling coils, compressor coolers, and sample coolers are evaluated with their respective systems. All other associated piping, components and instrumentation contained within the flow paths and systems described above are included in the RBCCW system evaluation boundary.

UFSAR References

Dresden Station UFSAR Section(s): 9.2.3, Table 9.2-3

Quad Cities Station UFSAR Section(s): 9.2.3, Table 9.2-3

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-20, LR-DRE-M-353, and LR-DRE-M-367-2.

Quad Cities Station:

LR-QDC-M-33-1, LR-QDC-M-33-2, LR-QDC-M-75-1, and LR-QDC-M-75-2

System Intended Functions

Primary containment isolation – provide primary containment isolation for those portions of the system that interface with the primary containment.

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Credited in regulated event(s) – at Dresden only, provides cooling water to the shutdown cooling heat exchangers to achieve and maintain cold shutdown during an Appendix R fire. At Dresden and Quad Cities the system also contains components that are relied upon for compliance with 10 CFR 50.49 (EQ).

Component Groups Requiring Aging Management Review

Table 2.3.3-17 Component Groups Requiring Aging Management Review – Reactor Building Closed Cooling Water System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.3.1.22
Flow Elements (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.2.68
Heat Exchangers (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.2.116, 3.3.2.117
Heat Exchangers (Dresden only)	Heat Transfer	3.3.2.112
Heat Exchangers (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.3.1.5, 3.3.2.77, 3.3.2.78
Manifolds (Dresden only)	Pressure Boundary	3.3.1.13, 3.3.2.40, 3.3.2.128
NSR Vents or Drains, Piping and Valves (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.2.130
Orifice Bodies (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.13

Table 2.3.3-17 Component Groups Requiring Aging Management Review – Reactor Building Closed Cooling Water System (Continued)

Component Group	Component Intended Function	Aging Management Ref
Piping and Fittings	Pressure Boundary	3.3.1.5, 3.3.1.13, 3.3.2.40, 3.3.2.137, 3.3.2.159
Piping and Fittings (spatial interaction) (Quad Cities only) (Includes flow elements)	Leakage Boundary (spatial)	3.3.1.5, 3.3.1.13, 3.3.2.40
Piping and Fittings (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.1.5, 3.3.1.13, 3.3.2.40
Pumps (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.13, 3.3.2.174
Tanks (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.13, 3.3.2.211
Thermowells (Dresden only)	Pressure Boundary	3.3.1.13, 3.3.2.23, 3.3.2.40, 3.3.2.220, 3.3.2.221, 3.3.2.225
Tubing (Dresden only)	Pressure Boundary	3.3.1.13, 3.3.2.233
Valves	Pressure Boundary	3.3.1.5, 3.3.1.13, 3.3.2.40, 3.3.2.267, 3.3.2.286, 3.3.2.293
Valves (spatial interaction)	Leakage Boundary (spatial)	3.3.1.5, 3.3.1.13, 3.3.2.266, 3.3.2.267
Valves (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.1.5, 3.3.1.13, 3.3.2.267

Aging management review results for the reactor building closed cooling water system are provided in Section 3.3.

2.3.3.18 Turbine Building Closed Cooling Water System (In-Scope for Dresden Only)

System Purpose

The turbine building closed cooling water (TBCCW) system provides the means for heat rejection from systems located in the turbine building and crib house.

System Operation

The TBCCW system is a closed loop system consisting of pumps, heat exchangers, an expansion tank, and necessary control and support equipment. The system removes heat from the following loads: circulating water pumps, feed pump lube oil and mechanical seal coolers, condensate and condensate booster pump seal coolers, CRD pump seal coolers, instrument air compressors, resin transfer air compressors, service air compressors, radwaste sparging air compressors, EHC oil coolers, bus duct coolers, and main generator alternator exciter cooler. Station service water (evaluated with service water system) provides the cooling medium on the tube side of the TBCCW heat exchangers.

System Evaluation Boundary

The evaluation boundary for the TBCCW system encompasses the closed loop cooling system supply and return lines up to the inlet and outlet water connections on the equipment it services. The system consists of pumps, heat exchangers, an expansion tank, piping, valves and necessary control and support equipment. Its loads include circulating water pumps, feed pump lube oil and mechanical seal coolers, condensate and condensate booster pump seal coolers, CRD pump seal coolers, instrument air compressors, resin transfer air compressors, service air compressors, radwaste sparging air compressors, EHC oil coolers, bus duct coolers, and main generator alternator exciter cooler.

UFSAR References

Dresden Station UFSAR Section(s): 9.2.7, Table 9.2-4

Quad Cities Station UFSAR Section(s): Not applicable

License Renewal Boundary Diagram References

Dresden Station: LR-DRE-M-21, LR-DRE-M-354-1, LR-DRE-M-354-2

Quad Cities Station: Not applicable.

System Intended Functions

Credited in regulated event(s) – (Dresden only) provides flow path for control rod drive pump cooling during an Appendix R fire.

Component Groups Requiring Aging Management Review

Table 2.3.3-18 Component Groups Requiring Aging Management Review – Turbine Building Closed Cooling Water System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting (Dresden only)	Pressure Boundary	3.3.1.22
Heat Exchangers (Dresden only)	Pressure Boundary	3.3.2.91, 3.3.2.92
Piping and Fittings (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.13, 3.3.2.137
Valves (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.13, 3.3.2.267

Aging management review results for the turbine building closed cooling water system are provided in Section 3.3.

2.3.3.19 Demineralized Water Makeup System

System Purpose

The purpose of the demineralized water makeup system is to provide reactor quality water for use in power plant systems, equipment, and service drops.

System Operation

Well water flows from the well water transfer pumps (evaluated with the well water system at Quad Cities) through the makeup demineralizers, to either the clean or contaminated demin water storage tank. While Dresden has permanently installed demineralizers, both Dresden and Quad Cities utilize portable units. The clean demin transfer pumps take suction from the clean demin storage tank and supply clean demin water through a distribution header to plant systems and equipment, such as makeup to the standby liquid control system, the plant heating system, the RBCCW and TBCCW systems. It also supplies various system loop seals, the Unit 1 fuel building (Dresden only), sample panels, and clean demin service drops inside primary containment and throughout the plant. Containment isolation valves are provided for the clean demin lines that penetrate the containment. Additionally, at Dresden, the distribution header provides makeup water to the isolation condenser and emergency makeup water to the fuel pools. Also at Dresden, the isolation condenser makeup pumps take suction from the clean demin water storage tank and discharge into a common header that supplies both isolation condensers.

System Evaluation Boundary

The demineralized water makeup system boundary begins with the suction of the makeup demin well water transfer pumps at Dresden, and with the supply header to the makeup demineralizers at Quad Cities. The boundary ends at the final end uses of the clean demin water, whether it was drawn from either the clean demin water storage tank or the distribution header. Within this boundary certain loops or components were evaluated with other systems. The clean demin water storage tank was evaluated with the isolation condenser system at Dresden and with the condensate and condensate storage system at Quad Cities. At Dresden, the isolation condenser makeup pumps and associated piping were evaluated with the isolation condenser system. Also at Dresden, the safety related clean demin water fill valves for the isolation condenser, located in the common pathway from both the discharge header and the isolation condenser makeup pumps, were evaluated with the isolation condenser system. Valves utilized to provide a cross tie between the two Unit 2/3 contaminated condensate storage tanks at Dresden were evaluated with the condensate and condensate storage system. The safety related primary containment isolation valves for the clean demin water supply lines to the Dresden Unit 2 and Unit 3 fuel pools, that are part of the fuel pool system, were evaluated with the demineralized water makeup system. At Quad Cities, valves in the clean demin water supply lines to certain process samplers, such as the offgas particulate sampler, the chemical equipment rooms, and the containment oxygen analyzers, were evaluated with the process sampling system. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the demineralized water makeup system evaluation boundary.

UFSAR References

Dresden Station UFSAR Section(s): 9.2.4

Quad Cities Station UFSAR Section(s): 9.2.4

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE-M-20, LR-DRE-M-31, LR-DRE-M-33, LR-DRE-M-35-1, LR-DRE-M-39, LR-DRE-M-177A-1, LR-DRE-M-353, LR-DRE-M-359, LR-DRE-M-362, LR-DRE-M-364, LR-DRE-M-366,LR-DRE-M-369,LR-DRE-M-419A-1,LR-DRE-M-423-1,LR-DRE-M-423-2, LR-DRE-M-423-3, LR-DRE-M-423-4, LR-DRE-M-423-5, LR-DRE-M-1234-3, and LR-DRE-M-1239-3

Quad Cities Station:
LR-QDC-M-58-1, LR-QDC-M-58-3, LR-QDC-M-58-4, LR-QDC-M-459A-2, and LR-QDC-M-462A-2

System Intended Functions

Primary containment isolation – provides containment isolation for those portions of the system that interface with the primary containment.

Isolation condenser alternate make up water (Dresden) – provides alternate makeup water to the isolation condenser.

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.3-19 Component Groups Requiring Aging Management Review - Makeup Demineralizer System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.3.1.22, 3.3.2.18
Flow Elements (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.3.2.40, 3.3.2.72
NSR Vents or Drains, Piping and Valves (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.2.130
Piping and Fittings	Pressure Boundary	3.3.1.5, 3.3.1.8, 3.3.1.25, 3.3.2.22, 3.3.2.29, 3.3.2.43, 3.3.2.143, 3.3.2.302
Piping and Fittings (spatial interaction)	Leakage Boundary (spatial)	3.3.1.5, 3.3.2.143

Table 2.3.3-19 Component Groups Requiring Aging Management Review - Makeup Demineralizer System (Continued)

Component Group	Component Intended Function	Aging Management Ref
Piping and Fittings (attached support)	Structural Integrity (attached)	3.3.1.5, 3.3.2.143
Pumps (Dresden only)	Pressure Boundary	3.3.2.182, 3.3.2.300
Pumps (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.3.2.182, 3.3.2.300
Restricting Orifices (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.2.188
Restricting Orifices (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.3.2.23, 3.3.2.186
Strainers (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.3.1.5, 3.3.2.209
Valves	Pressure Boundary	3.3.1.5, 3.3.2.23, 3.3.2.29, 3.3.2.272
Valves (spatial interaction)	Leakage Boundary (spatial)	3.3.1.5, 3.3.1.25, 3.3.2.40, 3.3.2.257, 3.3.2.272
Valves (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.1.5, 3.3.2.272

Aging management review results for makeup demineralizer system are provided in Section 3.3.

2.3.3.20 Residual Heat Removal Service Water (Quad Cities Only)

System Purpose

The residual heat removal service water (RHRSW) system operates in conjunction with the residual heat removal (RHR) system to remove heat from the suppression chamber through the containment cooling mode of RHR or to remove heat from the reactor coolant through the reactor shutdown cooling mode of RHR. Additionally, the RHRSW provides safety related cooling water to the train "B" control room HVAC refrigerant condensing unit as a back-up during a loss of offsite power and provides a cross-tie to the opposite unit to achieve safe shutdown in an Appendix "R" Fire. The RHR system also provides an auxiliary function during refueling to assist in removal of heat from the spent fuel pool.

System Operation

The residual heat removal service water (RHRSW) system at Quad Cities is an open loop cooling water system consisting of four two-stage pump sets, associated valves, piping, and instrumentation and controls, divided into two independent loops. The RHRSW system removes heat from the RHR heat exchangers, which are evaluated as part of the RHRSW system. When RHRSW is flowing, the pressure on the tube side of the RHR heat exchanger is maintained above the shell side to prevent reactor water leakage into the service water and thereby into the discharge bay. Additionally, the RHRSW provides safety related cooling water to the train "B" control room HVAC refrigerant condensing unit (evaluated with control room HVAC) as a back-up during a loss of offsite power and provides a cross tie to the opposite unit to achieve safe shutdown in an Appendix R fire.

System Evaluation Boundary

The RHRSW system starts at the crib house RHRSW intake bay. Suction piping brings the strained river water to two RHRSW pump suction manifolds. The four RHRSW pump sets (two pumps driven by one motor) take suction from the manifolds. The flow continues to the RHRSW pumps. The RHRSW pumps are located in watertight vaults. Each vault has one vault cooler for each pump in the vault. Cooling water for each vault cooler is supplied by its associated pump discharge. In the suction paths are the returns from the RHRSW pump vault coolers and pump seal supplies. The pumps discharge to the RHR heat exchangers, which are evaluated as part of the RHRSW system. A cross-tie to the opposite unit is also provided. The RHRSW system ends with a common discharge to the service water system discharge (evaluated with service water). All associated piping, components (including vents, drains, test taps and relief valves) and instrumentation contained within the flow paths and systems described above are included in the RHRSW system evaluation boundary.

UFSAR References

Dresden Station UFSAR Section(s): Not applicable.

Quad Cities Station UFSAR Section(s): 3.4.1.2, 5.4.7, 6.2.2, 6.3, 9.1.3.2, 9.2.1, 9.2.5

License Renewal Boundary Diagram References

Dresden Station: Not applicable.

Quad Cities Station: LR-QDC-M-22-1, LR-QDC-M-37, LR-QDC-M-39-3, LR-QDC-M-79, LR-QDC-M-81-3, and LR-QDC-M-725-3.

System Intended Functions

Containment and component cooling - provides a heat sink for the RHR system via the RHR heat exchangers to support containment cooling after a loss of coolant accident (LOCA).

Back-up cooling – provides safety related back-up cooling to the 'B' train of the control room HVAC refrigeration units as a back-up during a loss of offsite power and LOCA.

Credited in regulated events - provides a heat sink for the RHR system via the RHR heat exchangers to support ATWS actions and in the Appendix R fire safe shutdown analysis. The subsystems between the units can be connected by a normally isolated crosstie line that is credited in the plant's fire protection safe shutdown analysis. The system also contains components that are relied upon for compliance with 10 CFR Part 50.49 (EQ).

Preclude adverse effects on safety-related SSC's – Non-safety related components that could be a hazard to safety related SSC's maintain sufficient integrity so that the intended function of safety related SSC's is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.3-20 Component Groups Requiring Aging Management Review – Residual Heat Removal Service Water System (Quad Cities only)

Component Group	Component Intended Function	Aging Management Ref
Air Handlers Heating/Cooling (Aux&RW HVAC) (Quad Cities only)	Pressure Boundary	3.3.2.8, 3.3.2.9
Air Handlers Heating/Cooling (Aux&RW HVAC) (Quad Cities only)	Heat Transfer	3.3.2.7
Closure Bolting (Quad Cities only)	Pressure Boundary	3.3.1.22
Dampeners (Quad Cities only)	Pressure Boundary	3.3.1.15, 3.3.1.27, 3.3.2.24
Ducts & Fittings, Access Doors, Closure Bolts, Equip Frames (Quad Cities only)	Pressure Boundary	3.3.1.5
Heat Exchangers (Quad Cities only)	Pressure Boundary	3.3.2.26, 3.3.2.31, 3.3.2.89, 3.3.2.90, 3.3.2.118, 3.3.2.119, 3.3.2.120, 3.3.2.121

Table 2.3.3-20 Component Groups Requiring Aging Management Review – Residual Heat Removal Service Water System (Quad Cities only) (Continued)

Component Group	Component Intended Function	Aging Management Ref
Heat Exchangers (Quad Cities only)	Heat Transfer	3.3.2.88, 3.3.2.113, 3.3.2.115
NSR Vents or Drains, Piping and Valves (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.2.130
Orifice Bodies (Quad Cities only)	Pressure Boundary	3.3.1.15, 3.3.2.40, 3.3.2.41
Orifice Bodies (Quad Cities only)	Throttle	3.3.1.15
Piping and Fittings (Quad Cities only)	Pressure Boundary	3.3.1.5, 3.3.1.15, 3.3.1.16, 3.3.2.26, 3.3.2.44, 3.3.2.141, 3.3.2.168
Piping and Fittings (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.1.5, 3.3.1.15, 3.3.2.26
Pulsation Dampeners (Quad Cities only)	Pressure Boundary	3.3.1.15, 3.3.2.24, 3.3.2.41
Pumps (Quad Cities only)	Pressure Boundary	3.3.1.15, 3.3.2.26, 3.3.2.173, 3.3.2.183, 3.3.2.184
Sight Glasses (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.1.5, 3.3.1.15
Strainer Bodies (Quad Cities only)	Pressure Boundary	3.3.1.5, 3.3.1.15
Strainer Screens (Quad Cities only)	Filter	3.3.1.15
Thermowells (Quad Cities only)	Pressure Boundary	3.3.1.5, 3.3.1.15
Tubing (Quad Cities only)	Pressure Boundary	3.3.1.15, 3.3.2.40, 3.3.2.41
Tubing (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.1.15, 3.3.2.40, 3.3.2.41
Valves (Quad Cities only)	Pressure Boundary	3.3.1.5, 3.3.1.15, 3.3.1.27, 3.3.2.23, 3.3.2.24, 3.3.2.26, 3.3.2.31, 3.3.2.40, 3.3.2.41, 3.3.2.279, 3.3.2.280, 3.3.2.297, 3.3.2.300
Valves (attached support) (Quad Cities only)	Structural Integrity (attached)	3.3.1.5, 3.3.1.15, 3.3.1.27, 3.3.2.23, 3.3.2.24, 3.3.2.26, 3.3.2.31, 3.3.2.276, 3.3.2.277, 3.3.2.300

Aging management review results for the residual heat removal service water system are provided in Section 3.3.

2.3.3.21 Containment Cooling Service Water System (Dresden Only)

System Purpose

The containment cooling service water (CCSW) System removes heat from the primary containment by providing cooling water to the low pressure coolant injection (LPCI) heat exchangers. CCSW, working with LPCI, limits suppression chamber bulk water temperature assuring: 1) Suppression chamber hydrodynamic loads during blowdown are limited maintaining structure and equipment integrity; 2) Complete steam condensation during a loss of coolant accident limiting long term primary containment pressure; and 3) Adequate NPSH for ECCS pumps maintaining long term primary containment pressure control. The Unit 2 CCSW loops provide a safety related source of service water to the control room air conditioning condensers. The CCSW system also supplies a safety related source of river water to the LPCI and HPCI room coolers as a backup to the service water system.

System Operation

The containment cooling service water (CCSW) system is an open loop system consisting of four pumps, associated valves, piping, and instrumentation and controls. The system removes heat from the low pressure coolant injection (LPCI) system (evaluated with LPCI system) through the LPCI heat exchangers. The CCSW pumps develop sufficient head to maintain the cooling water heat exchanger tube side outlet pressure greater than the LPCI subsystem pressure on the shell side (evaluated with LPCI system). Maintaining this pressure differential prevents reactor water leakage into the service water and thereby into the river. The Unit 2 CCSW loops provide a safety related source of service water to the control room air conditioning condensers (evaluated with control room HVAC). The CCSW system also supplies a safety related source of river water to the LPCI and HPCI room coolers (evaluated with ECCS corner room HVAC) as a backup to the service water system.

System Evaluation Boundary

The evaluation boundary for the CCSW system begins with each pair of CCSW pumps taking suction from the crib house via separate supply piping. Two CCSW pumps discharge into a common header that routes the cooling water to that loop's associated heat exchanger. At the heat exchanger, heat is transferred from the LPCI subsystem (evaluated with LPCI system) to the CCSW system, and subsequently to the river. System piping is arranged to form two separate, two-pump flow networks (loops) to the piping downstream of the differential pressure control valve on the discharge of the heat exchanger. At this point, both piping loops merge into a common discharge line to the service water discharge header (evaluated with service water system).

UFSAR References

Dresden Station UFSAR Section(s): 3.4.1.2, 6.2.2, 6.3.1.2, 6.4, 9.2.1

Quad Cities Station UFSAR Section(s): Not applicable.

License Renewal Boundary Diagram References

Dresden Station: LR-DRE-M-22, LR-DRE-M-29-1, LR-DRE-M-29-2,

LR-DRE-M-360-1, LR-DRE-M-360-2, and LR-DRE-M-3121

Quad Cities Station: Not applicable.

System Intended Functions

Containment and component cooling - provides containment cooling function, cooling to the ECCS room coolers and the containment cooling service water vault coolers to maintain room temperatures.

Back-up cooling – (Dresden Unit 2 only) containment cooling service water provides safety related back-up cooling to the 'B' train of the control room HVAC refrigeration units.

Credited in regulated events - provides redundancy in suppression chamber cooling during an ATWS event, operates without reliance upon external sources of power (SBO), is credited in the Appendix R fire safe shutdown analysis and the system contains components that are relied upon for compliance with 10 CFR Part 50.49 (EQ).

Preclude adverse effects on safety-related SSC's – Non-safety related components that could be a hazard to safety related SSC's maintain sufficient integrity so that the intended function of safety related SSC's is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.3-21 Component Groups Requiring Aging Management Review – Containment Cooling Service Water System (Dresden only)

Component Group	Component Intended Function	Aging Management Ref
Air Handlers Heating/Cooling (Aux&RW HVAC) (Dresden only)	Pressure Boundary	3.3.2.8, 3.3.2.9
Air Handlers Heating/Cooling (Aux&RW HVAC) (Dresden only)	Heat Transfer	3.3.2.7
Closure Bolting (Dresden only)	Pressure Boundary	3.3.1.22
Ducts & Fittings, Access Doors, Closure Bolts, Equip Frames (Dresden only)	Pressure Boundary	3.3.1.5
Flow Elements (Dresden only)	Pressure Boundary	3.3.1.15, 3.3.2.40, 3.3.2.41
Heat Exchangers (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.2.80, 3.3.2.81
Heat Exchangers (Dresden only)	Heat Transfer	3.3.2.79

Table 2.3.3-21 Component Groups Requiring Aging Management Review – Containment Cooling Service Water System (Dresden only) (Continued)

Component Group	Component Intended Function	Aging Management Ref
Orifice Bodies (Dresden only)	Pressure Boundary	3.3.1.15, 3.3.2.40, 3.3.2.41
Piping and Fittings (Dresden only) (includes manifolds, tubes, and thermowells)	Pressure Boundary	3.3.1.5, 3.3.1.15, 3.3.1.16, 3.3.2.26, 3.3.2.40, 3.3.2.141
Piping and Fittings (attached support) (Dresden only)	Structural Integrity (attached)	3.3.1.5, 3.3.1.15, 3.3.2.26
Pumps (Dresden only)	Pressure Boundary	3.3.2.31, 3.3.2.179, 3.3.2.180
Strainer Bodies (Dresden only)	Pressure Boundary	3.3.2.208, 3.3.2.300
Strainer Screens (Dresden only)	Filter	3.3.1.15
Thermowells (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.15
Tubing (Dresden only)	Pressure Boundary	3.3.1.15, 3.3.2.40, 3.3.2.41
Valves (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.1.15, 3.3.1.27, 3.3.2.23, 3.3.2.24, 3.3.2.26, 3.3.2.31, 3.3.2.40, 3.3.2.41, 3.3.2.279, 3.3.2.280, 3.3.2.300
Valves (attached support) (Dresden only)	Structural Integrity (attached)	3.3.2.279, 3.3.2.280, 3.3.2.300

Aging management review results for the containment cooling service water system are provided in Section 3.3.

2.3.3.22 Ultimate Heat Sink

System Purpose

The purpose of the ultimate heat sink (UHS) system is to provide sufficient cooling water to the station, when the normal heat sink (the river) is unavailable, to permit operation of the containment cooling service water (CCSW) system at Dresden, the residual heat removal service water (RHRSW) at Quad Cities, and the diesel generator cooling water (DGCW) pumps at both stations.

System Operation

With the failure of the Dresden Island Lock and Dam on the Illinois River, or Lock and Dam No.14 on the Mississippi River, the corresponding river (Kankakee or Mississippi) level would fall below the high point of the intake flume. In the event of a loss of this water source, there is a limited supply of water (the UHS) trapped, by design, in the intake and discharge canals.

At Dresden, the natural topography forms the UHS basin with the level stabilizing at elevation 495'-0," the high point of the intake flume. The CCSW pumps take suction from the center compartment of the crib house at elevation 498'-0," above the UHS basin level. This necessitates isolating the compartment and raising its water level. To accomplish this the wire mesh screens in the compartment openings are replaced with stop logs. The dewatering valves are opened to allow water from the crib house forebay to flood the trash rake refuge pit, the refuge pumps are lined up to discharge to the center compartment, and the pumps are operated to flood the compartment above the CCSW pump suction. A CCSW pump (evaluated with the CCSW system) is placed in service, discharging to the containment cooling heat exchanger, and then to the discharge canal. The deicing valve is opened, allowing flow from the discharge canal back to the forebay. A portable, low-head, high-volume engine-driven pump could makeup the loss of the impounded river water due to evaporation.

At Quad Cities, the natural topography of the intake flume, along with the weir gate located in the discharge canal, forms the UHS basin. The level in the basin stabilizes at elevation 565'-0," the high point of the intake flume. The RHRSW (evaluated with the RHR SW system) and DGCW pumps (evaluated with the diesel generator service water system) take suction from the center compartment of the crib house at elevation 456'-6," below the UHS basin level. The pumps discharge to their assigned loads, and then to the discharge flume upstream of the weir. The gate on the ice-melt line is opened, allowing flow from the discharge flume back to the intake flume. The water impounded in the intake and discharge flumes is then used as an evaporative heat sink. With the loss of the dam, river water backflow would occur from the river, through the 16' diameter discharge piping connecting the river to the discharge flume, to the downstream base of the weir gate. Portable, diesel-driven pumps take suction from downstream of the weir and discharge into the center compartment of the crib house to makeup the loss of the impounded river water due to evaporation.

System Evaluation Boundary

The ultimate heat sink evaluation boundary for Dresden consists of the topography basin between the high points of the intake and discharge flumes. It includes the screen wash refuge sump pumps and associated piping (from the screen wash system). And it also includes the stop logs, dewatering valves and associated piping, and the ice melt gate and associated piping (from the circulating water system). The ultimate heat sink evaluation boundary for Quad Cities consists of the topography basin between the high point of the intake flume and the weir gate located in the discharge canal. It also includes (from the circulating water system) the ice melt gate and associated piping, and the discharge piping connecting the discharge flume to the river. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the ultimate heat sink evaluation boundary.

UFSAR References

Dresden Station UFSAR Section(s): 9.2.5

Quad Cities Station UFSAR Section(s): 9.2.5

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-1, and LR-DRE-M-36

Quad Cities Station:

LR-QDC-M-1

System Intended Functions

Ultimate cooling water supply – provides sufficient cooling water to the station to permit operation of the containment cooling service water system and the diesel generator cooling water pumps when the normal heat sink (the river) is unavailable. (Dresden)

Ultimate cooling water supply – provides sufficient cooling water to the station to permit operation of the residual heat removal service water pumps and the diesel generator cooling water pumps when the normal heat sink (the river) is unavailable. (Quad Cities)

Component Groups Requiring Aging Management Review

Table 2.3.3-22 Component Groups Requiring Aging Management Review – Ultimate Heat Sink

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting (Dresden only)	Pressure Boundary	3.3.1.22
Piping and Fittings	Pressure Boundary	3.3.1.5, 3.3.1.15, 3.3.2.28, 3.3.2.141
Pump Casings (Dresden only)	Pressure Boundary	3.3.2.172, 3.3.2.300
Valves (Dresden only)	Pressure Boundary	3.3.2.278, 3.3.2.300

Aging management review results for the ultimate heat sink are provided in Section 3.3.

2.3.3.23 Fuel Pool Cooling and Filter Demineralizer System (In-Scope for Dresden Only)

System Purpose

The purpose of the fuel pool cooling and filter demineralizer system is to remove heat from the spent fuel and to maintain fuel storage pool water clarity. During refueling operations the system may be used to maintain the water clarity of the reactor refueling cavity also.

System Operation

Water from the fuel storage pool (evaluated with the reactor building structure) overflows via scuppers and an adjustable weir into two crosstied skimmer surge tanks. The skimmer surge tanks drain into a common suction header for the fuel pool cooling pumps. Two parallel flow paths exist from the header, each with a fuel pool cooling pump taking suction from the header and discharging through a fuel pool cooling heat exchanger. Cooling water to the heat exchangers is supplied from the reactor building closed cooling water system (evaluated with the reactor building closed cooling water system) A crosstie line exists on the pump discharge piping in order to operate either pump with either heat exchanger. The heat exchangers discharge into a common header, that first flows through the fuel pool filter, and then through the fuel pool demineralizer. The fuel pool demineralizer discharges back into the fuel storage pool through two lines and spargers within the pool. The return lines to the fuel storage pool enter near the top and have openings in the piping about six inches below the pool surface to act as anti-siphon devices, to preclude uncontrolled draining of the pool during a pipe break. During refueling operations, the system may be aligned via manual valves to discharge into the reactor refueling cavity. The shutdown cooling system (evaluated with the shutdown cooling system) may be connected in parallel with the fuel pool cooling and filter demineralizer system during periods of extremely high heat loads, such as immediately after refueling or a full core discharge into the fuel storage pool. A clean demineralized water supply (evaluated with the demineralized water makeup system) passes through a safety related primary containment isolation valve that is part of the fuel pool cooling and filter demineralized system, to supply makeup water.

System Evaluation Boundary

The fuel pool cooling and filter demineralizer system boundary begins with the skimmer surge tanks, continues through the fuel pool cooling pumps, fuel pool heat exchangers, fuel pool filter, fuel pool demineralizer, and ends with the fuel storage pool spargers. It also includes the portion of the makeup water to the fuel pool cooling and filter demineralizer system downstream of the safety related primary containment isolation valve that is part of the system (evaluated with the demineralized water makeup system). All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the fuel pool cooling and filter demineralizer system.

UFSAR References

Dresden Station UFSAR Section(s): 9.1.3

Quad Cities Station UFSAR Section(s): Not applicable

License Renewal Boundary Diagram References

Dresden Station: LR-DRE-M-31, LR-DRE-M-362

Quad Cities Station: Not applicable.

System Intended Functions

Preclude adverse effects on safety-related SSC's – (Dresden only) Components that could be a hazard to safety related SSC's maintain sufficient integrity so that the intended function of safety related SSC's is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.3-23 Component Groups Requiring Aging Management Review – Fuel Pool Cooling and Filter Demineralizer System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting (Dresden only)	Pressure Boundary	3.3.1.22
Piping and Fittings (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.2.145
Piping and Fittings (spatial interaction) (Dresden only)	Leakage Boundary (spatial)	3.3.1.1, 3.3.2.40
Sight Glasses (spatial interaction) (Dresden only)	Leakage Boundary (spatial)	3.3.1.5, 3.3.2.198, 3.3.2.199
Valves (Dresden only)	Pressure Boundary	3.3.1.5, 3.3.2.273
Valves (spatial interaction) (Dresden only)	Leakage Boundary (spatial)	3.3.1.5, 3.3.2.273

Aging management review results for the fuel pool cooling and filter demineralizer system are provided in Section 3.3.

2.3.3.24 Plant Heating System

System Purpose

The purpose of the plant heating system is to supply steam for plant and area heating during cold weather periods, and for miscellaneous functions such as steam cleaning and carbon dioxide or nitrogen vaporizing. Additionally, the Dresden plant heating system supplies steam to the shutdown cooling system during its operation in the reactor heating mode.

System Operation

The plant heating boiler feedwater pumps take suction from the heating system deaerating tank and discharge into the plant heating boilers. In addition, Quad Cities has a small, summer boiler with its own feedwater pumps taking suction from the deaerating tank. The boilers produce steam that flows into a common distribution header. The header discharges steam through a pressure control valve to various loops throughout the plant. Separate loops supply loads in each major building, such as the turbine buildings, reactor buildings, crib house, and radwaste building. From these loops steam flows to loads such as ventilation heating coils, area space heaters, vaporizers, and steam drops for uses such as steam cleaning. The condensate from the loads passes through steam traps to condensate return units located in the major building areas. The condensate return units pump the condensate back to the heating system deaerating tank at Dresden, and to the condensate receiving tank at Quad Cities. At Quad Cities, the condensate receiving tank pumps then pump the condensate back to the deaerating tank. At Dresden, each reactor building heating steam supply loop also provides steam to its unit’s shutdown heat exchangers (evaluated with the shutdown cooling system) for use in the reactor heating mode. As the steam supplied to the shutdown heat exchangers condenses, it drains via a steam trap to the reactor building equipment drain tank.

System Evaluation Boundary

The plant heating system boundary begins with the deaerating tank, and continues through the boiler feedwater pumps, boilers, pressure control valves, heating loads, steam traps, condensate return units, condensate receiving tank (Quad Cities only) and back to the deaerating tank. At Quad Cities, the pressure control valve that provides plant service water to the control room heating, ventilation and air conditioning system is considered as part of the plant heating system. This valve and associated piping were evaluated with the residual heat removal service water system. All other associated piping, components and instrumentation contained within the flow paths and systems described above are included in the plant heating system evaluation boundary.

UFSAR References

Dresden Station UFSAR Section(s): None

Quad Cities Station UFSAR Section(s): None

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE-M-32, LR-DRE-M-175-2, LR-DRE-M-175-3, LR-DRE-M-472-1, LR-DRE-M-472-2, and LR-DRE-M-363.

Quad Cities Station:
LR-QDC-M-22-1, LR-QDC-M-55-1, LR-QDC-M-56-1, LR-QDC-M-56-2, LR-QDC-M-56-3, LR-QDC-M-56-4, and LR-QDC-M-90.

System Intended Functions

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.3-24 Component Groups Requiring Aging Management Review – Plant Heating System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.3.1.22
Filters/Strainers (spatial interaction)	Leakage Boundary (spatial)	3.3.2.57, 3.3.2.300
NSR Vents or Drains, Piping and Valves (spatial interaction) (Dresden only)	Leakage Boundary (spatial)	3.3.2.130
Piping and Fittings (spatial interaction)	Leakage Boundary (spatial)	3.3.1.5, 3.3.2.26, 3.3.2.142
Pumps (spatial interaction)	Leakage Boundary (spatial)	3.3.2.181, 3.3.2.300
Sight Glasses (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.3.1.5, 3.3.2.197
Tanks (spatial interaction)	Leakage Boundary (spatial)	3.3.1.5, 3.3.2.214
Thermowells (spatial interaction) (Dresden only)	Leakage Boundary (spatial)	3.3.2.2, 3.3.2.40
Traps (spatial interaction)	Leakage Boundary (spatial)	3.3.2.31, 3.3.2.229, 3.3.2.300
Tubing (spatial interaction)	Leakage Boundary (spatial)	3.3.2.34, 3.3.2.40, 3.3.2.41, 3.3.2.243, 3.3.2.252

Table 2.3.3-24 Component Groups Requiring Aging Management Review – Plant Heating System (Continued)

Component Group	Component Intended Function	Aging Management Ref
Valves (spatial interaction)	Leakage Boundary (spatial)	3.3.1.5, 3.3.2.23, 3.3.2.24, 3.3.2.31, 3.3.2.263, 3.3.2.271, 3.3.2.282, 3.3.2.300

Aging management review results for the plant heating system are provided in Section 3.3.

2.3.3.25 Containment Atmosphere Monitoring System

System Purpose

The containment atmosphere monitoring (CAM) system provides the ability to monitor hydrogen, oxygen, and gross gamma radiation levels in the containment following a LOCA, and provides necessary indication and trip signals.

System Operation

During CAM system operation, containment atmosphere is withdrawn through piping connected to primary containment penetrations for obtaining both a drywell and suppression chamber air sample. Hydrogen and oxygen concentration are measured outside the primary containment (evaluated with the primary containment structure) and the sample returned to the primary containment. The sample withdrawal lines in both cases are heat traced to prevent condensation in the sample lines which would cause measurement inaccuracies. A check valve is installed in the return discharge line for primary containment. In addition, a check valve is installed in each reagent and calibration gas line for primary containment. The containment atmosphere monitoring system consists of oxygen and hydrogen analyzer process instrumentation and various indication and annunciation instruments, primary containment monitoring panels, and gross gamma detector channels (from detector to annunciator and computer points). The system is automatically activated upon the occurrence of a LOCA, or manually by an operator. The system initiates a primary containment group 2 isolation on high radiation.

System Evaluation Boundary

The containment atmosphere monitoring system evaluation boundary contains safety related drywell/suppression chamber air sample oxygen and hydrogen analyzer process instrumentation, including various indicators/annunciators, sample piping into and out of the drywell, drywell/suppression chamber sample path valves, suppression chamber sample path valves, primary containment monitoring panels, gross gamma radiation detectors and heat tracing. Also included are calibration gas valves, bottles, sample test points/valves and computer points associated with the system. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the containment atmosphere monitoring system evaluation boundary.

UFSAR References

Dresden Station UFSAR Section(s): 6.2.5.3.2, 7.3.2.2.7

Quad Cities Station UFSAR Section(s): 6.2.5.2

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE-M-706-1 and LR-DRE-M-706-2

Quad Cities Station:
LR-QDC-M-641-1 and LR-QDC-M-641-2.

System Intended Functions

Primary containment isolation – provides containment isolation for those portions of the system that interface with the primary containment.

Support ESF function(s) - provides signals to indicate and alarm high hydrogen, oxygen, and high gross gamma radiation levels in the containment following a LOCA, and provide provides a primary containment Group 2 isolation signal on high radiation conditions.

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Credited in regulated event(s) – The system contains components that are relied upon for compliance with 10 CFR 50.49, (EQ).

Component Groups Requiring Aging Management Review

Table 2.3.3-25 Component Groups Requiring Aging Management Review – Containment Atmosphere Monitoring System

Component	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.3.1.22
Filters/Strainers	Filter	3.3.2.230
Flexible Hoses	Pressure Boundary	3.3.2.64, 3.3.2.67
NSR Vents or Drains, Piping and Valves (attached support)	Structural Integrity (attached)	3.3.2.130
Piping and Fittings (attached support)	Structural Integrity (attached)	3.3.2.40, 3.3.2.166
Piping and Fittings	Pressure Boundary	3.3.2.40, 3.3.2.166
Pumps	Pressure Boundary	3.3.2.40, 3.3.2.299
Restricting Orifices	Pressure Boundary	3.3.2.40, 3.3.2.193
Sample Pumps	Pressure Boundary	3.3.2.40, 3.3.2.195
Tubing	Pressure Boundary	3.3.2.40, 3.3.2.42, 3.3.2.248, 3.3.2.254, 3.3.2.255
Valves	Pressure Boundary	3.3.2.23, 3.3.2.40, 3.3.2.42, 3.3.2.260, 3.3.2.289, 3.3.2.295
Valves (attached support)	Structural Integrity (attached)	3.3.2.40, 3.3.2.295

Aging management review results for the system are provided in Section 3.3.

2.3.3.26 Nitrogen Containment Atmosphere Dilution System

System Purpose

The nitrogen containment atmosphere dilution (NCAD) system provides two redundant, single failure proof, independent flow paths for purging the primary containment with nitrogen to provide post-accident combustible gas control. The NCAD system injects gaseous nitrogen into the primary containment to purge the containment of oxygen and hydrogen to maintain the mixture below combustible levels.

System Operation

The NCAD system is a manually operated system comprised of redundant flow paths. It is operated locally by opening a manual valve near the nitrogen supply equipment. The containment purge and vent valves can be aligned to inject nitrogen into the drywell or suppression chamber (evaluated with the primary containment structure) for either flow path. At Quad Cities, the NCAD system is made up of two independent, redundant flow paths for each unit. Each flow path in turn can supply gaseous nitrogen to either the drywell or suppression chamber. One flow path runs from the unit’s corresponding electric vaporizer and taps back into the nitrogen inerting system piping just upstream of nitrogen purge vaporization valve, on the non-safety related side. The other flow path runs from the opposite unit's electric vaporizer and taps back into the normal nitrogen makeup system just upstream of the nitrogen makeup valve. Either flowpath can be supplied by the nitrogen atmospheric vaporizer. At Dresden, there is a normal and emergency supply line. The normal NCAD line begins with the drywell nitrogen purge and inerting system (DNPIS) (evaluated with the drywell nitrogen inerting system) connection downstream of the pressure regulating station at the discharge side of the makeup line atmospheric vaporizer. Then, from the pressure regulating stations to the nitrogen supply header. The emergency NCAD line begins with the discharge of the nitrogen auxiliary tank and taps into the emergency truck connection upstream of the makeup line atmospheric vaporizer.

System Evaluation Boundary

The NCAD system evaluation boundary at Dresden includes the normal and emergency supply lines. The NCAD system evaluation boundary at Quad Cities includes the two independent, redundant flow paths. All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the NCAD system evaluation boundary.

UFSAR References

Dresden Station UFSAR Section(s): 6.2.5.3.3

Quad Cities Station UFSAR Section(s): 6.2.5.3

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-25, LR-DRE-M-356, and LR-DRE-M-4215

Quad Cities Station:

LR-QDC-M-34-2, LR-QDC-M-34-3, and LR-QDC-M-76-2

System Intended Functions

Supports ESF function(s) – provides capability to maintain a non-explosive atmosphere in the primary containment following a design basis accident. Backup to nitrogen inerting system for post LOCA operations. (Quad Cities only)

Component Groups Requiring Aging Management Review

Table 2.3.3-26 Component Groups Requiring Aging Management Review - Nitrogen Containment Atmosphere Dilution System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.3.1.22, 3.3.2.18
Restricting Orifices (Dresden only)	Pressure Boundary	3.3.2.35, 3.3.2.189
Restricting Orifices (Dresden only)	Throttle	3.3.2.189
Tubing	Pressure Boundary	3.3.2.34, 3.3.2.35, 3.3.2.239
Valves	Pressure Boundary	3.3.2.23, 3.3.2.25, 3.3.2.260

Aging management review results for the nitrogen containment atmosphere dilution system are provided in Section 3.3.

2.3.3.27 Drywell Nitrogen Inerting System

System Purpose

The drywell nitrogen inerting (DNI) system, also known as the drywell nitrogen purge and inerting system, is provided to maintain the drywell in a nitrogen inerted condition as a means of inhibiting the formation of a combustible gas mixture under LOCA conditions. The system is not safety related; however, it can be used for post-LOCA hydrogen control. The system also serves as a backup to the pump-back system to maintain the required drywell-to-suppression chamber differential pressure and provide nitrogen to the NCAD system.

System Operation

The drywell nitrogen inerting system consists of a liquid nitrogen storage tank, nitrogen vaporizers, associated piping, isolation valves, and pressure regulators. Nitrogen is supplied to three possible types of vaporizers. Steam powered vaporizers, which use plant heating steam (evaluated with plant heating system) to ensure supply temperatures do not damage nitrogen piping during periods of large demand, exist at both Dresden and Quad Cities, although Quad Cities typically uses electrically powered vaporizers installed for the same purpose. Additionally, each site has atmospheric vaporizers for periods of low demand. Flow regulating valves are also installed to limit low nitrogen supply temperatures. Nitrogen to the drywell is supplied through the drywell purge inlet line while air is vented to the reactor building ventilation system (evaluated with HVAC – reactor building) or the standby gas treatment system (evaluated with standby gas treatment system). A similar method is used for inerting the suppression chamber. The containment is deinerted by admitting air into the containment as the containment atmosphere is vented to the reactor building ventilation system or the standby gas treatment system.

System Evaluation Boundary

The Quad Cities drywell nitrogen inerting system evaluation boundary begins with the bulk nitrogen storage tank that feeds to the common nitrogen supply header, which is connected to a nitrogen supply line from the diesel generator room. The nitrogen supply header connects to a common nitrogen purge vaporizer and also feeds the nitrogen purge vaporizer in the diesel generator room. The nitrogen from the nitrogen purge vaporizer is split to feed both units. The evaluation boundary ends with its connection to the instrument air system downstream of the drywell pneumatic check valves for Units 1 and 2 (evaluated with instrument air system), at the compressor inlet moisture separator isolation valve of the pumpback air system (evaluated with drywell/suppression chamber dp pumpback air system), at the nitrogen makeup cross-tie valve (evaluated with nitrogen containment atmosphere dilution), at the nitrogen supply from the vaporizer to the Unit 1 and Unit 2 drywell and suppression chamber valves (evaluated with nitrogen containment atmosphere dilution) and at the Unit 1 and Unit 2 connection to the high radiation sampling system valves for containment air sampling (evaluated with high radiation sampling system).

The Dresden drywell nitrogen inerting system evaluation boundary begins with the nitrogen primary tank and nitrogen auxiliary tank to the common nitrogen supply header to the nitrogen purge vaporizer and then routed to both units. The evaluation boundary ends at the compressor upstream suction valve of the pumpback air system (evaluated with drywell/suppression chamber dp pumpback air system), at the NCAD makeup/inerting bypass pressure relief valve inlet isolation valve (evaluated with NCAD), at the nitrogen auxiliary tank emergency supply to vaporizer check valve (evaluated with NCAD), at the nitrogen bypass supply to nitrogen inerting isolation valve (evaluated with NCAD), at the nitrogen bypass supply to nitrogen normal makeup isolation valve (evaluated with NCAD), at the drywell/suppression chamber nitrogen makeup valve (evaluated with primary containment and suppression chamber piping) and at the drywell/suppression chamber nitrogen purge and pumpback suction valve (evaluated with primary containment and suppression chamber piping). All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the drywell nitrogen inerting system.

UFSAR References

Dresden Station UFSAR Section(s): 6.2.5

Quad Cities Station UFSAR Section(s): 6.2.5

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-25, LR-DRE-M-356, and LR-DRE-M-4215.

Quad Cities Station:

LR-QDC-M-24-12, LR-QDC-M-34-2, LR-QDC-M-34-3, LR-QDC-M-71-7,

LR-QDC-M-76-2, LR-QDC-M-1056-4, and LR-QDC-M-1061-4.

System Intended Functions

Supports ESF function(s) – reduces and maintains a low concentration of oxygen in the primary containment. It can also be used, if available, for post-LOCA hydrogen concentration control.

Credited in regulated event(s) – credited in mitigation of the Appendix R fire event by establishing the inert drywell environment in which a design basis fire cannot occur.

Component Groups Requiring Aging Management Review

Table 2.3.3-27 Component Groups Requiring Aging Management Review – Drywell Nitrogen Inerting System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.3.1.22
Filters/Strainers (Dresden only)	Pressure Boundary	3.3.2.25, 3.3.2.51
Filters/Strainers (Dresden only)	Filter	3.3.2.51
Flow Elements	Pressure Boundary	3.3.1.5, 3.3.2.40, 3.3.2.69, 3.3.2.71
Isolation Barriers	Pressure Boundary	3.3.2.124, 3.3.2.125
Piping and Fittings	Pressure Boundary	3.3.1.5, 3.3.2.27, 3.3.2.138, 3.3.2.161
Tanks (includes vaporizers)	Pressure Boundary	3.3.1.5, 3.3.2.21, 3.3.2.40, 3.3.2.210, 3.3.2.212
Thermowells	Pressure Boundary	3.3.1.5, 3.3.2.223
Traps (Quad Cities only)	Pressure Boundary	3.3.1.5, 3.3.2.228
Tubing	Pressure Boundary	3.3.2.22, 3.3.2.34, 3.3.2.35, 3.3.2.40, 3.3.2.43, 3.3.2.231, 3.3.2.239, 3.3.2.248
Valves	Pressure Boundary	3.3.1.5, 3.3.2.23, 3.3.2.25, 3.3.2.40, 3.3.2.260, 3.3.2.262, 3.3.2.268, 3.3.2.273, 3.3.2.289, 3.3.2.295

Aging management review results for the drywell nitrogen inerting system are provided in Section 3.3.

2.3.3.28 Safe Shutdown Makeup Pump System (Quad Cities Only)

System Purpose

The safe shutdown makeup pump (SSMP) system at Quad Cities is a common system that provides cooling water to the Unit 1 or Unit 2 reactor core in the event that the reactor becomes isolated from the main condenser simultaneously with a loss of the feedwater system.

System Operation

The SSMP system consists of a motor-driven pump, associated valves, piping and instrumentation. The preferred water source to the pump is the contaminated condensate storage tank (evaluated with the condensate and condensate storage system). An alternate source of makeup water is available from the fire header (evaluated with the fire protection system). The SSMP discharge is delivered to the reactor vessel (evaluated with the reactor vessel) via the HPCI system pump discharge line (evaluated with the HPCI system).

System Evaluation Boundary

The SSMP system evaluation boundary begins with the SSMP suction line from the common RCIC suction line (evaluated with RCIC system) from the condensate storage tank, the suction line from the fire header (evaluated with the fire protection system), through the SSMP pump. The discharge path runs from the output side of the pump through one of two valves to either the Unit 1 “B” feedwater line (evaluated with the feedwater system), upstream of the outboard primary containment isolation valves, or to the Unit 2 HPCI pump discharge line (evaluated with the HPCI system). The system evaluation boundary also includes the safe shutdown pump room cooler and its associated piping from the service water system (evaluated with service water system) and the alternate supply from the fire header (evaluated with fire protection). All associated piping, components and instrumentation contained within the flow paths and systems described above are included in the SSMP system evaluation boundary.

UFSAR References

Dresden Station UFSAR Section(s): Not Applicable

Quad Cities Station UFSAR Section(s): 5.4.6.5

License Renewal Boundary Diagram References

Dresden Station: Not Applicable

Quad Cities Station:

LR-QDC-M-70

System Intended Functions

Pressure boundary - maintains pressure boundary integrity at interface with HPCI system piping to support injection of cooling water to the RPV.

Credited for regulated event(s) - provides cooling water injection into the reactor pressure vessel credited in mitigation of the Appendix R fire event.

Component Groups Requiring Aging Management Review

Table 2.3.3-28 Component Groups Requiring Aging Management Review – Safe Shutdown Makeup Pump System (Quad Cities only)

Component Group	Component Intended Function	Aging Management Ref
Air Handlers Heating/Cooling (Aux&RW HVAC) (Quad Cities only)	Pressure Boundary	3.3.2.8, 3.3.2.9, 3.3.2.21
Air Handlers Heating/Cooling (Aux&RW HVAC) (Quad Cities only)	Heat Transfer	3.3.2.7
Closure Bolting (Quad Cities only)	Pressure Boundary	3.3.1.22
Ducts & Fittings, Access Doors, Closure Bolts, Equip Frames (Quad Cities only)	Pressure Boundary	3.3.1.5
Filters/Strainers (Quad Cities only)	Pressure Boundary	3.3.1.5, 3.3.1.19
Filters/Strainers (Quad Cities only)	Filter	3.3.1.19
Piping and Fittings (Quad Cities only) (includes spectacle flanges)	Pressure Boundary	3.3.1.5, 3.3.1.8, 3.3.1.19, 3.3.2.40
Pumps (Quad Cities only)	Pressure Boundary	3.3.1.5, 3.3.1.8
Restricting Orifices (Quad Cities only)	Pressure Boundary	3.3.1.8, 3.3.1.25, 3.3.2.40
Restricting Orifices (Quad Cities only)	Throttle	3.3.1.8, 3.3.1.25
Valves (Quad Cities only)	Pressure Boundary	3.3.1.5, 3.3.1.8, 3.3.1.19

Aging management review results are presented in Section 3.3 for the additional functions of the safe shutdown makeup pump system

2.3.4 Steam and Power Conversion Systems

The section of the application addresses scoping and screening results for the following systems:

- Main steam system
- Feedwater system
- Condensate and condensate storage systems
- Main condenser
- Main turbine and auxiliary systems
- Turbine oil (In-scope for Quad Cities only)
- Main generator and auxiliaries (In-scope for Quad Cities only)

2.3.4.1 Main Steam System

System Purpose

The main steam system delivers steam from the reactor pressure vessel to the main turbine and balance of plant auxiliary steam loads.

System Operation

The main steam system transports steam generated by the reactor through the reactor pressure vessel nozzles to the main turbine via main steam stop valves in each of four lines. At Quad Cities only, the main steam system supplies steam to the HPCI and RCIC systems. Steam can be lined up to bypass the main turbine via bypass valves to the main condenser when required (e.g. during plant startup). Main steam isolation valves (MSIVs) and venturi type flow restrictors are installed to minimize reactor coolant inventory loss in the event of a main steam line break. The MSIVs also limit radiation release rates to prevent exceeding the 10 CFR 100 guidelines in the event of main steam line break outside the primary containment. Relief valves and safety valves located on the main steam piping inside primary containment are provided for reactor pressure vessel over pressure protection. The relief valves will operate automatically if steam pressure exceeds the relief valve setpoint or upon receipt of a signal from the automatic depressurization system. They can also be operated manually to reduce reactor pressure vessel pressure. The main steam system also provides steam to the main turbine gland seals, steam jet air ejectors, off-gas pre-heaters and booster air ejectors, as well as the radwaste maximum recycle re-boiler.

System Evaluation Boundary

The main steam system evaluation boundary starts at the four steam lines at the reactor pressure vessel nozzles and runs to and includes the turbine main steam stop valves and the turbine bypass valves via an equalization header. Each main steam line is equipped with safety valves, at least one relief valve, a venturi type flow restrictor followed by an MSIV inside and outside the primary containment. At Quad Cities only, a connection is provided for supplying steam to the HPCI and RCIC turbines, which are evaluated with the HPCI and RCIC systems. Downstream of the inboard and outboard MSIVs are drain lines to the main condenser. The boundary includes piping and components from gland seal for the main turbine, steam jet air ejectors, off-gas pre-heater and booster air ejectors, and radwaste maximum recycle re-boiler. The evaluation boundary includes piping between the reactor pressure vessel and outboard isolation valve, including the main steam line drain piping, even though they are considered part of the reactor coolant pressure boundary. All associated piping, components, and instrumentation contained within the flow paths and systems described above are included in the evaluation boundary. Solenoids, accumulators, pressure controllers, and position switches associated with the automatic depressurization system and manual isolation valves for instruments associated with the feedwater level control system are evaluated as part of the main steam system.

UFSAR References

Dresden Station UFSAR Section(s): 10.3

Quad Cities Station UFSAR Section(s): 10.3

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE--M-12-1, LR-DRE-M-12-2, LR-DRE-M-25, LR-DRE-M-37-2, LR-DRE-M-345-1, LR-DRE-M-345-2, LR-DRE-M-356, and LR-DRE-M-367-2.

Quad Cities Station:
LR-QDC-CID-13-2, LR-QDC-CID-60-2, LR-QDC-M-13-1, LR-QDC-M-13-2, LR-QDC-M-34-1, LR-QDC-M-60-1, LR-QDC-M-60-2, and LR-QDC-M-76-1.

System Intended Functions

Pressure boundary – maintains the integrity of the reactor coolant pressure boundary and provides steam-line isolation to support the reactor coolant pressure boundary.

Core cooling – in conjunction with the Automatic Depressurization System, supports emergency core cooling by depressurizing the reactor pressure vessel as required to support low pressure coolant injection and core spray operation.

Overpressure protection – provides overpressure protection in transient or accident events that increase pressure in the reactor pressure vessel.

Primary containment isolation – provides containment isolation for those portions of the system that interface with the primary containment.

Supports ESF function(s) – provides process signals for initiation of ESF functions, limits coolant inventory loss rate in some LOCA events, and (at Quad Cities only) provides steam supply for operation of the HPCI and RCIC systems.

Credited in regulated event(s) – provides overpressure protection, reactor vessel isolation capability and pressure control capability credited in mitigation of the Appendix R fire, ATWS, and SBO events. The system also contains components that are relied upon for compliance with 10 CFR 50.49, (EQ).

Post accident plateout of MSIV seat leakage – provides surfaces for plateout of iodine releases resulting from MSIV bypass leakage.

Limit steam line flow - This function limits potential radioactive release by restricting steam flow during a steam line rupture outside of primary containment. Flow is also limited to ensure integrity of dryers in order to prevent restriction of MSIV closure.

Steam flow measurement – The main steam system provides main steam flow input for primary containment isolation.

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.4-1 Component Groups Requiring Aging Management Review - Main Steam

Component Group	Component Intended Function	Aging Management Ref
Accumulators	Pressure Boundary	3.4.2.1
Closure Bolting	Pressure Boundary	3.1.1.1, 3.4.1.6
Dampeners (Quad Cities only)	Pressure Boundary	3.1.1.15, 3.4.2.11, 3.4.2.13
Filters/Strainers (Quad Cities only)	Filter	3.4.2.13, 3.4.2.17
Flexible Hoses	Pressure Boundary	3.4.2.18, 3.4.2.19
Flow Elements	Pressure Boundary	3.1.1.11, 3.4.2.6
Flow Elements	Throttle	3.1.1.15
NSR Vents or Drains, Piping and Valves (spatial interaction)	Leakage Boundary (spatial)	3.4.2.30
NSR Vents or Drains, Piping and Valves (attached support)	Structural Integrity (attached)	3.4.2.30
Piping and Fittings	Pressure Boundary	3.1.1.1, 3.1.1.11, 3.1.1.15, 3.2.1.3, 3.2.1.5, 3.4.1.1, 3.4.1.3, 3.4.1.4, 3.4.1.5, 3.4.2.5, 3.4.2.6, 3.4.2.11, 3.4.2.12, 3.4.2.13, 3.4.2.34
Piping and Fittings (small bore)	Pressure Boundary	3.1.1.5, 3.4.1.3, 3.4.2.5, 3.4.2.6, 3.4.2.11, 3.4.2.13
Restricting Orifices	Pressure Boundary	3.4.1.3, 3.4.1.4, 3.4.1.5
Rupture Discs	Pressure Boundary	3.2.1.3, 3.2.1.5, 3.4.2.13
Tanks (Quad Cities only)	Pressure Boundary	3.1.1.15, 3.4.2.11, 3.4.2.13
Thermowells	Pressure Boundary	3.1.1.15, 3.2.1.3, 3.2.1.5, 3.4.2.6, 3.4.2.12
Tubing (attached support)	Structural Integrity (attached)	3.1.2.42, 3.4.2.13
Vacuum Breakers	Pressure Boundary	3.2.1.3, 3.2.1.5, 3.4.2.13
Valves	Pressure Boundary	3.1.1.1, 3.1.1.11, 3.1.1.15, 3.4.1.2, 3.4.1.3, 3.4.1.4, 3.4.1.5, 3.4.2.5, 3.4.2.6, 3.4.2.11, 3.4.2.12, 3.4.2.13, 3.4.2.51, 3.4.2.53
Valves (attached support) (Dresden only)	Structural Integrity (attached)	3.4.2.13, 3.4.2.53

Aging management review results for the main steam system are provided in Section 3.1 for the reactor coolant pressure boundary functions and Sections 3.2 and 3.4 for the additional main steam system functions.

2.3.4.2 Feedwater System

System Purpose

The feedwater system (in conjunction with the condensate system) delivers condensate from the condenser to the reactor at a rate of water equivalent to what is being generated into steam by boil-off and removed by the main steam system.

System Operation

The feedwater system consists of reactor feed pumps (RFPs), feedwater regulating valves (FWRVs), high pressure feedwater heaters, piping, isolation valves, controls and instrumentation, and subsystems that supply the reactor with regenerative feedwater heating in a closed steam cycle. Three horizontal motor driven feed pumps are provided. All three RFPs need to be in service during normal full load operation. The RFPs take suction on a common header downstream of the low-pressure feedwater heaters (evaluated with the condensate and condensate storage system). The normal flowpath is through the RFP discharge check valve, combining into a common header upstream of the FWRVs, through the FWRVs, combining into a common header downstream of the FWRVs. Flow then goes through a common header upstream of the high-pressure (HP) feedwater heaters, through the HP heaters and associated inlet and outlet isolation MOVs to a common header. A HP heater bypass line is sized to carry 33% of rated flow. Flow then goes to the A and B feedwater lines and associated isolation MOV at the outlet of the HP heaters. Flow then passes through the A and B feedwater headers in parallel through two outboard isolation check valves, one inboard isolation check valve, and an inboard manual isolation valve in each line. Feedwater flow is finally directed into the reactor vessel (evaluated with the reactor vessel). Two FWRVs, whose positions are determined by the feedwater level control system are provided for normal power operation and are normally set to automatically maintain reactor water level. One low-flow regulating valve is used for lower power operation.

Other systems using the feedwater lines to provide a flow path to the reactor vessel are the high pressure coolant injection (HPCI) (evaluated as an ESF system) and reactor water cleanup systems (RWCU) (evaluated as an auxiliary system). At Dresden, HPCI and RWCU tap into the B feedwater line to inject fluid during emergency operations (HPCI) or as a return path for water removed from the vessel (RWCU). At Quad Cities, the reactor core injection cooling (RCIC) system (evaluated as an ESF system) and the RWCU system tap into the “A” feedwater line to either inject fluid during emergency operations (RCIC) or as a return path for water removed from the vessel (RWCU). Also at Quad Cities, HPCI and the safe shutdown makeup pump system (evaluated as an auxiliary system) tap into the B feedwater line to either inject fluid during emergency operations (HPCI) or as an injection path for the discharge of the safe shutdown makeup pump.

System Evaluation Boundary

The feedwater system starts with the reactor feed pumps and ends with the reactor feedwater connection at the reactor pressure vessel nozzles (evaluated with the reactor vessel). The system includes the reactor feed pumps, the pump discharge check and motor operated valves, the feedwater regulating valves, the tube side of the high-pressure (HP) feedwater heaters and its inlet and outlet motor operated isolation valves, the HP feedwater heater bypass valve, the motor operated inlet valve for the A and B feedwater lines, and the inboard and outboard A and B feedwater line check valves, and associated piping, and instrumentation. The reactor feed pump discharge flow elements, the feedwater regulating valves and the low flow feedwater regulating valve, considered part of the feedwater control system, are all evaluated as part of the feedwater system.

UFSAR References

Dresden Station UFSAR Section(s): 10.4.7

Quad Cities Station UFSAR Section(s): 10.4.7

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-14, LR-DRE-M-19, LR-DRE-M-174-3, LR-DRE-M-177-2, LR-DRE-M-347, LR-DRE-M-352, LR-DRE-M-419-2, and LR-DRE-M-420.

Quad Cities Station:

LR-QDC-M-15-1, LR-QDC-M-15-2, LR-QDC-M-62-1, and LR-QDC-M-62-2.

System Intended Functions

Flowpath - provide flowpath into the reactor pressure vessel for high pressure coolant injection, reactor water cleanup and for Quad Cities only, reactor core isolation cooling and safe shutdown makeup pump flow.

Pressure boundary – maintain the integrity of the reactor coolant pressure boundary.

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.4-2 Component Groups Requiring Aging Management Review - Feedwater System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.1.1.1, 3.4.1.6
NSR Vents or Drains, Piping and Valves (attached support)	Structural Integrity (attached)	3.4.2.30
Piping and Fittings	Pressure Boundary	3.1.1.1, 3.1.1.11, 3.4.1.1, 3.4.1.2, 3.4.1.3, 3.4.1.4, 3.4.2.5, 3.4.2.6
Piping and Fittings (spatial interaction)	Leakage Boundary (spatial)	3.4.1.2, 3.4.1.3, 3.4.1.4, 3.4.2.5, 3.4.2.6
Piping and Fittings (attached support)	Structural Integrity (attached)	3.4.1.2, 3.4.1.3, 3.4.1.4, 3.4.2.5, 3.4.2.6
Piping and Fittings (small bore) (Quad Cities only)	Pressure Boundary	3.1.1.5, 3.4.1.3, 3.4.2.5, 3.4.2.6
Valves	Pressure Boundary	3.1.1.1, 3.1.1.11, 3.1.1.15, 3.4.1.3, 3.4.2.5, 3.4.2.6, 3.4.2.11, 3.4.2.12, 3.4.2.13
Valves (attached support)	Structural Integrity (attached)	3.4.1.2, 3.4.1.3, 3.4.1.4, 3.4.2.5, 3.4.2.13, 3.4.2.54

Aging management review results for the feedwater system are provided in Section 3.1 for the reactor coolant pressure boundary functions and Section 3.4 for the additional feedwater system functions.

2.3.4.3 Condensate and Condensate Storage Systems

System Purpose

The condensate and condensate storage systems (in conjunction with the feedwater system) works to provide water of a quality and quantity required for operation of the power plant. The condensate and condensate booster pump portion of the system supply reactor quality water to the suction of the reactor feedwater pumps. The condensate storage system's contaminated condensate storage tanks (CCSTs) ensure reactor quality water is available for makeup requirements, and are designed to ensure a minimum of 90,000 gallons of water is available from each CCST for use by HPCI. The condensate and feedwater systems' pumping functions are not credited to support safe shutdown or to perform any reactor safety function. The CCSTs are also credited for providing makeup to the reactor via the CRD pumps (at Dresden) or the reactor core isolation cooling (RCIC) and safe shutdown makeup pump (SSMP) systems (at Quad Cities) for safe shutdown scenarios in the Fire Protection Plan.

System Operation

The condensate flowpath includes pumps, valves and piping required to transfer the condensate from the condenser hotwell (evaluated with the main condenser) to the suction of the reactor feed pump (RFP) (evaluated with the feedwater system). Four condensate pumps take suction from both sides of the condenser hotwell through a common header. Condensate then flows through the following components, before reaching the RPFs: steam jet air ejector (SJAE) condensers, off-gas condensers, condensate demineralizers, condensate booster pump and the low pressure heaters. The condensate pumps discharge through check valves to the tube side of two SJAE condensers or a 50% capacity bypass line. The bypass line provides a flowpath in the event a SJAE condenser requires maintenance and is orificed to simulate the pressure drop through one condenser. From the SJAE condenser, condensate flows through two gland seal steam condensers (tube side) to the tube side of two off-gas condensers. The condensate then passes through the condensate demineralizers. The demineralizers discharge into a common condensate booster pump suction header from which a condensate reject line to the CCSTs is provided with a normal and emergency reject valve, controlled by hotwell level instrumentation. The condensate booster pump is driven by the same motor that drives its associated condensate pump. A penetration downstream of the booster pump provides recirculation to the main condenser to provide minimum flow required for the SJAE condenser to operate efficiently, provide minimum flow required for the condensate and condensate booster pumps, to cleanup the main condenser water prior to startup and to maintain adequate differential pressure across the demineralizers. Water flows through the tube sides of three parallel strings of low pressure feedwater heaters. The low-pressure feedwater heater bypass line can carry 33% flow during heater string maintenance and is orificed to approximate the pressure drop through one heater string.

System Evaluation Boundary

The condensate and condensate storage systems were evaluated from the outlet of the condenser hotwell through the condensate pumps and condensate booster pumps up to the inlet of the reactor feedwater pump suction piping. System components included in the evaluation were the condensate pumps, the condensate pump discharge check valves, tube side of the SJAE condensers, tube side of the gland seal condensers and the tube side of the off-gas condensers. Other system components included in the evaluation were the condensate demineralizer motor operated inlet and outlet valves, condensate demineralizer bypass valve, condensate demineralizer post strainers and the tube side of the low pressure heaters. Also included were the manual isolation valves for the all of the aforementioned condensers, the motor operated inlet and outlet valves for all of the low pressure heater strings, the low pressure heater motor operated bypass valve and the associated system component piping and instrumentation. The common system boundary evaluation also includes the CCSTs, standpipes, associated instrumentation, condensate transfer pumps, associated distribution piping and valves. For Dresden only, valves in the main condenser system providing pressure boundary to control rod drive pump suction supply are evaluated with the condensate and condensate storage system, and an isolation valve from the condensate and condensate storage back up supply line is evaluated with the core spray system. The condensate demineralizers were not included in this evaluation; they were evaluated separately in the Auxiliary Systems.

UFSAR References

Dresden Station UFSAR Section(s): 9.2.6 and 10.4.7

Quad Cities Station UFSAR Section(s): 9.2.6 and 10.4.7

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE-M-15, LR-DRE-M-27, LR-DRE-M-29-1, LR-DRE-M-35-1, LR-DRE-M-348, LR-DRE-M-358, LR-DRE-M-360-1, and LR-DRE-M-366.

Quad Cities Station:
LR-QDC-M-16-1, LR-QDC-M-16-2, LR-QDC-M-16-4, LR-QDC-M-16-5, LR-QDC-M-17, LR-QDC-M-36, LR-QDC-M-39-2, LR-QDC-M-63-1, LR-QDC-M-63-2, LR-QDC-M-63-4, LR-QDC-M-64, and LR-QDC-M-81-2.

System Intended Functions

Support ESF function(s) – provide reactor grade water to HPCI, RCIC (at Quad Cities), core spray, LPCI (at Dresden), and RHR (at Quad Cities).

Credited in regulated events – provide water to support mitigating actions for Appendix R fire, SBO, and ATWS.

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.4-3 Component Groups Requiring Aging Management Review - Condensate and Condensate Storage System

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.4.1.6, 3.4.2.2
Piping and Fittings	Pressure Boundary	3.4.1.2, 3.4.1.3, 3.4.1.4, 3.4.2.3, 3.4.2.4, 3.4.2.7, 3.4.2.8, 3.4.2.11, 3.4.2.14, 3.4.2.15, 3.4.2.31, 3.4.2.35
Piping and Fittings (spatial interaction) (Dresden only)	Leakage Boundary (spatial)	3.4.1.2, 3.4.1.3, 3.4.1.4
Piping and Fittings (attached support) (Quad Cities only)	Structural Integrity (attached)	3.4.1.2, 3.4.1.3, 3.4.1.4
Tanks	Pressure Boundary	3.4.2.39, 3.4.2.40, 3.4.2.41, 3.4.2.42
Thermowells (Dresden only)	Pressure Boundary	3.4.1.2, 3.4.1.3, 3.4.1.4, 3.4.2.4, 3.4.2.45
Tubing	Pressure Boundary	3.4.2.3, 3.4.2.4, 3.4.2.11, 3.4.2.14, 3.4.2.46, 3.4.2.47
Valves	Pressure Boundary	3.4.1.3, 3.4.2.3, 3.4.2.4, 3.4.2.7, 3.4.2.11, 3.4.2.14, 3.4.1.2, 3.4.1.4, 3.4.2.49, 3.4.2.54
Valves (attached support) (Quad Cities only)	Structural Integrity (attached)	3.4.1.2, 3.4.1.3, 3.4.1.4

Aging management review results for the condensate and condensate storage system are provided in Section 3.4.

2.3.4.4 Main Condenser

System Purpose

The main condenser provides a heat sink for the turbine exhaust steam, turbine bypass steam, and other flows. It also deaerates and stores the condensate for reuse after a period of radioactive decay. Additionally, the main condenser provides for post accident containment holdup and plateout of main steam isolation valve (MSIV) bypass leakage. The purpose of the main condenser includes the following functions:

- to provide a heat sink for the turbine exhaust steam.
- to condense the bypass steam after a turbine trip.
- to accommodate feedwater heater drains, extraction steam, and steam line condensate routed to the condenser during operation with feedwater heaters out of service.
- to retain the condensate for a brief time to allow for the decay of short-lived isotopes.
- to deaerate the condensate and remove fission product gases, hydrogen, and oxygen.
- to provide adequate net positive suction head for condensate pumps.
- to provide for iodine plateout and radioactive decay prior to release.

The main condenser provides a surface for iodine plateout and the hold-up volume for radioactive decay prior to release that is credited in analysis of the Control Rod Drop Accident and in evaluation of the MSIV bypass leakage following a design basis LOCA.

System Operation

The main condenser is a divided water flow, single-pass, multi-pressure, deaerating type with capacity for reverse flow for each half of the condenser. The condenser is designed to accept bypass steam up to approximately 30% of throttle steam flow. During plant operation, steam, after expanding through the low-pressure turbine, exhausts through the bottom of the turbine casing to the condenser. The divided water flow permits circulating water to be reversed periodically through each bank of tubes in each half of the condenser for cleaning purposes. The condenser shell is supported on the turbine foundation mat. An expansion joint is fitted between each low-pressure turbine exhaust hood and condenser inlet connection. The condenser is divided into three separate compartments by two division plates. Cold circulating water enters the cold compartment, which has 100% condensing capacity. The intermediate compartment has 99% condensing capacity because of the warmer temperature of the circulating water. The warm compartment has 97-98% condensing capacity. The excess steam is called reheat steam and is used for deaerating purposes. The reheat steam heats the condensate streams at the weir plate to a boiling temperature that liberates the dissolved noncondensable gases. Vent pipes passing through the lower de-aerating weir plate and collecting tray vent the noncondensable gases to the tube bundle in the intermediate compartment. The air in-leakage and noncondensable gases are transported via vent pipes to the air cooler trays, which extend the entire condenser length. These gases are removed by the main condenser evacuation system. The condenser hotwells retain the condensate for a brief time to allow short-lived radioactive isotopes to decay to levels that eliminate the need for shielding of the condensate pumps. This condensate retention is accomplished by a series of baffles and tunnel arrangements at the condensate outlet. The condensate is pumped from outlet pipes by the condensate pumps.

System Evaluation Boundary

The evaluation boundary for the main condenser is the condenser shell and condenser tubes, from the low-pressure turbine exhaust inlets and main steam drain line inlets to the condenser hotwell. Included within this boundary are the instrumentation and controls to monitor condenser status and performance. The hotwell sample pumps and components from the main condenser to the sample panel is included in the boundary. The boundary does not include any components of the condensate and condensate storage system, condensate demineralizer subsystem, steam jet air ejector subsystem, turbine bypass subsystem or the circulating water system

UFSAR References

Dresden Station UFSAR Section(s): 10.4 and 15.6.5.5.2

Quad Cities Station UFSAR Section(s): 10.4 and 15.6.5.5.3

License Renewal Boundary Diagram References

Dresden Station:
LR-DRE-M-15, LR-DRE-M-43-1, LR-DRE-M-177-1, LR-DRE-M-348, LR-DRE-M-371-1, and LR-DRE-M-419-1.

Quad Cities Station:
LR-QDC-M-16-1, LR-QDC-M-16-2, LR-QDC-M-16-4, LR-QDC-M-17, LR-QDC-M-63-1, LR-QDC-M-63-2, LR-QDC-M-63-4, LR-QDC-M-64, LR-QDC-M-459-3, and LR-QDC-M-462-3.

System Intended Functions

Post accident containment, holdup and plateout of MSIV bypass leakage - The main condenser provides for post accident containment, holdup and plateout of MSIV bypass leakage.

Component Groups Requiring Aging Management Review

Table 2.3.4-4 Component Groups Requiring Aging Management Review - Main Condenser

Component Group	Component Intended Function	Aging Management Ref
Main Condenser Hotwells, False Floors (includes hatches)	Containment Holdup and Plateout	3.4.2.24, 3.4.2.27
Main Condenser Tubes, Tubesheets (includes hatches)	Containment Holdup and Plateout	3.4.2.25, 3.4.2.26
Main Condenser Waterboxes, Hatches	Containment Holdup and Plateout	3.4.2.27, 3.4.2.28

Aging management review results for the main condenser are provided in Section 3.4.

2.3.4.5 Main Turbine and Auxiliary Systems

System Purpose

The main turbine converts the thermodynamic energy of reactor steam into rotational mechanical energy to drive the main generator. The main turbine consists of one high-pressure (HP) section, and three low pressure (LP) sections. The main turbine is supported by auxiliary systems. The gland sealing system provides gland-sealing steam to the HP and LP turbine glands to prevent steam from entering the turbine building and non-condensables from entering the condenser. The exhaust hood spray system provides cooling water to the condenser exhaust hood at low load, when steam flow through the last few turbine stages is low and insufficient cooling is provided. The turbine electrohydraulic control (EHC) system provides high-pressure fluid and logic to control the turbine main stop valves (MSVs), turbine control valves (TCVs), combined intermediate valves, bypass valves, and the reactor pressure through pressure regulators.

System Operation

Steam is delivered from the reactor pressure vessel through four main steam lines and the turbine throttle to the main stop valves (MSVs) (evaluated with the main steam system), or to other loads on the main steam system (max-recycle reboiler, seal steam regulator, steam jet air ejector regulator, bypass valves, off-gas preheater, and off-gas booster jet/dilution steam). From the MSVs, steam is passed through the steam chest (the area from below the seats of the MSVs to the TCV seats), through the TCVs, and then through the HP turbine section. The steam is then routed to four moisture separators, where steam drying occurs. The dry steam is admitted through six combined intercept valves to the LP turbine sections and exhausted to the main condenser (evaluated with the main condenser system). Extraction steam for the feedwater heaters is drawn from various LP turbine stages. The four TCVs are welded to the MSVs and are completely throttleable as determined by the EHC system. The TCVs regulate the steam to the turbine within the capability of the reactor to supply steam and thereby control reactor pressure. Additionally, the TCVs provide control for rolling, synchronizing and loading the main turbine and generator.

System Evaluation Boundary

The main turbine and auxiliary systems starts with the steam chest (after the main stop valves) and ends at the condenser. Main turbine includes the steam chest, turbine control valves, turbine bypass valves, main turbine, moisture separator tanks, combined intermediate valves, LP turbines, and associated piping, valves, instrumentation, and controls. Auxiliary systems include EHC, off-gas booster air ejectors, turbine gland sealing, exhaust hood spray, gland seal exhaust, steam supply to steam jet air ejectors and associated piping, valves, instrumentation, and controls.

UFSAR References

Dresden Station UFSAR Section(s): 7.7.4, 10.2

Quad Cities Station UFSAR Section(s): 7.7.4, 10.2

License Renewal Boundary Diagram References

Dresden Station:

LR-DRE-M-5650-2 and LR-DRE-M-5650-5.

Quad Cities Station:

LR-QDC-M-2022-2 and LR-QDC-M-2022-4.

System Intended functions

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.4-5 Component Groups Requiring Aging Management Review - Main Turbine and Auxiliary Systems

Component Group	Component Intended Function	Aging Management Ref
Accumulators (spatial interaction)	Leakage Boundary (spatial)	3.4.2.11, 3.4.2.29
Closure Bolting	Pressure Boundary	3.4.1.6
Piping and Fittings (spatial interaction)	Leakage Boundary (spatial)	3.4.2.11, 3.4.2.36
Tubing (spatial interaction) (Includes flex hoses)	Leakage Boundary (spatial)	3.4.2.11, 3.4.2.48
Valves (spatial interaction) (Includes flex hoses)	Leakage Boundary (spatial)	3.4.2.11, 3.4.2.55

Aging management review results for the main turbine and auxiliary systems are provided in Section 3.4.

2.3.4.6 Turbine Oil System (In-Scope for Quad Cities Only)

System Purpose

The turbine oil system at Quad Cities supplies lubricating oil to the main turbine and generator journal bearings, the thrust bearing and thrust bearing wear detector, the mechanical overspeed trip device (for testing), the overspeed trip reset, and the turning gear. The lubricating oil storage and transfer system is used to store, transfer, and purify lubricating oil for use within the main turbine oil system, hydrogen seal oil system, reactor feed pumps and recirculation MG set lubricating oil systems.

System Operation

The turbine oil system supplies all the necessary lubricating oil to the main turbine and its support systems to allow the turbine to be operated properly. The system is required to be in service during startups, normal operations, shutdowns, and at any time the turbine is on the turning gear. And depending upon the operating status of the turbine, the turbine oil system uses one of the six oil pumps to transfer oil from the lube oil reservoir to the turbine and generator components. The oil is cooled and filtered as necessary prior to delivery to the components.

System Evaluation Boundary

Major system components include main turbine oil reservoir, oil driven turbine and booster pump, baffler valves, emergency bearing oil pump, turning gear oil pump, motor suction pump, turbine oil coolers, vapor extractor and the turbine oil filters & pumps. Also included are the turbine bearing oil lift pumps, emergency seal oil pump, main seal oil pump, recirculating seal oil pump, seal oil vacuum pump, hydrogen seal oil vacuum tank and the bulk lubricating oil storage and transfer system. Additionally, included are the system piping, valves, instrumentation and controls to fill the turbine oil tank and to supply oil from the turbine oil tank to the main turbine lubricating oil system, hydrogen seal oil system, reactor feed pumps, HPCI turbine lubricating oil and recirculation MG set lubricating oil systems.

At Quad Cities, portions of this system are in proximity to safety related electrical components.

UFSAR References

Dresden Station UFSAR Section(s): Not applicable

Quad Cities Station UFSAR Section(s): None

License Renewal Boundary Diagram References

Dresden Station: Not applicable

Quad Cities Station:

LR-QDC-M-46-3, LR-QDC-M-48-1, LR-QDC-M-48-5, and LR-QDC-M-48-9.

System Intended functions

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.4-6 Component Groups Requiring Aging Management Review - Turbine Oil System (In-scope for Quad Cities only)

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting (Quad Cities only)	Pressure Boundary	3.4.1.6
Filters/Strainers (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.4.1.3, 3.4.2.16
Piping and Fittings (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.4.1.3, 3.4.2.32
Piping and Fittings (attached support) (Quad Cities only)	Structural Integrity (attached)	3.4.1.3, 3.4.2.32
Pump Casings (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.4.2.9, 3.4.2.37
Tanks (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.4.1.3, 3.4.2.43
Valves (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.4.1.3, 3.4.2.50
Valves (attached support) (Quad Cities only)	Structural Integrity (attached)	3.4.1.3, 3.4.2.50

Aging management review results for the turbine oil system are provided in Section 3.4.

2.3.4.7 Main Generator and Auxiliaries (In-Scope for Quad Cities Only)

System Purpose

The main generator and auxiliaries system consists of the main generator, main generator exciter, main generator stator coolers, and the isolated phase bus system. The stator water cooling system removes the heat produced in the generator armature conductors caused by heating losses and removes the heat produced in the main generator field rectifiers. The isolated phase bus system cools the conductors, which connect the generator to the main transformer.

System Operation

The main generator converts the mechanical energy of the turbine into electrical energy. The main generator exciter provides regulated excitation to the generator field windings to control generator output voltage and current. The main generator stator coolers provide clean, de-ionized cooling water to the stator and exciter during plant operation. The isolated phase bus electrically connects the main generator and the unit auxiliary transformer, and cools the main phase conductors.

System Evaluation Boundary

The system evaluation includes the main generator and auxiliaries components located in the Turbine Building. The auxiliary systems include the main generator exciter, stator water cooling system, and the isolated phase bus system.

At Quad Cities, portions of this system are in proximity to safety related electrical components.

UFSAR References

Dresden Station UFSAR Section(s): Not applicable

Quad Cities Station UFSAR Section(s): 8.3

License Renewal Boundary Diagram References

Dresden Station: Not applicable

Quad Cities Station:

LR-QDC-M-2020-1 and LR-QDC-M-2020-2.

System Intended functions

Preclude adverse effects on safety-related SSCs – Non-safety related components that could be a hazard to safety related SSCs maintain sufficient integrity so that the intended function of safety related SSCs is not adversely affected.

Component Groups Requiring Aging Management Review

Table 2.3.4-7 Component Groups Requiring Aging Management Review - Main Generator and Auxiliaries (In-scope for Quad Cites only)

Component Group	Component Intended Function	Aging Management Ref
Closure Bolting	Pressure Boundary	3.4.1.6
Heat Exchangers (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.4.2.11, 3.4.2.20
Housings (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.4.2.11, 3.4.2.21
Piping and Fittings (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.4.2.11, 3.4.2.33
Pumps (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.4.2.11, 3.4.2.38
Tanks (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.4.2.11, 3.4.2.44
Valves (spatial interaction) (Quad Cities only)	Leakage Boundary (spatial)	3.4.2.11, 3.4.2.52

Aging management review results for the main generator and auxiliaries system are provided in Section 3.4.

2.4 SCOPING AND SCREENING RESULTS: STRUCTURES

The structure scoping and screening results consist of lists of components and component groups that require aging management review, arranged by structure. Brief descriptions and intended functions are provided for structures within the scope of license renewal. For each in-scope structure, components or component groups requiring an aging management review are provided.

In addition to the structures within the scope of license renewal presented in this section, the commodities identified as component supports and insulation within the scope of license renewal were evaluated.

For each structure within the scope of license renewal, this section provides the following information:

- Purpose of the structure,
- A description of the structure,
- Reference to the applicable UFSAR sections,
- Reference to the applicable license renewal boundary diagrams,
- The intended functions of the structure within the scope of license renewal,
- A listing of the components or component groups that require aging management review, and the associated component intended functions and aging management reference to the aging management review results provided in Chapter 3.

A discussion of component groups, and component intended functions is provided in Section 3.0.

For component supports and insulation (commodities), this section provides the following information:

- A general description of commodity,
- A listing of the components or component groups that require aging management review and associated component intended functions and aging management reference to the aging management review results provided in Chapter 3.

Section 3.5 provides the results of the aging management reviews for the identified component groups.

The containments, structures, and component supports scoping and screening results are provided for the following structures and commodity groups:

- Primary Containment
- Reactor Building
- Main Control Room and Auxiliary Electric Equipment Room
- Turbine Building

- Diesel Generator Buildings
- Station Blackout Building and Yard Structures
- Isolation Condenser Pump House (Dresden Only)
- Makeup Demineralizer Building (Dresden Only)
- Radwaste Floor Drain Surge Tank
- Miscellaneous Foundations
- Crib House
- Unit 1 Crib House (Dresden only)
- Station Chimney
- Cranes and Hoists
- Component Supports Commodity Group
- Insulation Commodity Group

2.4.1 Primary Containment

Purpose of the Structure

The primary containment provides a barrier that controls the release of fission products to the secondary containment in the event of a loss-of-coolant accident. It also provides structures and a water pool that limit the pressure increase in the containment in the event of a loss-of-coolant accident.

Description of the Structure

The primary containment is a General Electric Mark I design. It consists of a drywell; a pressure suppression chamber that is partially filled with water, and a vent system connecting the drywell and the suppression chamber. The design, fabrication, and inspection of the primary containment was in accordance with the requirements of the ASME Pressure Vessel Code, Section III, Class B. It is a Class I structure.

The drywell is a steel pressure vessel with a spherical lower section and a cylindrical upper section. A portion of the lower spherical section is embedded in concrete. This embedment, in combination with upper lateral supports that are attached to the cylindrical section, forms the reactor support system. The drywell houses the reactor vessel, the reactor coolant recirculation system, and branch connections of the reactor primary system. It includes structural steel framing, a concrete radiation shield wall between the reactor pressure vessel and the drywell walls, a removable steel head, an equipment hatch and other access hatches, a personnel airlock with two mechanically interlocked doors, and penetrations.

The drywell head is removed during refueling operations. The head is held in place by bolts and is sealed with a double gasket, tongue-and-groove arrangement that permits checks for leak tightness without pressurizing the entire containment.

The pressure suppression chamber is a toroidal shaped, steel pressure vessel encircling the base of drywell. The suppression chamber is commonly called the torus and includes internal steel framing, supports, access hatches, and penetrations. The suppression chamber is mounted on support structures that transmit loads to the concrete foundation of the reactor building.

Eight circular vent lines form a connection between the drywell and the pressure suppresion chamber. These drywell vent lines are connected to a header which is contained within the air space of the suppression chamber. The header downcomers terminate below the suppression chamber water level.

The primary containment also contains structural interfacing components of the electrical penetrations.

UFSAR References

Dresden Station UFSAR Section(s): 6.2.1, 3.2.1

Quad Cities Station UFSAR Section(s): 6.2.1, 3.2.1

License Renewal Boundary Diagram References

Dresden Station: LR-DRE-M-1

Quad Cities Station: LR-QDC-M-1

Structure Intended Functions

Primary Containment - controls the release of fission products to the secondary containment in the event of design basis loss-of-coolant accidents (LOCA) so that offsite consequences are within acceptable limits.

Physical support and protection - provides physical support and protection for safety related components and components relied upon to demonstrate compliance with Fire Protection, ATWS, and SBO regulated events. The structure also contains components that are relied upon for compliance with 10 CFR 50.49, (EQ).

Pressure suppression - provides sufficient air and water volumes to absorb the energy released to the containment in the event of design basis events so that the pressure is within acceptable limits.

Water source - provides a source of water for emergency core cooling systems.

Radiation shielding - biological shield wall between the reactor pressure vessel and the drywell walls provides protection to personnel and components from radiation.

Component Groups Requiring Aging Management Review

Table 2.4-1 Component Groups Requiring Aging Management Review - Primary Containment

Component	Component Intended Function	Aging Management Ref
Beam Seats	Structural Support	3.5.1.20
Concrete & Grout	Structural Support	3.5.1.29
Concrete Slabs	Structural Support	3.5.1.20, 3.5.1.27
Concrete Walls	Structural Support	3.5.1.20, 3.5.1.27
Concrete Walls	Shelter, Protection, Shielding	3.5.1.20, 3.5.1.27
Containment Penetrations (Electrical)	Structural Support	3.5.1.3
Containment Penetrations (Electrical)	Fission Product Barrier	3.5.1.3
Containment Penetrations (Electrical)	Structural Pressure Barrier	3.5.1.3
Containment Penetrations (Mechanical)	Structural Support	3.5.1.3
Containment Penetrations (Mechanical)	Fission Product Barrier	3.5.1.3

Table 2.4-1 Component Groups Requiring Aging Management Review - Primary Containment (Continued)

Component	Component Intended Function	Aging Management Ref
Containment Penetrations (Mechanical)	Structural Pressure Barrier	3.5.1.3
Containment Penetrations Bellows	Fission Product Barrier	3.5.1.2
Containment Penetrations Bellows	Structural Pressure Barrier	3.5.1.2
Downcomers	Structural Pressure Barrier	3.5.1.12, 3.5.1.14
Drywell Expansion Foam	Expansion/Separation	3.5.2.8
Drywell Heads	Structural Support	3.5.1.12, 3.5.1.14
Drywell Heads	Fission Product Barrier	3.5.1.12, 3.5.1.14
Drywell Heads	Structural Pressure Barrier	3.5.1.12, 3.5.1.14
Drywells	Structural Support	3.5.1.12, 3.5.1.14
Drywells	Fission Product Barrier	3.5.1.12, 3.5.1.14
Drywells	Structural Pressure Barrier	3.5.1.12, 3.5.1.14
Hatches	Structural Pressure Barrier	3.5.1.4, 3.5.1.5
Misc. Steel (Includes Stairs, Ladders, Platforms, Gratings)	Non-S/R Structural Support	3.5.1.20
Penetration Sleeves, Penetration Bellows	Structural Support	3.5.1.1
Penetration Sleeves, Penetration Bellows	Fission Product Barrier	3.5.1.1
Penetration Sleeves, Penetration Bellows	Structural Pressure Barrier	3.5.1.1
Seals	Structural Pressure Barrier	3.5.1.6
Steel Embedments	Structural Support	3.5.1.20
Steel Panels and Cabinets	Structural Support	3.5.1.20
Structural Steel	Structural Support	3.5.1.20
Suppression Chambers	Structural Support	3.5.1.12, 3.5.1.13, 3.5.1.14
Suppression Chambers	Fission Product Barrier	3.5.1.12, 3.5.1.13, 3.5.1.14
Suppression Chambers	Structural Pressure Barrier	3.5.1.12, 3.5.1.13, 3.5.1.14
Thermowells	Structural Support	3.5.2.15
Thermowells	Structural Pressure Barrier	3.5.2.15
Vent Headers	Structural Support	3.5.1.13
Vent Headers	Structural Pressure Barrier	3.5.1.12, 3.5.1.13, 3.5.1.14
Vent Line Bellows	Structural Support	3.5.1.13
Vent Line Bellows	Structural Pressure Barrier	3.5.1.13, 3.5.1.17
Vent Lines	Structural Support	3.5.1.12, 3.5.1.14

Table 2.4-1 Component Groups Requiring Aging Management Review - Primary Containment (Continued)

Component	Component Intended Function	Aging Management Ref
Vent Lines	Structural Pressure Barrier	3.5.1.12, 3.5.1.14
Walls, Ceilings, Floors	Fire Barrier	3.3.1.28

Aging management review results for the primary containment are provided in Sections 3.3 and 3.5.

2.4.2 Reactor Building

Purpose of the Structure

The reactor building serves as the secondary containment. The secondary containment, in conjunction with other engineered safeguards and nuclear safety systems, limits the release of radioactive materials ensuring that site exposure resulting from a postulated design basis accident will remain below 10 CFR 100 guideline values. The reactor building provides secondary containment when the primary containment is in service and provides primary containment during reactor refueling and maintenance operations when the primary containment system is open.

Description of the Structure

A single seismic Class I reactor building for each unit completely encloses both reactor and primary containment structures, and auxiliary systems of the nuclear steam supply system. A major sub-structure within the reactor building is a reinforced concrete biological shield that surrounds each reactor and drywell portion of the primary containment. Additionally, the building houses the spent fuel pool, steam dryer/moisture separator storage pool, the new fuel storage vault, reactor cavity, reactor auxiliary equipment, refueling equipment and reactor servicing equipment. The reactor building consists of monolithic reinforced concrete floors and walls from its foundation to the refueling floor, with a separation wall between the two units. Above this floor, is the common refueling floor, where the building superstructure consisting of structural steel framing, sealed sheet metal siding and a pre-cast concrete roof, provide secondary containment integrity. The building is designed to contain positive internal pressure without structural failure and without pressure relief. Blowoff panels are installed as part of the reactor building superstructure siding to relieve pressure and control potential damage under short-term tornado loads. Personnel interlock/airlock access control doors have seals and are electrically controlled so that only one door in an “airlock” can be open at a time.

The containment barrier function of the reactor building is achieved through design and construction low leakage of air through the interlock/airlock doors, pipe and electrical penetration seals, and the building walls and roof. During normal operation, pressure in the building is automatically maintained at a slight negative pressure by controlling the exhaust to minimize exfiltration of airborne radioactive contamination, even under high wind conditions. The reactor building ventilation system (evaluated with HVAC-reactor building) is isolated on a secondary containment isolation signal.

Other structural components evaluated in this section include the reactor building penetrations and doors, equipment access building, the spent fuel pool, high density spent fuel racks, crane rails and the new fuel storage vault with associated components.

Reactor building structural items evaluated in other areas include the refueling platforms (evaluated with refueling equipment) and reactor building cranes (evaluated with cranes and hoists).

UFSAR References
Dresden Station UFSAR Section(s): 6.2.3, 3.2.1
Quad Cities Station UFSAR Section(s): 6.2.3, 3.2.1

License Renewal Boundary Diagram References
Dresden Station: LR-DRE-M-1
Quad Cities Station: LR-QDC-M-1

Structure Intended Functions

Containment - controls the potential release of fission products to the external environment so that offsite consequences of design basis events are within acceptable limits. The reactor building provides secondary containment function when the primary containment is required to be in service and provides primary containment function during reactor refueling and maintenance operations when the primary containment systems are open.

Physical support and protection - provides physical support for safety related and non-safety related components, protection for all personnel and safety related components; and components relied upon to demonstrate compliance with regulated events.

Protection and Radiation Shielding – provides leak-tight boundary to protect public health and safety in the event of postulated design basis events and radiation shielding.

Protection – provides protection for safe storage of new fuel.

Component Groups Requiring Aging Management Review

Table 2.4-2 Component Groups Requiring Aging Management Review – Reactor Building

Component	Component Intended Function	Aging Management Ref
Blowout Panels	Structural Pressure Barrier	3.5.1.20
Caulking/Sealants	Structural Pressure Barrier	3.5.2.4
Concrete & Grout	Structural Support	3.5.1.29
Concrete Beams	Structural Support	3.5.1.20, 3.5.1.21
Concrete Columns	Structural Support	3.5.1.20, 3.5.1.21, 3.5.1.27
Concrete Curbs	Direct Flow	3.5.1.20
Concrete Slabs	Structural Support	3.5.1.20, 3.5.1.21
Concrete Slabs	Shelter, Protection, Shielding	3.5.1.20
Concrete Walls	Structural Support	3.5.1.20, 3.5.1.21
Concrete Walls	Shelter, Protection, Shielding	3.5.1.20
Concrete Walls	Structural Pressure Barrier	3.5.1.20
Door Seals	Flood Barrier	3.5.2.7

Table 2.4-2 Component Groups Requiring Aging Management Review – Reactor Building (Continued)

Component	Component Intended Function	Aging Management Ref
Fire Doors	Fire Barrier	3.3.2.4
Fire Doors (Dresden only)	Shelter, Protection, Shielding	3.3.2.4
Fire Proofing	Fire Barrier	3.3.2.62
Fire Wrap	Fire Barrier	3.3.2.63
Foundations	Structural Support	3.5.1.20, 3.5.1.21, 3.5.1.25, 3.5.1.26
Liners	Structural Pressure Barrier	3.5.1.23
Masonry Walls	Structural Support	3.5.1.24
Masonry Walls	Fire Barrier	3.3.2.129
Masonry Walls	Missile Barrier	3.5.1.24
Masonry Walls	HELB Shielding	3.5.1.24
Metal Decking	Structural Support	3.5.1.20
Metal Siding	Shelter, Protection, Shielding	3.5.1.20
Metal Siding	Structural Pressure Barrier	3.5.1.20
Misc. Steel (Includes Grating, Ladders)	Structural Support	3.5.1.20
Misc. Steel (Includes Kick Plates, Ladders, Platforms, Stairs, Railing)	Non-S/R Structural Support	3.5.1.20
Neutron-Absorbing Sheets	Absorb Neutrons	3.3.1.12, 3.3.1.9
New Fuel Racks	Structural Support	3.5.2.10
Penetration Seals (Includes Secondary Containment Boot Seal)	Fire Barrier	3.3.1.18
Penetration Sleeves	Non-S/R Structural Support	3.5.1.20
Precast Concrete Panels	Structural Support	3.5.1.20
Precast Concrete Panels	Shelter, Protection, Shielding	3.5.1.20
Roofing	Shelter, Protection, Shielding	3.5.2.11
Secondary Containment Boot Seals	Structural Pressure Barrier	3.5.2.12
Steel Doors	Shelter, Protection, Shielding	3.5.1.20
Steel Doors	Flood Barrier	3.5.1.20
Steel Doors	Structural Pressure Barrier	3.5.1.20
Steel Embedments	Structural Support	3.5.1.20
Steel Panels and Cabinets	Structural Support	3.5.1.20
Steel Plates	Missile Barrier	3.5.1.20
Steel Plates	HELB Shielding	3.5.1.20
Steel Plates	Direct Flow	3.5.1.20

Table 2.4-2 Component Groups Requiring Aging Management Review – Reactor Building (Continued)

Component	Component Intended Function	Aging Management Ref
Storage Racks	Structural Support	3.3.1.11
Structural Steel (includes flued head anchor support)	Structural Support	3.5.1.20
Structural Steel	HELB Shielding	3.5.1.20
Structural Steel	Pipe Whip Restraint	3.5.1.20
Walls, Ceilings, Floors	Fire Barrier	3.3.1.28

Aging management review results for the reactor building are provided in Sections 3.3 and 3.5.

2.4.3 Main Control Room and Auxiliary Electric Equipment Room

Purpose of the Structure

The main control room and auxiliary electric equipment room provide protection and structural support of the control equipment required for normal station operation and for shutdown of the plant under abnormal conditions. The main control room provides protection to safety related components and to operating personnel from radiation exposure, High Energy Line Break (HELB), tornado and internally generated missiles.

Description of the Structure

The main control room and auxiliary electric equipment room contain the controls for normal station operation and for shutdown of the plant under abnormal conditions. This combined structure is seismic Class I, primarily reinforced concrete, capable of accommodating loading conditions imposed during any design basis accident (DBA) without failure.

The Dresden main control room has reinforced concrete and reinforced concrete block walls, with a reinforced concrete floor and ceiling. The Dresden auxiliary electric equipment room serves as the cable spreading room for both units and houses the computer room. It is classified as a Class II structure, which has been investigated to assure that the integrity of the Class I items is not compromised. The auxiliary electrical equipment room is a reinforced concrete structure, with structural steel support elements, a reinforced concrete floor, and the control room floor above, as its ceiling.

The Quad Cities main control room, auxiliary electrical equipment room, and the cable spreading room complex is a Class I structure, designed to accommodate loading conditions imposed during any DBA without failure. The control room is heavy-walled, constructed of ordinary reinforced concrete and of magnetite (high density) concrete, with ordinary reinforced concrete for the control room and cable spreading room floor, and ordinary reinforced concrete for the auxiliary electric equipment room floor slab, and roof slab. The cable spreading room is located directly below the main control room and above the auxiliary electrical equipment room and is used solely for the routing of instrument and control cables. The auxiliary electrical equipment room contains alternate safe shutdown equipment and the cabling from the tunnel below to the cable spreading room.

UFSAR References

Dresden Station UFSAR Section(s): 3.1.1.3.1, 3.2.1, 3.2.6 and 6.4

Quad Cities Station UFSAR Section(s): 3.1.3.1, 3.2.1 and 6.4

License Renewal Boundary Diagram References

Dresden Station: None

Quad Cities Station: None

Structure Intended Functions

Physical support and protection - provides physical support and protection for safety related components and components relied upon to demonstrate compliance with Fire Protection and SBO regulated events.

Personnel protection - provides shelter, protection and radiation shielding for essential operating personnel.

Component Groups Requiring Aging Management Review

Table 2.4-3 Component Groups Requiring Aging Management Review - Main Control Room and Auxiliary Electric Equipment Room

Component	Component Intended Function	Aging Management Ref
Concrete & Grout	Structural Support	3.5.1.29
Concrete Beams (Quad Cities only)	Structural Support	3.5.1.20
Concrete Columns (Quad Cities only)	Structural Support	3.5.1.20
Concrete Curbs (Dresden only)	Direct Flow	3.5.1.20
Concrete Manholes (Dresden only)	Structural Support	3.5.1.20
Concrete Manholes (Dresden only)	Fire Barrier	3.3.1.28
Concrete Slabs	Structural Support	3.5.1.20, 3.5.1.21, 3.5.1.27
Concrete Slabs	Shelter, Protection, Shielding	3.5.1.20, 3.5.1.21, 3.5.1.27
Concrete Walls	Structural Support	3.5.1.20
Concrete Walls	Shelter, Protection, Shielding	3.5.1.20, 3.5.1.27
Fire Doors	Fire Barrier	3.3.2.4
Fire Proofing	Fire Barrier	3.3.2.62
Fire Wrap	Fire Barrier	3.3.2.63
Foundations	Structural Support	3.5.1.20, 3.5.1.21, 3.5.1.25, 3.5.1.26
Masonry Walls	Structural Support	3.5.1.24
Masonry Walls	Fire Barrier	3.3.2.129
Masonry Walls	Shelter, Protection, Shielding	3.5.1.24

Table 2.4-3 Component Groups Requiring Aging Management Review - Main Control Room and Auxiliary Electric Equipment Room (Continued)

Component	Component Intended Function	Aging Management Ref
Masonry Walls	Missile Barrier	3.5.1.24
Metal Decking (Dresden only)	Structural Support	3.5.1.20
Misc. Steel (Quad Cities only)	Structural Support	3.5.1.20
Penetration Seals	Fire Barrier	3.3.1.18
Penetration Seals (Dresden only)	Non-S/R Structural Support	3.3.1.18
Penetration Sleeves	Non-S/R Structural Support	3.5.1.20
Roofing (Quad Cities only)	Shelter, Protection, Shielding	3.5.2.11
Steel Embedments	Structural Support	3.5.1.20
Steel Panels and Cabinets	Structural Support	3.5.1.20
Structural Steel	Structural Support	3.5.1.20
Walls, Ceilings, Floors	Fire Barrier	3.3.1.28

Aging management review results for the main control room and auxiliary electric equipment room are provided in Sections 3.3 and 3.5.

2.4.4 Turbine Building

Purpose of the Structure

The purpose of the turbine building is the protection of the main turbine generators and other plant equipment from environmental hazards and missiles, as well as providing structural support for equipment and radiation shielding.

Description of the Structure

The turbine building is a common structure shared by both units at each station. Located in one side or half of the turbine building are the turbine-generator, exciter, condenser, feedwater heaters, feedwater and condensate pumps, demineralizer system, condenser circulating system and electrical switchgear. Duplicate equipment and systems for the other unit are located in the other half of the building. Each turbine building superstructure consists of a structural steel frame-type design with metal siding from the turbine floor up. All Class I components in the turbine building are located in levels below the turbine main floor within a reinforced concrete structure with capabilities similar to the reactor building. Large equipment, located in the superstructure, is designed and supported to preclude failure that could damage equipment related to the ECCS systems or cause significant release of radioactivity.

The building is a Class II structure and provides Class I protection in areas where Class I items and associated SSCs are located. Exceptions are the swing emergency diesel generators and Quad Cities emergency diesel generators which are evaluated in the Emergency Diesel Generator Room. The turbine building cranes are evaluated with cranes and hoists.

UFSAR References

Dresden Station UFSAR Section(s): 1.2.2.2

Quad Cities Station UFSAR Section(s): 1.2.2.2

License Renewal Boundary Diagram References

Dresden Station: LR-DRE-M-1

Quad Cities Station: LR-QDC-M-1

Structure Intended Functions

Physical support and protection - provides physical support and protection for safety related components and components relied upon to demonstrate compliance with regulated events.

Radiation shielding – provides shielding that protects personnel and components from radiation.

Protection – provides missile barrier protection for internally and externally generated events and the flood protection for SSCs.

Component Groups Requiring Aging Management Review

Table 2.4-4 Component Groups Requiring Aging Management Review - Turbine Building

Component	Component Intended Function	Aging Management Ref
Caulking/Sealants	Structural Pressure Barrier	3.5.2.3, 3.5.2.4
Concrete & Grout	Structural Support	3.5.1.29
Concrete & Grout	Non-S/R Structural Support	3.5.1.29
Concrete Beams	Structural Support	3.5.1.20, 3.5.1.27
Concrete Columns	Structural Support	3.5.1.20
Concrete Curbs	Direct Flow	3.5.1.20
Concrete Manholes	Structural Support	3.5.1.20
Concrete Manholes	Shelter, Protection, Shielding	3.5.1.20
Concrete Slabs	Structural Support	3.5.1.20
Concrete Walls	Structural Support	3.5.1.20
Fire Doors	Fire Barrier	3.3.1.18
Fire Proofing	Fire Barrier	3.3.2.62
Fire Wrap	Fire Barrier	3.3.2.63
Foundations	Structural Support	3.5.1.20, 3.5.1.25, 3.5.1.26
Masonry Walls	Structural Support	3.5.1.24
Masonry Walls	Fire Barrier	3.3.2.129
Masonry Walls	Shelter, Protection, Shielding	3.5.1.24
Masonry Walls	Non-S/R Structural Support	3.5.1.24
Metal Siding	Shelter, Protection, Shielding	3.5.1.20
Misc. Steel (Includes Gratings, Ladders, Platforms, Railings, Stairs, Kickplates)	Non-S/R Structural Support	3.5.1.20
Penetration Seals	Flood Barrier	3.3.1.18
Penetration Sleeves	Non-S/R Structural Support	3.5.1.20
Precast Concrete Panels	Structural Support	3.5.1.20
Precast Concrete Panels	Non-S/R Structural Support	3.5.1.20
Roofing	Shelter, Protection, Shielding	3.5.2.11
Steel Doors	Flood Barrier	3.5.1.20
Steel Doors	Non-S/R Structural Support	3.5.1.20
Steel Doors	Structural Pressure Barrier	3.5.1.20
Steel Embedments	Structural Support	3.5.1.20
Steel Panels and Cabinets	Structural Support	3.5.1.20
Steel Plates	Shelter, Protection, Shielding	3.5.1.20

Table 2.4-4 Component Groups Requiring Aging Management Review - Turbine Building (Continued)

Component	Component Intended Function	Aging Management Ref
Steel Plates (Dresden only)	Flood Barrier	3.5.1.20
Structural Steel	Structural Support	3.5.1.20
Structural Steel	Pipe Whip Restraint	3.5.1.20
Walls, Ceilings, Floors	Fire Barrier	3.3.1.28

Aging management review results for the turbine building are provided in Sections 3.3 and 3.5.

2.4.5 Diesel Generator Buildings

Purpose of the Structure

The purpose of the diesel generator buildings is to provide structural support and protection of the emergency diesel generators and HPCI system components (Dresden only).

Description of the Structure

The diesel generator buildings contain the Dresden Unit 2/3 (swing) emergency diesel generator and HPCI building (a structure which includes both the swing diesel generator and HPCI system components) and the Quad Cities Unit 1,2 and 1/2 (swing) emergency diesel generator rooms.

The Dresden diesel generator and HPCI building is a Class I concrete structure that houses the Unit 2/3 (swing) emergency diesel generator, HPCI system equipment, and other safe shutdown equipment. It abuts the Unit 3 reactor building and shares the reactor building wall on its north side. The Dresden Unit 2 and 3 emergency diesel generators are housed in the turbine building.

The three Quad Cities diesel generator rooms are Class II concrete structures that have been evaluated to ensure that they have the capability to protect safety related components. The rooms provide structural support and protection of the emergency diesel generators as well as fire protection of adjacent safety related structures. The Unit 1 diesel generator room is located in the southeast corner of the Unit 1 section of the turbine building. The Unit 2 diesel generator room is located in the northeast corner of Unit 2 section of the turbine building. The Unit 1/2 (swing) diesel generator room is adjacent to the reactor building. It is centered on the reactor building east wall, which is shared with the Unit 1/2 diesel generator room.

UFSAR References

Dresden Station UFSAR Section(s): 3.2.6, 8.3

Quad Cities Station UFSAR Section(s): 3.2.6, 8.3.1.6.1

License Renewal Boundary Diagram References

Dresden Station: LR-DRE-M-1

Quad Cities Station: LR-QDC-M-1

Structure Intended Functions

Physical support and protection - provides physical support and protection for safety related components and components relied upon to demonstrate compliance with Fire Protection and SBO (Dresden only) regulated events.

Containment - provide leak-tight barrier protecting the health and safety of the public in the event of any postulated design basis events and also provides secondary containment boundary (Dresden 2/3 Diesel Generator Room only)

Protection – provides missile barrier for internally and externally generated events.

Component Groups Requiring Aging Management Review

Table 2.4-5 Component Groups Requiring Aging Management Review - Diesel Generator Buildings

Component	Component Intended Function	Aging Management Ref
Concrete & Grout	Structural Support	3.5.1.29
Concrete & Grout	Non-S/R Structural Support	3.5.1.29
Concrete Beams (Dresden only)	Structural Support	3.5.1.20
Concrete Curbs	Direct Flow	3.5.1.20
Concrete Shield Plugs (Dresden only)	Structural Support	3.5.1.20
Concrete Shield Plugs (Dresden only)	Fire Barrier	3.3.1.28
Concrete Shield Plugs (Dresden only)	Shelter, Protection, Shielding	3.5.1.20
Concrete Slabs	Structural Support	3.5.1.20
Concrete Slabs	Shelter, Protection, Shielding	3.5.1.20
Concrete Walls	Structural Support	3.5.1.20, 3.5.1.21, 3.5.1.27
Concrete Walls	Shelter, Protection, Shielding	3.5.1.20
Concrete Walls	Missile Barrier	3.5.1.20
Fire Doors	Fire Barrier	3.3.1.18
Fire Proofing	Fire Barrier	3.3.2.62
Fire Wrap	Fire Barrier	3.3.2.63
Foundations (Dresden only)	Structural Support	3.5.1.20
Masonry Walls (Dresden only)	Structural Support	3.5.1.24
Masonry Walls (Dresden only)	Fire Barrier	3.3.2.129
Metal Decking (Quad Cities only)	Structural Support	3.5.1.20
Misc. Steel (Includes Ladders, Railings, Stairs, Gratings, Kick Plates, Platforms)	Non-S/R Structural Support	3.5.1.20

Table 2.4-5 Component Groups Requiring Aging Management Review - Diesel Generator Buildings (Continued)

Component	Component Intended Function	Aging Management Ref
Penetration Seals (Dresden only)	Fire Barrier	3.3.1.18
Penetration Sleeves	Non-S/R Structural Support	3.5.1.20
Penetration Sleeves (Quad Cities only)	Structural Pressure Barrier	3.5.1.20
Steel Doors (Dresden only)	Structural Pressure Barrier	3.5.1.20
Steel Embedments	Structural Support	3.5.1.20
Steel Panels and Cabinets	Structural Support	3.5.1.20
Steel Plates (Dresden only)	Direct Flow	3.5.1.20
Structural Steel	Structural Support	3.5.1.20
Walls, Ceilings, Floors	Fire Barrier	3.3.1.28

Aging management review results for the diesel generator buildings are provided in Sections 3.3 and 3.5.

2.4.6 Station Blackout Building and Yard Structures

Purpose of the Structure

The station blackout (SBO) building provides structural support and environmental protection for the station blackout diesel generators and associated components. The off site power structures are to provide sufficient capacity and capability to start and operate safety related equipment.

Description of the Structure

Other structures grouped with the SBO building for evaluation are off site power structures and foundations associated with the reserve auxiliary transformers, dead end structures, bus duct supports and intermediate transmission towers. For both stations, the bus duct supports, dead end structures, and intermediate transmission towers are either galvanized or coated steel with reinforced concrete foundations. Foundations for supporting transformers and circuit breakers are also reinforced concrete.

Structural boundaries include the physical extent of the circuit breaker foundations in the switchyards serving the RATs, 345-kV and 138-kV dead end structures, intermediate transmission towers serving the RATs, 345 k-V and 138-kV dead end structures serving adjacent to the reserve auxiliary transformer (Dresden only), reserve auxiliary transformer foundations, and bus ducts and their supports and foundations terminating at the turbine building (Dresden) and terminating at the diesel generator/turbine building (Quad Cities)

The Dresden station blackout building is a Class I structure that houses the station blackout diesel generators and safety related components, including the Unit 2 alternate 125VDC batteries. It is a heavy-walled reinforced concrete structure capable of protecting its contents from weather-related events that could initiate a station blackout event. The underground diesel oil tank foundation on the eastern side of the building supports the 15,000-gallon fuel tank.

The Quad Cities station blackout building protects the diesel generators and associated components from weather-related events, which could initiate a station blackout event and provides physical isolation from safety related components. It is a two-floor structure consisting of a reinforced concrete ground floor slab/foundation, steel framed exterior walls with corrugated metal siding, metal deck supported concrete slab second floor, and a roof consisting of a fully-adhered single-ply system on rigid insulation supported by metal decking. The station blackout diesel generators are supported within the building by independent reinforced concrete foundation slabs. Fire-rated block walls separate the Unit 1 and 2 diesel generator rooms, the day tank rooms and the second floor battery rooms.

UFSAR References

Dresden Station UFSAR Section(s): 9.5.9, 8.2

Quad Cities Station UFSAR Section(s): 8.3.1.9, 8.2

License Renewal Boundary Diagram References

Dresden Station: LR-DRE-M-1
Quad Cities Station: LR-QDC-M-1

Structure Intended Functions

Physical support and protection – provides physical support and protection for safety related components and components relied upon to demonstrate compliance with the SBO regulated event. For Quad Cities only, the structure also provides protection for components relied upon to demonstrate compliance with the Fire Protection regulated event.

Component Groups Requiring Aging Management Review

Table 2.4-6 Component Groups Requiring Aging Management Review - Station Blackout Building

Component	Component Intended Function	Aging Management Ref
Bus Duct Covers	Non-S/R Structural Support	3.5.2.1
Bus Duct Supports	Non-S/R Structural Support	3.5.2.2
Caulking/Sealants	Flood Barrier	3.5.2.4
Caulking/Sealants	Expansion/Separation	3.5.2.4
Concrete & Grout	Structural Support	3.5.1.29
Concrete & Grout	Non-S/R Structural Support	3.5.1.29
Concrete Curbs	Direct Flow	3.5.1.20
Concrete Manholes (Quad Cities only)	Structural Support	3.5.1.20
Concrete Manholes (Quad Cities only)	Non-S/R Structural Support	3.5.1.20, 3.5.1.21
Concrete Slabs	Structural Support	3.5.1.20
Concrete Slabs	Non-S/R Structural Support	3.5.1.20
Concrete Walls (Dresden only)	Structural Support	3.5.1.20, 3.5.1.21
Dead End Structures	Non-S/R Structural Support	3.5.2.6
Doors	Fire Barrier	3.3.1.18
Fire Proofing	Fire Barrier	3.3.2.62
Fire Wrap	Fire Barrier	3.3.2.63
Foundations	Structural Support	3.5.1.20, 3.5.1.25, 3.5.1.26, 3.5.1.27
Foundations	Non-S/R Structural Support	3.5.1.20
Masonry Walls	Structural Support	3.5.1.24
Masonry Walls	Fire Barrier	3.3.2.129
Masonry Walls	Shelter, Protection, Shielding	3.5.1.24
Metal Decking	Non-S/R Structural Support	3.5.1.20
Metal Decking (Dresden only)	Shelter, Protection, Shielding	3.5.1.20

Table 2.4-6 Component Groups Requiring Aging Management Review - Station Blackout Building (Continued)

Component	Component Intended Function	Aging Management Ref
Metal Siding	Shelter, Protection, Shielding	3.5.1.20
Misc. Steel (Dresden only- Includes Gratings, Kick Plates, Ladders, Platforms, Railings)	Non-S/R Structural Support	3.5.1.20
Penetration Seals (Quad Cities only)	Fire Barrier	3.3.1.18
Penetration Seals (Quad Cities only)	Expansion/Separation	3.3.1.18
Penetration Sleeves	Non-S/R Structural Support	3.5.1.20
Steel Embedments	Structural Support	3.5.1.20
Steel Panels and Cabinets	Non-S/R Structural Support	3.5.1.20
Steel Piles (Dresden only)	Non-S/R Structural Support	3.3.2.207
Steel Plates (Dresden only)	Missile Barrier	3.5.1.20
Structural Steel	Non-S/R Structural Support	3.5.1.20
Transmission Towers	Non-S/R Structural Support	3.5.2.16
Walls, Ceilings, Floors	Fire Barrier	3.3.1.28

Aging management review results for the station blackout structure and foundation are provided in Sections 3.3 and 3.5.

2.4.7 Isolation Condenser Pump House (Dresden Only)

Purpose of the Structure

The isolation condenser pump house provides structural support and environmental protection for the two diesel-driven isolation condenser makeup water pumps.

Description of the Structure

The isolation condenser pump house is a two-floor Class II structure with the first floor at grade and the other floor being a reinforced concrete basement. The above grade exterior north, south and east walls, as well as, an interior center wall are constructed of concrete block. The west wall that borders the reactor building is reinforced concrete with Rodofoam seismic gap separation at the reactor building wall. The first floor is a metal deck supported concrete slab. The roof consists of single-ply system on rigid insulation supported by metal decking on structural steel.

UFSAR References

Dresden Station UFSAR Section(s): 5.4.6

Quad Cities UFSAR Section(s): Not Applicable

License Renewal Boundary Diagram References

Dresden Station: LR-DRE-M-1

Quad Cities Station: Not applicable

Structure Intended Functions

Credited in regulated events – provides physical support and protection for components relied upon to demonstrate compliance with the Fire Protection regulated event.

Component Groups Requiring Aging Management Review

Table 2.4-7 Component Groups Requiring Aging Management Review - Isolation Condenser Pump House (Dresden Only)

Component	Component Intended Function	Aging Management Ref
Caulking/Sealants (Dresden only)	Expansion/Separation	3.5.2.4
Concrete Duct Banks (Dresden only)	Non-S/R Structural Support	3.5.1.21
Concrete Walls (Dresden only)	Structural Support	3.5.1.20
Concrete Walls (Dresden only)	Non-S/R Structural Support	3.5.1.20

Table 2.4-7 Component Groups Requiring Aging Management Review - Isolation Condenser Pump House (Dresden only) (Continued)

Component	Component Intended Function	Aging Management Ref
Doors (Dresden only)	Fire Barrier	3.3.1.18
Foundations (Dresden only)	Structural Support	3.5.1.20, 3.5.1.21, 3.5.1.25, 3.5.1.26
Foundations (Dresden only)	Non-S/R Structural Support	3.5.1.20
Masonry Walls (Dresden only)	Structural Support	3.5.1.24
Masonry Walls (Dresden only)	Fire Barrier	3.3.2.129
Masonry Walls (Dresden only)	Non-S/R Structural Support	3.5.1.24
Metal Decking (Dresden only)	Shelter, Protection, Shielding	3.5.1.20
Metal Decking (Dresden only)	Non-S/R Structural Support	3.5.1.20
Penetration Seals (Dresden only)	Fire Barrier	3.3.1.18
Penetration Sleeves (Dresden only)	Non-S/R Structural Support	3.5.1.20
Roofing (Dresden only)	Shelter, Protection, Shielding	3.5.2.11
Seismic Gap Filler (Dresden only)	Expansion/Separation	3.5.2.13
Steel Doors (Dresden only)	Non-S/R Structural Support	3.5.1.20
Steel Embedments (Dresden only)	Structural Support	3.5.1.20
Steel Panels and Cabinets (Dresden only)	Non-S/R Structural Support	3.5.1.20
Structural Steel (Dresden only)	Non-S/R Structural Support	3.5.1.20
Walls, Ceilings, Floors (Dresden only)	Fire Barrier	3.3.1.28

Aging management review results for the isolation condenser pump house are provided in Sections 3.3 and 3.5.

2.4.8 Makeup Demineralizer Building (Dresden Only)

Purpose of the Structure

The makeup demineralizer building at Dresden provides support and protection for instrumentation required for remote monitoring of water level of the "B" contaminated condensate storage tank in the event that the main control room is evacuated due to fire.

Description of the Structure

The makeup demineralizer building is a pre-engineered steel building that includes an interior reinforced concrete slab, anchor bolt hardware, instrument rack, and foundation support for the level indicator.

UFSAR References

Dresden Station UFSAR Section(s): None

Quad Cities Station UFSAR Section(s): Not Applicable

License Renewal Boundary Diagram References

Dresden Station: LR-DRE-M-1

Quad Cities Station: Not applicable

Structure Intended Functions

Physical support and protection - provides physical support and protection for components relied upon to demonstrate compliance with the Fire Protection regulated event.

Component Groups Requiring Aging Management Review

Table 2.4-8 Component Groups Requiring Aging Management Review - Makeup Demineralizer Building (Dresden Only)

Component	Component Intended Function	Aging Management Ref
Concrete & Grout (Dresden only)	Structural Support	3.5.1.29
Concrete Slabs (Dresden only)	Structural Support	3.5.1.20
Concrete Slabs (Dresden only)	Non-S/R Structural Support	3.5.1.20
Foundations (Dresden only)	Structural Support	3.5.1.20, 3.5.1.21, 3.5.1.25, 3.5.1.26, 3.5.1.27

Table 2.4-8 Component Groups Requiring Aging Management Review - Makeup Demineralizer Building (Dresden Only) (Continued)

Component	Component Intended Function	Aging Management Ref
Foundations (Dresden only)	Non-S/R Structural Support	3.5.1.20
Metal Decking (Dresden only)	Shelter, Protection, Shielding	3.5.1.20
Metal Siding (Dresden only)	Shelter, Protection, Shielding	3.5.1.20
Steel Doors (Dresden only)	Shelter, Protection, Shielding	3.5.1.20
Steel Panels and Cabinets (Dresden only)	Structural Support	3.5.1.20
Structural Steel (Dresden only)	Non-S/R Structural Support	3.5.1.20

Aging management review results for the makeup demineralizer building are provided in Section 3.5.

2.4.9 Radwaste Floor Drain Surge Tank

Purpose of the Structure

The floor drain surge tank provides the necessary surge volume for the floor drain system, which collects potentially radioactive liquids.

Description of the Structure

The above ground, floor drain surge tank has thick reinforced concrete walls for shielding and electric heaters to prevent freezing during cold weather. The tank bottom is sloped to reduce sludge buildup. The floor drain surge tank is a Class I structure and is, therefore, not considered an above-ground tank for the purpose of the curies content requirements. The floor drain surge tank includes the attached pump house structure, foundations, floors, walls, roof, and stainless steel liner.

UFSAR References

Dresden Station UFSAR Section(s): 11.2, 3.2.1

Quad Cities Station UFSAR Section(s): 11.2

License Renewal Boundary Diagram References

Dresden Station: LR-DRE-M-1

Quad Cities Station: LR-QDC-M-1

Structure Intended Functions

Radioactive fluid containment - provides physical barrier and support to contain potentially radioactive liquid waste.

Component Groups Requiring Aging Management Review

Table 2.4-9 Component Groups Requiring Aging Management Review - Radwaste Floor Drain Surge Tank

Component	Component Intended Function	Aging Management Ref
Concrete Manholes	Structural Support	3.5.1.20, 3.5.1.21
Concrete Manholes	Shelter, Protection, Shielding	3.5.1.20
Concrete Slabs	Structural Support	3.5.1.20
Concrete Slabs	Shelter, Protection, Shielding	3.5.1.20

Table 2.4-9 Component Groups Requiring Aging Management Review - Radwaste Floor Drain Surge Tank (Continued)

Component	Component Intended Function	Aging Management Ref
Concrete Walls	Structural Support	3.5.1.20
Concrete Walls	Shelter, Protection, Shielding	3.5.1.20
Foundations	Structural Support	3.5.1.20, 3.5.1.21, 3.5.1.25, 3.5.1.26
Liners	Structural Pressure Barrier	3.5.1.28
Steel Embedments	Structural Support	3.5.1.20

Aging management review results for the radwaste floor drain surge tank and foundations are provided in Section 3.5.

2.4.10 Miscellaneous Foundations

Purpose of the Structure

Contaminated Condensate Storage Tank Foundations

The contaminated condensate storage tank foundations provide physical support for the non-safety related contaminated condensate storage tanks.

Diesel Generator Fuel Oil Storage Tank Foundations

The diesel generator fuel oil storage tank foundations provide structural support for the safety related diesel generator fuel oil storage tanks.

Description of the Structure

Contaminated Condensate Storage Tank Foundations

The condensate storage facilities provide a storage volume for clean and potentially contaminated water of suitable quality for use in the reactor and other systems throughout the plant. The Dresden condensate storage facilities ensure that an adequate amount of water is available from each contaminated condensate storage tank for use by HPCI pumps. The Quad Cities condensate storage facilities ensure that an adequate amount of water is available from each contaminated condensate storage tank for use by HPCI, RCIC, and safe shutdown pumps. The contaminated condensate storage tank foundations are Class II reinforced concrete foundations and include anchor bolts.

Diesel Generator Fuel Oil Storage Tank Foundations

Each diesel generator fuel oil storage tank, except for the Quad Cities Unit 1 fiberglass tank, is supported on three reinforced concrete foundation pads and anchored with anchor bolts (four per pad). The Quad Cities Unit 1 fiberglass tank is anchored to two reinforced concrete foundations and is restrained in place by stainless steel straps and turnbuckle assemblies.

UFSAR References

Dresden Station UFSAR Section(s): 9.2.6, 9.5.4

Quad Cities Station UFSAR Section(s): 9.2.6, 9.5.4

License Renewal Boundary Diagram References

Dresden Station: LR-DRE-M-1

Quad Cities Station: LR-QDC-M-1

Structure Intended Functions

Contaminated Condensate Storage Tank Foundations

Regulated event component support - provide support for components relied upon to demonstrate compliance with Fire Protection, ATWS, and SBO regulated events.

Diesel Generator Fuel Oil Storage Tank Foundations

Safety-related component support – provide support for safety related, seismically qualified fuel oil storage tanks provided for emergency diesel generators.

Component Groups Requiring Aging Management Review

Table 2.4-10 Component Groups Requiring Aging Management Review - Miscellaneous Foundations

Component	Component Intended Function	Aging Management Ref
Caulking/Sealants (Dresden only)	Shelter, Protection, Shielding	3.5.2.4
Foundations	Structural Support	3.5.1.20, 3.5.1.21, 3.5.1.25, 3.5.1.26
Foundations	Non-S/R Structural Support	3.5.1.20, 3.5.1.26
Steel Embedments (Quad Cities only)	Structural Support	3.5.1.20

Aging management review results for miscellaneous foundations are provided in Section 3.5.

2.4.11 Crib House

Purpose of the Structure

The crib house serves as the entry point for river water into plant systems. It protects and supports the pumps and pipes which deliver river water to the plant.

Description of the Structure

The crib house is a reinforced concrete structure with a concrete block and steel superstructure. It contains the circulating water, service water and diesel driven fire pumps.

At Dresden, the crib house includes the diesel generator cooling water pumps and the suction piping for the containment cooling service water system pumps. The diesel generator cooling water pumps and the containment cooling service water system are safety related. The crib house also contains stop logs that can be used to isolate the compartment and raise its water level where the containment cooling service water system pump and the diesel fire pump take their suction. The crib house is classified as Class II, and was investigated to assure that the integrity of the Class I items is not compromised.

At Quad Cities, the crib house includes the suction lines for the RHR service water system. The RHR service water system is safety related. The crib house is classified as Class II, and was investigated to assure that it will not fail and isolate the plant from the river water source. For license renewal evaluation purposes the Quad Cities discharge flume weir wall that forms one of the boundaries of the ultimate heat sink is included as part of the crib house.

UFSAR References

Dresden Station UFSAR Section(s): 1.2.2.2, 3.3.2.3.2, 3.8.5, 9.2.5, 9.5.5

Quad Cities Station UFSAR Section(s): 3.3, 3.8.6, 9.2.5

License Renewal Boundary Diagram References

Dresden Station: LR-DRE-M-1

Quad Cities Station: LR-QDC-M-1

Structure Intended Functions

Physical support and protection - provides physical support and protection for safety related components and components relied upon to demonstrate compliance with the Fire Protection regulated event.

Heat sink – provides heat sink during SBO or design basis events.

Water source – provides source of cooling water for plant shutdown.

Component Groups Requiring Aging Management Review

Table 2.4-11 Component Groups Requiring Aging Management Review - Crib House

Component	Component Intended Function	Aging Management Ref
Concrete & Grout	Structural Support	3.5.1.29
Concrete & Grout	Non-S/R Structural Support	3.5.1.29
Concrete Canal Weirs (Quad Cities only)	Heat Sink	3.5.1.22
Concrete Curbs	Direct Flow	3.5.1.22
Concrete Slabs	Structural Support	3.5.1.22, 3.5.1.26
Concrete Slabs	Shutdown Cooling Water	3.5.1.22
Concrete Slabs	Heat Sink	3.5.1.22, 3.5.1.26
Concrete Stairs	Structural Support	3.5.1.22
Concrete Stairs	Non-S/R Structural Support	3.5.1.22
Concrete Walls	Structural Support	3.5.1.22
Concrete Walls	Non-S/R Structural Support	3.5.1.22
Concrete Walls	Shutdown Cooling Water	3.5.1.22
Concrete Walls	Heat Sink	3.5.1.22
Fire Doors (Dresden only)	Fire Barrier	3.3.1.18
Foundations	Structural Support	3.5.1.22, 3.5.1.26
Foundations	Non-S/R Structural Support	3.5.1.22
Masonry Walls	Structural Support	3.5.1.24
Masonry Walls	Shelter, Protection, Shielding	3.5.1.24
Metal Siding (Dresden only)	Shelter, Protection, Shielding	3.5.1.22
Misc. Steel (Dresden only)	Non-S/R Structural Support	3.5.1.22
Misc. Steel (Dresden only)	Direct Flow	3.5.1.22
Precast Concrete Panels	Structural Support	3.5.1.22
Precast Concrete Panels	Shelter, Protection, Shielding	3.5.1.22
Roofing	Shelter, Protection, Shielding	3.5.2.11
Steel Embedments	Structural Support	3.5.1.20, 3.5.1.22
Steel Embedments (Dresden only)	Non-S/R Structural Support	3.5.1.22
Steel Panels and Cabinets	Structural Support	3.5.1.20, 3.5.1.22
Steel Panels and Cabinets (Quad Cities only)	Non-S/R Structural Support	3.5.1.20
Steel Plates (Dresden only)	Direct Flow	3.5.1.22

Table 2.4-11 Component Groups Requiring Aging Management Review - Crib House (Continued)

Component	Component Intended Function	Aging Management Ref
Steel Sump Screens (Quad Cities only)	Non-S/R Structural Support	3.5.1.22
Structural Steel	Non-S/R Structural Support	3.5.1.22
Walls, Ceilings, Floors	Fire Barrier	3.3.1.28

Aging management review results for the crib house are provided in Sections 3.3 and 3.5.

2.4.12 Unit 1 Crib House (Dresden Only)

Purpose of the Structure

The Unit 1 crib house at Dresden supports a diesel-driven fire pump, which is required to support the Unit 2 and 3 fire protection system.

Description of the Structure

The diesel-driven fire pump assembly is located on a reinforced concrete floor slab and takes its suction from the center bay of the Unit 1 crib house. The diesel engine is supported by a reinforced concrete pedestal and anchored by cast-in-place anchor bolts. The fire pump support consists of a steel leveling/bearing plate, on grout, with cast-in-place anchor bolts. The anchor bolts and the steel leveling bearing plate are evaluated in the component support commodity group.

UFSAR References

Dresden Station UFSAR Section(s): 9.2.2

Quad Cities Station UFSAR Section(s): Not Applicable

License Renewal Boundary Diagram References

Dresden Station: LR-DRE-M-1

Structure Intended Functions

Credited in regulated events - provides physical support for components relied upon to demonstrate compliance with the Fire Protection regulated events.

Component Groups Requiring Aging Management Review

Table 2.4-12 Component Groups Requiring Aging Management Review - Unit 1 Crib House (Dresden Only)

Component	Component Intended Function	Aging Management Ref
Concrete & Grout (Dresden only)	Non-S/R Structural Support	3.5.1.29
Concrete Slabs (Dresden only)	Structural Support	3.5.1.22
Concrete Slabs (Dresden only)	Non-S/R Structural Support	3.5.1.22
Walls, Ceilings, Floors (Dresden only)	Fire Barrier	3.3.1.28

Aging management review results for the Unit 1 crib house are provided in Sections 3.3 and 3.5.

2.4.13 Station Chimney

Purpose of the Structure

The station chimney provides an elevated discharge point for treated gaseous radioactive effluents.

Description of the Structure

The chimney is a 310 foot tall tapered structure that contains and/or directs the release of fission products. The reinforced concrete chimney is founded on bedrock. The lower section of the chimney is divided into 5 cells, comprised of reinforced concrete walls that provide a holdup volume for the gland exhausters.

UFSAR References

Dresden Station UFSAR Section(s): 11.3

Quad Cities Station UFSAR Section(s): 11.3

License Renewal Boundary Diagram References

Dresden Station: LR-DRE-M-1

Quad Cities Station: LR-QDC-M-1

Structure Intended Functions

Elevated release - provides for the discharge of treated gaseous waste to meet the requirements of 10 CFR 100.

Pressure control path - provides a secondary pressure control path for primary containment.

Component Groups Requiring Aging Management Review

Table 2.4-13 Component Groups Requiring Aging Management Review - Station Chimney

Component	Component Intended Function	Aging Management Ref
Caulking/Sealants	Gaseous Release Path	3.5.2.4
Concrete Slabs	Structural Support	3.5.1.20
Concrete Walls	Structural Support	3.5.1.20
Concrete Walls	Gaseous Release Path	3.5.1.20
Foundations	Structural Support	3.5.1.20, 3.5.1.21, 3.5.1.25, 3.5.1.26

Table 2.4-13 Component Groups Requiring Aging Management Review - Station Chimney (Continued)

Component	Component Intended Function	Aging Management Ref
Masonry Walls (Quad Cities only)	Structural Pressure Barrier	3.5.2.9
Masonry Walls (Quad Cities only)	Gaseous Release Path	3.5.2.9
Misc. Steel (Includes Platforms, Ladders, Railings)	Non-S/R Structural Support	3.5.1.20
Steel Doors (Dresden only)	Gaseous Release Path	3.5.1.20
Steel Embedments	Structural Support	3.5.1.20
Steel Plates	Gaseous Release Path	3.5.1.20
Structural Steel	Non-S/R Structural Support	3.5.1.20

Aging management review results for the station chimney are provided in Section 3.5.

2.4.14 Cranes and Hoists

Purpose of the Structure

Cranes and hoists provide systems for lifting, transporting and handling of loads.

Description of the Structure

Cranes and hoists include those cranes and hoists, whose failure could affect safety related components, except for the refueling bridge platform, which is covered with refueling equipment. Cranes and hoists include the reactor building crane, the turbine building cranes, smaller capacity cranes and hoists, and jib cranes that are located in various parts of the reactor and turbine buildings. Cranes and hoists and jib cranes are classified as Safety Class II components.

The reactor building crane services the operating floor, which is shared by both units. It is a bridge-type crane equipped with a 125-ton main hoist and a 9-ton auxiliary hoist and can reach major component storage areas on the operating floor. The reactor building crane is used for lifting and transporting the spent fuel cask between the spent fuel pools and the cask decontamination work area and handling other equipment and reactor components accessible from the refueling floor. The crane hoist system consists of a dual load path through the hoist gear train, the reeving system, and the hoist load block along with restraints at critical points to provide load retention and minimization of uncontrolled motions of the load in the event of failure of any single hoist component. Redundancy has also been designed into the hoist, trolley brakes, the spent fuel cask lifting devices, and crane control components.

The two turbine building overhead cranes are equipped with a 175-ton hoist, with a 25-ton auxiliary hoist (for the south crane at Quad Cities and the west crane at Dresden) and a 125-ton hoist with a 10-ton auxiliary hoist (for the north crane at Quad Cities and the east crane at Dresden).

UFSAR References

Dresden Station UFSAR Section(s): 9.1.4

Quad Cities Station UFSAR Section(s): 9.1.4

License Renewal Boundary Diagram References

Dresden Station: None

Quad Cities Station: None

Structure Intended Functions

Lifting and transporting loads - provide a safe means for handling safety related components and loads above or near safety related components.

Component Groups Requiring Aging Management Review

Table 2.4-14 Component Groups Requiring Aging Management Review - Cranes and Hoists

Component	Component Intended Function	Aging Management Ref
Cranes	Structural Support	3.3.1.3
Cranes	Non-S/R Structural Support	3.3.1.14
Rails	Non-S/R Structural Support	3.3.1.14

Aging management review results for cranes and hoists are provided in Section 3.3.

2.4.15 Component Supports Commodity Group

Commodity Group Description

The component support commodity group consists of support members (includes support members, welds, bolted connections and support anchorage to building structures), high strength bolting for Class I supports, and miscellaneous supports (includes constant aand variable load springs, guides, stops, slidiing surfaces, design clearances, vibration isolators and clevis pins). Grout (which includes reinforced concrete, grout and masonry) is evaluated as a component group within structures. The compoonent supports commodity group includes:

- Supports for ASME Class I, 2 and 3 piping and components
- Supports for ASME Class MC components including suppression chamber seismic restraints, suppression chamber support saddles and columns, and vent system supports
- Supports for cable trays, conduit, HVAC ducts, tube track, instrument tubing and non-ASME piping and components
- Anchorage of racks, panels, cabinets and enclosures for electrical equipment and instrumentation
- Supports for emergency diesel generator, HVAC system components, and miscellaneous mechanical equipment
- Supports for platforms, pipe whip restraints, jet impingement shields, masonry walls, and miscellaneous structures

Component Groups Requiring Aging Management Review

Table 2.4-15 Component Groups Requiring Aging Management Review - Component Supports

Component	Component Intended Function	Aging Management Ref
Anchorage to Buildings, Including Bolted/Welded Connections	Structural Support	3.5.1.29, 3.5.1.30
Anchorage to Buildings, Including Bolted/Welded Connections	Non-S/R Structural Support	3.5.1.29
Bolting	Structural Support	3.5.1.32
Clevis Pins: Suppression chamber Columns, Vent Systems, ESF Lines	Structural Support	3.5.2.5
Instrument Racks, Frames, Panels, Etc,	Structural Support	3.5.1.29
Instrument Racks, Frames, Panels, Etc,	Non-S/R Structural Support	3.5.1.29

Table 2.4-15 Component Groups Requiring Aging Management Review - Component Supports (Continued)

Component	Component Intended Function	Aging Management Ref
Raceways	Structural Support	3.5.1.29
Sliding Surfaces	Structural Support	3.5.1.31
Support Members (Includes Spring Hangers)	Structural Support	3.2.2.79, 3.2.2.80, 3.2.2.81, 3.5.1.29, 3.5.1.31, 3.5.2.14
Support Members	Non-S/R Structural Support	3.5.1.29
Vibration Isolation Elements (Quad Cities only)	Structural Support	3.5.1.29

Aging management review results for component supports are provided in Sections 3.2 and 3.5.

2.4.16 Insulation Commodity Group

Commodity Group Description

The insulation commodity group consists of the following categories of insulation:

- Mirror insulation inside containment
- Insulation and jacketing inside containment
- Insulation and jacketing outside containment
- Asbestos insulation outside containment
- Anti-sweat insulation outside containment
- Outdoor insulation and jacketing

Plant areas where systems and equipment in the scope of license renewal require temperature control include inside the drywell, the ECCS pump rooms, the outboard MSIV rooms and on outdoor heat-traced piping for freeze protection. Plant areas where insulation jacketing is subjected to periodic wetting is limited to outdoor heat-traced piping.

Insulation materials in use at the stations include both originally installed materials and replacement materials. These include metallic reflective insulation, asbestos, fiberglass batts, calcium silicate, quilted fiberglass blankets, preformed fiberglass and closed cell foam. Outdoor insulation installed over electric heat tracing consists of either calcium silicate or preformed fiberglass with aluminum jacketing.

Insulation requiring aging management consists of asbestos and fiberglass batt insulation located in the drywell, ECCS pump rooms, outboard MSIV rooms, and outdoor insulation and jacketing installed over heat-traced piping.

Component Groups Requiring Aging Management Review

Table 2.4-16 Component Groups Requiring Aging Management Review – Insulation

Component	Component Intended Function	Aging Management Ref
Insulation	Insulating Characteristics	3.2.2.44, 3.2.2.45, 3.2.2.46, 3.2.2.47, 3.3.2.122, 3.4.2.22
Insulation Jacketing	Insulation Jacket Integrity	3.2.2.48, 3.3.2.123, 3.4.2.23

Aging management review results for insulation are provided in Sections 3.2, 3.3, and 3.4.

2.5 SCOPING AND SCREENING RESULTS: ELECTRICAL AND INSTRUMENTATION AND CONTROLS SYSTEMS

The guidance provided in NEI 95-10, Appendix B was used to define passive, long-lived electrical commodities subject to aging management review. After applying the scoping and screening criteria discussed in Section 2.1.2, the electrical commodities requiring aging management reviews were determined.

Passive and long-lived electrical and instrumentation and controls component groups were evaluated using the plant “spaces” approach, whereby aging effects are identified, and bounding environmental conditions were used to evaluate the identified aging effect(s) with respect to component function.

The License Renewal Application is a joint application and discussions are applicable to both Dresden, Units 2 and 3 and Quad Cities, Units 1 and 2 unless otherwise noted. Clear statements (such as Dresden only or Quad Cities only) are made if discussions are applicable to only one station.

The following electrical commodity groups were determined to require aging management review.

- Cables and connections (splices, connectors, fuse blocks, and terminal blocks)
- Bus ducts
- High voltage transmission conductors and insulators
- Electrical penetrations

This section presents the results of the system scoping and screening processes for electrical commodities and provides the following information:

- A general description of the commodity,
- A listing of the components or component groups that require aging management review, associated component intended functions and a reference to the Section 3.6 Table that provides the aging management review results.

The interface of electrical and instrumentation and controls components with other types of components and the assessments of these interfacing components are provided in the appropriate mechanical or structural sections. For example, the assessment of electrical racks, panels, frames, cabinets, cable trays, conduit, and their supports is provided in the structural assessment documented in Section 2.4.

2.5.1 Electrical Commodities

2.5.1.1 Cables and Connections

Using the "spaces" approach, all electrical insulated cables and connections were evaluated for aging management based on the comparison of material property capability with environmental conditions. As appropriate, electrical cables and connections were excluded from aging management if they were identified as feeding an electrical component that performed no license renewal intended function.

The function of insulated cables and connections is to electrically connect specified sections of an electrical circuit to deliver voltage, current or signals. Electrical cables and their connections are reviewed as commodity groups. The types of connections included in this review are splices, connectors, fuse blocks, and terminal blocks.

2.5.1.2 Bus Ducts

The bus ducts within the scope of license renewal are the bus ducts used for safety related systems and those associated with the 4160V power feeds between the reserve auxiliary transformers (RATs) and switchgear. The bus ducts utilize pre-assembled raceway (enclosure) design with conductors supported by electrical insulators.

The function of the bus ducts is to electrically connect power supplies and load centers to deliver voltage and current.

The function of bus duct insulators is to support and insulate the bus bar conductors.

2.5.1.3 High Voltage Transmission Conductors and Insulators

The high voltage transmission conductors and insulators within scope of license renewal rule are those associated with the power feeds from the switchyard to reserve auxiliary transformers (RATs).

The function of the high voltage transmission conductors is to supply power to the plant systems through the reserve auxiliary transformers (RATs).

The function of high voltage insulators is to support and insulate the high voltage transmission conductors.

2.5.1.4 Electrical Penetrations

Electrical penetrations perform the functions of primary containment boundary (pressure integrity) and electrical continuity across the primary containment boundary. All primary containment electrical penetrations are included in the scope of the Rule. The electrical continuity function for penetrations is managed under the Environmental Qualification (EQ) Program which is discussed in Section 4.4, Environmental Qualification of Electrical Equipment (EQ). The pressure boundary function of every primary containment electrical penetration is evaluated in Section 2.4.1, Primary Containment, and the aging management program is referenced in Section 3.5.1.3.

2.5.1.5 Electrical Equipment Subject to 10 CFR 50.49 Environmental Qualification (EQ) Requirements

Electrical equipment subject to 10 CFR 50.49 environmental qualification (EQ) requirements are managed under the environmental qualification (EQ) program which is discussed in Section 4.4, Environmental Qualification of Electrical Equipment (EQ).

Component Groups Requiring Aging Management Review

Table 2.5-1 Component Groups Requiring Aging Management Review – Electrical Commodities

Component	Component Intended Function	Aging Management Ref
Connectors	Electrical Continuity	3.6.1.2
Electrical Cables	Electrical Continuity	3.6.1.2, 3.6.1.3, 3.6.1.4
Electrical Equipment Subject to 10 CFR 50.49 (EQ) Requirements	Electrical Continuity	3.6.1.1
Electrical Equipment Subject to 10 CFR 50.49 (EQ) Requirements	Insulate	3.6.1.1
Fuse Blocks	Electrical Continuity	3.6.1.2
High Voltage Transmission Conductors	Electrical Continuity	3.6.2.1
Insulators	Insulate	3.6.2.2, 3.6.2.3
Splices	Electrical Continuity	3.6.1.2
Terminal Blocks	Electrical Continuity	3.6.1.2

Aging management review results for electrical and instrumentation and controls components are provided in Section 3.6.

CHAPTER 3
AGING MANAGEMENT REVIEW RESULTS

List of Tables

3.0 AGING MANAGEMENT REVIEW RESULTS

10 CFR 54.21 (a)(3) requires a demonstration that the effects of aging will be adequately managed so that the intended functions will be maintained consistent with the current licensing basis throughout the period of extended operation. This section is provided in accordance with the requirements of 10 CFR 54.21 (a)(3). This section also discusses the layout of this application relative to aging management review results.

Aging Management Reviews included in NUREG-1801

Aging management reviews were performed by determining the aging effects of components or component groups, manufactured from similar materials, residing in similar environments. Aging management programs or activities were assigned to manage the applicable aging effect. Once this evaluation was completed, each component group, material, and environment was compared to the aging management review results performed in NUREG-1801. This evaluation was performed at the line-item level presented in NUREG-1801, Volume 2.

The NUREG-1801 line-item review was performed to assure that the component groups, materials, environments and aging effects referenced in NUREG –1801 are applicable to Dresden and Quad Cities and that the aging management program described in NUREG-1801 is appropriate.

In addition, a separate review of the aging management program and activities was performed to determine if the referenced program or activity contains the same program elements that are presented in Chapters X and XI of NUREG-1801. The results of this program element review are presented in Appendix B of this application.

The aging management review results for component groups, material, aging effects and aging mechanisms combinations that are consistent with evaluations performed in NUREG-1801 are provided in Tables 3.X-1. These tables are provided in the format and content described in Chapter 3 of NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” for all BWR applicable lines.

The overall conclusion provided in the discussion column of Tables 3.X-1 is a combination of each of these individual reviews, as discussed below.

Each line of the 3.X-1 Tables provides a discussion column that identifies:

- Consistent with NUREG-1801

  Identifies aging management reviews that are consistent with the NUREG-1801. This means that the component group, material and environment are applicable, the aging effect and aging mechanisms identified require management, the aging management program identified is appropriate, and the review results of the key elements provided in Appendix B concludes that the program elements are consistent with those elements provided in Chapters X and IX of NUREG-1801. In these cases, the discussion states – "Consistent with NUREG-1801."

- Exceptions to NUREG-1801

  Identifies those evaluations which are an exception to the NUREG-1801 aging effects or aging management program or activity and provides reference to the section in Section 3 and/or Appendix B which provides further explanation and justification. In these cases, the exception refers to all of the NUREG-1801 line items. In these cases, the discussion states – "Exception to NUREG-1801."

- Consistent to NUREG-1801, with exception

  Identifies those evaluations where several of the individual NUREG-1801 line items may be evaluated as consistent with NUREG-1801 while other line items in the same NUREG-1800 line may be evaluated as exception to NUREG-1801. In these cases, the discussion states – "Consistent with NUREG-1801 with exception identified in Section 3.X.X and or Appendix B."

- Further evaluations recommended by NUREG-1801

  Identifies and provides reference to sections providing further evaluation of aging management recommended by NUREG-1801

- Notes to clarify

  Provides notes to clarify NUREG-1801 Volume 2 line items such as those that are not in scope of license renewal or not installed at Dresden or Quad Cities

Plant-specific aging management activities recommended by NUREG-1801 are identified in the Aging Management Program column in each of the 3.X-1 Tables by including the plant-specific program in parentheses.

Aging Management Reviews not included in NUREG-1801

Aging management review results are provided in Tables 3.X-2 for components and component groups made of materials not included in NUREG-1801 or aging effects and aging mechanisms not identified in NUREG-1801. These tables provide the results of the aging management review and identify the component group, material, environment, aging effects requiring management, and the aging management program credited for managing the aging effect.

Component, Material, and Environment Considerations

Component Group

Components were grouped together to perform the aging management reviews. Component groups used in aging management reviews are discussed in further detail in the beginning of each 3.X section.

Materials

Components made of materials not included in NUREG-1801 were evaluated separately and the results of the aging management reviews are provided in Tables 3.X-2.

Environment

The aging management reviews for components and structures were evaluated based upon component groupings in common environments. Environments used are consistent with the environments presented in NUREG-1801.

Operating Experience

A review of plant-specific operating experience was conducted to identify aging effects requiring management. Industry-wide operating experience since the preparation of NUREG-1801 was also reviewed to identify aging effects requiring management. These reviews concluded that the aging effects identified by plant-specific and industry-wide operating experience were consistent with those identified in NUREG-1801. On-going review of plant-specific and industry operating experience is performed in accordance with corrective action programs and operating experience programs.

3.1 AGING MANAGEMENT OF REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

The following systems are evaluated as part of the reactor vessel, internals, and reactor coolant system:

- Reactor vessel and internals
- Recirculation system, recirculation flow control, and MG sets
- Reactor vessel head vent system
- Nuclear boiler instrumentation system
- Head spray system (Dresden only)
- Reactor coolant pressure boundary components of the following systems:

  High pressure coolant injection system

  Core spray system

  Reactor core isolation cooling system (Quad Cities only)

  Isolation condenser (Dresden only)

  Residual heat removal system (Quad Cities only)

  Low pressure coolant injection system (Dresden only)

  Standby liquid control system

  Shutdown cooling system (Dresden only)

  Control rod drive hydraulic system

  Reactor water cleanup system

  Main steam system

  Feedwater system

Components Evaluated Consistent with NUREG-1801

The components or component groups requiring aging management review in each of the reactor vessel, internals, and reactor coolant systems listed above are presented in Chapter 2. Aging management reviews were performed to assure that the component groups, materials, environments and aging effects referenced in NUREG-1801 are applicable to Dresden and Quad Cities and that the aging management program described in NUREG-1801 is appropriate.

When the components or component group and associated evaluations corresponded with an individual line item in NUREG-1801, Volume 2, the component aging management results were included in the appropriate line items in Table 3.1-1 “Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the reactor vessel, internals, and reactor coolant system“ for the reactor vessel, internals, and reactor coolant system. Each line in Table 3.1-1 “Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the reactor vessel, internals, and reactor coolant system“ matches a line in Chapter 3 of NUREG-1800, that is applicable to Boiling Water Reactors (BWR) or to both Boiling Water Reactors and Pressurized Water Reactors (BWR/PWR).

Not all component types in the Dresden and Quad Cities reactor vessel, internals, and reactor coolant system are listed in NUREG-1801, Volume 2. However, the aging management reviews presented in NUREG-1801, Volume 2 were applied to additional component types if the following criteria were satisfied:

- Constructed of the same material as components in the NUREG-1801 line item,
- Assigned the same Component Intended Function as components in the NUREG-1801 line item,
- Located in the same environment as components in the NUREG-1801 line item,
- Have exhibited the same aging effects identified in the NUREG-1801 line item, and

Component types meeting these criteria have been included in the presentation of aging management review results in Table 3.1-1 “Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the reactor vessel, internals, and reactor coolant system.” The third column of the table shows the component types included in each evaluation line. “NUREG-1801 Components” are those that correspond exactly with component types in NUREG-1801, Volume 2. “Evaluated with NUREG-1801 Components” shows the component types that meet the four (4) criteria above, and therefore share the same evaluation characteristics. Dresden and Quad Cities do not have all of the component types identified in NUREG-1801, Volume 2. The component type entries in the third column of Table 3.1-1 “Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the reactor vessel, internals, and reactor coolant system“ under the “NUREG-1801 Components” heading are those components or component groups that were identified and evaluated at Dresden and Quad Cities.

Other Components Evaluated

Table 3.1-2 “Aging management review results for the reactor vessel, internals, and reactor coolant system that are not addressed in NUREG-1801” presents the aging management review results for the remainder of the reactor vessel, internals, and reactor coolant system components. These entries result from aging management review results where the component type, material, environment or aging effect/mechanism differs from NUREG-1801, Volume 2 line item entries. Each line in Table 3.1-2 includes a line reference number, component group, material, environment, aging effect/mechanism, aging management program and discussion.

3.1.1 Aging management programs evaluated in NUREG-1801 that are relied upon for license renewal for the reactor vessel, internals, and reactor coolant system

Table 3.1-1 “Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the reactor vessel, internals, and reactor coolant system“ shows the component groups and aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the reactor vessel, internals, and reactor coolant system.

Table 3.1-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the reactor vessel, internals, and reactor coolant system

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1.1	Reactor coolant pressure boundary components	NUREG-1801 Components Closure Bolting Core Plates Core Spray Lines and Spargers Incore Instrumentation Dry Tubes and Guide Tubes Jet Pump Assemblies (Does not include Sensing Lines) Nozzle Safe Ends Nozzles Orificed Fuel Supports Penetrations Piping and Fittings Pumps Support Skirts and Attachment Welds Top Guides Top Head Enclosure (Head Flanges) Valves Vessel Bottom Heads Vessel Shells	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Further Evaluation of cumulative fatigue damage is described in Section 3.1.1.1.1, Section 4.3.1(reactor pressure vessel), Section 4.3.2 (reactor vessel internals), Section 4.3.3.1 (Dresden 3 RCPB piping), Section 4.3.3.2 (RCPB piping) and Section 4.3.4 (environmental effects of fatigue).

Table 3.1-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the reactor vessel, internals, and reactor coolant system (Continued)

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1.2	Isolation condenser	NUREG-1801 Components Isolation Condensers	Loss of material due to general, pitting, and crevice corrosion	Inservice inspection (B.1.1); water chemistry (B.1.2)	Yes, plant specific	Consistent with NUREG-1801, with exception. The exceptions to water chemistry and inservice inspection are described in Section 3.1.1.2.3. The exceptions to ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD are described in Section B.1.1. The exceptions to Water Chemistry are described in Section B.1.2. Further evaluation of Loss of Material due to Pitting and Crevice Corrosion BWR PWR is described in Section 3.1.1.1.2. Dresden only has isolation condenser.
3.1.1.3	Pressure vessel ferritic materials that have a neutron fluence greater than 10^{17} n/cm^2 (E>1 MeV)	NUREG-1801 Components Nozzles Vessel Shells	Loss of fracture toughness due to neutron irradiation embrittlement	TLAA, evaluated in accordance with Appendix G of 10 CFR 50 and RG 1.99	Yes, TLAA	Further evaluation of “Loss of Fracture Toughness due to Neutron Embrittlement” are described in Section 3.1.1.1.3 and Section 4.2.
3.1.1.4	Reactor vessel beltline shell and welds	NUREG-1801 Components Vessel Shells	Loss of fracture toughness due to neutron irradiation embrittlement	Reactor vessel surveillance (B.1.22)	Yes, plant specific	Consistent with NUREG-1801. Further evaluation of Loss of Fracture Toughness due to Neutron Irradiation Embrittlement BWR PWR is described in Section 3.1.1.1.4.

Table 3.1-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the reactor vessel, internals, and reactor coolant system (Continued)

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1.5	Small-bore reactor coolant system and connected systems piping	NUREG-1801 Components Piping and Fittings (small bore) Evaluated with NUREG-1801 Components Valves (small bore)	Crack initiation and growth due to SCC, intergranular SCC, and thermal and mechanical loading	Inservice inspection (B.1.1); water chemistry (B.1.2); one-time inspection (B.1.23)	Yes, parameters monitored/ inspected and detection of aging effects are to be further evaluated	Consistent with NUREG-1801, with exception. The exceptions to ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD are described in Section B.1.1. The exceptions to Water Chemistry are described in Section B.1.2. Further evaluation of Crack Initiation and Growth due to Thermal and Mechanical Loading or Stress Corrosion Cracking BWR PWR is described in Section 3.1.1.1.5.
3.1.1.6	Jet pump sensing line, and reactor vessel flange leak detection line	NUREG-1801 Components Pipes Valves	Crack initiation and growth due to SCC, inter-granular stress corrosion cracking (IGSCC), or cyclic loading	Plant specific	Yes, plant specific	Further evaluation of Crack Initiation and Growth due to Thermal and Mechanical Loading or Stress Corrosion Cracking BWR PWR is described in Section 3.1.1.1.6. Jet pump sensing lines are not in scope for license renewal at Dresden and Quad Cities. Dresden leak detection line is carbon steel and is evaluated as a non-NUREG-1801 item, Quad Cities leak detection line is stainless steel.

Table 3.1-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the reactor vessel, internals, and reactor coolant system (Continued)

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1.7	Isolation condenser	NUREG-1801 Components Isolation Condensers	Crack initiation and growth due to stress corrosion cracking (SCC) or cyclic loading;	Inservice inspection (B.1.1); water chemistry (B.1.2)	Yes, plant specific	Consistent with NUREG-1801, with exception. The exceptions to water chemistry and inservice inspection are described in Section 3.1.1.2.3. The exceptions to ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD are described in Section B.1.1. The exceptions to Water Chemistry are described in Section B.1.2. Further evaluation of Crack Initiation and Growth due to Thermal and Mechanical Loading or Stress Corrosion Cracking BWR PWR is described in Section 3.1.1.1.7. Dresden only has Isolation Condenser.
3.1.1.8	Reactor vessel closure studs and stud assembly	NUREG-1801 Components Top Head Enclosure (Closure Studs and Nuts)	Crack initiation and growth due to SCC and/or IGSCC	Reactor head closure studs (B.1.3)	No	Consistent with NUREG-1801, with exception. The exceptions to Reactor Head Closure Studs are described in Section B.1.3.
3.1.1.9	CASS pump casing and valve body	NUREG-1801 Components Pumps Valves	Loss of fracture toughness due to thermal aging embrittlement	Inservice inspection (B.1.1)	No	Consistent with NUREG-1801, with exception. The exceptions to ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD are described in Section B.1.1.
3.1.1.10	CASS piping	See Discussion Column	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No	CASS piping does not exist at Dresden or Quad Cities.

Table 3.1-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the reactor vessel, internals, and reactor coolant system (Continued)

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1.11	BWR piping and fittings; steam generator components	NUREG-1801 Components Piping and Fittings Valves Evaluated with NUREG-1801 Components Filters/Strainers Flow Elements Valves	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion (B.1.11)	No	Consistent with NUREG-1801, with exception. The exceptions to flow accelerated corrosion are described in Section 3.1.1.2.2.
3.1.1.12	Reactor coolant pressure boundary (RCPB) valve closure bolting, manway and holding bolting, and closure bolting in high pressure and high temperature systems	NUREG-1801 Components Closure Bolting Flanges	Loss of material due to wear; loss of preload due to stress relaxation; crack initiation and growth due to cyclic loading and/or SCC	Bolting integrity (B.1.12)	No	Consistent with NUREG-1801, with exception. The exceptions to loss of preload aging degradation are described in Section 3.1.1.2.1. The exceptions to Bolting Integrity are described in Section B.1.12.
3.1.1.13	Feedwater and control rod drive (CRD) return line nozzles	NUREG-1801 Components Nozzles	Crack initiation and growth due to cyclic loading	Feedwater nozzle (B.1.5); CRD return line nozzle (B.1.6)	No	Consistent with NUREG-1801, with exception. The exceptions to BWR Control Rod Drive Return Line Nozzle are described in Section B.1.6.

Table 3.1-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the reactor vessel, internals, and reactor coolant system (Continued)

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1.14	Vessel shell attachment welds	NUREG-1801 Components Vessel Shell Attachment Welds	Crack initiation and growth due to SCC, IGSCC	BWR vessel ID attachment welds (B.1.4); water chemistry (B.1.2)	No	Consistent with NUREG-1801, with exception. The exceptions to Water Chemistry are described in Section B.1.2. The exceptions to BWR Vessel ID Attachment Welds are described in Section B.1.4.
3.1.1.15	Nozzle safe ends, recirculation pump casing, connected systems piping and fittings, body and bonnet of valves	NUREG-1801 Components Nozzle Safe Ends Piping and Fittings Pumps Valves Evaluated with NUREG-1801 Components Dampeners Flow Elements Tanks Thermowells Tubing	Crack initiation and growth due to SCC, IGSCC	BWR stress corrosion cracking (B.1.7); water chemistry (B.1.2)	No	Consistent with NUREG-1801, with exception. The exceptions to Water Chemistry are described in Section B.1.2. The exceptions to BWR Stress Corrosion Cracking are described in Section B.1.7.
3.1.1.16	Penetrations	NUREG-1801 Components Penetrations Penetrations (Control Rod Drive Stub Tubes)	Crack initiation and growth due to SCC, IGSCC, cyclic loading	BWR penetrations (B.1.8); water chemistry (B.1.2)	No	Consistent with NUREG-1801, with exception. The exceptions to Water Chemistry are described in Section B.1.2. The exceptions to BWR Penetrations are described in Section B.1.8.

Table 3.1-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the reactor vessel, internals, and reactor coolant system (Continued)

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1.17	Core shroud and core plate, support structure, top guide, core spray lines and spargers, jet pump assemblies, control rod drive housing, nuclear instrumentation guide tubes	NUREG-1801 Components Control Rod Drive Housings Control Rod Guide Tubes Core Plates and Bolts Core Shrouds (Upper, Central, Lower) Core Spray Lines and Spargers Incore Instrumentation Dry Tubes and Guide Tubes Jet Pump Assemblies (Does not include Sensing Lines) Reactor Internals Modification/Repair Hardware Shroud Support Structures Top Guides Evaluated with NUREG-1801 Components Orificed Fuel Support Pieces	Crack initiation and growth due to SCC, IGSCC, IASCC	BWR vessel internals (B.1.9); water chemistry (B.1.2)	No	Consistent with NUREG-1801, with exception. The exceptions to Water Chemistry are described in Section B.1.2. The exceptions to BWR Vessel Internals are described in Section B.1.9. LPCI couplings are not used at Dresden or Quad Cities.

Table 3.1-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the reactor vessel, internals, and reactor coolant system (Continued)

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.1.1.18	Core shroud and core plate access hole cover (welded and mechanical covers)	NUREG-1801 Components Access Hole Covers (Mechanical) Access Hole Covers (Welded)	Crack initiation and growth due to SCC, IGSCC, IASCC	ASME Section XI inservice inspection (B.1.1); water chemistry (B.1.2)	No	Consistent with NUREG-1801, with exception. The exceptions to ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD are described in Section B.1.1. The exceptions to Water Chemistry are described in Section B.1.2. Dresden only has welded access hole covers.
3.1.1.19	Jet pump assembly castings; orificed fuel support	NUREG-1801 Components Jet Pump Assemblies (Does not include Sensing Lines) Orificed Fuel Support Pieces	Loss of fracture toughness due to thermal aging and neutron embrittlement	Thermal aging and neutron irradiation embrittlement (B.1.10)	No	Consistent with NUREG-1801.
3.1.1.20	Unclad top head and nozzles	See Discussion Column	Loss of material due to general, pitting, and crevice corrosion	Inservice inspection (B.1.1); water chemistry (B.1.2)	No	Dresden and Quad Cities have cladded top head enclosure and nozzles, therefore this aging effect is not applicable.

3.1.1.1 Further evaluation of aging management as recommended by NUREG-1801 for the reactor vessel, internals, and reactor coolant system

3.1.1.1.1 Cumulative Fatigue Damage BWR/PWR (NUREG-1800, Section 3.1.2.2.1)

Cumulative fatigue damage of the reactor pressure vessel, reactor vessel internals, and reactor coolant pressure boundary components is a TLAA as defined in 10 CFR 54.3. Cumulative fatigue damage of reactor pressure vessel, reactor vessel internals, and reactor coolant pressure boundary components is required to be evaluated in accordance with 10 CFR 54.21(c)(1). The TLAA evaluation cumulative fatigue damage for the reactor pressure vessel, reactor vessel internals, and reactor coolant pressure boundary components is addressed as follows:

- Reactor vessel fatigue analysis is addressed in Section 4.3.1.
- The only reactor vessel internals components with a thermal fatigue analysis are the Quad Cities core shroud and shroud repair hardware. Fatigue analysis of the reactor vessel internals is addressed in Section 4.3.2.
- Dresden Unit 3 RCPB piping with an ASME Section III Class 1 fatigue analysis is addressed in Section 4.3.3.1.
- Fatigue of RCPB piping designed in accordance with USAS B31.1 is addressed in Section 4.3.3.2.
- Environmental effects of fatigue are addressed in Section 4.3.4.
- Recirculation pumps are designed to ASME Section III Class C (1965). ASME Section III Class C (1965) does not require evaluation of thermal or pressure cycle fatigue.
- No fatigue analysis exists for RCPB valves or valve closure bolting.

3.1.1.1.2 Loss of Material due to Pitting and Crevice Corrosion BWR/PWR (NUREG-1800, Section 3.1.2.2.2.2)

NUREG-1801 indicates that ASME Section XI inservice inspections (ISI), control of isolation condenser water chemistry and specified augmentation activities are adequate to mitigate loss of material and to provide detection of aging effects. NUREG-1801 specifies that aging management program XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," should be augmented to detect loss of material due to pitting or crevice corrosion. The plant-specific aging management program Heat Exchanger Test and Inspection Activities (B.2.6) will augment aging management program ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.1.1.) The heat exchanger test and inspection activities include temperature and radioactivity monitoring of the shell side water, and eddy current testing of the tubes to ensure intended function is maintained.

3.1.1.1.3 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement BWR/PWR (NUREG-1800, Section 3.1.2.2.3.1)

Evaluation of loss of fracture toughness is a TLAA as defined in 10 CFR 54.3. Loss of fracture toughness for the reactor pressure vessel shell is evaluated in accordance with 10 CFR 54.21(c)(1). Fluence at the reactor pressure vessel nozzles is less than $1x10^{17}$ n/cm^2 at the end of the license renewal term. Therefore reactor pressure vessel nozzles are not evaluated for loss of fracture toughness due to neutron embrittlement. The TLAA evaluation of the loss of fracture toughness for the reactor pressure vessel shell is addressed in Section 4.2. Loss of fracture toughness for the reactor pressure vessel shell and nozzles will be managed with the Reactor Vessel Surveillance Program (B.1.22.)

3.1.1.1.4 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement BWR/PWR (NUREG-1800, Section 3.1.2.2.3.2)

Loss of fracture toughness due to neutron irradiation embrittlement will be managed using aging management program Reactor Vessel Surveillance (B.1.22.) The proposed withdrawal schedule will be provided prior to the period of extended operation.

3.1.1.1.5 Crack Initiation and Growth due to Thermal and Mechanical Loading or Stress Corrosion Cracking BWR/PWR (NUREG-1800, Section 3.1.2.2.4.1)

An inspection of small bore reactor coolant piping is to be conducted in accordance with aging management program One-Time Inspection (B.1.23) to verify that service-induced weld cracking is not occurring in the small-bore piping less than 4 inches including pipe, fittings and branch connections. A review of the Dresden and Quad Cities Risk Informed Inservice Inspection (RISI) Evaluations on degradation mechanism assessment demonstrated that only Dresden had a high failure potential on a small bore pipe due to thermal fatigue. The inspection will consist of an ultrasonic exam on one of the two-inch drain lines off the Dresden main steam header. These lines are Class 1 and within the scope of License Renewal. The aging mechanisms cited by the report for these lines are thermal stratification, cycling, and stripping (TASCS), thermal transients (TT), and flow accelerated corrosion. RISI lists the reasons for susceptibility to TASCS and TT due to the potential for two phase flow and because hot steam line condensate will flow into the line that is at ambient temperature whenever the drain valve is open. Selection of these lines will also minimize exposure levels.

3.1.1.1.6 Crack Initiation and Growth due to Thermal and Mechanical Loading or Stress Corrosion Cracking BWR/PWR (NUREG-1800, Section 3.1.2.2.4.2)

The reactor vessel flange leak detection line at Quad Cities is a Class 2 line. The line is stainless steel and therefore, it is susceptible to cracking due to stress corrosion cracking and intergranular stress corrosion cracking. Quad Cities ISI Program, Relief Request PR-02 (relief granted per SER dated 9/15/95), proposes an alternate inspection of the reactor vessel flange leak detection line. The alternate examination utilizes a VT-2 visual examination on the line during vessel flood-up during a refueling outage.

3.1.1.1.7 Crack Initiation and Growth due to Thermal and Mechanical Loading or Stress Corrosion Cracking BWR/PWR (NUREG-1800, Section 3.1.2.2.4.3)

NUREG-1801 indicates that ASME Section XI inservice inspections (ISI) and control of isolation condenser water chemistry, and specified augmentation activities are adequate to mitigate cracking and to provide detection of aging effects. NUREG-1801 specifies that aging management program XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," should be augmented to detect cracking due to SCC or cyclic loading. The plant-specific aging management program Heat Exchanger Test and Inspection Activities (B.2.6) will augment aging management program ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.1.1.) The heat exchanger test and inspection activities include temperature and radioactivity monitoring of the shell side water, and eddy current testing of the tubes to ensure intended function is maintained.

3.1.1.2 Aging management programs or evaluations that are different than those described in NUREG-1801 for the reactor vessel, internals, and reactor coolant system

3.1.1.2.1 Exception for loss of preload aging degradation (GALL Items IV.C1.2-e and IV.C1.3-f)

EPRI 1003056, Rev. 3, "Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools," states that loss of preload mechanisms are typically addressed during installation and subsequent maintenance of closure bolting. As the loss of pre-load in a bolt is a design driven effect, it is not an applicable aging effect and does not require aging management.

3.1.1.2.2 Exception to NUREG-1801 for flow accelerated corrosion

Flow accelerated corrosion is an applicable aging mechanism for the main steam lines and feedwater lines. However, carbon steel components in the reactor vessel head vent, core spray, shutdown cooling (Dresden only), HPCI, RCIC (Quad Cities only), and nuclear boiler instrumentation (Quad Cities only) systems are not susceptible to flow accelerated corrosion and do not require aging management. This exception is based on the following:

1. EPRI NSAC-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program," allows an exclusion from flow accelerated corrosion for systems that operate less than 2% of plant operating time.

2. EPRI TR-114882, "Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools," states that flow rates less than 6 ft/sec do not need to be considered for FAC.

3. NUREG-1557, "Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal," states that erosion/corrosion in HPCI and the RCIC turbine steam supply piping is non-significant due to the low flow range.

4. Dresden and Quad Cities operate these systems less than 2% of plant operating time or at flow rates less than 6 ft/sec. Additionally, plant experience has not revealed flow accelerated corrosion in these lines.

3.1.1.2.3 Exception to GALL for XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD", for Class 1 components; and XI.M2 "Water Chemistry"; and Chapter XI.M1 AMP augmentation activities

NUREG-1801 lines IV.C1.4-a and IV.C1.4-b specify the use of ASME Section XI Inspection for Class 1 components; Water Chemistry controls; and additional augmentation activities to detect cracking or loss of material, and verify effectiveness of the program.

At Dresden, the isolation condenser is classified as ISI Class 2 on the tube side and ISI Class 3 on the shell side, so ISI Class 1 inspection requirements do not apply. VT-2 examinations of the reactor coolant pressure boundary are conducted during system pressure testing. The water chemistry controls specified in NUREG-1801, XI.M2 are not applied to the isolation condenser shell. However, the water chemistry controls specified in Water Chemistry (B.1.2) are applied to the makeup water supplied to the isolation condenser. Additionally, aging management program ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.1.1) is augmented with plant-specific Heat Exchanger Test and Inspection activities (B.2.6).

3.1.2 Components or aging effects that are not addressed in NUREG-1801 for the reactor vessel, internals, and reactor coolant system

Table 3.1-2, "Aging management review results for the reactor vessel, internals, and reactor coolant system that are not addressed in NUREG-1801" contains aging management review results for the reactor vessel, internals, and reactor coolant systems for component groups that are not addressed in NUREG-1801.

Table 3.1-2 Aging management review results for the reactor vessel, internals, and reactor coolant system that are not addressed in NUREG-1801

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.1.2.1	Closure Bolting	Low-Alloy Steel	Containment Nitrogen	Crack initiation and growth/ Cyclic loading	Bolting Integrity (B.1.12)	NUREG-1801 does not address closure bolting in a containment nitrogen environment.
3.1.2.2	Closure Bolting	Low-Alloy Steel	Containment Nitrogen	Loss of material/ Wear	Bolting Integrity (B.1.12)	NUREG-1801 does not address closure bolting in a containment nitrogen environment.
3.1.2.3	Component External Surfaces (piping and fittings, filters/strainers, valves, flow elements, restricting orifices)	Carbon Steel	Air, moisture, and humidity < 100°C (212°F)	Loss of material/ General corrosion	Bolting Integrity (B.1.12) and Structures Monitoring Program (B.1.30)	NUREG-1801 does not address carbon steel in a plant indoor environment for Reactor Coolant System components. Bolting Integrity (B.1.12) is used to manage piping and fittings, filter/strainers, valves, flow elements, and restricting orifices. In addition, Structures Monitoring Program (B.1.30) inspects piping adjacent to selected supports.
3.1.2.4	Component External Surfaces (piping and fittings, nozzles, penetrations, tubing, valves, top head enclosure head and nozzles, vessel shells)	Carbon Steel	Containment Nitrogen	None	None	NUREG-1801 does not address carbon steel components in a containment nitrogen environment. A nitrogen ambient environment is not conducive to promoting aging degradation.
3.1.2.5	Component External Surfaces (sight glasses)	Glass	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address sight glass components. A plant indoor environment is not conducive to promoting aging degradation of glass.
3.1.2.6	Component External Surfaces (sight glasses)	Glass	Containment Nitrogen	None	None	NUREG-1801 does not address sight glass components. A nitrogen environment is not conducive to promoting aging degradation of glass.

Table 3.1-2 Aging management review results for the reactor vessel, internals, and reactor coolant system that are not addressed in NUREG-1801

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.1.2.7	Component External Surfaces (filters/strainers, piping and fittings, dampeners, tubing, valves)	Stainless Steel	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address stainless steel in a plant indoor environment. Plant indoor environment is not an aggressive wetted environment conducive to promoting aging degradation of stainless steel components.
3.1.2.8	Component External Surfaces (piping and fittings, flow element, pumps, tanks, tubing, filters/strainers, restricting orifices, thermowells, valves)	Stainless Steel	Containment Nitrogen	None	None	NUREG-1801 does not address stainless steel in a containment nitrogen environment. A nitrogen environment is not conducive to promoting aging degradation.
3.1.2.9	Dampeners	Stainless Steel	Lubricating oil (with contaminants and/or moisture)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.1.2.10	Filters/Strainers	Stainless Steel Casting	Oxygenated water, up to 288°C (550°F)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address this environment for CRD filter located around CRD housing containing CST water. This also serves as a reactor coolant pressure boundary.
3.1.2.11	Filters/Strainers	Stainless Steel Casting	Oxygenated water, up to 288°C (550°F)	Crack initiation and growth/ Stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address this environment for CRD filters/strainers located around CRD housing containing CST water. This also serves as a reactor coolant pressure boundary.

Table 3.1-2 Aging management review results for the reactor vessel, internals, and reactor coolant system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.1.2.12	Filters/Strainers	Carbon Steel	Lubricating oil (with contaminants and/or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.1.2.13	Filters/Strainers	Stainless Steel	288°C (550°F) reactor coolant water	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address non-safety related components in a reactor coolant water.
3.1.2.14	Flow Elements	Carbon Steel	25-288°C (77-550°F) demineralized water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non-safety related components in a demineralized water environment.
3.1.2.15	Isolation Condensers	Tubes: stainless steel	Tube side: Steam; shell side: Demineralized water	Buildup of Deposit/Fouling	Heat Exchanger Test and Inspection Activities (B.2.6)	NUREG-1801 does not identify fouling as an applicable aging effect. Fouling is identified in EPRI/SANDIA, and is considered applicable due to construction and operating conditions.
3.1.2.16	Nozzles	SA508 Cl2 with or without Stainless Steel Cladding	288°C (550°F) steam	None	None	NUREG-1801 addresses cumulative fatigue damage (and embrittlement for the shell) as a TLAA. There are no other aging effects for these nozzles.
3.1.2.17	Nozzles	SA508 Cl2 with or without Stainless Steel Cladding	Up to 288°C (550°F) reactor coolant water	None	None	NUREG-1801 addresses cumulative fatigue damage (and embrittlement for the shell) as a TLAA. There are no other aging effects for these nozzles..

Table 3.1-2 Aging management review results for the reactor vessel, internals, and reactor coolant system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.1.2.18	NSR Vents or Drains, Piping and Valves	Carbon Steel, Stainless Steel, Brass or Bronze	Air, moisture, humidity, and leaking fluid	Loss of material/ Corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address NSR vents or drains in an air, moisture, humidity and leaking fluid environment.
3.1.2.19	Pipes	Carbon Steel	Leaking reactor coolant water and/or steam up to 288°C (550°F)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801, item IV A1.1-d aging effects do not apply to Dresden since vessel flange leak detection line is constructed of carbon steel.
3.1.2.20	Piping and Fittings	Carbon Steel	25-288°C (77-550°F) demineralized water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non-safety related components in a demineralized water environment.
3.1.2.21	Piping and Fittings	Carbon Steel	Lubricating oil (with contaminants and/or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.1.2.22	Piping and Fittings	Carbon Steel	Wet Gas	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components in a wet gas environment.

Table 3.1-2 Aging management review results for the reactor vessel, internals, and reactor coolant system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.1.2.23	Piping and Fittings	Stainless Steel	25-288°C (77-550°F) demineralized water	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address non-safety related components in a demineralized water environment.
3.1.2.24	Piping and Fittings	Stainless Steel	288°C (550°F) reactor coolant water	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address non-safety related components in a reactor coolant water environment.
3.1.2.25	Piping and Fittings	Stainless Steel	Oxygenated water, up to 288°C (550°F)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address this environment for CRD piping and fittings located around CRD housing containing CST water. This also serves as a reactor coolant pressure boundary.
3.1.2.26	Piping and Fittings	Stainless Steel	Oxygenated water, up to 288°C (550°F)	Crack initiation and growth/ Stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address this environment for CRD piping and fittings located around CRD housing containing CST water. This also serves as a reactor coolant pressure boundary.

Table 3.1-2 Aging management review results for the reactor vessel, internals, and reactor coolant system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.1.2.27	Restricting Orifices	Carbon Steel	Lubricating oil (with contaminants and/or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.1.2.28	Restricting Orifices	Stainless Steel	Oxygenated water, up to 288°C (550°F)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non-safety related components in an oxygenated water environment.
3.1.2.29	Restricting Orifices	Stainless Steel	Oxygenated water, up to 288°C (550°F)	Crack initiation and growth/ Stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address this environment for CRD restricting orifices located around CRD housing containing CST water. This also serves as a reactor coolant pressure boundary.
3.1.2.30	Sight Glasses	Glass	25-288°C (77-550°F) demineralized water	None	None	NUREG-1801 does not address glass components. Glass in this environment does not have any applicable aging effect.
3.1.2.31	Sight Glasses	Glass	Lubricating oil (with contaminants and/or moisture)	None	None	NUREG-1801 does not address glass components in an oil environment. An oil environment is not conducive to promoting aging degradation of glass components. Glass in this environment does not have any applicable aging effect.

Table 3.1-2 Aging management review results for the reactor vessel, internals, and reactor coolant system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.1.2.32	Sight Glasses	Glass	Wet Gas	None	None	NUREG-1801 does not address glass components in a wet gas environment. A wet gas environment is not conducive to promoting aging degradation of glass components. Glass in this environment does not have any applicable aging effect.
3.1.2.33	Support Skirts and Attachment Welds	SA-302 Gr B Welds Low Alloy Steel	Containment nitrogen	None	None	NUREG-1801 addresses cumulative fatigue damage as a TLAA. There are no other aging effects for support skirts and attachment welds. A nitrogen ambient environment is not conducive to promoting aging degradation.
3.1.2.34	Tanks	Stainless Steel	Oxygenated water, up to 288°C (550°F)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address this environment for CRD accumulator (tanks) containing CST water. This also serves as a reactor coolant pressure boundary.
3.1.2.35	Tanks	Stainless Steel	Oxygenated water, up to 288°C (550°F)	Crack initiation and growth/ Stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address this environment for CRD accumulator (tanks) located around CRD housing containing CST water. This also serves as a reactor coolant pressure boundary.
3.1.2.36	Top Head Enclosure (Head Flanges)	SA336 with or without Stainless Steel Cladding	288°C (550°F) steam	None	None	NUREG-1801 addresses cumulative fatigue damage as a TLAA. There are no other aging effects for the Top Head Enclosure head flanges.

Table 3.1-2 Aging management review results for the reactor vessel, internals, and reactor coolant system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.1.2.37	Top Head Enclosure (Top Heads)	SA302Gr B with Stainless Steel Cladding	288°C (550°F) steam	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.1.1), for Class 1 components; and Water Chemistry (B.1.2)	NUREG-1801 does not address cladded top head enclosure.
3.1.2.38	Tubing	Stainless Steel	25-288°C (77-550°F) demineralized water	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address non-safety related components in a demineralized water environment.
3.1.2.39	Tubing	Stainless Steel	Oxygenated water, up to 288°C (550°F)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non safety-related components in an oxygenated water environment.
3.1.2.40	Tubing	Stainless Steel	Oxygenated water, up to 288°C (550°F)	Crack initiation and growth/ Stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address non safety-related components in an oxygenated water environment.
3.1.2.41	Tubing	Stainless Steel	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address stainless steel components in a saturated air environment.

Table 3.1-2 Aging management review results for the reactor vessel, internals, and reactor coolant system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.1.2.42	Tubing	Stainless Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel piping components in a warm moist air environment.
3.1.2.43	Valves	Carbon Steel	25-288°C (77-550°F) demineralized water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non-safety related components in a demineralized water environment.
3.1.2.44	Valves	Carbon Steel	Leaking reactor coolant water and/or steam up to 288°C (550°F)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801, item IV A1.1-d aging effects do not apply to Dresden since vessel flange leak detection line is constructed of carbon steel.
3.1.2.45	Valves	Carbon Steel	Lubricating oil (with contaminants and/or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.1.2.46	Valves	Carbon Steel	Oxygenated water, up to 288°C (550°F)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel components in an oxygenated water environment.
3.1.2.47	Valves	Carbon Steel	Treated water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non-safety related components in a treated water environment.

Table 3.1-2 Aging management review results for the reactor vessel, internals, and reactor coolant system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.1.2.48	Valves	Carbon Steel	Wet Gas	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components in a wet gas environment.
3.1.2.49	Valves	Stainless Steel	288°C (550°F) reactor coolant water	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address non-safety related components in a reactor coolant water environment.
3.1.2.50	Valves	Stainless Steel	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address stainless steel valves in a saturated air environment.
3.1.2.51	Valves	Stainless Steel	Wet Gas	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components in a wet gas environment.
3.1.2.52	Valves	Stainless Steel Casting	Oxygenated water, up to 288°C (550°F)	Crack initiation and growth/ Stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address this environment for CRD valves located around CRD housing containing CST water. This also serves as a reactor coolant pressure boundary.

Table 3.1-2 Aging management review results for the reactor vessel, internals, and reactor coolant system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.1.2.53	Valves	Stainless Steel Casting	Oxygenated water, up to 288°C (550°F)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address this environment for CRD valves located around CRD housing containing CST water. This also serves as a reactor coolant pressure boundary.
3.1.2.54	Vessel Bottom Heads	SA302Gr B with 308 309 308L 309L Cladding and Low Alloy Steel Weldments	Up to 288°C (550°F) reactor coolant water	None	None	NUREG-1801 addresses cumulative fatigue damage (and embrittlement for the shell) as a TLAA. There are no other aging effects for the vessel bottom heads.
3.1.2.55	Vessel Shells	SA302Gr B with 308 309 308L 309L Cladding and Low Alloy Steel Weldments	288°C (550°F) steam	None	None	NUREG-1801 addresses cumulative fatigue damage (and embrittlement for the shell) as a TLAA. There are no other aging effects for the vessel shells.
3.1.2.56	Vessel Shells	SA336 with Stainless Steel Cladding	288°C (550°F) steam	None	None	NUREG-1801 addresses cumulative fatigue damage (and embrittlement for the shell) as a TLAA. There are no other aging effects for the vessel shells.
3.1.2.57	Vessel Shells	SA533Gr B with 308 309 308L 309L Cladding Low Alloy Steel Weldments	288°C (550°F) steam	None	None	NUREG-1801 addresses cumulative fatigue damage (and embrittlement for the shell) as a TLAA. There are no other aging effects for the vessel shells.
3.1.2.58	Penetrations Drain line	Carbon steel	Up to 288°C (550°F) reactor coolant water	None	None	NUREG-1801 addresses cumulative fatigue damage as a TLAA. There are no other aging effects for these penetrations.

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.1.2.59	Top Head Enclosure (Top Head Nozzles)	SA-508 C12 with Stainless Steel Cladding	288°C (550°F) steam	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.1.1), for Class 1 components; and Water Chemistry (B.1.2)	NUREG-1801 does not address cladded top head enclosure nozzles.
3.1.2.60	Nozzle Safe Ends	Carbon Steel	Up to 288°C (550°F) reactor coolant water	None	None	NUREG-1801 addresses cumulative fatigue damage as a TLAA. There are no other aging effects for these nozzles.
3.1.2.61	Vessel Shells	SA-302 Gr B with 308, 309, 308L, 309L and Low Alloy Steel Weldments	288°C (550°F) reactor coolant water $5X10^8$ - $5X10^9$ $n/cm^2 \bullet s$	None	None	NUREG-1801 addresses cumulative fatigue damage (and embrittlement for the shell) as a TLAA. There are no other aging effects for vessel shells.
3.1.2.62	Penetrations	SA-336 or SA-508 Cl 2	Up to 288°C (550°F) reactor coolant water	None	None	NUREG-1801 addresses cumulative fatigue damage as a TLAA. There are no other aging effects for these penetrations.
3.1.2.63	Penetrations	SB-166	Up to 288°C (550°F) reactor coolant water	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	BWR Penetrations (B.1.8); and Water Chemistry (B.1.2)	NUREG-1801 does not address SB-166 penetrations.

3.2. AGING MANAGEMENT OF ENGINEERED SAFETY FEATURES

The following systems are evaluated as part of the engineered safety features:

- High pressure coolant injection system
- Core spray system
- Containment isolation components and primary containment piping system
- Reactor core isolation cooling system (Quad Cities only)
- Isolation condenser (Dresden only)
- Residual heat removal system (Quad Cities only)
- Low pressure coolant injection (Dresden only)
- Standby gas treatment system
- Anticipated transients without scram system
- Automatic depressurization system (evaluated under main steam)

Components Evaluated Consistent with NUREG-1801

The components or component groups requiring aging management review in each of the engineered safety features systems listed above are presented in Chapter 2. Aging management reviews were performed to assure that the component groups, materials, environments and aging effects referenced in NUREG –1801 are applicable to Dresden and Quad Cities and that the aging management program described in NUREG-1801 is appropriate.

When the components or component group and associated evaluations corresponded with an individual line item in NUREG-1801, Volume 2, the component aging management results were included in the appropriate line items in Table 3.2-1, "Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the engineered safety features" for engineered safety features systems. Each line in Table 3.2-1, "Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the engineered safety features" matches a line in Chapter 3 of NUREG-1800, that is applicable to Boiling Water Reactors (BWR) or to both Boiling Water Reactors and Pressurized Water Reactors (BWR/PWR).

Not all component types in the Dresden and Quad Cities engineered safety features systems are listed in NUREG-1801, Volume 2. However, the aging management reviews presented in NUREG-1801, Volume 2 were applied to additional component types if the following criteria were satisfied:

- Constructed of the same material as components in the NUREG-1801 line item,
- Assigned the same component intended function as components in the NUREG-1801 line item,

- Located in the same environment as components in the NUREG-1801 line item, and

- Have exhibited the same aging effects identified in the NUREG 1801 line item.

Component types meeting these criteria have been included in the presentation of aging management review results in Table 3.2-1, “Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the engineered safety features.” The third column of the table shows the component types included in each evaluation line. “NUREG 1801 Components” are those that correspond exactly with component types in NUREG 1801, Volume 2. “Evaluated with NUREG 1801 Components” shows the component types that meet the four (4) criteria above, and therefore share the same evaluation characteristics. Dresden and Quad Cities do not have all of the component types identified in NUREG 1801, Volume 2. The component type entries in the third column of Table 3.2-1 under the “NUREG 1801 Components” heading are those components or component groups that were identified and evaluated at Dresden and Quad Cities.

Other Components Evaluated

Table 3.2-2, "Aging management review results for the engineered safety features that are not addressed in NUREG-1801” presents the aging management review results for the remainder of the engineered safety features systems components. These entries result from aging management review results where the component type, material, environment or aging effect/mechanism differs from NUREG-1801, Volume 2 line item entries. Each line in Table 3.2-2, "Aging management review results for the engineered safety features that are not addressed in NUREG-1801” includes a line reference number, component group, material, environment, aging effect/mechanism, aging management program and discussion.

3.2.1 Aging management programs evaluated in NUREG-1801 that are relied upon for license renewal for the engineered safety features

Table 3.2-1, "Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the engineered safety features" shows the component groups and aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the engineered safety features systems.

Table 3.2-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the engineered safety features

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.2.1.1	Piping, fittings, and valves in emergency core cooling system	NUREG-1801 Components Piping and Fittings	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Further evaluation of cumulative fatigue damage is provided in Section 3.2.1.1.1 and Section 4.3.3.2.
3.2.1.2	Piping, fittings, pumps, and valves in emergency core cooling system	NUREG-1801 Components ECCS Suction Headers Piping and Fittings Pumps Valves Evaluated with NUREG-1801 Components Filters/Strainers Flow Elements Flow Orifices Restricting Orifices Tanks Thermowells Traps	Loss of material due to general corrosion	Water chemistry (B.1.2) and one-time inspection (B.1.23)	Yes, detection of aging effects is to be further evaluated	Consistent with NUREG-1801, with exception. The exceptions to Water Chemistry are described in Section B.1.2. Further evaluation of Loss of Material due to General Corrosion is described in Section 3.2.1.1.2.

Table 3.2-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the engineered safety features (Continued)

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.2.1.3	Components in containment spray (PWR only), standby gas treatment (BWR only), containment isolation, and emergency core cooling systems	NUREG-1801 Components Doors, Closure Bolts, Equip Frames Fan Housings Housings and Supports Isolation Barriers Piping and Fittings Evaluated with NUREG-1801 Components Isolation Barriers Rupture Discs Thermowells Vacuum Breakers	Loss of material due to general corrosion	Plant specific	Yes, plant specific	Further evaluation of Loss of Material due to General Corrosion is described in Section 3.2.1.1.3. External surfaces of carbon steel components identified in NUREG-1801 item V.E.1-b. are evaluated in Table 3.2.2.

Table 3.2-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the engineered safety features (Continued)

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.2.1.4	Piping, fittings, pumps, and valves in emergency core cooling system	NUREG-1801 Components ECCS Suction Headers Piping and Fittings Pumps Valves Evaluated with NUREG-1801 Components Filters/Strainers Flow Elements Flow Orifices Restricting Orifices Tanks Thermowells Traps	Loss of material due to pitting and crevice corrosion	Water chemistry (B.1.2) and one-time inspection (B.1.23)	Yes, detection of aging effects is to be further evaluated	Consistent with NUREG-1801, with exception. The exceptions to Water Chemistry are described in Section B.1.2. Further evaluation of Local Loss of Material due to Pitting and Crevice Corrosion is described in Section 3.2.1.1.4.
3.2.1.5	Components in containment spray (PWR only), standby gas treatment (BWR only), containment isolation, and emergency core cooling systems	NUREG-1801 Components Isolation Barriers Piping and Fittings Evaluated with NUREG-1801 Components Isolation Barriers Rupture Discs Thermowells Vacuum Breakers	Loss of material due to pitting and crevice corrosion	Plant specific	Yes, plant specific	Further evaluation of Local Loss of Material due to Pitting and Crevice Corrosion is described in Section 3.2.1.1.5.
3.2.1.6	Containment isolation valves and associated piping	NUREG-1801 Components Isolation Barriers	Loss of material due to microbiologically influenced corrosion	Plant specific	Yes, plant specific	Further evaluation of Local Loss of Material due to Microbiologically Influenced Corrosion is described in Section 3.2.1.1.6.

Table 3.2-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the engineered safety features (Continued)

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.2.1.7	Seals in standby gas treatment system	NUREG-1801 Components Flex Collars, Doors and Damper Seals Seals	Changes in properties due to elastomer degradation	Plant specific	Yes, plant specific	Further evaluation of Changes in Properties due to Elastomer Degradation is described in Section 3.2.1.1.7.
3.2.1.8	Drywell and suppression chamber spray system nozzles and flow orifices	See Discussion Column	Plugging of nozzles and flow orifices due to general corrosion	Plant specific	Yes, plant specific	NUREG-1801 specifies carbon steel material. Dresden and Quad Cities have brass/bronze spray nozzles.
3.2.1.9	Piping and fittings of CASS in emergency core cooling system	See Discussion Column	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No	CASS material does not exist at Dresden or Quad Cities.
3.2.1.10	Components serviced by open-cycle cooling system	See Discussion Column	Local loss of material due to corrosion and/or buildup of deposit due to biofouling	Open-cycle cooling water system (B.1.13)	No	Material does not match materials at Dresden or Quad Cities.
3.2.1.11	Components serviced by closed-cycle cooling system	See Discussion Column	Loss of material due to general, pitting, and crevice corrosion	Closed-cycle cooling water system (B.1.14)	No	Material does not match materials at Dresden or Quad Cities.
3.2.1.12	Emergency core cooling system valves and lines to and from HPCI and RCIC pump turbines	NUREG-1801 Components Piping and Fittings Valves	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion (B.1.11)	No	Consistent with NUREG-1801, with exception. The exceptions to flow accelerated corrosion are described in Section 3.2.1.2.1.

Table 3.2-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the engineered safety features (Continued)

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.2.1.13	Pumps, valves, piping, and fittings in emergency core cooling systems	NUREG-1801 Components Piping and Fittings Valves Evaluated with NUREG-1801 Components Dampeners Diffusers Filters/Strainers Flow Elements Pumps Restricting Orifices Tanks Thermowells Tubing	Crack initiation and growth due to SCC and IGSCC	Water chemistry (B.1.2) and BWR stress corrosion cracking (B.1.7)	No	Consistent with NUREG-1801, with exception. The exceptions to BWR stress corrosion cracking are described in Section 3.2.1.2.2. The exceptions to Water Chemistry are described in Section B.1.2. The exceptions to BWR Stress Corrosion Cracking are described in Section B.1.7.
3.2.1.14	Closure bolting in high pressure or high temperature systems	NUREG-1801 Components Closure Bolting	Loss of material due to general corrosion, loss of preload due to stress relaxation and crack initiation and growth due to cyclic loading or SCC	Bolting integrity (B.1.12)	No	Consistent with NUREG-1801, with exception. The exceptions to Bolting Integrity are described in Section B.1.12.

3.2.1.1 Further evaluation of aging management as recommended by NUREG-1801 for the engineered safety features systems

3.2.1.1.1 Cumulative Fatigue Damage (NUREG-1800, Section 3.2.2.2.1)

Cumulative fatigue damage of ESF piping is a TLAA as defined in 10 CFR 54.3. Cumulative fatigue damage of ESF piping is required to be evaluated in accordance with 10 CFR 54.21(c)(1). The TLAA evaluation of ESF piping outside the RCPB is addressed in Section 4.3.3.2.

3.2.1.1.2 Loss of Material due to General Corrosion (NUREG-1800, Section 3.2.2.2.2.1)

An inspection of selected components exposed to a stagnant flow water environment will be conducted in accordance with aging management program One-Time Inspection (B.1.23.) The inspection of selected components will verify the effectiveness of the chemistry control program to manage loss of material due to general corrosion in low flow or stagnant flow areas by ensuring that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation.

HPCI components were selected to provide typical samples of the aging effects seen in the ESF systems. Examinations will be conducted on carbon steel components in an area where typically stagnant flow is present but occasionally there is flow, which will cause replenishment of the oxygen supply. Inspections will be conducted on the HPCI torus suction check valves and the HPCI booster pumps. The carbon steel HPCI torus suction check valves are exposed to torus water while the carbon steel HPCI booster pumps are exposed to condensate storage tank water. Each will undergo a visual examination.

3.2.1.1.3 Loss of Material due to General Corrosion (NUREG-1800, Section 3.2.2.2.2.2)

An inspection in accordance with One-Time Inspection (B.1.23) of standby gas treatment system (SGTS) ducts and components will be performed. The one-time inspection will provide assurance that penetrating corrosion of SGTS components is not occurring at an unacceptable rate. The inspection will consist of VT-3 visual inspections for the presence of general corrosion in selected standby gas treatment components.

An inspection in accordance with One-Time Inspection (B.1.23) of carbon steel piping most likely to experience a loss of material in the Dresden and Quad Cities safety relief discharge piping, Dresden and Quad Cities HPCI systems, Dresden LPCI (spray piping) system, Quad Cities RHR (spray piping), and the Quad Cities RCIC system that are exposed to a containment atmosphere environment (wet gas) will be performed. The safety relief discharge piping at Dresden and Quad Cities is carbon steel. The water level in the suppression chamber fluctuates, subjecting the section of safety relief discharge piping, HPCI piping, and RCIC piping at the waterline to repeated wetting and drying, and therefore making it more susceptible to general corrosion, pitting and crevice corrosion in that area. The examination will consist of four ultrasonic tests to detect reduction in wall thickness due to loss of material on the inside of the piping at the waterline. Engineering will specify one safety relief discharge piping inspection location at Dresden and another inspection location at Quad Cities. Engineering will also specify one HPCI piping inspection location at Dresden and another inspection location at Quad Cities. An evaluation of the inspection results will be performed to determine that there is no unacceptable loss of material for the selected piping in the Dresden and Quad Cities safety relief discharge piping, Dresden and Quad Cities HPCI systems, Dresden LPCI (spray piping) system, Quad Cities RHR (spray piping), and the Quad Cities RCIC system components exposed to a containment atmosphere environment (wet gas).

Containment isolation barriers (penetration piping and isolation valves) will be inspected per the requirements of aging management program 10 CFR 50, Appendix J (B.1.28), to verify the pressure retaining integrity of individual penetrations. Containment isolation barriers subject to loss of material due to general corrosion are found in the following license renewal systems:

- Primary containment and suppression pool piping
- Traversing incore probe system
- Drywell equipment and floor drain system
- Atmospheric containment air dilution
- Service and emergency breathing air system
- Instrument air system

Components Requiring Aging Management for Loss of Material due to General Corrosion

Component Group	Aging Management Program		
	One-Time Inspection (B.1.23) Ventilation and SBGT Systems	One-Time Inspection (B.1.23) Piping Exposed to Containment Atmosphere (Wet Gas)	10 CFR 50, Appendix J (B.1.28)
Duct & Fittings, Access Doors, Closure Bolts, and Equipment Frames	X		
Fan Housings	X		
Filter Housing and Supports	X		
Isolation Barriers		X	X
Piping and Fittings		X	
Rupture Discs		X	
Thermowells		X	
Vacuum Breakers		X	

3.2.1.1.4 Local Loss of Material due to Pitting and Crevice Corrosion (NUREG-1800, Section 3.2.2.2.3.1)

An inspection of selected components exposed to a stagnant flow water environment will be conducted in accordance with aging management program One-Time Inspection (B.1.23.) The inspection of selected components will verify the effectiveness of the chemistry control program in low flow or stagnant flow areas. The inspections ensure that significant degradation due to pitting and crevice corrosion is not occurring and the component intended function will be maintained during the extended period of operation. Examinations will be conducted on components in areas where typically stagnant flow is present but occasionally there is flow, which will cause replenishment of the oxygen supply. Inspections will be conducted on the HPCI torus suction check valves, the HPCI booster pumps, and the control rod drive (CRD) scram valves. These components were selected to provide typical samples of the aging effects seen in the ESF systems. The carbon steel HPCI torus suction check valves are exposed to torus water, while the carbon steel HPCI booster pumps and the stainless steel CRD scram valves are exposed to condensate storage tank water. Each will undergo a visual examination.

3.2.1.1.5 Local Loss of Material due to Pitting and Crevice Corrosion (NUREG-1800, Section 3.2.2.2.3.2)

An inspection in accordance with One-Time Inspection (B.1.23) of carbon steel piping most likely to experience a loss of material in the Dresden and Quad Cities safety relief discharge piping, Dresden and Quad Cities HPCI systems, Dresden LPCI (spray piping) system, Quad Cities RHR (spray piping), and the Quad Cities RCIC system that are exposed to a containment atmosphere environment (wet gas) will be performed. The safety relief discharge piping at Dresden and Quad Cities is carbon steel. The water level in the suppression chamber fluctuates, subjecting the section of safety relief discharge piping, HPCI piping, and RCIC piping at the waterline to repeated wetting and drying, and therefore making it more susceptible to general corrosion, pitting and crevice corrosion in that area. The examination will consist of four ultrasonic tests to detect reduction in wall thickness due to loss of material on the inside of the piping at the waterline. Engineering will specify one safety relief discharge piping inspection location at Dresden and another inspection location at Quad Cities. Engineering will also specify one HPCI piping inspection location at Dresden and another inspection location at Quad Cities. An evaluation of the inspection results will be performed to determine that there is no unacceptable loss of material for the selected piping in the Dresden and Quad Cities safety relief discharge piping, Dresden and Quad Cities HPCI systems, Dresden LPCI (spray piping) system, Quad Cities RHR (spray piping), and the Quad Cities RCIC system components exposed to a containment atmosphere environment (wet gas).

Containment isolation barriers (penetration piping and isolation valves) will be inspected per the requirements of aging management program 10 CFR 50, Appendix J (B.1.28), to verify the pressure retaining integrity of individual penetrations. Containment isolation barriers subject to loss of material due to pitting and crevice corrosion are found in the following license renewal systems:

- Primary containment and suppression pool piping
- Traversing incore probe system

- Drywell equipment and floor drain system
- Atmospheric containment air dilution
- Service and emergency breathing air system
- Instrument air system

Components Requiring Aging Management for Loss of Material due to Pitting and Crevice Corrosion

Component Group	Aging Management Program	
	One-Time Inspection: (B.1.23) Piping Exposed to Containment Atmosphere (wet gas)	10 CFR 50 Appendix J (B.1.28)
Isolation Barriers	X	X
Piping and Fittings	X	
Rupture Discs	X	
Thermowells	X	
Vacuum Breakers	X	

3.2.1.1.6 Local Loss of Material due to Microbiologically Influenced Corrosion (NUREG-1800, Section 3.2.2.2.4)

Management of aging due to local loss of material resulting from microbiologically influenced corrosion (MIC) in the drywell equipment drain sump and drywell floor drain sump containment isolation barriers is performed in accordance with aging management program 10 CFR Part 50, Appendix J (B.1.28.) No other containment isolation barriers are subject to loss of material due to MIC or biofouling.

3.2.1.1.7 Changes in Properties due to Elastomer Degradation (NUREG-1800, Section 3.2.2.2.5)

Aging management of standby gas treatment system elastomers will be performed by the periodic inspection of ventilation system elastomers in accordance with plant-specific aging management program Periodic Inspection of Ventilation System Elastomers (B.2.3.)

3.2.1.2 Aging management programs or evaluations that are different than those described in NUREG-1801

3.2.1.2.1 Exception to NUREG-1801 for flow accelerated corrosion

Flow accelerated corrosion is an applicable aging mechanism for the Quad Cities HPCI steam line drains. However, carbon steel components in the ATWS, isolation condenser, core spray, LPCI (Dresden only), RHR (Quad Cities only), primary containment and suppression pool piping, HPCI (except as previously noted) and RCIC (Quad Cities only) systems are not susceptible to flow accelerated corrosion and do not require aging management. This exception is based on the following:

1. EPRI NSAC-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program," allows an exclusion from flow accelerated corrosion for systems that operate less than 2% of plant operating time.

2. EPRI TR-114882, "Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools," states that flow rates less than 6 ft/sec do not need to be considered for flow accelerated corrosion .

3. NUREG-1557, "Summary of Technical Information and Agreements from Nuclear Management and Resources Council Industry Reports Addressing License Renewal," states that erosion/corrosion in HPCI and the RCIC turbine steam supply piping is non-significant due to the low flow range.

4. Dresden and Quad Cities operate these systems less than 2% of plant operating time or at flow rates of less than 6 ft/sec. Additionally, plant experience has not revealed flow accelerated corrosion in these lines.

3.2.1.2.2 Exception to GALL for XI.M7, "BWR Stress Corrosion Cracking" and XI.M2, "Water Chemistry"

NUREG-1801 program XI.M7, "BWR Stress Corrosion Cracking," does not apply to the segments of ECCS systems which are stainless steel and contain torus water. EPRI TR-1003056, "Mechanical Tools" Appendix A, states that cracking due to SCC and IGSCC is not likely in a high purity environment below 200 degrees F. NUREG-1801 program XI.M7, "BWR Stress Corrosion Cracking," applies to piping that contains reactor coolant at a temperature above 200 degrees F. The ECCS piping that contains torus water does not reach this level of temperature. Therefore, aging management program XI.M7 does not apply. Aging management program Water Chemistry (B.1.2) alone will manage aging due to cracking by controlling chloride and sulfate contaminants.

3.2.2 Components or aging effects that are not addressed in NUREG-1801 for the engineered safety features

Table 3.2-2 "Aging management review results for the engineered safety features that are not addressed in NUREG-1801" contains aging management review results for the engineered safety features systems for component groups that are not addressed in NUREG-1801.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.1	Closure Bolting	Low-Alloy Steel	Containment Nitrogen	Crack initiation and growth/ Cyclic loading	Bolting Integrity (B.1.12)	NUREG-1801 does not address closure bolting in a containment nitrogen environment.
3.2.2.2	Piping and Fittings	Stainless Steel	Air and steam up to 320°C (608°F) (Primarily Air)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address NSR components in an air and steam environment.
3.2.2.3	Tanks	Brass or Bronze	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address brass or bronze components in a saturated air environment.
3.2.2.4	Closure Bolting	Low-Alloy Steel	Containment Nitrogen	Loss of material/ Wear	Bolting Integrity (B.1.12)	NUREG-1801 does not address closure bolting in a containment nitrogen environment.
3.2.2.5	Sight Glasses	Glass	Air and steam up to 320°C (608°F)	None	None	NUREG-1801 does not address glass components in an air and steam environment. An air and steam environment is not conducive to promoting aging degradation of glass components. Glass in this environment does not have any applicable aging effect.
3.2.2.6	Closure Bolting	Low-Alloy Steel	Outdoor ambient conditions	Loss of Material/General corrosion and wear	Bolting Integrity (B.1.12)	NUREG-1801 does not address closure bolting in an outdoor environment.
3.2.2.7	Valves	Brass or Bronze	Air and steam up to 320°C (608°F)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address brass or bronze components in an air and steam environment.
3.2.2.8	Component External Surfaces (turbine casings)	Alloy Steel Casting (A217)	Air, moisture, and humidity < 100°C (212°F)	Loss of material/ General corrosion	Bolting Integrity (B.1.12)	NUREG-1801 does not address ESF System alloy steel components in a plant indoor environment.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.9	Component External Surfaces (piping and fittings)	Aluminum	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address aluminum in a plant indoor environment. Plant indoor environment is not an aggressive wetted environment conducive to promoting aging degradation of aluminum components.
3.2.2.10	Component External Surfaces (piping and fittings, valves, tanks)	Aluminum	Outdoor ambient conditions	Loss of material/ Pitting	Bolting Integrity (B.1.12) and Structures Monitoring Program (B.1.30)	NUREG-1801 does not address aluminum component external surfaces in an outdoor environment. Bolting Integrity (B.1.12) is used to manage piping and fittings, valves, and tanks. In addition, Structures Monitoring Program (B.1.30) inspects piping adjacent to selected supports.
3.2.2.11	Component External Surfaces (valves, tanks)	Brass or Bronze	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address brass or bronze components in a plant indoor environment. Copper alloy materials are not subject to any viable aging mechanism in this environment.
3.2.2.12	Component External Surfaces (piping and fittings, valves, spray nozzles)	Brass or Bronze	Containment Nitrogen	None	None	NUREG-1801 does not address brass or bronze components in a containment nitrogen environment. A containment nitrogen environment is not conducive to promoting aging degradation.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.13	Component External Surfaces (piping and fittings, valves, restricting orifices, thermowells, traps, tubing, filters/ strainers, tanks, tubing)	Carbon Steel	Air, moisture, and humidity where surface temp > 100°C (212°F)	None	None	NUREG-1801 does not address high temperature carbon steel external surfaces. A temperature limit of 212°F applies to all materials as moisture must be present for corrosion to occur.
3.2.2.14	Component External Surfaces (piping and fittings, valves, tubing, isolation barriers)	Carbon Steel	Containment Nitrogen	None	None	NUREG-1801 does not address carbon steel in a containment nitrogen environment. A containment nitrogen environment is not conducive to promoting aging degradation.
3.2.2.15	Component External Surfaces (piping and fittings, valves)	Carbon Steel	Outdoor ambient conditions	Loss of material/ General pitting crevice corrosion	Bolting Integrity (B.1.12)	NUREG-1801 does not address ESF System carbon steel component external surfaces in an outdoor environment.
3.2.2.16	Component External Surfaces (carbon steel components)	Carbon Steel	Soil and groundwater	Loss of Material/ General pitting crevice and microbiologically influenced corrosion	Buried Piping and Tanks Inspection (B.1.25)	NUREG-1801 does not address buried SGTS carbon steel piping external surfaces.
3.2.2.17	Component External Surfaces (pumps, valves, tanks)	Cast Iron	Air, moisture, and humidity < 100°C (212°F)	Loss of material/ Pitting and crevice corrosion	Bolting Integrity (B.1.12)	NUREG-1801 does not address cast iron component external surfaces in an indoor environment.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.18	Component External Surfaces (tubing)	Copper	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address copper in a plant indoor environment. The plant indoor environment is not an aggressive wetted environment conducive to promoting aging degradation of copper components.
3.2.2.19	Component External Surfaces (tubing)	Copper	Containment Nitrogen	None	None	NUREG-1801 does not address copper components in a containment nitrogen environment. A containment nitrogen environment is not conducive to promoting aging degradation.
3.2.2.20	Component External Surfaces (sight glasses)	Glass	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address glass in a plant indoor environment. A plant indoor environment is not conducive to promoting aging degradation of glass components. Glass in this environment does not have any applicable aging effect.
3.2.2.21						Line left intentionally blank.
3.2.2.22	Component External Surfaces (piping and fittings, valves, tubing, dampeners, filters/strainers, rupture discs, flow elements, restricting orifices, thermowells, isolation barriers, tanks, pumps,)	Stainless Steel	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address stainless steel engineered safety features components in a plant indoor environment. Plant indoor environment is not conducive to promoting aging degradation of stainless steel components.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.23	Component External Surfaces (piping and fittings, valves, dampeners, rupture discs, tubing)	Stainless Steel	Air, moisture, and humidity where surface temp > 100°C (212°F)	None	None	NUREG-1801 does not address high temperature stainless steel external surfaces. A temperature limit of 212°F applies to all materials as moisture must be present for corrosion to occur.
3.2.2.24	Component External Surfaces (piping and fittings, valves, isolation barriers, tubing, flow elements)	Stainless Steel	Containment Nitrogen	None	None	NUREG-1801 does not address stainless steel in a containment nitrogen environment. A containment nitrogen environment is not conducive to promoting aging degradation
3.2.2.25	Component External Surfaces (piping and fittings, valves)	Stainless Steel	Outdoor ambient conditions	Loss of material/ Pitting and crevice corrosion	Bolting Integrity (B.1.12) and Structures Monitoring Program (B.1.30)	NUREG-1801 does not address outdoor stainless steel ECCS supply piping external surfaces. Bolting Integrity (B.1.12) is used to manage piping and fittings and valves. In addition, Structures Monitoring Program (B.1.30) inspects piping adjacent to selected supports.
3.2.2.26	Component External Surfaces (piping and fittings)	Stainless Steel	Soil and groundwater	Loss of material/ Pitting crevice and microbiologically influenced corrosion	Buried Piping and Tanks Inspection (B.1.25)	NUREG-1801 does not address buried stainless steel ECCS supply piping external surfaces.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.27	Component External Surfaces (piping and fittings, valves)	Steel Chrome Moly	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address chrome-moly steel in a plant indoor environment. Plant indoor environment is not an aggressive wetted environment conducive to promoting aging degradation of chrome-moly steel components.
3.2.2.28	Component External Surfaces (piping and fittings)	Steel Chrome Moly	Air, moisture, and humidity where surface temp > 100°C (212°F)	None	None	NUREG-1801 does not address high temperature chrome-moly steel external surfaces. A temperature limit of 212°F applies to all materials as moisture must be present for corrosion to occur.
3.2.2.29	Dampeners	Stainless Steel	25-288°C (77-550°F) demineralized water	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address non safety related components in a demineralized water environment.
3.2.2.30	Dampeners	Stainless Steel	Air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel dampeners in an air environment.
3.2.2.31	Filters/Strainers	Stainless Steel	Internal: occasional exposure to moist air; external: ambient plant air environment	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components with an internal exposure to occasional moisture and external exposure to plant air.
3.2.2.32	Filters/Strainers	Carbon Steel	Lubricating oil (with contaminants and/or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.33	Flexible Hoses	Elastomers Neoprene and Similar Materials	Lubricating oil (with contaminants and/or moisture)	Hardening and loss of strength/ Elastomer degradation	One-Time Inspection (B.1.23)	NUREG-1801 does not address elastomers, neoprene or similar material components in a lubricating oil environment.
3.2.2.34	Flexible Hoses	Elastomers Neoprene and Similar Materials	Moist air	Hardening and loss of strength/ Elastomer degradation	One-Time Inspection (B.1.23)	NUREG-1801 does not address elastomers in a moist air environment.
3.2.2.35	Flexible Hoses	Elastomers Neoprene and Similar Materials	Moist containment atmosphere (air/nitrogen), steam, or demineralized water	Hardening and loss of strength/ Elastomer degradation	One-Time Inspection (B.1.23)	NUREG-1801 does not address elastomers in a moist containment atmosphere environment.
3.2.2.36	Flexible Hoses	Elastomers Neoprene and Similar Materials	Saturated air	Hardening and loss of strength/ Elastomer degradation	One-Time Inspection (B.1.23)	NUREG-1801 does not address elastomers in a saturated air environment.
3.2.2.37	Flow Elements	Stainless Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel flow elements in a warm, moist air environment.
3.2.2.38	Heat Exchangers	Tubes: Admiralty Brass	Tube Side: Condensate (demineralized water); Shell side: Lubricating oil	Buildup of Deposit/Fouling	Heat Exchanger Test and Inspection Activities (B.2.6)	NUREG-1801 does not address lube oil coolers.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.39	Heat Exchangers	Tubes: Admiralty Brass	Tube Side: Condensate (demineralized water); Shell side: Reactor coolant water and Warm moist air	Buildup of Deposit/Fouling	Heat Exchanger Test and Inspection Activities (B.2.6)	NUREG-1801 does not address admiralty brass components or air environments for ECCS heat exchangers.
3.2.2.40	Heat Exchangers	Tubes: Admiralty Brass; Tubesheet: Brass; Channel Head: Carbon Steel; Shell: Carbon Steel	Tube Side: Condensate (demineralized water); Shell side: Lubricating oil	Loss of Material/General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Selective Leaching	Water Chemistry (B.1.2), Selective Leaching of Materials (B.1.24), Lube Oil Monitoring Activities, Heat Exchanger Test and Inspection Activities (B.2.6)	NUREG-1801 does not address lube oil coolers. Galvanic corrosion, erosion or FAC, and wear are not included in NUREG-1801 for ECCS heat exchangers.
3.2.2.41	Heat Exchangers	Tubes: Admiralty Brass; Tubesheet: Brass; Channel Head: Carbon Steel; Shell: Carbon Steel	Tube Side: Condensate (demineralized water); Shell side: Lubricating oil	Cracking/Mech Fatigue, SCC	Water Chemistry (B.1.2), Lube Oil Monitoring Activities, Heat Exchanger Test and Inspection Activities (B.2.6)	NUREG-1801 does not address lube oil coolers.
3.2.2.42	Heat Exchangers	Tubes: Admiralty Brass; Tubesheet: Carbon Steel; Channel Head: Carbon Steel; Shell: Carbon Steel	Tube Side: Condensate (demineralized water); Shell side: Reactor coolant water and Warm moist air	Loss of Material/General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Selective Leaching, Pitting Corrosion, Crevice Corrosion	Water Chemistry (B.1.2), Selective Leaching of Materials (B.1.24), Heat Exchanger Test and Inspection Activities	NUREG-1801 does not address admiralty brass components or air environments for ECCS heat exchangers.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.43	Heat Exchangers	Tubes: Admiralty Brass; Tubesheet: Carbon Steel; Channel Head: Carbon Steel; Shell: Carbon Steel	Tube Side: Condensate (demineralized water); Shell side: Reactor coolant water and Warm moist air	Cracking/Mech Fatigue, SCC	Water Chemistry (B.1.2), Heat Exchanger Test and Inspection Activities (B.2.6)	NUREG-1801 does not address admiralty brass components or air environments for ECCS heat exchangers.
3.2.2.44	Insulation	Asbestos	Air, moisture, and humidity < 100°C (212°F)	Insulation degradation/Loss of insulating characteristics	Structures Monitoring Program (B.1.30)	NUREG-1801 does not address indoor asbestos insulation degradation.
3.2.2.45	Insulation	Fiberglass	Air, moisture, and humidity < 100°C (212°F)	Insulation degradation/Loss of insulating characteristics	Structures Monitoring Program (B.1.30)	NUREG-1801 does not address indoor fiberglass insulation degradation.
3.2.2.46	Insulation	NUKON quilted fiberglass	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address NUKON quilted fiberglass insulation in a plant indoor environment. Plant indoor environment is not conducive to promoting aging degradation of quilted fiberglass insulation.
3.2.2.47	Insulation	Stainless Steel Mirror Insulation	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address stainless steel mirror insulation in a plant indoor environment. Stainless steel materials are not subject to any viable aging mechanism in the absence of aggressive chemical species.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.48	Insulation Jacketing	Stainless Steel	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address stainless steel insulation jacketing in a plant indoor environment. Stainless steel materials are not subject to any viable aging mechanism in the absence of aggressive chemical species.
3.2.2.49	Isolation Barriers	Brass or Bronze	Saturated air	Loss of material/ Pitting and crevice corrosion	10 CFR Part 50, Appendix J (B.1.28)	NUREG-1801 does not address brass or bronze components in a saturated air environment.
3.2.2.50	Isolation Barriers	Carbon Steel	Saturated air	Loss of material/ General pitting crevice corrosion	10 CFR Part 50, Appendix J (B.1.28)	NUREG-1801 does not address carbon steel components in a saturated air environment.
3.2.2.51	Isolation Barriers	Carbon Steel	Warm, moist air	Loss of material/ General pitting crevice corrosion	10 CFR Part 50, Appendix J (B.1.28)	NUREG-1801 does not address carbon steel components in a warm, moist air environment.
3.2.2.52	Isolation Barriers	Stainless Steel	Saturated air	Loss of material/ Pitting and crevice corrosion	10 CFR Part 50, Appendix J (B.1.28)	NUREG-1801 does not address stainless steel components in a saturated air environment.
3.2.2.53	Isolation Barriers	Stainless Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	10 CFR Part 50, Appendix J (B.1.28)	NUREG-1801 does not address stainless steel components in a warm, moist air environment.
3.2.2.54	Manifolds	Stainless Steel	Internal: occasional exposure to moist air; external: ambient plant air environment	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components with an internal exposure to occasional moisture and external exposure to plant air.
3.2.2.55	NSR Vents or Drains, Piping and Valves	Carbon Steel, Stainless Steel, Brass or Bronze	Air, moisture, humidity, and leaking fluid	Loss of material/ Corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address NSR vents or drains in an air, moisture, humidity and leaking fluid environment.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.56	Piping and Fittings	Aluminum	25-288°C (77-550°F) demineralized water	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address aluminum in a reactor grade water environment.
3.2.2.57	Piping and Fittings	Carbon Steel	25-288°C (77-550°F) demineralized water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non safety-related components in a demineralized water environment.
3.2.2.58	Piping and Fittings	Carbon Steel	Air and steam up to 320°C (608°F) (Primarily Steam)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address loss of material in an air and steam environment.
3.2.2.59	Piping and Fittings	Carbon Steel	Lubricating oil (with contaminants and/or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.2.2.60	Piping and Fittings	Carbon Steel	Saturated air	Loss of material/ General and pitting corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address carbon steel components in a saturated air environment.
3.2.2.61	Piping and Fittings	Carbon Steel	Treated water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non safety related components in a treated water environment.
3.2.2.62	Piping and Fittings	Carbon Steel	Warm, moist air	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel piping and fittings in a warm, moist air environment.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.63	Piping and Fittings	Stainless Steel	25-288°C (77-550°F) demineralized water	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address non safety related components in a demineralized water environment.
3.2.2.64	Piping and Fittings	Stainless Steel	Air and steam up to 320°C (608°F) (Primarily Steam)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address stainless steel piping and fittings in an air and steam environment.
3.2.2.65	Piping and Fittings	Stainless Steel	Lubricating oil (with contaminants and/or moisture)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in a lubricating oil environment.
3.2.2.66	Piping and Fittings	Stainless Steel	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address stainless steel components in a saturated air environment.
3.2.2.67	Piping and Fittings	Stainless Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel valves in a warm, moist air environment.
3.2.2.68	Piping and Fittings	Steel Chrome Moly	25-288°C (77-550°F) demineralized water	Loss of material/ Pitting	Water Chemistry (B.1.2)	NUREG-1801 does not address chrome moly in a reactor grade water environment.
3.2.2.69	Pumps	Cast Iron	25-288°C (77-550°F) demineralized water	Loss of Material/ Selective Leaching	Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address cast iron in a reactor grade water environment.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.70	Pumps	Cast Iron	Lubricating oil (with contaminants and/or moisture)	Loss of material/ General pitting crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address cast iron components in a lubricating oil environment.
3.2.2.71	Pumps	Cast Iron	Lubricating oil (with contaminants and/or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.2.2.72	Restricting Orifices	Carbon Steel	25-288°C (77-550°F) demineralized water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non safety related components in a demineralized water environment.
3.2.2.73	Restricting Orifices	Carbon Steel	Lubricating oil (with contaminants and/or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.2.2.74	Restricting Orifices	Stainless Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel valves in a warm, moist air environment.
3.2.2.75	Sight Glasses	Carbon Steel	25-288°C (77-550°F) demineralized water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non safety related components in a demineralized water environment.
3.2.2.76	Sight Glasses	Glass	25-288°C (77-550°F) demineralized water	None	None	NUREG-1801 does not address glass components. Glass in this environment does not have any applicable aging effect.
3.2.2.77	Sight Glasses	Glass	Lubricating oil (with contaminants and/or moisture)	None	None	NUREG-1801 does not address glass components in an oil environment. An oil environment is not conducive to promoting aging degradation of glass components. Glass in this environment does not have any applicable aging effect.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.78	Spray Nozzles	Brass or Bronze	Air	Plugging of flow orifice and spray nozzles/ General corrosion	Periodic Testing of Drywell and Torus Spray Nozzles (B.2.4)	NUREG-1801 does not address components made of brass or bronze.
3.2.2.79	Support Members	Carbon Steel	25-288°C (77-550°F) demineralized water	Loss of material/ General galvanic pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address support members in submerged environments.
3.2.2.80	Support Members	Stainless Steel	25-288°C (77-550°F) demineralized water	Cracking/ Stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address support members in submerged environments.
3.2.2.81	Support Members	Stainless Steel	25-288°C (77-550°F) demineralized water	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address support members in submerged environments.
3.2.2.82	Tanks	Aluminum	25-288°C (77-550°F) demineralized water	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address aluminum in a reactor grade water environment.
3.2.2.83	Tanks	Carbon Steel	Air and steam up to 320°C (608°F) (Primarily Steam)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address loss of material in an air and steam environment.
3.2.2.84	Tanks	Carbon Steel	Lubricating oil (with contaminants and/or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.2.2.85	Tanks	Cast Iron	25-288°C (77-550°F) demineralized water	Loss of Material/ Selective Leaching	Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address cast iron in a reactor grade water environment.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.86	Thermowells	Carbon Steel	Lubricating oil (with contaminants and/or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.2.2.87	Thermowells	Carbon Steel	Warm, moist air	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel thermowells in a warm, moist air environment.
3.2.2.88	Traps	Carbon Steel	Air and steam up to 320°C (608°F) (Primarily Steam)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address loss of material in an air and steam environment.
3.2.2.89	Tubing	Carbon Steel	25-288°C (77-550°F) demineralized water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non safety related components in a demineralized water environment.
3.2.2.90	Tubing	Copper	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address copper components in a saturated air environment.
3.2.2.91	Tubing	Copper	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address copper components in a warm, moist air environment.
3.2.2.92	Tubing	Stainless Steel	25-288°C (77-550°F) demineralized water	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address non safety related components in a demineralized water environment.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.93	Tubing	Stainless Steel	288°C (550°F) reactor coolant water	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address non safety related components in a reactor coolant water environment.
3.2.2.94	Tubing	Stainless Steel	Air and steam up to 320°C (608°F) (Primarily Steam)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address stainless steel tubing in an air and steam environment.
3.2.2.95	Tubing	Stainless Steel	Internal: occasional exposure to moist air; external: ambient plant air environment	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components with an internal environment of occasional exposure to moist air and external environment of ambient plant air.
3.2.2.96	Tubing	Stainless Steel	Lubricating oil (with contaminants and/or moisture)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in a lubricating oil environment.
3.2.2.97	Tubing	Stainless Steel	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address stainless steel components in a saturated air environment.
3.2.2.98	Tubing	Stainless Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel tubing in a warm, moist air environment.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.99	Turbine Casings	Alloy Steel Casting (A217)	Air and steam up to 320°C (608°F)	Loss of material/ General, pitting, and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address alloy steel turbine casings in an air and steam environment.
3.2.2.100	Valves	Aluminum	25-288°C (77-550°F) demineralized water	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address aluminum in a reactor grade water environment.
3.2.2.101	Valves	Brass or Bronze	Internal: occasional exposure to moist air; external: ambient plant air environment	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address brass or bronze components in a moist air or ambient plant air environment.
3.2.2.102	Valves	Brass or Bronze	Lubricating oil (with contaminants and/or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.2.2.103	Valves	Brass or Bronze	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address brass or bronze components in a saturated air environment.
3.2.2.104	Valves	Carbon Steel	25-288°C (77-550°F) demineralized water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non safety related components in a demineralized water environment.
3.2.2.105	Valves	Carbon Steel	Air	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel valves in an air environment.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.106	Valves	Carbon Steel	Air and steam up to 320°C (608°F) (Primarily Steam)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address loss of material in an air and steam environment.
3.2.2.107	Valves	Carbon Steel	Air and steam up to 320°C (608°F)	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel valves in an air and steam environment.
3.2.2.108	Valves	Carbon Steel	Internal: occasional exposure to moist air; external: ambient plant air environment	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel valves with an internal exposure to occasional moisture and external exposure to plant air.
3.2.2.109	Valves	Carbon Steel	Lubricating oil (with contaminants and/or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.2.2.110	Valves	Carbon Steel	Saturated air	Loss of material/ General and pitting corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address carbon steel components in a saturated air environment.
3.2.2.111	Valves	Carbon Steel	Treated water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non safety related components in a treated water environment.
3.2.2.112	Valves	Carbon Steel	Warm, moist air	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel valves in a warm, moist air environment.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.113	Valves	Carbon Steel	Wet Gas	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components in a wet gas environment.
3.2.2.114	Valves	Cast Iron	Internal: occasional exposure to moist air; external: ambient plant air environment	Loss of material/ General, pitting, and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address cast iron components in a moist air or ambient plant air environment.
3.2.2.115	Valves	Stainless Steel	25-288°C (77-550°F) demineralized water	Crack initiation and growth/ Stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address non safety related components in a demineralized water environment.
3.2.2.116	Valves	Stainless Steel	288°C (550°F) reactor coolant water	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address non safety related components in a reactor coolant water environment.
3.2.2.117	Valves	Stainless Steel	Air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel valves in an air environment.
3.2.2.118	Closure Bolting	Low-Alloy Steel	Air, moisture, and humidity less than 100°C (212°F)	Loss of material/ General corrosion and wear	Bolting Integrity (B.1.12)	NUREG-1801 does not address closure bolting for low pressure, low temperature indoor system applications.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.119	Valves	Stainless Steel	Air and steam up to 320°C (608°F) (Primarily Steam)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address stainless steel valves in an air and steam environment.
3.2.2.120	Valves	Stainless Steel	Internal: occasional exposure to moist air; external: ambient plant air environment	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components with an internal exposure to occasional moisture and external exposure to plant air.
3.2.2.121	Valves	Stainless Steel	Lubricating oil (with contaminants and/or moisture)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in a lubricating oil environment.
3.2.2.122	Valves	Stainless Steel	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address stainless steel components in a saturated air environment.
3.2.2.123	Valves	Stainless Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel valves in a warm, moist air environment.
3.2.2.124	Valves	Stainless Steel	Wet Gas	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components in a wet gas environment.
3.2.2.125	Dampeners	Stainless Steel	Air and steam up to 320°C (608°F) (Primarily Air)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address loss of material in an air and steam environment.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.126	Piping and Fittings	Carbon Steel	Air and steam up to 320°C (608°F) (Primarily Air)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address loss of material in an air and steam environment.
3.2.2.127	Restricting Orifices	Cast Iron	Internal: occasional exposure to moist air; external: ambient plant air environment	Loss of material/ General corrosion, pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address cast iron components in a moist air or ambient plant air environment.
3.2.2.128	Restricting Orifices	Carbon Steel	Air and steam up to 320°C (608°F) (Primarily Air)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address loss of material in an air and steam environment.
3.2.2.129	Rupture Discs	Stainless Steel	Air and steam up to 320°C (608°F) (Primarily Air)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address loss of material in an air and steam environment.
3.2.2.130	Tanks	Carbon Steel	Air and steam up to 320°C (608°F) (Primarily Air)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address loss of material in an air and steam environment.
3.2.2.131	Traps	Carbon Steel	Air and steam up to 320°C (608°F) (Primarily Air)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address loss of material in an air and steam environment.
3.2.2.132	Tubing	Stainless Steel	Air and steam up to 320°C (608°F) (Primarily Air)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address loss of material in an air and steam environment.

Table 3.2-2 Aging management review results for the engineered safety features that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.2.2.133	Valves	Carbon Steel	Air and steam up to 320°C (608°F) (Primarily Air)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address loss of material in an air and steam environment.
3.2.2.134						Line left intentionally blank.
3.2.2.135	Valves	Stainless Steel	Air and steam up to 320°C (608°F) (Primarily Air)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address loss of material in an air and steam environment.
3.2.2.136						Line left intentionally blank.
3.2.2.137	Component External Surfaces (piping and fittings, filters/strainers, flow orifices, heat exchangers, pumps, restricting orifices, tanks, thermowells, valves, sight glasses, tubing, isolation barriers, traps, isolation condensers, ECCS suction headers, flow elements)	Carbon Steel	Air, moisture, and humidity < 100°C (212°F)	Loss of material/ General corrosion	Bolting Integrity (B.1.12) and Structures Monitoring Program (B.1.30)	NUREG-1801 does not address carbon steel component external surfaces in an indoor environment. Bolting Integrity (B.1.12) is used to manage piping and fittings, filter/strainers, flow orifices, heat exchangers, pumps, restricting orifices, tanks, thermowells, valves, sight glasses, tubing, turbine casings, isolation barriers, traps, isolation condensers, ECCS suction headers, flow elements, and accumulators. In addition, Structures Monitoring Program (B.1.30) inspects piping adjacent to selected supports.

3.3 AGING MANAGEMENT OF AUXILIARY SYSTEMS

The following systems are evaluated as part of the auxiliary systems:

- Refueling equipment system
- Shutdown cooling system (Dresden only)
- Control rod drive hydraulic system
- Reactor water cleanup system
- Fire protection system
- Emergency diesel generator and auxiliaries
- HVAC-Main control room
- HVAC-Reactor building
- ECCS corner room HVAC
- Station blackout building HVAC
- Station blackout system (diesels and auxiliaries)
- Diesel generator cooling water system
- Diesel fuel oil system
- Process sampling system
- Carbon dioxide system
- Service water system
- Reactor building closed cooling water system
- Turbine closed cooling water system (in-scope Dresden only)
- Residual heat removal service water system (Quad Cities only)
- Containment cooling service water system (Dresden only)
- Ultimate heat sink
- Fuel pool cooling and filter demineralizer system (in-scope for Dresden only)
- Plant heating system
- Containment atmosphere monitoring system
- Nitrogen containment atmosphere dilution system
- Drywell nitrogen inerting system
- Safe shutdown makeup pump system (Quad Cities only)
- Standby liquid control system
- Cranes and hoists
- Demineralized water makeup system

Components Evaluated Consistent with NUREG-1801

The components or component groups requiring aging management review in each of the auxiliary systems listed above are presented in Chapter 2. Aging management reviews were performed to assure that the component groups, materials, environments and aging effects referenced in NUREG-1801 are applicable to Dresden and Quad Cities and that the aging management program described in NUREG-1801 is appropriate.

When the components or component group and associated evaluations corresponded with an individual line item in NUREG-1801, Volume 2, the component aging management results were included in the appropriate line items in Table 3.3-1 "Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems" for auxiliary systems. Each line in Table 3.3-1 matches a line in Chapter 3 of NUREG-1800, that is applicable to Boiling Water Reactors (BWR) or to both Boiling Water Reactors and Pressurized Water Reactors (BWR/ PWR).

Not all component types in the Dresden and Quad Cities Auxiliary Systems are listed in NUREG-1801, Volume 2. However, the aging management reviews presented in NUREG-1801, Volume 2 were applied to additional component types if the following criteria were satisfied:

- Constructed of the same material as components in the NUREG-1801 line item,
- Assigned the same Component Intended Function as components in the NUREG-1801 line item,
- Located in the same environment as components in the NUREG-1801 line item, and
- Have exhibited the same aging effects identified in the NUREG-1801 line item.

Component types meeting these criteria have been included in the presentation of aging management review results in Table 3.3-1 "Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems." The third column of the table shows the component types included in each evaluation line. "NUREG-1801 Components" are those that correspond exactly with component types in NUREG-1801, Volume 2. "Evaluated with NUREG-1801 Components" shows the component types that meet the four (4) criteria above, and therefore share the same evaluation characteristics. Dresden and Quad Cities do not have all of the component types identified in NUREG-1801, Volume 2. The component type entries in the third column of Table 3.3-1 under the "NUREG-1801 Components" heading are those components or component groups that were identified and evaluated at Dresden and Quad Cities.

<u>Other Components Evaluated</u>

Table 3.3-2 "Aging management review results for the auxiliary systems that are not addressed in NUREG-1801" presents the aging management review results for the remainder of the auxiliary systems components. These entries result from aging management review results where the component type, material, environment or aging effect/mechanism differs from NUREG-1801, Volume 2 line item entries. Each line in Table 3.3-2 includes a line reference number, component group, material, environment, aging effect/mechanism, aging management program and discussion.

3.3.1 Aging management programs evaluated in NUREG-1801 that are relied upon for license renewal for the auxiliary systems

Table 3.3-1 "Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems" shows the component groups and aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems.

Table 3.3-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.3.1.1	Components in spent fuel pool cooling and cleanup	NUREG-1801 Components Piping and Fittings	Loss of material due to general, pitting, and crevice corrosion	Water chemistry (B.1.2) and one-time inspection (B.1.23)	Yes, detection of aging effects is to be further evaluated	Consistent with NUREG-1801, with exception. The exceptions to Water Chemistry are described in Section B.1.2. Further evaluation of Loss of Material due to General, Pitting, and Crevice Corrosion is described in Section 3.3.1.1.1. Components evaluated in NUREG-1801 lines VII.A4.2-a, 3-a, 4-b, 5-a and 6-a are not in the scope of license renewal. Only Dresden has these components in the scope of license renewal.
3.3.1.2	Linings in spent fuel pool cooling and cleanup system; seals and collars in ventilation systems	NUREG-1801 Components Flex Collars, Doors and Damper Seals Seals	Hardening, cracking and loss of strength due to elastomer degradation; loss of material due to wear	Plant specific	Yes, plant specific	Further evaluation of Hardening and Cracking or Loss of Strength due to Elastomer Degradation or Loss of Material due to Wear is described in Section 3.3.1.1.5. Elastomer linings of valves for water systems evaluated in NUREG-1801, line VII.A4, are not within the scope of license renewal. Elastomers are not in the scope of license renewal for ECCS room coolers at Dresden and Quad Cities.

Table 3.3-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.3.1.3	Components in load handling, chemical and volume control system (PWR), and reactor water cleanup and shutdown cooling systems (older BWR)	NUREG-1801 Components Cranes Piping and Fittings	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Further Evaluation of cumulative fatigue damage is provided in Section 3.3.1.1.6, Section 4.3.3.2 (piping) and Section 4.7.1 (crane).
3.3.1.4	Heat exchangers in reactor water cleanup system (BWR); high pressure pumps in chemical and volume control system (PWR)	See Discussion Column	Crack initiation and growth due to SCC or cracking	Plant specific	Yes, plant specific	Components are not in the scope of license renewal

Table 3.3-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.3.1.5	Components in ventilation systems, diesel fuel oil system, and emergency diesel generator systems; external surfaces of carbon steel components	NUREG-1801 Components Air Accumulator Vessels Air Handlers Heating/ Cooling (CR HVAC) Carbon Steel Components Doors, Closure Bolts, Equip Frames Ducts & Fittings, Access Doors, Closure Bolts, Equip Frames Filters/ Strainers Housings and Supports Mufflers Piping and Fittings Evaluated with NUREG-1801 Components Dampeners, Filters/ Strainers, Flame Arrestors, Lubricators, Pumps, Valves, Heat Exchangers, Restricting Orifices, Orifice Bodies, Sight Glasses, Sprinklers, Tanks, Strainer bodies, Thermowells, Tubing, Diffusers, Flow Elements Traps	Loss of material due to general, pitting, and crevice corrosion, and MIC	Plant specific	Yes, plant specific	Further evaluation of Loss of Material due to General, Microbiologically Influenced, Pitting, and Crevice Corrosion is described in Section 3.3.1.1.7. The primary containment heating and cooling coils evaluated in NUREG-1801, line VII.F3.2-a are not within the scope of license renewal. Dresden and Quad Cities do not have diesel fuel oil system valves or pumps, identified in NUREG-1801, lines VII.H1.2 and VII.H1.3, located outdoors. Dresden and Quad Cities do not have material-environment combination evaluated in NUREG-1801 lines VII.F1.2-a, VII.F2.2-a, VII.F2.4-a, VII.F4.2-a. Filter housing and supports identified in NUREG-1801, line VII.F2.4-a are not included in Dresden and Quad Cities auxiliary and radwaste area ventilation systems.

Table 3.3-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.3.1.6	Components in reactor coolant pump oil collect system of fire protection	See Discussion Column	Loss of material due to galvanic, general, pitting, and crevice corrosion	One-time inspection (B.1.23)	Yes, detection of aging effects is to be further evaluated	Lube oil components for Dresden and Quad Cities are addressed as non NUREG-1801 lines.
3.3.1.7	Diesel fuel oil tanks in diesel fuel oil system and emergency diesel generator system	NUREG-1801 Components Tanks Evaluated with NUREG-1801 Components Filters/ Strainers Piping and Fittings Pumps Thermowells Valves	Loss of material due to general, pitting, and crevice corrosion, MIC, and biofouling	Fuel oil chemistry (B.1.21) and one-time inspection (B.1.23)	Yes, detection of aging effects is to be further evaluated	Consistent with NUREG-1801, with exception. The exceptions to Fuel Oil Chemistry are described in Section B.1.21. Further evaluation of Loss of Material due to General, Pitting, Crevice, and Microbiologically Influenced Corrosion and Biofouling is described in Section 3.3.1.1.8.
3.3.1.8	Piping, pump casing, and valve body and bonnets in shutdown cooling system (older BWR)	NUREG-1801 Components Piping and Fittings Pumps Evaluated with NUREG-1801 Components Dampeners Filters/ Strainers Flow Elements Restricting Orifices Sight Glasses Tanks Thermowells Tubing Valves	Loss of material due to pitting and crevice corrosion	Water chemistry (B.1.2) and one-time inspection (B.1.23)	Yes, detection of aging effects is to be further evaluated	Consistent with NUREG-1801, with exception. The exceptions to Water Chemistry are described in Section B.1.2. Further evaluation of Loss of Material due to General, Pitting, and Crevice Corrosion is described in Section 3.3.1.1.2.

Table 3.3-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.3.1.9	Neutron absorbing sheets in spent fuel storage racks	NUREG-1801 Components Neutron-Absorbing Sheets	Reduction of neutron absorbing capacity and loss of material due to general corrosion (Boral, boron steel)	Plant specific	Yes, plant specific	Further evaluation of Reduction of Neutron-Absorbing Capacity and Loss of Material due to General Corrosion is described in Section 3.3.1.1.3. Boral is used only at Dresden.
3.3.1.10	New fuel rack assembly	See Discussion Column	Loss of material due to general, pitting, and crevice corrosion	Structures monitoring (B.1.30)	No	The new fuel racks at Dresden and Quad Cities are not made of carbon steel. ThereforeNUREG-1801, line VII.A1.1-a, does not apply.
3.3.1.11	Spent fuel storage racks and valves in spent fuel pool cooling and cleanup	NUREG-1801 Components Storage Racks	Crack initiation and growth due to stress corrosion cracking	Water chemistry (B.1.2)	No	Consistent with NUREG-1801, with exception. The exceptions to Water Chemistry are described in Section B.1.2.
3.3.1.12	Neutron absorbing sheets in spent fuel storage racks	NUREG-1801 Components Neutron-Absorbing Sheets	Reduction of neutron absorbing capacity due to Boraflex degradation	Boraflex monitoring (B.1.36)	No	Consistent with NUREG-1801. Boraflex is used only at Quad Cities.

Table 3.3-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.3.1.13	Components in or serviced by closed-cycle cooling water system	NUREG-1801 Components Orifice Bodies Piping and Fittings Pumps Tanks Valves Evaluated with NUREG-1801 Components Manifolds Piping and Fittings Pumps Tanks Thermowells Tubing Valves	Loss of material due to general, pitting, and crevice corrosion, and MIC	Closed-cycle cooling water system (B.1.14)	No	Consistent with NUREG-1801. The fuel pool cooling heat exchangers identified in NUREG-1801, line VII.A4.4 are not in the scope of license renewal. The reactor water cleanup system non-regenerative heat exchangers identified in NUREG-1801, line VII.E3.4 are not in the scope of license renewal. The control room, auxiliary and radwaste , primary containment, and diesel generator building HVAC systems identified in NUREG-1801, lines VII.F1.3-a, VII.F2.3-a, VII.F3.3-a, and VII.F4.3-a do not include hot or cold chemically treated water environments. Only Dresden has components in the scope of license renewal that apply to this line item.
3.3.1.14	Cranes including bridge and trolleys and rail system in load handling system	NUREG-1801 Components Cranes Evaluated with NUREG-1801 Components Rails	Loss of material due to general corrosion and wear	Overhead heavy load and light load handling systems (B.1.15)	No	Consistent with NUREG-1801, with exception. The exceptions to Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems are described in Section B.1.15.

Table 3.3-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.3.1.15	Components in or serviced by open-cycle cooling water systems	NUREG-1801 Components Orifice Bodies Piping and Fittings Pumps Strainer Bodies Valves Evaluated with NUREG-1801 Components Dampeners Flow Elements Flow Orifices Pulsation Dampeners Sight Glasses Strainer Screens Thermowells Tubes Tubing	Loss of material due to general, pitting, crevice, and galvanic corrosion, MIC, and biofouling; buildup of deposit due to biofouling	Open-cycle cooling water system (B.1.13)	No	Consistent with NUREG-1801, with exception. The exceptions to biofouling for a(2) components are described in Section 3.3.1.2.2. The exceptions to Open-Cycle Cooling Water System are described in Section B.1.13. Dresden and Quad Cities do not have material-environment combination evaluated in NUREG-1801 lines VII.C1.3-a and VII.C1.3-b. NUREG-1801, line VII.C3.2-a does not apply to valve material at Dresden. Quad Cities does not have any ultimate heat sink valves in the scope of license renewal. Brass and bronze instrument pulsation dampeners are only used at Quad Cities. Only Quad Cities heat exchangers use this material/environment combination.
3.3.1.16	Buried piping and fittings	NUREG-1801 Components Piping and Fittings	Loss of material due to general, pitting, and crevice corrosion, and MIC	Buried piping and tanks surveillance or Buried piping and tanks inspection (B.1.25)	No Yes, detection of aging effects and operating experience are to be further evaluated	Consistent with NUREG-1801, with exception. The exceptions to Buried Piping and Tanks Inspection are described in Section B.1.25. Further evaluation of Loss of Material due to General, Pitting, Crevice, and Microbiologically Influenced Corrosion is described in Section 3.3.1.1.4.

Table 3.3-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.3.1.17	Components in compressed air system	NUREG-1801 Components Piping and Fittings Valves Evaluated with NUREG-1801 Components Filters/ Strainers Tubing	Loss of material due to general and pitting corrosion	Compressed air monitoring (B.1.16)	No	Consistent with NUREG-1801, with exception. The exceptions to Compressed Air Monitoring are described in Section B.1.16. The instrument air system components evaluated in NUREG-1801, lines VII.D.3-a, 4-a, 5-a, and 6-a are not within the scope of license renewal.
3.3.1.18	Components (doors and barrier penetration seals) and concrete structures in fire protection	NUREG-1801 Components Doors Fire Doors Penetration Seals Evaluated with NUREG-1801 Components Fire dampers	Loss of material due to wear; hardening and shrinkage due to weathering	Fire protection (B.1.18)	No	Consistent with NUREG-1801, with exception. The exceptions to Fire Protection are described in Section B.1.18. Elastomers are not in the scope of license renewal for Intake structures (crib house) at Dresden and Quad Cities. Primary containment does not contain fire doors as identified in NUREG-1801, line VII.G.5-c.
3.3.1.19	Components in water-based fire protection	NUREG-1801 Components Filters/ Strainers Fire Hydrants Piping and Fittings Pumps Sprinklers Valves	Loss of material due to general, pitting, crevice, and galvanic corrosion, MIC, and biofouling	Fire water system (B.1.19)	No	Consistent with NUREG-1801, with exception. The exceptions to Fire Water System are described in Section B.1.19.

Table 3.3-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.3.1.20	Components in diesel fire system	NUREG-1801 Components Piping and Fittings Evaluated with NUREG-1801 Components Filters/ Strainers Tanks Valves	Loss of material due to galvanic, general, pitting, and crevice corrosion	Fire protection (B.1.18) and fuel oil chemistry (B.1.21)	No	Consistent with NUREG-1801, with exception. The exceptions to Fire Protection are described in Section B.1.18. The exceptions to Fuel Oil Chemistry are described in Section B.1.21.
3.3.1.21	Tanks in diesel fuel oil system	See Discussion Column	Loss of material due to general, pitting, and crevice corrosion	Aboveground carbon steel tanks (B.1.20)	No	There are no aboveground carbon steel tanks in the diesel fuel oil system
3.3.1.22	Closure bolting	NUREG-1801 Components Closure Bolting	Loss of material due to general corrosion; crack initiation and growth due to cyclic loading and SCC	Bolting integrity (B.1.12)	No	Consistent with NUREG-1801, with exception. The exceptions to Bolting Integrity are described in Section B.1.12.
3.3.1.23	Components in contact with sodium penta-borate solution in standby liquid control system (BWR)	NUREG-1801 Components Piping and Fittings Pumps Tanks Valves Evaluated with NUREG-1801 Components Dampeners Thermowells Tubing	Crack initiation and growth due to SCC	Water chemistry (B.1.2)	No	Consistent with NUREG-1801, with exception. The exceptions to the application of the water chemistry program are described in Section 3.3.1.2.3. The exceptions to Water Chemistry are described in Section B.1.2.

Table 3.3-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.3.1.24	Components in reactor water cleanup system	NUREG-1801 Components Piping and Fittings	Crack initiation and growth due to SCC and IGSCC	Reactor water cleanup system inspection (B.1.17)	No	Consistent with NUREG-1801, with exception. The exceptions to BWR Reactor Water Cleanup System are described in Section B.1.17. The RWCU pumps identified in NUREG-1801, lines VII.E3.2-a, b, and c are not in the scope of license renewal.
3.3.1.25	Components in shutdown cooling system (older BWR)	NUREG-1801 Components Piping and Fittings Valves Evaluated with NUREG-1801 Components Dampeners Filters/ Strainers Flow Elements Pumps Restricting Orifices Tubing	Crack initiation and growth due to SCC	BWR stress corrosion cracking (B.1.7) and water chemistry (B.1.2)	No	Consistent with NUREG-1801, with exception. The exceptions to Water Chemistry are described in Section B.1.2. The exceptions to BWR Stress Corrosion Cracking are described in Section B.1.7.
3.3.1.26	Components in shutdown cooling system (older BWR)	NUREG-1801 Components Heat Exchangers	Loss of material due to pitting and crevice corrosion, and MIC	Closed-cycle cooling water system (B.1.14)	No	Consistent with NUREG-1801. Only Dresden heat exchangers use this material/ environment combination.

Table 3.3-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.3.1.27	Components (aluminum bronze, brass, cast iron, cast steel) in open-cycle and closed-cycle cooling water systems, and ultimate heat sink	NUREG-1801 Components Piping and Fittings Valves Evaluated with NUREG-1801 Components Dampeners	Loss of material due to selective leaching	Selective leaching of materials (B.1.24)	No	Consistent with NUREG-1801, with exception. The exceptions to Selective Leaching of Materials are described in Section B.1.24. Brass or copper-nickel identified in NUREG-1801 line VII.C3.1-a (piping) and 3.2-a (valves) are not used in ultimate heat sink piping at Dresden or Quad Cities. Components in NUREG-1801 item VIIC1.1-c (buried cast iron components) are managed for selective leaching rather than general corrosion. Dresden and Quad Cities do not have material-environment combination evaluated in NUREG-1801 lines VII.C1.3-a. Dresden and Quad Cities pump and piping material is carbon steel in the open-cycle cooling water system, as discussed in NUREG-1801, lines VII.C1.5-a and VII.C3.1-a, and is therefore not susceptible to selective leaching. Dresden and Quad Cities pump material is carbon steel in the closed-cycle cooling water system, as discussed in NUREG-1801, lines VII.C2.3-a, and is therefore not susceptible to selective leaching. Brass and bronze instrument pulsation dampeners are only used at Quad Cities.

Table 3.3-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the auxiliary systems (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.3.1.28	Fire barriers, walls, ceilings and floors in fire protection	NUREG-1801 Components Walls, Ceilings, Floors Evaluated with NUREG-1801 Components Concrete Manholes Concrete Shield Plugs	Concrete cracking and spalling due to freeze-thaw, aggressive chemical attack, and reaction with aggregates; loss of material due to corrosion of embedded steel	Fire protection (B.1.18) and structures monitoring (B.1.30)	No	Consistent with NUREG-1801, with exception. The exceptions to structural aging effects due to aggressive chemical attack, reaction with aggregates, freeze-thaw & corrosion of embedded steel are described in Section 3.3.1.2.1. The exceptions to Fire Protection are described in Section B.1.18.

3.3.1.1 Further Evaluation of aging management as recommended by NUREG-1801 for the auxiliary systems

3.3.1.1.1 Loss of Material due to General, Pitting, and Crevice Corrosion (NUREG-1800, Section 3.3.2.2.1.1)

To confirm the effectiveness of aging management program Water Chemistry (B.1.2), a one-time inspection of the Dresden fuel pool cooling and filter demineralizer system will be performed. The one-time inspection will be either a visual or ultrasonic examination of a stainless steel component or piping in the system. The inspection will inspect for general, pitting, and crevice corrosion.

3.3.1.1.2 Loss of Material due to General, Pitting, and Crevice Corrosion (NUREG-1800, Section 3.3.2.2.1.2)

An inspection of selected components exposed to a stagnant flow water environment will be conducted in accordance with aging management program One-Time Inspection (B.1.23.) The inspection of selected components will verify the effectiveness of the chemistry control program to manage loss of material due to general, pitting, and crevice corrosion in low flow or stagnant flow areas by ensuring that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. Examinations will be conducted on carbon and stainless steel components in areas where typically stagnant flow is present but occasionally there is flow, which will cause replenishment of the oxygen supply. Inspections will be conducted on the HPCI torus suction check valves, the HPCI booster pumps, and the control rod drive (CRD) scram valves. These components were selected to provide representative samples of the aging effects seen in the shutdown cooling system. The carbon steel HPCI torus suction check valves are exposed to torus water while the carbon steel HPCI booster pumps and the stainless steel CRD scram valves are exposed to condensate storage tank water. Both will undergo a visual exam. This inspection is also credited for those components exposed to reactor coolant, which are outside NUREG-1801, line IV.C1.

3.3.1.1.3 Reduction of Neutron-Absorbing Capacity and Loss of Material due to General Corrosion (NUREG-1800, Section 3.3.2.2.10)

Reduction of neutron-absorbing capacity and loss of material due to general corrosion could occur at Dresden, in the boral neutron absorbing material in the spent fuel storage racks. Aging management program Water Chemistry (B.1.2) manages general corrosion. A one-time inspection of boral coupon test specimens at Dresden has been performed. The one-time inspection confirmed that significant aging degradation has not occurred and that neutron absorbing capability has not been reduced.

3.3.1.1.4 Loss of Material due to General, Pitting, Crevice, and Microbiologically Influenced Corrosion (NUREG-1800, Section 3.3.2.2.11)

Aging management program Buried Piping and Tanks Inspection (B.1.25) relies on industry practice, frequency of pipe excavations, and operating experience to manage the aging of buried components. Since Dresden and Quad Cities infrequently expose buried components during yard excavation activities, additional testing and inspection activities are credited. Specific aging management activities for buried components include:

1) Preventive measures (coatings and wrappings) to mitigate corrosion,

2) Periodic inspections of buried piping and tanks whenever they are excavated during yard area excavations (Structures Monitoring),

3) Periodic pressure testing of buried Class 3 cooling water piping,

4) Periodic pressure testing of buried Fire Mains,

5) Periodic inspection of the inside of buried fuel oil storage tanks (including 'one-time' ultrasonic test),

6) 'One-time' tank bottom ultrasonic test of an outdoor aluminum tank on an earthen foundation,

7) Periodic inspection of the ground above buried commodities for seepage and settling, and

8) 'One-time' excavation and inspection of a section of fire main piping if Item 2) doesn't occur prior to year 39.

3.3.1.1.5 Hardening and Cracking or Loss of Strength due to Elastomer Degradation or Loss of Material due to Wear (NUREG-1800, Section 3.3.2.2.2)

Aging management of control room, emergency diesel generator building, station blackout diesel generator building, and reactor building (using the requirements of the containment ventilation) ventilation system elastomers will be performed by the periodic inspection of elastomers in accordance with plant-specific aging management program Periodic Inspection of Ventilation System Elastomers (B.2.3.)

3.3.1.1.6 Cumulative Fatigue Damage (NUREG-1800, Section 3.3.2.2.3)

Cumulative fatigue damage of auxiliary system piping and load handling cranes is a TLAA as defined in 10 CFR 54.3. RWCU pumps identified by NUREG-1801 lines VII E3.2-b and VII E3.2-c are not in the scope of license renewal and are not evaluated as a TLAA. Cumulative fatigue damage of auxiliary system piping and load handling cranes is required to be evaluated in accordance with 10 CFR 54.21(c)(1). The TLAA evaluation of auxiliary system piping outside the RCPB is addressed in Section 4.3.3.2. The TLAA evaluation of load handling cranes is addressed in Section 4.7.1.

3.3.1.1.7 Loss of Material due to General, Microbiologically Influenced, Pitting, and Crevice Corrosion (NUREG-1800, Section 3.3.2.2.5)

Emergency diesel generator and station blackout diesel fuel oil piping external surface aging management in outdoor ambient conditions will be managed with system engineer walkdowns performed by Bolting Integrity (B.1.12) and Buried Piping and Tanks Inspection (B.1.25) aging management programs.

Emergency diesel generator and station blackout diesel auxiliaries, diesel generator cooling water, diesel fuel oil, shutdown cooling water (Dresden only), reactor water cleanup, process sampling, standby liquid control, control rod drive hydraulic system, fire protection (fire water portions), service water, ultimate heat sink, reactor building closed cooling water, turbine building closed cooling water (Dresden only), fuel pool cooling and filter demineralizer system (Dresden only), residual heat removal service water (Quad Cities only), containment cooling service water (Dresden only), plant heating system, drywell nitrogen inerting, demineralized water makeup, and safe shutdown makeup pump system (Quad Cities only) components external surface in a sheltered environment with warm, moist air will be managed either by the Structures Monitoring Program (B.1.30) or system engineer walkdowns performed by Bolting Integrity (B.1.12), and Structures Monitoring Program (B.1.30) aging management activities.

A visual inspection of selected components will be performed in accordance with the requirements of the Fire Protection (B.1.18) aging management program to verify the integrity of halon system (Dresden only), cardox system, and fire protection diesel-driven fire pump fuel oil subsystem external surfaces in a sheltered environment with warm, moist air.

Emergency diesel generator and station blackout diesel combustion air, and exhaust air, diesel fuel oil, drywell nitrogen inerting (nitrogen storage tank and outdoor components), and nitrogen containment atmosphere dilution external surface aging management of components in outdoor ambient conditions will be managed with system engineer walkdowns performed by Bolting Integrity (B.1.12). Emergency diesel generator and station blackout diesel combustion air, starting air, and exhaust air interior air environments will be managed with a One-Time Inspection (B.1.23) of compressed gas systems.

Air handler cooling coils for main control room HVAC, reactor building HVAC, SBO building HVAC (includes cooler frames and housings), and ECCS room cooler HVAC (includes cooler frames and housings) will be managed with Open-Cycle Cooling Water System (B.1.13) or Heat Exchanger Test and Inspection Activities (B.2.6) aging management programs depending on the cooling fluid environment. Other ventilation system metallic components, including equipment frames and housings, within the scope of license renewal for main control room HVAC, reactor building HVAC, and SBO building HVAC (with exception of cooler frames and housing), and emergency diesel generator HVAC will be managed with a One-Time Inspection (B.1.23) of the ventilation systems.

Components Requiring Aging Management for Loss of Material Due to General Corrosion

	Aging Management Program						
	Buried Piping and Tanks Inspection (B.1.25)	Bolting Integrity (B.1.12)	Structural Monitoring (B.1.30)	Fire Protection (B.1.18)	One-Time Inspection: Compressed Gas (B.1.23)	One-Time Inspection: Ventilation Systems (B.1.23)	Air Handler Cooling Coil AMPs (See Note 1)
Air Accumulator Vessels					X		
Air Handler Heating/Cooling Coils							X
Carbon Steel Components (See Note 2)		X	X	X			
Doors, Closure Bolts, Equipment Frames						X	X
Filter/Strainers					X		
Flame Arrestors					X		
Housing and Supports						X	
Lubricators					X		
Mufflers					X		
Piping and Fittings	X	X			X		
Pumps					X		
Valves					X		

Notes

1. Air handler cooling coil aging management programs include:
   - Open-Cycle Cooling Water (B.1.13)
   - Heat Exchanger Testing & Inspection (B.2.6)

2 Carbon Steel Components include: diffusers, filter/strainers, flow elements, heat exchangers, lubricators, piping & fittings, pumps, restricting orifices, sight glasses, sprinklers, tanks, thermowells, traps, tubing, and valves.

3.3.1.1.8 Loss of Material due to General, Pitting, Crevice, and Microbiologically Influenced Corrosion and Biofouling (NUREG-1800, Section 3.3.2.2.7)

An inspection will be performed in accordance with aging management program One-Time Inspection (B.1.23) to verify the effectiveness of Fuel Oil Chemistry (B.1.21) aging management program at preventing loss of material. An UT examination of the lower portion of one carbon steel underground fuel oil storage tank and one day tank at each facility will be performed. The Quad Cities Unit I underground fuel oil storage tank is constructed of fiberglass and aging management for loss of material is discussed separately as a non-NUREG-1801 item.

Activities to prevent biofouling of the fuel oil systems are performed in accordance with aging management program Fuel Oil Chemistry (B.1.21). Preventive activities include routine sampling to provide assurance that contaminant levels, including water, are kept at acceptable levels for fuel oil system components, and the addition of a biocide to the underground fuel oil storage tanks with each new fuel delivery.

3.3.1.2 Aging management programs or evaluations that are different than those described in NUREG-1801

3.3.1.2.1 Exception to structural aging effects due to aggressive chemical attack, reaction with aggregates, freeze-thaw & corrosion of embedded steel

Structures Monitoring Program (B.1.30) and Fire Protection (B.1.18) are required to manage the structural aging effects for concrete in accessible areas. No aging management is required to manage the following structural aging effects for concrete in inaccessible areas.

- Cracking and spalling due to aggressive chemical attack

Dresden and Quad Cities ground water test data obtained during construction, the 1980's, 1990's, and 2000's shows that the below-grade environment is not aggressive based on NUREG-1801 criteria with chlorides less than 500 ppm, sulfates less than 1500 ppm, and pH greater than 5.5. Examination of representative samples of below-grade concrete, when excavated for any reason, is included as part of aging management program Structures Monitoring Program (B.1.30.) To ensure conditions are maintained throughout the period of extended operations, aging management program Structures Monitoring Program (B.1.30) will be enhanced to include monitoring of below-grade water chemistry to demonstrate that the environment remains non-aggressive. Existing plant procedures will be used to periodically sample pH, chlorides, and sulfates.

- Cracking and spalling due to reaction with aggregates

Dresden and Quad Cities documented evidence demonstrates that the concrete used meets the requirements of ACI 201.2R-77 with no evidence of reactive aggregates. Therefore, concrete cracking and spalling due to reaction with aggregates in inaccessible areas are not applicable and no aging management is required.

- Cracking and spalling due to freeze-thaw

Dresden and Quad Cities are located in severe weathering conditions. Dresden and Quad Cities have documented evidence to show that the concrete air content is between 3% and 6%. Plant inspections did not show freeze-thaw degradation. Therefore, concrete cracking and spalling due to freeze-thaw in inaccessible areas are not applicable and no aging management is required.

- Loss of material due to corrosion of embedded steel

Dresden and Quad Cities ground water test data obtained during construction, the 1980's, 1990's, and 2000's shows that the below-grade environment is not aggressive based on NUREG-1801 criteria with chlorides less than 500 ppm, sulfates less than 1500 ppm, and pH greater than 5.5. Examination of representative samples of below-grade concrete, when excavated for any reason, is included as part of aging management program Structures Monitoring Program (B.1.30.) To ensure conditions are maintained throughout the period of extended operations, aging management program Structures Monitoring Program (B.1.30) will be enhanced to include monitoring of below-grade water chemistry to demonstrate that the environment remains non-aggressive. Existing plant procedures will be used to periodically sample pH, chlorides, and sulfates.

3.3.1.2.2 Exception to biofouling for a(2) components

Biofouling of components included in scope for criteria a(2) in a raw water environment does not require aging management to maintain the component intended function.

3.3.1.2.3 Exception to GALL for XI.M2, "Water Chemistry"

Aging of standby liquid control (SBLC) system components not in the reactor coolant pressure boundary section of the SBLC system relies on monitoring and control of SBLC makeup water chemistry. The makeup water is monitored in lieu of the storage tank because the sodium pentaborate maintained in the storage tank would mask most of the chemistry parameters monitored. The effectiveness of the water chemistry program will be verified by a one-time VT-3 inspection of a Dresden SBLC pump discharge valve and a Quad Cities SBLC pump casing as discussed in aging management program One-Time Inspection (B.1.23.)

3.3.2 Components or aging effects that are not addressed in NUREG-1801 for the auxiliary systems

Table 3.3-2, “Aging management review results for the auxiliary systems that are not addressed in NUREG-1801” contains aging management review results for the auxiliary systems for component groups that are not addressed in NUREG-1801.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.1	Accumulators	Carbon Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen and CO2) is not conducive to promoting aging degradation of carbon steel components.
3.3.2.2	Thermowells	Stainless Steel	Saturated Steam/ Condensate	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in a plant heating steam environment.
3.3.2.3	Accumulators	Carbon Steel	Sodium pentaborate solution at 21 - 32 °C (70 - 90°F) (˜24,500 ppm B)	Loss of material/ General galvanic pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not consider carbon steel components in a sodium pentaborate environment.
3.3.2.4	Fire Doors	Steel	Indoor and outdoor environments	Loss of material/ Wear	Fire Protection (B.1.18)	NUREG-1801 does not address Control Building and Reactor Building fire doors.
3.3.2.5	Accumulators	Stainless Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen) is not conducive to promoting aging degradation of stainless steel components.
3.3.2.6	Air Accumulator Vessels	Stainless Steel	Moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in a moist air environment.
3.3.2.7	Air Handlers Heating/ Cooling (Aux&RW HVAC)	Tubes: Copper	Tube side: Open cycle cooling water (raw water); Shell side: Warm moist air	Buildup of Deposit/ Fouling	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address raw water environments for air handler units in auxiliary systems.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.8	Air Handlers Heating/ Cooling (Aux&RW HVAC)	Tubes: Copper; Tubesheet: Stainless Steel; End Bells: Carbon Steel; Fins: Aluminum	Tube side: Open cycle cooling water (raw water); Shell side: Warm moist air	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Pitting Corrosion, Crevice Corrosion	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address stainless steel, carbon steel, aluminum subcomponents; or raw water environments; for air handler units in auxiliary systems.
3.3.2.9	Air Handlers Heating/ Cooling (Aux&RW HVAC)	Tubes: Copper; Tubesheet: Stainless Steel; End Bells: Carbon Steel; Fins: Aluminum	Tube side: Open cycle cooling water (raw water); Shell side: Warm moist air	Cracking/ Mech Fatigue, SCC	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address stainless steel, carbon steel, aluminum subcomponents; or raw water environments; for air handler units in auxiliary systems.
3.3.2.10	Air Handlers Heating/ Cooling (CR HVAC)	Tubes: Copper	Tube side: Refrigerant; Shell side: Warm moist air	Buildup of Deposit/ Fouling	Heat Exchanger Test and Inspection Activities (B.2.6)	NUREG-1801 does not address fouling for air handler units.
3.3.2.11	Air Handlers Heating/ Cooling (CR HVAC)	Tubes: Copper; Tubesheet: Copper; End Bells: Copper; Fins: Aluminum	Tube side: Refrigerant; Shell side: Warm moist air	Cracking/ Mech Fatigue	Heat Exchanger Test and Inspection Activities (B.2.6)	NUREG-1801 does not address aluminum subcomponents, or refrigerant environments for air handler units in auxiliary systems.
3.3.2.12	Air Handlers Heating/ Cooling (DGB HVAC)	Tubes: Copper	Tube side: Open cycle cooling water (raw water); Shell side: Warm moist air	Buildup of Deposit/ Fouling	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address raw water environments for air handler units in auxiliary systems.
3.3.2.13	Air Handlers Heating/ Cooling (DGB HVAC)	Tubes: Copper	Tube side: Refrigerant; Shell side: Warm moist air	Buildup of Deposit/ Fouling	Heat Exchanger Test and Inspection Activities (B.2.6)	NUREG-1801 does not address fouling for air handler units.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.14	Air Handlers Heating/ Cooling (DGB HVAC)	Tubes: Copper; End Bells: Galvanized Steel; Fins: Aluminum	Tube side: Refrigerant; Shell side: Warm moist air	Loss of Material/ Galvanic Corrosion, Wear, Pitting Corrosion, Crevice Corrosion	Heat Exchanger Test and Inspection Activities (B.2.6)	NUREG-1801 does not address aluminum or galvanized steel subcomponents, or refrigerant environments for air handler units in auxiliary systems.
3.3.2.15	Air Handlers Heating/ Cooling (DGB HVAC)	Tubes: Copper; Tubesheet: Carbon Steel; End Bells: Carbon Steel; Fins: Aluminum	Tube side: Open cycle cooling water (raw water); Shell side: Warm moist air	Cracking/ Mech Fatigue, SCC	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address carbon steel or aluminum subcomponents; or raw water environments; for air handler units in auxiliary systems.
3.3.2.16	Air Handlers Heating/ Cooling (DGB HVAC)	Tubes: Copper; Tubesheet: Carbon Steel; End Bells: Carbon Steel; Fins: Aluminum	Tube side: Open cycle cooling water (raw water); Shell side: Warm moist air	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Pitting Corrosion, Crevice Corrosion	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address carbon steel or aluminum subcomponents; or raw water environments; for air handler units in auxiliary systems.
3.3.2.17	Closure Bolting	Cast Iron	Soil and groundwater	Loss of Material/ Pitting, Crevice Corrosion, Selective Leaching and MIC	Selective Leaching of Materials (B.1.24) and Buried Piping & Tanks Inspection (B.1.25)	NUREG-1801 does not address buried Fire Main cast iron bolting.
3.3.2.18	Closure Bolting	High Strength Low Alloy Steel	Outdoor ambient conditions	Loss of Material/ General corrosion and wear	Bolting Integrity (B.1.12)	NUREG-1801 does not address high strength low alloy steel bolting in an outdoor environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.19	Closure Bolting	High Strength Low Alloy Steel	Raw water (submerged)	Loss of material/ General, pitting, crevice, microbiologically influenced corrosion and macro organisms	Bolting Integrity (B.1.12)	NUREG-1801does not address pump casing bolting in a submerged (raw water) environment.
3.3.2.20	Closure Bolting	High Strength Low Alloy Steel	Soil and groundwater	Loss of material/ Pitting crevice and microbiologically influenced corrosion	Buried Piping and Tanks Inspection (B.1.25)	NUREG-1801 does not address buried Fire Main high strength low alloy steel bolting.
3.3.2.21	Component External Surfaces (filters/strainers, piping and fittings, heat exchangers, lubricators, vaporizers (tanks), air handlers (heating/cooling))	Aluminum	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address aluminum in plant indoor environment. Aluminum is a reactive metal, but it develops an aluminum oxide film that protects it from further corrosion. Therefore, no viable aging effects exist in this environment.
3.3.2.22	Component External Surfaces (piping and fittings, tubing)	Aluminum	Outdoor ambient conditions	Loss of material/ Pitting	Bolting Integrity (B.1.12)	NUREG-1801 does not address aluminum components in plant outdoor ambient conditions.
3.3.2.23	Component External Surfaces (valves, filters/ strainers, thermowells, dampers, piping and fittings, restricting orifices)	Brass or Bronze	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address brass or bronze in the plant indoor environment. The plant indoor environment is not an aggressive wetted environment conducive to promoting aging degradation of brass or bronze components.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.24	Component External Surfaces (valves, dampeners)	Brass or Bronze	Air, moisture, humidity, and leaking fluid	Loss of material/ Pitting and crevice corrosion	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address Auxiliary System brass or bronze components in a high moisture (pump vault) indoor environment.
3.3.2.25	Component External Surfaces (valves, filters/strainers)	Brass or Bronze	Outdoor ambient conditions	Loss of material/ Pitting and crevice corrosion	Bolting Integrity (B.1.12)	NUREG-1801 does not address auxiliary system brass or bronze components in an outdoor environment.
3.3.2.26	Component External Surfaces (piping and fittings, sprinklers, pumps, valves, thermowells, heat exchangers)	Carbon Steel	Air, moisture, humidity, and leaking fluid	Loss of material/ General pitting crevice corrosion	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address auxiliary system carbon steel components in a high moisture (pump vault) indoor environment.
3.3.2.27	Component External Surfaces (piping and fittings, valves)	Carbon Steel	Containment Nitrogen	None	None	NUREG-1801 does not address carbon steel components in a containment nitrogen environment. Containment nitrogen is not conducive to promoting aging degradation of carbon steel.
3.3.2.28	Component External Surfaces (piping and fittings)	Carbon Steel	Encased in Concrete	None	None	NUREG-1801 does not address steel piping encased in concrete. EPRI Tools Appendix E, Rev. 3 concludes that concrete is not conducive to promoting aging degradation of carbon steel components.
3.3.2.29	Component External Surfaces (piping and fittings, filters/strainers, mufflers, heat exchangers, equipment enclosures, valves)	Carbon Steel	Outdoor ambient conditions	Loss of material/ General pitting crevice corrosion	Bolting Integrity (B.1.12)	NUREG-1801 does not address carbon steel components in plant outdoor ambient conditions.
3.3.2.30	Component External Surfaces (piping and fittings)	Carbon Steel	Outdoor ambient conditions	Loss of material/ General pitting crevice corrosion	Fire Water System (B.1.19)	NUREG-1801 does not address carbon steel fire protection piping in plant outdoor ambient conditions.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.31	Component External Surfaces (valves, filters/strainers, pumps, heat exchangers, traps)	Cast Iron	Air, moisture, humidity, and leaking fluid	Loss of material/ Pitting and crevice corrosion	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address auxiliary system cast iron components in a high moisture (pump vault) indoor environment.
3.3.2.32	Component External Surfaces (fire hydrants, valves)	Cast Iron	Outdoor ambient conditions	Loss of material/ Pitting and crevice corrosion	Fire Water System (B.1.19)	NUREG-1801 does not address cast iron fire protection components in an outdoor environment.
3.3.2.33	Component External Surfaces (piping and fittings)	Cast Iron	Soil and groundwater	Loss of Material/ Pitting, Crevice Corrosion, Selective Leaching and MIC	Selective Leaching of Materials (B.1.24) and Buried Piping & Tanks Inspection (B.1.25)	NUREG-1801 does not address buried cast iron fire main pipe fittings.
3.3.2.34	Component External Surfaces (tubing, heat exchangers, piping and fittings)	Copper	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address copper in the plant indoor environment. The plant indoor environment is not an aggressive wetted environment conducive to promoting aging degradation of copper components.
3.3.2.35	Component External Surfaces (tubing, restricting orifices)	Copper	Outdoor ambient conditions	Loss of material/ Pitting and crevice corrosion	Bolting Integrity (B.1.12)	NUREG-1801 does not address copper components in plant outdoor ambient conditions.
3.3.2.36	Component External Surfaces (sight glasses)	Glass	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address glass components in a plant indoor environment. A plant indoor environment is not conducive to promoting aging degradation of glass components. Glass in this environment does not have any applicable aging effect.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.37	Component External Surfaces (valves)	Iron Ductile	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address ductile iron valves in a plant indoor environment. The plant indoor environment is not an aggressive wetted environment conducive to promoting aging degradation of ductile iron components.
3.3.2.38	Component External Surfaces (sprinklers, piping and fittings)	Iron Malleable	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address malleable iron components in the plant indoor environment. The plant indoor environment is not an aggressive wetted environment conducive to promoting aging degradation of malleable iron components.
3.3.2.39	Component External Surfaces (piping and fittings)	Polyvinyl Chloride (PVC)	Soil and groundwater	None	None	NUREG-1801 does not address PVC in a buried environment. PVC is relatively unaffected by water, concentrated alkaline, and non-oxidizing acids, oils, and ozone. Therefore, no viable aging effects exist in this environment.
3.3.2.40	Component External Surfaces (valves, piping and fittings, filters/ strainers, restricting orifices, orifice bodies, , dampeners, accumulators, flow elements, tubing, manifolds, rupture discs, pumps, housing and supports, thermowells, tanks, vaporizers (tanks))	Stainless Steel	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address stainless steel alloys in the plant indoor environment. Stainless steel materials are not subject to any viable aging mechanism in the absence of aggressive chemical species.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.41	Component External Surfaces (restricting orifices, orifice bodies, valves, tubing, pulsation dampeners)	Stainless Steel	Air, moisture, humidity, and leaking fluid	Loss of material/ Pitting and crevice corrosion	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address auxiliary system stainless steel components in a high moisture (pump vault) indoor environment.
3.3.2.42	Component External Surfaces (piping and fittings, valves, tubing)	Stainless Steel	Containment Nitrogen	None	None	NUREG-1801 does not address stainless steel components in a containment nitrogen environment. Containment nitrogen is not conducive to promoting aging degradation of stainless steel alloys.
3.3.2.43	Component External Surfaces (piping and Fittings, tubing)	Stainless Steel	Outdoor ambient conditions	None	None	NUREG-1801 does not address stainless steel in the plant outdoor environment. Stainless steel materials are not subject to any viable aging mechanism in the absence of aggressive chemical species.
3.3.2.44	Component External Surfaces (valves, piping and fittings)	Steel Saran Lined	Air, moisture, humidity, and leaking fluid	Loss of material/ General pitting crevice corrosion	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address saran lined steel components in a plant indoor environment. The external surface of the components is unlined carbon steel.
3.3.2.45	Component External Surfaces (valves, piping and fittings)	Titanium	Air, moisture, humidity, and leaking fluid	None	None	NUREG-1801 does not address titanium components. The high moisture (pump vault) indoor environment does not promote aging degradation of these titanium components as they are not exposed to high chloride concentrations at high temp.
3.3.2.46	Dampeners	Brass or Bronze	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address brass or bronze components in a warm, moist air environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.47	Dampeners	Stainless Steel	Sodium pentaborate solution at 21 - 32 °C (70 - 90°F) (˜24,500 ppm B)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not consider loss of material for components in a sodium pentaborate environment.
3.3.2.48	Diffusers	Carbon Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (refrigerant) is not conducive to promoting aging degradation of carbon steel components.
3.3.2.49	Duct Fittings, Hinges, Latches	Aluminum-Zinc Alloy	Warm, moist air	None	None	NUREG-1801 does not address components made of aluminum-zinc alloy. A warm, moist air environment is not conducive to promoting aging degradation of aluminum-zinc alloy components.
3.3.2.50	Filters/ Strainers	Aluminum	Diesel fuel oil	Loss of Material/ General pitting crevice and microbiologically influenced corrosion	Fuel Oil Chemistry (B.1.21)	NUREG-1801 does not address aluminum components in a fuel oil environment.
3.3.2.51	Filters/ Strainers	Brass or Bronze	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen) is not conducive to promoting aging degradation of brass or bronze components.
3.3.2.52	Filters/ Strainers	Brass or Bronze	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address brass or bronze components in a saturated air environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.53	Filters/ Strainers	Carbon Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (refrigerant) is not conducive to promoting aging degradation of carbon steel components.
3.3.2.54	Filters/ Strainers	Cast Iron	Diesel fuel oil	Loss of Material/ General pitting crevice and microbiologically influenced corrosion	Fuel Oil Chemistry (B.1.21)	NUREG-1801 does not address cast iron components in a fuel oil environment.
3.3.2.55	Filters/ Strainers	Cast Iron	Moist air	Loss of material/ General, pitting, and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address cast iron components in a moist air environment.
3.3.2.56	Filters/ Strainers	Cast Iron	Raw water	Loss of Material/ Selective Leaching	Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address selective leaching of cast iron in fire protection systems.
3.3.2.57	Filters/ Strainers	Cast Iron	Saturated Steam/ Condensate	Loss of material/ General corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address cast iron components in a plant heating steam environment.
3.3.2.58	Filters/ Strainers	Stainless Steel	Moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in a moist air environment.
3.3.2.59	Debris Screens	Stainless Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel debris screens and strainer elements.
3.3.2.60	Filters/ Strainers	Carbon Steel	Lubricating oil (with contaminants and/ or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.61	Fire Hydrants	Cast Iron	Raw water	Loss of Material/ Selective Leaching	Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address selective leaching of cast iron in fire protection systems.
3.3.2.62	Fire Proofing	Cementitious Fire Proofing	Indoor	None	None	NUREG-1801 does not address cementitious fireproofing in an indoor environment. A non-aggressive, vibration free plant indoor environment is not conductive to promoting aging of cementitious fireproofing.
3.3.2.63	Fire Wrap	Ceramic Fiber	Indoor	None	None	NUREG-1801 does not address ceramic fiber. Fire wrap in a plant indoor environment is not conducive to promoting aging of ceramic fiber firewrap.
3.3.2.64	Flexible Hoses	Elastomers Neoprene and Similar Materials	Dry Gas	Hardening and loss of strength/ Elastomer degradation	One-Time Inspection (B.1.23)	NUREG-1801 does not address components in a dry gas (moisture free) environment.
3.3.2.65	Flexible Hoses	Elastomers Neoprene and Similar Materials	Moist air	Hardening and loss of strength/ Elastomer degradation	One-Time Inspection (B.1.23)	NUREG-1801 does not address elastomers in a moist air environment.
3.3.2.66	Flexible Hoses	Elastomers Neoprene and Similar Materials	Saturated air	Hardening and loss of strength/ Elastomer degradation	One-Time Inspection (B.1.23)	NUREG-1801 does not address elastomers in a saturated air environment.
3.3.2.67	Flexible Hoses	Elastomers Neoprene and Similar Materials	Warm, moist air	Hardening and loss of strength/ Elastomer degradation	One-Time Inspection (B.1.23)	NUREG-1801 does not address flexible hoses in a warm, moist air environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.68	Flow Elements	Carbon Steel	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider cracking in closed cooling water environments.
3.3.2.69	Flow Elements	Carbon Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen) is not conducive to promoting aging degradation of carbon steel components.
3.3.2.70	Flow Elements	Stainless Steel	Chemically treated demineralized water <90°C (194°F) (Glycol based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider stainless steel in a diesel generator cooling water subsystem.
3.3.2.71	Flow Elements	Stainless Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen) is not conducive to promoting aging degradation of stainless steel components.
3.3.2.72	Flow Elements	Stainless Steel	Treated water	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non-safety related components in a treated water environment.
3.3.2.73	Flow Elements	Stainless Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel flow elements in a warm, moist air environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.74	Fuel Grapples	Stainless Steel	Chemically treated oxygenated water	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address the stainless steel fuel grapple.
3.3.2.75	Fuel Pool Gates	Aluminum	Chemically treated oxygenated water	Loss of material/ General and pitting corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address aluminum in a fuel pool environment.
3.3.2.76	Fuel Preparation Machines	Aluminum	Chemically treated oxygenated water	Loss of material/ General and pitting corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address aluminum in a fuel pool environment.
3.3.2.77	Heat Exchangers	Channel Head: Carbon Steel; Shell: Carbon Steel	Tube side: Open cycle cooling water (raw water); Shell side: Closed cooling water (treated water)	Crack initiation and growth/ Stress corrosion cracking	Open-Cycle Cooling Water System (B.1.13), Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not evaluate this heat exchanger and intended function combination.
3.3.2.78	Heat Exchangers	Channel Head: Carbon Steel; Shell: Carbon Steel	Tube side: Open cycle cooling water (raw water); Shell side: Closed cooling water (treated water)	Loss of Material/ General, MIC, Erosion/ FAC, Wear, Pitting, Crevice Corrosion	Open-Cycle Cooling Water System (B.1.13), Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not evaluate this heat exchanger and intended function combination.
3.3.2.79	Heat Exchangers	Tubes: 70-30 Cu-Ni	Tube side: Open cycle cooling water (raw water); Shell side: Torus Water (demineralized water)	Buildup of Deposit/ Fouling	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address Cu-Ni tubing for LPCI heat exchangers. Fouling from non-microbiological sources in not included in NUREG-1801.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.80	Heat Exchangers	Tubes: 70-30 Cu-Ni; Tubesheet: Carbon Steel; Channel Head: Carbon Steel; Shell: Carbon Steel	Tube side: Open cycle cooling water (raw water); Shell side: Torus Water (demineralized water)	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Selective Leaching, Pitting Corrosion, Crevice Corrosion	Water Chemistry (B.1.2), Open-Cycle Cooling Water System (B.1.13), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address Cu-Ni tubing for LPCI heat exchangers.
3.3.2.81	Heat Exchangers	Tubes: 70-30 Cu-Ni; Tubesheet: Carbon Steel; Channel Head: Carbon Steel; Shell: Carbon Steel	Tube side: Open cycle cooling water (raw water); Shell side: Torus Water (demineralized water)	Cracking/ Mech Fatigue, SCC	Water Chemistry (B.1.2), Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address Cu-Ni tubing for LPCI heat exchangers.
3.3.2.82	Heat Exchangers	Tubes: 90-10 Cu-Ni	Tube side: Open cycle cooling water (raw water); Shell side: Refrigerant	Buildup of Deposit/ Fouling	Open-Cycle Cooling Water System (B.1.13)	This heat exchanger does not match the material/ environment combination evaluated in NUREG-1801.
3.3.2.83	Heat Exchangers	Tubes: 90-10 Cu-Ni	Tube side: Reactor coolant water; Shell side: Closed-cycle cooling water	Buildup of Deposit/ Fouling	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address Cu-Ni tubes for shutdown cooling heat exchangers.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.84	Heat Exchangers	Tubes: 90-10 Cu-Ni; Shell: Carbon Steel	Tube side: Open cycle cooling water (raw water); Shell side: Refrigerant	Cracking/ Mech Fatigue, SCC	Open-Cycle Cooling Water System (B.1.13)	This heat exchanger does not match the material/ environment combination evaluated in NUREG-1801.
3.3.2.85	Heat Exchangers	Tubes: 90-10 Cu-Ni; Shell: Carbon Steel	Tube side: Open cycle cooling water (raw water); Shell side: Refrigerant	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Pitting Corrosion, Crevice Corrosion	Open-Cycle Cooling Water System (B.1.13)	This heat exchanger does not match the material/ environment combination evaluated in NUREG-1801.
3.3.2.86	Heat Exchangers	Tubes: 90-10 Cu-Ni; Tubesheet: Carbon Steel; Channel Head: Carbon Steel; Shell: Carbon Steel	Tube side: Reactor coolant water; Shell side: Closed-cycle cooling water	Cracking/ Mech Fatigue, SCC	Water Chemistry (B.1.2), Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address Cu-Ni tubes for shutdown cooling heat exchangers.
3.3.2.87	Heat Exchangers	Tubes: 90-10 Cu-Ni; Tubesheet: Carbon Steel; Channel Head: Carbon Steel; Shell: Carbon Steel	Tube side: Reactor coolant water; Shell side: Closed-cycle cooling water	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Pitting Corrosion, Crevice Corrosion	Water Chemistry (B.1.2), Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address Cu-Ni tubes for shutdown cooling heat exchangers. General corrosion, galvanic corrosion, erosion or FAC, and wear are not included in NUREG-1801 for shutdown cooling heat exchangers.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.88	Heat Exchangers	Tubes: Admiralty Brass	Tube side: Open cycle cooling water (raw water); Shell side: Torus Water (demineralized water)	Buildup of Deposit/ Fouling	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address admiralty brass tubing for RHR system auxiliary heat exchangers.
3.3.2.89	Heat Exchangers	Tubes: Admiralty Brass; Channel Head: Cast Iron; Shell: Carbon Steel	Tube side: Open cycle cooling water (raw water); Shell side: Torus Water (demineralized water)	Cracking/ Mech Fatigue, SCC	Water Chemistry (B.1.2), Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address admiralty brass or cast iron subcomponents for RHR system auxiliary heat exchangers.
3.3.2.90	Heat Exchangers	Tubes: Admiralty Brass; Channel Head: Cast Iron; Shell: Carbon Steel	Tube side: Open cycle cooling water (raw water); Shell side: Torus Water (demineralized water)	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Selective Leaching, Pitting Corrosion, Crevice Corrosion	Water Chemistry (B.1.2), Open-Cycle Cooling Water System (B.1.13), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address admiralty brass or cast iron subcomponents for RHR system auxiliary heat exchangers. Galvanic corrosion, erosion or FAC, and wear are not included in NUREG-1801 for ECCS heat exchangers.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.91	Heat Exchangers	Tubes: Admiralty Brass; Tubesheet: Carbon Steel; Channel Head: Carbon Steel	Raw, untreated fresh water	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Selective Leaching, Pitting Corrosion, Crevice Corrosion	Open-Cycle Cooling Water System (B.1.13), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address admiralty brass tubes or carbon steel tubesheets for open-cycle heat exchangers in auxiliary systems.
3.3.2.92	Heat Exchangers	Tubes: Admiralty Brass; Tubesheet: Carbon Steel; Channel Head: Carbon Steel	Raw, untreated fresh water	Cracking/ Mech Fatigue, SCC	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address admiralty brass tubes or carbon steel tubesheets for open-cycle heat exchangers in auxiliary systems.
3.3.2.93	Heat Exchangers	Tubes: Austenitic Stainless Steel	Tube side: Open cycle cooling water (raw water); Shell side: Closed cooling water (treated water)	Buildup of Deposit/ Fouling	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address stainless steel tubing in heat exchangers between open/ closed-cycle cooling systems.
3.3.2.94	Heat Exchangers	Tubes: Austenitic Stainless Steel; Tubesheet: Carbon Steel; Shell: Carbon Steel	Tube side: Open cycle cooling water (raw water); Shell side: Closed cooling water (treated water)	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Pitting Corrosion, Crevice Corrosion	Open-Cycle Cooling Water System (B.1.13), Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address stainless steel tubing in heat exchangers between open/ closed-cycle cooling systems. Erosion or FAC, and wear are not included in NUREG-1801 auxiliary systems for heat exchangers.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.95	Heat Exchangers	Tubes: Austenitic Stainless Steel; Tubesheet: Carbon Steel; Shell: Carbon Steel	Tube side: Open cycle cooling water (raw water); Shell side: Closed cooling water (treated water)	Cracking/ Mech Fatigue, SCC	Open-Cycle Cooling Water System (B.1.13), Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address stainless steel tubing in heat exchangers between open/ closed-cycle cooling systems.
3.3.2.96	Heat Exchangers	Tubes: Copper	Tube side: Closed cooling water; Shell side: Warm moist air	Buildup of Deposit/ Fouling	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address copper tubes or air environments for auxiliary system heat exchangers.
3.3.2.97	Heat Exchangers	Tubes: Copper	Tube side: Glycol-based cooling water (closed-cycle cooling water); Shell side: Lubricating oil	Buildup of Deposit/ Fouling	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address lube oil coolers.
3.3.2.98	Heat Exchangers	Tubes: Copper	Tube Side: Glycol-based cooling water (closed-cycle cooling water); Shell side: Warm moist air	Buildup of Deposit/ Fouling	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address copper tubes or air environments for auxiliary system heat exchangers.
3.3.2.99	Heat Exchangers	Tubes: Copper	Tube side: Lubricating Oil; Shell side: Closed cooling water	Buildup of Deposit/ Fouling	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address lube oil coolers.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.100	Heat Exchangers	Tubes: Copper; Channel Head: Cast Steel; Shell: Cast Steel	Tube side: Closed cooling water; Shell side: Warm moist air	Cracking/ Mech Fatigue, SCC	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address copper tubes or air environments for auxiliary system heat exchangers.
3.3.2.101	Heat Exchangers	Tubes: Copper; Channel Head: Cast Steel; Shell: Cast Steel	Tube side: Closed cooling water; Shell side: Warm moist air	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Pitting Corrosion, Crevice Corrosion	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address copper tubes or air environments for auxiliary system heat exchangers.
3.3.2.102	Heat Exchangers	Tubes: Copper; Channel Head: Cast Steel; Shell: Cast Steel	Tube Side: Glycol-based cooling water (closed-cycle cooling water); Shell side: Warm moist air	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Pitting Corrosion, Crevice Corrosion	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address copper tubes, cast steel subcomponents or air environments for auxiliary system heat exchangers.
3.3.2.103	Heat Exchangers	Tubes: Copper; Channel Head: Cast Steel; Shell: Cast Steel	Tube Side: Glycol-based cooling water (closed-cycle cooling water); Shell side: Warm moist air	Cracking/ Mech Fatigue	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address copper tubes, cast steel subcomponents or air environments for auxiliary system heat exchangers.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.104	Heat Exchangers	Tubes: Copper; Tubesheet: Carbon Steel; Channel Head: Carbon Steel; Fins: Aluminum	Tube Side: Glycol-based cooling water (closed-cycle cooling water); Shell side: Warm moist air	Cracking/ Mech Fatigue	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address copper tubes, aluminum subcomponents or air environments for auxiliary systems heat exchangers.
3.3.2.105	Heat Exchangers	Tubes: Copper; Tubesheet: Carbon Steel; Channel Head: Carbon Steel; Fins: Aluminum	Tube Side: Glycol-based cooling water (closed-cycle cooling water); Shell side: Warm moist air	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Pitting Corrosion, Crevice Corrosion	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address copper tubes, aluminum subcomponents or air environments for auxiliary systems heat exchangers.
3.3.2.106	Heat Exchangers	Tubes: Copper; Tubesheet: Carbon Steel; Channel Head: Cast Iron; Shell: Carbon Steel	Tube side: Glycol-based cooling water (closed-cycle cooling water); Shell side: Lubricating oil	Cracking/ Mech Fatigue	Closed-Cycle Cooling Water System (B.1.14), Lube Oil Monitoring Activities (B.2.5)	NUREG-1801 does not address lube oil coolers.
3.3.2.107	Heat Exchangers	Tubes: Copper; Tubesheet: Carbon Steel; Channel Head: Cast Iron; Shell: Carbon Steel	Tube side: Glycol-based cooling water (closed-cycle cooling water); Shell side: Lubricating oil	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Selective Leaching	Closed-Cycle Cooling Water System (B.1.14), Selective Leaching of Materials (B.1.24), Lubricating Oil Monitoring Activities (B.2.5)	NUREG-1801 does not address lube oil coolers. Erosion or FAC, and wear are not addressed in NUREG-1801 for heat exchangers.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.108	Heat Exchangers	Tubes: Copper; Tubesheet: Carbon Steel; Channel Head: Cast Iron; Shell: Carbon Steel	Tube side: Lubricating Oil; Shell side: Closed cooling water	Cracking/ Mech Fatigue, SCC	Closed-Cycle Cooling Water System (B.1.14), Lube Oil Monitoring Activities (B.2.5)	NUREG-1801 does not address lube oil coolers.
3.3.2.109	Heat Exchangers	Tubes: Copper; Tubesheet: Carbon Steel; Channel Head: Cast Iron; Shell: Carbon Steel	Tube side: Lubricating Oil; Shell side: Closed cooling water	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Pitting Corrosion, Crevice Corrosion	Closed-Cycle Cooling Water System (B.1.14), Lube Oil Monitoring Activities (B.2.5)	NUREG-1801 does not address lube oil coolers.
3.3.2.110	Heat Exchangers	Tubes: Cu-Ni; Tubesheet: Carbon Steel; Channel Head: Carbon Steel; Shell: Cu-Ni	Tube side: Open cycle cooling water (raw water); Shell side: Closed cooling water (treated water)	Cracking/ Mech Fatigue, SCC	Open-Cycle Cooling Water System (B.1.13), Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address carbon steel tubesheets, or Cu-Ni shells in heat exchangers between open/ closed-cycle cooling systems in auxiliary systems.
3.3.2.111	Heat Exchangers	Tubes: Cu-Ni; Tubesheet: Carbon Steel; Channel Head: Carbon Steel; Shell: Cu-Ni	Tube side: Open cycle cooling water (raw water); Shell side: Closed cooling water (treated water)	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Selective Leaching, Pitting Corrosion, Crevice Corrosion	Open-Cycle Cooling Water System (B.1.13), Closed-Cycle Cooling Water System (B.1.14), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address carbon steel tubesheets, or Cu-Ni shells in heat exchangers between open/ closed-cycle cooling systems. Erosion or FAC and wear are not included in NUREG-1801 auxiliary systems for heat exchangers.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.112	Heat Exchangers	Tubes: stainless steel	Tube side: Open cycle cooling water (raw water); Shell side: Closed cooling water (treated water)	Buildup of Deposit/ Fouling	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address stainless steel tubing in heat exchangers between open/ closed-cycle cooling systems.
3.3.2.113	Heat Exchangers	Tubes: stainless steel	Tube side: Open cycle cooling water (raw water); Shell side: Torus Water (demineralized water)	Buildup of Deposit/ Fouling	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address stainless steel tubing in auxiliary system heat exchangers exposed to raw water environments.
3.3.2.114	Heat Exchangers	Tubes: stainless steel	Tube side: Reactor coolant water; Shell side: Closed-cycle cooling water	Buildup of Deposit/ Fouling	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address fouling of stainless steel tubing in shutdown cooling system heat exchangers.
3.3.2.115	Heat Exchangers	Tubes: stainless steel	Tube side: Torus Water (demineralized water); Shell side: Open cycle cooling water (raw water)	Buildup of Deposit/ Fouling	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address stainless steel tubing in auxiliary system heat exchangers exposed to raw water environments.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.116	Heat Exchangers	Tubes: stainless steel; tubesheet: carbon steel; channel head: carbon steel; shell: carbon steel	Tube side: Open cycle cooling water (raw water); Shell side: Closed cooling water (treated water)	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Pitting Corrosion, Crevice Corrosion	Open-Cycle Cooling Water System (B.1.13), Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address stainless steel tubing in heat exchangers between open/ closed-cycle cooling systems. Erosion or FAC, and wear are not included in NUREG-1801 auxiliary systems for heat exchangers.
3.3.2.117	Heat Exchangers	Tubes: stainless steel; tubesheet: carbon steel; channel head: carbon steel; shell: carbon steel	Tube side: Open cycle cooling water (raw water); Shell side: Closed cooling water (treated water)	Cracking/ Mech Fatigue, SCC	Open-Cycle Cooling Water System (B.1.13), Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address stainless steel tubing in heat exchangers between open/ closed-cycle cooling systems.
3.3.2.118	Heat Exchangers	Tubes: Stainless Steel; Tubesheet: Cast Iron; Channel Head: Cast Iron; Shell: Carbon Steel	Tube side: Open cycle cooling water (raw water); Shell side: Torus Water (demineralized water)	Cracking/ Mech Fatigue, SCC	Water Chemistry (B.1.2), Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address stainless steel tubing in auxiliary system heat exchangers exposed to a raw water environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.119	Heat Exchangers	Tubes: Stainless Steel; Tubesheet: Cast Iron; Channel Head: Cast Iron; Shell: Carbon Steel	Tube side: Open cycle cooling water (raw water); Shell side: Torus Water (demineralized water)	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Selective Leaching, Pitting Corrosion, Crevice Corrosion	Water Chemistry (B.1.2), Open-Cycle Cooling Water System (B.1.13), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address stainless steel tubing in auxiliary system heat exchangers exposed to a raw water environment. Erosion or FAC, and wear are not included in NUREG-1801 for ECCS heat exchangers.
3.3.2.120	Heat Exchangers	Tubes: Stainless Steel; Tubesheet: Cast Iron; Shell: Cast Iron	Tube side: Torus Water (demineralized water); Shell side: Open cycle cooling water (raw water)	Loss of Material/ General Corrosion, Galvanic Corrosion, MIC, Erosion or FAC, Wear, Selective Leaching, Pitting Corrosion, Crevice Corrosion	Water Chemistry (B.1.2), Open-Cycle Cooling Water System (B.1.13), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address stainless steel tubing in auxiliary system heat exchangers exposed to raw water environments. Galvanic corrosion, erosion or FAC, and wear are not included in NUREG-1801 for ECCS heat exchangers.
3.3.2.121	Heat Exchangers	Tubes: Stainless Steel; Tubesheet: Cast Iron; Shell: Cast Iron	Tube side: Torus Water (demineralized water); Shell side: Open cycle cooling water (raw water)	Cracking/ Mech Fatigue, SCC	Water Chemistry (B.1.2), Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address stainless steel tubing in auxiliary system heat exchangers exposed to raw water environments.
3.3.2.122	Insulation	Closed-Cell Foam	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address closed-cell foam insulation. Closed-cell foam insulation is susceptible to degradation when exposed to UV light. Plant indoor environment is not conducive to promoting aging degradation of closed-cell foam insulation.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.123	Insulation Jacketing	Aluminum	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address aluminum insulation jacketing. Aluminum is reactive, but develops an oxide film that protects it from further corrosion. No viable aging effects exist in the indoor environment for aluminum insulation jacketing.
3.3.2.124	Isolation Barriers	Carbon Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	10 CFR Part 50, Appendix J (B.1.28)	NUREG-1801 does not address carbon steel components in a warm, moist air environment.
3.3.2.125	Isolation Barriers	Stainless Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	10 CFR Part 50, Appendix J (B.1.28)	NUREG-1801 does not address stainless steel components in a warm, moist air environment.
3.3.2.126	Lubricators	Aluminum	Moist air	Loss of material/ General and pitting corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components made of aluminum.
3.3.2.127	Lubricators	Glass	Moist air	None	None	NUREG-1801 does not address glass components. A diesel staring air environment is not conducive to promoting aging degradation of glass components. Glass in this environment does not have any applicable aging effect.
3.3.2.128	Manifolds	Stainless Steel	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider cracking in closed cooling water environments.
3.3.2.129	Masonry Walls	Concrete Block	Ambient environment inside building	Cracking/ Restraint; shrinkage; creep; aggressive environment	Fire Protection (B.1.18) and Structures Monitoring Program (B.1.30)	NUREG-1801 does not address masonry walls with an aging management program of fire protection.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.130	NSR Vents or Drains, Piping and Valves	Carbon Steel, Stainless Steel, Brass or Bronze	Air, moisture, humidity, and leaking fluid	Loss of material/ Corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address NSR vents or drains in an air, moisture, humidity and leaking fluid environment.
3.3.2.131	Pipe Joint Gaskets	Rubber	Soil and groundwater	Change in Material Properties/ Elastomer degradation and loss of resiliency	Buried Piping and Tanks Inspection (B.1.25)	NUREG-1801 does not address Auxiliary System rubber pipe gasket material in a buried environment.
3.3.2.132	Piping and Fittings	Aluminum	Moist air	Loss of material/ General and pitting corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components made of aluminum.
3.3.2.133	Piping and Fittings	Brass or Bronze	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Cracking/ Stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider brass or bronze in closed cooling water environments.
3.3.2.134	Piping and Fittings	Brass or Bronze	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Loss of Material/ General, Galvanic, Pitting, Crevice Corrosion, Selective Leaching and MIC	Closed-Cycle Cooling Water System (B.1.14), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not consider brass or bronze in closed cooling water environments.
3.3.2.135	Piping and Fittings	Brass or Bronze	Diesel fuel oil	Loss of material/ General pitting crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address brass or bronze components in a fuel oil environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.136	Piping and Fittings	Brass or Bronze	Lubricating oil (with contaminants and/ or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.3.2.137	Piping and Fittings	Carbon Steel	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider cracking in closed cooling water environments.
3.3.2.138	Piping and Fittings	Carbon Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (halon, CO2, or nitrogen) is not conducive to promoting aging degradation of carbon steel components.
3.3.2.139	Piping and Fittings	Carbon Steel	Lubricating oil (with contaminants and/ or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel components in a lubricating oil environment.
3.3.2.140	Piping and Fittings	Carbon Steel	Oxygenated water 93°C - 288°C (200°F-550°F)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel components in oxygenated water environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.141	Piping and Fittings	Carbon Steel	Raw water (submerged)	Loss of material/ General, pitting, crevice, microbiologically influenced corrosion and macro organisms	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address Auxiliary System carbon steel piping in a submerged (raw water) environment.
3.3.2.142	Piping and Fittings	Carbon Steel	Saturated Steam/ Condensate	Loss of material/ General corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel components in a plant heating steam environment.
3.3.2.143	Piping and Fittings	Carbon Steel	Treated water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non-safety related components in a treated water environment.
3.3.2.144	Piping and Fittings	Carbon Steel	Warm, moist air	Loss of material/ General (carbon steel only) pitting and crevice corrosion	Fire Water System (B.1.19)	NUREG-1801 does not address warm, moist air as an environment in which carbon steel is to be managed.
3.3.2.145	Piping and Fittings	Carbon Steel	Warm, moist air	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel piping and fittings in a warm, moist air environment.
3.3.2.146	Piping and Fittings	Carbon Steel	Wet Gas	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components in a wet gas environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.147	Piping and Fittings	Cast Iron	Chemically treated demineralized water <90°C (194°F)	Loss of material/ General, pitting and crevice corrosion, selective leaching and microbiologically influenced corrosion	Closed-Cycle Cooling Water System (B.1.14), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not consider cast iron in a diesel generator cooling water subsystem.
3.3.2.148	Piping and Fittings	Cast Iron	Lubricating oil (with contaminants and/ or moisture)	Loss of material/ General pitting crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address cast iron components in a lubricating oil environment.
3.3.2.149	Piping and Fittings	Cast Iron	Moist air	Loss of material/ General, pitting, and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not include cast iron components in a moist air environment.
3.3.2.150	Piping and Fittings	Cast Iron	Raw water	Loss of Material/ Selective Leaching	Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address selective leaching of cast iron in fire protection systems.
3.3.2.151	Piping and Fittings	Copper	Chemically treated demineralized water <90°C (194°F) (Glycol based chemical treatment)	Loss of material/ Crevice, galvanic, pitting corrosion and MIC	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider copper in a diesel generator cooling water subsystem.
3.3.2.152	Piping and Fittings	Copper	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. This refrigerant environment is not conducive to promoting aging degradation of copper components. Copper is not subject to any viable aging mechanism in this environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.153	Piping and Fittings	Iron Ductile	Raw water	Loss of material/ General galvanic pitting crevice microbiologically influenced corrosion and biofouling	Fire Water System (B.1.19)	NUREG-1801 does not address ductile iron.
3.3.2.154	Piping and Fittings	Iron Ductile	Soil and groundwater	Loss of material/ Pitting crevice and microbiologically influenced corrosion	Buried Piping and Tanks Inspection (B.1.25)	NUREG-1801 does not address buried ductile iron fire main piping.
3.3.2.155	Piping and Fittings	Iron Malleable	Chemically treated demineralized water <90°C (194°F) (Glycol based chemical treatment)	Loss of Material/ General, Galvanic, Pitting, Crevice Corrosion, Selective Leaching and MIC	Closed-Cycle Cooling Water System (B.1.14), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not consider malleable iron in a diesel generator cooling water subsystem.
3.3.2.156	Piping and Fittings	Iron Malleable	Lubricating oil (with contaminants and/ or moisture)	Loss of material/ General pitting crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address malleable iron components in a lubricating oil environment.
3.3.2.157	Piping and Fittings	Iron Malleable	Raw water	Loss of material/ General galvanic pitting crevice microbiologically influenced corrosion and biofouling	Fire Water System (B.1.19)	NUREG-1801 does not address malleable iron.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.158	Piping and Fittings	Polyvinyl Chloride (PVC)	Raw water	None	None	NUREG-1801 does not address PVC in a raw water environment. PVC is relatively unaffected by water, concentrated alkaline, and non-oxidizing acids, oils, and ozone. No viable aging effects exist for PVC in this environment.
3.3.2.159	Piping and Fittings	Stainless Steel	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider cracking in closed cooling water environments.
3.3.2.160	Piping and Fittings	Stainless Steel	Chemically treated demineralized water <90°C (194°F) (Glycol based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider stainless steel in a diesel generator cooling water subsystem.
3.3.2.161	Piping and Fittings	Stainless Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen) is not conducive to promoting aging degradation of stainless steel components.
3.3.2.162	Piping and Fittings	Stainless Steel	Lubricating oil (with contaminants and/ or moisture)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in a lubricating oil environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.163	Piping and Fittings	Stainless Steel	Moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in diesel generator subsystems.
3.3.2.164	Piping and Fittings	Stainless Steel	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address stainless steel components in a saturated air environment.
3.3.2.165	Piping and Fittings	Stainless Steel	Sodium pentaborate solution at 21 - 32 °C (70 - 90°F) (˜24,500 ppm B)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not consider loss of material for components in a sodium pentaborate environment.
3.3.2.166	Piping and Fittings	Stainless Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel piping and fittings in a warm, moist air environment.
3.3.2.167	Piping and Fittings	Stainless Steel	Wet Gas	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components in a wet gas environment.
3.3.2.168	Piping and Fittings	Steel Saran Lined	Raw, untreated salt water or fresh water	Loss of Material/ General, pitting, crevice, galvanic, erosion, & MIC Flow Blockage/ Biofouling, silting & corrosion product buildup	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address saran-lined steel material for open-cycle cooling water piping. Material is being conservatively treated as carbon steel.
3.3.2.169	Piping and Fittings	Titanium	Raw, untreated salt water or fresh water	Flow Blockage/ Biofouling	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address titanium as a material for open-cycle cooling water piping.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.170	Pump Casings	Cast Iron	Raw water (submerged)	Loss of material/ General, pitting, crevice, microbiologically influenced corrosion and macro organisms	Open-Cycle Cooling Water System (B.1.13) or Fire Water System (B.1.19)	NUREG-1801 does not address auxiliary system cast iron pump casings in a submerged (raw water) environment.
3.3.2.171	Pump Casings	Cast Iron	Raw water (submerged)	Loss of Material/ Selective Leaching	Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address auxiliary system cast iron pump casings in a submerged (raw water) environment.
3.3.2.172	Pump Casings	Cast Iron	Raw, untreated fresh water	Loss of material/ General, pitting and crevice corrosion, selective leaching and microbiologically influenced corrosion	Open-Cycle Cooling Water System (B.1.13), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address cast iron material for ultimate heat sink pumps.
3.3.2.173	Pump Casings	Iron Cast (Lined)	Air, moisture, humidity, and leaking fluid	Loss of material/ Pitting and crevice corrosion	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address cooling water lined cast iron pump casings in a high moisture (pump vault) indoor environment. The external surfaces are unlined cast iron.
3.3.2.174	Pumps	Carbon Steel	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider cracking in closed cooling water environments.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.175	Pumps	Cast Iron	Chemically treated demineralized water <90°C (194°F)	Loss of Material/ General, Galvanic, Pitting, Crevice Corrosion, Selective Leaching and MIC	Closed-Cycle Cooling Water System (B.1.14), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not consider cast iron in a diesel generator cooling water subsystem.
3.3.2.176	Pumps	Cast Iron	Diesel fuel oil	Loss of Material/ General pitting crevice and microbiologically influenced corrosion	Fuel Oil Chemistry (B.1.21)	NUREG-1801 does not address cast iron components in a fuel oil environment.
3.3.2.177	Pumps	Cast Iron	Lubricating oil (with contaminants and/ or moisture)	Loss of material/ General pitting crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address cast iron components in a lubricating oil environment.
3.3.2.178	Pumps	Cast Iron	Raw water	Loss of Material/ Selective Leaching	Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address selective leaching of cast iron in fire protection.
3.3.2.179	Pumps	Cast Iron	Raw, untreated salt water or fresh water	Loss of material/ General, pitting, crevice, microbiologically influenced corrosion and macro organisms	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address cast iron material for open-cycle cooling water pumps.
3.3.2.180	Pumps	Cast Iron	Raw, untreated salt water or fresh water	Loss of Material/ Selective Leaching	Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address cast iron material for open-cycle cooling water pumps.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.181	Pumps	Cast Iron	Saturated Steam/ Condensate	Loss of material/ General corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address cast iron components in a plant heating steam environment.
3.3.2.182	Pumps	Cast Iron	Treated water	Loss of Material/ Selective Leaching	Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address non-safety related components in a treated water environment.
3.3.2.183	Pumps	Iron Cast (Lined)	Raw, untreated salt water or fresh water	Loss of Material/ Selective Leaching	Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address lined cast iron material for open-cycle cooling water pumps. Loss of material aging is evaluated as for unlined cast iron for conservatism.
3.3.2.184	Pumps	Iron Cast (Lined)	Raw, untreated salt water or fresh water	Loss of material/ General, pitting, crevice, microbiologically influenced corrosion and macro organisms	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address lined cast iron material for open-cycle cooling water pumps. Loss of material aging is evaluated as for unlined cast iron for conservatism.
3.3.2.185	Pumps	Stainless Steel	Sodium pentaborate solution at 21 - 32 °C (70 - 90°F) (˜24,500 ppm B)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not consider loss of material for components in a sodium pentaborate environment.
3.3.2.186	Restricting Orifices	Brass or Bronze	25-288°C (77-550°F) demineralized water	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address non-safety related components in a demineralized water environment.
3.3.2.187	Restricting Orifices	Carbon Steel	Lubricating oil (with contaminants and/ or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.188	Restricting Orifices	Carbon Steel	Treated water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non-safety related components in a treated water environment.
3.3.2.189	Restricting Orifices	Copper	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. This nitrogen environment is not conducive to promoting aging degradation of copper components. Copper is not subject to any viable aging mechanism in this environment.
3.3.2.190	Restricting Orifices	Stainless Steel	Chemically treated demineralized water <90°C (194°F) (Glycol based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider stainless steel in a diesel generator cooling water subsystem.
3.3.2.191	Restricting Orifices	Stainless Steel	Diesel fuel oil	Loss of material/ Pitting and crevice corrosion	Fuel Oil Chemistry (B.1.21)	NUREG-1801 does not address stainless steel components in a fuel oil environment.
3.3.2.192	Restricting Orifices	Stainless Steel	Moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in a moist air environment.
3.3.2.193	Restricting Orifices	Stainless Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel piping components in a warm, moist air environment.
3.3.2.194	Rupture Discs	Stainless Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen) is not conducive to promoting aging degradation of stainless steel components.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.195	Sample Pumps	Stainless Steel	Moist containment atmosphere (air/ nitrogen), steam, or demineralized water	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in moist containment atmosphere (air/ nitrogen)environment.
3.3.2.196	Sight Glasses	Carbon Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (refrigerant) is not conducive to promoting aging degradation of carbon steel components.
3.3.2.197	Sight Glasses	Carbon Steel	Saturated Steam/ Condensate	Loss of material/ General corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel components in a building heating steam environment.
3.3.2.198	Sight Glasses	Carbon Steel	Warm, moist air	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components in a warm, moist air environment.
3.3.2.199	Sight Glasses	Carbon Steel	Wet Gas	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components in a wet gas environment.
3.3.2.200	Sight Glasses	Glass	Chemically treated demineralized water <90°C (194°F)	None	None	NUREG-1801 does not address glass components in a chemically treated water environment. Glass is not subject to any viable aging effect in this environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.201	Sight Glasses	Glass	Fuel oil	None	None	NUREG-1801 does not address glass components in a fuel oil environment. A fuel oil environment is not conducive to promoting aging degradation of glass components. Glass in this environment does not have any applicable aging effect.
3.3.2.202	Sight Glasses	Glass	Lubricating oil (with contaminants and/ or moisture)	None	None	NUREG-1801 does not address glass components in a lubricating oil environment. A lubricating oil environment is not conducive to promoting aging degradation of glass components. Glass in this environment does not have any applicable aging effect.
3.3.2.203	Sight Glasses	Glass	Sodium pentaborate solution at 21 - 32 °C (70 - 90°F) (˜24,500 ppm B)	None	None	NUREG-1801 does not address glass components in a sodium pentaborate environment. Glass is not subject to any viable aging effect in this environment.
3.3.2.204	Sight Glasses	Glass	Wet Gas	None	None	NUREG-1801 does not address glass components in a wet gas environment. A wet gas environment is not conducive to promoting aging degradation of glass components. Glass in this environment does not have any applicable aging effect.
3.3.2.205	Sprinklers	Brass or Bronze	Warm, moist air	Loss of material/ Pitting and crevice corrosion	Fire Water System (B.1.19)	NUREG-1801 does not address brass and bronze in a warm, moist air as an environment for the fire protection system.
3.3.2.206	Sprinklers	Carbon Steel	Warm, moist air	Loss of material/ General (carbon steel only) pitting and crevice corrosion	Fire Water System (B.1.19)	NUREG-1801 does not address warm, moist air as an environment for carbon steel.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.207	Steel Piles	Carbon Steel	Soil and groundwater	None	None	NUREG-1801 does not address carbon steel piles in a soil and ground water environment. The intended function of steel piles driven in undisturbed soils are not affected by corrosion.
3.3.2.208	Strainer Bodies	Cast Iron	Raw, untreated salt water or fresh water	Loss of Material/ General, pitting, crevice, galvanic, erosion, & MIC Flow Blockage/ Biofouling, silting & corrosion product buildup	Open-Cycle Cooling Water System (B.1.13), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address cast iron material for open-cycle cooling water strainers.
3.3.2.209	Strainers	Carbon Steel	Treated water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non-safety related components in a treated water environment.
3.3.2.210	Tanks	Aluminum	Dry Gas	None	None	NUREG-1801 does not address aluminum components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen) is not conducive to promoting aging degradation of aluminum components.
3.3.2.211	Tanks	Carbon Steel	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider cracking in closed cooling water environments.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.212	Tanks	Carbon Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen and CO2) is not conducive to promoting aging degradation of carbon steel components.
3.3.2.213	Tanks	Carbon Steel	Outdoor ambient conditions	Loss of material/ General pitting crevice corrosion	Aboveground Carbon Steel Tanks (B.1.20)	NUREG-1801 does not address outdoor carbon steel storage tanks other than for diesel fuel oil service.
3.3.2.214	Tanks	Carbon Steel	Saturated Steam/ Condensate	Loss of material/ General corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel components in a plant heating steam environment.
3.3.2.215	Tanks	Carbon Steel	Soil and groundwater	Loss of Material/ General pitting crevice and microbiologically influenced corrosion	Buried Piping and Tanks Inspection (B.1.25)	NUREG-1801 does not address buried carbon steel storage tanks.
3.3.2.216	Tanks	Carbon Steel	Wet Gas	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components in a wet gas environment.
3.3.2.217	Tanks	Fiberglass	Fuel oil, water (as contaminant)	Buildup of deposit/ Biofouling	Fuel Oil Chemistry (B.1.21)	NUREG-1801 does not address fiberglass components in a fuel oil environment.
3.3.2.218	Tanks	Fiberglass	Soil and groundwater	None	None	NUREG-1801 does not address fiberglass tanks in a buried environment. Ultraviolet radiation will age fiberglass, however, this aging mechanism is not applicable to underground storage tanks. Therefore, no viable aging effects exist in this environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.219	Tanks	Stainless Steel	Sodium pentaborate solution at 21 - 32 °C (70 - 90°F) (˜24,500 ppm B)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not consider loss of material for components in a sodium pentaborate environment.
3.3.2.220	Thermowells	Brass or Bronze	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Cracking/ Stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider brass or bronze in closed cooling water environments.
3.3.2.221	Thermowells	Brass or Bronze	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Loss of Material/ General, Galvanic, Pitting, Crevice Corrosion, Selective Leaching and MIC	Closed-Cycle Cooling Water System (B.1.14), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not consider brass or bronze in closed cooling water environments.
3.3.2.222	Thermowells	Brass or Bronze	Chemically treated demineralized water <90°C (194°F) (Glycol based chemical treatment)	Loss of Material/ Galvanic, Pitting, Crevice Corrosion, Selective Leaching and MIC	Closed-Cycle Cooling Water System (B.1.14), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not consider brass or bronze in Diesel Generator cooling water subsystem.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.223	Thermowells	Carbon Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen) is not conducive to promoting aging degradation of carbon steel components.
3.3.2.224	Thermowells	Carbon Steel	Lubricating oil (with contaminants and/ or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.3.2.225	Thermowells	Stainless Steel	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider cracking in closed cooling water environments.
3.3.2.226	Thermowells	Stainless Steel	Chemically treated demineralized water <90°C (194°F) (Glycol based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider stainless steel in a diesel generator cooling water subsystem.
3.3.2.227	Thermowells	Stainless Steel	Sodium pentaborate solution at 21 - 32 °C (70 - 90°F) (˜24,500 ppm B)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not consider loss of material for components in a sodium pentaborate environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.228	Traps	Carbon Steel	Warm, moist air	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel piping components in a warm, moist air environment.
3.3.2.229	Traps	Cast Iron	Saturated Steam/ Condensate	Loss of material/ General corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address cast iron components in a plant heating steam environment.
3.3.2.230	Filters/ Strainers	Stainless Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel filters/strainers in a warm, moist air environment.
3.3.2.231	Tubing	Aluminum	Dry Gas	None	None	NUREG-1801 does not address aluminum components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen) is not conducive to promoting aging degradation of aluminum components.
3.3.2.232	Tubing	Brass or Bronze	Moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address brass or bronze components in a moist air environment.
3.3.2.233	Tubing	Carbon Steel	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider cracking in closed cooling water environments.
3.3.2.234	Tubing	Carbon Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (CO2) is not conducive to promoting aging degradation of carbon steel components.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.235	Tubing	Carbon Steel	Lubricating oil (with contaminants and/ or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.3.2.236	Tubing	Copper	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Cracking/ Stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider cracking in closed cooling water environments.
3.3.2.237	Tubing	Copper	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Loss of material/ Crevice, galvanic, pitting corrosion and MIC	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider copper in a diesel generator cooling water subsystem.
3.3.2.238	Tubing	Copper	Chemically treated demineralized water <90°C (194°F) (Glycol based chemical treatment)	Loss of material/ Crevice, galvanic, pitting corrosion and MIC	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider copper in a diesel generator cooling water subsystem.
3.3.2.239	Tubing	Copper	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen) is not conducive to promoting aging degradation of copper components.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.240	Tubing	Copper	Lubricating oil (with contaminants and/ or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address copper components in a lubricating oil environment.
3.3.2.241	Tubing	Copper	Moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address copper tubing as a material of construction.
3.3.2.242	Tubing	Copper	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address copper tubing as a material of construction.
3.3.2.243	Tubing	Copper	Saturated Steam/ Condensate	Loss of material/ General corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address copper components in a plant heating steam environment.
3.3.2.244	Tubing	Copper	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address copper components in a warm, moist air environment.
3.3.2.245	Tubing	Stainless Steel	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider cracking in closed cooling water environments.
3.3.2.246	Tubing	Stainless Steel	Chemically treated demineralized water <90°C (194°F) (Glycol based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider stainless steel in a diesel generator cooling water subsystem.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.247	Tubing	Stainless Steel	Diesel fuel oil	Loss of material/ Pitting and crevice corrosion	Fuel Oil Chemistry (B.1.21)	NUREG-1801 does not address stainless steel components in a fuel oil environment.
3.3.2.248	Tubing	Stainless Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen) is not conducive to promoting aging degradation of stainless steel components.
3.3.2.249	Tubing	Stainless Steel	Lubricating oil (with contaminants and/ or moisture)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in a lubricating oil environment.
3.3.2.250	Tubing	Stainless Steel	Moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel piping components in a moist air environment.
3.3.2.251	Tubing	Stainless Steel	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address stainless steel components in a saturated air environment.
3.3.2.252	Tubing	Stainless Steel	Saturated Steam/ Condensate	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in a plant heating steam environment.
3.3.2.253	Tubing	Stainless Steel	Sodium pentaborate solution at 21 - 32 °C (70 - 90°F) (˜24,500 ppm B)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not consider loss of material for components in a sodium pentaborate environment.
3.3.2.254	Tubing	Stainless Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel piping components in a warm moist air environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.255	Tubing	Stainless Steel	Wet Gas	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components in a wet gas environment.
3.3.2.256	Turbochargers	Cast Iron	Hot diesel engine exhaust gases containing moisture and particulates	Loss of material/ General, pitting, and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address cast iron components in an engine exhaust gas environment.
3.3.2.257	Valves	Brass or Bronze	25-288°C (77-550°F) demineralized water	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address non-safety related components in a demineralized water environment.
3.3.2.258	Valves	Brass or Bronze	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Cracking/ Stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider brass or bronze in closed cooling water environments.
3.3.2.259	Valves	Brass or Bronze	Diesel fuel oil	Loss of material/ General pitting crevice corrosion	Fuel Oil Chemistry (B.1.21)	NUREG-1801 does not address brass or bronze components in a fuel oil environment.
3.3.2.260	Valves	Brass or Bronze	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen) is not conducive to promoting aging degradation of brass or bronze components.
3.3.2.261	Valves	Brass or Bronze	Moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address brass or bronze components in a moist air environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.262	Valves	Brass or Bronze	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address brass or bronze components in a saturated air environment.
3.3.2.263	Valves	Brass or Bronze	Saturated Steam/ Condensate	Loss of material/ General corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address brass or bronze components in a plant heating steam environment.
3.3.2.264	Valves	Brass or Bronze	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address brass or bronze components in a warm, moist air environment.
3.3.2.265	Valves	Brass or Bronze	Warm, moist air	Loss of material/ Pitting and crevice corrosion	Fire Water System (B.1.19)	NUREG-1801 does not address warm, moist air as an environment for brass and bronze.
3.3.2.266	Valves	Carbon Steel	25-288°C (77-550°F) demineralized water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non-safety related components in a demineralized water environment.
3.3.2.267	Valves	Carbon Steel	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider cracking in closed cooling water environments.
3.3.2.268	Valves	Carbon Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen and CO2) is not conducive to promoting aging degradation of carbon steel components.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.269	Valves	Carbon Steel	Lubricating oil (with contaminants and/ or moisture)	Loss of material/ General galvanic pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address BWR components in a lubricating oil environment.
3.3.2.270	Valves	Carbon Steel	Oxygenated water 93°C - 288°C (200°F-550°F)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel components in an oxygenated water environment.
3.3.2.271	Valves	Carbon Steel	Saturated Steam/ Condensate	Loss of material/ General corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel components in a plant heating steam environment.
3.3.2.272	Valves	Carbon Steel	Treated water	Loss of material/ General, pitting, and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address non-safety related components in a treated water environment.
3.3.2.273	Valves	Carbon Steel	Warm, moist air	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel valves in a warm, moist air environment.
3.3.2.274	Valves	Carbon Steel	Wet Gas	Loss of material/ General (carbon steel only) pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components in a wet gas environment.
3.3.2.275	Valves	Cast Iron	Chemically treated demineralized water <90°C (194°F)	Loss of Material/ General, Galvanic, Pitting, Crevice Corrosion, Selective Leaching and MIC	Closed-Cycle Cooling Water System (B.1.14), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not consider cast iron in a diesel generator cooling water subsystem.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.276	Valves	Cast Iron	Raw water	Loss of Material/ General, pitting, crevice, galvanic, erosion, & MIC Flow Blockage/ Biofouling, silting & corrosion product buildup	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address cast iron valves in a raw water environment.
3.3.2.277	Valves	Cast Iron	Raw water	Loss of Material/ Selective Leaching	Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address cast iron valves in a raw water environment.
3.3.2.278	Valves	Cast Iron	Raw, untreated fresh water	Loss of material/ General, pitting and crevice corrosion, selective leaching and microbiologically influenced corrosion	Open-Cycle Cooling Water System (B.1.13), Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address cast iron material for ultimate heat sink valves.
3.3.2.279	Valves	Cast Iron	Raw, untreated salt water or fresh water	Loss of Material/ Selective Leaching	Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address cast iron valves in a raw water environment.
3.3.2.280	Valves	Cast Iron	Raw, untreated salt water or fresh water	Loss of Material/ General, pitting, crevice, galvanic, erosion, & MIC Flow Blockage/ Biofouling, silting & corrosion product buildup	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address cast iron valves in a raw water environment.
3.3.2.281	Valves	Cast Iron	Raw, untreated salt water or fresh water	Loss of Material/ General, pitting, crevice, galvanic, erosion, and MIC	One-Time Inspection (B.1.23)	NUREG-1801 does not address NSR cast iron valves in a raw water environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.282	Valves	Cast Iron	Saturated Steam/ Condensate	Loss of material/ General corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address cast iron components in a plant heating steam environment.
3.3.2.283	Valves	Cast Iron	Warm, moist air	Loss of material/ General, pitting, and crevice corrosion	Fire Water System (B.1.19)	NUREG-1801 does not address warm, moist air as an environment for cast iron.
3.3.2.284	Valves	Iron Ductile	Dry Gas	None	None	NUREG-1801 does not address ductile iron components in a dry gas (moisture free) environment. A moisture free gaseous environment (refrigerant) is not conducive to promoting aging degradation of ductile iron components.
3.3.2.285	Valves	Stainless Steel	25-288°C (77-550°F) demineralized water	Crack initiation and growth/ Stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address non-safety related components in a demineralized water environment.
3.3.2.286	Valves	Stainless Steel	Chemically treated demineralized water < 90°C (194°F) (TTA-Nitrite based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider cracking in closed cooling water environments.
3.3.2.287	Valves	Stainless Steel	Chemically treated demineralized water <90°C (194°F) (Glycol based chemical treatment)	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not consider stainless steel in a diesel generator cooling water subsystem.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.288	Valves	Stainless Steel	Diesel fuel oil	Loss of material/ Pitting and crevice corrosion	Fuel Oil Chemistry (B.1.21)	NUREG-1801 does not address stainless steel components in a fuel oil environment.
3.3.2.289	Valves	Stainless Steel	Dry Gas	None	None	NUREG-1801 does not address components in a dry gas (moisture free) environment. A moisture free gaseous environment (nitrogen) is not conducive to promoting aging degradation of stainless steel components.
3.3.2.290	Valves	Stainless Steel	Lubricating oil (with contaminants and/ or moisture)	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in a lubricating oil environment.
3.3.2.291	Valves	Stainless Steel	Moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel valves in a moist air environment.
3.3.2.292	Valves	Stainless Steel	Oxygenated water 93°C - 288°C (200°F-550°F)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address crevice and pitting corrosion of stainless steel in treated water environment.
3.3.2.293	Valves	Stainless Steel	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address stainless steel valves in a saturated air environment.
3.3.2.294	Valves	Stainless Steel	Sodium pentaborate solution at 21 - 32 °C (70 - 90°F) (˜24,500 ppm B)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not consider loss of material for components in a sodium pentaborate environment.
3.3.2.295	Valves	Stainless Steel	Warm, moist air	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel valves in a warm, moist air environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.296	Valves	Stainless Steel	Wet Gas	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components in a wet gas environment.
3.3.2.297	Valves	Steel Saran Lined	Raw, untreated salt water or fresh water	Loss of Material/ General, pitting, crevice, galvanic, erosion, & MIC Flow Blockage/ Biofouling, silting & corrosion product buildup	Open-Cycle Cooling Water System (B.1.13)	NUREG-1801 does not address saran-lined steel material for open-cycle cooling water valves.
3.3.2.298	Valves	Titanium	Raw, untreated salt water or fresh water	None	None	NUREG-1801 does not address titanium components. Raw, untreated fresh water is not an aggressive environment conducive to promoting aging degradation of titanium valves as it does not a contain high chlorides concentration at high temperature (>165°F).
3.3.2.299	Pumps	Stainless Steel	Wet Gas	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address components in a wet gas environment.
3.3.2.300	Component External Surfaces (piping and fittings, valves, sprinklers, filters/strainers, pumps, strainer bodies, ejectors, turbochargers)	Cast Iron	Air, moisture, and humidity < 100°C (212°F)	Loss of material/ Pitting and crevice corrosion	Bolting Integrity (B.1.12)	NUREG-1801 does not address cast iron in the plant indoor environment.

Table 3.3-2 Aging management review results for the auxiliary systems that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.3.2.301	Valves (Fire Protection)	Cast Iron	Raw water	Loss of Material/ Selective Leaching	Selective Leaching of Materials (B.1.24)	NUREG-1801 does not address cast iron valves in a raw water environment.
3.3.2.302	Piping and Fittings	Aluminum	<90° C (<194°F) treated water	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address aluminum in a reactor grade water environment.
3.3.2.303	Heat Exchangers	Tubes: Stainless Steel; Channel Head: Cast Steel; Shell: Cast Steel	Tube Side: Reactor Coolant water; Shell side: Closed-cycle cooling water	Cracking/ Mech Fatigue, SCC	Water Chemistry (B.1.2), Closed-Cycle Cooling Water System (B.1.14)	NUREG-1801 does not address cast steel subcomponents or cracking in shutdown cooling system heat exchangers.

3.4 AGING MANAGEMENT OF STEAM AND POWER CONVERSION SYSTEM

The following systems are evaluated as part of the steam and power conversion system:

- Main steam system
- Feedwater system
- Condensate and condensate storage system
- Main condenser
- Main turbine and auxiliaries
- Turbine oil system (in-scope for Quad Cities only)
- Main generator and auxiliaries (in-scope for Quad Cities only)

Components Evaluated Consistent with NUREG-1801

The components or component groups requiring aging management review in each of the steam and power conversion systems listed above are presented in Chapter 2. Aging management reviews were performed to assure that the component groups, materials, environments and aging effects referenced in NUREG –1801 are applicable to Dresden and Quad Cities and that the aging management program described in NUREG-1801 is appropriate.

When the components or component group and associated evaluations corresponded with an individual line item in NUREG-1801, Volume 2, the component aging management results were included in the appropriate line items in Table 3.4-1 “Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the steam and power conversion system” for the steam and power conversion system. Each line in Table 3.4-1 matches a line in Chapter 3 of NUREG-1800, that is applicable to Boiling Water Reactors (BWR) or to both Boiling Water Reactors and Pressurized Water Reactors (BWR/PWR).

Not all component types in the Dresden and Quad Cities steam and power conversion system are listed in NUREG-1801, Volume 2. However, the aging management reviews presented in NUREG-1801, Volume 2 were applied to additional component types if the following criteria were satisfied:

- Constructed of the same material as components in the NUREG-1801 line item,
- Assigned the same Component Intended Function as components in the NUREG-1801 line item,
- Located in the same environment as components in the NUREG-1801 line item, and
- Have exhibited the same aging effects identified in the NUREG-1801 line item.

Component types meeting these criteria have been included in the presentation of aging management review results in Table 3.4-1 "Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the steam and power conversion system." The third column of the table shows the component types included in each evaluation line. "NUREG-1801 Components" are those that correspond exactly with component types in NUREG-1801, Volume 2. "Evaluated with NUREG-1801 Components" shows the component types that meet the four (4) criteria above, and therefore share the same evaluation characteristics. Dresden and Quad Cities do not have all of the component types identified in NUREG-1801, Volume 2. The component type entries in the third column of Table 3.4-1 "Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the steam and power conversion system" under the "NUREG-1801 Components" heading are those components or component groups that were identified and evaluated at Dresden and Quad Cities.

Other Components Evaluated

Table 3.4-2 "Aging management review results for the steam and power conversion system that are not addressed in NUREG-1801" presents the aging management review results for the remainder of the steam and power conversion system components. These entries result from aging management review results where the component type, material, environment or aging effect/mechanism differs from NUREG-1801, Volume 2 line item entries. Each line in Table 3.4-2 includes a line reference number, component group, material, environment, aging effect/mechanism, aging management program and discussion.

3.4.1 Aging management programs evaluated in NUREG-1801 that are relied upon for license renewal for the steam and power conversion system

Table 3.4-1, "Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the steam and power conversion system" shows the component groups and aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the steam and power conversion systems.

Table 3.4-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the steam and power conversion system

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.4.1.1	Piping and fittings in main feedwater line, steam line and AFW piping (PWR only)	NUREG-1801 Components Piping and Fittings	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Further Evaluation of cumulative fatigue damage is provided in Section 3.4.1.1.1 and Section 4.3.3.2.

Table 3.4-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the steam and power conversion system (Continued)

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.4.1.2	Piping and fittings, valve bodies and bonnets, pump casings, tanks, tubes, tubesheets, channel head and shell (except main steam system)	NUREG-1801 Components Piping and Fittings Valves Evaluated with NUREG-1801 Components Thermowells	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Water chemistry (B.1.2) and one-time inspection (B.1.23)	Yes, detection of aging effects is to be further evaluated	Consistent with NUREG-1801, with exception. The exceptions to Water Chemistry are described in Section B.1.2. Further evaluation of Loss of Material due to General, Pitting, and Crevice Corrosion is described in Section 3.4.1.1.2. Condensate pumps identified in NUREG-1801, line VIII.E.3-a and feedwater pumps identified in line VIII.D2.3-b are not in the scope of license renewal. Piping and fittings for the steam turbine and extraction steam systems identified in NUREG-1801, lines VIII.A.1-b and VIII.C1-b are not in the scope of license renewal. The condensate cleanup system components identified in NUREG-1801 line VII.E.6-a are not in the scope of license renewal. The condensate coolers / condensers in the condensate system identified in NUREG-1801, lines VIII.E.4-a and VIII.E.4-d and valves in the extraction steam system identified in line VIII.C.2-b are not in the scope of license renewal. Dresden and Quad Cities do not use carbon steel or stainless steel for the CST tanks as identified in NUREG-1801, lines VIII.E.5-a and VIII.E.5-b. Dresden and Quad Cities CST tanks are aluminum.

Table 3.4-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the steam and power conversion system (Continued)

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.4.1.3	External surface of carbon steel components	NUREG-1801 Components Carbon Steel Components (piping and fittings, valves, restricting orifices, thermowells, filters/strainers, tanks)	Loss of material due to general corrosion	Plant specific	Yes, plant specific	Further evaluation of General Corrosion is described in Section 3.4.1.1.3.
3.4.1.4	Carbon steel piping and valve bodies	NUREG-1801 Components Piping and Fittings Valves Evaluated with NUREG-1801 Components Restricting Orifices Thermowells	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion (B.1.11)	No	Consistent with NUREG-1801, with exception. The exceptions to flow accelerated corrosion are described in Section 3.4.1.2.1. Piping and fittings for the steam turbine and extraction steam system identified in NUREG-1801, VIII.A.1-a and VIII.C.1-a are not in the scope of license renewal. Valves for the extraction steam system and feedwater pumps in the feedwater system discussed in NUREG-1801, lines VIII.C.2-a and VIII.D2.3-a are not in the scope of license renewal.
3.4.1.5	Carbon steel piping and valve bodies in main steam system	NUREG-1801 Components Piping and Fittings Valves Evaluated with NUREG-1801 Components Restricting Orifices	Loss of material due to pitting and crevice corrosion	Water chemistry (B.1.2)	No	Consistent with NUREG-1801, with exception. The exceptions to Water Chemistry are described in Section B.1.2.

Table 3.4-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the steam and power conversion system (Continued)

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.4.1.6	Closure bolting in high-pressure or high-temperature systems	NUREG-1801 Components Closure Bolting	Loss of material due to general corrosion; crack initiation and growth due to cyclic loading and/or SCC.	Bolting integrity (B.1.12)	No	Consistent with NUREG-1801, with exception. The exceptions to Bolting Integrity are described in Section B.1.12.
3.4.1.7	Heat exchangers and coolers/ condensers serviced by open-cycle cooling water	See Discussion Column	Loss of material due to general (carbon steel only), pitting, and crevice corrosion, MIC, and biofouling; buildup of deposit due to biofouling	Open-cycle cooling water system (B.1.13)	No	Components are not in the scope of license renewal.
3.4.1.8	Heat exchangers and coolers/ condensers serviced by open-cycle cooling water	See Discussion Column	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Closed-cycle cooling water system (B.1.14)	No	Components are not in the scope of license renewal.
3.4.1.9	External surface of aboveground condensate storage tank	See Discussion Column	Loss of material due to general (carbon steel only), pitting, and crevice corrosion	Aboveground carbon steel tanks (B.1.20)	No	NUREG-1801 specifies carbon steel for the CST, however, Dresden and Quad Cities have tanks constructed of aluminum.
3.4.1.10	External surface of buried condensate storage tank and AFW piping	See Discussion Column	Loss of material due to general, pitting, and crevice corrosion and MIC	Buried piping and tanks surveillance or Buried piping and tanks inspection (B.1.25)	No, Yes, detection of aging effects and operating experience are to be further evaluated	Carbon steel material and buried environment does not exist at Dresden or Quad Cities

3.4.1.1 Further evaluation of aging management as recommended by NUREG-1801 for the steam and power conversion system

3.4.1.1.1 Cumulative Fatigue Damage (NUREG-1800, Section 3.4.2.2.1)

Cumulative fatigue damage of steam line and main feedwater piping is a TLAA as defined in 10 CFR 54.3. Cumulative fatigue damage of steam line and feedwater piping is required to be evaluated in accordance with 10 CFR 54.21(c)(1). The TLAA evaluation of steam line and feedwater piping outside the RCPB is addressed in Section 4.3.3.2.

3.4.1.1.2 Loss of Material due to General, Pitting, and Crevice Corrosion (NUREG-1800, Section 3.4.2.2.2)

An inspection of selected components exposed to a stagnant flow water environment will be conducted in accordance with aging management program One-Time Inspection (B.1.23.) The inspection of selected components will verify the effectiveness of the chemistry control program to manage loss of material due to general, pitting, and crevice corrosion in low flow or stagnant flow areas by ensuring that significant degradation is not occurring and the component intended function will be maintained during the extended period of operation. Examinations will be conducted on carbon and stainless steel components in an area where typically stagnant flow is present but occasionally there is flow, which will cause replenishment of the oxygen supply. Inspections will be conducted on the HPCI torus suction check valves, the HPCI booster pumps, and the control rod drive (CRD) scram valves. The carbon steel HPCI torus suction check valves are exposed to torus water and will undergo a visual exam followed by an ultrasonic exam if significant corrosion is observed, while the carbon steel HPCI booster pumps and the stainless steel CRD scram valves are exposed to condensate storage tank water and will undergo a visual examination. These components provide representative samples of the aging effects seen in steam and power conversion system.

3.4.1.1.3 General Corrosion (NUREG-1800, Section 3.4.2.2.4)

Aging management of the external surface of the main steam, feedwater, condensate and condensate storage system components in a sheltered environment with moist, warm air will be managed either by the Structures Monitoring Program (B.1.30) or by system engineer walkdowns performed by the Bolting Integrity (B.1.12) aging management activities.

3.4.1.2 Aging management programs or evaluations that are different than those described in NUREG-1801 for the steam and power conversion systems

3.4.1.2.1 Exception to NUREG-1801 for flow accelerated corrosion

Flow accelerated corrosion is an applicable aging mechanism for the main steam lines and the feedwater lines. Carbon steel components in the condensate system are not susceptible to flow accelerated corrosion and do not require aging management. This exception is based on the following:

1. EPRI TR-114882, "Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools," states that flow rates less than 6 ft/sec do not need to be considered for FAC.

2. Dresden and Quad Cities flow rates in the condensate system are less than 6 ft/sec. Additionally, plant experience has not revealed flow accelerated corrosion in these lines.

For the evaluation of the HPCI and RCIC steam lines, an exception is taken to NUREG-1801 for item VIII.B2.1-b. The HPCI and RCIC steam piping and fittings are evaluated with NUREG-1801 line V.D2.1-f (Lines to HPCI and RCIC pump turbine) with the results presented in Table 3.2-1, reference number 3.2.1.12.

3.4.2 Components or aging effects that are not addressed in NUREG-1801 for the steam and power conversion system

Table 3.4-2, “Aging management review results for the steam and power conversion system” that are not addressed in NUREG-1801“ contains aging management review results for the steam and power conversion system for component groups that are not addressed in NUREG-1801.

Table 3.4-2 Aging management review results for the steam and power conversion system that are not addressed in NUREG-1801

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.4.2.1	Accumulators	Stainless Steel	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address stainless steel components in a saturated air environment.
3.4.2.2	Closure Bolting	High Strength Low Alloy Steel	Outdoor ambient conditions	Loss of Material/General corrosion and wear	Bolting Integrity (B.1.12)	NUREG-1801 does not address closure bolting in an outdoor environment.
3.4.2.3	Component External Surfaces (piping and fittings, valves, tubing)	Aluminum	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address aluminum components in a plant indoor environment. Plant indoor environment is not an aggressive wetted environment conducive to promoting aging degradation of aluminum components.
3.4.2.4	Component External Surfaces (piping and fittings, valves, thermowells, tubing)	Aluminum	Outdoor ambient conditions	Loss of material/ Pitting	Structures Monitoring Program (B.1.30)	NUREG-1801 does not address aluminum component external surfaces in an outdoor environment.
3.4.2.5	Component External Surfaces (piping and fittings, valves)	Carbon Steel	Air, moisture, and humidity where surface temp > 100°C (212°F)	None	None	NUREG-1801 does not address high temperature carbon steel external surfaces. A temperature limit of 212°F applies to all materials as moisture must be present for corrosion to occur.
3.4.2.6	Component External Surfaces (piping and fittings, valves, flow elements, thermowells)	Carbon Steel	Containment Nitrogen	None	None	NUREG-1801 does not address carbon steel in a containment nitrogen environment. A containment nitrogen environment is not conducive to promoting aging degradation.

Table 3.4-2 Aging management review results for the steam and power conversion system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.4.2.7	Component External Surfaces (piping and fittings, valves)	Carbon Steel	Outdoor ambient conditions	Loss of material/ General, pitting, and crevice corrosion	Bolting Integrity (B.1.12)	NUREG-1801 does not address carbon steel component external surfaces in an outdoor environment.
3.4.2.8	Component External Surfaces (piping and fittings)	Carbon Steel	Soil and groundwater	Loss of Material/ General pitting crevice and microbiologically influenced corrosion	Buried Piping and Tanks Inspection (B.1.25)	NUREG-1801 does not address buried steam and power conversion system carbon steel components.
3.4.2.9	Component External Surfaces (pump casings)	Cast Iron	Air, moisture, and humidity < 100°C (212°F)	Loss of material/ Pitting and crevice corrosion	Bolting Integrity (B.1.12)	NUREG-1801 does not address cast iron component external surfaces in an indoor environment.
3.4.2.10						Line left intentionally blank.
3.4.2.11	Component External Surfaces (piping and fittings, valves, dampeners, tanks, tubing, accumulators, heat exchangers, housings, pumps)	Stainless Steel	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address stainless steel in a plant indoor environment. Plant indoor environment is not an aggressive wetted environment conducive to promoting aging degradation of stainless steel components.
3.4.2.12	Component External Surfaces (piping and fittings, valves, thermowells)	Stainless Steel	Air, moisture, and humidity where surface temp > 100°C (212°F)	None	None	NUREG-1801 does not address high temperature stainless steel external surfaces. A temperature limit of 212°F applies to all materials as moisture must be present for corrosion to occur.

Table 3.4-2 Aging management review results for the steam and power conversion system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.4.2.13	Component External Surfaces (piping and fittings, valves, dampeners, filters/strainers, rupture discs, tanks, tubing, vacuum breakers)	Stainless Steel	Containment Nitrogen	None	None	NUREG-1801 does not address stainless steel in a containment nitrogen environment. A containment nitrogen environment is not conducive to promoting aging degradation.
3.4.2.14	Component External Surfaces (piping and fittings, valves, tubing)	Stainless Steel	Outdoor ambient conditions	Loss of material/ Pitting and crevice corrosion	Bolting Integrity (B.1.12)	NUREG-1801 does not address steam and power conversion system stainless steel component external surfaces in an outdoor environment.
3.4.2.15	Component External Surfaces (piping and fittings)	Stainless Steel	Soil and groundwater	Loss of material/ Pitting and crevice corrosion	Buried Piping and Tanks Inspection (B.1.25)	NUREG-1801 does not address steam and power conversion system buried stainless steel component external surfaces.
3.4.2.16	Filters/Strainers	Carbon Steel	Generator Hydrogen Seal Oil	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel components in a generator hydrogen seal oil environment.
3.4.2.17	Filters/Strainers	Stainless Steel	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address stainless steel components in a saturated air environment.
3.4.2.18	Flexible Hoses	Elastomers Neoprene and Similar Materials	Containment Nitrogen	None	None	NUREG-1801 does not address flexible hose elastomer materials in a containment nitrogen environment. A containment nitrogen environment is not conducive to promoting aging degradation.
3.4.2.19	Flexible Hoses	Elastomers Neoprene and Similar Materials	Saturated air	Hardening and loss of strength/ Elastomer degradation	One-Time Inspection (B.1.23)	NUREG-1801 does not address elastomers in a saturated air environment.

Table 3.4-2 Aging management review results for the steam and power conversion system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.4.2.20	Heat Exchangers	Stainless Steel	Demineralized Water - Stator Liquid Cooling	Loss of material/ Pitting and crevice corrosion	Main Generator Stator Cooling Water Chemistry (B.2.7)	NUREG-1801 does not address components in the main generator stator cooling environment.
3.4.2.21	Housings	Stainless Steel	Demineralized Water - Stator Liquid Cooling	Loss of material/ Pitting and crevice corrosion	Main Generator Stator Cooling Water Chemistry (B.2.7)	NUREG-1801 does not address NSR components in the main generator stator cooling environment.
3.4.2.22	Insulation	Calcium Silicate	Outdoor ambient conditions	None	None	NUREG-1801 does not address calcium silicate insulation. Plant outdoor environment is not conducive to promoting aging degradation of jacketed calcium silicate insulation.
3.4.2.23	Insulation Jacketing	Aluminum Jacketing	Outdoor ambient conditions	Insulation degradation/Loss of jacket leak-tight integrity	Structures Monitoring Program (B.1.30)	NUREG-1801 does not address aluminum insulation jacketing of outdoor piping.
3.4.2.24	Main Condenser Hotwells, False Floors	Carbon Steel	Steam	None	None	NUREG-1801 does not address main condenser components. There are no main condenser aging effects requiring management. Therefore, no aging management is required to maintain the intended function of containment holdup plate-out.
3.4.2.25	Main Condenser Tubes, Tubesheets	Stainless Steel	Open-cycle cooling water (raw water) side	None	None	NUREG-1801 does not address main condenser components. There are no main condenser aging effects requiring management. Therefore, no aging management is required to maintain the intended function of containment holdup plate-out.

Table 3.4-2 Aging management review results for the steam and power conversion system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.4.2.26	Main Condenser Tubes, Tubesheets	Stainless Steel	Steam	None	None	NUREG-1801 does not address main condenser components. There are no main condenser aging effects requiring management. Therefore, no aging management is required to maintain the intended function of containment holdup plate-out.
3.4.2.27	Main Condenser Waterboxes, Hatches	Carbon Steel	Air, moisture, and humidity < 100°C (212°F)	None	None	NUREG-1801 does not address the main condenser components. There are no main condenser aging effects requiring management. Therefore, no aging management is required to maintain the intended function of containment holdup plate-out.
3.4.2.28	Main Condenser Waterboxes, Hatches	Carbon Steel	Open-cycle cooling water (raw water) side	None	None	NUREG-1801 does not address main condenser components. There are no main condenser aging effects requiring management. Therefore, no aging management is required to maintain the intended function of containment holdup plate-out.
3.4.2.29	Accumulators	Stainless Steel	Turbine EHC Fluid	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in a turbine EHC fluid environment.
3.4.2.30	NSR Vents or Drains, Piping and Valves	Carbon Steel, Stainless Steel, Brass or Bronze	Air, moisture, humidity, and leaking fluid	Loss of material/ Corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address NSR vents or drains in an air, moisture, humidity and leaking fluid environment.
3.4.2.31	Piping and Fittings	Aluminum	<90°C (<194°F) treated water	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address aluminum in a reactor grade water environment.

Table 3.4-2 Aging management review results for the steam and power conversion system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.4.2.32	Piping and Fittings	Carbon Steel	Generator Hydrogen Seal Oil	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel components in a hydrogen seal oil environment.
3.4.2.33	Piping and Fittings	Stainless Steel	Demineralized Water - Stator Liquid Cooling	Loss of material/ Pitting and crevice corrosion	Main Generator Stator Cooling Water Chemistry (B.2.7)	NUREG-1801 does not address NSR components in the main generator stator cooling environment.
3.4.2.34	Piping and Fittings	Stainless Steel	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address stainless steel piping and fittings in a saturated air environment.
3.4.2.35	Piping and Fittings	Stainless Steel	Treated water (BWRs: reactor coolant; PWRs: secondary side water)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address crevice and pitting corrosion of stainless steel in treated water environment.
3.4.2.36	Piping and Fittings	Stainless Steel	Turbine EHC Fluid	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in a turbine EHC fluid environment.
3.4.2.37	Pumps	Cast Iron	Generator Hydrogen Seal Oil	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address cast iron components in a generator hydrogen seal oil environment.
3.4.2.38	Pumps	Stainless Steel	Demineralized Water - Stator Liquid Cooling	Loss of material/ Pitting and crevice corrosion	Main Generator Stator Cooling Water Chemistry (B.2.7)	NUREG-1801 does not address NSR components in the main generator stator cooling environment.

Table 3.4-2 Aging management review results for the steam and power conversion system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.4.2.39	Tanks	Aluminum	<90°C (<194°F) treated water	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address aluminum in a reactor grade water environment.
3.4.2.40	Tanks	Aluminum	Outdoor ambient conditions	Loss of material/ Pitting	Structures Monitoring Program (B.1.30)	NUREG-1801 does not address outdoor aluminum storage tanks.
3.4.2.41	Tanks	Aluminum	Outdoor ambient conditions	Cracking/ Stress corrosion cracking	Structures Monitoring Program (B.1.30)	NUREG-1801 does not address outdoor aluminum storage tanks.
3.4.2.42	Tanks	Aluminum	Soil and groundwater	Loss of material/ Pitting	Buried Piping and Tanks Inspection (B.1.25)	NUREG-1801 does not address outdoor aluminum tanks resting on the ground (tank bottom).
3.4.2.43	Tanks	Carbon Steel	Generator Hydrogen Seal Oil	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel components in a generator hydrogen seal oil environment.
3.4.2.44	Tanks	Stainless Steel	Demineralized Water - Stator Liquid Cooling	Loss of material/ Pitting and crevice corrosion	Main Generator Stator Cooling Water Chemistry (B.2.7)	NUREG-1801 does not address NSR components in the main generator stator cooling environment.
3.4.2.45	Thermowells	Aluminum	<90°C (<194°F) treated water	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address aluminum in a reactor grade water environment.
3.4.2.46	Tubing	Aluminum	<90°C (<194°F) treated water	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address aluminum in a reactor grade water environment.

Table 3.4-2 Aging management review results for the steam and power conversion system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.4.2.47	Tubing	Stainless Steel	Treated water (BWRs: reactor coolant; PWRs: secondary side water)	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address crevice and pitting corrosion of stainless steel in treated water environment.
3.4.2.48	Tubing	Stainless Steel	Turbine EHC Fluid	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in a turbine EHC fluid environment.
3.4.2.49	Valves	Aluminum	<90°C (<194°F) treated water	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2)	NUREG-1801 does not address aluminum in a reactor grade water environment.
3.4.2.50	Valves	Carbon Steel	Generator Hydrogen Seal Oil	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address carbon steel components in a generator hydrogen seal oil environment.
3.4.2.51	Valves	Stainless Steel	288°C (550°F) steam	Crack initiation and growth/ Stress corrosion cracking intergranular stress corrosion cracking	Water Chemistry (B.1.2)	NUREG-1801 does not address crevice and pitting corrosion of stainless steel in treated water environment.
3.4.2.52	Valves	Stainless Steel	Demineralized Water - Stator Liquid Cooling	Loss of material/ Pitting and crevice corrosion	Main Generator Stator Cooling Water Chemistry (B.2.7)	NUREG-1801 does not address NSR components in the main generator stator cooling environment.

Table 3.4-2 Aging management review results for the steam and power conversion system that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.4.2.53	Valves	Stainless Steel	Saturated air	Loss of material/ Pitting and crevice corrosion	Compressed Air Monitoring (B.1.16)	NUREG-1801 does not address stainless steel components in a saturated air environment.
3.4.2.54	Valves	Stainless Steel	Treated water	Loss of material/ Pitting and crevice corrosion	Water Chemistry (B.1.2) and One-Time Inspection (B.1.23)	NUREG-1801 does not address crevice and pitting corrosion of stainless steel in treated water environment.
3.4.2.55	Valves	Stainless Steel	Turbine EHC Fluid	Loss of material/ Pitting and crevice corrosion	One-Time Inspection (B.1.23)	NUREG-1801 does not address stainless steel components in a turbine EHC fluid environment.

3.5 AGING MANAGEMENT OF CONTAINMENTS, STRUCTURES AND COMPONENT SUPPORTS

The following tables provide the results of aging management reviews for containments, structures, and component supports component groups within the scope of license renewal for the structures and commodity groups listed below:

- Primary containment
- Reactor building
- Main control room and auxiliary electric equipment room
- Turbine building
- Diesel generator buildings
- Station blackout building and yard structures
- Isolation condenser pump house (Dresden only)
- Makeup demineralizer building (Dresden only)
- Radwaste floor drain surge tank
- Miscellaneous foundations
- Crib house
- Unit 1 crib house (Dresden only)
- Station chimney
- Cranes and hoists
- Component supports commodity group
- Insulation commodity group

Aging management programs and activities are discussed in Appendix B.

The aging management reviews for this section have incorporated the proposed NRC guidance provided in Enclosure 2 to the letter from Christopher I. Grimes, Chief, Licensing and Standardization Branch, Office of Nuclear Reactor Regulation, to Mr. Alan Nelson, Nuclear Energy Institute, “Proposed Staff Guidance on the Position of the GALL Report Presenting One Acceptable Way to Manage Aging Effects for License Renewal,” dated November 23, 2001. Table 3.5-1 incorporates revisions provided in Enclosure 2 to this letter.

<u>Components Evaluated Consistent with NUREG-1801</u>

The components or component groups associated with containments, structures and component supports requiring aging management review are presented in Chapter 2. Aging management reviews were performed to assure that the component groups, materials, environments and aging effects referenced in NUREG-1801 are applicable to Dresden and Quad Cities and that the aging management program described in NUREG-1801 is appropriate. The characteristics of the components or component group, the material of construction, and the environment were the determining factors.

When the components or component groups and associated evaluations corresponded with an individual line item in NUREG-1801, Volume 2, the component aging management results were included in the appropriate line items in Table 3.5-1 “Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for containments, structures and component supports." Each line in Table 3.5-1 matches a line in Chapter 3 of NUREG-1800, that is applicable to Boiling Water Reactors (BWR) or to both Boiling Water Reactors and Pressurized Water Reactors (BWR/ PWR).

Not all component types in the Dresden and Quad Cities containments, structures and component supports are listed in NUREG-1801, Volume 2. However, the aging management reviews presented in NUREG-1801, Volume 2 were applied to additional component types if the following criteria were satisfied:

- Constructed of the same material as components in the NUREG-1801 line item,
- Assigned the same Component Intended Function as components in the NUREG-1801 line item,
- Located in the same environment as components in the NUREG-1801 line item, and
- Have exhibited the same aging effects identified in the NUREG-1801 line item.

Component types meeting these criteria have been included in the presentation of aging management review results in Table 3.5-1 "Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for containments, structures and component supports." The third column of the table shows the component types included in each evaluation line. "NUREG-1801 Components" are those that correspond exactly with component types in NUREG-1801, Volume 2. "Evaluated with NUREG-1801 Components" shows the component types that meet the four (4) criteria above, and therefore share the same evaluation characteristics. Dresden and Quad Cities do not have all of the component types identified in NUREG-1801, Volume 2. The component type entries in the third column of Table 3.5-1 under the "NUREG-1801 Components" heading are those components or component groups that were identified and evaluated at Dresden and Quad Cities.

Other Components Evaluated

Table 3.5-2 "Aging management review results for containments, structures, and component supports that are not addressed in NUREG-1801" presents the aging management review results for the remainder of the containments, structures and component supports components types. These entries result from aging management review results where the component type, material, environment or aging effect/ mechanism differs from NUREG-1801, Volume 2 line item entries. Each line in Table 3.5-2 includes a line reference number, component group, material, environment, aging effect/mechanism, aging management program and discussion.

3.5.1 Aging management programs evaluated in NUREG-1801 that are relied upon for license renewal for containments, structures and component supports.

Table 3.5-1, “Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for containments, structures, and component supports“ shows the component groups and aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the containments, structures and component supports.

Table 3.5-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for containments, structures and component supports

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1.1	Penetration sleeves, penetration bellows, and dissimilar metal welds	NUREG-1801 Components Penetration Sleeves Penetration Bellows	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Further evaluation of cumulative fatigue damage due to fatigue is provided in Section 3.5.1.1.4 and Section 4.6.
3.5.1.2	Penetration sleeves, bellows, and dissimilar metal welds.	NUREG-1801 Components Containment Penetrations Bellows	Cracking due to cyclic loading, or crack initiation and growth due to SCC	Containment ISI (B.1.26) and Containment leak rate test (B.1.28)	Yes, detection of aging effects is to be evaluated	Consistent with NUREG-1801, with exception. The exceptions to ASME Section XI, Subsection IWE are described in Section B.1.26. Further evaluation of Cracking due to Cyclic Loading and SCC is described in Section 3.5.1.1.5. CLB fatigue analyses exist for penetration bellows. Therefore, NUREG-1801 line II.B4.1-c does not apply.
3.5.1.3	Penetration sleeves, penetration bellows, and dissimilar metal welds	NUREG-1801 Components Containment Penetrations (Electrical) Containment Penetrations (Mechanical)	Loss of material due to corrosion	Containment ISI (B.1.26) and Containment leak rate test (B.1.28)	No	Consistent with NUREG-1801, with exception. The exceptions to ASME Section XI, Subsection IWE are described in Section B.1.26.
3.5.1.4	Personnel airlock and equipment hatch	NUREG-1801 Components Hatches	Loss of material due to corrosion	Containment ISI (B.1.26) and Containment leak rate test (B.1.28)	No	Consistent with NUREG-1801, with exception. The exceptions to ASME Section XI, Subsection IWE are described in Section B.1.26. NUREG-1801, line II.B4.2-a includes "coatings program, if credited." Protective Coating Monitoring and Maintenance (B.1.32) is credited for managing aging effects inside containment.

Table 3.5-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for containments, structures and component supports (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1.5	Personnel airlock and equipment hatch	NUREG-1801 Components Hatches	Loss of leak tightness in closed position due to mechanical wear of locks, hinges and closure mechanism	Containment leak rate test (B.1.28) and Plant Technical Specifications	No	Consistent with NUREG-1801.
3.5.1.6	Seals, gaskets, and moisture barriers	NUREG-1801 Components Seals	Loss of sealant and leakage through containment due to deterioration of joint seals, gaskets, and moisture barriers	Containment ISI (B.1.26) and Containment leak rate test (B.1.28)	No	Consistent with NUREG-1801, with exception. The exceptions to ASME Section XI, Subsection IWE are described in Section B.1.26.
3.5.1.7	Concrete elements: foundation, walls, dome.	See Discussion Column	Aging of accessible and inaccessible concrete areas due to leaching of calcium hydroxide, aggressive chemical attack, and corrosion of embedded steel	Containment ISI (B.1.26)	Yes, a plant specific aging management program is required for inaccessible areas as stated	Not applicable for a Mark I containment
3.5.1.8	Concrete elements: foundation	See Discussion Column	Cracks, distortion, and increases in component stress level due to settlement	Structures Monitoring (B.1.30)	No, if within the scope of the applicant's structures monitoring program	Not applicable for a Mark I containment
3.5.1.9	Concrete elements: foundation	See Discussion Column	Reduction in foundation strength due to erosion of porous concrete subfoundation	Structures Monitoring (B.1.30)	No, if within the scope of the applicant's structures monitoring program	Not applicable for a Mark I containment

Table 3.5-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for containments, structures and component supports (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1.10	Concrete elements: foundation, dome, and wall	See Discussion Column	Reduction of strength and modulus due to elevated temperature	Plant specific	Yes, for any portions of concrete containment that exceed specified temperature limits	Not applicable for a Mark I containment
3.5.1.11	Prestressed containment: tendons and anchorage components	See Discussion Column	Loss of prestress due to relaxation, shrinkage, creep, and elevated temperature	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Further evaluation of loss of prestress is provided in Section 3.5.1.1.9.
3.5.1.12	Steel elements: liner plate, containment shell	NUREG-1801 Components Downcomers Drywell Heads Drywells Suppression Chambers Evaluated with NUREG-1801 Components Vent Headers Vent Lines	Loss of material due to corrosion in accessible and inaccessible areas	Containment ISI (B.1.26) and Containment leak rate test (B.1.28)	Yes, if corrosion is significant for inaccessible areas	Consistent with NUREG-1801, with exception. The exceptions to NUREG-1801 for evaluation of ECCS suction header are described in Section 3.5.1.2.9. The exceptions to ASME Section XI, Subsection IWE are described in Section B.1.26. Further evaluation of Loss of Material due to Corrosion in Inaccessible Areas of Steel Containment Shell or Liner Plate is described in Section 3.5.1.1.3. NUREG-1801, lines II.B2.1.1-a, II.B2.2.2-a, II.B3.1.1-a, and II.B3.2.2-a do not apply to Mark I containments.
3.5.1.13	Steel elements: vent header, drywell head, torus, downcomers, pool shell	NUREG-1801 Components Suppression Chambers Vent Headers Vent Line Bellows	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Further evaluation of cumulative fatigue damage due to fatigue is provided in Section 3.5.1.1.4 and Section 4.6.

Table 3.5-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for containments, structures and component supports (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1.14	Steel elements: protected by coating	NUREG-1801 Components Downcomers Drywell Heads Drywells Suppression Chambers Evaluated with NUREG-1801 Components Vent Headers Vent Lines	Loss of material due to corrosion in accessible areas only	Protective coating monitoring and maintenance (B.1.32)	No	Consistent with NUREG-1801, with exception. The exceptions to NUREG-1801 for evaluation of ECCS suction header are described in Section 3.5.1.2.9. NUREG-1801, lines II.B2.1.1-a, II.B2.2.2-a, II.B3.1.1-a, and II.B3.2.2-a do not apply to Mark I containments.
3.5.1.15	Prestressed containment: tendons and anchorage components	See Discussion Column	Loss of material due to corrosion of prestressing tendons and anchorage components	Containment ISI (B.1.26)	No	Not applicable for a Mark I containment
3.5.1.16	Concrete elements: foundation, dome, and wall	See Discussion Column	Scaling, cracking, and spalling due to freeze-thaw; expansion and cracking due to reaction with aggregate	Containment ISI (B.1.26)	No, if stated conditions are satisfied for inaccessible areas	Not applicable for a Mark I containment

Table 3.5-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for containments, structures and component supports (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1.17	Steel elements: vent line bellows, vent headers, downcomers	NUREG-1801 Components Vent Line Bellows	Cracking due to cyclic loads or Crack initiation and growth due to SCC	Containment ISI (B.1.26) and Containment leak rate test (B.1.28)	Yes, detection of aging effects is to be evaluated	Consistent with NUREG-1801, with exception. The exceptions to ASME Section XI, Subsection IWE are described in Section B.1.26. Further evaluation of Cracking due to Cyclic Loading and SCC is described in Section 3.5.1.1.5. CLB fatigue analyses exist therefore NUREG-1801 line II.B1.1.1-b (Mark 1 Containment steel elements) does not apply NUREG-1801, lines II.B2.1.1-b, and II.B2.2.2-c do not apply to Mark I containments.
3.5.1.18	Steel elements: Suppression chamber liner	See Discussion Column	Crack initiation and growth due to SCC	Containment ISI (B.1.26) and Containment leak rate test (B.1.28)	No	Not applicable for a Mark I containment
3.5.1.19	Steel elements: drywell head and downcomer pipes	See Discussion Column	Fretting and lock up due to wear	Containment ISI (B.1.26)	No	Material does not exist at Dresden or Quad Cities

Table 3.5-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for containments, structures and component supports (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1.20	All Groups except Group 6: accessible interior/ exterior concrete & steel components	NUREG-1801 Components Beam Seats Blowout Panels Concrete Beams Concrete Columns Concrete Curbs Concrete Manholes Concrete Shield Plugs Concrete Slabs Concrete Walls Foundations Metal Decking Metal Siding Misc. Steel Penetration Sleeves Precast Concrete Panels Steel Doors Steel Embedments Steel Panels and Cabinets Steel Plates Structural Steel	All types of aging effects	Structures Monitoring (B.1.30)	No, if within the scope of the applicant's structures monitoring program and a plant-specific aging management program is required for inaccessible areas as stated	Consistent with NUREG-1801. Further evaluation of Aging of Structures Not Covered by Structures Monitoring Program is described in Section 3.5.1.1.6. Dresden and Quad Cities are Group 2 reactor buildings with steel superstructures NUREG-1801, line III.A2.2-a.) Dresden and Quad Cities do not use stainless steel lined, carbon steel tanks as evaluated in NUREG-1801, line III.A8.2-a. Fuel storage facility, refueling canal identified in NUREG-1801, lines III.A5.1-a,b,c,d,f are evaluated as part of the reactor building in NUREG-1801, lines III.A2.1-a,b,c,d,f. Spent fuel pool identified in NUREG-1801, line III.A5.2-a is evaluated as part of the reactor building in NUREG-1801, line III.A.2.2-a.

Table 3.5-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for containments, structures and component supports (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1.21	Groups 1-3, 5, 7-9: inaccessible concrete components, such as exterior walls below grade and foundation	NUREG-1801 Components Concrete Beams Concrete Columns Concrete Manholes Concrete Slabs Concrete Walls Foundations Evaluated with NUREG-1801 Components Concrete Duct Banks	Aging of inaccessible concrete areas due to aggressive chemical attack, and corrosion of embedded steel	Plant specific	Yes, a plant specific aging management program is required for inaccessible areas as stated	Further evaluation of Aging Management of Inaccessible Areas is described in Section 3.5.1.1.7. Fuel storage facility, refueling canal identified in NUREG-1801, lines III.A5.1-e,g are evaluated as part of the reactor building in NUREG-1801, lines III.A2.1-e,g.

Table 3.5-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for containments, structures and component supports (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1.22	Group 6: all accessible/ inaccessible concrete, steel, and earthen components	NUREG-1801 Components Concrete Curbs Concrete Slabs Concrete Stairs Concrete Walls Foundations Metal Siding Misc. Steel Precast Concrete Panels Steel Embedments Steel Panels and Cabinets Steel Plates Steel Sump Screens Structural Steel Evaluated with NUREG-1801 Components Concrete Canal Weirs	All types of aging effects, including loss of material due to abrasion, cavitation, and corrosion	Inspection of Water-Control Structures (B.1.31) or FERC/ US Army Corps of Engineers dam inspections and maintenance	No	Consistent with NUREG-1801, with exception. The exceptions to structural aging effect for concrete due to settlement are described in Section 3.5.1.2.1. The exceptions to structural aging effect due to freeze-thaw are described in Section 3.5.1.2.2. The exceptions to structural aging effect due to leaching of calcium hydroxide are described in Section 3.5.1.2.3. The exceptions to structural aging effect due to reaction with aggregates are described in Section 3.5.1.2.4. The exceptions to structural aging effect due to abrasion and cavitation are described in Section 3.5.1.2.5. The exceptions to structural aging effects due to corrosion of embedded steel are described in Section 3.5.1.2.6. The exceptions to structural aging effects due to aggressive chemical attack are described in Section 3.5.1.2.7. Earthen water control structures evaluated in NUREG-1801, line III.A6.4-a are not in the scope of license renewal.
3.5.1.23	Group 5: liners	NUREG-1801 Components Liners	Crack initiation and growth from SCC and loss of material due to crevice corrosion	Water Chemistry Program (B.1.2) and Monitoring of spent fuel pool water level	No	Consistent with NUREG-1801.
3.5.1.24	Groups 1-3, 5, 6: all masonry block walls	NUREG-1801 Components Masonry Walls	Cracking due to restraint, shrinkage, creep, and aggressive environment	Masonry Wall (B.1.29)	No	Consistent with NUREG-1801. Fuel storage facility, refueling canal identified in NUREG-1801, line III.A5.3-a are evaluated as part of the reactor building in NUREG-1801, line III.A2.3-a.

Table 3.5-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for containments, structures and component supports (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1.25	Groups 1-3, 5, 7-9: foundation	NUREG-1801 Components Concrete Slabs Foundations	Cracks, distortion, and increases in component stress level due to settlement	Structures Monitoring (B.1.30)	No, if within the scope of the applicant's structures monitoring program	Consistent with NUREG-1801. Further evaluation of Cracking, Distortion, and Increase in Component Stress Level due to Settlement; Reduction of Foundation Strength due to Erosion of Porous Concrete Subfoundations is described in Section 3.5.1.1.1. Fuel storage facility, refueling canal identified in NUREG-1801, lines III.A5.1-h is evaluated as part of the reactor building in NUREG-1801, lines III.A2.1-h.
3.5.1.26	Groups 1-3, 5-9: foundation	NUREG-1801 Components Concrete Slabs Foundations	Reduction in foundation strength due to erosion of porous concrete subfoundation	Structures Monitoring (B.1.30)	No, if within the scope of the applicant's structures monitoring program	Consistent with NUREG-1801. Further evaluation of Cracking, Distortion, and Increase in Component Stress Level due to Settlement; Reduction of Foundation Strength due to Erosion of Porous Concrete Subfoundations is described in Section 3.5.1.1.1. Fuel storage facility, refueling canal identified in NUREG-1801, lines III.A5.1-i is evaluated as part of the reactor building in NUREG-1801, lines III.A2.1-i.
3.5.1.27	Groups 1-5: concrete	NUREG-1801 Components Concrete Beams Concrete Columns Concrete Slabs Concrete Walls Foundations	Reduction of strength and modulus due to elevated temperature	Plant specific	Yes, for any portions of concrete that exceed specified temperature limits	Further evaluation of Reduction of Strength and Modulus of Concrete Structures due to Elevated Temperature is described in Section 3.5.1.1.2. Fuel storage facility, refueling canal identified in NUREG-1801, lines III.A5.1-j is evaluated as part of the reactor building in NUREG-1801, lines III.A2.1-j.

Table 3.5-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for containments, structures and component supports (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1.28	Groups 7, 8: liners	NUREG-1801 Components Liners	Crack Initiation and growth due to SCC; Loss of material due to crevice corrosion	Plant specific	Yes	The exceptions to NUREG-1801 for cracking due to crack initiation and growth due to SSC and loss of material due to crevice corrosion are described in Section 3.5.1.2.8. Dresden and Quad Cities do not use steel tanks lined with stainless as identified in NUREG-1801, line III.A8.2-b.
3.5.1.29	All Groups: support members: anchor bolts, concrete surrounding anchor bolts, welds, grout pad, bolted connections, etc.	NUREG-1801 Components Anchorage to Buildings, Including Bolted/ Welded Connections Concrete & Grout Instrument Racks, Frames, Panels, Etc, Support Members Vibration Isolation Elements Evaluated with NUREG-1801 Components Raceways	Aging of component supports	Structures Monitoring (B.1.30)	No, if within the scope of the applicant's structures monitoring program	Consistent with NUREG-1801.
3.5.1.30	Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds	NUREG-1801 Components Anchorage to Buildings, Including Bolted/ Welded Connections	Cumulative fatigue damage (CLB fatigue analysis exists)	TLAA evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA	Further evaluation of cumulative fatigue damage due to fatigue is provided in Section 3.5.1.1.8 and Section 4.6.

Table 3.5-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for containments, structures and component supports (Continued)

Ref No	Component	Components Evaluated	Aging Effect/ Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.5.1.31	Groups B1.1, B1.2, and B1.3: support members: anchor bolts, welds, guides, stops, and vibration isolators	NUREG-1801 Components Sliding Surfaces Support Members	Loss of material due to environmental corrosion; loss of mechanical function due to corrosion, distortion, dirt, overload, etc.	ISI (B.1.27)	No	Consistent with NUREG-1801, with exception. The exceptions to ASME Section XI, Subsection IWF" are described in Section 3.5.1.2.11.
3.5.1.32	Group B1.1: high strength low-alloy bolts	NUREG-1801 Components Bolting	Crack initiation and growth due to SCC	Bolting integrity (B.1.12)	No	Consistent with NUREG-1801, with exception. The exceptions to bolting integrity are described in Section 3.5.1.2.10. The exceptions to Bolting Integrity are described in Section B.1.12.

3.5.1.1 Further evaluation of aging management as recommended by NUREG-1801 for containments, structures and component supports

3.5.1.1.1 Cracking, Distortion, and Increase in Component Stress Level due to Settlement; Reduction of Foundation Strength due to Erosion of Porous Concrete Subfoundations (NUREG-1800, Section 3.5.2.2.1.2)

Cracks, distortion, increase in component stress level due to settlement are not applicable to Dresden and Quad Cities concrete structures and no aging management is required. The Dresden and Quad Cities licensing basis does not include a program to monitor concrete for settlement nor is a de-watering system in place. Dresden and Quad Cities structures are founded on rock or naturally compacted soil with no documented changes in groundwater conditions or a history of settlement.

Reduction in foundation strength, cracking, and differential settlement due to erosion of porous concrete subfoundation are not applicable to Dresden and Quad Cities and no aging management is required. Dresden and Quad Cities evaluations of Information Notices 97-11 and 98-26 concluded that no porous materials were used. The Dresden and Quad Cities licensing basis does not include a program to monitor concrete for settlement nor is a de-watering system in place. Dresden and Quad Cities structures are founded on rock or natural compacted soil and there is no documented change in groundwater conditions or history of settlement.

3.5.1.1.2 Reduction of Strength and Modulus of Concrete Structures due to Elevated Temperature (NUREG-1800, Section 3.5.2.2.1.3)

Reduction of strength and modulus due to elevated temperature are not applicable for Dresden and Quad Cities concrete structures and no aging management is required. Dresden and Quad Cities normal operating temperatures are less than 150°F general and are less than 200°F local.

3.5.1.1.3 Loss of Material due to Corrosion in Inaccessible Areas of Steel Containment Shell or Liner Plate (NUREG-1800, Section 3.5.2.2.1.4)

Corrosion of containment steel elements in inaccessible areas will be confirmed as insignificant in accordance with aging management program ASME Section XI, Subsection IWE (B.1.26.)

Since Dresden Unit 3 had more water leakage in the sand pocket area than Quad Cities and Dresden Unit 2, UT examinations were performed on Dresden Unit 3 sand pocket area in 1988. The examinations indicated that significant corrosion was not occurring, and it was concluded that corrosion is insignificant at Dresden Unit 2 and Quad Cities as well. A UT examination of the same locations at Dresden Unit 3 is conducted as an augmented inspection in accordance with aging management program ASME Section XI, Subsection IWE (B.1.26) to confirm that significant corrosion is not occurring.

A general visual inspection of the moisture barrier at the junction of the steel drywell shell and the concrete floor is performed once each inspection period in accordance with aging management program ASME Section XI, Subsection IWE (B.1.26.)

Dresden and Quad Cities documentation demonstrates that concrete meeting the requirements of ACI 318-63 and the guidance of ACI 201.2R-77 was used for the concrete in contact with the embedded drywell shell at the sand pocket location. The concrete is monitored for penetrating cracks that provide a path for water seepage in accordance with Structures Monitoring Program (B.1.30.)

3.5.1.1.4 Cumulative Fatigue Damage (NUREG-1800, Section 3.5.2.2.1.6)

Fatigue analyses of BWR Mark I and Mark II containment steel elements, penetration sleeves, and penetration bellows are TLAAs as defined in 10 CFR 54.3. Dresden and Quad Cities are Mark I containments. Cumulative fatigue damage of BWR Mark I containment steel elements, penetration sleeves, and penetration bellows are required to be evaluated in accordance with 10 CFR 54.21(c). The TLAA evaluation of cumulative fatigue damage is addressed in Section 4.6.

3.5.1.1.5 Cracking due to Cyclic Loading and SCC (NUREG-1800, Section 3.5.2.2.1.7)

For Mark 1 Containment steel elements and stainless steel containment penetrations (NUREG-1801 Items II.B1.1.1-d, and II.B4.1-d), stress corrosion cracking (SCC) is a concern for dissimilar metal welds, exposed to a corrosive environment. These components are in a sheltered environment, outside containment and inside the reactor building and are not exposed to a corrosive environment. Therefore, existing requirements for Appendix J leak rate testing (B.1.28) and Containment ISI plan surface inspections, in accordance with ASME Section XI, Subsection IWE (B.1.26), are adequate to detect cracking. In addition, other factors associated with SCC with regard to temperature, pressure and concentrated chlorides are not at threshold levels at the installed locations.

ASME Section XI, Subsection IWE weld examination categories E-B and E-F have been removed from the ASME Section XI, 1998 Edition. Both of these weld categories are considered to be part of the containment boundary surface in the current Dresden Containment Inservice Inspection (CISI) Program (ASME Section XI, Subsection IWE 1998 Edition) and Quad Cities CISI Programs and are subject to the examination requirements of Category E-A.

3.5.1.1.6 Aging of Structures Not Covered by Structures Monitoring Program (NUREG-1800, Section 3.5.2.2.2.1)

Structures Monitoring Program (B.1.30) is required to manage the following structural aging effects for accessible areas:

- Loss of material and cracking due to freeze-thaw of concrete.
- Increase in porosity and permeability and loss of strength due to leaching of calcium hydroxide of concrete.
- Increase in porosity and permeability, cracking, and loss of material (spalling, scaling) due to aggressive chemical attack of concrete.
- Expansion and cracking due to reaction with aggregates of concrete.

- Cracking, loss of bond, and loss of material (spalling, scaling) due to corrosion of embedded steel of concrete.

No aging management is required to manage the following structural aging effects for inaccessible areas.

- Loss of material and cracking due to freeze-thaw of concrete.

For loss of material and cracking due to freeze-thaw of concrete in inaccessible areas, no aging management is required. Dresden and Quad Cities are located in severe weathering conditions. Dresden and Quad Cities have documented evidence to show that the concrete air content is between 3% and 6%. Plant inspections did not show freeze-thaw degradation. Therefore, loss of material and cracking due to freeze-thaw of concrete in inaccessible areas are not applicable and no aging management is required.

- Increase in porosity and permeability and loss of strength due to leaching of calcium hydroxide of concrete.

For increase in porosity and permeability and loss of strength due to leaching of calcium hydroxide of concrete in inaccessible areas, no plant-specific aging management is required. Dresden and Quad Cities concrete is not exposed to flowing water and there is documented evidence that the concrete used was constructed in accordance with the recommendations in ACI 201.2R-77 for durability. Therefore, increase in porosity and permeability and loss of strength due to leaching of calcium hydroxide of concrete in inaccessible areas are not applicable and no plant-specific aging management is required.

- Expansion and cracking due to reaction with aggregates of concrete.

For expansion and cracking due to reaction with aggregates of concrete in inaccessible areas, no aging management is required. Dresden and Quad Cities documented evidence demonstrates that the concrete used meets the requirements of ACI 201.2R-77 with no evidence of reactive aggregates. Therefore, expansion and cracking due to reaction with aggregates of concrete in inaccessible areas are not applicable and no aging management is required.

- Loss of Material in Drywell Radial Beam Lubrite Baseplates

Aging management of loss of material due to galvanic corrosion, lock-up or wear of lubrite baseplates will be performed by One-Time Inspection (B.1.23) The torus saddle support lubrite baseplates will be visually inspected to verify unacceptable loss of material due to galvanic corrosion, lock-up or wear has not occurred. The drywell radial beam lubrite baseplates and torus saddle support lubrite baseplates are comprised of the same materials and are exposed to similar environments. The drywell radial beam lubrite baseplates are not accessible for inspection; therefore the inspection of the torus saddle support lubrite baseplates will be used as a representative inspection for aging of the drywell radial beam lubrite baseplates.

3.5.1.1.7 Aging Management of Inaccessible Areas (NUREG-1800, Section 3.5.2.2.2.2)

No plant-specific aging management is required to manage the following structural aging effects for inaccessible areas:

- Increase in porosity and permeability, cracking, and loss of material (spalling, scaling) due to aggressive chemical attack of concrete.
- Cracking, loss of bond, and loss of material (spalling, scaling) due to corrosion of embedded steel of concrete.

Dresden and Quad Cities ground water test data obtained during construction, the 1980’s, 1990’s, and 2000’s shows that the below-grade environment is not aggressive based on NUREG-1801 criteria with chlorides less than 500 ppm, sulfates less than 1500 ppm, and pH greater than 5.5. Examination of representative samples of below-grade concrete, when excavated for any reason, is included as part of the Structures Monitoring Program (B.1.30.) To ensure conditions are maintained throughout the period of extended operations, the Structures Monitoring Program (B.1.30) will be enhanced to include monitoring of below-grade water chemistry to demonstrate that the environment remains non-aggressive. Existing plant procedures will be used to periodically sample pH, chlorides, and sulfates.

3.5.1.1.8 Cumulative Fatigue Damage due to Cyclic Loading (NUREG-1800, Section 3.5.2.2.3.2)

Fatigue of component support members, anchor bolts, and welds for Groups B1.1, B1.2, and B1.3 component supports is a TLAA as defined in 10 CFR 54.3 only if a CLB fatigue analysis exists. Dresden and Quad Cities piping and component supports were designed to ASME Section VIII and ANSI USAS B31.1. Dresden Unit 3 ASME III Class I replacement piping was analyzed to Subsection NB, 1980 Edition including Summer 1982 Addenda. None of these codes required formal fatigue analysis of supports or design of supports for fatigue effects. Some ASME III Class MC support components were the subject of fatigue analysis in support of the Mark I “New Loads” program. Cumulative fatigue damage of ASME III Class MC support components are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The TLAA evaluation of the ASME III Class MC support components is addressed in Section 4.6.

3.5.1.1.9 Loss of Prestress due to Relaxation, Shrinkage, Creep, and Elevated Temperature (NUREG-1800, Section 3.5.2.2.1.5)

Loss of prestress forces due to relaxation, shrinkage, creep, and elevated temperature for PWR prestressed concrete containments and BWR Mark II prestressed concrete containments is a TLAA as defined in 10 CFR 54.3. Loss of prestress is not applicable to Dresden and Quad Cities Mark I containments.

3.5.1.2 Aging management programs or evaluations that are different than those described in NUREG-1801 for structures and component supports

3.5.1.2.1 Exception to structural aging effect for concrete due to settlement

The Dresden and Quad Cities licensing basis does not include a program to monitor concrete for settlement nor is a de-watering system in place. Dresden and Quad Cities structures are founded on rock or naturally compacted soil with no documented changes in groundwater conditions or a history of settlement. Cracks, distortion and increase in component stress level due to settlement are not applicable and no aging management is required.

3.5.1.2.2 Exception to structural aging effect due to freeze-thaw

For loss of material and cracking due to freeze-thaw of concrete in accessible areas, the aging management program is RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants (B.1.31.)

For loss of material and cracking due to freeze-thaw of concrete in inaccessible areas, no aging management is required.

Dresden and Quad Cities are located in severe weathering conditions. Dresden and Quad Cities have documented evidence to show that the concrete air content is between 3% and 6%. Plant inspections did not show freeze-thaw degradation. Therefore, loss of material and cracking due to freeze-thaw of concrete in inaccessible areas are not applicable and no aging management is required.

3.5.1.2.3 Exception to structural aging effect due to leaching of calcium hydroxide

For increase in porosity and permeability and loss of strength due to leaching of calcium hydroxide of concrete in accessible areas, the aging management program is RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants (B.1.31.)

For increase in porosity and permeability and loss of strength due to leaching of calcium hydroxide of concrete in inaccessible areas, no aging management is required.

Dresden and Quad Cities concrete is not exposed to flowing water and there is documented evidence that the concrete used was constructed in accordance with the recommendations in ACI 201.2R-77 for durability. Therefore, increase in porosity and permeability and loss of strength due to leaching of calcium hydroxide of concrete in inaccessible areas are not applicable and no plant-specific aging management is required.

3.5.1.2.4 Exception to structural aging effect due to reaction with aggregates

For expansion and cracking due to reaction with aggregates of concrete in accessible areas, the aging management program is RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants (B.1.31.)

For expansion and cracking due to reaction with aggregates of concrete in inaccessible areas, no aging management is required.

Dresden and Quad Cities documented evidence demonstrates that the concrete used meets the requirements of ACI 201.2R-77 with no evidence of reactive aggregates.

Therefore, expansion and cracking due to reaction with aggregates of concrete in inaccessible areas are not applicable and no aging management is required.

3.5.1.2.5 Exception of structural aging effect due to abrasion and cavitation

For loss of material due to abrasion and cavitation of concrete in accessible areas, the aging management program is RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants (B.1.31.)

For loss of material due to abrasion and cavitation of concrete in inaccessible areas, no aging management is required.

Dresden and Quad Cities water velocity at 3.68 feet per second (fps) is less than the industry abrasion erosion threshold velocity of 4 fps and less than the industry cavitation threshold velocity of 25 fps. Therefore, loss of material due to abrasion and cavitation of concrete in inaccessible areas is not applicable and no aging management is required.

3.5.1.2.6 Exception of structural aging effects due to corrosion of embedded steel

For cracking, loss of bond, and loss of material (spalling, scaling) due to corrosion of embedded steel of concrete in accessible areas, the aging management program is RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants (B.1.31.)

For cracking, loss of bond, and loss of material (spalling, scaling) due to corrosion of embedded steel of concrete in inaccessible areas, no aging management is required.

Dresden and Quad Cities ground water test data obtained during construction, the 1980's, 1990's, and 2000's shows that the below-grade environment is not aggressive based on NUREG-1801 criteria with chlorides less than 500 ppm, sulfates less than 1500 ppm, and pH greater than 5.5.

Examination of representative samples of below-grade concrete, when excavated for any reason, is included as part of the Structures Monitoring Program (B.1.30.)

To ensure conditions are maintained throughout the period of extended operations, the Structures Monitoring Program (B.1.30) will be enhanced to include monitoring of below-grade water chemistry to demonstrate that the environment remains non-aggressive. Existing plant procedures will be used to periodically sample pH, chlorides, and sulfates.

3.5.1.2.7 Exception of structural aging effects due to aggressive chemical attack

For increase in porosity and permeability, cracking, and loss of material (spalling, scaling) due to aggressive chemical attack of concrete in accessible areas, the aging management program is RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants (B.1.31.)

For increase in porosity and permeability, cracking, and loss of material (spalling, scaling) due to aggressive chemical attack of concrete in inaccessible areas, no aging management is required.

Dresden and Quad Cities ground water test data obtained during construction, the 1980's, 1990's, and 2000's shows that the below-grade environment is not aggressive based on NUREG-1801 criteria with chlorides less than 500 ppm, sulfates less than 1500 ppm, and pH greater than 5.5.

Examination of representative samples of below-grade concrete, when excavated for any reason, is included as part of the Structures Monitoring Program (B.1.30.)

To ensure conditions are maintained throughout the period of extended operations, the Structures Monitoring Program (B.1.30) will be enhanced to include monitoring of below-grade water chemistry to demonstrate that the environment remains non-aggressive. Existing plant procedures will be used to periodically sample pH, chlorides, and sulfates.

3.5.1.2.8 Exception to NUREG-1801 for cracking due to crack initiation and growth due to SSC and loss of material due to crevice corrosion

Stainless steel liners in the Dresden and Quad Cities floor drain surge tanks are not susceptible to cracking due to crack initiation and growth due to SSC or loss of material due to crevice corrosion and do not require aging management.

The floor drain surge tanks are vented and constructed of reinforced concrete with a stainless steel liner. The floor drain surge tank liner is not considered susceptible to SCC, since the tanks are vented (low service pressure), concentrated chlorides in the effluent are not expected, and the temperature of effluents would be ambient (less than threshold temperature of 140 degrees F for SCC). Stainless steel is susceptible to crevice corrosion given a sufficiently narrow crevice in the presence of oxygen. Crevice corrosion most frequently occurs in joints, and connections, or points of contact between metals and nonmetals, such as gasket surfaces, lap joints, and under bolt heads where contaminants can concentrate. The stainless steel liner has all welded seams and plug welds for anchorage, with all welds ground smooth. Therefore, the occurrence of crevice corrosion in the tank liner is not expected due to its configuration.

In addition, the floor drain surge tank has a drain system installed between the liner and concrete that would intercept leakage from behind the liner plate weld seams and drain the leakage to the attached pump house room. There have been no documented corrective action requests related to aging associated with the stainless steel liner plate drains.

3.5.1.2.9 Exception to NUREG-1801 for evaluation of ECCS Suction Header

For the evaluation of the ECCS suction header, an exception is taken to NUREG-1801 for item II.B.1.1.1-a. The ECCS suction header piping and fittings connected to the suppression chamber are evaluated with NUREG-1801 line V.D2.1.a (lines to suppression chamber) with the results presented in Table 3.2-1, reference numbers 3.2.1.2 and 3.2.1.4.

3.5.1.2.10 Exception to GALL for XI.M18, "Bolting Integrity"

Dresden and Quad Cities recirculation piping loop component supports inside the containment have ASTM 193 Grade B7 high strength low alloy steel bolting. The specification for ASTM 193 Grade B7 lists minimum yield strength of 105 ksi. with no upper yield strength installed. No documented evidence of cracking of ASTM 193 Grade B7 high strength low alloy steel bolting could be found in the Dresden and Quad Cities recirculation piping loop component supports operating experience data. Therefore, aging management of support members, including bolted connections for Class I piping and components will be managed by aging management program ASME Section XI, Subsection IWF (B.1.27.)

3.5.1.2.11 Exception to GALL for XI.S3, "ASME Section XI, Subsection IWF"

Aging of downcomer bracing will be managed with the inspections performed by the aging management program ASME Section XI, Subsection IWE (B.1.26.)

3.5.2 Components or aging effects that are not addressed in NUREG-1801 for the containments, structures and component supports

Table 3.5-2, “Aging management review results for containments, structures, and component supports that are not addressed in NUREG-1801“ contains aging management review results for the containments, structures, and component supports that are not addressed in NUREG-1801.

Table 3.5-2 Aging management review results for containments, structures, and component supports that are not addressed in NUREG-1801

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.5.2.1	Bus Duct Covers	Galvanized or Coated Carbon Steel	Weather exposed	Loss of material/ General corrosion	Structures Monitoring Program (B.1.30)	NUREG-1801 Chapter III does not address offsite power structures for coping with SBO.
3.5.2.2	Bus Duct Supports	Galvanized or Coated Carbon Steel	Weather exposed	Loss of material/ General corrosion	Structures Monitoring Program (B.1.30)	NUREG-1801 Chapter III does not address offsite power structures for coping with SBO.
3.5.2.3	Caulking/ Sealants	Silicone Rubber	Various	Hardening cracking/ Elastomer degradation	Structures Monitoring Program (B.1.30)	NUREG-1801 does not address caulking/ sealants used with turbine building water tight doors.
3.5.2.4	Caulking/ Sealants	Silicone Rubber	Weather exposed	Change in Material Properties/ Loss of resiliency, loss of strength, loss of elasticity	Structures Monitoring Program (B.1.30)	NUREG-1801 does not have this component.
3.5.2.5	Clevis Pins: Torus Columns, Vent Systems, ESF Lines	Carbon Steel or Stainless Steel	Submerged (torus grade water) and inside or outside containment	Loss of Material/ Mechanical wear	ASME Section XI, Subsection IWF (B.1.27)	Torus support columns applicable to Dresden only. NUREG-1801 does not address mechanical wear of clevis pins.
3.5.2.6	Dead End Structures	Galvanized or Coated Carbon Steel	Weather exposed	Loss of material/ General corrosion	Structures Monitoring Program (B.1.30)	NUREG-1801 Chapter III does not address offsite power structures for coping with SBO.
3.5.2.7	Door Seals	Silicone Rubber	Various	Hardening cracking/ Elastomer degradation	Structures Monitoring Program (B.1.30)	NUREG-1801 Chapter III does not address this specific secondary containment pressure boundary component.
3.5.2.8	Drywell Expansion Foam	Polyurethane	Outside containment	Hardening/ Radiation exposure	Time-Limited Aging Analysis evaluated for the period of extended operation	NUREG-1801 does not address line item for drywell air gap fill material. Plant-specific TLAA is described in Section 4.7.4.

Table 3.5-2 Aging management review results for containments, structures, and component supports that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.5.2.9	Masonry Walls	Concrete Block	Various	Cracking/ Restraint; shrinkage; creep; aggressive environment	Masonry Wall Program (B.1.29)	NUREG-1801 does not address these components for a Group 9 structure.
3.5.2.10	New Fuel Racks	Aluminum	Various	None	None	NUREG-1801 does not address aluminum material for new fuel racks. Aluminum is reactive, but develops an aluminum oxide film that protects it from further corrosion. No viable aging effects exist in this indoor environment for the new fuel storage racks.
3.5.2.11	Roofing	Vapor barrier coal tar pitch rigid insulation felt gravel or single ply hypalon pavers	Weather exposed	Separation and water in-leakage/ weathering	Structures Monitoring Program (B.1.30)	NUREG-1801 does not have this component .
3.5.2.12	Secondary Containment Boot Seals	Silicone Rubber	Various	Hardening cracking/ Elastomer degradation	Structures Monitoring Program (B.1.30)	NUREG-1801 Chapter III does not address this specific secondary containment pressure boundary component.
3.5.2.13	Seismic Gap Filler	Polyethylene	Weather exposed	Change in Material Properties/ Loss of resiliency, loss of strength, loss of elasticity	Structures Monitoring Program (B.1.30)	NUREG-1801 does not have this component.
3.5.2.14	Support Members	Stainless Steel	Inside or outside containment	Loss of material/ Pitting and crevice corrosion	ASME Section XI, Subsection IWF (B.1.27)	NUREG-1801 does not address stainless steel support members. Stainless steel pipe support stanchions are used on the recirc piping 28" lines at Dresden and Quad Cities.

Table 3.5-2 Aging management review results for containments, structures, and component supports that are not addressed in NUREG-1801 (Continued)

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.5.2.15	Thermowells	Stainless Steel; Dissimilar Metal Welds	Inside or outside containment	Loss of material/ General galvanic pitting and crevice corrosion	ASME Section XI, Subsection IWE (B.1.26)	NUREG-1801 does not address stainless steel thermowells installed in torus.
3.5.2.16	Transmission Towers	Galvanized or Coated Carbon Steel	Weather exposed	Loss of material/ General corrosion	Structures Monitoring Program (B.1.30)	NUREG-1801 Chapter III does not address offsite power structures for coping with SBO.

3.6 AGING MANAGEMENT OF ELECTRICAL AND INSTRUMENTATION AND CONTROLS

This section provides the results of aging management reviews for electrical and instrumentation and control components within the scope of license renewal. Aging management programs and activities are discussed in Appendix B.

Components Evaluated Consistent with NUREG-1801

The electrical and instrumentation and control components or component groups requiring aging management review are presented in Section 2.5. Aging management reviews were performed to assure that the component groups, materials, environments and aging effects referenced in NUREG-1801 are applicable to Dresden and Quad Cities and that the aging management program described in NUREG-1801 is applicable to Boiling Water Reactors (BWR) or to both Boiling Water Reactors and Pressurized Water Reactors (BWR/PWR).

When the components or component group and associated evaluations corresponded with an individual line item in NUREG-1801, Volume 2, the component aging management results were included in the appropriate line items in Table 3.6-1 “Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the electrical and instrumentation and control components.” Each line in Table 3.6-1 matches a line in Chapter 3 of NUREG-1800 that is applicable to Boiling Water Reactors (BWR).

Not all the electrical and instrumentation and control component types at Dresden and Quad Cities are listed in NUREG-1801, Volume 2. However, the aging management reviews presented in NUREG-1801, Volume 2 were applied to additional component types if the following criteria were satisfied:

- Constructed of the similar material as components in the NUREG-1801 line item,
- Assigned the same component intended function as components in the NUREG-1801 line item,
- Located in the same environment as components in the NUREG-1801 line item, and
- Have exhibited the same aging effects identified in the NUREG 1801 line item.

Component types meeting these criteria have been included in the presentation of aging management review results in Table 3.6-1 “Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for electrical and instrumentation and control components.” The third column of the table shows the component types included in each evaluation line. “NUREG 1801 Components” are those that correspond exactly with component types in NUREG 1801, Volume 2. “Evaluated with NUREG 1801 Components” shows the component types that meet the criteria above, and therefore share the same evaluation characteristics.

<u>Other Components Evaluated</u>

Table 3.6-2, "Aging management review results for the electrical and instrumentation and control components that are not addressed in NUREG-1801" presents the aging management review results for the remainder of the electrical and instrumentation and control components. These entries result from aging management review where the component type, material, environment or aging effect/mechanism differs from NUREG-1801, Volume 2 line item entries. Table 3.6-2 includes a line reference number, component group, material, environment, aging effect/mechanism, aging management program and discussion.

3.6.1 Aging management programs evaluated in NUREG-1801 that are relied upon for license renewal for the electrical and instrumentation and control components.

Table 3.6-1, "Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the electrical and instrumentation and control components" shows the component groups and aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the electrical and instrumentation and control components.

Table 3.6-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the electrical and instrumentation and control components

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.6.1.1	Electrical equipment subject to 10 CFR 50.49 environmental qualification (EQ) requirements	NUREG-1801 Components Electrical Equipment Subject to 10 CFR 50.49 (EQ) Requirements	Degradation due to various aging mechanisms	Environmental qualification of electric components (B.1.35)	Yes, TLAA	Further evaluation of degradation due to various aging mechanisms is provided in Section 3.6.1.1.1 and Section 4.4.
3.6.1.2	Electrical cables and connections not subject to 10 CFR 50.49 EQ requirements	NUREG-1801 Components Electrical Cables Evaluated with NUREG-1801 Components Connectors Fuse Blocks Splices Terminal Blocks	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced insulation resistance (IR); electrical failure caused by thermal/ thermoxidative degradation of organics; radiolysis and photolysis (ultraviolet [UV] sensitive materials only) of organics; radiation-induced oxidation; moisture intrusion	Aging management program for electrical cables and connections not subject to 10 CFR 50.49 EQ requirements (B.1.33)	No	Consistent with NUREG-1801.

Table 3.6-1 Aging management programs evaluated in NUREG-1801 that are relied on for license renewal for the electrical and instrumentation and control components (Continued)

Ref No	Component	Components Evaluated	Aging Effect/Mechanism	Aging Management Program	Further Evaluation Recommended	Discussion
3.6.1.3	Electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements that are sensitive to reduction in conductor insulation resistance (IR)	NUREG-1801 Components Electrical Cables	Embrittlement, cracking, melting, discoloration, swelling, or loss of dielectric strength leading to reduced IR; electrical failure caused by thermal/ thermoxidative degradation of organics; radiation-induced oxidation; moisture intrusion	Aging management program for electrical cables used in instrumentation circuits not subject to 10 CFR 50.49 EQ requirements	No	Consistent with NUREG-1801, with exception. The exceptions to instrumentation cable insulation are described in Section 3.6.1.2.1.
3.6.1.4	Inaccessible medium-voltage (2 kV to 15 kV) cables (e.g., installed in conduit or direct buried) not subject to 10 CFR 50.49 EQ requirements	NUREG-1801 Components Electrical Cables	Formation of water trees, localized damage leading to electrical failure (breakdown of insulation); water trees caused by moisture intrusion	Aging management program for inaccessible medium-voltage cables not subject to 10 CFR 50.49 EQ requirements	No	Consistent with NUREG-1801, with exception. The exceptions to formation of water treeing are described in Section 3.6.1.2.2. Only Dresden has 4160V cables in the scope of license renewal that are routed in underground ducts.

3.6.1.1 Further evaluation of aging management as recommended by NUREG-1801 for the electrical and instrumentation and control components

3.6.1.1.1 Electrical Equipment Subject to Environmental Qualification (NUREG-1800, Section 3.6.2.2.1)

Environmental qualification is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). Degradation due to various aging mechanisms for electrical equipment subject to 10 CFR 50.49 environmental qualification requirements is evaluated in accordance with 10 CFR 54.21(c)(1). The TLAA evaluation for electrical equipment subject to 10 CFR 50.49 environmental qualification requirements is provided in Section 4.4.

3.6.1.2 Aging management programs or evaluations that are different than those described in NUREG-1801 for the electrical and instrumentation and control components

3.6.1.2.1 Exception to Instrumentation Cable Insulation

Sensitive instrumentation circuit cable insulations were reviewed for their resilience against temperature, radiation and moisture environments. All cable insulation materials were assessed to have 60-year temperature and radiation thresholds greater than the bounding plant environments for which cables and connections are installed. The specified aging effects are not expected and therefore, no aging management is required. However, the cables of sensitive instrumentation circuits not subject to 10 CFR 50.49 requirements will be managed for aging due to adverse localized environments, as they are included in cables that are managed for aging per Item 3.6.1.2 of Table 3.6-1 and aging management program Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements (B.1.33.)

3.6.1.2.2 Exception to formation of water treeing

Five medium-voltage power cables at Dresden are exposed to significant moisture and significant voltage (subject to system voltage more than 25% of the time). Prior to the extended period of operation, these five medium-voltage power cables will be replaced with cables that are resistant to insulation degradation due to water treeing, and therefore no aging management is required.

3.6.2 Components or aging effects that are not addressed in NUREG-1801 for the electrical and instrumentation and control components

Table 3.6-2, "Aging management review results for the electrical and instrumentation and control components that are not addressed in NUREG-1801" contains aging management review results for the electrical and instrumentation and control components that are not addressed in NUREG-1801.

Table 3.6-2 Aging management review results for the electrical and instrumentation and control components that are not addressed in NUREG-1801

Ref No	Component Group	Material	Environment	Aging Effect/ Mechanism	Aging Management Program	Discussion
3.6.2.1	High Voltage Transmission Conductors	Aluminum Conductor Steel Reinforced	Outdoors: sun, weather, humidity, and moisture	Loss of material/ Corrosion	None	NUREG-1801 does not address aluminum conductor. The plant outdoor environment is not subject to heavy industry air pollution or saline environment. Aluminum is reactive, but develops an aluminum oxide film that protects it from further corrosion.
3.6.2.2	Insulators	Polyester Glass	Indoor and outdoor environments	Embrittlement cracking melting discoloration swelling or loss of dielectric strength leading to reduced insulation resistance electrical failure	Periodic Visual Inspection of Electrical Bus Duct Insulation (B.2.2)	NUREG-1801 does not address polyester glass.
3.6.2.3	Insulators	Porcelain	Indoor and outdoor environments	None	None	NUREG-1801 does not address porcelain. The plant outdoor environment is not subject to heavy industrial air pollution or saline environment. Plant indoor and outdoor environments are not conducive to promoting aging degradation of porcelain components.

CHAPTER 4
TIME-LIMITED AGING ANALYSES

List of Tables

4.0 TIME-LIMITED AGING ANALYSES

4.1 INTRODUCTION

This chapter presents descriptions of the Time-Limited Aging Analyses (TLAAs) for Dresden and Quad Cities in accordance with 10 CFR 50.34(a) and 54.21(c). The chapter is divided into sections, each containing a number of TLAAs on a common general category:

- Neutron Embrittlement of the Reactor Vessel and Internals
- Metal Fatigue of the Reactor Vessel, Internals, and Primary Coolant Boundary Piping and Components
- Environmental Qualification of Electrical Equipment (EQ)
- Loss of Prestress in Concrete Containment Tendons
- Fatigue of the Primary Containment, Attached Piping, and Components
- Other Plant-Specific TLAAs

Information about the TLAAs in a general category is presented within each section, as follows:

Applicability:	The plants to which this TLAA applies are identified.
Summary Description:	A brief description of the TLAA topic is provided.
Analysis:	A description of the current license analysis is provided.
Disposition:	The disposition is provided and classified in accordance with 10 CFR 54.21(c)(1) as: ▪ Validation, ▪ Revision, or ▪ Aging Management

NUREG-1801 identifies numerous aging effects that require evaluation as possible TLAAs in accordance with 10 CFR 54.21(c). Each of these is summarized in NUREG-1800 and presented in Section 3 of this LRA and referenced to the appropriate TLAA section.

4.1.1 Identification of TLAAs

The scope and methods for identifying TLAAs are consistent with the NUREG-1800 *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants.* (SRP)

Under the 10 CFR 54 License Renewal Rule (the Rule), an analysis, calculation, or evaluation is a “Time-Limited Aging Analysis” (TLAA) only if it meets all six of the defining criteria per 10 CFR 54.3(a). These are:

(1) Involve systems, structures, and components within the scope of license renewal,

(2) Consider the effects of aging;
(3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;
(4) Were determined to be relevant by the licensee in making a safety determination;
(5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, and
(6) Are contained or incorporated by reference in the CLB (current licensing basis).

A list of potential generic TLAAs was assembled from the SRP, industry guidance and experience, including

- The NUREG-1800 *Standard Review Plan for License Renewal*
- The NEI 95-10 *Industry Guideline for Implementing the Requirements of 10 CFR 54 the License Renewal Rule*
- The 10 CFR 54 Final Rule "Statement of Considerations," and
- Prior license renewal applications.

The Dresden and Quad Cities current licensing basis (CLB) was searched to confirm the occurrence of plant-specific TLAAs and to identify additional unit-specific TLAAs. The CLB search included the following documents:

- Updated Final Safety Analysis Reports (UFSARs)
- Operating License and License Conditions
- Technical Specifications
- Technical Requirements Manuals
- Safety Evaluation Reports (SERs)
- Exelon and NRC Licensing Correspondence
- Licensing basis program documents, such as the ISI and EQ.

The resulting list of potential TLAAs was reviewed (screened) against the six 10 CFR 54.3(a) criteria with the aid of supporting documents, such as:

- Environmental Qualification Binders
- ISI reports (ASME XI Summaries of Reportable Indications)
- Design Basis Documents
- Drawings
- Specifications
- Calculations
- Containment Plant Unique Analysis Report (PUAR)
- Procedures
- Supporting databases.

The supporting sources confirmed the screening and provided the information needed for dispositions.

The Rule requires that these TLAAs be evaluated to demonstrate that

(i) The analyses remain valid for the period of extended operation;
(ii) The analyses have been projected to the end of the period of extended operation; or
(iii) The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

Each TLAA was dispositioned by one of these three methods.

4.1.2 Identification of Exemptions

The rule requires a list of plant-specific exemptions granted pursuant to 10 CFR 50.12 and is based on time-limited aging analyses as defined in §54.3.

A search of docketed correspondence, the operating licenses, and the Updated Final Safety Analysis Reports (UFSARs) was made to identify all exemptions in effect. Each exemption in effect was evaluated to determine if it involved a TLAA as defined in 10 CFR 54.3. There are no exemptions based on time-limited aging analyses in effect.

4.1.3 Summary of Results

Six general categories of TLAAs applicable to Dresden and Quad Cities were identified in Sections 4.2 through 4.7 of this chapter, with their dispositions. A summary is presented in Table 4.1-1. The table includes a reference to the applicable section of this report that discusses the TLAA.

Table 4.1-1: List of Time-Limited Aging Analyses (TLAAs)

TLAA	Description	Applies to	Disposition Category	Section
1.	Neutron Embrittlement of the Reactor Vessel and Internals			4.2
	Reactor Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement	Dresden and Quad Cities	Revision of the analysis	4.2.1
	Adjusted Reference Temperature for Reactor Vessel Materials Due to Neutron Embrittlement	Dresden and Quad Cities	Revision of the analysis	4.2.2
	Reflood Thermal Shock Analysis of the Reactor Vessel	Dresden and Quad Cities	Validation of the analysis for the period of extended operation	4.2.3
	Reflood Thermal Shock Analysis of the Reactor Vessel Core Shroud and Repair Hardware	Dresden and Quad Cities	Validation of the analysis for the period of extended operation	4.2.4
	Reactor Vessel Thermal Limit Analysis: Operating Pressure – Temperature Limits	Dresden and Quad Cities	Revision of the analysis	4.2.5
	Reactor Vessel Circumferential Weld Examination Relief	Dresden only	Revision of the analysis	4.2.6
	Reactor Vessel Axial Weld Failure Probability	Dresden only	Revision of the analysis	4.2.7
2.	Metal Fatigue of the Reactor Vessel, Internals, and Reactor Coolant Pressure Boundary Piping and Components			4.3
	Reactor Vessel Fatigue Analyses	Dresden and Quad Cities	Management of the aging effect	4.3.1
	Fatigue Analysis of Reactor Vessel Internals			4.3.2
	Low-Cycle Thermal Fatigue Analyses of the Core Shroud and Repair Hardware	Quad Cities only	Validation of the analysis for the period of extended operation	4.3.2.1
	High-Cycle Flow-Induced Vibration Fatigue Analysis of Jet Pump Riser Braces	Dresden only	Revision of the analysis (repair)	4.3.2.2

Table 4.1-1: List of Time-Limited Aging Analyses (TLAAs) (Continued)

TLAA	Description	Applies to	Disposition Category	Section
	Reactor Coolant Pressure Boundary Piping and Component Fatigue Analysis			4.3.3
	ASME Section III Class I Reactor Coolant Pressure Boundary Piping and Component Fatigue Analysis	Dresden only	Management of the aging effect	4.3.3.1
	Reactor Coolant Pressure Boundary Piping and Components Designed to B31.1, ASME Section III Class 2 and 3, or ASME Section VIII Class B and C	Dresden and Quad Cities	Validation of the analysis for the period of extended operation	4.3.3.2
	Fatigue Analysis of the Isolation Condenser	Dresden only	Validation of the analysis for the period of extended operation	4.3.3.3
	Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)	Dresden and Quad Cities	Revision of the analysis	4.3.4
3.	Environmental Qualification of Electrical Equipment (EQ)	Dresden and Quad Cities	Management of the aging effect	4.4
4.	Loss of Prestress in Concrete Containment Tendons (The Dresden and Quad Cities containments have no prestress tendons.)	NA	Not applicable	4.5
5.	Fatigue of the Primary Containment, Attached Piping, and Components			4.6
	Fatigue Analyses of the Suppression Chamber, Vents, and Downcomers	Dresden and Quad Cities	Validation of the analysis for the period of extended operation, and Management of the aging effect	4.6.1

Table 4.1-1: List of Time-Limited Aging Analyses (TLAAs) (Continued)

TLAA	Description	Applies to	Disposition Category	Section
	Fatigue Analysis of SRV Discharge Piping Inside the Suppression Chamber, External Suppression Chamber Attached Piping, and Associated Penetrations	Dresden and Quad Cities	Validation of the analysis for the period of extended operation, and Management of the aging effect	4.6.2
	Drywell-to-Suppression Chamber Vent Line Bellows Fatigue Analyses	Dresden and Quad Cities	Validation of the analysis for the period of extended operation	4.6.3
	Primary Containment Process Penetration Bellows Fatigue Analysis	Dresden and Quad Cities	Validation of the analysis for the period of extended operation	4.6.4
6.	Other Plant-Specific TLAAs			4.7
	Reactor Building Crane Load Cycles	Dresden and Quad Cities	Validation of the analysis for the period of extended operation	4.7.1
	Metal Corrosion			4.7.2
	Corrosion Allowance for Power Operated Relief Valves	Quad Cities only	Validation of the analysis for the period of extended operation	4.7.2.1
	Degradation Rates of Inaccessible Exterior Drywell Plate Surfaces	Dresden and Quad Cities	Revision of the analysis	4.7.2.2
	Galvanic Corrosion in the Containment Shell and Attached Piping Components due to Stainless Steel ECCS Suction Strainers	Dresden and Quad Cities	Revision of the analysis	4.7.2.3
	Crack Growth Calculation of a Postulated Flaw in the Heat Affected Zone of an Arc Strike in the Suppression Chamber Shell	Dresden and Quad Cities	Validation of the analysis for the period of extended operation	4.7.3
	Radiation Degradation of Drywell Shell Expansion Gap Polyurethane Foam	Dresden and Quad Cities	Validation of the analysis for the period of extended operation	4.7.4
	High-Energy Line Break Postulation Based on Fatigue Cumulative Usage Factor (Usage factors were not used in the break location criteria for these plants.)	NA	Not applicable	4.7.5

4.2 NEUTRON EMBRITTLEMENT OF THE REACTOR VESSEL AND INTERNALS

The ferritic materials of the reactor vessel are subject to embrittlement due to high energy neutron exposure. Embrittlement means the material has lower toughness (e.g. will absorb less strain energy during a crack or rupture), thus allowing a crack to propagate more easily under load.

Toughness (indirectly measured in foot-pounds of absorbed energy in a Charpy impact test) is temperature dependent. In most materials, toughness increases with temperature up to a maximum value called the “upper-shelf energy,” or USE. Neutron embrittlement results in a decrease to the USE of reactor vessel steels. To reduce the potential for brittle fracture during vessel operation, account for the changes in material toughness as a function of neutron radiation exposure (fluence), and account for a reduction in toughness, operating pressure-temperature limit curves (P-T curves) are included in plant Technical Specifications. The P-T curves account for the decrease in material toughness associated with a given fluence in the future. The fluence is used to predict the loss in toughness of the reactor vessel materials. Based on the projected drop in toughness for a given fluence, the P-T curves are generated to provide a minimum temperature limit to which the vessel can be pressurized.

An initial nil-ductility reference temperature (RT_{NDT}) is determined for vessel materials before exposure to neutron radiation raises this transition temperature. This increase or shift in the initial nil-ductility reference temperature (ΔRT_{NDT}) means higher temperatures are required for the material to continue to act in a ductile manner. The P-T curves are determined by the RT_{NDT} and ΔRT_{NDT} values for the licensed operating period along with appropriate margins.

The reactor vessel ΔRT_{NDT} and USE, calculated on the basis of neutron fluence, are part of the licensing basis, and support safety determinations. Therefore, these calculations are TLAAs. The increases in RT_{NDT} (ΔRT_{NDT}) affect the bases for relief from circumferential weld inspection and its associated supporting calculation of limiting axial weld conditional failure probability. As such, circumferential weld examination relief and axial weld failure probability are also TLAAs. Section 4.2 includes the following TLAA discussions related to the issue of neutron embrittlement:

- Reactor Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement
- Adjusted Reference Temperature for Reactor Vessel Materials Due to Neutron Embrittlement
- Reflood Thermal Shock Analysis of the Reactor Vessel
- Reflood Thermal Shock Analysis of the Reactor Vessel Core Shroud and Repair Hardware
- Reactor Vessel Thermal Limit Analysis: Operating Pressure–Temperature Limits
- Reactor Vessel Circumferential Weld Examination Relief
- Reactor Vessel Axial Weld Failure Probability

4.2.1 Reactor Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

Upper shelf energy (USE) is the standard industry parameter used to indicate the maximum toughness of a material at high temperature. 10 CFR 50 Appendix G requires the predicted end-of-life Charpy impact test USE for reactor vessel materials to be at least 50 ft-lb (absorbed energy), unless an approved analysis supports a lower value. Initial unirradiated test data are not available for the Dresden or Quad Cities reactor vessels to demonstrate a minimum 50 ft-lb USE by standard methods. End-of-life fracture energy was evaluated by using an equivalent margin analysis (EMA) methodology approved by the NRC in NEDO-32205-A. This analysis confirmed that an adequate margin of safety against fracture, equivalent to 10 CFR 50 Appendix G requirements, does exist.

The end-of-life upper shelf energy calculations satisfy the criteria of 10 CFR 54.3(a). As such, these calculations are a TLAA.

Analysis

The Dresden and Quad Cities reactor vessels were designed for a 40-year life with an assumed neutron exposure of less than 10^{19} n/cm^2 from energies exceeding 1 MeV. The current licensing basis calculations use realistic calculated fluences that are lower than this limiting value. The design basis value of 10^{19} n/cm^2 bounds calculated fluences for the original 40-year term for all four units.

The tests performed on reactor vessel materials under the code of record provided limited Charpy impact data. It was not possible to develop original Charpy impact test USE values using the ASME III NB-2300, Summer 1972 (and later) methods invoked by 10 CFR 50 Appendix G. Therefore, alternative methods approved by the NRC in NEDO-32205-A, were used to demonstrate compliance with the 40-year 50 ft-lb USE requirement.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

Fluences were calculated for the Dresden and Quad Cities reactor vessels for the extended 60-year (54 EFPY) licensed operating periods, using the methodology of NEDC-32983P, “General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation,” which was approved by the NRC in a letter dated September 14, 2001 from S.A. Richards (NRC) to J.F. Klapproth (GE). (Reference 4.2) One bounding fluence calculation was performed for Dresden and Quad Cities. Peak fluences were calculated at the vessel inner surface (inner diameter), for purposes of evaluating USE. The value of neutron fluence was also calculated for the 1/4T location into the vessel wall measured radially from the inside diameter (ID), using Equation 3 from Paragraph 1.1 of Regulatory Guide 1.99, Revision 2. This 1/4T depth is recommended in the ASME Boiler and Pressure Vessel Code Section XI, Appendix G, 1998 Edition, Addendum 2000, Sub-article G-2120 as the maximum postulated defect depth.

The 54 EFPY USE was evaluated by an equivalent margin analysis (EMA) using the 54 EFPY calculated fluence and Dresden and Quad Cities surveillance capsule results. EPRI TR-113596, "BWR Reactor Pressure Vessel Inspection and Flaw Evaluation Guidelines," BWRVIP-74, September 1999, performs a generic analysis and determines that the percent reduction in Charpy USE for the limiting BWR/3-6 plates and BWR/2-6 welds are 23.5 per cent and 39 per cent respectively. Summary tables 4.2.1-1 through 4.2.1-8 provide results of the equivalent margin analysis for limiting welds and plates on the reactor vessels at both plants. The results show that the limiting USE EMA percent is less than the BWRVIP-74 EMA percent acceptance criterion in all cases.

A report summarizing the results of the equivalent margin analysis will be submitted for NRC approval by December 31, 2003. This report will be the basis for demonstrating compliance to 10 CFR 50 Appendix G for the Dresden and Quad Cities reactor vessels. Exelon will manage the 54 EFPY USE values in conjunction with the surveillance capsule results from the BWRVIP Integrated Surveillance Program (BWRVIP reports 78 and 86). See the Reactor Vessel Surveillance Program (B.1.22).

Table 4.2.1-1: Equivalent Margin Analysis for Dresden Unit 2 Plate Material

BWR/3-6 PLATE
Surveillance Plate USE: %Cu = 0.19 1st Capsule Fluence = 1.3×10^{16} n/cm^2 2nd Capsule Fluence = 5.2×10^{16} n/cm^2 1st Capsule Measured % Decrease = 9 (Charpy Curves) 2nd Capsule Measured % Decrease = 13 (Charpy Curves) 1st Capsule R.G. 1.99 Predicted % Decrease = 6 (R.G. 1.99, Figure 2) 2nd Capsule R.G. 1.99 Predicted % Decrease = 9 (R.G. 1.99, Figure 2)
Limiting Beltline Plate USE: %Cu = 0.23 54 EFPY 1/4T Fluence = 3.9×10^{17} n/cm^2 R.G. 1.99 Predicted % Decrease = 15.5 (R.G. 1.99, Figure 2) Adjusted % Decrease = 21 (R.G. 1.99, Position 2.2)
21 $\leq$ 23.5% , so vessel plates are bounded by equivalent margin analysis

Table 4.2.1-2: Equivalent Margin Analysis for Dresden Unit 2 Weld Material

BWR/2-6 WELD
<u>Surveillance Weld USE:</u> %Cu = 0.17 1st Capsule Fluence = 1.3×10^{16} n/cm^2 2nd Capsule Fluence = 5.2×10^{16} n/cm^2 1st Capsule Measured % Decrease = N/A (Charpy Curves) 2nd Capsule Measured % Decrease = N/A (Charpy Curves) 1st Capsule R.G. 1.99 Predicted % Decrease = 7 (R.G. 1.99, Figure 2) 2nd Capsule R.G. 1.99 Predicted % Decrease = 9 (R.G. 1.99, Figure 2)
<u>Limiting Beltline Weld USE:</u> %Cu = 0.24 54 EFPY 1/4T Fluence = 3.9×10^{17} n/cm^2 R.G. 1.99 Predicted % Decrease = 18.5 (R.G. 1.99, Figure 2) Adjusted % Decrease = N/A (R.G. 1.99, Position 2.2)
18.5 $\leq$ 39% , so vessel welds are bounded by equivalent margin analysis

Table 4.2.1-3: Equivalent Margin Analysis for Dresden Unit 3 Plate Material

BWR/3-6 PLATE
Surveillance Plate USE: %Cu = 0.13 1st Capsule Fluence = 9.3×10^{15} n/cm^2 2nd Capsule Fluence = 2.9×10^{16} n/cm^2 3rd Capsule Fluence = 7.1×10^{16} n/cm^2 1st Capsule Measured % Decrease = 0 (Charpy Curves) 2nd Capsule Measured % Decrease = - 4 (increase) (Charpy Curves) 3rd Capsule Measured % Decrease = - 11 (increase) (Charpy Curves) 1st Capsule R.G. 1.99 Predicted % Decrease = 4 (R.G. 1.99, Figure 2) 2nd Capsule R.G. 1.99 Predicted % Decrease = 6 (R.G. 1.99, Figure 2) 3rd Capsule R.G. 1.99 Predicted % Decrease = 7 (R.G. 1.99, Figure 2)
Limiting Beltline Plate USE: %Cu = 0.23 54 EFPY 1/4T Fluence = 3.9×10^{17} n/cm^2 R.G. 1.99 Predicted % Decrease = 15.5 (R.G. 1.99, Figure 2) Adjusted % Decrease = N/A (R.G. 1.99, Position 2.2)
15.5 $\leq$ 23.5% , so vessel plates are bounded by equivalent margin analysis

Table 4.2.1-4: Equivalent Margin Analysis for Dresden Unit 3 Weld Material

BWR/2-6 WELD
Surveillance Weld USE: %Cu = 0.20 1st Capsule Fluence = 9.3×10^{15} n/cm^2 2nd Capsule Fluence = 2.9×10^{16} n/cm^2 3rd Capsule Fluence = 7.1×10^{16} n/cm^2 1st Capsule Measured % Decrease = - 7 (increase) (Charpy Curves) 2nd Capsule Measured % Decrease = - 51 (increase) (Charpy Curves) 3rd Capsule Measured % Decrease = 0 (Charpy Curves) 1st Capsule R.G. 1.99 Predicted % Decrease = 7 (R.G. 1.99, Figure 2) 2nd Capsule R.G. 1.99 Predicted % Decrease = 9 (R.G. 1.99, Figure 2) 3rd Capsule R.G. 1.99 Predicted % Decrease = 11 (R.G. 1.99, Figure 2)
Limiting Beltline Weld USE: %Cu = 0.34 54 EFPY 1/4T Fluence = 2.9×10^{17} n/cm^2 R.G. 1.99 Predicted % Decrease = 21.5 (R.G. 1.99, Figure 2) Adjusted % Decrease = N/A (R.G. 1.99, Position 2.2)
21.5 $\leq$ 39% , so vessel welds are bounded by equivalent margin analysis

Table 4.2.1-5: Equivalent Margin Analysis for Quad Cities Unit 1 Plate Material

BWR/3-6 PLATE
Surveillance Plate USE: %Cu = 0.22 1st Capsule Fluence = 1.03×10^{16} n/cm^2 2nd Capsule Fluence = 5.5×10^{16} n/cm^2 1st Capsule Measured % Decrease = 0 (Charpy Curves) 2nd Capsule Measured % Decrease = 0 (Charpy Curves) 1st Capsule R.G. 1.99 Predicted % Decrease = 7 (R.G. 1.99, Figure 2) 2nd Capsule R.G. 1.99 Predicted % Decrease = 10 (R.G. 1.99, Figure 2)
Limiting Beltline Plate USE: %Cu = 0.27 54 EFPY 1/4T Fluence = 2.9×10^{17} n/cm^2 R.G. 1.99 Predicted % Decrease = 16.5 (R.G. 1.99, Figure 2) Adjusted % Decrease = N/A (R.G. 1.99, Position 2.2)
16.5 $\leq$ 23.5% , so vessel plates are bounded by equivalent margin analysis

Table 4.2.1-6: Equivalent Margin Analysis for Quad Cities Unit 1 Weld Material

BWR/2-6 WELD
Surveillance Weld USE: %Cu = 0.17 1st Capsule Fluence = 1.03×10^{16} n/cm^2 2nd Capsule Fluence = 5.5×10^{16} n/cm^2 1st Capsule Measured % Decrease = 5 (Charpy Curves) 2nd Capsule Measured % Decrease = 12 (Charpy Curves) 1st Capsule R.G. 1.99 Predicted % Decrease = 7 (R.G. 1.99, Figure 2) 2nd Capsule R.G. 1.99 Predicted % Decrease = 10 (R.G. 1.99, Figure 2)
Limiting Beltline Weld USE: %Cu = 0.27 54 EFPY 1/4T Fluence = 2.9×10^{17} n/cm^2 R.G. 1.99 Predicted % Decrease = 18.5 (R.G. 1.99, Figure 2) Adjusted % Decrease = 17.5 (R.G. 1.99, Position 2.2)
18.5 $\leq$ 39% , so vessel welds are bounded by equivalent margin analysis

Table 4.2.1-7: Equivalent Margin Analysis for Quad Cities Unit 2 Plate Material

BWR/3-6 PLATE
Surveillance Plate USE: %Cu = 0.09 1st Capsule Fluence = 1.69×10^{16} n/cm^2 2nd Capsule Fluence = 6.6×10^{16} n/cm^2 1st Capsule Measured % Decrease = 2 (Charpy Curves) 2nd Capsule Measured % Decrease = - 9 (increase) (Charpy Curves) 1st Capsule R.G. 1.99 Predicted % Decrease = 4 (R.G. 1.99, Figure 2) 2nd Capsule R.G. 1.99 Predicted % Decrease = 6 (R.G. 1.99, Figure 2)
Limiting Beltline Plate USE: %Cu = 0.18 54 EFPY 1/4T Fluence = 2.9×10^{17} n/cm^2 R.G. 1.99 Predicted % Decrease = 12 (R.G. 1.99, Figure 2) Adjusted % Decrease = N/A (R.G. 1.99, Position 2.2)
12 $\leq$ 23.5% , so vessel plates are bounded by equivalent margin analysis

Table 4.2.1-8: Equivalent Margin Analysis for Quad Cities Unit 2 Weld Material

BWR/2-6 WELD
Surveillance Weld USE: %Cu = 0.14 1st Capsule Fluence = 1.69 x 10^{16} n/cm^2 2nd Capsule Fluence = 6.6 x 10^{16} n/cm^2 1st Capsule Measured % Decrease = 12 (Charpy Curves) 2nd Capsule Measured % Decrease = 32 (Charpy Curves) 1st Capsule R.G. 1.99 Predicted % Decrease = 7 (R.G. 1.99, Figure 2) 2nd Capsule R.G. 1.99 Predicted % Decrease = 9 (R.G. 1.99, Figure 2)
Limiting Beltline Weld USE: %Cu = 0.24 54 EFPY 1/4T Fluence = 3.9 x 10^{17} n/cm^2 R.G. 1.99 Predicted % Decrease = 18.5(R.G. 1.99, Figure 2) Adjusted % Decrease = 39 (R.G. 1.99, Position 2.2)
39 $\leq$ 39% , so vessel welds are bounded by equivalent margin analysis

4.2.2 Adjusted Reference Temperature for Reactor Vessel Materials Due to Neutron Embrittlement

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

The initial RT_{NDT}, nil-ductility reference temperature, is the temperature at which a non-irradiated metal (ferritic steel) changes in fracture characteristics going from ductile to brittle behavior. RT_{NDT} was evaluated according to the procedures in the ASME Code, Paragraph NB-2331. Neutron embrittlement raises the initial nil-ductility reference temperature. 10 CFR 50 Appendix G defines the fracture toughness requirements for the life of the vessel. The shift to the initial nil-ductility reference temperature (ΔRT_{NDT}) is evaluated as the difference in the 30 ft-lb index temperatures from the average Charpy curves measured before and after irradiation. This increase (ΔRT_{NDT}) means that higher temperatures are required for the material to continue to act in a ductile manner. The adjusted reference temperature (ART) is defined as RT_{NDT} + ΔRT_{NDT} + margin. The margin is defined in Regulatory Guide 1.99, Revision 2. The P-T curves are developed from adjusted reference temperatures (ART) for the vessel materials. These are determined by the unirradiated RT_{NDT} and by the ΔRT_{NDT} calculations for the licensed operating period. Regulatory Guide 1.99 defines the calculation methods for ΔRT_{NDT}, ART, and end-of-life USE.

The ΔRT_{NDT} and ART calculations meet the criteria of 10 CFR 54.3(a). As such, they are TLAAs.

Analysis

The Dresden and Quad Cities reactor vessels were designed for a 40-year life with an assumed neutron exposure of less than 10^{19} n/cm^2 from energies exceeding 1 MeV. (References 4.3 and 4.4). The current licensing basis calculations use realistic calculated fluences that are lower than this limiting value. The design basis value of 10^{19} n/cm^2 bounds calculated fluences for the original 40-year term for all four units. The ΔRT_{NDT} values were determined using the embrittlement correlations defined in Regulatory Guide 1.99, Revision 2.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

Fluences were calculated for the Dresden and Quad Cities reactor vessels for the extended 60-year (54 EFPY) licensed operating periods, using the methodology of NEDC-32983P, “General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation,” which was approved by the NRC in a letter dated September 14, 2001 from S.A. Richards (NRC) to J.F. Klapproth (GE). (Reference 4.2) One bounding calculation was performed for Dresden and Quad Cities. Peak fluences were calculated at the vessel inner surface (inner diameter) for purposes of evaluating USE and ART. The value of neutron fluence was also calculated for the 1/4T location into the vessel wall measured radially from the inside diameter (ID), using Equation 3 from Paragraph 1.1 of Regulatory Guide 1.99, Revision 2. This 1/4T depth is recommended in the ASME Boiler and Pressure Vessel Code Section XI, Appendix G Sub-article G-2120 as the maximum postulated defect depth.

The 54 EFPY ΔRT_{NDT} for all belt line materials were calculated based on the embrittlement correlation found in Regulatory Guide 1.99, Revision 2. The peak fluence, $\Delta RT_{NDT,\ and}$ ART values for the 60-year (54 EFPY) license operating period are presented in Tables 4.2.2-1 through 4.2.2-2. These tables show that the limiting 54 EFPY ARTs allow P-T limits that will provide reasonable operational flexibility. Exelon will manage the 54 EFPY ΔRT_{NDT} and ART values in conjunction with the surveillance capsule results from the BWRVIP Integrated Surveillance Program (BWRVIP reports 78 and 86). See the Reactor Vessel Surveillance Program (B.1.22) for additional details.

It should be noted that the ΔRT_{NDT} and ART values are provided for one limiting material. Code Case N-588 was used for development of the associated Pressure-Temperature Operating curves, which caused the axial weld to be the limiting material. Due to the refinement in the approved methodology used to calculate the 54 EFPY fluence, the material with the limiting ART is the axial weld, with the exception of Dresden Unit 3 where the axial weld and girth weld ART values are identical. Code Case N-588 is required for Dresden Unit 3 and causes the axial weld to become the limiting material; therefore, information for the single limiting material is presented in Tables 4.2.2-1 and 4.2.2-2.

Table 4.2.2-1 54 EFPY Analysis Results for Dresden Units 2 & 3

Parameter	Unit 2	Unit 3
Peak Surface Fluence (n/cm^2)	5.7×10^{17} n/cm^2	5.7×10^{17} n/cm^2
1/4T Fluence (n/cm^2)	3.9×10^{17} n/cm^2	3.9×10^{17} n/cm^2
ΔRT_{NDT} (°F)	36	36
ART (°F)	104	104

Table 4.2.2-2 54 EFPY Analysis Results for Quad Cities Units 1 & 2

Parameter	Unit 1	Unit 2
Peak Surface Fluence (n/cm^2)	5.7×10^{17} n/cm^2	5.7×10^{17} n/cm^2
1/4T Fluence (n/cm^2)	3.9×10^{17} n/cm^2	3.9×10^{17} n/cm^2
ΔRT_{NDT} (°F)	36	36
ART (°F)	104	104

4.2.3 Reflood Thermal Shock Analysis of the Reactor Vessel

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

The Dresden and Quad Cities UFSARs describe an end-of-life thermal shock analysis performed on the reactor vessels for a design basis LOCA followed by a low-pressure coolant injection. The effects of embrittlement assumed by this thermal shock analysis will change with an increase in the licensed operating period. This analysis satisfies the criteria of 10 CFR 54.3(a). As such, this analysis is a TLAA.

Analysis

For the current operating period, a thermal shock analysis was originally performed on the reactor vessel components at Dresden and Quad Cities. The analysis assumed a design basis LOCA followed by a low-pressure coolant injection accounting for the full effects of neutron embrittlement at the end of life (40 years). The analysis showed that the total maximum vessel irradiation (1MeV) at the mid-core inside of the vessel to be 2.4×10^{17} n/cm^2 which was below the threshold level of any nil-ductility temperature shift for the vessel material. As a result, it was concluded that the irradiation effects on all locations of the reactor vessels could be ignored. However, this analysis only bounded 40 years of operation.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The original analysis has since been superseded by an analysis for BWR-6 vessels that is applicable to the Dresden and Quad Cities BWR-3 reactor vessels. (Reference 4.8) This revised analysis assumes end-of-life material toughness, which in turn depends on end-of-life adjusted reference temperature (ART). The critical location for fracture mechanics analysis is at ¼ of the vessel thickness (from the inside, 1/4T). For this event, the peak stress intensity occurs at approximately 300 seconds after the LOCA.

The analysis shows that at 300 seconds into the thermal shock event, the temperature of the vessel wall at 1.5 inches deep (which is 1/4T) is approximately 400°F (200°C). The 54 EFPY ART values described in Section 4.2.2 and tabulated in Tables 4.2.2-1 and 4.2.2-2 list the ARTs for the limiting material (weld metal) of the Dresden and Quad Cities reactor vessels. The worst-case calculated reactor vessel beltline material ART is 104°F which is well below the 400°F (204°C) 1/4T temperature predicted for the thermal shock event at the time of peak stress intensity. Therefore, the revised analysis is valid for the period of extended operation.

4.2.4 Reflood Thermal Shock Analysis of the Reactor Vessel Core Shroud and Repair Hardware

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

Radiation embrittlement may affect the ability of reactor vessel internals, particularly the core shroud and repair hardware, to withstand a low-pressure coolant injection (LPCI) thermal shock transient. Core shroud repair hardware was installed on the Dresden and Quad Cities core shrouds after 20 years of operation when cracks were found on the shroud. The analysis of core shroud strain due to reflood thermal shock is a TLAA because it is part of the current licensing basis, supports a safety determination, and is based on the calculated lifetime neutron fluence.

Analysis

The reactor vessel core shrouds were evaluated for a LPCI reflood thermal shock transient considering the embrittlement effects of lifetime radiation exposure (32 EFPY). The core shrouds receive the maximum irradiation on the inside surface opposite the midpoint of the fuel centerline. The total integrated neutron fluence at end of life at inside surface of the shroud is anticipated to be 2.7×10^{20} nvt (greater than 1 MeV). The maximum thermal shock stress in this region will be 155,700 psi equivalent to 0.57% strain. This strain range of 0.57% was calculated at the midpoint of the shroud, the zone of highest neutron irradiation. The calculated strain range of 0.57% represents a considerable margin of safety below measured values of percent reduction in area for annealed Type 304 stainless steel irradiated to 1×10^{21} nvt (greater than 1 MeV).

The value of percent reduction in area for Type 304 stainless steel is a minimum of approximately 38% for a temperature of 550°F with a neutron fluence of 1×10^{21} nvt (greater than 1 MeV) and a reduction in area of 52.5% for a temperature of 750°F with a neutron fluence of 6.9×10^{21} nvt (greater than 1 MeV). At lower values of temperature or neutron fluence, the percent reduction in area is generally higher. Therefore, thermal shock effects on the shroud at the point of highest irradiation level will not jeopardize the proper functioning of the shroud following the design basis accident (DBA) during the current licensed operating period (40 years).

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

As discussed above, core shroud components were evaluated for a reflood thermal shock event, considering the embrittlement effects of lifetime radiation exposure. The analysis includes the most irradiated point on the inner surface of the shroud where the calculated value of fluence for 40-year operating period as below the threshold (3.0×10^{20} n/cm^2) for material property changes due to irradiation. However, using the approved fluence methodology discussed in Section 4.2.2, the 54 EFPY fluence at the most irradiated point on the core shroud was calculated to be 5.85×10^{20} n/cm^2.

The allowable value of the thermal strain for irradiation levels in excess of 1×10^{21} n/cm^2 and for this faulted event is at least 20% (Reference 4.9). Since this value is far in excess of the calculated thermal shock strain amplitude of (0.6/2)% or 0.3%, the calculated thermal shock strain at the most irradiated location is acceptable considering the embrittlement effects for a 60-year operating period. Therefore, the peak thermal shock strain location is acceptable considering the embrittlement effects for a 60-year (54 EFPY) operating period.

The shroud repair hardware was designed with a 40-year design life. Since the hardware was installed more than 20 years into operation, the design life includes the period of extended operation and no additional evaluation is necessary.

4.2.5 Reactor Vessel Thermal Limit Analyses: Operating Pressure – Temperature Limits

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

The Adjusted Reference Temperature (ART) is the value of Initial RT_{NDT} + ΔRT_{NDT} + margins for uncertainties at a specific location. Neutron embrittlement increases the ART. Thus, the minimum temperature at which a reactor vessel is allowed to be pressurized increases. The ART of the limiting beltline material is used to correct the beltline P-T limits to account for irradiation effect.

10 CFR Part 50 Appendix G requires reactor vessel thermal limit analyses to determine operating pressure-temperature (P-T) limits for boltup, hydrotest, pressure tests and normal operating and anticipated operational occurrences. Operating limits for pressure and temperature are required for three categories of operation: 1) hydrostatic pressure tests and leak tests, referred to as Curve A; 2) non-nuclear heat-up / cooldown and low-level physics tests, referred to as Curve B; and 3) core critical operation, referred to as Curve C. Pressure/temperature limits are developed for three vessel regions: the upper vessel region, the core beltline region, and the lower vessel bottom head region.

The calculations associated with generation of the P-T curves satisfy the criteria of 10 CFR 54.3(a). As such, this topic is a TLAA.

Analysis

The Dresden and Quad Cities Technical Specifications contain P-T limit curves for heat-up, cooldown, and in-service leakage and hydrostatic testing and also limit the maximum rate of change of reactor coolant temperature. The criticality curves provide limits for both heat-up and criticality calculated for a 32 EFPY operating period. Because of the relationship between the P-T limits and the fracture toughness transition of the reactor vessel, both Dresden and Quad Cities will require new P-T limits to be calculated and approved before the extended period of operation.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

Revised P-T limits will be prepared and submitted to the NRC for approval prior to the start of the extended period of operation using an approved fluence methodology for Dresden and Quad Cities. The analysis will utilize Code Case N-640 and N-588 (Dresden Unit 3 only). Exelon will manage the P-T curves using approved fluence calculations when there are changes in power of core design in conjunction with surveillance capsule results from the BWRVIP Integrated Surveillance Program (BWRVIP reports 78 and 86). See the Reactor Vessel Surveillance Program (B.1.22) for additional discussion.

4.2.6 Reactor Vessel Circumferential Weld Examination Relief

Applicability

This section applies to Dresden only.

Summary Description

Relief from reactor vessel circumferential weld examination requirements under Generic Letter 98-05 is based on probabilistic assessments that predict an acceptable probability of failure per reactor operating year. The analysis is based on reactor vessel metallurgical conditions as well as flaw indication sizes and frequencies of occurrence that are expected at the end of a licensed operating period.

Dresden has received this relief for the remaining 40 year licensed operating period. Quad Cities never submitted a relief request for the remaining 40 year licensed operating period. (Reference 4.27) As such, the supporting evaluations only apply to Dresden. The circumferential weld examination relief analysis meets the requirements of 10CFR54.3(a) and is a TLAA.

Analysis

Dresden received NRC approval for a technical alternative which eliminated the reactor vessel circumferential shell weld inspections for the current license term. The basis for this relief request was an analysis that satisfied the limiting conditional failure probability for the circumferential welds at the expiration of the current license, based on BWRVIP-05 and the extent of neutron embrittlement. The anticipated changes in metallurgical conditions expected over the extended licensed operating period require an additional analysis for 54 EFPY and approval by the NRC to extend this relief request.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

The USNRC evaluation of BWRVIP-05 utilized the FAVOR code to perform a probabilistic fracture mechanics (PFM) analysis to estimate the RPV shell weld failure probabilities. Three key assumptions of the PFM analysis are: 1) the neutron fluence that was the estimated end–of-life mean fluence, 2) the chemistry values are mean values based on vessel types, and 3) the potential for beyond-design-basis events is considered. Table 4.2.6-1 provides a comparison of the Dresden reactor vessel limiting circumferential weld parameters to those used in the NRC analysis for the first two key assumptions. Data provided in Table 4.2.6-1 was supplied from Tables 2.6-4 and 2.6-5 of the Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-05 Report.

Although the chemistry composition and chemistry factor for unit 3 are higher than the limits of the NRC Analysis; the 54 EFPY fluence results are considerably lower for both Dresden Units 2 and 3. As a result, the shifts in reference temperature for both units are lower than the 54 EFPY shift from the NRC analysis. In addition, the unirradiated reference temperatures for both Dresden units are lower. The combination of unirradiated reference temperature ($RT_{NDT(U)}$) and shift (ΔRT_{NDT} w/o margin) yields adjusted reference temperatures that are considerably lower than the NRC mean analysis values. Therefore, the RPV shell weld embrittlement due to fluence has a negligible effect on the probabilities of RPV shell weld failure. The Mean RT_{NDT} values for both units at 54 EFPY are bounded by the 64 EFPY Mean RT_{NDT} provided the NRC. Although a conditional failure probability has not been calculated, the fact that the Dresden 54 EFPY values are less than the 64 EFPY value provided by the NRC leads to the conclusion that the Dresden RPV conditional failure probability is bounded by the NRC analysis.

The procedures and training used to limit cold over-pressure events will be the same as those approved by the NRC when Dresden requested the BWRVIP-05 technical alternative be used for the current term (Reference 4.14). An extension of this relief for Dresden for the 60-year period will be submitted to the NRC for approval prior to the period of extended operation.

Table 4.2.6-1 Effects for Irradiation on RPV Circumferential Weld Properties Dresden Units 2 & 3

Group	B&W 64 EFPY	Dresden Unit 2 54 EFPY	Dresden Unit 3 54 EFPY
Cu%	0.31	0.23	0.34
Ni%	0.59	0.59	0.68
CF	196.7	168	221
Fluence at clad/weld interface (10^{19} n/cm^2)	0.19	0.042	0.041
ΔRT_{NDT} w/o margin (°F)	109.4	36	47
$RT_{NDT(U)}$ (°F)	20	10	-5
Mean RT_{NDT} (°F)	129.4	46	42
P(FIE) NRC	4.83×10^{-4}	---	---
P(FIE) BWRVIP	---	---	---

4.2.7 Reactor Vessel Axial Weld Failure Probability

Applicability

This section applies to Dresden only.

Summary Description

The Boiling Water Reactor Owner's Group Vessel and Internals Program recommendations for inspection of reactor vessel shell welds (BWRVIP-05, Reference 4.14) contain generic analyses supporting an NRC SER (Reference 4.15) conclusion that the generic-plant axial weld failure rate is no more than 5×10^{-6} per reactor year. BWRVIP-05 showed that this axial weld failure rate of 5×10^{-6} per reactor year is orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds as described in Section 4.2.6

Dresden received relief from the circumferential weld inspections for the remaining 40 year licensed operating period. Quad Cities never submitted a relief request for the remaining 40-year license operating period. As such, the supporting evaluations only apply to Dresden.

Analysis

As stated in Section 4.2.6, Dresden Station received NRC approval for a technical alternative which eliminated the reactor vessel circumferential shell weld inspections for the current license term. The basis for this relief request was an analysis that satisfied the limiting conditional failure probability for the circumferential welds at the expiration of the current license, based on BWRVIP-05 and the extent of neutron embrittlement. The NRC SER associated with BWRVIP-05 (Reference 4.15) concluded that the reactor vessel failure frequency due to failure of the limiting axial welds in the BWR fleet at the end of 40 years of operation is less than 5×10^{-6} per reactor year. This failure frequency is dependent upon given assumptions of flaw density, distribution, and location. The failure frequency also assumes that "essentially 100%" of the reactor vessel axial welds will be inspected.

Due to various obstructions within the reactor vessel, Dresden has not been able to meet the "essentially 100%" inspection requirement. As such, an analysis was performed to assess the effect on the probability of fracture due to the actual inspection performed on the vessel axial welds and to determine if the coverage was sufficient in the inspection of regions contributing to the majority of the risk. The analysis included an estimate and comparison of the probability of failure for the cases of "essentially 100%" inspection and the limited scope inspections on the Dresden Unit 2 and Unit 3 vessel axial welds. The analysis concluded that the conditional probabilities of failure due to a low temperature over pressurization event are very small, 7.4×10^{-10} and 7.88×10^{-12} on a per year basis for Dresden Unit 2 and Unit 3, respectively. The conditional probability of failure with the "essentially 100%" inspections were an order of magnitude lower than that for the actual inspection coverage. However, the analysis only applies to the current 40-year operating period. The anticipated changes in metallurgical conditions expected over the extended licensed operating period require an additional analysis for 54 EFPY and approval by the NRC to extend the reactor vessel circumferential weld inspection relief request.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

Table 4.2.7-1 compares the limiting axial weld 54 EFPY properties for Dresden Units 2 and 3 against the values taken from Table 2.6-5 found in the NRC SER for BWRVIP-05 and associated supplement to the SER (Reference 4.16). The SER supplement required the limiting axial weld to be compared with data found in Table 3 of the document. For Dresden, the comparison was made to the Clinton plant information. The supplemental SER stated that the axial welds for the Clinton plant are the limiting welds for the BWR fleet and vessel failure probability calculations determined for Clinton should bound those for the BWR fleet.

The limiting axial welds at Dresden are all electroslag welds with similar chemistry. The Dresden limiting weld chemistry, chemistry factor (CF), and 54 EFPY mean RT_{NDT} values are within the limits of the values assumed in the analysis performed by the NRC staff in the March 7, 2000 BWRVIP-05 SER supplement and the 64 EFPY limits and values obtained from Table 2.6.5 of the SER.

As stated above, the probability of a failure event P_{FE} calculated by the NRC BWRVIP-05 SER and its supplements depends in part on an assumption that 90 per cent of axial welds can be inspected. Less than 90 per cent of axial welds can be examined at Dresden. As such, an analysis was performed for 54 EFPY to assess the effect on the probability of fracture due to the actual inspection performed on the vessel axial welds and to determine if the coverage was sufficient in the inspection of regions contributing to the majority of the risk. The analysis included the estimate and comparison of the probability of failure for the cases of “essentially 100%“ inspection and the limited scope inspections on the Dresden Unit 2 and Unit 3 vessel axial welds. The analysis concluded that the conditional probabilities of failure due to a low temperature over pressurization event are very small, 3.89×10^{-8} and 5.07×10^{-8} on a per year basis for Dresden Unit 2 and Unit 3, respectively. The evaluation shows that the calculated unit-specific axial weld conditional failure probabilities at 54 EFPY for Dresden are less than the failure probabilities calculated by the NRC staff in the SER at 64 EFPY and the limiting Clinton values found in Table 3 of the SER supplement. The probability of failure of an axial weld at Dresden will therefore provide adequate margin above the probability of failure of a circumferential weld, in support of relief from inspection of circumferential welds, for the extended licensed operating period.

Table 4.2.7-1 Effects for Irradiation on RPV Axial Weld Properties

Dresden Units 2 & 3

Value	B&W 64 EFPY	SER Supplement (Clinton)	DRE 2 54 EFPY	DRE 3 54 EFPY
Cu%	0.25	0.10	0.24	0.24
Ni%	0.35	1.08	0.37	0.37
CF	142.5	----	141	141
Fluence x 10^{19} n/cm^2	0.35	0.69	0.057	0.057
ΔRT_{NDT} ºF	88.9	121	36	36
$RT_{NDT(U)}$ ºF	10	-30	23	23
Mean RT_{NDT} ºF	98.9	91	59	59
P(FIE) NRC	1.87×10^{-1}	2.73×10^{-3}	3.89×10^{-8}	5.07×10^{-8}
$P_{(FIE)}$ BWRVIP	----	1.52×10^{-3}	----	----

4.3 METAL FATIGUE OF THE REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT PRESSURE BOUNDARY PIPING AND COMPONENTS

A cyclically loaded metal component may fail because of fatigue even though the cyclic stresses are considerably less than the static design limit. Some design codes therefore contain explicit metal fatigue calculations or design limits, such as the ASME Boiler and Pressure Vessel Code and the ANSI piping codes. Cyclic or fatigue design of other components may not be to these codes, but may use similar methods. These analyses, calculations and designs to cycle count limits or to fatigue usage factor limits may be TLAAs.

Fatigue analyses are presented in the following groupings:

- Reactor Vessel Fatigue Analyses
- Reactor Vessel Internals Fatigue Analysis
    - Low Cycle Fatigue of Core Shroud and Repair Hardware
    - High Cycle Fatigue of Jet Pump Riser Braces
- Reactor Coolant Pressure Boundary Piping and Component Fatigue Analysis
    - Piping and Components Designed in Accordance With ASME Section III Class 1
    - Piping and Components Designed in Accordance with USAS B31.1, ASME Section III Class 2 and 3, or ASME Section III Class B and C
    - Isolation Condenser Fatigue Analysis
- Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)

NUREG-1801 identifies numerous fatigue related aging effects that require evaluation as possible TLAAs in accordance with 10 CFR 54.21(c). Each of these is summarized in NUREG-1800 and presented in Section 3 of this LRA and referenced to the appropriate TLAA section.

4.3.1 Reactor Vessel Fatigue Analyses

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

Reactor vessel fatigue analyses of the vessel support skirt, shell, upper and lower heads, closure flanges, nozzles and penetrations, nozzle safe ends, and closure studs depend on assumed numbers and severity of normal and upset-event pressure and thermal operating cycles to predict end-of-life fatigue usage factors.

These assumed cycle counts and fatigue usage factors are based on 40 years of operation. Calculation of fatigue usage factors is part of the current licensing basis and is used to support safety determinations. The reactor vessel fatigue analyses are TLAAs.

Analysis

The original reactor pressure vessel stress report included a fatigue analysis for the reactor vessel components based on a set of design basis duty cycles. These duty cycles are listed in Table 3.9-1 of the Dresden and Quad Cities UFSARs. The original 40-year analyses demonstrated that the cumulative usage factors (CUF) for the critical components would remain below the ASME Code Section III allowable value of 1.0.

A reanalysis was performed for reactor vessel cumulative fatigue usage factors as a part of Extended Power Uprate (EPU) implementation at all four Dresden and Quad Cities units. (References 4.10 and 4.11) A subset of the bounding reactor vessel components was evaluated as a part of this re-analysis. The resulting fatigue cumulative usage factors (CUFs) for these limiting components supersede the values determined in the original reactor vessel analyses. The current bounding-case analysis (worst CUFs for all four reactor vessels) lists the following values for the 40-year CUFs for limiting components.

Shroud Support	0.820
Support Skirt	0.862
Feedwater Nozzle (Safe End)	0.748
Closure Studs	0.750

The original code analysis of the reactor vessel included fatigue analysis of the feedwater and control rod drive hydraulic system return line nozzles. After several years of operation, it was discovered that both the control rod drive hydraulic system return line nozzles and the feedwater nozzles were subject to cracking caused by a number of factors including rapid thermal cycling. Consequently, the control rod drive hydraulic system return line nozzles were capped and removed from service. As such, they are no longer subject to rapid thermal aging. A reanalysis was later performed on the feedwater nozzles along with modifications to reduce or eliminate the causes. This revised analysis included the appropriate effects from rapid thermal cycling that were attributed to the original cause of cracking in the feedwater nozzles. Both Dresden and Quad Cities also follow the improved BWR Owners Group inspection and management methods.

Disposition: Aging Management, 10 CFR 54.21(c)(1)(iii)

The fatigue cumulative usage factors (CUFs) of the reactor vessel, including the support skirt, shell, upper and lower heads, closure assembly, nozzles and penetrations, and nozzle safe ends will be managed by the Metal Fatigue of Reactor Coolant Pressure Boundary (B.1.34) aging management program. This program will monitor CUFs through either stress-based fatigue (SBF) monitoring or cycle-based fatigue monitoring (CBF).

Stressed-based fatigue monitoring consists of computing a “real time” stress history for a given component from actual temperature, pressure, and flow histories via a finite element based Green’s Function approach. CUF is then computed from the computed stress history using appropriate cycle counting techniques and appropriate ASME Code, Section III fatigue analysis methodology. SBF monitoring is intended to duplicate the methodology used in the governing ASME Code Section III stress report for the component in question, but uses actual transient severity in place of design basis transient severity.

Cycle-based fatigue monitoring consists of a two step process: (a) automated cycle counting, and (b) CUF computation based on the counted cycles. The automated cycle counting counts each transient that is defined in the plant licensing basis based upon the mechanistic process or sequence of events experienced by the plant (as determined from monitored plant instruments). The approach is conservative because it assumes each actual transient has a severity equal to that assumed in the design basis. The unique severity of any transient identified by the aging management program software is captured for each monitored component, for ready comparison to design basis transient severity. Transients defined in the Dresden and Quad Cities UFSAR are identified and implemented into the aging management program software. CUF computation calculates fatigue directly from counted transients and parameters for the monitored components. CUF is computed via a design-basis fatigue calculation where the numbers of cycles are substituted for assumed design basis number of cycles. The CUF calculations are conservative in that design basis transient severity is assumed.

All governing reactor vessel and Class 1 fatigue analyses have been reviewed to establish a comprehensive and bounding set of reactor vessel locations for inclusion in the Metal Fatigue of Reactor Coolant Pressure Boundary (B.1.34) aging management program. All locations with 40-year CUFs expected to exceed 0.4 will be included in the program. Those locations are listed in Table 4.3.1-1 below. The associated CUFs are all based upon the original 40 year analysis. In the most limiting cases (CUF > .40), the 40-year post EPU analysis is listed.

All necessary plant transient events will be tracked to ensure that the CUF remains less than 1.0 for all monitored components. In the event fatigue cumulative usage factor is predicted to exceed 1.0 for any component prior to 60 years of operation, appropriate corrective action will be taken in accordance with the Exelon Corrective Action Program (B.2.1). The required implementing actions will be completed prior to the period of extended operation. Dresden and Quad Cities have programs in place to track operating thermal and pressure cycles and to assess their effect on vessel fatigue. The requirements from these procedures will be incorporated into the Metal Fatigue of Reactor Coolant Pressure Boundary (B.1.34) aging management program.

Table 4.3.1-1 Fatigue Monitoring Locations for Reactor Pressure Vessel Components

Component	Computed Fatigue Usage Factor (Pre-EPU)	Computed Fatigue Usage Factor (Post EPU)	Monitoring Technique
Recirculation Outlet Nozzle	0.270	Note 4	CBF (NUREG/CR-6260 component)
Recirculation Inlet Nozzle	0.301	Note 4	CBF (NUREG/CR-6260 component)
Feedwater Nozzle	0.538	.748	SBF (NUREG/CR-6260 component)
Core Spray Nozzle	0.079	Note 4	CBF (NUREG/CR-6260 component)
Support Skirt	0.940	0.862	SBF
Shroud Support	0.630	0.820	CBF
Closure Stud Bolts	0.79	0.75	CBF
Vessel Shell	0.141	Note 4	CBF (NUREG/CR-6260 component)

Notes:
1. CBF = Cycle-Based Fatigue and SBF = Stress-Based Fatigue
2. EPU = Extended Power Uprate
3. The components listed as a “NUREG/CR-6260 component” will be monitored for GSI –190. See Section 4.3.4
4. Only locations with 40-year CUF expected to exceed 0.400 were computed for the EPU Project.

4.3.2 Fatigue Analysis of Reactor Vessel Internals

A review of the current licensing basis found no fatigue analysis on the reactor vessel internals with the following exceptions listed below.

This section describes TLAAs arising within:

- Low-Cycle Thermal Fatigue Analysis of the Core Shroud and Repair Hardware

- High-Cycle Flow-Induced Vibration Fatigue Analysis of Jet Pump Riser Braces

4.3.2.1 Low-Cycle Thermal Fatigue Analysis of the Core Shroud and Repair Hardware

Applicability

This section applies to Quad Cities only.

Summary Description

Only one analysis of low-cycle fatigue of reactor vessel internals was identified which includes an evaluation of the core shroud and core shroud repair hardware at Quad Cities. The core shroud repair SER for Quad Cities states that the limiting upset loading condition is the cold feedwater transient and that this event is the only case that produces any fatigue in need of consideration. As such, the analysis is a TLAA.

Analysis

A review of licensing basis documents found no evidence of analyses of pressure or thermal cycle fatigue for the core plate, top guide, jet pump assemblies (other than high-cycle fatigue in the riser braces – see Section 4.3.2.2), fuel supports, in-core instrumentation tubes, or control rod drive assemblies. Low-cycle mechanical fatigue was mentioned only for the tie rod stabilizers in the core shroud repair evaluations.

The USNRC Safety Evaluation of the core shroud repair for Quad Cities (Reference 4.1) states:

> "The limiting upset loading condition event that ComEd evaluated is the cold feedwater transient that is classified as an upset loading condition. During this transient, due to the injection of cold feedwater into the shroud annulus, a maximum difference of 133° F between the hot core shroud and the cooler tie rod stabilizer assembly components could exist. This would cause an increase in the tensile load on the stabilizers and an increase in the compressive load on the core shroud. This condition was evaluated and it was determined that the stresses in the stabilizer and in the core shroud for this condition would be both less than the ASME Code upset allowable stress and less than the material yield stress, thus preventing permanent deformation, which is acceptable. ComEd also determined that this event is the only case that produces any fatigue requiring consideration. For this event, the maximum calculated fatigue usage was found to be insignificant compared to the allowable usage and is, therefore, acceptable."

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The current predicted cumulative usage factors for the Quad Cities core shrouds and core shroud repair hardware were found to be not significant, less than 0.11. A 20-year increase in service life will not raise the usage factors significantly. Therefore, the design of core shroud repair hardware for fatigue effects is valid for the extended licensed operating period.

4.3.2.2 High-Cycle Flow-Induced Vibration Fatigue Analysis of Jet Pump Riser Braces

Applicability

This section applies to Dresden Unit 2 only.

Summary Description

Flow rate through the reactor vessel leads to vibration of internal components. Certain conditions, such as increased core flow or single recirculation loop operation, can significantly increase vibration levels. To address this concern, the Dresden reactor vessel internals were instrumented and tested for vibration levels during initial plant startup. Criteria were established such that vibration levels were assured to remain below material endurance limits over the life of the plant. Various operating conditions were evaluated, including those associated with increased core flow and transient unbalanced flow conditions. Of the conditions tested, evaluation of transient unbalanced flow conditions was found to be limiting at Dresden Unit 2 for the jet pump riser braces. Although reactor internals are not code pressure boundary components, the evaluation of the Dresden Unit 2 jet pump riser braces used methods and fatigue curves similar to those of ASME Section III Class 1 fatigue analyses.

Analysis

Acceptable vibration levels for the Dresden reactor internals were based initially on the amplitude that could be permitted on a continuous basis for a 40-year plant life. Despite the fact that a 40-year basis was described in the underlying evaluations, this criterion is not time-dependent.

The original design addressed high-cycle fatigue effects in the internals and determined that these effects are acceptable over the life of the plant. The UFSAR description of the original design mentions the "reactor core support structure" and excitation by the recirculation and jet pumps, and that the jet pump lines were not a significant safety question. However, except for fatigue in the Dresden Unit 2 jet pump riser braces, the original evaluation of specific components such as the core plate, top guide, jet pump assemblies, fuel supports, or control rod drive assemblies used displacement criteria determined by a fatigue endurance that is not time-limited.

Extended power uprate (EPU) analysis found that the Dresden and Quad Cities reactor internal components with the exception of the Dresden Unit 2 jet pump riser braces, can operate at EPU conditions for a 60-year plant life without (1) exceeding the original design vibration criteria or fatigue usage during balance flow operation; or (2) developing resonance problems due to recirculation pump vane passing frequency (VPF) at EPU conditions.

The EPU project evaluated possible effects of power uprate, including increased core flow, increased pump vane-passing frequency (VPF), and other factors, and found that, with some possible exceptions (including the Dresden Unit 2 riser braces), the stress ranges of internals would remain within the original 10,000 psi endurance limit. Among the exceptions, only the evaluation of the Dresden Unit 2 jet pump riser braces included fatigue.

The final EPU report on flow induced vibration of reactor internals found that the Dresden Unit 2 jet pump riser braces might be damaged by recirculation pump vane passing frequency (VPF) vibration if operation is permitted in the maximum extended load line limit analysis (MELLA) region. Dresden Unit 2 has only one brace per riser with leaves that are thinner than the other three units. The EPU report therefore stated that significant operation of Dresden Unit 2 in a region at which the VPF is resonant with the brace leaves might produce fatigue cracks and failures in the riser braces. This might be a problem for continuous operation in the MELLA region. The Quad Cities riser brace design is identical to the Dresden Unit 3 brace design which shows no resonance effects and presents no concern.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

The Dresden Unit 2 riser braces will be repaired or replaced prior to the period of extended operation and will be qualified for the extended licensed operating period.

4.3.3 Reactor Coolant Pressure Boundary Piping and Component Fatigue Analysis

This section describes fatigue-related TLAAs arising within design analyses of:

- ASME Section III Class 1 Reactor Coolant Boundary Piping and Component Fatigue Analysis
- Reactor Coolant Pressure Boundary Piping and Components Designed to USAS B31.1, ASME Section III Class 2 and 3, or ASME Section VIII Class B and C
- Fatigue Analysis of the Isolation Condenser

4.3.3.1 ASME Section III Class 1 Reactor Coolant Pressure Boundary Piping and Component Fatigue Analysis

Applicability

This section applies to Dresden.

Summary Description

The Dresden Unit 3 recirculation piping was replaced under the IGSCC mitigation program. The pipe replacement included reactor pressure coolant boundary piping in the shutdown cooling system, low pressure coolant injection system, isolation condenser system, and reactor water cleanup system. As a part of this replacement, fatigue evaluation of the replaced portions of piping was performed. The analyses demonstrate that the 40-year cumulative usage factors (CUF) for the critical components of the replaced piping are below the ASME Code Section III allowable value of 1.0. The current analyses of record are TLAAs.

Analysis

Other than special cases under the Mark I containment "New Loads" program (see Section 4.6), the only reactor coolant pressure boundary piping that has been analyzed for fatigue at either Dresden or Quad Cities is the Dresden Unit 3 recirculation system piping. This included portions of the connected shutdown cooling, low pressure coolant injection, isolation condenser, and reactor water cleanup piping that form the reactor coolant pressure boundary.

The validation of all remaining Dresden and Quad Cities USAS B31.1 piping design is discussed in Section 4.3.3.2. All "Class I" piping at Dresden and Quad Cities was originally designed to USAS B31.1, 1967 Edition. The Dresden Unit 3 recirculation and other connected piping were replaced under the GL 88-01 IGSCC correction program. The replacement piping was analyzed to ASME Section III Class 1 rules. For an acceptable fatigue design, the ASME Code limits the CUF to less than 1.0. The ASME Section III Class 1 code analyses for the Dresden Unit 3 recirculation line and attached large bore piping replacement inside the drywell show calculated cumulative usage factors (CUFs) that exceed 0.400. These are shown in Table 4.3.3.1-1 below.

Disposition: Aging Management, 10CFR54.21(c)(1)(iii)

The Dresden Unit 3 reactor coolant pressure boundary piping will be managed by the Metal Fatigue of Reactor Coolant Pressure Boundary (B.1.34) aging management program. This program will monitor CUFs through cycle-based fatigue monitoring (CBF).

Cycle-based fatigue monitoring consists of a two step process: (a) automated cycle counting, and (b) CUF computation based on the counted cycles. The automated cycle counting counts each transient that is defined in the plant licensing basis based upon the mechanistic process or sequence of events experienced by the plant (as determined from monitored plant instruments). The approach is conservative because it assumes each actual transient has a severity equal to that assumed in the design basis. The unique severity of any transient identified by the aging management program software is captured for each monitored component, for ready comparison to design basis transient severity. Transients defined in the Dresden and Quad Cities UFSAR are identified and implemented into the aging management program software. CUF computation calculates fatigue directly from counted transients and parameters for the monitored components. CUF is computed via a design-basis fatigue calculation where the numbers of cycles are substituted for assumed design basis number of cycles. The CUF calculations are conservative in that design basis transient severity is assumed.

All governing Dresden Unit 3 reactor coolant pressure boundary piping fatigue analyses have been reviewed to establish a comprehensive and bounding set of primary coolant boundary piping locations for inclusion in the Metal Fatigue of Reactor Coolant Pressure Boundary (B.1.34) aging management program. All locations with 40-year CUFs expected to exceed 0.4 will be included in the program. Those locations are listed in Table 4.3.3.1-1 below. The associated CUFs are all based upon the original 40-year analysis.

All applicable plant transient events will be tracked to ensure that the CUF remains less than 1.0 for all monitored components. In the event fatigue cumulative usage is predicted to exceed 1.0 for any component prior to 60 years of operation, appropriate corrective action will be taken in accordance with the Exelon Corrective Action Program (B.2.1). The required implementing actions will be completed prior to the period of extended operation. The requirements from these procedures will be incorporated into the Metal Fatigue of Reactor Coolant Pressure Boundary (B.1.34) aging management program.

Table 4.3.3.1-1 Fatigue Monitoring Locations for Dresden Unit 3 Reactor Recirculation Piping

Component	Calculated Fatigue Usage Factor	Fatigue Monitoring Technique
Recirculation Suction Line	0.475	CBF
Recirculation Discharge Line (Loop B)	0.411	CBF
Shutdown Cooling Line	0.499	CBF
Loop A; Shutdown Cooling (SDC)-Isolation Condenser (ISCO) Tee	0.696	CBF
Shutdown Cooling Guard Pipe Lug	0.711	CBF
Shutdown Cooling Penetration Flued Head	0.600	CBF
2” Reactor Water Cleanup Bypass Line	0.473	CBF

4.3.3.2 Reactor Coolant Pressure Boundary Piping and Components Designed to USAS B31.1, ASME Section III Class 2 and 3, or ASME Section VIII Class B and C

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

The reactor coolant pressure boundary (RCPB) piping for all of the Quad Cities and Dresden units was designed to USAS B31.1 except for some piping in Dresden Unit 3 as discussed in Section 4.3.3.1. The RCPB and non-RCPB piping in the scope of license renewal that is designed to USAS B31.1 or ASME Section III Class 2 and 3 requires the application of a stress range reduction factor to the allowable stress range for secondary (expansion and displacement) stresses to account for thermal cyclic conditions.

The codes and standards to which the Dresden and Quad Cities units were designed and constructed did not invoke fatigue analyses for piping or component supports, nor for their welds, bolted connections, or anchors. The only exceptions are some ASME Class MC containment piping support and penetration analyses for "new loads," discussed in Section 4.6, and the replaced RCPB piping in Dresden Unit 3, discussed in Section 4.3.3.1.

Analysis

USAS B31.1 (1967) is the original design code for RCPB and non-RCPB piping at the Quad Cities and Dresden units. At Dresden, piping from the reactor vessel to the first isolation or stop valve were analyzed based on ASME Section I (1965, Winter 1966 Addenda), and Nuclear Code Cases N-1 through N-11, (Reference 4.3 and 4.4). None of these codes, addenda, or cases invoke fatigue analyses. However, ASME Section III Class 2 and 3, and USAS B31.1 piping require the application of a stress range reduction factor to the allowable stress range for expansion stresses to account for cyclic thermal conditions. The allowable secondary stress range is 1.0 S_A for 7,000 equivalent full-temperature thermal cycles or less and is reduced in steps to 0.5 S_A for greater than 100,000 cycles.

With the exception of containment vent and process bellows, no components in the scope of license renewal designed to ASME Section III or Section VIII require design for cyclic thermal loading. Therefore, cycle counting is not a consideration for the following components.

<u>Recirculation Pumps</u>: The reactor recirculation pumps are designed per ASME Section III, Class C (1965) (Reference 4.3 and 4.4). Evaluation for thermal cycles is not required and none were performed (Reference 4.3 and 4.4).

<u>Quad Cities RHR System</u>: The RHR heat exchangers at Quad Cities were designed to ASME Section III (1966), Class C requirements on the shell side and Section VIII on the tube side, neither of which require design for cyclic thermal loading. The review found no other RHR components with indications of design for fatigue. The RHR system

design therefore includes no TLAAs other than the piping design for USAS B31.1 stress range reduction factors.

Other than Dresden Unit 3 piping described in 4.3.3.1 above, all of the Dresden and Quad Cities RCPB and connected piping, including the portions of the main steam and SRV discharge lines inside the drywell, were designed using USAS B31.1 Power Piping Code stress range reduction factors for a finite number of thermal cycles, but have no fatigue analyses. USAS B31.1 does not require explicit fatigue analysis. Therefore CUF values were not originally calculated for the remainder of the Dresden and Quad Cities "Class I" piping.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The assumed thermal cycle count for the analyses used in the codes associated with piping and components can conservatively be approximated by the thermal cycles used in the reactor vessel fatigue analysis. These thermal cycles are listed in UFSAR Table 3.9-1. The total count of all these listed thermal cycles is less than 2200 for a 40-year plant. For the 60-year extended operating period the number of thermal cycles for piping analyses would be proportionally increased to less than 3300, a fraction of the 7000-cycle threshold. The existing piping analyses within the scope of license renewal containing assumed thermal cycle counts are valid for the period of extended operation.

4.3.3.3 Fatigue Analysis of the Isolation Condenser

Applicability

This section applies to Dresden only.

Summary Description

The Dresden isolation condensers provide core cooling when the reactor vessel becomes isolated from the turbine and the main condenser. Fatigue evaluation of the isolation condensers was performed as a part of original component design. The analyses demonstrate that the 40-year cumulative usage factors (CUF) for the critical components of the isolation condenser are below the ASME Code Section III allowable value of 1.0. The fatigue analysis of the Dresden isolation condensers is a TLAA because it is part of the current licensing basis, is used to support a safety determination, and is based on the cycles predicted for a 40-year plant life.

Analysis

The isolation condenser and the supporting system piping and components were specified for 250 shutdown depressurization cycles, with 40 to 240 depressurizations expected in 40 years, and for 250 thermal shock events in 40 years. The design analysis for thermal shock assumed 280 events, a margin of 30 cycles or 12%. The limiting thermal shock event is caused by an isolation event that raises the fluid on the secondary side of the isolation condenser to boiling at atmospheric pressure, or 212°F, followed by injection of makeup water as cold as 70°F into the secondary side. The code analysis shows that each isolation condenser operating cycle includes a thermal shock event.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The thermal shock is the most limiting of the design transients for the isolation condenser which is bounded by 280 isolation condenser operations. Since the isolation condenser is infrequently operated, records of the number of actual isolation condenser operations were not kept during the first 20 years of operation. Record keeping of isolation condenser operations began in 1996. Since that time, there have been 3 isolation condenser operations on Unit 2 and 4 operations on Unit 3. This would indicate an operating rate of 0.58 isolation condenser operations per year (4 operations ÷ 7 years).

There have been a total of 164 reactor scrams on Unit 2 from initial operation (1970) to August 2002. Unit 2 has the bounding number of scrams between the two units. Conservatively assuming that there was an isolation condenser operation every time that a unit scrammed from 1970 to 2002 and assuming an actual operating rate of 0.58 operations per year through the extended period of operation (60 years), the total number of expected isolation condenser operations would be 181 [164 + (0.58 x 28)]. This projected cycle count falls below the 280 isolation condenser operating design limit. Therefore, the design analysis remains valid for the extended period of operation.

4.3.4 Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

ASME Section III uses stress versus allowable cycle curves (S-N curves) based on tests in air to determine a fatigue usage factor. Generic Safety Issue (GSI) 190 addresses the effects of reactor coolant environment on fatigue life of components and piping. The environment of a stressed component can affect fatigue life. Although GSI 190 is resolved, NUREG-1800 states "The applicant's consideration of the effects of coolant environment on component fatigue life for license renewal is an area of review" (Section 4.3.1.2). The GSI-190 review requirements are therefore imposed by NUREG-1800 and do not depend on the individual plant licensing basis.

Analysis

The NRC staff assessed the impact of reactor water environment on fatigue life at high fatigue usage locations and presented the results in NUREG/CR-6260, "Application of NUREG/CR-5999, 'Interim Fatigue Curves to Selected Nuclear Power Plant Components'," March 1995. Formulas currently acceptable to the staff for calculating the environmental correction factors for carbon and low-alloy steels are contained in NUREG/CR-6583, "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels," and those for austenitic stainless steels are contained in NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design of Austenic Stainless Steels."

In order to comply with the requirements of NUREG/CR-5999, Exelon would be required to perform plant specific calculations at Dresden and Quad Cities for the locations identified in NUREG/CR-6260 for the older vintage BWR plants. For each of these locations, detailed environmental fatigue calculations would have to be performed using the appropriate F_{en} relationship from NUREG/CR-6583 (for carbon/low alloy steels) and NUREG/CR-5704 (for stainless steels), as appropriate for the material for each location. The detailed calculations must include the calculation of an appropriate F_{en} factor for each individual load pair in the governing fatigue calculation so that an overall multiplier on CUF for environmental effects can be determined for each location.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

Exelon will perform plant-specific calculations for Dresden and Quad Cities for the locations identified in NUREG/CR 6260 for older-vintage BWR. These locations are:

- Reactor Vessel (Lower Head to Shell Transition)
- Feedwater Nozzle
- Recirculation System (RHR Return Line Tee)
- Core Spray System (Nozzle and Safe End)

- Residual Heat Removal Line (Tapered Transition)
- Limiting Class 1 Location in a Feedwater Line

The list above does not specifically include the feedwater line (RCIC tee) location identified in NUREG/CR-6260 for the older-vintage GE plant. Dresden does not have a RCIC system. Quad Cities does have a RCIC system; however, the RCIC tee location is located outside the containment in the Class 2 portion of the feedwater line. Therefore, for both plants, the limiting Class 1 feedwater piping location will be evaluated for environmental fatigue effects.

For each location, detailed environmental fatigue calculations will be performed using the appropriate F_{en} relationships from NUREG/CR 6583 for carbon and low-alloy steels and from NUREG/CR 5704 for stainless steels, as appropriate for the material. The calculations will determine an appropriate F_{en} for each individual load pair in the governing fatigue calculation, so that an overall multiplier on cumulative usage factor (CUF) for environmental effects can be determined for each location. These calculations will be completed prior to the period of extended operation, and appropriate corrective action will be taken if the resulting projected end-of life CUF values exceed 1.0.

Exelon reserves the right to modify this position in the future based on the results of industry activities currently underway, or based on other results of improvements in methodology, subject to NRC approval prior to changes in this position.

4.4 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT (EQ)

10 CFR 50.49, Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants (Reference 4.22), requires that certain electrical and instrument and control (I&C) equipment located in harsh environments be qualified to perform their safety related functions in those harsh environments after the effects of in-service aging.

NUREG-1801 identifies numerous environmentally related aging effects that require evaluation as possible TLAAs in accordance with 10 CFR 54.21(c). Each of these is summarized in NUREG-1800 and presented in Section 3 of this LRA and referenced to the appropriate TLAA section.

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

10 CFR 50.49(e)(5) contains provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. 10 CFR 50.49(e)(5) also requires component replacement or maintenance prior to the end of designated life, unless additional life is established through ongoing qualification. 10 CFR 50.49(k) and (l) permit different qualification criteria to apply based on plant vintage. Supplemental EQ regulatory guidance for compliance with these different qualification criteria is provided in the Regulatory Guide 1.89, Revision 1, "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants," (Reference 4.25), the Division of Operating Reactors (DOR) Guidelines (Reference. 4.23) and NUREG-0588 (Reference. 4.24).

All operating plants must meet the requirements of 10 CFR 50.49 for certain electrical and I&C components important to safety. 10 CFR 50.49 defines the scope of components to be included, requires the preparation and maintenance of a qualification binder that includes component performance specifications, electrical characteristics, and environmental conditions. 10 CFR 50.49(f) establishes four methods of demonstrating qualification for aging and accident conditions. Compliance with 10 CFR 50.49 provides evidence that the component will perform its intended functions during and after a design basis accident after experiencing the effects of in-service aging.

The Dresden and Quad Cities EQ programs were established to demonstrate that certain electrical components located in harsh plant environment (that is, those areas of the plant that could be subject to the harsh environmental effects of loss of coolant accident [LOCA], high energy line breaks [HELB] or post-LOCA radiation) are qualified to perform their safety function operation in those harsh environments after the effects of in service aging. The Dresden and Quad Cities station EQ program complies with the requirements of 10 CFR 50.49, or DOR Guidelines for that equipment presently qualified to DOR guidelines. For Dresden and Quad Cities, the EQ-related equipment is identified in controlled equipment data bases and equipment qualification binders.

Dresden and Quad Cities EQ program manages component thermal, radiation and cyclic aging as applicable, through the aging evaluations based on 10 CFR 50.49, and DOR guidelines for those components presently qualified to DOR guidelines.

GSI 168, Environmental Qualification of Electrical Components

NRC guidance for addressing GSI-168 for license renewal is contained in the June 2, 1998, NRC letter to NEI. (Reference 4.26) In this letter, the NRC states, "With respect to addressing GSI-168 for license renewal, until completion of an ongoing research program and staff evaluations, the potential issues associated with GSI-168 and their scope have not been defined to the point that a license renewal applicant can reasonably be expected to address them at this time. Therefore, an acceptable approach described in the Statements of Consideration is to provide technical rationale demonstrating that the current licensing basis for environmental qualification pursuant to 10 CFR 50.49 will be maintained in the period of extended operation. Although the Statements of Consideration also indicates that an applicant should provide a brief description of one or more reasonable options that would be available to adequately manage the effects of aging, the staff does not expect an applicant to provide the options at this time." Consistent with the above NRC guidance, no additional information is required to address GSI-168 in a renewal application at this time.

Analysis

Aging evaluations of electrical components in the Dresden and Quad Cities EQ program that specify qualification of at least 40 years are TLAAs. As such, a reanalysis as described below will be applied to EQ components now qualified for the current operating term of 40 years.

The reanalysis of an aging evaluation may be performed to extend the qualification by reducing margin or excess conservatism incorporated in the prior evaluation. Reanalysis of an aging evaluation to extend the qualification of a component may be performed as part of the EQ program. While a component life limiting condition may be due to thermal, radiation, or cyclical aging, the vast majority of component aging limits are based on thermal conditions. Conservatism may exist in aging evaluation parameters, such as the assumed ambient temperature of the component, unrealistically low activation energy, or in the application of a component (de-energized versus energized). The important attributes of reanalysis will include analytical methods, data collection and conservative reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met), as discussed below.

Analytical Methods: The analytical models used in the reanalysis of an aging evaluation are the same as those previously applied during the prior evaluation. The Arrhenius methodology is an acceptable thermal model for performing a thermal aging evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose). For license renewal, one acceptable method of establishing the 60-year normal radiation dose is to multiply the 40-year normal radiation dose by 1.5 (that is, 60 years/40 years). The result is added to the accident radiation dose to obtain the total integrated dose for the component. For cyclical aging, a similar approach may be used. Other models may be justified on a case-by-case basis.

Data Collection and Reduction Methods: Reducing excess conservatism in the component service conditions (for example, temperature, radiation, cycles) used in the prior aging evaluation is the chief method used for a reanalysis. Temperature data used in an aging evaluation is to be conservative and based on plant design temperatures or on actual plant temperature data. When used, plant temperature data can be obtained in several ways, including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, and temperature sensors on large motors (while the motor is not running). When used, a representative number of temperature measurements are conservatively evaluated to establish the temperatures used in an aging evaluation. Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation, or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation. Any changes to material activation energy values as part of a reanalysis are justified on a case-specific basis. Similar methods of reducing excess conservatism in the component service conditions used in prior aging evaluations may be used for radiation and cyclical aging.

Underlying Assumptions: EQ component aging evaluations contain sufficient conservatism to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the normal operating environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.

Acceptance Criteria and Corrective Actions: The reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component is maintained, replaced, or re-qualified prior to exceeding the period for which the current qualification remains valid. A reanalysis is performed in a timely manner (that is, sufficient time is available to maintain, replace, or re-qualify the component if the reanalysis is unsuccessful).

Disposition: Aging Management, 10 CFR 54.21(c)(1)(iii)

The existing EQ programs will be continued for the extended operating period. Continuing the existing EQ programs provides reasonable assurance that the aging effects will be managed and that the EQ components will continue to perform their intended functions for the period of extended operation. See Environmental Qualification (EQ) of Electrical Components (B.1.35) for additional information .

4.5 LOSS OF PRESTRESS IN CONCRETE CONTAINMENT TENDONS

None of the Dresden or Quad Cities containments have prestress tendons. As such, this topic is not a TLAA.

4.6 FATIGUE OF PRIMARY CONTAINMENT, ATTACHED PIPING, AND COMPONENTS

The primary containments for Quad Cities and Dresden were designed in accordance with the ASME Code, Section III, 1965 Edition with addenda up to and including Winter 1965. Subsequently, while performing large-scale testing for the Mark III containment system and in-plant testing for Mark I primary containment systems, new suppression chamber hydrodynamic loads were identified. The "new loads" are related to the postulated loss of coolant accident (LOCA) and safety relief valve (SRV) operation. Subsequent to the original design, elements of the Dresden and Quad Cities containments were reanalyzed in response to discoveries by General Electric and others of unevaluated loads due to design basis events and SRV discharge. The load definitions include assumed pressure and temperature cycles resulting from SRV discharge and design basis loss of coolant accident (LOCA) events. This re-evaluation was in two parts: generic analyses applicable to each of the several classes of BWR containments, and Mark I Containment Program plant-unique analyses (PUA). The scope of the analyses included the pressure suppression chambers (shells and welds), the drywell-to-pressure suppression chamber vents (header and downcomers), SRV discharge piping, other piping attached to the pressure suppression chamber, penetrations, and vent bellows.

In the absence of hydrodynamic loads, fatigue is not a concern in containment design except at penetrations or other stress concentration areas. Drywell shell plates were not evaluated for fatigue in the original design and the PUA did not reevaluate the drywell, drywell penetrations, or process penetration bellows, all of which are attached to the drywell. The licensing and design basis documents do not reflect the existence of any fatigue analysis for the drywell or its penetrations. However, the drywell process bellows were specified for a finite number of operating cycles, as were replacement bellows for Quad Cities.

Containment Fatigue analyses are presented for the following groups:

- Fatigue Analysis of the Suppression Chamber, Vents, and Downcomers
- Fatigue Analysis of SRV Discharge Piping Inside the Suppression Chamber, External Suppression Chamber Attached Piping, and Associated Penetrations
- Drywell-to-Suppression Chamber Vent Line Bellows Fatigue Analyses
- Primary Containment Process Penetration Bellows Fatigue Analysis

NUREG-1801 identifies numerous fatigue related aging effects that require evaluation as possible TLAAs in accordance with 10 CFR 54.21(c). Each of these is summarized in NUREG-1800 and presented in Section 3 of this LRA and referenced to the appropriate TLAA section.

4.6.1 Fatigue Analysis of the Suppression Chamber, Vents, and Downcomers

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

New hydrodynamic loads were identified subsequent to the original design for the containment suppression chamber vents. These additional loads result from blowdown into the suppression chamber during a postulated loss-of-coolant accident (LOCA) and during safety relief valve (SRV) operation during plant transients. The results of analyses of these effects were presented in the Plant Unique Analysis Report (PUAR) for each plant. The suppression chamber, and suppression chamber vents including the vent headers and downcomers were modified in order to reestablish the original design safety margins.

The Dresden and Quad Cities PUARs (Volumes 2 and 3) describe fatigue analyses of suppression chamber and suppression chamber vents, including the vent headers and downcomers. (References 4.5 and 4.6) Fatigue analyses of Class MC support members are included in the PUAR summaries of limiting-location usage factors. The analyses assumed a limited number of SRV actuations, based on a survey of plant data extrapolated to 40 years, and are therefore TLAAs.

Analysis

Dresden Suppression Chamber Shells and Associated Welds

There are five SRVs installed at Dresden and Quad Cities. The current design basis analysis assumed 300 SRV actuations of all five SRVs simultaneously during normal operation condition (NOC), plus 25 cycles for an intermediate-break accident (IBA) or 50 for a small-break accident (SBA) which ever was more bounding. The design basis also included an Operating Basis Earthquake (OBE) which was assumed the equivalent of 600 SRV cycles. The limiting analysis involves the 50 small break accidents rather the 25 cycles for an intermediate-break accident. In this limiting case, NOC + (SBA + OBE) , the worst-location fatigue cumulative usage factor is 0.50 for the shells and 0.80 for the associated welds. SRV actuations are the largest contributor, 0.34, to the fatigue cumulative usage factor for the shells; and SBA + OBE is the largest contributor, 0.72, to the fatigue cumulative usage factor for the associated welds.

Dresden Suppression Chamber Vent Headers, Downcomers, and Associated Welds

The current design basis analysis assumed 550 SRV actuations of all five SRVs simultaneously during normal operation (NOC), plus 1000 cycles for OBE, plus 25 IBA or 50 SBA cycles. NOC + (SBA + OBE) is the limiting case, for which the worst-location fatigue cumulative usage factor is 0.92 for the vent headers (at the intersection with the downcomers) and 0.26 for the associated welds. The contribution of the SRV discharge loads to the fatigue cumulative usage factor of the vent headers and associated welds is insignificant (~ 0.00).

Quad Cities Suppression Chamber Shells and Associated Welds

The current design basis analysis assumed 550 SRV actuations of all five SRVs simultaneously during normal operation (NOC), plus 600 cycles for OBE, plus 25 IBA or 50 SBA cycles. NOC + (SBA + OBE) is the limiting case, for which the worst-location fatigue cumulative usage factor is 0.52 for the shells and 0.40 for the associated welds. SRV actuations and SBA + OBE are equal contributors to the fatigue cumulative usage factor for the shells, 0.25 from SRV and 0.27 from SBA + OBE. SRV actuations are the largest contributor, 0.26, to the fatigue cumulative usage factor for the associated welds.

Quad Cities Suppression Chamber Vent Headers, Downcomers, and Associated Welds

The current design basis assumed 550 SRV actuations of all five SRVs simultaneously during normal operation (NOC), plus 1000 cycles for OBE, plus 25 IBA or 50 SBA cycles. NOC + (SBA + OBE) is the limiting case, for which the worst-location usage factor is 0.92 for the vent headers (at the intersection with the downcomers) and 0.26 for the associated welds. The contribution of the SRV discharge loads to the usage factor of the vent headers and associated welds is insignificant (~ 0.00).

Table 4.6.1-1 presents the cumulative usage factors (CUFs) based on calculations using the above inputs. In addition, it assumes that each SRV “actuation” results in one thermal, one pressure, and five dynamic load cycles. However, for the vent header and associated welds, the effect of normal operating condition SRV discharge loads on the fatigue usage factor is insignificant.

Table 4.6.1-1 Summary of CUFs for Suppression Chamber Shell, Vent Header Including Vents and Welds

	Limiting 40-year CUF	**Assumed SRV Actuations**
Dresden Suppression Chamber Shell	0.50	300
Dresden Suppression Chamber Shell Associated Welds	0.80	300
Dresden Suppression Chamber Vent Header	0.92	550
Dresden Suppression Chamber Vent Header Associated Welds	0.26	550
Quad Cities Suppression Chamber Shell	0.52	550
Quad Cities Suppression Chamber Shell Associated Welds	0.40	550
Quad Cities Suppression Chamber Vent Header	0.92	550
Quad Cities Suppression Chamber Vent Header Associated Welds	0.26	550

Disposition: Validation, 10 CFR 54.21(c)(1)(i) and Aging Management, 10 CFR 54.21(c)(1)(iii)

For most shell, vent, and penetration locations, the predicted 40-year usage factor (CUF) is less than 0.666, and the analysis for that location is validated by

$$(U_{max,\,40} < 0.666) \times 60/40 \Rightarrow (U_{max,\,60} < 0.999), < 1.0$$

However, a CUF of 0.666 provides no analytical or event margin. This validation will therefore be applied only to locations with a calculated 40-year CUF less than 0.4. The locations whose 40-year CUF is at least 0.4 will be included in the aging management program, as described below.

Since only the SRV load cases contribute to fatigue during normal operation, normal operation may continue so long as the contribution from SRV discharges has not exceeded 1.0 minus the contribution expected from the postulated worst-case LOCA plus OBE event. The high-usage-factor locations in the containment shell, vents, and penetrations will be also managed by the Metal Fatigue of Reactor Coolant Pressure Boundary (B.1.34) aging management program. Specifically, SRV lifts will be monitored to ensure that CUFs remain less than 1.0. For the primary containment, this program will monitor CUFs through Cycle-Based Fatigue Monitoring (CBF).

Cycle-based fatigue monitoring consists of a two step process: (a) automated cycle counting, and (b) CUF computation based on the counted cycles. The automated cycle counting counts each transient that is defined in the plant licensing basis based upon the mechanistic process or sequence of events experienced by the plant (as determined from monitored plant instruments). The approach is conservative because it assumes each actual transient has a severity equal to that assumed in the design basis. The unique severity of any transient identified by the aging management program software is captured for each monitored component, for ready comparison to design basis transient severity. Transients defined in the Dresden and Quad Cities UFSAR are identified and implemented into the aging management program software. CUF computation calculates fatigue directly from counted transients and parameters for the monitored components. CUF is computed via a design-basis fatigue calculation where the numbers of cycles are substituted for assumed design basis number of cycles. The CUF calculations are conservative in that design basis transient severity is assumed.

All governing fatigue analyses have been reviewed to establish a comprehensive and bounding set of Primary Containment locations for inclusion in the Metal Fatigue of Reactor Coolant Pressure Boundary (B.1.34) aging management program. All locations with 40-year CUFs expected to exceed 0.4 will be included in the program. Those locations are listed in Table 4.6.1-2 below. The associated CUFs are all based upon the original 40 year analysis.

All necessary plant transient events will be tracked to ensure that the CUF remains less than 1.0 for all monitored components. In the event fatigue usage is predicted to exceed 1.0 for any component prior to 60 years of operation, appropriate corrective action will be taken in accordance with the Exelon Corrective Action Program (B.2.1). The required implementing actions will be completed prior to the period of extended operation.

Table 4.6.1-2 Fatigue Monitoring Locations for Primary Containment Components

Plant	Primary Containment Location	Fatigue Usage Factor
Dresden 2 and 3	Suppression Chamber Weld	0.800
Dresden 2 and 3	Vent Header at the Downcomer-Vent Header Intersection	0.920
Quad Cities 1 and 2	Suppression Chamber Shell	0.520

The locations listed above apply to all of our Quad Cities/Dresden Units and represent/bound all other locations for both plants.

4.6.2 Fatigue Analysis of SRV Discharge Piping Inside the Suppression Chamber, External Suppression Chamber Attached Piping, and Associated Penetrations

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

There are 5 safety relief valves (SRV) located on the main steam piping for each unit at Dresden and Quad Cities. When open, steam discharges from each SRV through piping routed through the drywell to the suppression chamber. The SRV discharge piping enters the suppression chamber through penetrations on the suppression chamber vent header where the steam is discharged to the suppression chamber water through a T-quencher attached to the suppression chamber. Additionally, there are a number of external piping systems attached to the suppression chamber shell. The Plant-Unique Analysis Reports for each site describe the fatigue analyses of SRV discharge lines, T-quenchers, the SRV discharge line penetrations through the vent lines, suppression chamber shell (torus) attached piping (TAP) systems, and the associated penetrations. These analyses assume a limited number of SRV actuations throughout the 40-year life for the plant and are therefore TLAAs.

Analysis

The suppression chamber assembly is often referred to as the “torus” and is used interchangeably with the phrase “suppression chamber” in the discussions that follow. The Dresden and Quad Cities analyses of fatigue effects of Mark I cyclic “new loads” on SRV discharge lines (SRVDL), internal to the suppression chamber, and on torus attached piping (TAP) external to the suppression chamber can be summarized as follows:

External TAP and Class 2 and 3 SRV Discharge Lines

Both the TAP and the SRV discharge lines in the suppression chamber are covered by a generic fatigue analysis for Class 2/3 piping which was submitted for NRC approval by the Mark I containment owners group. The generic analysis assumed 800 SRV actuations for a 40-year plant lifetime. Other thermal cycles due to normal operating conditions were considered to be negligible. The analysis concluded that the SRV discharge lines and TAP would have a fatigue cumulative usage factor of less than 0.5 at the end of 40 years of operation.

SRV Discharge Line-Vent Line Penetrations (“Sleeves”) and Associated Sections of the SRV Discharge Lines

Fatigue analyses of these Class MC components shows that the maximum fatigue usage factors for these piping system locations are 0.18 for Quad Cities and 0.09 for Dresden (References 4.17 and 4.18). The analysis performed considered two separate cases in order to identify bounding analysis. Fatigue analyses assumed 220 SRV actuations for “Case 1,” 110 for “Case 2.”

Case 1: All 220 SRV actuations are single-valve first actuations (i.e. no subsequent actuations). This maximizes the number of cycles of full-range $|T_A - T_B|$ stress.

Case 2: 110 isolation events, each with 1 first actuation and 1 subsequent actuation. This maximizes the number of cycles in which mechanical and thermal expansion loads from a subsequent actuation add to the thermal transient stress from the first actuation.

Case 2 can result in a higher usage factor than the “220 actuation” Case 1: Case 2 is in fact not 110 “actuations” but 110 “events” of two actuations each, and upon analysis, the Case 2 event definition is found to include more-severe loads than Case 1.

Calculations of Class MC penetration components and attachments assumed 220 SRV actuations of 5 dynamic cycles each, plus 4,050 cycles due to condensation oscillation or chugging. The calculations determined the allowable alternating stress intensity S_A, from Figure I–9.0 of the ASME Code for that number of load cycles. This method did not calculate a usage factor, but instead limited the alternating stress intensity S_A to a value that ensured that the usage factor will remain less than 1.0 for the number of cycles assumed to occur during the design life.

<u>Analysis for ECCS Suction Strainer Modifications</u>

To address concerns associated with potential plugging and unacceptable head loss, ECCS suction strainers were replaced at Dresden and Quad Cities. Larger ECCS suction strainers required reinforcement of ECCS suction header-strainer penetrations X-303A, B, C, and D at Dresden and of penetrations X-204A, B, C, and D at Quad Cities Unit 2. The additional loads required revised analyses of the stress and fatigue effects.

The Dresden fatigue analyses of these ECCS penetrations were based on the generic Mark I owners group analysis. The analyses assumed 300 SRV actuations in 40 years, plus 25 cycles for intermediate-break accident (IBA) and 50 for a small-break accident (SBA). However, improvements in the SRV actuation load definition model permitted the significant stress reversals for each SRV actuation to be reduced from five to three. This revised analysis assumed five operating basis earthquake (OBE) events.

For Dresden Unit 2 the worst-location worst-case fatigue cumulative usage factors are 0.0836 at the suppression chamber insert-nozzle combination and 0.1423 at the corresponding pad, of which the largest contributions, 0.040 and 0.0571, respectively, are from the 300 normal-operating-condition (NOC) SRV events.

For Dresden Unit 3, the worst-location worst-case usage factors are 0.0570 at the suppression chamber insert-nozzle combination, and 0.0832 at the corresponding pad, of which the largest contributions, 0.0216 and 0.0348, respectively, are from the 300 NOC SRV events.

The Quad Cities Unit 2 fatigue analysis of these ECCS penetrations was based on generic analysis, MPR-751, “Mark I Owners Group Generic Fatigue Evaluation Report - 1982.” The analysis assumed 550 SRV actuations in 40 years, plus 25 cycles for intermediate-break accident (IBA) and 50 for a small-break accident (SBA). However, improvements in the SRV actuation load definition model permitted the significant stress reversals for each actuation to be reduced from five to three. This revised analysis assumed five operating basis earthquake (OBE) events, in agreement with MPR-751. The worst-location worst-case maximum 40-year usage factor is 0.3087, of which the largest contribution, 0.1713, is from the 550 NOC SRV events.

Table 4.6.2-1 provides a summary of the maximum cumulative usage factors discussed for the suppression pool shell attached piping, SRV discharge lines, and penetrations. The analysis assumes that each SRV "actuation" results in one thermal, one pressure, and five (later reduced to three) dynamic load cycles. Note that at both plants, Case 2 is limiting. A CUF "<1.0" means the usage factor was not calculated. The calculation determined an S_A limit from the number of load cycles that ensures that the usage factor is less than 1.0.

Disposition: Validation, 10 CFR 54.21(c)(1)(i) and Aging Management, 10 CFR 54.21(c)(1)(iii)

For most safety relief valve (SRV) discharge lines, T-quenchers, the SRV discharge line penetrations through the vent lines, suppression chamber-attached piping (TAP) systems, and the associated penetration locations the predicted 40-year usage factor (CUF) is less than 0.666, and the analysis for that location is validated by

$$(U_{max,\,40} < 0.666) \times 60/40 \Rightarrow (U_{max,\,60} < 0.999), < 1.0$$

However, a CUF of 0.666 provides no analytical or event margin. Therefore, this validation will be applied only to locations with a calculated 40-year CUF less than 0.4. The locations whose 40-year CUF is at least 0.4, will be included in the aging management program, described below.

Since only the SRV load cases contribute to fatigue during normal operation, normal operation may continue so long as the contribution from SRV lifts has not exceeded 1.0 minus the contribution expected from the postulated worst-case LOCA plus OBE event. This will be confirmed for the duration of the extended operating period by monitoring fatigue at the high-usage-factor containment system locations using the Metal Fatigue of Reactor Coolant Pressure Boundary (B.1.34) aging management program. Specifically, SRV lifts will be monitored to ensure that CUFs remain less than 1.0 for the Dresden and Quad Cities Class MC SRVDL-Vent Line Penetration Components and Welds listed in Table 4.6.2-1. The Suppression Chamber Attached Piping (TAP) and Class 2 and 3 SRVDL's listed in Table 4.6.2-1 is bounded and covered by the suppression chamber shell which is monitored by the Metal Fatigue of Reactor Coolant Pressure Boundary (B.1.34) aging management program shown in Table 4.6.1-2. This program will be initiated prior to the period of extended operation.

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Table 4.6.2-1 Summary of CUFs for Shell-Attached Piping, SRVDLs, and Penetrations

Scope	Limiting 40-year CUF	Assumed SRV Actuations
Suppression Chamber Attached Piping (TAP) and Class 2 and 3 SRVDLs	0.5	800, 5 cycles each
Dresden Class MC Portions of SRVDLs at the Vent Line Penetrations ("Sleeves")	0.061 0.093	Case 1: 220 Case 2: 110 events, 220 actuations@5 cycles each
Quad Cities Class MC Portions of SRVDLs at the Vent Line Penetrations ("Sleeves")	0.042 0.18	Case 1: 220 Case 2: 110 events, 220 actuations @5 cycles each
Dresden and Quad Cities Class MC SRVDL-Vent Line Penetration Components and Welds	<1.0 (See Note Below)	220, 5 cycles each, plus 4,050 OBE, chugging, and condensation oscillation (CO) cycles
Dresden Unit 2 Revised Class MC Analysis of ECCS Suction Penetrations for Strainer Modifications	Suppression chamber-insert-nozzle: 0.08 Pad plate: 0.14	300, 3 cycles each
Dresden Unit 3 Revised Class MC Analysis of ECCS Suction Penetrations for Strainer Modifications	Suppression Chamber, insert, nozzle: 0.06 Pad plate: 0.08	300, 3 cycles each
Quad Cities Unit 2 Revised Class MC Analysis of ECCS Suction Penetrations for Strainer Modifications	0.31	550, 3 cycles each

Note: A CUF "<1.0" means the usage factor was not calculated.

4.6.3 Drywell-to-Suppression Chamber Vent Line Bellows Fatigue Analyses

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

The drywell-to-suppression chamber vent line bellows are included in the Mark I Containment Long-Term Program (LTP) plant-unique analyses. A fatigue analysis of the drywell-to-suppression chamber vent line bellows demonstrates their adequacy to accommodate thermal and internal pressure load cycles for the life of the plant.

Analysis

Mark I containment designs include a drywell-to-suppression chamber vent line. A bellows assembly is provided at the penetration of the vent line to the suppression chamber. The bellows allows differential movement of the vent system and suppression chamber to occur without developing significant interaction loads. The analysis of the drywell-to-suppression chamber vent line bellows assumed a total of 150 load cycles for the life of the plant. The bellows are adequate for fatigue since the bellows have a rated capacity of 1,000 cycles at maximum displacement.

Dresden has external vacuum breakers which include 24-inch bellows at their penetration attachment. Quad Cities has no corresponding bellows configuration. The bellows of the external vacuum breakers are included in the vent system analysis. The differences between Dresden and Quad Cities include the external vacuum breakers. The effect of these differences in the overall vent system analysis was found to be insignificant, and therefore the Dresden analyses were based on Quad Cities geometry. An NRC SER concluded that the owner's review of the Dresden external vacuum breakers to the stress criteria of ASME III Division 1 Subsection NC (1977, Summer 1977 Addenda) demonstrated that the design is adequate. (Reference 4.7) "The fatigue usage factors in the vacuum breakers and their attachments, (and at Dresden, their penetrations) are therefore bounded by other analyzed locations and are not limiting."

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

For the vent line bellows, 150 thermal and internal pressure load cycles were specified for the life of the plant. The expected total load cycles for the extended operation period would therefore be 150 x (60/40) = 225 cycles. This is less than 25 percent of the rated capacity of 1000 cycles. The fatigue adequacy of these bellows is therefore validated.

The rated capacity of the drywell-to-shell vent line bellows is adequate for the number of pressure and temperature cycles expected during the extended licensed operating period.

4.6.4 Primary Containment Process Penetrations Bellows Fatigue Analysis

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

Containment pipe penetrations that must accommodate thermal movement have expansion bellows. The bellows are designed for a minimum number of operating thermal cycles over the design life at containment normal, test, and limiting design pressures.

Analysis

At Dresden and Quad Cities the only containment process piping expansion joints are those between the drywell shell penetrations and process piping subject to significant thermal expansion and contraction. These containment penetration process bellows have been designed for 7000 operating and thermal cycles.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

Thermal cycles on the bellows are imposed by thermal cycles experienced by the attached piping. The assumed thermal cycle count for the analyses used in the codes associated with piping and components can conservatively be approximated by the thermal cycles used in the reactor vessel fatigue analysis. These thermal cycles are listed in UFSAR Table 3.9-1. The total count of all these listed thermal cycles is less than 2200 for a 40-year plant. For the 60-year extended operating period, the number of thermal cycles for piping analyses would be proportionally increased to less than 3300, a fraction of the 7000-cycle threshold. The containment penetration bellows fatigue analysis is therefore valid for the 60-year extended operating period.

4.7 OTHER PLANT-SPECIFIC TLAAS

NUREG-1801 identifies numerous aging effects that require evaluation as possible TLAAs in accordance with 10 CFR 54.21(c). Each of these is summarized in NUREG-1800 and presented in Section 3 of this LRA and referenced to the appropriate TLAA section.

Section 4.7 of this application contains a discussion on the following plant-specific TLAAs:

- Reactor Building Crane Load Cycles
- Metal Corrosion Allowances
  - Corrosion Allowances for Power Operated Relief Valves
  - Degradation Rates of Inaccessible Exterior Drywell Plate Surfaces
  - Galvanic Corrosion in the Containment Shell and Attached Piping Components Due to Stainless Steel ECCS Suction Strainers
- Crack Growth Calculation of a Postulated Flaw in the Heat Affected Zone of an Arc Strike in the Containment Shell
- Radiation Degradation of Drywell Shell Expansion Gap Polyurethane Foam

High-Energy Line Break postulation based on fatigue cumulative usage factor is listed as a possible TLAA in NUREG-1800. However, this is not a TLAA for either Dresden or Quad Cities. Section 4.7.5 provides addition information supporting this conclusion.

4.7.1 Reactor Building Crane Load Cycles

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

The reactor building overhead cranes at Dresden and Quad Cities were designed to meet or exceed the design fatigue loading requirements of the Crane Manufacturers Association of America (CMAA) Specification 70, Class A1. This evaluation of expected cycles over the 40-year life is the basis of a safety determination and is therefore a TLAA.

Analysis

The service classification for the reactor building overhead crane is CMAA Class A1 and is designed for 100,000 loading cycles. The weldments are categories B and C, which permit a stress range of 28,000 - 33,000 psi (References 4.3 and 4.4).

The maximum allowable stress in the girders with rated load is 17,600 PSI and the minimum stress (no load) is approximately 2,400 PSI. The maximum stress range in the girders will not exceed 15,200 PSI. Since the maximum stress permitted in other weldments is 14,000 PSI, they have a smaller range and better fatigue resistance than the girders. These ranges are satisfactory for approximately 2 million loading cycles. This would be equivalent to approximately 50,000 125-ton loads per year handled in the center of span over a 40-year period.

It is estimated that these cranes will see fewer than 5,000 cycles at rated capacity and a larger number of cycles at significantly less than rated capacity. For this reason, fatigue life is not significant to the operation of this equipment (References. 4.19, 4.20, and 4.21).

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The reactor building cranes are designed to CMAA-70 Class A1. The design evaluations show that all components are qualified for 100,000 loading cycles (i.e., 100,000 lifts at rated capacity). The 40-year estimated cycles are at most 5,000 rated-capacity load cycles and 7,500 if extended to a 60-year life. The 60-year 7,500-cycle estimate remains a small fraction of the 100,000 cycle minimum design. Therefore, fatigue life is not significant to the operation of this equipment and remains valid for the period of extended operation.

4.7.2 Metal Corrosion

General corrosion is the result of a chemical or electrochemical reaction between a material and an aggressive environment. General corrosion is normally characterized by uniform attack resulting in material dissolution and sometimes corrosion product buildup. The metal can thin down and fail by either penetration or lack of cross sectional area to support the required load. Certain plant components have a design life limited by corrosion rates. As such, these design limitations may be a TLAA.

Section 4.7.2 of the application addresses the following corrosion related TLAAs:

- Corrosion Allowance for Power Operated Relief Valves
- Degradation Rates of Inaccessible Exterior Drywell Plate Surfaces
- Galvanic Corrosion in the Containment Shell and Attached Piping Components Due to Stainless Steel ECCS Suction Strainers

4.7.2.1 Corrosion Allowance for Power Operated Relief Valves

Applicability

This section applies to Quad Cities Unit 2 only.

Summary Description

Power Operated Relief Valves (PORVs) installed on the Quad Cities Unit 2 reactor coolant pressure boundary for overpressurization relief replacement PORVs were designed with a corrosion allowance valid for 40 years of operation. The specification is cited in Quad Cities UFSAR Section 5.2.2 and as such, the corrosion design allowance for the PORVs is a TLAA.

Analysis

A modification was installed on Quad Cities Unit 2 to replace the main steam line electromatic relief valves with PORVs. The electromatic relief valves were replaced for purposes other than general corrosion damage. The specification for the PORVs is cited in Quad Cities UFSAR § 5.2.2. The replacement PORVs were designed with a corrosion allowance designed for 40 years of operation. The corrosion rate and allowances remain applicable for the remaining design life.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The PORVs were installed on Quad Cities Unit 2 in 1995. Since the valves were installed more than 20 years into the current license, the values remain valid for the period of extended operation.

4.7.2.2 Degradation Rates of Inaccessible Exterior Drywell Plate Surfaces

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

The Mark I containment design includes an inaccessible "sand pocket" around the drywell. The potential for degradation of the containment exists due to conditions that allow the introduction of water into the annulus (expansion gap) between the containment and the primary containment shield wall. Water can be introduced due to leakage of the refuel cavity past the refueling bellows drain line expansion joints during refueling or due to the introduction of water at other drywell penetrations. This water migrates to the sand pocket under the bottom elevation of the containment and then passes through the sand pocket drain lines. If the drain lines become clogged, the water remains in the sand pocket and creates an environment that may be corrosive to the containment steel plates.

In response to Generic Letter 87-05, "Potential Degradation of Mark I Drywell," Dresden and Quad Cities projected corrosion rates for the steel drywell plates in this area and determined that the wall thickness was sufficient for the remainder of the 40-year license period. The evaluation of remaining life of the drywell steel thickness based on a specified corrosion rate is a TLAA.

Analysis

In response to Generic Letter 87-05, both Dresden and Quad Cities performed an inspection of the drain lines on all four units to detect leakage in the pocket region. As a result of these actions, it was determined that the sand pocket drains were clogged on Dresden Unit 3 and an evaluation of actual plate thickness was performed on this unit.

The design of the Dresden Unit 3 containment vessel is such that margin exists between the required shell thickness and the actual thickness of the steel plate provided. A reevaluation of the required shell thickness in the region of the sand pocket was performed based on loads and data compatible with the original certified containment vessel stress report by Chicago Bridge & Iron Company. It was determined that the thickness of the plates in the sand pocket region may be reduced to approximately ¼ inch below the nominal and still be within ASME Code allowable stress limits.

Actual UT thickness measurements were made of the Dresden Unit 3 drywell steel plate at the sand pocket level. All thickness measurements were on the high side (above the nominal 1.0625 inch). The measurements were taken during the 18th year of operation for Dresden Unit 3. In response to GL 87-05, conservative estimates of corrosion rates that might occur were made. Starting with the minimum as-found steel plate thickness of 1.08 inch and assuming a corrosion rate of 0.01 inch per year, it was concluded that 27 years of service would remain before the effects of corrosion on stresses would become significant. The corrosion rate was based upon a worst case rate of 10 mils per year for fresh river water.

The final response to Generic Letter 87-05 concluded that the amount of moisture found in the sand pocket drains at Quad Cities was negligible in comparison to Dresden Unit 3 and that it was not expected that any corrosion occurred on either unit. The final response also concluded that there was no reason to expect adverse thickness of the drywell liner on Dresden Unit 2. However, the Dresden Unit 3 plate thickness estimates were used to bound Dresden Unit 2 and both units at Quad Cities. The conservatively analyzed years of service life (18 + 27 = 45) would not be sufficient for the extended license period.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

The corrosion rate assumptions used in the calculation will be confirmed by a UT inspection prior to the period of extended operation. The inspection will be performed at Dresden and the results will be used to revise the corrosion calculation and validate that an acceptable wall thickness will remain to the end of the 60 year license operating period.

4.7.2.3 Galvanic Corrosion in the Containment Shell and Attached Piping Components due to Stainless Steel ECCS Suction Strainers

Applicability

This section applies to Dresden and Quad Cities

Summary Description

To address concerns associated with potential plugging and unacceptable head loss, Exelon replaced the ECCS suction strainers at Dresden and Quad Cities. This modification resulted in contact between the carbon steel support flanges and stainless steel components. The effects of galvanic corrosion on these components were evaluated in a calculation. The evaluation of the effects of galvanic corrosion on the ECCS suction strainer flanges and other components in containment shell is based on a predicted corrosion rate for the plant lifetime, and is a TLAA.

Analysis

The modification to install larger ECCS strainers required drilling new bolt holes and enlarging existing bolt holes in each of the original strainer support flanges to provide sufficient bolting for the larger replacement strainers. The holes drilled in the carbon steel flanges were not coated to protect them from corrosion. Previously, coatings prevented contact and corrosion loss. The existing design calculation for the modified strainer support flanges assumes a corrosion loss of 4 mils/year for 33 years which is not sufficient to encompass the entire period of extended operation.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii)

The corrosion rate assumptions used in the calculation will be confirmed by a ultrasonic inspection prior to the period of extended operation. One bounding inspection will be performed and results will be used to validate the corrosion rate for both sites. Based upon the results of the inspection, a revised galvanic corrosion calculation will be performed to ensure acceptable wall thickness to the end of the 60-year licensed operating period.

4.7.3 Crack Growth Calculation of a Postulated Flaw in the Heat Affected Zone of an Arc Strike in the Suppression Chamber Shell

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

Dresden and Quad Cities evaluated the effect of an arc strike flaw on the suppression chamber wall based on a common analysis. Postulated crack growth rates due to operation were evaluated to be acceptable over a 40-year life.

No other flaw evaluation TLAAs were found for Dresden and Quad Cities.

Analysis

In 1990, arc strikes found in the Dresden and Quad Cities suppression chamber walls were evaluated using a common analysis. The evaluation included a crack growth calculation and assumed 850 load cycles due to SRV and other operations in 40 years of plant life. A further evaluation was performed in 1997 and it was determined that the flaw depth of the arc strike at Dresden was not of sufficient depth to warrant any final repairs. Assuming an operating limit of 850 SRV load cycles, no further action was warranted. A UT measurement was performed at Quad Cities that validated that there is no flaw in the heat affected zone of the original arc strike. An evaluation was performed and it was determined that further repairs or inspections were not warranted.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The number of SRV actuations used in the Quad Cities containment analysis is 550 for 40 years compared with the 850 actuations assumed in the flaw evaluation calculation. The expected number of SRV actuations from the year the flaw was repaired, 1990, to the end of the extended operating period in 2032 would be at most (550/40) x (2032-1990) = 577.5 <850. Therefore the evaluation remains valid for Quad Cities for the extended period of operation.

The number of SRV actuations used in the Dresden containment analysis is 300 for 40 years compared with the 850 actuation assumed in the flaw evaluation calculation, the Dresden TLAA can be validated similarly. The current license for Dresden Unit 3 will expire in 2011, and the Dresden flaw was repaired in 1991. The expected number of SRV actuations from the year the flaw was repaired to the end of the extended operating period in 2031 would be at most (300/40) x (2031-1991) = 300 <850. Therefore the evaluation remains valid for Dresden for the extended period of operation.

4.7.4 Radiation Degradation of Drywell Shell Expansion Gap Polyurethane Foam

Applicability

This section applies to Dresden and Quad Cities.

Summary Description

The steel drywell shell is largely enclosed within the structural and shielding concrete of the reactor building. To accommodate thermal expansion, compressible foam was used to form an expansion gap between the concrete and the drywell shell. An analysis evaluated the increase in external compressive loads on the drywell exterior, due to additional compression of this foam, for normal, refuel, and accident conditions. The effect on this analysis of a postulated increase in the foam stiffness resulting from radiation dose is a TLAA.

Analysis:

The polyurethane foam material was chosen for its resistance to the environmental conditions likely to exist during its service life. The polyurethane foam was chosen such that the effects of compression during a loss of coolant accident resulting in thermal expansion of the drywell would not exceed ASME Code allowable limits. Polyurethane foam samples, similar to those used in the gap, were irradiated in a test lab at various levels, from 10^7 and 10^9 rads. The test results established that there was no detectable change in resilience below 10^8 rads. The original design considered the effects of a 40-year lifetime dose of 2.5×10^7 rads on the foam material. As such, the resilient characteristics of the polyurethane foam will remain in tact during the 40-year design life.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

A 20-year increase in the design lifetime to 60 years, combined with approved increases in power rating, would conservatively result in a total radiation exposure of 4.2×10^7 rads. This is less than half of the 10^8 rads qualified radiation exposure. Therefore, the material properties will remain within the limits assumed by the original design analysis, in accordance with the aging assumptions assumed by the original design, for the 60-year extended operating period.

4.7.5 High-Energy Line Break Postulation Based on Fatigue Cumulative Usage Factor

Summary Description

This issue is included only because it is listed as a possible TLAA in NUREG-1800. Neither Dresden nor Quad Cities postulated break locations are based on a fatigue usage factor criterion, nor are any break locations based on any other evaluation of fatigue effects. This is not a TLAA for either Dresden or Quad Cities.

4.8 REFERENCES

4.1 USNRC *Supplemental Safety Evaluation Regarding the Proposed Core Shroud Repair - Quad Cities Nuclear Power Station, Units 1 and 2 (TAC Nos. M91301 and M91302).*" Attached to a letter from R. M. Pulsifer (USNRC) to D. L. Farrar (ComEd), September 11, 1995.

4.2 USNRC *Safety Evaluation by the Office of Nuclear Reactor Regulation, GE Nuclear Energy Topical Report NEDC-32983P, 'General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations,' Project No. 710.* Attached to NRC Letter MFN 01-050, Stuart A. Richards, Director, Project Directorate IV, to James F. Klapproth, Manager, Engineering and Technology, GE Nuclear Energy, "Safety Evaluation for NEDC-32983P, 'General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation,' (TAC No. MA9891)," September 14, 2001.

4.3 *Dresden Station Updated Final Safety Analysis Report,* Revision 4. Exelon Nuclear: 2001. Cited throughout this chapter as "Dresden UFSAR."

4.4 *Quad Cities Station Updated Final Safety Analysis Report,* Revision 6. Exelon Nuclear: 2001. Cited throughout this chapter as "Quad Cities UFSAR."

4.5 *Dresden Nuclear Power Station Units 2 and 3 Plant Unique Analysis Report.* Volumes 1-4, 6, and 7, NUTECH Engineers, Inc. Report COM-02-041, May 1983. Volume 5, Sargent and Lundy, June 1983.

4.6 *Quad Cities Nuclear Power Station Units 1 and 2, Mark I Plant Unique Analysis Report.* Volumes 1-4, 6, and 7, NUTECH Engineers, Inc. Report COM-02-039, May 1983. Volume 5, Sargent and Lundy, June 1983

4.7 Dresden SER 861219, *Safety Evaluation by the Office of Nuclear Reactor Regulation Relating to Mark I Containment Program - Vacuum Breaker Integrity, Commonwealth Edison Company, Dresden Nuclear Power Station Units 2 and 3, Docket Nos.: 50-237 and 249.* With attached Franklin Research Center Technical Evaluation Report TER-C5506-324, *Structural Evaluation of the Vacuum Breakers (Mark I Containment Program), Commonwealth Edison Company, Dresden Nuclear Power Station Units 2 and 3,* July 17, 1986; and attached Continuum Dynamics, Inc. Tech Note No. 84-30, *Mark I Wetwell to Drywell Differential Pressure Load and Vacuum Breaker Response for the Dresden Nuclear Power Station Units 2 and 3,* Revision 0, January 1985. All attached to an NRC Letter, John A. Zwolinski, Director, BWR Project Directorate #1, Division of BWR Licensing, to Dennis L. Farrar, Director of Nuclear Licensing, Commonwealth Edison, "Structural Integrity of Vacuum Breakers (TAC 57152 and 57153)," December 19, 1986.

4.8 Ranganath, S., "Fracture Mechanics Evaluation of a Boiling Water Reactor Vessel Following a Postulated Loss of Coolant Accident," Fifth International Conference on Structural Mechanics in Reactor Technology, Berlin, Germany, August 1979, Paper G1/5.

This work was docketed as an attachment to a PP&L letter of October 8, 1991 for Dockets 50-387 and 50-388 (Susquehanna Steam Electric Station) and its attached "Response to Request for Additional Information Enforcement Action 89-042 on Reactor Vessel Cooldown Rate," Revision 0, October 10, 1991 [PDR ACN 9910110101]. This RAI response cites the work as "GE SASR 89-40 Reference 7-9." The GE report is

> General Electric Report SASR 89-40, "Pressure-Temperature Curve Basis for Susquehanna Steam Electric Station Units 1 and 2," June 1989.

General Electric Report SASR 89-40 contains proprietary information belonging to both General Electric and PP&L, and has not been released or reviewed. This Susquehanna citation is however an example of application of the 1979 Ranganath paper to a BWR-4, although the sample case was a BWR-6.

4.9 General Electric Document Y1002A602, “304 Stainless Steel, Irradiated,” Revision 3, October 16, 1985.

4.10 General Electric Nuclear Energy NEDO-32962 Revision 1, DRF A22-00103-13, Class I, August 2001, “Safety Analysis Report for Dresden 2 & 3 Extended Power Uprate”

4.11 General Electric Nuclear Energy NEDO-32961 Revision 1, DRF A22-00103-13, Class I, August 2001, “Safety Analysis Report for Quad Cities 1 & 2 Extended Power Uprate”

4.12 ASTM Special Technical Publication No. 426, *The Effects of Radiation on Structural Materials.* Philadelphia: ASTM, 1966. (Cited by D and QC UFSAR § 3.9.5.3.2, not separately reviewed.)

4.13 L. A. Waldman and M. Doumas, “Fatigue and Burst Tests on Irradiated In-Pile Stainless Steel Pressure Tubes,” *Nuclear Applications,* Vol. 1, October 1965. (Cited by D and QC UFSAR § 3.9.5.3.2, not separately reviewed.)

4.14 BWRVIP-05, EPRI Report TR-105697, *BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05).* For the Boiling Water Reactor Owners Group (Proprietary), September 28, 1995, with supplementing letters of June 24 and October 29, 1996, May 16, June 4, June 13, and December 18, 1997, and January 13, 1998. Cited by the SRP-LR [NUREG 1800,] as Section 4.2 Reference 4. (EPRI Proprietary Information)

4.15 BWRVIP-05 SER (Final), USNRC letter from Gus C. Lainas to Carl Terry, Niagara Mohawk Power Company, BWRVIP Chairman, “Final Safety Evaluation of the BWRVIP Vessel and Internals Project BWRVIP-05 Report,” (TAC No. M93925), July 28, 1998. Cited by the SRP-LR [NUREG 1800] as Section 4.2 Reference 5.

4.16 BWRVIP-05 SER (Supplement), USNRC letter from Jack R. Strosnider, Jr., to Carl Terry, BWRVIP Chairman, “Supplement to Final Safety Evaluation of the BWRVIP Vessel and Internals Project BWRVIP-05 Report,” (TAC No. MA3395), March 7, 2000.

4.17 S&L Report EMD-033967 for Quad Cities, *Piping Stress Analysis Report, Mark I Program Plant Unique Analysis of the SRV Discharge Piping System in the Suppression Chamber, Quad Cities Nuclear Power Station, Units 1&2.* Revision 0, March 23, 1983.

4.18 S&L Report EMD-036594 for Dresden, *Piping Stress Analysis Report, Mark I Program Plant Unique Analysis of the SRV Discharge Piping System in the Suppression Chamber, Dresden Nuclear Power Station, Units 2&3.* Revision 0, February 16, 1983.

4.19 Letter from G. A. Abrell (ComEd Nuclear Licensing Administrator) to NRC, "Supplement to Dresden Special Report No. 41 and Quad Cities Special Report No. 16, …" December 8, 1975.

4.20 Letter from G. A. Abrell (ComEd Nuclear Licensing Administrator) to NRC, "Supplement to Dresden Special Report No. 41 and Quad Cities Special Report No. 16, …" February 9, 1976.

4.21 Quad Cities SER 77012701, "Am Nos. 37/35 Modified Crane Handling System," January 27, 1977.

4.22 Title 10 US Code, Part 50, Section 49 (10 CFR 50.49), “Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants.”

4.23 DOR Guidelines, “Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors,” U.S. Nuclear Regulatory Commission, June 1979.

4.24 NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment,” U.S. Nuclear Regulatory Commission, July 1981.

4.25 Regulatory Guide 1.89, Revision 1, “Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, June 1984.

4.26 C.I. Grimes (NRC) letter to D. Walters (NEI), “Guidance on Addressing GSI-168 for License Renewal,” Project 690, June 1998.

4.27 USNRC Safety Evaluation by the Office of Nuclear Reactor Regulations, Related to Alternative to Inspection of Reactor Pressure Vessel Circumferential Welds, Dresden Nuclear Power Station, Units 2 and 3. Attached to NRC letter from Anthony J. Mendiola, Chief Section 2 Project Directorate III to Oliver D. Kingsley, Dresden-Authorization for Proposed Alternative Reactor Pressure Vessel Circumferential Weld Examinations (TAC NOS. MA6228 and MA6229) dated February 25, 2000.

APPENDIX A
UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) SUPPLEMENT

APPENDIX A UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) SUPPLEMENT

Introduction

The summary descriptions of aging management program activities presented in this Appendix A represent commitments for managing aging of the systems, structures and components within the scope of license renewal during the period of extended operation. This appendix also provides summary descriptions of time-limited aging analyses. These summary descriptions of aging management program activities and time-limited aging analyses will be incorporated in the Updated Final Safety Analysis Reports for the Dresden Nuclear Power Station and the Quad Cities Nuclear Power Station following issuance of the renewed operating license.

A separate Appendix A Updated Final Safety Analysis Supplement has been provided for Dresden, Units 2 and 3 and for Quad Cities, Units 1 and 2.

Dresden Units 2 and 3

Updated Final Safety Analysis Supplement

A.1 AGING MANAGEMENT PROGRAMS

A.1.1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

The ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD aging management program consists of periodic volumetric and visual examinations of components for assessment, identification of signs of degradation, and establishment of corrective actions. Prior to the period of extended operation the program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda.

A.1.2 Water Chemistry

The water chemistry aging management program consists of monitoring and control of water chemistry to keep peak levels of various contaminants below system-specific limits based on industry-recognized guidelines of EPRI TR-103515, “BWR Water Chemistry Guidelines.” To mitigate aging effects on component surfaces that are exposed to water as process fluid, the chemistry programs are used to control water chemistry for impurities (e.g., chlorides, and sulfates) that accelerate corrosion.

A.1.3 Reactor Head Closure Studs

The reactor head closure studs aging management program includes inservice inspection (ISI). This program also includes preventive actions and inspection techniques for BWRs. Prior to the period of extended operation the program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda. The reactor head studs are not metal-plated, and have had manganese phosphate coatings applied.

A.1.4 BWR Vessel ID Attachment Welds

The BWR vessel ID attachment welds aging management program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved Boiling Water Reactor Vessel and Internals Project (BWRVIP)-48, “Vessel ID Attachment Weld Inspection and Evaluation Guidelines,” and/or ASME Section XI; and (b) monitoring and control of reactor coolant water chemistry in accordance with industry-recognized guidelines of EPRI TR-103515, “BWR Water Chemistry Guidelines.” Prior to the period of extended operation the program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda.

A.1.5 BWR Feedwater Nozzle

The BWR feedwater nozzle aging management program includes enhancing the inservice inspections (ISI) specified in the ASME Code, Section XI, with the recommendation of General Electric (GE) NE-523-A71-0594, “Alternate BWR Feedwater Nozzle Inspection Requirements," to perform periodic ultrasonic testing inspection of critical regions of the BWR feedwater nozzles.

A.1.6 BWR Control Rod Drive Return Line Nozzle

The BWR control rod drive return line nozzle aging management program consists of previously implemented system modifications and inservice inspections that manage the aging effect of cracking in the control rod drive return line nozzles. The control rod drive return line nozzles have been capped. Inservice inspections are performed consistent with ASME Section XI requirements. No augmented inspections in accordance with NUREG-0619, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking,” or the alternative recommendations of GE NE-523-A71-0594, “Alternate BWR Feedwater Nozzle Inspection Requirements," are required. Prior to the period of extended operation the program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda.

A.1.7 BWR Stress Corrosion Cracking

The BWR stress corrosion cracking aging management program to manage intergranular stress corrosion cracking (IGSCC) in boiling water reactor coolant pressure boundary piping four inches and larger nominal pipe size made of stainless steel (SS) is delineated, in part, in NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," Revision 2, BWRVIP 75, “Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules,” and Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-01, “NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping," and its Supplement 1. The program includes (a) replacements and preventive measures to mitigate IGSCC and (b) inspections to monitor IGSCC and its effects. Water chemistry is monitored and maintained in accordance with industry-recognized guidelines in EPRI TR-103515, “BWR Water Chemistry Guidelines.” Prior to the period of extended operation the program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda.

A.1.8 BWR Penetrations

The BWR penetrations aging management program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved Boiling Water Reactor Vessel and Internals Project (BWRVIP)-49, “Instrument Penetration Inspection and Flaw Evaluation Guidelines,” and BWRVIP-27, “BWR Standby Liquid Control System/Core Plate Delta-P Inspection and Flaw Evaluation Guidelines,” documents and (b) monitoring and control of reactor coolant water chemistry in accordance with industry-recognized guidelines of EPRI TR-103515, “BWR Water Chemistry Guidelines,” to ensure the long-term integrity and safe operation of boiling water reactor vessel internal components. Prior to the period of extended operation the program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda.

A.1.9 BWR Vessel Internals

The BWR vessel internals aging management program includes (a) inspection and flaw evaluation in conformance with the guidelines of applicable and staff-approved Boiling Water Reactor Vessel and Internals Project (BWRVIP) documents, and with ASME Section XI; and (b) monitoring and control of reactor coolant water chemistry in accordance with industry-recognized guidelines of EPRI TR-103515, "BWR Water Chemistry Guidelines," to ensure the long-term integrity and safe operation of boiling water reactor vessel internal components. Prior to the period of extended operation the inservice inspection program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda.

A.1.10 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)

The thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) aging management program consists of (1) determination of the susceptibility of cast austenitic stainless steel components to thermal aging embrittlement, (2) accounting for the synergistic effects of thermal aging and neutron irradiation, and (3) implementing a supplemental examination program, as necessary. The program is being implemented prior to the period of extended operation.

A.1.11 Flow-Accelerated Corrosion

The flow-accelerated corrosion aging management program consists of (1) appropriate analysis and baseline inspections, (2) determination of the extent of thinning, and replacement or repair of components, and (3) follow-up inspections to confirm or quantify effects, and to take longer-term corrective actions. This program is in response to NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning." The program relies on implementation of the EPRI NSAC-202L, "Recommendations for an Effective Flow Accelerated Corrosion Program," Revision 2 guidelines. Prior to the period of extended operation the program will be revised to include main steam piping within the scope of license renewal.

A.1.12 Bolting Integrity

This bolting integrity aging management program incorporates industry recommendations of EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," and includes periodic visual inspections for external surface degradation that may be caused by loss of material or cracking of the bolting, or by an adverse environment. Inspection of inservice inspection Class 1, 2, and 3 components is conducted in accordance with ASME Section XI. Prior to the period of extended operation the inservice inspection program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda. In addition, the program will include inspections of bolted joints of diesel generator system components and of components in locations containing high humidity or moisture.

Program activities address the guidance contained in EPRI TR-104213, “Bolted Joint Maintenance and Applications Guide,” but do not specifically identify its use. Non-safety component inspections rely on detection of visible leakage during preventive maintenance and routine observation. The program does not address structural and component support bolting. The aging management of structural bolting is covered by the structures monitoring program. Aging management of ASME Section XI Class 1, 2, and 3 and Class MC support members, including mechanical connections is covered by the “ASME Section XI, Subsection IWF” aging management program.

A.1.13 Open-Cycle Cooling Water System

The open-cycle cooling water system aging management program includes (a) surveillance and control of biofouling, (b) tests to verify heat transfer, (c) a routine inspection and maintenance program, including system flushing and chemical treatment, (d) periodic inspections for leakage, loss of material, and blockage, (e) engineering evaluations and heat sink performance assessments, and (f) assessments of the overall heat sink program. These evaluations and assessments produced specific component and programmatic corrective actions. The program provides assurance that the open-cycle cooling water system is in compliance with General Design Criteria, and with quality assurance requirements, to ensure that the open-cycle cooling water system can be managed for an extended period of operation. This program is in response to and uses the test and inspection guidelines of NRC Generic Letter 89-13, “Service Water System Problems Affecting Safety-Related Equipment.” Prior to the period of extended operation, the scope of the program will be increased to include inspection of additional heat exchangers and sub-components, external surfaces of various submerged pumps and piping, cooling water pump linings, and components in the pump vaults that have a high humidity or moisture environment.

A.1.14 Closed-Cycle Cooling Water System

The closed-cycle cooling water system aging management program relies on preventive measures to minimize corrosion by maintaining inhibitors and by performing non-chemistry monitoring consisting of inspection and nondestructive examinations (NDEs) based on industry-recognized guidelines of EPRI TR-107396, “Closed Cooling Water Chemistry Guidelines,” for closed-cycle cooling water systems. Station maintenance inspections and NDE provide condition monitoring of heat exchangers exposed to closed-cycle cooling water environments. Prior to the period of extended operation, the program will be enhanced to include procedure revisions that provide for monitoring of specific chemistry parameters in order to meet EPRI TR-107396 guidance.

A.1.15 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems

The inspection of overhead heavy load and light load (related to refueling) handling systems aging management program confirms the effectiveness of the maintenance monitoring program and the effects of past and future usage on the structural reliability of cranes and hoists. Administrative controls ensure that only allowable loads are handled, and fatigue failure of structural elements is not expected. A time-limited aging analysis concludes that there are no fatigue concerns for reactor building overhead cranes during the period of extended operation. The bridge, trolley, and other structural components are visually inspected on a routine basis for degradation. These cranes are included in the corporate structural monitoring program (which complies with the 10 CFR 50.65 maintenance rule) and in various station procedures. Prior to the period of extended operation, the program will be enhanced to include inspections for rail wear and proper crane travel on rails, and corrosion of crane structural components.

A.1.16 Compressed Air Monitoring

The compressed air monitoring aging management program consists of inspection, monitoring, and testing of the entire system, including (1) pressure decay testing, visual inspections, and walkdowns of various system locations; and (2) preventive monitoring that checks air quality at various locations in the system to ensure that dewpoint, particulates, and suspended hydrocarbons are kept within the specified limits. This program is consistent with responses to NRC Generic Letter 88-14, “Instrument Air Supply Problems,” and ANSI/ISA-S7.3-1975, "Quality Standard for Instrument Air." Prior to the period of extended operation, the program will be enhanced to include inspections of instrument air distribution piping based on EPRI TR-108147, “Compressor and Instrument Air System Maintenance Guide,” and blowdown of instrument air distribution piping.

A.1.17 BWR Reactor Water Cleanup System

The BWR reactor water cleanup (RWCU) system aging management program monitors and controls reactor water chemistry based on industry-recognized guidelines of EPRI TR-103515, “BWR Water Chemistry Guidelines," to reduce the susceptibility of RWCU piping to stress corrosion cracking (SCC) and intergranular stress corrosion cracking (IGSCC). RWCU system piping has been replaced with piping that is resistant to intergranular stress corrosion cracking, in response to NRC Generic Letter 88-01, “NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping," concerns. In addition, all actions requested in NRC Generic Letter 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance,” have been completed. Therefore, inservice inspection in accordance with ASME Section XI is not required.

A.1.18 Fire Protection

The fire protection aging management program includes a fire barrier inspection program and a diesel-driven fire pump inspection program. The fire barrier inspection program requires periodic visual inspection of fire barrier penetration seals and wraps, fire barrier walls, ceilings, and floors; flood barrier penetration seals that also serve as fire barrier seals; and periodic visual inspection and functional tests of fire rated doors to ensure that their operability is maintained. The program includes surveillance tests of fuel oil systems for the diesel-driven fire pumps and isolation condenser diesel-driven makeup pumps to ensure that the fuel supply lines can perform intended functions. The program also includes visual inspections and periodic operability tests of halon and carbon dioxide fire suppression systems based on NFPA codes.

Prior to the period of extended operation, the program will be revised to include:

- Inspection of oil spill barriers
- Inspection of external surfaces of the halon system and the carbon dioxide system
- Periodic capacity tests of the isolation condenser makeup pumps
- Specific fuel supply leak inspection criteria for fire pumps and isolation condenser makeup pumps during tests
- Specific inspection criteria for fire doors
- Inspection frequencies for fire doors and spill barriers

A.1.19 Fire Water System

The fire water system aging management program provides fire system header and hydrant flushing, system performance (flow and pressure) testing, and inspections, on a periodic basis, and for injection of chemical agents during or subsequent to flushing to minimize biofouling. System performance tests measure hydraulic resistance and compare results with previous testing. This approach eliminates the need for tests at maximum design flow and pressure. Internal inspections are conducted on system components when disassembled to identify evidence of corrosion or biofouling. Fire header pressure is maintained through a crosstie with the service water system. Significant leakage (exceeding the capacity of this line) would be identified by automatic start of the fire pumps, which would initiate immediate investigation and corrective action. Inspection and surveillance testing is performed in accordance with procedures based on applicable NFPA codes. Where code deviations are required or desirable, the intent of the code is maintained by documented technical justifications.

Sprinkler test requirements will be modified prior to the period of extended operation to include sprinkler sampling in accordance with NFPA 25, “Inspection, Testing and Maintenance of Water-Based Fire Protection Systems,” Section 2-3.1. Samples will be submitted to a testing laboratory prior to being in service 50 years. This testing will be repeated at intervals not exceeding 10 years.

Prior to the period of extended operation the program will be revised to include external surface inspections of submerged fire pumps, outdoor hydrants, and outdoor transformer deluge systems; and periodic non-intrusive wall thickness measurements of selected portions of the fire water system at intervals that do not exceed every 10 years.

A.1.20 Aboveground Carbon Steel Tanks

The aboveground carbon steel tanks aging management program manages corrosion of outdoor nitrogen tanks. Paint is a corrosion preventive measure, and periodic visual inspections monitor degradation of the paint and any resulting metal degradation. Carbon steel tanks in the scope of license renewal are above ground and not directly supported by earthen or concrete foundations. Therefore, inspection of the sealant or caulking at the tank-foundation interface, and inspection of inaccessible tank locations and on-grade tank bottoms do not apply. Prior to the period of extended operation the program will be revised to include documentation of results of periodic system engineer walkdowns of the nitrogen tanks.

A.1.21 Fuel Oil Chemistry

The fuel oil chemistry aging management program relies on a combination of surveillance and maintenance procedures. Monitoring and controlling fuel oil contamination maintains the fuel oil quality. Exposure to fuel oil contaminants such as water and microbiological organisms is minimized by routine draining and cleaning of fuel oil tanks, and by fuel oil sampling and analysis, including analysis of new fuel before its introduction into the storage tanks. A biocide is added to the fuel oil storage tanks during each new fuel delivery. Sampling and testing of diesel fuel oil is in accordance with industry-recognized ASTM methods and standards. Emergency diesel generator fuel oil analysis acceptance criteria are contained in the Technical Specifications and are based on industry-recognized ASTM methods and standards.

A.1.22 Reactor Vessel Surveillance

The reactor vessel surveillance aging management program includes periodic testing of metallurgical surveillance samples to monitor the progress of neutron embrittlement of the reactor pressure vessel as a function of neutron fluence, in accordance with Regulatory Guide (RG) 1.99, “Radiation Embrittlement of Reactor Vessel Materials," Revision 2.

Prior to the period of extended operation the program will be consistent with BWRVIP-78, “Integrated Surveillance Program,” and BWRVIP-86, “BWR Integrated Surveillance Program Implementation Plan.” The program will ensure coupon availability during the period of extended operation, and provide for saving withdrawn coupons for future reconstitution.

A.1.23 One-Time Inspection

The one-time inspection aging management program includes inspections of a number of samples of the piping and components listed below. The inspections are scheduled for implementation prior to the period of extended operation to manage aging effects of selected components within the scope of license renewal. The purpose of the inspection is to determine if a specified aging effect is occurring. If the aging effect is occurring, an evaluation is performed to determine the effect it will have on the ability of affected components to perform their intended functions for the period of extended operation, and appropriate corrective action is taken. The program includes the following one-time inspections:

- Inspection of a sample of Class I piping less than four inch nominal pipe size (NPS) exposed to reactor coolant for cracking.
- Inspection of a sample of torus saddle Lubrite baseplates for galvanic corrosion, wear, and lockup to confirm the condition of the inaccessible drywell radial beam Lubrite baseplates.
- Inspection of a sample of spent fuel pool cooling and demineralizer system components for corrosion in stagnant locations to verify effective water chemistry controls.
- Inspection a sample of piping exposed to the containment atmosphere (safety relief valve discharge piping and HPCI turbine exhaust sample locations) for loss of material.
- Inspection of a sample of condensate and torus water components for corrosion in stagnant locations to verify effective water chemistry control.
- Inspection of a sample of compressed gas system piping components for corrosion and a sample of compressed gas system flexible hoses for elastomer degradation.
- Inspection of a sample of lower sections of carbon steel fuel oil and lubricating oil tanks for reduced thickness.
- Inspection of a sample of fuel oil and lubricating oil piping and components for corrosion.
- Inspection of a sample of standby gas treatment and ventilation system components for loss of material.
- Inspection of a sample of stainless steel standby liquid control (SBLC) system components not in the reactor coolant pressure boundary of the SBLC system for cracking, to verify effective water chemistry control.
- Inspection of a sample of HPCI turbine lubricating oil hoses for age-related degradation.

- Inspection of a sample of non-safety related vents and drains including their valves and associated piping, for age-related degradation leading to a loss of structural integrity.

- Inspection of a sample of 10 CFR 54.4(a)(2) components for corrosion for which the component, material, environment, aging effect, or their combination is not specifically identified in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report."

A.1.24 Selective Leaching of Materials

The selective leaching of materials aging management program includes numerous one-time inspections of components of the different susceptible materials selected from each of the applicable environments to determine if loss of material due to selective leaching is occurring. If selective leaching is occurring the program requires evaluation of the effect it will have on the ability of the affected components to perform their intended functions for the period of extended operation, and of the need to expand the test sample. For systems subjected to environments where water is not treated (i.e., the open-cycle cooling water system) the program also follows the guidance of NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

A.1.25 Buried Piping and Tanks Inspection

The buried piping and tanks inspection aging management program includes (1) preventive measures to mitigate corrosion, and (2) periodic inspection to manage the effects of corrosion on the pressure-retaining capacity of buried carbon steel piping and tanks. The program includes the use of piping and component coatings and wrappings, periodic pressure testing, buried tank leakage checks, inspections of buried tank interior surfaces, and inspections of the ground above buried tanks and piping.

Prior to the period of extended operation a one-time visual inspection of the external surface of a buried piping section, a one-time internal ultrasonic inspection of a sampling of the buried steel tanks, and a one-time internal ultrasonic inspection on the bottom of an outdoor aluminum storage tank will be performed.

A.1.26 ASME Section XI, Subsection IWE

The ASME Section XI, Subsection IWE aging management program consists of periodic visual examination for signs of degradation, and limited surface or volumetric examination when augmented examination is required. The program covers steel containment shells and their integral attachments; containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure-retaining bolting. The program includes assessment of damage and corrective actions. The program utilizes an approved relief request that permits utilization of the 1998 Edition of Subsection IWE of ASME Section XI in its entirety instead of the 1992 Edition and Addenda.

A.1.27 ASME Section XI, Subsection IWF

The ASME Section XI, Subsection IWF aging management program consists of periodic visual examination of ASME Section XI Class 1, 2, and 3 component and piping supports for signs of degradation, evaluation, and establishment of corrective actions. The program is in accordance with ASME Section XI, Subsection IWF, 1989 Edition, and Code Case N-491-1. Prior to the period of extended operation the program will include ASME Class MC component supports consistent with NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Chapter III, Section B1.3.

A.1.28 10 CFR Part 50, Appendix J

The 10 CFR Part 50, Appendix J aging management program monitors leakage rates through the containment pressure boundary, including the drywell and torus, penetrations, fittings, and other access openings, in order to detect degradation of containment pressure boundary. Corrective actions are taken if leakage rates exceed acceptance criteria. The Appendix J program also manages changes in material properties of gaskets, o-rings, and packing materials for the containment pressure boundary access points. The containment leak rate tests are performed in accordance with the regulations and guidance provided in 10 CFR 50 Appendix J Option B, Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program," NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J," and ANSI/ANS 56.8, "Containment System Leakage Testing Requirements."

A.1.29 Masonry Wall Program

This masonry wall aging management program consists of inspections, based on IE Bulletin 80-11, "Masonry Wall Design," and plant-specific monitoring proposed by IN 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11," for managing cracking of masonry walls. This program is part of the structures monitoring program.

A.1.30 Structures Monitoring Program

The structures monitoring aging management program includes periodic inspection and monitoring of the condition of structures; supports not included in the "ASME Section XI, Subsection IWF" aging management program; and external surfaces of mechanical and electrical components. The program ensures that aging degradation leading to loss of intended functions will be detected and that the extent of degradation can be determined. This program was developed under 10 CFR 50.65 and is based on NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2 and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2.

Prior to the period of extended operation the program will be revised to include:

- Inspections of structural steel components in secondary containment, flood barriers, electrical panels and racks, junction boxes, instrument panels and racks, and offsite power structural components and their foundations.

- Periodic reviews of chemistry data on below-grade water to confirm that the environment remains non-aggressive for aggressive chemical attack of concrete or corrosion of embedded steel.

- Inspection of a sample of non-insulated indoor piping external surfaces at locations immediately adjacent to periodically inspected piping supports.

- Reference to specific insulation inspection criteria for existing cold weather preparation and inspection procedures for outdoor insulation, and the establishment of new inspections for various indoor area piping and equipment insulation.

- Addition of specific inspection parameters for non-structural joints, roofing, grout pads and isolation gaps.

- Extension of inspection criteria to the structural steel, concrete, masonry walls, equipment foundations, and component support sections of the program.

A.1.31 RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants

The RG 1.127, “Inspection of Water-Control Structures Associated with Nuclear Power Plants,” aging management program consists of inspection and surveillance of structural steel elements (exposed to raw water) and concrete (exposed and not exposed to raw water) that are in the Unit 1 and Unit 2 and 3 crib houses and within the scope of license renewal. The activities are based on Regulatory Guide 1.127, Revision 1, and are part of the structures monitoring program. Prior to the period of extended operation the program will be revised to include monitoring crib house concrete walls and slabs with opposing sides in contact with river water, to emphasize inspection for structural integrity of concrete and steel components, and to identify specific types of components to be inspected.

A.1.32 Protective Coating Monitoring and Maintenance Program

The protective coating monitoring and maintenance aging management program consists of guidance for selection, application, inspection, and maintenance of Service Level I protective coatings. This program is implemented in accordance with Regulatory Guide 1.54, “Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants,” Revision 0, ANSI N101 4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities," and the guidance of EPRI TR-109937, "Guidelines on Nuclear Safety-Related Coating." Prior to the period of extended operation the program will be revised to include thorough visual inspection of Service Level 1 coatings near sumps or screens for the emergency core cooling system, pre-inspection review of previous reports so that trends can be identified, and analysis of suspected causes of any coating failures.

A.1.33 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

The electrical cables and connections not subject to 10 CFR 50.49 environmental qualification requirements aging management program manages aging of cables and connections which might be susceptible to aging during the period of extended operation. A sample of accessible electrical cables and connections installed in adverse localized environments are visually inspected at least once every 10 years for indications of accelerated insulation aging. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for a subject electrical cable or connection. This is a new program initiated prior to the period of extended operation.

A.1.34 Metal Fatigue of Reactor Coolant Pressure Boundary

The metal fatigue and reactor coolant pressure boundary aging management program ensures that the design fatigue usage factor limit will not be exceeded during the period of extended operation. The program will be enhanced prior to the period of extended operation. The enhanced program calculates and tracks cumulative usage factors for bounding locations in the reactor coolant pressure boundary (reactor pressure vessel and Class I piping), containment torus, torus vents, and torus attached piping and penetrations. The program also tracks isolation condenser fatigue stress cycles. The enhanced program uses the EPRI-licensed FatiguePro® cycle counting and fatigue usage factor tracking computer program, which provides for calculation of stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles, and automated calculation and tracking of fatigue cumulative usage factors.

A.1.35 Environmental Qualification (EQ) of Electrical Components

The effects of aging on the intended functions will be adequately managed per the requirements of 10 CFR 54.21 (c)(1)(iii). The existing environmental qualification (EQ) program will manage aging of electrical equipment within the scope of 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," for the period of extended operation. The program establishes, demonstrates, and documents the level of qualification, qualified configurations, maintenance, surveillance and replacements necessary to meet 10 CFR 50.49. A qualified life is determined for equipment within the scope of the program and appropriate actions such as replacement or refurbishment are taken prior to or at the end of the qualified life of the equipment so that the aging limit is not exceeded.

A.2 PLANT-SPECIFIC AGING MANAGEMENT PROGRAMS

A.2.1 Corrective Action Program

The 10 CFR Part 50, Appendix B program provides corrective actions, confirmation processes, and administrative controls for aging management programs for license renewal. Prior to the period of extended operation the scope of the program will be expanded to include non-safety related structures and components that are subject to an aging management review for license renewal. The corrective action program applies to all plant systems, structures and components (both safety related and non-safety related) within the scope of license renewal. Administrative controls are in place for existing aging management programs and activities. Administrative controls will also be applied to new and enhanced programs and activities as they are implemented. As a minimum, these programs and activities are or will be performed in accordance with written procedures that are or will be reviewed and approved in accordance with the Quality Assurance Program.

A.2.2 Periodic Inspection of Non-EQ, Non-Segregated Electrical Bus Ducts

This program inspects the non-segregated bus ducts that connect the reserve auxiliary transformers to the 4160V ESF buses. They are normally energized, and therefore the bus duct insulation material will experience temperature rise due to energization, which may cause age-related degradation during the period of extended operation. These bus ducts are in scope of license renewal but are not subject to 10 CFR 50.49 environmental qualification requirements.

An inspection program will be established prior to the period of extended operation. The program will provide for inspection of the bus ducts. This inspection program considers the technical information and guidance provided in IEEE Standard P1205, “IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations,” SAND 96-0344, “Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations,” and EPRI TR-109619, "Guideline for the Management of Adverse Localized Equipment Environments." Non-segregated bus duct internal components and materials will be inspected for signs of aging degradation that indicate possible loss of insulation function. Repair or rework is initiated as required to maintain the operating functions of the bus ducts.

A.2.3 Periodic Inspection of Ventilation System Elastomers

The periodic inspection of ventilation system elastomers aging management program provides for routine inspections of certain elastomers in the standby gas treatment, reactor building ventilation, station blackout diesel generator building ventilation, and main control room ventilation systems. Prior to the period of extended operation an existing program for inspection of ventilation system elastomers will be enhanced. The program will include inspections for cracking, loss of material, or other evidence of aging of all flexible boots, access door seals and gaskets, and filter seals and gaskets in the components of these systems that are within the scope of license renewal. The scope of inspections will also include RTV silicone used as a duct sealant, in systems within the scope of license renewal.

A.2.4 Periodic Testing of Drywell and Torus Spray Nozzles

Carbon steel piping upstream of the drywell and torus spray nozzles is subject to possible general corrosion. The periodic flow tests of drywell and torus spray nozzles address a concern that rust from the possible general corrosion may plug the spray nozzles. These periodic tests verify that the drywell and torus spray nozzles are free from plugging that could result from corrosion product buildup from upstream sources.

A.2.5 Lubricating Oil Monitoring Activities

The lubricating oil monitoring activities aging management program manages corrosion, loss of material, and cracking in lubricating oil heat exchangers in the scope of license renewal by monitoring physical and chemical properties in lubricating oil. Sampling, testing, and monitoring verify lubricating oil properties. Oil analysis permits identification of specific wear mechanisms, contamination, and oil degradation within operating machinery.

These activities apply to the emergency diesel generator, station blackout diesel generator, and HPCI oil coolers. The complete aging management program for these oil coolers also includes secondary-side (heat sink) chemistry controls, performance monitoring, and inspections. Those portions of the lubricating oil heat exchanger management program are described in:

- Section A.1.14, “Closed-Cycle Cooling Water System,” for the diesel generator and station blackout diesel generator oil coolers; and in
- Section A.2.6, “Heat Exchanger Test and Inspection Activities,” for the HPCI oil coolers.

A.2.6 Heat Exchanger Test and Inspection Activities

The heat exchanger test and inspection activities aging management program provides condition monitoring, inspection, and performance testing to manage loss of material, cracking, and buildup of deposits in heat exchangers in the scope of license renewal, that are not tested and inspected under “Open-Cycle Cooling Water” or “Closed-Cycle Cooling Water” aging management programs. For the isolation condensers this program also includes the augmentation activities identified in NUREG-1801, “Generic Aging Lessons Learned (GALL) Report,” lines IV.C1.4-a and IV.C1.4-b.

These are new activities that will be implemented prior to the period of extended operation.

The isolation condenser test and inspection augmentation activities detect cracking due to stress corrosion cracking or cyclic loading, and detect loss of material due to pitting and crevice corrosion. These are ISI augmentation activities, outside the ISI program, not augmented ISI activities within the ISI program. These augmentation activities verify that significant degradation is not occurring, and therefore that the intended function of the isolation condenser is maintained during the extended period of operation. These augmentation activities consist of temperature and radioactivity monitoring of the shell-side (cooling) water, and eddy current testing of tubes.

These activities include tests, inspections, and monitoring and trending of test results to confirm that aging effects are managed. To ensure that system and component functions are maintained, these components are also being included in the scope of other activities which provide inservice inspection and performance monitoring, and primary and secondary-side (water and oil) chemistry controls.

- Inservice inspection is described in Section A.1.1, “ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD.”
- Management of water chemistry is described in Section A.1.2, “Water Chemistry.”
- Management of the primary, oil side of the HPCI lubricating oil coolers is described in Section A.2.5, “Lubricating Oil Monitoring Activities.”

A.3 TIME-LIMITED AGING ANALYSIS SUMMARIES

In the descriptions of this section, Class I and Class II are the Dresden safety classifications described in UFSAR Section 3.2.

A.3.1 Neutron Embrittlement of the Reactor Vessel and Internals

The ferritic materials of the reactor vessel are subject to embrittlement due to high energy neutron exposure. Reactor vessel neutron embrittlement is a TLAA.

A.3.1.1 Reactor Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement

The reactor vessel end-of-life neutron fluence has been recalculated for a 60-year (54 EFPY) extended licensed operating period.

The 54 EFPY USE was evaluated by an equivalent margin analysis (EMA) using the 54 EFPY calculated fluence and the Dresden surveillance capsule results, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.1.2 Adjusted Reference Temperature for Reactor Vessel Materials Due to Neutron Embrittlement

The reactor vessel materials peak fluence, $\Delta RT_{NDT,}$ and ART values for the 60-year (54 EFPY) license operating period were calculated in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.1.3 Reflood Thermal Shock Analysis of the Reactor Vessel

The effects of a reflood thermal shock described in UFSAR Section 3.9.5.3.3 were examined. An alternative analysis confirms that the effects remain acceptable for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.3.1.4 Reflood Thermal Shock Analysis of the Reactor Vessel Core Shroud and Repair Hardware

Radiation embrittlement may affect the ability of reactor vessel internals, particularly the core shroud and repair hardware, to withstand a low-pressure coolant injection (LPCI) thermal shock transient. Embrittlement effects are evaluated for the maximum-fluence beltline region of the core shroud, where the maximum event strain is about 0.57 percent [UFSAR Section 3.9.5.3.2], and design of the core shroud repair tie rod stabilizer assemblies included an investigation of possible embrittlement effects.

The effects of the increase in neutron fluence with a 54 EFPY life at uprated power were evaluated, and the allowable strain for this faulted event remains a considerable margin above the expected strain.

The core shroud repair tie rod stabilizer assemblies were designed for a 40-year life, which will not be exceeded at the end of the extended licensed operating period.

The existing analyses of the effects of embrittlement in the internals have been evaluated and remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.3.1.5 Reactor Vessel Thermal Limit Analyses: Operating Pressure – Temperature Limits

Revised pressure-temperature (P-T) limits for a 60-year licensed operating life will be prepared using available Dresden capsule data and submitted to the NRC for approval prior to the start of the extended period of operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.1.6 Reactor Vessel Circumferential Weld Examination Relief

Relief has been granted from the requirements for inspection of RPV circumferential welds for the remainder of the current 40-year licensed operating period. The justification for relief is consistent with Boiling Water Reactor Vessel and Internals Program BWRVIP-05, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations," guidelines. Application for an extension of this relief for the 60-year period of extended operation will be submitted prior to the end of the current operating license term.

The procedures and training that will be used to limit the frequency of cold over-pressure events to the number specified in the SER for the RPV circumferential weld relief request extension, during the license renewal term, are the same as those approved for use in the current period.

The analyses associated with reactor vessel circumferential weld examination relief will be projected to the end of the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.1.7 Reactor Vessel Axial Weld Failure Probability

BWRVIP-05, "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations," estimated the 40-year end-of-life failure probability of a limiting reactor vessel axial weld, showed that it was orders of magnitude greater than the 40-year end-of-life circumferential weld failure probability, and used this analysis to justify relief from inspection of the circumferential welds, as described in Section A.3.1.6 above.

The re-evaluation of the axial weld failure probability for 60 years depends on vessel ΔRT_{NDT} calculations. The NRC staff review and the NRC staff and BWRVIP calculations of the test-case failure probabilities assume that 90 percent of axial welds will be inspected. At Dresden, less than 90 percent of axial welds can be inspected. As such, an analysis was performed for 54 EFPY to assess the effect on the probability of fracture due to the actual inspection performed on the vessel axial welds and to determine if the coverage was sufficient in the inspection of regions contributing to the majority of the risk.

The evaluation shows that the calculated unit-specific axial weld conditional failure probabilities at 54 EFPY for Dresden are less than the failure probabilities calculated by the NRC staff in the NRC BWRVIP-05 SER at 64 EFPY and the limiting Clinton values found in Table 3 of the SER supplement. The projected probability of failure of an axial weld at Dresden will therefore provide adequate margin above the probability of failure of a circumferential weld, in support of relief from inspection of circumferential welds, for the extended licensed operating period, in accordance with the requirements of 10 CFR 54.21 (c)(1)(ii).

A.3.2 Metal Fatigue

The thermal and mechanical fatigue analyses of mechanical components have been identified as TLAAs for Dresden. Specific components have been designed considering transient cycle assumptions, as listed in vendor specifications and the Dresden UFSAR.

A.3.2.1 Reactor Vessel Fatigue Analyses

Unit 2 and Unit 3 reactor vessel fatigue analyses depend on cycle count assumptions that assume a 40-year operating period. The effects of fatigue in the reactor vessel will be managed for the period of extended operation by the fatigue management program for cycle counting and fatigue usage factor tracking, as described in Section A.1.34.

This aging management program will ensure that fatigue effects in vessel pressure boundary components will be adequately managed and will be maintained within code design limits for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.3.2.2 Fatigue Analysis of Reactor Vessel Internals

A.3.2.2.1 High-Cycle Flow-Induced Vibration Fatigue Analysis of Jet Pump Riser Braces

The original design addressed high-cycle fatigue in the internals. Except for fatigue in the Dresden Unit 2 jet pump riser braces, the original evaluation of specific components such as the core plate, top guide, jet pump assemblies, fuel supports, or control rod drive assemblies used a displacement criterion, which is not time-limited.

The Dresden Unit 2 riser braces will be repaired or replaced prior to the period of extended operation and will be qualified for the extended licensed operating period, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.2.3 Reactor Coolant Pressure Boundary Piping and Component Fatigue Analysis

A.3.2.3.1 ASME Section III Class 1 Reactor Coolant Pressure Boundary Piping and Component Fatigue Analysis

Other than special cases under the Mark I containment "New Loads" program, the only piping which has received a fatigue analysis is the Dresden Unit 3 recirculation piping replaced under the IGSCC mitigation program, including some connected shutdown cooling, low pressure coolant injection, isolation condenser, and reactor water cleanup piping.

The effects of fatigue in Class I primary system piping, including the piping analyzed to ASME Section III Class 1 criteria, will be managed for the period of extended operation by the fatigue management cycle counting and fatigue usage factor tracking program, as part of the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program described in Section A.1.34. The fatigue management cycle counting and fatigue usage tracking program will apply to piping whose calculated usage factor exceeds 0.4. This aging management activity will ensure that fatigue effects in pressure boundary components will be adequately managed and will be maintained within code design limits for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.3.2.3.2 Reactor Coolant Pressure Boundary Piping and Components Designed to USAS B31.1, ASME Section III Class 2 and 3, or ASME Section VIII Class B and C

Except for the Dresden Unit 3 recirculation piping described in A.3.2.3.1, all other primary system or reactor coolant pressure boundary (RCPB) piping systems were designed to USAS B31.1, 1967 Edition, as were the safety relief valve (SRV) discharge lines inside the drywell. Neither the USAS B31.1 piping design nor the additional nuclear code and code case rules applied to this piping invoke a fatigue analysis, but USAS B31.1 does apply a stress range reduction factor based on an assumed finite number of equivalent full-range thermal cycles for the design life. The B31.1 designs are therefore TLAAs because they are part of the current licensing basis, are used to support a safety determination, and depend on a specific number of cycles which might change with a change in licensed operating life.

The assumed number of design lifetime equivalent full-range thermal cycles determines the allowable stress range (the stress range reduction factor) for design of all Class I and Class II USAS B31.1 or ASME Class 2 or 3 piping. With the exception of containment vent and process bellows, no components in the scope of license renewal designed to ASME Section III or Section VIII require design for cyclic thermal loading.

The number of thermal cycles assumed for design of Class I and II piping has been evaluated and the existing stress range reduction factor remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.2.3.3 Fatigue Analysis of the Isolation Condenser

The isolation condensers and the supporting system piping and components were specified for 250 shutdown depressurization cycles and for 250 thermal shock events in 40 years.

A review of isolation condenser operations since 1990 and a conservative estimate of earlier condenser operations based on number of unit scrams concluded that the projected total cycle count for 60 years is well below the number of design cycles.

The analyses of the effects of thermal cycle and thermal shock events on the Dresden isolation condenser systems and components have been evaluated and remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.3.2.4 Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)

Generic Safety Issue (GSI) 190 was identified by the NRC because of concerns about potential effects of reactor water environments on component fatigue life during the period of extended operation.

Exelon will perform plant-specific calculations for the applicable locations identified in NUREG/CR 6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," for older-vintage BWR plants, to assess the potential effects of reactor coolant on component fatigue life in accordance with 10 CFR 54.21(c)(1)(ii). The calculations of current and projected cumulative usage factors (CUFs) under this program will include appropriate environmental fatigue effect (F_{EN}) factors. Appropriate corrective action will be taken if the resulting projected end-of-life CUF values exceed 1.0.

Exelon reserves the right to modify this position in the future based on the results of industry activities currently underway, or based on other results of improvements in methodology, subject to NRC approval prior to changes in this position.

A.3.3 Environmental Qualification of Electrical Equipment

Electrical equipment included in the Dresden Environmental Qualification Program which has a specified qualified life of at least 40 years involves time-limited aging analyses for license renewal. The aging effects of this equipment will be managed in the Environmental Qualification Program discussed in Section A.1.35, “Environmental Qualification (EQ) of Electrical Components,” in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.3.4 Containment Fatigue

The Dresden Mark I containments were originally designed to stress limit criteria without fatigue analyses. However, the discovery of significant hydrodynamic loads (“new loads”) caused by safety relief valve (SRV) and small, intermediate, and design basis pipe break discharges into the suppression pool required the reanalysis of the suppression chamber, vents, and attached piping and internal structures, including some fatigue analyses at limiting locations. These fatigue analyses of the suppression chamber, and its internals, and vents in each unit include assumed pressure, temperature, seismic, and SRV cycles, and combinations thereof. The scope of the analyses included the suppression chamber, the drywell-to-suppression chamber vents, SRV discharge piping, other piping attached to the suppression chamber and its penetrations, and the drywell-to-suppression chamber vent bellows.

A.3.4.1 Fatigue Analysis of the Suppression Chamber, Vents, and Downcomers

For low cumulative usage factor (CUF) locations (40-year CUF < 0.4) the Dresden new loads analyses of each suppression chamber and its associated vents and downcomers have been evaluated and remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

For higher cumulative usage factor locations in the analyses of the suppression chamber and suppression chamber vents and downcomers (40-year CUF ≥ 0.4) the effects of fatigue will be managed for the period of extended operation by the fatigue management cycle counting and fatigue usage factor tracking program, as described in Section A.1.34.

The fatigue management activities will ensure that fatigue effects in containment pressure boundary components are adequately managed and are maintained within code design limits for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.3.4.2 Fatigue Analysis of SRV Discharge Piping Inside the Suppression Chamber, External Suppression Chamber Attached Piping, and Associated Penetrations

SRV discharge lines and external suppression chamber attached piping and associated penetrations were analyzed separately from the suppression chamber, vents and downcomers.

The disposition of these analyses is the same as described for the suppression chamber, vents and downcomers in Section A.3.4.1 above.

A.3.4.3 Drywell-to-Suppression Chamber Vent Line Bellows Fatigue Analyses

A fatigue analysis of the drywell-to-suppression chamber vent line bellows was performed assuming 150 thermal and internal pressure load cycles for the 40-year life of the plant. The drywell-to-suppression chamber vent line bellows have a rated capacity of 1,000 cycles at maximum displacement.

The Dresden new loads fatigue analysis of the drywell-to-suppression chamber vent line bellows have been evaluated and remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.4.4 Primary Containment Process Penetrations Bellows Fatigue Analysis

The only containment process piping expansion joints subject to significant thermal expansion and contraction are those between the drywell shell penetrations and process piping. These are designed for a stated number of operating and thermal cycles.

The thermal cycle designs of Dresden containment process penetration bellows have been evaluated and remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.5 Other Plant-Specific TLAAs

A.3.5.1 Reactor Building Crane Load Cycles

The reactor building overhead cranes in Dresden were designed to meet or exceed the design criteria of the Crane Manufacturers Association of America (CMAA) Specification 70, "Specifications for Electric Overhead Traveling Cranes," Class A1. These cranes are capable of a minimum of 100,000 cycles at the full rated load of 125 tons. Correspondence with the NRC stated that over their 40-year life these cranes would most probably see fewer than 5,000 cycles at a maximum of 100 tons and a larger number of cycles at significantly less than 100 tons.

The load cycle designs of Dresden reactor building cranes have been evaluated and remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.5.2 Metal Corrosion Allowances

A.3.5.2.1 Degradation Rates of Inaccessible Exterior Drywell Plate Surfaces

In its response to Generic Letter 87-05, "Request for Additional Information Assessment of Licensee Measures to Mitigate and/ or Identify Degradation of Mark I Drywells," Commonwealth Edison evaluated the potential effects of corrosion on exterior drywell steel surfaces in the "sand pockets" of Dresden Unit 3 and found that 27 years of service remained before corrosion at the assumed rate would have a significant adverse effect on design basis stresses. The evaluation concluded that the findings were applicable to Dresden Unit 2 and Quad Cities Units 1 and 2 as well.

The calculation will be revised for the realistic environment and for a full 60-year design life, in accordance with 10 CFR 54.21(c)(1)(ii). A UT inspection will validate assumptions used in the calculation. These actions will be completed before the period of extended operation.

A.3.5.2.2 Galvanic Corrosion in the Containment Shell and Attached Piping Components due to Stainless Steel ECCS Suction Strainers

The Dresden ECCS suction strainers have been replaced with larger strainers. The replacement strainers are stainless steel. The modification included drilling new bolt holes and enlarging the existing bolt holes in each of the existing carbon steel strainer support flanges to provide sufficient bolting for the larger replacement strainers. The holes in the carbon steel flanges are not coated to protect them from corrosion. The calculation of corrosion effects assumes a corrosion allowance of 4 mils/year and assumes a design life of 33 years, which is just short of the 60-year extended operating period.

The corrosion rate assumptions used in the calculation will be confirmed by an ultrasonic inspection prior to the period of extended operation. Based on the results of the inspection, a revised galvanic corrosion calculation will be performed to validate acceptable wall thickness to the end of the 60-year licensed operating period, in accordance with 10 CFR 54.21(c)(1)(ii).

A.3.5.3 Crack Growth Calculation of a Postulated Flaw in the Heat Affected Zone of an Arc Strike in the Suppression Chamber Shell

The Dresden Unit 3 torus contains an area that was damaged by an arc strike. The flaw was ground smooth and evaluated by a calculation. This calculation showed that the Dresden flaw was bounded by a Quad Cities Unit 2 arc strike, and therefore by its analysis. The Quad Cities analysis included a crack growth calculation. A further evaluation was performed in 1997 and it was determined that the flaw depth of the arc strike at Dresden was not of sufficient depth to warrant any final repairs.

The supporting Quad Cities crack growth calculation has been evaluated and remains valid for Dresden for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.5.4 Radiation Degradation of Drywell Shell Expansion Gap Polyurethane Foam

The steel drywell shell is largely enclosed within the structural and shielding concrete of the reactor containment building. To accommodate thermal expansion, compressible foam was used to form an expansion gap between the concrete and the drywell shell. A confirming analysis contained in the UFSAR evaluates the increase in external compressive loads on the drywell exterior, due to additional compression of this foam, for accident-condition thermal expansion of the drywell. The load depends on the stress-strain curve of the foam, and the validity of this confirming analysis of the Dresden drywells therefore depends on the stiffness of the polyurethane foam. The analysis would require validation if the foam became stiffer (higher compressive stress for the same strain) as a result of increased radiation exposure from extended plant operation.

The expected radiation exposure of the foam has been evaluated and remains below the significant damage threshold at the end of the period of extended operation. The evaluation of thermal expansion compressive loads therefore also remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.6 References for Section A.3

1. *Dresden Nuclear Power Station Units 2 and 3, Quad Cities Nuclear Power Station Units 1 and 2, License Renewal Project, TLAA Technical Report.* Revision 0, June 2002. Prepared by Parsons Energy and Chemicals, Inc. for the General Electric Company.

2. *Dresden Nuclear Power Station Units 2 and 3, Quad Cities Nuclear Power Station Units 1 and 2, License Renewal Project, Potential TLAA Review Results Package.* Revision 0, June 2002. Prepared by Parsons Energy and Chemicals, Inc. for the General Electric Company.

Quad Cities Units 1 and 2

Updated Final Safety Analysis Supplement

A.1 AGING MANAGEMENT PROGRAMS

A.1.1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program consists of periodic volumetric, surface, and visual examinations of components for assessment, identification of signs of degradation, and establishment of corrective actions. Prior to the period of extended operation the program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda.

A.1.2 Water Chemistry

The water chemistry aging management program consists of monitoring and control of water chemistry to keep peak levels of various contaminants below system-specific limits based on industry-recognized guidelines of EPRI TR-103515, "BWR Water Chemistry Guidelines." To mitigate aging effects on component surfaces that are exposed to water as process fluid, the chemistry programs are used to control water chemistry for impurities (e.g., chlorides, and sulfates) that accelerate corrosion.

A.1.3 Reactor Head Closure Studs

The reactor head closure studs aging management program includes inservice inspection (ISI). This program also includes preventive actions and inspection techniques for BWRs. Prior to the period of extended operation the program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda. The reactor head studs are not metal-plated, and have had manganese phosphate coatings applied.

A.1.4 BWR Vessel ID Attachment Welds

The BWR vessel ID attachment welds aging management program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved Boiling Water Reactor Vessel and Internals Project (BWRVIP)-48, "Vessel ID Attachment Weld Inspection and Evaluation Guidelines," and/or ASME Section XI; and (b) monitoring and control of reactor coolant water chemistry in accordance with industry-recognized guidelines of EPRI TR-103515, "BWR Water Chemistry Guidelines." Prior to the period of extended operation the program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda.

A.1.5 BWR Feedwater Nozzle

The BWR feedwater nozzle aging management program includes enhancing the inservice inspections (ISI) specified in the ASME Code, Section XI, with the recommendation of General Electric (GE) NE-523-A71-0594, "Alternate BWR Feedwater Nozzle Inspection Requirements," to perform periodic ultrasonic testing inspection of critical regions of the BWR feedwater nozzles.

A.1.6 BWR Control Rod Drive Return Line Nozzle

The BWR control rod drive return line nozzle aging management program consists of previously implemented system modifications and inservice inspections that manage the aging effect of cracking in the control rod drive return line nozzles. The control rod drive return line nozzles have been capped. Inservice inspections are performed consistent with ASME Section XI requirements. No augmented inspections in accordance with NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," or the alternative recommendations of GE NE-523-A71-0594, "Alternate BWR Feedwater Nozzle Inspection Requirements," are required. Prior to the period of extended operation the program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda.

A.1.7 BWR Stress Corrosion Cracking

The BWR stress corrosion cracking aging management program to manage intergranular stress corrosion cracking (IGSCC) in boiling water reactor coolant pressure boundary piping four inches and larger nominal pipe size made of stainless steel (SS) is delineated, in part, in NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," Revision 2, BWRVIP 75, "Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules," and Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping", and its Supplement 1. The program includes (a) replacements and preventive measures to mitigate IGSCC and (b) inspections to monitor IGSCC and its effects. Water chemistry is monitored and maintained in accordance with industry-recognized guidelines in EPRI TR-103515, "BWR Water Chemistry Guidelines." Prior to the period of extended operation the program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda.

A.1.8 BWR Penetrations

The BWR penetrations aging management program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved Boiling Water Reactor Vessel and Internals Project (BWRVIP)-49, "Instrument Penetration Inspection and Flaw Evaluation Guidelines," and BWRVIP-27, "BWR Standby Liquid Control System/Core Plate Delta-P Inspection and Flaw Evaluation Guidelines," documents and (b) monitoring and control of reactor coolant water chemistry in accordance with industry-recognized guidelines of EPRI TR-103515, "BWR Water Chemistry Guidelines," to ensure the long-term integrity and safe operation of boiling water reactor vessel internal components. Prior to the period of extended operation the program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda.

A.1.9 BWR Vessel Internals

The BWR vessel internals aging management program includes (a) inspection and flaw evaluation in conformance with the guidelines of applicable and staff-approved Boiling Water Reactor Vessel and Internals Project (BWRVIP) documents, and with ASME Section XI; and (b) monitoring and control of reactor coolant water chemistry in accordance with industry-recognized guidelines of EPRI TR-103515, “BWR Water Chemistry Guidelines,” to ensure the long-term integrity and safe operation of boiling water reactor vessel internal components. Prior to the period of extended operation the inservice inspection program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda.

A.1.10 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)

The thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) aging management program consists of (1) determination of the susceptibility of cast austenitic stainless steel components to thermal aging embrittlement, (2) accounting for the synergistic effects of thermal aging and neutron irradiation, and (3) implementing a supplemental examination program, as necessary. The program is being implemented prior to the period of extended operation.

A.1.11 Flow-Accelerated Corrosion

The flow-accelerated corrosion aging management program consists of (1) appropriate analysis and baseline inspections, (2) determination of the extent of thinning, and replacement or repair of components, and (3) follow-up inspections to confirm or quantify effects, and to take longer-term corrective actions. This program is in response to NRC Generic Letter 89-08, “Erosion/Corrosion-Induced Pipe Wall Thinning.” The program relies on implementation of the EPRI NSAC-202L, “Recommendations for an Effective Flow Accelerated Corrosion Program,” Revision 2 guidelines. Prior to the period of extended operation the program will be revised to include main steam piping within the scope of license renewal.

A.1.12 Bolting Integrity

This bolting integrity aging management program incorporates industry recommendations of EPRI NP-5769, “Degradation and Failure of Bolting in Nuclear Power Plants,” and includes periodic visual inspections for external surface degradation that may be caused by loss of material or cracking of the bolting, or by an adverse environment. Inspection of inservice inspection Class 1, 2, and 3 components is conducted in accordance with ASME Section XI. Prior to the period of extended operation the inservice inspection program will be revised to be consistent with ASME Section XI, 1995 Edition through the 1996 Addenda. In addition, the program will include inspections of bolted joints of diesel generator system components and of components in locations containing high humidity or moisture.

Program activities address the guidance contained in EPRI TR-104213, "Bolted Joint Maintenance and Applications Guide," but do not specifically identify its use. Non-safety component inspections rely on detection of visible leakage during preventive maintenance and routine observation. The program does not address structural and component support bolting. The aging management of structural bolting is covered by the structures monitoring program. Aging management of ASME Section XI Class 1, 2, and 3 and Class MC support members, including mechanical connections, is covered by the "ASME Section XI, Subsection IWF" aging management program.

A.1.13 Open-Cycle Cooling Water System

The open-cycle cooling water system aging management program includes (a) surveillance and control of biofouling, (b) tests to verify heat transfer, (c) a routine inspection and maintenance program, including system flushing and chemical treatment, (d) periodic inspections for leakage, loss of material, and blockage, (e) engineering evaluations and heat sink performance assessments, and (f) assessments of the overall heat sink program. These evaluations and assessments produced specific component and programmatic corrective actions. The program provides assurance that the open-cycle cooling water system is in compliance with General Design Criteria, and with quality assurance requirements, to ensure that the open-cycle cooling water system can be managed for an extended period of operation. This program is in response to and uses the test and inspection guidelines of NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment." Prior to the period of extended operation, the scope of the program will be increased to include inspection of additional heat exchangers and sub-components, external surfaces of various submerged pumps and piping, cooling water pump linings, and components in the pump vaults that have a high humidity or moisture environment.

A.1.14 Closed-Cycle Cooling Water System

The closed-cycle cooling water system aging management program relies on preventive measures to minimize corrosion by maintaining inhibitors and by performing non-chemistry monitoring consisting of inspection and nondestructive examinations (NDEs) based on industry-recognized guidelines of EPRI TR-107396, "Closed Cooling Water Chemistry Guidelines," for closed-cycle cooling water systems. Station maintenance inspections and NDE provide condition monitoring of heat exchangers exposed to closed-cycle cooling water environments. Prior to the period of extended operation, the program will be enhanced to include procedure revisions that provide for monitoring of specific chemistry parameters in order to meet EPRI TR-107396 guidance.

A.1.15 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems

The inspection of overhead heavy load and light load (related to refueling) handling systems aging management program confirms the effectiveness of the maintenance monitoring program and the effects of past and future usage on the structural reliability of cranes and hoists. Administrative controls ensure that only allowable loads are handled, and fatigue failure of structural elements is not expected. A time-limited aging analysis concludes that there are no fatigue concerns for reactor building overhead cranes during the period of extended operation. The bridge, trolley, and other structural components are visually inspected on a routine basis for degradation. These cranes are included in the corporate structural monitoring program (which complies with the 10 CFR 50.65 maintenance rule) and in various station procedures. Prior to the period of extended operation, the program will be enhanced to include inspections for rail wear and proper crane travel on rails, and corrosion of crane structural components.

A.1.16 Compressed Air Monitoring

The compressed air monitoring aging management program consists of inspection, monitoring, and testing of the entire system, including (1) pressure decay testing, visual inspections, and walkdowns of various system locations; and (2) preventive monitoring that checks air quality at various locations in the system to ensure that dewpoint, particulates, and suspended hydrocarbons are kept within the specified limits. This program is consistent with responses to NRC Generic Letter 88-14, “Instrument Air Supply Problems,” and ANSI/ISA-S7.3-1975, "Quality Standard for Instrument Air." Prior to the period of extended operation, the program will be enhanced to include inspections of instrument air distribution piping based on EPRI TR-108147, “Compressor and Instrument Air System Maintenance Guide.”

A.1.17 BWR Reactor Water Cleanup System

The BWR reactor water cleanup (RWCU) system aging management program monitors and controls reactor water chemistry based on industry-recognized guidelines of EPRI TR-103515, “BWR Water Chemistry Guidelines," to reduce the susceptibility of RWCU piping to stress corrosion cracking (SCC) and intergranular stress corrosion cracking (IGSCC). RWCU system piping has been replaced with piping that is resistant to intergranular stress corrosion cracking, in response to NRC Generic Letter 88-01, “NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping," concerns. In addition, all actions requested in NRC Generic Letter 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance,” have been completed. Therefore, inservice inspection in accordance with ASME Section XI is not required.

A.1.18 Fire Protection

The fire protection aging management program includes a fire barrier inspection program and a diesel-driven fire pump inspection program. The fire barrier inspection program requires periodic visual inspection of fire barrier penetration seals and wraps, fire barrier walls, ceilings, and floors; flood barrier penetration seals that also serve as fire barrier seals; and periodic visual inspection and functional tests of fire rated doors to ensure that their operability is maintained. The program includes surveillance tests of fuel oil systems for the diesel-driven fire pumps to ensure that the fuel supply line can perform intended functions. The program also includes visual inspections and periodic operability tests of the carbon dioxide fire suppression system based on NFPA codes.

Prior to the period of extended operation, the program will be revised to include:

- Inspection of oil spill barriers
- Inspection of external surfaces of the carbon dioxide systems
- Specific fuel supply leak inspection criteria for fire pumps
- Specific inspection criteria for fire doors

A.1.19 Fire Water System

The fire water system aging management program provides fire system header and hydrant flushing, system performance (flow and pressure) testing, and inspections, on a periodic basis; and for injection of chemical agents during or subsequent to flushing to minimize biofouling. System performance tests measure hydraulic resistance and compare results with previous testing. This approach eliminates the need for tests at maximum design flow and pressure. Internal inspections are conducted on system components when disassembled to identify evidence of corrosion or biofouling. Fire header pressure is maintained through a crosstie with the service water system. Significant leakage (exceeding the capacity of this line) would be identified by automatic start of the fire pumps, which would initiate immediate investigation and corrective action. Inspection and surveillance testing is performed in accordance with procedures based on applicable NFPA codes. Where code deviations are required or desirable, the intent of the code is maintained by technical justifications.

Sprinkler test requirements will be modified prior to the period of extended operation to include sprinkler sampling in accordance with NFPA 25, “Inspection, Testing and Maintenance of Water-Based Fire Protection Systems,” Section 2-3.1. Samples will be submitted to a testing laboratory prior to being in service 50 years. This testing will be repeated at intervals not exceeding 10 years.

Prior to the period of extended operation the program will be revised to include external surface inspections of submerged fire pumps, outdoor hydrants, and outdoor transformer deluge systems; and periodic non-intrusive wall thickness measurements of selected portions of the fire water system at intervals that do not exceed every 10 years.

A.1.20 Aboveground Carbon Steel Tanks

The aboveground carbon steel tanks aging management program manages corrosion of outdoor nitrogen tanks. Paint is a corrosion preventive measure, and periodic visual inspections monitor degradation of the paint and any resulting metal degradation. Carbon steel tanks in the scope of license renewal are above ground and not directly supported by earthen or concrete foundations. Therefore, inspection of the sealant or caulking at the tank-foundation interface, and inspection of inaccessible tank locations and on-grade tank bottoms do not apply. Prior to the period of extended operation the program will be revised to include documentation of results of periodic system engineer walkdowns of the nitrogen tanks.

A.1.21 Fuel Oil Chemistry

The fuel oil chemistry aging management program relies on a combination of surveillance and maintenance procedures. Monitoring and controlling fuel oil contamination maintains the fuel oil quality. Exposure to fuel oil contaminants such as water and microbiological organisms is minimized by routine draining and cleaning of fuel oil tanks, and by fuel oil sampling and analysis, including analysis of new oil before its introduction into the storage tanks. A biocide is added to the fuel oil storage tanks during each new fuel delivery. Sampling and testing of fuel oil is in accordance with industry-recognized ASTM methods and standards. Emergency diesel generator fuel oil analysis acceptance criteria are contained in the Technical Specifications and are based on industry-recognized ASTM methods and standards.

A.1.22 Reactor Vessel Surveillance

The reactor vessel surveillance aging management program includes periodic testing of metallurgical surveillance samples to monitor the progress of neutron embrittlement of the reactor pressure vessel as a function of neutron fluence, in accordance with Regulatory Guide (RG) 1.99, “Radiation Embrittlement of Reactor Vessel Materials," Revision 2.

Prior to the period of extended operation the program will be consistent with BWRVIP-78, “Integrated Surveillance Program,” and BWRVIP-86, “BWR Integrated Surveillance Program Implementation Plan.” The program will ensure coupon availability during the period of extended operation, and provide for saving withdrawn coupons for future reconstitution.

A.1.23 One-Time Inspection

The one-time inspection aging management program includes inspections of a number of samples of the piping and components listed below. The inspections are scheduled for implementation prior to the period of extended operation to manage aging effects of selected components within the scope of license renewal. The purpose of the inspection is to determine if a specified aging effect is occurring. If the aging effect is occurring, an evaluation is performed to determine the effect it will have on the ability of affected components to perform their intended functions for the period of extended operation, and appropriate corrective action is taken. The program includes the following one-time inspections:

- Inspection of a sample of Class I piping less than four inch nominal pipe size (NPS) exposed to reactor coolant for cracking.
- Inspection of a sample of torus saddle Lubrite baseplates for galvanic corrosion, wear, and lockup to confirm the condition of the inaccessible drywell radial beam Lubrite baseplates.

- Inspection a sample of piping exposed to the containment atmosphere (safety relief valve discharge piping and HPCI turbine exhaust sample locations) for loss of material.

- Inspection of a sample of condensate and torus water components for corrosion in stagnant locations to verify effective water chemistry control.

- Inspection of a sample of compressed gas system piping components for corrosion and a sample of compressed gas system flexible hoses for elastomer degradation.

- Inspection of a sample of lower sections of carbon steel fuel oil and lubricating oil tanks for reduced thickness.

- Inspection of a sample of fuel oil and lubricating oil piping and components for corrosion.

- Inspection of a sample of standby gas treatment and ventilation system components for loss of material.

- Inspection of a sample of stainless steel standby liquid control (SBLC) system components not in the reactor coolant pressure boundary of the SBLC system for cracking, to verify effective water chemistry control.

- Inspection of a sample of HPCI turbine lubricating oil hoses for age-related degradation.

- Inspection of a sample of non-safety-related vents and drains including their valves and associated piping, for age-related degradation leading to a loss of structural integrity.

- Inspection of a sample of 10 CFR 54.4(a)(2) components for corrosion for which the component, material, environment, aging effect, or their combination is not specifically identified in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report."

A.1.24 Selective Leaching of Materials

The selective leaching of materials aging management program includes numerous one-time inspections of components of the different susceptible materials selected from each of the applicable environments to determine if loss of material due to selective leaching is occurring. If selective leaching is occurring the program requires evaluation of the effect it will have on the ability of the affected components to perform their intended functions for the period of extended operation, and of the need to expand the test sample. For systems subjected to environments where water is not treated (i.e., the open-cycle cooling water system) the program also follows the guidance of NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

A.1.25 Buried Piping and Tanks Inspection

The buried piping and tanks inspection aging management program includes (1) preventive measures to mitigate corrosion, and (2) periodic inspection to manage the effects of corrosion on the pressure-retaining capacity of buried carbon steel piping and tanks. The program includes the use of piping and component coatings and wrappings, periodic pressure testing, buried tank leakage checks, inspections of buried tank interior surfaces, and inspections of the ground above buried tanks and piping.

Prior to the period of extended operation a one-time visual inspection of the external surface of a buried piping section, a one-time internal ultrasonic inspection of a sampling of the buried steel tanks, and a one-time internal ultrasonic inspection on the bottom of an outdoor aluminum storage tank will be performed.

A.1.26 ASME Section XI, Subsection IWE

The ASME Section XI, Subsection IWE aging management program consists of periodic visual examination for signs of degradation, and limited surface or volumetric examination when augmented examination is required. The program covers steel containment shells and their integral attachments; containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure-retaining bolting. The program includes assessment of damage and corrective actions. The program complies with ASME Section XI Subsection IWE for steel containments (Class MC), 1992 Edition including 1992 Addenda.

A.1.27 ASME Section XI, Subsection IWF

The ASME Section XI, Subsection IWF aging management program consists of periodic visual examination of ASME Section XI Class 1, 2, and 3 component and piping supports for signs of degradation, evaluation, and establishment of corrective actions. The program is in accordance with ASME Section XI, Subsection IWF, 1989 Edition, and Code Case N-491-1. Prior to the period of extended operation the program will include ASME Class MC component supports consistent with NUREG-1801, “Generic Aging Lessons Learned (GALL) Report,” Chapter III, Section B1.3.

A.1.28 10 CFR Part 50, Appendix J

The 10 CFR Part 50, Appendix J aging management program monitors leakage rates through the containment pressure boundary, including the drywell and torus, penetrations, fittings, and other access openings; in order to detect degradation of containment pressure boundary. Corrective actions are taken if leakage rates exceed acceptance criteria. The Appendix J program also manages changes in material properties of gaskets, o-rings, and packing materials for the containment pressure boundary access points. The containment leak rate tests are performed in accordance with the regulations and guidance provided in 10 CFR 50 Appendix J Option B, Regulatory Guide 1.163, “Performance-Based Containment Leak-Testing Program," NEI 94-01, “Industry Guideline for Implementing Performance-Based Option of 10 CFR

Part 50 Appendix J," and ANSI/ANS 56.8, "Containment System Leakage Testing Requirements."

A.1.29 Masonry Wall Program

This masonry wall aging management program consists of inspections, based on IE Bulletin 80-11, "Masonry Wall Design," and plant-specific monitoring proposed by IN 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11," for managing cracking of masonry walls. This program is part of the structures monitoring program.

A.1.30 Structures Monitoring Program

The structures monitoring aging management program includes periodic inspection and monitoring of the condition of structures; supports not included in the "ASME Section XI, Subsection IWF" aging management program; and external surfaces of mechanical and electrical components. The program ensures that aging degradation leading to loss of intended functions will be detected and that the extent of degradation can be determined. This program was developed under 10 CFR 50.65 and is based on NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2 and Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 2.

Prior to the period of extended operation the program will be revised to include:

- Inspections of structural steel components in secondary containment, flood barriers, electrical panels and racks, junction boxes, instrument panels and racks, offsite power structural components and their foundations, and the discharge canal weir as part of the ultimate heat sink.

- Periodic reviews of chemistry data on below-grade water to confirm that the environment remains non-aggressive for aggressive chemical attack of concrete or corrosion of embedded steel.

- Inspection of a sample of non-insulated indoor piping external surfaces at locations immediately adjacent to periodically inspected piping supports.

- Reference to specific insulation inspection criteria for existing cold weather preparation and inspection procedures for outdoor insulation, and the establishment of new inspections for various indoor area piping and equipment insulation.

- Addition of specific inspection parameters for non-structural joints, roofing, grout pads and isolation gaps.

- Extension of inspection criteria to the structural steel, concrete, masonry walls, equipment foundations, and component support sections of the program.

A.1.31 RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants

The RG 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants," aging management program consists of inspection and surveillance of structural steel elements (exposed to raw water) and concrete (exposed and not exposed to raw water) that are in the crib house and discharge canal weir structure supporting the ultimate heat sink and within the scope of license renewal. The activities are based on Regulatory Guide 1.127, Revision 1, and are part of the structures monitoring program. Prior to the period of extended operation the program will be revised to include monitoring crib house concrete walls and slabs with opposing sides in contact with river water, and the discharge canal weir supporting the ultimate heat sink; to emphasize inspection for structural integrity of concrete and steel components; and to identify specific types of components to be inspected.

A.1.32 Protective Coating Monitoring and Maintenance Program

The protective coating monitoring and maintenance aging management program consists of guidance for selection, application, inspection, and maintenance of Service Level I protective coatings. This program is implemented in accordance with Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants," Revision 0, ANSI N101 4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities," and the guidance of EPRI TR-109937, "Guidelines on Nuclear Safety-Related Coating." Prior to the period of extended operation the program will be revised to include thorough visual inspection of Service Level 1 coatings near sumps or screens for the emergency core cooling system, pre-inspection review of previous reports so that trends can be identified, and analysis of suspected causes of any coating failures.

A.1.33 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

The electrical cables and connections not subject to 10 CFR 50.49 environmental qualification requirements aging management program manages aging of cables and connections which might be susceptible to aging during the period of extended operation. A sample of accessible electrical cables and connections installed in adverse localized environments are visually inspected at least once every 10 years for indications of accelerated insulation aging. An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for a subject electrical cable or connection. This is a new program initiated prior to the period of extended operation.

A.1.34 Metal Fatigue of Reactor Coolant Pressure Boundary

The metal fatigue and reactor coolant pressure boundary aging management program ensures that the design fatigue usage factor limit will not be exceeded during the period of extended operation. The program will be enhanced prior to the period of extended operation. The enhanced program calculates and tracks cumulative usage factors for bounding locations in the reactor coolant pressure boundary (reactor pressure vessel and Class I piping), containment torus, torus vents, and torus attached piping and penetrations. The enhanced program uses the EPRI-licensed FatiguePro® cycle counting and fatigue usage factor tracking computer program, which provides for calculation of stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles, and automated calculation and tracking of fatigue cumulative usage factors.

A.1.35 Environmental Qualification (EQ) of Electrical Components

The effects of aging on the intended functions will be adequately managed per the requirements of 10 CFR 54.21 (c)(1)(iii). The existing environmental qualification (EQ) program will manage aging of electrical equipment within the scope of 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” for the period of extended operation. The program establishes, demonstrates, and documents the level of qualification, qualified configurations, maintenance, surveillance and replacements necessary to meet 10 CFR 50.49. A qualified life is determined for equipment within the scope of the program and appropriate actions such as replacement or refurbishment are taken prior to or at the end of the qualified life of the equipment so that the aging limit is not exceeded.

A.1.36 Boraflex Monitoring

The Boraflex monitoring aging management program consists of (1) neutron attenuation testing (“blackness testing”) to determine gap formation, (2) sampling for the presence of silica in the spent fuel pool along with boron loss, and (3) analysis of criticality to assure that the required 5% subcriticality margin is maintained. This program is implemented in response to Generic Letter 96-04, “Boraflex Degradation in Spent Fuel Pool Storage Racks.” The Boraflex monitoring activities are based on the maintenance rule and on EPRI TR-108761, "A Synopsis of the Technology Developed to Address the Boraflex Degradation Issue."

A.2 PLANT-SPECIFIC AGING MANAGEMENT PROGRAMS

A.2.1 Corrective Action Program

The 10 CFR Part 50, Appendix B program provides corrective actions, confirmation processes, and administrative controls for aging management programs for license renewal. Prior to the period of extended operation the scope of the program will be expanded to include non-safety-related structures and components that are subject to an aging management review for license renewal. The corrective action program applies to all plant systems, structures and components (both safety-related and non-safety-related) within the scope of license renewal. Administrative controls are in place for existing aging management programs and activities. Administrative controls will also be applied to new and enhanced programs and activities as they are implemented. As a minimum, these programs and activities are or will be performed in accordance with written procedures that are or will be reviewed and approved in accordance with the Quality Assurance Program.

A.2.2 Periodic Inspection of Non-EQ, Non-Segregated Electrical Bus Ducts

This program inspects the non-segregated bus ducts that connect the reserve auxiliary transformers to the 4160V ESF buses. They are normally energized, and therefore the bus duct insulation material will experience temperature rise due to energization, which may cause age-related degradation during the period of extended operation. These bus ducts are in scope of license renewal but are not subject to 10 CFR 50.49 environmental qualification requirements

An inspection program will be established prior to the period of extended operation. The program will provide for inspection of the bus ducts. This inspection program considers the technical information and guidance provided in IEEE Standard P1205, “IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations,” SAND 96-0344, “Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations,” and EPRI TR-109619, "Guideline for the Management of Adverse Localized Equipment Environments." Non-segregated bus duct internal components and materials will be inspected for signs of aging degradation that indicate possible loss of insulation function. Repair or rework is initiated as required to maintain the operating functions of the bus ducts.

A.2.3 Periodic Inspection of Ventilation System Elastomers

The periodic inspection of ventilation system elastomers aging management program provides for routine inspections of certain elastomers in the standby gas treatment, reactor building ventilation, emergency diesel generator building ventilation, station blackout diesel generator building ventilation, and main control room ventilation systems. Prior to the period of extended operation an existing program for inspection of ventilation system elastomers will be enhanced. The program will include inspections for cracking, loss of material, or other evidence of aging of all flexible boots, access door seals and gaskets, and filter seals and gaskets in the components of these systems that are within the scope of license renewal. The scope of inspections will also include RTV silicone used as a duct sealant, in systems within the scope of license renewal.

A.2.4 Periodic Testing of Drywell and Torus Spray Nozzles

Carbon steel piping upstream of the drywell and torus spray nozzles is subject to possible general corrosion. The periodic flow tests of drywell and torus spray nozzles address a concern that rust from the possible general corrosion may plug the spray nozzles. These periodic tests verify that the drywell and torus spray nozzles are free from plugging that could result from corrosion product buildup from upstream sources.

A.2.5 Lubricating Oil Monitoring Activities

The lubricating oil monitoring activities aging management program manages corrosion, loss of material, and cracking in lubricating oil heat exchangers in the scope of license renewal by monitoring physical and chemical properties in lubricating oil. Sampling, testing, and trending verify lubricating oil properties. Oil analysis permits identification of specific wear mechanisms, contamination, and oil degradation within operating machinery.

These activities apply to the emergency diesel generator, station blackout diesel generator, and HPCI oil coolers. The complete aging management program for these oil coolers also includes secondary-side (heat sink) chemistry controls, performance monitoring, and inspections. Those portions of the lubricating oil heat exchanger management program are described in:

- Section A.1.14, “Closed-Cycle Cooling Water System,” for the diesel generator and station blackout diesel generator oil coolers; and in
- Section A.2.6, “Heat Exchanger Test and Inspection Activities,” for the HPCI oil coolers.

A.2.6 Heat Exchanger Test and Inspection Activities

The heat exchanger test and inspection activities aging management program provides condition monitoring, inspection, and performance testing to manage loss of material, cracking, and buildup of deposits in heat exchangers in the scope of license renewal, that are not tested and inspected under “Open-Cycle Cooling Water” or “Closed-Cycle Cooling Water” aging management programs.

These are new activities that will be implemented prior to the period of extended operation.

These activities include tests, inspections, and monitoring and trending of test results to confirm that aging effects are managed. To ensure that system and component functions are maintained, these components are also being included in the scope of other activities which provide inservice inspection and performance monitoring, and primary and secondary-side (water and oil) chemistry controls.

- Management of water chemistry is described in Section A.1.2, “Water Chemistry.”

- Management of the primary, oil side of the HPCI lubricating oil coolers is described in Section A.2.5, “Lubricating Oil Monitoring Activities.”

A.2.7 Generator Stator Water Chemistry Activities

The generator stator water chemistry activities aging management program manages loss of material and cracking aging effects by monitoring and controlling water chemistry. Generator stator water chemistry control maintains high purity water in accordance with General Electric guidelines for stator cooling water systems. Generator stator water is continuously monitored for conductivity and an alarm annunciates if conductivity increases to a predetermined limit.

A.3 TIME-LIMITED AGING ANALYSIS SUMMARIES

In the descriptions of this section, Class I and Class II are the Quad Cities safety classifications described in UFSAR Section 3.2.

A.3.1 Neutron Embrittlement of the Reactor Vessel and Internals

The ferritic materials of the reactor vessel are subject to embrittlement due to high energy neutron exposure. Reactor vessel neutron embrittlement is a TLAA.

A.3.1.1 Reactor Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement

The reactor vessel end-of-life neutron fluence has been recalculated for a 60-year (54 EFPY) extended licensed operating period.

The 54 EFPY USE was evaluated by an equivalent margin analysis (EMA) using the 54 EFPY calculated fluence and the Quad Cities surveillance capsule results in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.1.2 Adjusted Reference Temperature for Reactor Vessel Materials Due to Neutron Embrittlement

The reactor vessel materials peak fluence, $\Delta RT_{NDT,}$ and ART values for the 60-year (54 EFPY) license operating period were calculated in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.1.3 Reflood Thermal Shock Analysis of the Reactor Vessel

The effects of a reflood thermal shock described in UFSAR Section 3.9.5.3.3 were examined. An alternative analysis confirms that the effects remain acceptable for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.3.1.4 Reflood Thermal Shock Analysis of the Reactor Vessel Core Shroud and Repair Hardware

Radiation embrittlement may affect the ability of reactor vessel internals, particularly the core shroud and repair hardware, to withstand a low-pressure coolant injection (LPCI) thermal shock transient. Embrittlement effects are evaluated for the maximum-fluence beltline region of the core shroud, where the maximum event strain is about 0.57 percent [UFSAR Section 3.9.5.3.2], and design of the core shroud repair tie rod stabilizer assemblies included an investigation of possible embrittlement effects.

The effects of the increase in neutron fluence with a 54 EFPY life at uprated power were evaluated, and the allowable strain for this faulted event remains a considerable margin above the expected strain.

The core shroud repair tie rod stabilizer assemblies were designed for a 40-year life, which will not be exceeded at the end of the extended licensed operating period.

The existing analyses of the effects of embrittlement in the internals have been evaluated and remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.3.1.5 Reactor Vessel Thermal Limit Analyses: Operating Pressure – Temperature Limits

Revised pressure-temperature (P-T) limits for a 60-year licensed operating life will be prepared using available Quad Cities capsule data and submitted to the NRC for approval prior to the start of the extended period of operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(ii).

A.3.2 Metal Fatigue

The thermal and mechanical fatigue analyses of mechanical components have been identified as TLAAs for Quad Cities. Specific components have been designed considering transient cycle assumptions, as listed in vendor specifications and the Quad Cities UFSAR.

A.3.2.1 Reactor Vessel Fatigue

Unit 1 and Unit 2 reactor vessel fatigue analyses depend on cycle count assumptions that assume a 40-year operating period. The effects of fatigue in the reactor vessel will be managed for the period of extended operation by the fatigue management program for cycle counting and fatigue usage factor tracking, as described in Section A.1.34.

This aging management program will ensure that fatigue effects in vessel pressure boundary components will be adequately managed and will be maintained within code design limits for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.3.2.2 Fatigue Analysis of Reactor Vessel Internals

A.3.2.2.1 Low-Cycle Thermal Fatigue Analysis of the Core Shroud and Repair Hardware

Only one Quad Cities analysis of low-cycle fatigue in RPV internals exists: the evaluation of a standard design for repair of the core shroud. This analysis is a TLAA. The calculated fatigue effects are not significant.

The fatigue analysis of the core shroud repair has been evaluated and remains valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

A.3.2.3 Reactor Coolant Pressure Boundary Piping and Component Fatigue Analysis

A.3.2.3.1 Reactor Coolant Pressure Boundary Piping and Components Designed to USAS B31.1, ASME Section III Class 2 and 3, or ASME Section VIII Class B and C

All primary system and other reactor coolant pressure boundary (RCPB) piping systems were designed to USAS B31.1, 1967 Edition, as were the safety relief valve (SRV) discharge lines inside the drywell. The USAS B31.1 piping design does not invoke a fatigue analysis, but USAS B31.1 does apply a stress range reduction factor based on an assumed finite number of equivalent full-range thermal cycles for the design life. The B31.1 designs are therefore TLAAs because they are part of the current licensing basis, are used to support a safety determination, and depend on a specific number of cycles which might change with a change in licensed operating life.

The assumed number of design lifetime equivalent full-range thermal cycles determines the allowable stress range (the stress range reduction factor) for design of all Class I and Class II USAS B31.1 or ASME Class 2 or 3 piping. With the exception of containment vent and process bellows, no components in the scope of license renewal designed to ASME Section III or Section VIII require design for cyclic thermal loading.

The number of thermal cycles assumed for design of Class I and II piping has been evaluated and the existing stress range reduction factor remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.2.4 Effects of Reactor Coolant Environment on Fatigue Life of Components and Piping (Generic Safety Issue 190)

Generic Safety Issue (GSI) 190 was identified by the NRC because of concerns about potential effects of reactor water environments on component fatigue life during the period of extended operation.

Exelon will perform plant-specific calculations for the applicable locations identified in NUREG/CR 6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," for older-vintage BWR plants, to assess the potential effects of reactor coolant on component fatigue life in accordance with 10 CFR 54.21(c)(1)(ii). The calculations of current and projected cumulative usage factors (CUFs) under this program will include appropriate environmental fatigue effect (F_{EN}) factors. Appropriate corrective action will be taken if the resulting projected end-of-life CUF values exceed 1.0.

Exelon reserves the right to modify this position in the future based on the results of industry activities currently underway, or based on other results of improvements in methodology, subject to NRC approval prior to changes in this position.

A.3.3 Environmental Qualification of Electrical Equipment

Electrical equipment included in the Quad Cities Environmental Qualification Program which has a specified qualified life of at least 40 years involves time-limited aging analyses for license renewal. The aging effects of this equipment will be managed in the

Environmental Qualification Program discussed in Section A.1.35, "Environmental Qualification (EQ) of Electrical Components," in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.3.4 Containment Fatigue

The Quad Cities Mark I containments were originally designed to stress limit criteria without fatigue analyses. However, the discovery of significant hydrodynamic loads ("new loads") caused by safety relief valve (SRV) and small, intermediate, and design basis pipe break discharges into the suppression pool required the reanalysis of the suppression chamber, vents, and attached piping and internal structures, including some fatigue analyses at limiting locations. These fatigue analyses of the suppression chamber, and its internals, and vents in each unit include assumed pressure, temperature, seismic, and SRV cycles, and combinations thereof. The scope of the analyses included the suppression chamber, the drywell-to-suppression chamber vents, SRV discharge piping, other piping attached to the suppression chamber and its penetrations, and the drywell-to-suppression chamber vent bellows.

A.3.4.1 Fatigue Analysis of the Suppression Chamber, Vents, and Downcomers

For low cumulative usage factor (CUF) locations (40-year CUF < 0.4) the Quad Cities new loads analyses of each suppression chamber and its associated vents and downcomers have been evaluated and remain valid for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(i).

For higher cumulative usage factor locations in the analyses of the suppression chamber and suppression chamber vents and downcomers (40-year CUF $\geq$ 0.4) the effects of fatigue will be managed for the period of extended operation by the fatigue management cycle counting and fatigue usage factor tracking program, as described in Section A.1.34.

The fatigue management activities will ensure that fatigue effects in containment pressure boundary components are adequately managed and are maintained within code design limits for the period of extended operation, in accordance with the requirements of 10 CFR 54.21(c)(1)(iii).

A.3.4.2 Fatigue Analysis of SRV Discharge Piping Inside the Suppression Chamber, External Suppression Chamber Attached Piping, and Associated Penetrations

SRV discharge lines and external suppression chamber attached piping and associated penetrations were analyzed separately from the suppression chamber, vents and downcomers. The disposition of these analyses is the same as described for the suppression chamber, vents and downcomers in Section A.3.4.1 above.

A.3.4.3 Drywell-to-Suppression Chamber Vent Line Bellows Fatigue Analyses

A fatigue analysis of the drywell-to-suppression chamber vent line bellows was performed assuming 150 thermal and internal pressure load cycles for the 40-year life of the plant. The drywell-to-suppression chamber vent line bellows have a rated capacity of 1,000 cycles at maximum displacement.

The Quad Cities new loads fatigue analysis of the drywell-to-suppression chamber vent line bellows have been evaluated and remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.4.4 Primary Containment Process Penetrations Bellows Fatigue Analysis

The only containment process piping expansion joints subject to significant thermal expansion and contraction are those between the drywell shell penetrations and process piping. These are designed for a stated number of operating and thermal cycles.

The thermal cycle designs of Quad Cities containment process penetration bellows have been evaluated and remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.5 Other Plant-Specific TLAAs

A.3.5.1 Reactor Building Crane Load Cycles

The reactor building overhead cranes in Quad Cities were designed to meet or exceed the design criteria of the Crane Manufacturers Association of America (CMAA) Specification 70, “Specifications for Electric Overhead Traveling Cranes,” Class A1. These cranes are capable of a minimum of 100,000 cycles at the full rated load of 125 tons. Correspondence with the NRC stated that over their 40-year life these cranes would most probably see fewer than 5,000 cycles at a maximum of 100 tons, and a larger number of cycles at significantly less than 100 tons.

The load cycle designs of Quad Cities reactor building cranes have been evaluated and remain valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.5.2 Metal Corrosion Allowances

A.3.5.2.1 Corrosion Allowance for Power Operated Relief Valves

GE specification 25A5508, “Relief Valve, Power Operated,” for the Quad Cities Unit 2 replacement PORVs prescribes a corrosion allowance of 0.002 inches for stainless steel and 0.120 inches for carbon steel for a design life of 40 years. The specification is cited in Quad Cities UFSAR Section 5.2.2.

The corrosion allowance for the Quad Cities Unit 2 replacement PORVs has been evaluated and remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.5.2.2 Degradation Rates of Inaccessible Exterior Drywell Plate Surfaces

In its response to Generic Letter 87-05, “Request for Additional Information Assessment of Licensee Measures to Mitigate and/ or Identify Degradation of Mark I Drywells,” Commonwealth Edison evaluated the potential effects of corrosion on exterior drywell steel surfaces in the "sand pockets" of Dresden Unit 3 drywell and found that 27 years of service remained before corrosion at the assumed rate would have a significant adverse effect on design basis stresses. The evaluation concluded that the findings were applicable to Dresden Unit 2 and Quad Cities Units 1 and 2 as well.

The calculation will be revised for the realistic environment and for a full 60-year design life, in accordance with 10 CFR 54.21(c)(1)(ii). A UT inspection will validate assumptions used in the calculation. These actions will be completed before the period of extended operation.

A.3.5.2.3 Galvanic Corrosion in the Containment Shell and Attached Piping Components due to Stainless Steel ECCS Suction Strainers

The Quad Cities ECCS suction strainers have been replaced with larger strainers. The replacement strainers are stainless steel. The modification included drilling new bolt holes and enlarging the existing bolt holes in each of the existing carbon steel strainer support flanges to provide sufficient bolting for the larger replacement strainers. The holes in the carbon steel flanges are not coated to protect them from corrosion. The calculation of corrosion effects assumes a corrosion allowance of 4 mils/year and assumes a design life of 33 years, which is just short of the 60-year extended operating period.

The corrosion rate assumptions used in the calculation will be confirmed by an ultrasonic inspection prior to the period of extended operation. Based on the results of the inspection, a revised galvanic corrosion calculation will be performed to validate acceptable wall thickness to the end of the 60-year licensed operating period, in accordance with 10 CFR 54.21(c)(1)(ii).

A.3.5.3 Crack Growth Calculation of a Postulated Flaw in the Heat Affected Zone of an Arc Strike in the Suppression Chamber Shell

A calculation provides technical justification for continued operation of the Quad Cities Unit 2 torus which was damaged by an arc strike. The flaw has been ground smooth and NDE tested. It was initially assumed the damaged area would be repaired after two fuel cycles of operation. This time limit has been extended with appropriate NDE being performed to assure no cracks or other linear flaws exist in the affected area.

The crack growth calculation has been evaluated and remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.5.4 Radiation Degradation of Drywell Shell Expansion Gap Polyurethane Foam

The steel drywell shell is largely enclosed within the structural and shielding concrete of the reactor containment building. To accommodate thermal expansion, compressible foam was used to form an expansion gap between the concrete and the drywell shell. A confirming analysis contained in the UFSAR evaluates the increase in external compressive loads on the drywell exterior, due to additional compression of this foam, for accident-condition thermal expansion of the drywell. The load depends on the stress-strain curve of the foam, and the validity of this confirming analysis of the Quad Cities drywells therefore depends on the stiffness of the polyurethane foam. The analysis would require validation if the foam became stiffer (higher compressive stress for the same strain) as a result of increased radiation exposure from extended plant operation.

The expected radiation exposure of the foam has been evaluated and remains below the significant damage threshold at the end of the period of extended operation. The evaluation of thermal expansion compressive loads therefore also remains valid for the period of extended operation, in accordance with 10 CFR 54.21(c)(1)(i).

A.3.6 References for Section A.3

1. *Dresden Nuclear Power Station Units 2 and 3, Quad Cities Nuclear Power Station Units 1 and 2, License Renewal Project, TLAA Technical Report.* Revision 0, June 2002. Prepared by Parsons Energy and Chemicals, Inc. for the General Electric Company.

2. *Dresden Nuclear Power Station Units 2 and 3, Quad Cities Nuclear Power Station Units 1 and 2, License Renewal Project, Potential TLAA Review Results Package.* Revision 0, June 2002. Prepared by Parsons Energy and Chemicals, Inc. for the General Electric Company.

APPENDIX B
AGING MANAGEMENT PROGRAMS

List of Tables

APPENDIX B AGING MANAGEMENT PROGRAMS

Introduction

Section 3 of the LRA identifies the aging management program activities credited for managing aging effects. Section 4 of the LRA identifies the aging management program activities credited as a result of time-limited aging analyses. Appendix B describes the aging management program activities identified in Sections 3 and 4 of the LRA.

Table B-1 provides a list of the aging management programs evaluated in Chapters X and XI of NUREG-1801, “Generic Aging Lessons Learned (GALL) Report,” April 2001, and the corresponding Dresden and Quad Cities programs credited for aging management in this application.

Section B.1 provides descriptions of program activities and the results of reviews to determine their consistency with the aging management programs evaluated in NUREG-1801. A summary of enhancements to the Dresden and Quad Cities programs as well as exceptions to specific elements of the NUREG-1801 aging management programs are presented. In addition, plant-specific operating experience associated with the program activities is described.

Section B.2 provides evaluations of plant-specific aging management programs.

The aging management program activities are presented under the following headings:

B.1 AGING MANAGEMENT PROGRAMS EVALUATED IN NUREG-1801

B.2 PLANT-SPECIFIC AGING MANAGEMENT PROGRAMS

Table B-1 NUREG-1801 and Dresden and Quad Cities Aging Management Program Matrix

NUREG-1801 Program	Dresden and Quad Cities Programs	Appendix B Subsection
NUREG-1801 Chapter X		
X.M1 Metal Fatigue of Reactor Coolant Pressure Boundary	Metal Fatigue of Reactor Coolant Pressure Boundary	B.1.34
X.S1 Concrete Containment Tendon Prestress	Not Credited for License Renewal	
X.E1 Environmental Qualification (EQ) of Electrical Components	Environmental Qualification (EQ) of Electrical Components	B.1.35
NUREG-1801 Chapter XI		
XI.M1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	B.1.1
XI.M2 Water Chemistry	Water Chemistry	B.1.2
XI.M3 Reactor Head Closure Studs	Reactor Head Closure Studs	B.1.3
XI.M4 BWR Vessel ID Attachment Welds	BWR Vessel ID Attachment Welds	B.1.4
XI.M5 BWR Feedwater Nozzle	BWR Feedwater Nozzle	B.1.5
XI.M6 BWR Control Rod Drive Return Line Nozzle	BWR Control Rod Drive Return Line Nozzle	B.1.6
XI.M7 BWR Stress Corrosion Cracking	BWR Stress Corrosion Cracking	B.1.7
XI.M8 BWR Penetrations	BWR Penetrations	B.1.8
XI.M9 BWR Vessel Internals	BWR Vessel Internals	B.1.9
XI.M10 Boric Acid Corrosion	Not Credited for License Renewal	

Table B-1 NUREG-1801 and Dresden and Quad Cities Aging Management Program Matrix (Continued)

NUREG-1801 Program	Dresden and Quad Cities Programs	Appendix B Subsection
XI.M11 Nickel-Alloy Nozzles and Penetrations	Not Credited for License Renewal	
XI.M12 Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)	Not Credited for License Renewal	
XI.M13 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)	B.1.10
XI.M14 Loose Part Monitoring	Not Credited for License Renewal	
XI.M15 Neutron Noise Monitoring	Not Credited for License Renewal	
XI.M16 PWR Vessel Internals	Not Credited for License Renewal	
XI.M17 Flow-Accelerated Corrosion	Flow-Accelerated Corrosion	B.1.11
XI.M18 Bolting Integrity	Bolting Integrity	B.1.12
XI.M19 Steam Generator Tube Integrity	Not Credited for License Renewal	
XI.M20 Open-Cycle Cooling Water System	Open-Cycle Cooling Water System	B.1.13
XI.M21 Closed-Cycle Cooling Water System	Closed-Cycle Cooling Water System	B.1.14
XI.M22 Boraflex Monitoring	Boraflex Monitoring	B.1.36
XI.M23 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	B.1.15
XI.M24 Compressed Air Monitoring	Compressed Air Monitoring	B.1.16

Table B-1 NUREG-1801 and Dresden and Quad Cities Aging Management Program Matrix (Continued)

NUREG-1801 Program	Dresden and Quad Cities Programs	Appendix B Subsection
XI.M25 BWR Reactor Water Cleanup System	BWR Reactor Water Cleanup System	B.1.17
XI.M26 Fire Protection	Fire Protection	B.1.18
XI.M27 Fire Water System	Fire Water System	B.1.19
XI.M28 Buried Piping and Tanks Surveillance	Not Credited for License Renewal	
XI.M29 Aboveground Carbon Steel Tanks	Aboveground Carbon Steel Tanks	B.1.20
XI.M30 Fuel Oil Chemistry	Fuel Oil Chemistry	B.1.21
XI.M31 Reactor Vessel Surveillance	Reactor Vessel Surveillance	B.1.22
XI.M32 One-Time Inspection	One-Time Inspection	B.1.23
XI.M33 Selective Leaching of Materials	Selective Leaching of Materials	B.1.24
XI.M34 Buried Piping and Tanks Inspection	Buried Piping and Tanks Inspection	B.1.25
XI.S1 ASME Section XI, Subsection IWE	ASME Section XI, Subsection IWE	B.1.26
XI.S2 ASME Section XI, Subsection IWL	Not Credited for License Renewal	
XI.S3 ASME Section XI, Subsection IWF	ASME Section XI, Subsection IWF	B.1.27
XI.S4 10 CFR Part 50, Appendix J	10 CFR Part 50, Appendix J	B.1.28

Table B-1 NUREG-1801 and Dresden and Quad Cities Aging Management Program Matrix (Continued)

NUREG-1801 Program		Dresden and Quad Cities Programs	Appendix B Subsection
XI.S5	Masonry Wall Program	Masonry Wall Program	B.1.29
XI.S6	Structures Monitoring Program	Structures Monitoring Program	B.1.30
XI.S7	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants	RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants	B.1.31
XI.S8	Protective Coating Monitoring and Maintenance Program	Protective Coating Monitoring and Maintenance Program	B.1.32
XI.E1	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	B.1.33
XI.E2	Electrical Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Not Credited for License Renewal	
XI.E3	Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Not Credited for License Renewal	

B.1 AGING MANAGEMENT PROGRAMS EVALUATED IN NUREG-1801

B.1.1 ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD

Description

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program is part of the inservice inspection (ISI) program and provides for condition monitoring of reactor coolant pressure retaining piping and components within the scope of license renewal, except for the reactor pressure vessel. It also provides for condition monitoring of reactor internal components within the scope of license renewal, and the Dresden isolation condenser. The program is implemented through procedures that require examinations consistent with ASME Section XI and for Quad Cities an approved relief request PR-02.

The program includes:

- Cracking monitoring for susceptible inservice inspection Class 1 components subject to a steam or reactor water environment, through volumetric examinations of pressure retaining welds and their heat affected zones in piping components.
- Cracking monitoring of the Quad Cities reactor vessel flange leak detection line through monitoring for leaks during reactor vessel flood-up in accordance with relief request PR-02.
- Loss of fracture toughness monitoring for susceptible inservice inspection Class 1 components in reactor recirculation and reactor water cleanup systems through visual inspections of reactor recirculation and reactor water cleanup valves and reactor recirculation pumps for signs of this aging effect.
- Cracking monitoring for susceptible reactor internal components subject to a reactor water environment, through surface and volumetric examinations.
- Cracking monitoring of the Dresden isolation condenser through surface and volumetric examinations of pressure retaining nozzle welds and their heat affected zones that are subject to a steam or reactor water environment.

- Loss of material monitoring of portions of the Dresden isolation condenser subject to a steam or reactor water environment, through system pressure tests.

NUREG-1801 Consistency

With enhancement the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program is consistent with the ten elements of aging management program XI.M1, "ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that the aging of the isolation condenser is to be managed by ASME Section XI Inservice Inspection (ISI) Subsection IWB (for Class 1 components). However, the Dresden isolation condenser is ISI Class 2 on the tube side and ISI Class 3 on the shell side. Therefore, Subsections IWC and IWD are used, as Class 1 requirements do not apply.

Enhancements

- NUREG-1801 indicates this program is to use the 1995 Edition through the 1996 Addenda of ASME Section XI. The current Code of record for Dresden and Quad Cities is the 1989 Edition of ASME Section XI. The program will be revised to be consistent with the requirements of the 1995 Edition through the 1996 Addenda of ASME Section XI.

The enhancement is scheduled for implementation prior to the period of extended operation.

Operating Experience

Dresden and Quad Cities have both successfully identified indications of age-related degradation prior to the loss of the intended functions of the components, and have taken appropriate corrective actions through evaluation, repair or replacement of the components in accordance with ASME Section XI and station implementing procedures.

Conclusion

The ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD aging management program provides reasonable assurance that aging effects are adequately managed so that the intended functions of components within the scope of license renewal that are covered by this program are maintained during the period of extended operation.

B.1.2 Water Chemistry

Description

The water chemistry aging management program activities consist of measures that are used to manage aging of components exposed to reactor water, condensate and feedwater, control rod drive water, demineralized water storage tank water, condensate tank water, torus water (pressure suppression pool), and spent fuel pool water. The program activities provide for monitoring and controlling of water chemistry using station procedures and processes based on EPRI TR-103515, “BWR Water Chemistry Guidelines,” 2000 Revision for the prevention or mitigation of loss of material and cracking aging effects.

Control of water chemistry does not preclude loss of material due to general, crevice or pitting corrosion at locations of stagnant flow. The One-Time Inspection (B.1.23) aging management program includes provisions specified by NUREG-1801 for verification of chemistry control and confirmation of the absence of loss of material in stagnant flow areas in piping systems and components.

NUREG-1801 Consistency

With enhancements the water chemistry aging management program is consistent with the ten elements of aging management program XI.M2, "Water Chemistry," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that water chemistry control is in accordance with BWRVIP-29 for water chemistry in BWRs. BWRVIP-29 references the 1993 revision of EPRI TR-103515, "BWR Water Chemistry Guidelines." The Dresden and Quad Cities water chemistry programs are based on EPRI TR-103515-R2, which is the 2000 Revision of "BWR Water Chemistry Guidelines." EPRI periodically updates the water chemistry guidelines, as new information becomes available. EPRI TR-103515-R2 incorporates new information to develop proactive plant-specific water chemistry programs to minimize IGSCC.

NUREG-1801 indicates that hydrogen peroxide is monitored to mitigate degradation of structural materials. The Dresden and Quad Cities programs do not monitor for hydrogen peroxide because the rapid decomposition of hydrogen peroxide makes reliable data exceptionally difficult to obtain and EPRI TR-103515 Section 4.3.3, "Water Chemistry Guidelines for Power Operation," does not address monitoring for hydrogen peroxide.

NUREG-1801 indicates that dissolved oxygen is monitored. Consistent with the guidance provided in EPRI TR-103515, condensate storage tank, demineralized water storage tank water and torus (pressure suppression pool) water is not sampled for dissolved oxygen.

NUREG-1801 indicates that water quality (pH and conductivity) is maintained in accordance with established guidance. However, per EPRI TR-103515-R2, "BWR Water Chemistry Guidelines," Section 5.2.1.11, pH measurement accuracy in most BWR streams is generally suspect because of the dependence of the instrument reading on ionic strength of the sample solution. In addition, the monitoring of pH is not discussed in EPRI TR-103515-R2, Appendix B for condensate storage tank, demineralized water storage tank, or torus (pressure suppression pool) water. Therefore, pH is not monitored in condensate storage tank, demineralized water storage tank, or torus (pressure suppression pool) water.

Aging of SBLC system components not in the reactor coolant pressure boundary section of SBLC system relies on monitoring and control of SBLC makeup water chemistry. The makeup water is monitored in lieu of the storage tank, because the sodium pentaborate that is maintained in the storage tank would mask most of the chemistry parameters monitored. The effectiveness of the water chemistry program will be verified by a one-time VT-3 inspection of a Dresden SBLC pump discharge valve casing and a Quad Cities SBLC pump casing as discussed in the One-Time Inspection (B.1.23) aging management program.

Enhancements

- Procedures will be revised to provide for increased sampling to verify corrective actions taken to address abnormal chemistry conditions.
- The Quad Cities procedure for turbine building sample panel collection will be revised to assure maintenance of the chemistry integrity of samples.

Enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

Appropriate guidance for maintaining the contaminants below specific limits is provided. Periodic self-assessments of the water chemistry activities have been and continue to be performed to identify areas that need improvement to maintain the quality performance of the activity, such as improved condensate demineralizer element precoating to minimize feedwater iron excursions.

The water chemistry program has identified instances where parameters were outside the established specifications. Increased sampling and actions to bring the parameters back into specification were initiated. Additionally, previous inspections of test coupons have demonstrated the effectiveness of the chemistry program applied to the spent fuel pool in managing the aging of aluminum and stainless steel in this environment.

Conclusion

The water chemistry aging management program covers components within the scope of license renewal that are exposed to reactor water, condensate, feedwater, control rod drive water, condensate tank water, torus water, and spent fuel pool water.

The program when supplemented with one-time inspections as required by NUREG-1801 provides reasonable assurance that components in the scope of license renewal are maintained in their desired environment during the period of extended operation, which mitigates the aging effects of loss of material and cracking.

B.1.3 Reactor Head Closure Studs

Description

The reactor head closure studs aging management program provides for condition monitoring and preventive activities to manage stud cracking and loss of material. The program is implemented through station procedures based on the examination and inspection requirements specified in ASME Section XI, Table IWB-2500-1 and preventive measures described in Regulatory Guide 1.65, "Materials and Inspection for Reactor Vessel Closure Studs." The reactor head studs at Dresden and Quad Cities are not metal-plated, and have had manganese phosphate coatings applied.

NUREG-1801 Consistency

With enhancement the reactor head closure studs aging management program is consistent with the ten elements of aging management program XI.M3, "Reactor Head Closure Studs," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that program inspections are in accordance with the requirements of ASME Section XI, Subsection IWB, Table IWB 2500-1. The Dresden and Quad Cities programs utilize relief requests CR-13 and CR-11 respectively. These relief requests provide for a VT-1 visual inspection instead of a surface examination of reactor closure head nuts.

NUREG-1801 indicates that the reactor closure head studs receive volumetric examinations. Quad Cities reactor closure head studs are examined by end shot UT as approved in relief request CR-12.

Enhancements

- NUREG-1801 indicates this program is to use the 1995 Edition through the 1996 Addenda of ASME Section XI. The current Code of record for Dresden and Quad Cities is the 1989 Edition of ASME Section XI. The program will be revised to be consistent with the requirements of the 1995 Edition through the 1996 Addenda of ASME Section XI.

The enhancement is scheduled for implementation prior to the period of extended operation.

Operating Experience

The Dresden and Quad Cities inspection and testing methodologies have detected aging effects (cracking) prior to loss of their intended functions. Engineering evaluations have resulted in various specific component and programmatic corrective actions. The reactor head closure studs aging management program activities have detected aging degradation and implemented appropriate corrective actions to maintain system and component intended functions including prompt repair or replacement of degraded components prior to failure.

Conclusion

The reactor head closure studs aging management program provides reasonable assurance that loss of material and cracking aging effects are adequately managed so that the intended functions of components within the scope of license are maintained during the period of extended operation.

B.1.4 BWR Vessel ID Attachment Welds

Description

The BWR vessel ID attachment welds aging management program activities incorporate the inspection and evaluation recommendations of BWRVIP-48, “Vessel ID Attachment Weld Inspection and Evaluation Guidelines,” as well as the water chemistry recommendations of EPRI TR-103515-R2, “BWR Water Chemistry Guidelines.” The program is implemented through station procedures that provide for mitigation of cracking through water chemistry and monitoring for cracking through invessel examinations. Reactor vessel attachment weld inspections are implemented through station procedures that are part of inservice inspection and incorporate the requirements of ASME Section XI. The ASME inspections are enhanced with inspections consistent with BWRVIP-48.

NUREG-1801 Consistency

With enhancement, the BWR vessel ID attachment welds aging management program is consistent with the ten elements of aging management program XI.M4, "BWR Vessel ID Attachment Welds," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that water chemistry control is in accordance with BWRVIP-29, “BWR Water Chemistry Guidelines.” BWRVIP-29 references the 1993 revision of EPRI TR-103515, "BWR Water Chemistry Guidelines." The Dresden and Quad Cities water chemistry programs are based on EPRI TR-103515-R2, which is the 2000 revision. Section B.1.2 presents the Water Chemistry aging management program and the exceptions to the program as specified in NUREG-1801.

Enhancements

- NUREG-1801 indicates this program is to use the 1995 Edition through the 1996 Addenda of ASME Section XI. The current Code of record for Dresden and Quad Cities is the 1989 Edition of ASME Section XI. The program will be revised to be consistent with the requirements of the 1995 Edition through the 1996 Addenda of ASME Section XI.

This enhancement is scheduled for implementation prior to the period of extended operation.

Operating Experience

The Dresden and Quad Cities inspection and testing methodologies have not detected cracking attachment welds. However, the same inspection and testing methodologies have detected cracking on the vessel internals. Section B.1.9 presents the BWR Vessel Internals aging management program.

Conclusion

The BWR vessel ID attachment welds aging management program provides reasonable assurance that cracking aging effects are adequately managed so that the intended functions of vessel ID attachment welds are maintained during the period of extended operation.

B.1.5 BWR Feedwater Nozzle

Description

The BWR feedwater nozzle aging management program provides for monitoring of feedwater nozzles for cracking through station procedures based on ASME Section XI augmented by inspections performed in accordance with the inspection recommendations of General Electric NE-523-A71-0594, “Alternate BWR Feedwater Nozzle Inspection Requirements." The program addresses the requirements of NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," by implementation of General Electric NE-523-A71-0594.

The Dresden and Quad Cities feedwater nozzles have been modified to mitigate cracking by removing the stainless steel cladding.

NUREG-1801 Consistency

The BWR feedwater nozzle aging management program is consistent with the ten elements of aging management program XI.M5, "BWR Feedwater Nozzle," specified in NUREG-1801.

Operating Experience

The Dresden and Quad Cities inspection and testing methodologies have detected indications of cracking aging effects on feedwater nozzles prior to loss of their intended functions. These indications were evaluated by engineering and the cracks were repaired by grinding and re-examination or the sleeve was replaced as determined by the engineering evaluation, ASME Code, and station procedures.

Conclusion

The BWR feedwater nozzle aging management program provides reasonable assurance that the aging effects of cracking are adequately managed so that the intended functions of the feedwater nozzles are maintained during the period of extended operation.

B.1.6 BWR Control Rod Drive Return Line Nozzle

Description

The control rod drive return line nozzle aging management program consists of inservice inspections and previously implemented system modifications to manage the aging effect of cracking in the control rod drive return line nozzles. Dresden and Quad Cities have capped the control rod drive return line nozzles. Inservice inspections are performed consistent with ASME Section XI requirements. No augmented inspections are required.

NUREG-1801 Consistency

With enhancement the control rod drive return line nozzle aging management program is consistent with the ten elements of aging management program XI.M6, "BWR Control Rod Drive Return Line Nozzle," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that the program includes system modifications and enhanced inservice inspections. The Dresden and Quad Cities programs do not provide for the augmented inspections specified in NUREG-0619, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking,” because the control rod drive return line nozzles have been capped. Inservice inspections are performed consistent with ASME Section XI requirements.

Enhancements

- NUREG-1801 indicates this program is to use the 1995 Edition through the 1996 Addenda of ASME Section XI. The current Code of record for Dresden and Quad Cities is the 1989 Edition of ASME Section XI. The program will be revised to be consistent with the requirements of the 1995 Edition through the 1996 Addenda of ASME Section XI.

The enhancement is scheduled for implementation prior to the period of extended operation.

Operating Experience

The Dresden and Quad Cities inspection and testing methodologies have detected indication of cracking aging effects on control rod drive nozzles prior to loss of their intended functions. These indications were repaired by flaw removal or weld overlay as determined appropriate by engineering evaluation in accordance with ASME Code and station procedure requirements.

Conclusion

The control rod drive return line nozzle aging management program provides reasonable assurance that the aging effects of cracking are adequately managed so that the intended functions of control rod return line nozzles are maintained during the period of extended operation.

B.1.7 BWR Stress Corrosion Cracking

Description

BWR stress corrosion cracking aging management program mitigates intergranular stress corrosion cracking (IGSCC) in stainless steel reactor coolant pressure boundary components and piping four inches and greater nominal pipe size. Preventive measures include monitoring and controlling of water impurities by water chemistry program activities and providing replacement stainless steel components in the solution annealed condition with a maximum carbon content of 0.035 wt. % and a minimum ferrite level of 7.5 wt. %. Inspection and flaw evaluation are conducted in accordance with the inservice inspection program plan for the station.

The program is implemented through station procedures based on NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," GL 88-01, "NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping," and its Supplement 1, BWRVIP-75, "Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules," EPRI TR-103515-R2, "BWR Water Chemistry Guidelines," and ASME Section XI.

NUREG-1801 Consistency

With enhancement the BWR stress corrosion cracking aging management program is consistent with the ten elements of aging management program XI.M7, "BWR Stress Corrosion Cracking," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that water chemistry control is in accordance with BWRVIP-29 for water chemistry in BWRs. BWRVIP-29 references the 1993 revision of EPRI TR-103515, "BWR Water Chemistry Guidelines." The Dresden and Quad Cities water chemistry programs are based on EPRI TR-103515-R2, which is the 2000 revision. Section B.1.2 presents the Water Chemistry aging management program and the exceptions to the program as specified in NUREG-1801.

Enhancements

- NUREG-1801 indicates this program is to use the 1995 Edition through the 1996 Addenda of ASME Section XI. The current Code of record for Dresden and Quad Cities is the 1989 Edition of ASME Section XI. The program will be revised to be consistent with the requirements of the 1995 Edition through the 1996 Addenda of ASME Section XI.

The enhancement is scheduled for implementation prior to the period of extended operation.

Operating Experience

BWR stress corrosion cracking aging management program activities have detected flaw indications in reactor coolant pressure boundary piping prior to loss of intended functions of the components, such as indications on the reactor vessel safe ends, and IGSCC on recirculation piping. These indications were evaluated by engineering, and where necessary were repaired in accordance with ASME Section XI and station procedural requirements. Periodic self-assessments of program activities have been performed and will continue to be performed to identify areas that need improvement. When problems have been identified corrective actions have been taken to prevent recurrence.

Conclusion

The BWR stress corrosion cracking aging management program provides reasonable assurance that IGSCC aging effects are adequately managed so that the intended functions of the stainless steel components in the reactor coolant pressure boundary are maintained during the period of extended operation.

B.1.8 BWR Penetrations

Description

The BWR penetrations aging management program activities incorporate the inspection and evaluation recommendations of BWRVIP-27, "BWR Standby Liquid Control System/Core Plate Delta-P Inspection and Flaw Evaluation Guidelines," and BWRVIP-49, "Instrument Penetration Inspection and Flaw Evaluation Guidelines," as well as the water chemistry recommendations of EPRI TR-103515-R2, "BWR Water Chemistry Guidelines." The program is implemented through station procedures that provide for mitigation of cracking through the water chemistry and monitoring for cracking through inservice inspection examinations. Penetration inspections are implemented through station procedures that are part of inservice inspection and incorporate the requirements of ASME Section XI.

NUREG-1801 Consistency

With enhancement the BWR penetrations aging management program is consistent with the ten elements of aging management program XI.M8, "BWR Penetrations," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that water chemistry control is in accordance with BWRVIP-29 for water chemistry in BWRs. BWRVIP-29 references the 1993 revision of EPRI TR-103515, "BWR Water Chemistry Guidelines." The Dresden and Quad Cities water chemistry programs are based on EPRI TR-103515-R2, which is the 2000 revision. Section B.1.2 presents the Water Chemistry aging management program and the exceptions to the program as specified in NUREG-1801.

NUREG-1801 indicates that the standby liquid control system nozzles are inspected in accordance with the requirements of ASME Section XI, Subsection IWB. The Dresden and Quad Cities programs utilize relief request ISI CR-01 that provides for inspection of the inner radius of the standby liquid control system nozzle by a VT-2 examination instead of the normal volumetric inspection required by the ASME Code.

Enhancements

- NUREG-1801 indicates this program is to use the 1995 Edition through the 1996 Addenda of ASME Section XI. The current Code of record for Dresden and Quad Cities is the 1989 Edition of ASME Section XI. The program will be revised to be consistent with the requirements of the 1995 Edition through the 1996 Addenda of ASME Section XI.

The enhancement is scheduled for implementation prior to the period of extended operation.

Operating Experience

The Dresden and Quad Cities inspection and testing methodologies have detected cracking aging effects prior to loss of their intended functions. Engineering evaluations have resulted in various specific component and programmatic corrective actions. No instrument penetration or standby liquid control nozzle indications have been reported. These penetrations are covered by the same inspection procedure as used for vessel internals inservice inspection. Section B.1.9 presents the BWR Vessel Internals aging management program.

Conclusion

The BWR penetrations aging management program provides reasonable assurance that the aging effects of cracking are adequately managed so that the intended functions of instrument penetrations and the standby liquid control system nozzles are maintained during the period of extended operation.

B.1.9 BWR Vessel Internals

Description

The BWR vessel internals aging management program mitigates the effects of SCC, IGSCC, and IASCC in reactor pressure vessel internals through water chemistry activities that are implemented through station procedures and are consistent with the guidelines of EPRI TR-103515-R2, "BWR Water Chemistry Guidelines," 2000 Revision. The program also manages cracking of reactor pressure vessel internals through condition monitoring activities that consist of examinations implemented through station procedures consistent with the recommendations of the BWRVIP guidelines, as well as the requirements of ASME Section XI.

NUREG-1801 Consistency

With enhancement the BWR vessel internals aging management program is consistent with the ten elements of aging management program XI.M9, "BWR Vessel Internals," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that water chemistry control is in accordance with BWRVIP-29, “BWR Water Chemistry Guidelines.” BWRVIP-29 references the 1993 revision of EPRI TR-103515, "BWR Water Chemistry Guidelines." The Dresden and Quad Cities water chemistry programs are based on EPRI TR-103515-R2, which is the 2000 revision. Section B.1.2 presents the Water Chemistry aging management program and the exceptions to the program as specified in NUREG-1801.

Enhancements

- NUREG-1801 indicates this program is to use the 1995 Edition through the 1996 Addenda of ASME Section XI. The current Code of record for Dresden and Quad Cities is the 1989 Edition of ASME Section XI. The program will be revised to be consistent with the requirements of the 1995 Edition through the 1996 Addenda of ASME Section XI.

The enhancement is scheduled for implementation prior to the period of extended operation.

Operating Experience

BWR vessel internals aging management activities have detected aging degradation and implemented appropriate corrective actions to maintain system and component intended functions including prompt repair of degraded components prior to failure. This includes cracking adjacent to the Quad Cities access hole cover and cracking of the reactor internal portion of the core spray piping at Dresden Unit 3. After evaluation by engineering, these cracks were repaired in accordance with ASME Section XI and station procedure requirements.

Jet Pump Beam Assembly #20 failed at Quad Cities Unit 1 in January 2002. All similar beams have been replaced with improved heat treatment beams.

Conclusion

The BWR vessel internals aging management program provides reasonable assurance that stress corrosion cracking, intergranular stress corrosion cracking, and irradiation assisted stress corrosion cracking aging effects are adequately managed so that the intended functions of BWR vessel internals within the scope of license renewal are maintained during the period of extended operation.

B.1.10 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)

Description

The aging management program for thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) is a new program that provides for aging management of CASS reactor internal components within the scope of license renewal. The program is scheduled for implementation prior to the period of extended operation.

The program will include a component specific evaluation of the loss of fracture toughness. For those components where loss of fracture toughness may affect function of the component, an inspection will be performed as part of the station ISI program. This inspection will ensure the integrity of the CASS components exposed to the high temperature and neutron fluence present in the reactor environment.

NUREG-1801 Consistency

With enhancement the aging management program for thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) will be consistent with the ten elements of aging management program XI.M13, "Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)," specified in NUREG-1801.

Enhancements

The aging management program for thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) is a new program, and therefore, the entire program is an enhancement.

Enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

The aging management program for thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) is a new program, and therefore, no operating experience exists.

Conclusion

The aging management program provides reasonable assurance that thermal aging and neutron irradiation embrittlement aging effects are adequately managed so that the intended functions of cast austenitic stainless steel (CASS) components within the scope of license renewal are maintained during the period of extended operation.

<u>B.1.11 Flow-Accelerated Corrosion</u>

Description

The flow-accelerated corrosion (FAC) aging management program is based on EPRI guidelines in NSAC-202L-R2, “Recommendations for an Effective Flow Accelerated Corrosion Program.” The program predicts, detects, and monitors wall thinning in piping, fittings and valve bodies due to FAC.

Analytical evaluations and periodic examinations of locations that are most susceptible to wall thinning due to FAC are used to predict the amount of wall thinning in pipes and fittings. Program activities include analyses to determine critical locations, baseline inspections to determine the extent of thinning at these critical locations, and follow-up inspections to confirm the predictions. Repairs and replacements are performed as necessary.

NUREG-1801 Consistency

With enhancement the flow-accelerated corrosion aging management program is consistent with the ten elements of aging management program XI.M17, "Flow-Accelerated Corrosion," specified in NUREG-1801.

Enhancements

- NUREG-1801 indicates the FAC program is to be applied to the main steam piping within the scope of license renewal. The Dresden and Quad Cities FAC program will be expanded to include main steam piping within the scope of license renewal.

This enhancement is scheduled for implementation prior to the period of extended operation.

Operating Experience

Operating experience of flow-accelerated corrosion aging management program activities has shown that the program can determine susceptible locations for flow-accelerated corrosion, predict the component degradation, and detect the wall thinning in piping and valves due to flow-accelerated corrosion. In addition, the program provides for reevaluation, repair or replacement for locations where calculations indicate an area will reach minimum allowable thickness before the next inspection. Periodic self-assessments of the program have been performed which have identified opportunities for program improvements. Improvements made as a result of the self-assessments include the addition of small bore HPCI turbine inlet drain lines to the program at Quad Cities.

Conclusion

The flow-accelerated corrosion (FAC) aging management program provides reasonable assurance that wall thinning aging effects are adequately managed so that the intended functions of components within the scope of license renewal are maintained during the period of extended operation.

B.1.12 Bolting Integrity

Description

The bolting integrity aging management program provides for condition monitoring of selected pressure retaining bolted joints and external surfaces for piping and components within the scope of license renewal. The bolting integrity program incorporates industry recommendations addressed in EPRI NP-5769, “Degradation and Failure of Bolting in Nuclear Power Plants,” as part of the comprehensive corporate component pressure retaining bolting program. The program consists of visual inspections for external surface degradation that may be caused by loss of material or cracking of the bolting, or by an adverse environment. The activities are implemented through station procedures and predefined tasks. Inspection of inservice inspection Class 1, 2 and 3 components is conducted in accordance with ASME Section XI. Non-safety component inspections rely on detection of visible leakage during preventive maintenance and routine observation activities.

NUREG-1801 Consistency

With enhancements the bolting integrity aging management program is consistent with the ten elements of aging management program XI.M18, "Bolting Integrity," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that EPRI TR-104213, “Bolted Joint Maintenance and Applications Guide,” is used as a basis for evaluation of the structural integrity of non-safety related bolting. NUREG-1801 also indicates that the bolting integrity programs developed and implemented in accordance with commitments made in response to NRC communications on bolting events have provided an effective means of ensuring bolting reliability. These programs are documented in EPRI NP-5769 and TR-104213 and represent industry consensus. The Dresden and Quad Cities programs address the guidance contained in EPRI TR-104213 but do not specifically cite its use.

NUREG-1801 indicates that the program covers all bolting within the scope of license renewal including structural bolting. The Dresden and Quad Cities bolting integrity programs do not address structural bolting. The Structures Monitoring Program (B.1.30) covers aging management of structural bolting.

NUREG-1801 indicates that the program covers all bolting within the scope of license renewal including bolting for Class 1 NSSS component supports. The Dresden and Quad Cities bolting integrity programs do not address Class 1 NSSS component support bolts. Aging management of ASME Section XI Class 1, 2, and 3, and Class MC support members, including mechanical connections, is covered by the ASME Section XI, Subsection IWF (B.1.27) aging management program.

NUREG-1801 indicates that the program generally includes periodic inspection for loss of preload. The Dresden and Quad Cities programs do not include inspections for loss of preload because loss of preload in a mechanical joint is a design driven effect and not an aging effect.

Enhancements

- The program will be revised as necessary to be in accordance with the 1995 Edition through the 1996 Addenda, and approved relief requests instead of the 1989 Edition with approved relief requests, of ASME Section XI for inspection of inservice inspection Class 1, 2 and 3 components.
- The program will provide for formal inspections of bolted joints of diesel generator system components and performing periodic component bolted joint inspections in high-humidity/moisture areas (pump vaults).

Enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

Dresden and Quad Cities have experienced isolated cases of bolting degradation attributed to loss of material and cracking aging effects. No reactor coolant pressure boundary leakage due to boric acid induced degradation has been noted since both stations are BWRs. System engineer walkdowns have also identified incidental surface rust on exterior component surfaces. In all cases, the existing inspection and testing methodologies have discovered the deficiencies and corrective actions were implemented prior to loss of system or component intended functions.

Conclusion

The bolting integrity aging management program provides reasonable assurance that loss of material and cracking aging effects are adequately managed so that the intended functions of bolting for pressure retaining joints and external surfaces for piping and components within the scope of license renewal are maintained during the period of extended operation.

B.1.13 Open-Cycle Cooling Water System

Description

The open-cycle cooling water (OCCW) system aging management program primarily consists of station GL 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” programs that include system test procedures, system inservice testing procedures, periodic component inspections and piping nondestructive examinations (NDE). Other program activities include station maintenance inspections, component preventive maintenance (PM) testing and inservice inspection (ISI) procedures and inspections for the Class 3 portions of raw water systems in the scope of license renewal.

The OCCW aging management program activities provide for management of loss of material, cracking, buildup of deposits and flow blockage aging effects in cooling water systems that are tested and inspected in accordance with guidelines of GL 89-13. The program also provides for managing loss of material aging degradation on the outside surfaces of cooling water and ultimate heat sink (UHS) components in the scope of license renewal by condition monitoring of the accessible external surfaces of components in moist air (indoor) or submerged (raw water) environments.

The OCCW aging management program activities apply to raw cooling water system piping and components and include both condition monitoring and mitigating actions for managing loss of material, cracking and flow blockage aging effects. System and component tests, visual inspections, NDE, flushing, and chemical treatment are conducted to ensure that aging effects are managed such that system and component functions are maintained.

The station maintenance and PM procedures provide for condition monitoring of raw water components within the scope of license renewal. ISI procedures for the inservice inspection of the Class 3 portions of raw water systems provide for periodic leakage detection of the aboveground and buried piping and components and inspection of the aboveground piping and components.

External surfaces of buried cooling water and UHS components are not included in this program because they are included in the Buried Piping and Tanks Inspection (B.1.25) aging management program.

NUREG-1801 Consistency

With enhancements the open-cycle cooling water system aging management program is consistent with the ten elements of aging management program XI.M20, "Open-Cycle Cooling Water System," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that program testing and inspections are performed annually and during refueling outages. The Dresden and Quad Cities open-cycle cooling water system aging management program activities provide for adjustment of inspection intervals due to specific inspection results as stated in the response to GL 89-13.

Enhancements

- The program will provide for inspection of cooling water pump internal linings, additional heat exchangers and sub-components, inspection of external surfaces of various submerged pumps and piping.
- The program will provide for new periodic component inspections in the pump vaults that have a high humidity/moisture environment.

Enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

The open-cycle cooling water system aging management activities have detected aging degradation and implemented appropriate corrective actions to maintain system and component intended functions including prompt repair of degraded components prior to failure. Prior to the implementation of the station GL 89-13 program activities, component blockage was a recurring problem resulting in valves being unable to function. More recent periodic system flushing and component inspections and cleaning have detected minor levels of biofouling and silting, primarily in system drain lines, that was removed without any loss of system functions. Component external surfaces have experienced corrosion, primarily in the pump vaults. Flooding and high humidity conditions noted in pump vaults contribute to creating a corrosive external environment. In all cases, the degradation was identified and corrective actions implemented prior to loss of system or component intended functions.

Engineering evaluations have resulted in various specific component and programmatic enhancements and corrective actions. In addition, program assessments have been reviewed for heat sink performance.

Conclusion

The open-cycle cooling water system aging management program provides reasonable assurance that loss of material, cracking, buildup of deposits and flow blockage aging effects are adequately managed so that the intended functions of open-cycle cooling water system components within the scope of license renewal are maintained during the period of extended operation.

B.1.14 Closed-Cycle Cooling Water System

Description

The closed-cycle cooling water system aging management program activities manage loss of material, cracking, and buildup of deposit aging effects in system components in the scope of license renewal exposed to closed-cycle cooling water environments. The program provides for preventive, performance monitoring and condition monitoring activities that are implemented through station procedures. Preventive activities include measures to maintain water purity and the addition of corrosion inhibitors to minimize corrosion based on EPRI TR-107396, "Closed Cooling Water Chemistry Guidelines." Performance monitoring provides indications of degradation in closed-cycle cooling water systems, with plant operating conditions providing indications of degradation in normally operating systems. In addition, station maintenance inspections and NDE provide condition monitoring of heat exchangers exposed to closed-cycle cooling water environments.

Heat exchanger activities are based on EPRI Report 1003056, "Non Class 1 Mechanical Implementation Guideline and Mechanical Tools," Revision 3, November 2001, Appendix G, "Heat Exchangers," Sandia National Laboratory Report SAND 93-7070 UC-523, "Aging Management Guideline for Commercial Nuclear Power Plants – Heat Exchangers," and ASME OM-S/G-2000, Part 21, "Inservice Performance Testing of Heat Exchangers in Light-Water Reactor Power Plants."

NUREG-1801 Consistency

With enhancements the closed-cycle cooling water system aging management program is consistent with the ten elements of aging management program XI.M21, "Closed-Cycle Cooling Water System," specified in NUREG-1801.

Enhancements

- Procedure revisions will provide for monitoring of specific parameters in accordance with EPRI TR-107396 guidance. Procedure revisions include provisions for monitoring parameters such as pH, specific gravity, freeze point, reserve alkalinity, percent glycol and suspended solids in glycol based systems.

- Dresden procedure revisions will provide for monitoring pH and ammonia in the diesel generator jacket water.

Enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

The Dresden and Quad Cities stations have not experienced a loss of intended function failure of components due to corrosion product buildup or through-wall cracking for components within the scope of license renewal that are subject to closed-cycle cooling water system activities. Additionally, industry operating experience demonstrates that the use of corrosion inhibitors in closed-cycle cooling water systems that are monitored and maintained is effective in mitigating loss of material, cracking, and buildup of deposits.

The Dresden and Quad Cities closed-cycle cooling water system activities have detected loss of material, cracking, and buildup of deposit aging effects in heat exchangers prior to loss of system intended functions. Engineering evaluations have resulted in various specific component and programmatic corrective actions.

Conclusion

The closed-cycle cooling water system aging management program provides reasonable assurance that loss of material, cracking, and buildup of deposits aging effects are adequately managed so that the intended functions of components exposed to closed-cycle cooling water environments within the scope of license renewal are maintained during the period of extended operation.

B.1.15 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems

Description

This aging management program provides for visual inspections of overhead heavy load and light load (related to refueling) handling systems. The program, which is implemented through station procedures, manages loss of material of bridge and trolley structural components for systems within the scope of 10 CFR 54.4 and other load handling systems within the scope of license renewal.

NUREG-1801 Consistency

With enhancements, the inspection of overhead heavy load and light load (related to refueling) handling systems aging management program is consistent with the ten elements of aging management program XI.M23, "Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that the number and magnitude of lifts made by the crane are reviewed. The Dresden and Quad Cities programs do not provide for tracking of the number and magnitude of lifts because administrative controls are implemented to ensure that only allowable loads are handled and fatigue failure of structural elements is not expected. A time-limited aging analysis concludes that there are no fatigue concerns for reactor building overhead cranes during the period of extended operation. Frequent inspections of cranes for indications of functional failures are conducted.

Enhancements

- The program will provide for specific inspections for rail wear and proper crane travel on rails.

- The program will provide for specific inspections for corrosion of crane structural components.

Enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

This program has been successful in managing aging of structural components of overhead heavy load and light load (related to refueling) handling systems so that intended functions have been maintained. The Dresden and Quad Cities corrective action process records indicate that crane operational problems have been minor and do not indicate that there has been degradation of structural components.

Conclusion

The inspection of overhead heavy load and light load (related to refueling) handling systems aging management program provides reasonable assurance that loss of material aging effects are adequately managed so that the intended functions of crane structural components within the scope of license renewal are maintained during the period of extended operation.

B.1.16 Compressed Air Monitoring

Description

The compressed air monitoring aging management program activities manage loss of material due to general , crevice, and pitting corrosion for portions of the instrument air system within the scope of license renewal. Program activities consist of air quality testing, pressure decay testing, and visual inspections at various system locations. The activities are consistent with Dresden and Quad Cities responses to NRC Generic Letter 88-14, “Instrument Air Supply Problems,” and ANSI/ISA-S7.3-1975, "Quality Standard for Instrument Air." Testing and monitoring activities are implemented through station specific procedures and associated predefined tasks.

NUREG-1801 Consistency

With enhancements the compressed air monitoring aging management program is consistent with the ten elements of aging management program XI.M24, "Compressed Air Monitoring," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that the program is based on responses to GL 88-14 and INPO SOER 88-01, “Instrument Air System Failures,” as well as EPRI NP-7079-1990, EPRI TR-108147, “Compressor and Instrument Air System Maintenance Guide,” ASME OM-S/G-1998 and ANSI/ISA-S7.0.01-1996. The Dresden and Quad Cities programs are based on the guidance provided in the GL 88-14 and ANSI/ISA-S7.3-1975 documents, which are part of the current licensing basis. Enhancements include inspection of instrument air distribution piping based on EPRI TR-108147.

NUREG-1801 indicates that inservice inspection and testing is performed to verify proper air quality, and confirm that maintenance practices, emergency procedures and training are adequate to ensure that the intended function of the air system is maintained. Inservice inspections at Dresden and Quad Cities do not verify air quality because air quality testing is performed in accordance with specific procedures based on ANSI/ISA-S7.3-1975. Maintenance practices, emergency procedures, and training are plant performance issues that are not directly related to aging management of the instrument air systems. Aging management consists of air quality tests and pressure decay tests of MSIV and safety/relief valve pneumatic systems including accumulators, piping, and check valves, and periodic inspections to verify the integrity of the systems.

Enhancements

- The program will provide for new periodic inspections for those portions of instrument air distribution piping at Dresden and Quad Cities that are within the scope of the rule.

- The program will provide for periodic blowdowns of instrument air distribution piping at Dresden.

Enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

Dresden has experienced recent occurrences of corrosion, corrosion product buildup, and dirt buildup in instrument air system piping, positioners, and valve operators. The program enhancement of providing periodic blowdowns of instrument air distribution piping addresses this condition.

Quad Cities has not experienced a failure of a pneumatic component within the scope of license renewal due to corrosion, corrosion product buildup, or dirt buildup since 1993. The Quad Cities experience is consistent with the implementation of corrective actions in response to GL 88-14.

Dresden and Quad Cities have experienced equipment failures including MSIVs, dampers, and process valves due to instrument air leaks. These failures were to individual components and did not propagate to other components within the system. Dresden and Quad Cities have not experienced a common mode failure caused by the instrument air system. The Dresden and Quad Cities enhancements of performing predefined tasks that require periodic inspections of instrument air distribution piping address this condition.

Conclusion

The compressed air monitoring aging management program provides reasonable assurance that loss of material aging effects are adequately managed so that the intended functions of the instrument air components within the scope of license renewal are maintained during the period of extended operation.

B.1.17 BWR Reactor Water Cleanup System

Description

The aging management program for the BWR reactor water cleanup system provides for reactor water chemistry activities that reduce susceptibility to stress corrosion cracking (SCC) and intergranular stress corrosion cracking (IGSCC).

Dresden and Quad Cities have satisfactorily completed all actions requested in NRC GL 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance,” and have replaced the reactor water cleanup (RWCU) system piping with piping that is resistant to intergranular stress corrosion cracking in accordance with NRC GL 88-01, “NRC Position on Intergranular Stress Corrosion Cracking (IGSCC) in BWR Austenitic Stainless Steel Piping.” Therefore, inspection of RWCU piping is not required.

The reactor water chemistry activities provide for monitoring and controlling of water chemistry using station procedures and processes based on EPRI TR-103515, “BWR Water Chemistry Guidelines,” 2000 Revision to reduce susceptibility of RWCU piping to SCC and IGSCC.

NUREG-1801 Consistency

The BWR reactor water cleanup system aging management program is consistent with the ten elements of aging management program XI.M25, "BWR Reactor Water Cleanup System," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that water chemistry control is in accordance with BWRVIP-29, “BWR Water Chemistry Guidelines.” BWRVIP-29 references the 1993 revision of EPRI TR-103515, "BWR Water Chemistry Guidelines." The Dresden and Quad Cities water chemistry programs are based on EPRI TR-103515-R2, which is the 2000 revision. Section B.1.2 presents the Water Chemistry aging management program and the exceptions to the program as specified in NUREG-1801.

Operating Experience

Dresden and Quad Cities have replaced the reactor water cleanup (RWCU) system piping with piping that is resistant to intergranular stress corrosion cracking. No adverse trends have been detected in the chemistry program.

Conclusion

The BWR reactor water cleanup system aging management program provides reasonable assurance that the SSC and IGSCC aging effects are adequately managed so that the intended functions of RWCU components within the scope of license renewal are maintained during the period of extended operation.

B.1.18 Fire Protection

Description

The fire protection program provides for aging management of various fire protection related components within the scope of license renewal.

The program provides for visual inspection of fire barrier walls, ceilings and floors in structures within the scope of license renewal for the aging effects of cracking, spalling, and loss of material. The inspections are implemented through procedures that are part of the structures monitoring program.

The program provides for visual inspections of fire barrier penetration seals, fire wraps, and flood barrier penetration seals that also serve as fire barrier seals for signs of degradation, such as damage, holes, cracking, and loss of material, through periodic inspection, surveillance and maintenance activities. The inspections are implemented through station procedures. Flood barrier penetration seal inspections are part of the structures monitoring program.

The program provides or will provide for periodic visual inspections of fire doors for holes in skin, wear, or missing parts. Fire door clearances are checked during periodic inspections and when fire doors and components are repaired or replaced. Additionally, periodic functional tests of fire doors and periodic visual inspections for loss of barrier integrity of curb oil spill barriers are implemented through station procedures.

The program will provide for aging management of external surfaces of Dresden and Quad Cities carbon dioxide system components and Dresden halon system components for corrosion through periodic operability tests based on NFPA codes and visual inspections. Testing and inspections are implemented through predefined tasks and procedures.

The program will provide for managing loss of material aging effects for the fuel oil systems for the Dresden and Quad Cities diesel-driven fire pumps and the Dresden isolation condenser makeup pumps by the performance of periodic fuel oil system surveillance tests that are implemented through station procedures. The Dresden isolation condenser diesel-driven makeup pumps are similar in design and function to the diesel-driven fire pumps. Therefore, the fuel oil systems for the isolation condenser makeup pumps are included in the program.

NUREG-1801 Consistency

With enhancements the fire protection program is consistent with the ten elements of aging management program XI.M26, "Fire Protection," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that visual inspections performed at least once every refueling outage detect any sign of degradation of the concrete fire barrier walls, ceilings, and floors before there is a loss of the intended function. The Dresden and Quad Cities program requires inspection of concrete fire barriers once every five years. Station fire protection program inspections and other similar station inspections have not found any significant aging effect that required extensive corrective action for any concrete structure within the scope of license renewal. Typically, concrete cracks that have been observed have been attributed to normal concrete shrinkage occurring during construction and are non-active. Additionally, the environment surrounding Dresden and Quad Cities is non-aggressive for concrete, and industry guidance contained in ACI 349.3R-96, “Evaluation of Existing Nuclear Safety-Related Concrete Structures,” indicates that a five-year inspection frequency for concrete components is adequate for timely identification and correction of degraded conditions prior to loss of intended function.

NUREG-1801 indicates that VT-1 or equivalent seal inspections and VT-3 or equivalent fire door inspections are performed. Personnel performing seal and fire door inspections at Dresden and Quad Cities are not qualified to American Society for Nondestructive Testing requirements. However, personnel performing these inspections are trained and experienced in fire protection program requirements, and the quality of the fire barrier penetration seal and fire door inspections are equivalent to the VT-1 and VT-3 inspections as is evidenced by the history of identifying conditions requiring maintenance, repair or replacement.

NUREG-1801 indicates that fire doors are visually inspected at least once bi-monthly for holes in the skin of the door. Fire door clearances are also checked at least once bi-monthly as part of an inspection program. Function tests of fire doors are performed daily, weekly, or monthly to verify the operability of automatic hold-open, release, closing mechanisms, and latches. The Dresden and Quad Cities programs provide for an in-depth inspection for condition and operability of fire doors once per operating cycle. The frequency of the in-depth inspections is appropriate because the fire doors most likely to experience excessive wear are those that are subject to the most frequent use, such as those in normal and high traffic areas, and such doors are monitored by normal plant operation during periodic fire marshal tours, operator rounds, and security patrols.

NUREG-1801 indicates that a periodic function test and visual inspection performed at least once every six months detects degradation of the halon and carbon dioxide fire suppression systems before the loss of the component intended function. The Quad Cities and Dresden halon and carbon dioxide fire suppression systems are normally tested and inspected every 18 months. The Technical Requirements Manual permits a testing frequency of once every two years. This frequency is considered sufficient to ensure system availability and operability based on station operating history that indicates no occurrence of aging related events that have adversely affected system operation.

NUREG-1801 indicates that any signs of corrosion and mechanical damage of the halon or carbon dioxide fire suppression system are not acceptable. The Dresden and Quad Cities programs require that signs of aging degradation on the external surfaces of the halon or carbon dioxide fire suppression systems are evaluated and corrective action is taken as required. This approach provides reasonable assurance that corrective actions appropriate to the observations will be implemented prior to loss of system or component intended functions.

NUREG-1801 indicates that the performance of the fire pump is monitored during the periodic test to detect any degradation in the fuel supply lines, and that periodic testing provides data (e.g., pressure) necessary for trending. The Dresden and Quad Cities diesel-driven fire pump test results and the Dresden isolation condenser diesel-driven makeup pump test results are not trended. Instead, in the event the predetermined acceptance criteria are not met an engineering evaluation is conducted to determine the operability of the pump and the need for corrective action.

Enhancements

- The program will provide specific guidance at Dresden and Quad Cities to check fire doors for wear and holes in skin that could affect intended function during weekly tours.
- The program will provide for inspection for corrosion on external surfaces of piping and components for the Dresden and Quad Cities carbon dioxide systems and the Dresden halon system.
- The program will provide for performance of periodic capacity tests of the Dresden isolation condenser diesel-driven makeup pumps.
- The program will provide specific guidance for examining the fire pump and the Dresden isolation condenser makeup pump diesel fuel supply systems for leaks during pump tests.
- The program will provide for inspection of oil spill barriers for signs of damage that would break the integrity of the barrier at Quad Cities.
- Dresden procedures will identify the frequency of inspections of fire doors and spill barriers.

Enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

The fire protection program has been effective in identifying aging effects and taking appropriate corrective action.

Minor degradation such as minor cracks with water stains, pitting, and small amounts of leaching have been detected in concrete components in structures within the scope of license renewal including the Dresden reactor building and crib house, and the Quad Cities reactor building and circulating water intake bays. The observed degradation was evaluated and dispositioned based on program acceptance criteria and in accordance with the corrective action process.

The Dresden and Quad Cities experience with fire barrier penetration seals is consistent with the industry experience. Silicone foam fire barrier penetration seals are used at Dresden. Silicone foam fire barrier penetration seals are not used at Quad Cites.

Dresden and Quad Cities have experienced fire door component degradation due to wear and physical damage. Mitigating actions have been taken as appropriate. No reports of door assembly loss of material due to corrosion have been identified for either station.

Dresden and Quad Cities operating experience has shown no loss of material on the external surfaces of components in the halon and carbon dioxide systems that have adversely affected system operation.

The Dresden and Quad Cities diesel-driven fire pump fuel oil systems have experienced minor system leakage events that have been detected and corrected in a timely manner. In addition, Dresden reported high particulates in the Unit 1 fire pump fuel oil day tank, and Quad Cities reported a fuel oil separator drain line plugged with sediment. These were identified and corrected prior to loss of intended function of the fire pumps. There have been no reports of loss of material or flow blockage of the Dresden isolation condenser makeup pump fuel oil subsystem.

Conclusion

The fire protection program covers various fire protection related components within the scope of license renewal including fire barrier doors, walls, ceilings, floors, penetration seals and wraps. It also covers flood barrier penetration seals, curb oil spill barriers, external surfaces of Dresden and Quad Cities carbon dioxide system components, and external surfaces of Dresden halon system components. In addition, the program covers the fuel oil systems for Dresden and Quad Cities diesel-driven fire pumps and the Dresden isolation condenser makeup pumps.

The fire protection aging management program provides reasonable assurance that the cracking, spalling, loss of material and wear aging effects are adequately managed so that the intended functions of components within the scope of license renewal are maintained during the period of extended operation.

B.1.19 Fire Water System

Description

The fire water system aging management program provides for managing loss of material and biofouling aging effects on intended functions of the water-based fire protection system components within the scope of license renewal through periodic inspection, surveillance testing, and maintenance activities. Fire hose testing is not included in the program because fire hoses are consumables and not in the scope of license renewal. Fire hoses are periodically inspected or tested and replaced as necessary in accordance with station procedures.

Inspection and surveillance is performed in accordance with procedures based on NFPA codes. Where code deviations are required or desirable, the intent of the code is maintained based on documented technical justifications.

NUREG-1801 Consistency

With enhancements the fire water system aging management program is consistent with the ten elements of aging management program XI.M27, "Fire Water System," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that fire water systems are tested in accordance with NFPA codes and standards. The fire water systems at Dresden and Quad Cities used the NFPA codes as guidelines for their design. Similarly, inspection and periodic testing performed in accordance with corporate and station procedures were developed using NFPA codes as guidance. Where deviations are considered necessary or desirable, technical justifications are documented. Regardless of the codes of record, the credited inspections will be conducted at a frequency appropriate to detect aging degradation prior to loss of intended function.

NUREG-1801 indicates that NRC GL 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” recommends periodic flow testing of infrequently used loops of the fire water system at the maximum design flow to ensure that the system maintains its intended function. The Dresden and Quad Cities programs do not provide for flow tests at the maximum design flow and pressure because such tests are not practicable. Periodic flow and pressure testing is performed to determine the condition of underground fire protection piping. These tests measure hydraulic resistance and compare it with previous tests.

Enhancements

- The program will provide for periodic non-intrusive wall thickness measurements of selected portions of the fire water system at intervals that do not exceed every 10 years.

- Pump bay inspection procedures will provide for external surface inspections for submerged fire pumps.

- Fire hydrant test procedures will provide for external surface inspections for outdoor fire hydrants.

- The requirement for external surface inspections of outdoor transformer deluge system components will be added to deluge system test procedures.

- The program will provide for sampling of sprinklers in accordance with NFPA 25, “Inspection, Testing, and Maintenance of Water-Based Fire Protection Systems,” and submitting the samples to a testing laboratory prior to the sprinklers being in service 50 years. Subsequent testing is at intervals that do not exceed every 10 years.

Enhancements are scheduled for implementation at Dresden and Quad Cities prior to the period of extended operation.

Operating Experience

Instances of fire water system degradation due to corrosion and biofouling have been detected at Dresden and Quad Cities. Some cases have resulted in system leakage or blockage that affected system capability. Problems have been identified and appropriate corrective actions were taken to prevent recurrence. Station activities and assessments have resulted in the following enhancements for aging management of biofouling: procedure revisions for more frequent pump bay cleaning to reduce silt and clam buildup; monitoring for zebra mussel infestation; and new periodic inspections of components susceptible to blockage ("Y" strainers and check valves).

Conclusion

The fire water system aging management program provides reasonable assurance that the loss of material and biofouling aging effects are adequately managed so that the intended functions of fire water components within the scope of license renewal are maintained during the period of extended operation.

B.1.20 Aboveground Carbon Steel Tanks

Description

The aboveground carbon steel tanks aging management program provides for management of loss of material aging effects for outdoor carbon steel nitrogen storage tanks. The program provides for the application of paint as a corrosion preventive measure and for periodic visual inspections to monitor degradation of the paint and any resulting metal degradation.

The nitrogen storage tanks are aboveground and not directly supported by earthen or concrete foundations. Therefore, inspection of the sealant or caulking at the tank-foundation interface and inspection of inaccessible tank locations does not apply.

NUREG-1801 Consistency

With enhancement the aboveground carbon steel tanks aging management program is consistent with the ten elements of aging management program XI.M29, "Aboveground Carbon Steel Tanks," specified in NUREG-1801.

Enhancements

- The program will provide for initiating documentation of inspection results of periodic system engineer walkdowns of the nitrogen storage tanks.

The enhancement is scheduled for implementation at Dresden and Quad Cities prior to the period of extended operation.

Operating Experience

The Dresden and Quad Cities outdoor carbon steel nitrogen storage tanks have not experienced leakage or degradation due to loss of material.

Conclusion

The aboveground carbon steel tanks aging management program provides reasonable assurance that the aging effects of loss of material are adequately managed so that the intended functions of outdoor nitrogen storage tanks within the scope of license renewal are maintained during the period of extended operation.

B.1.21 Fuel Oil Chemistry

Description

The fuel oil chemistry aging management program provides for preventive activities that manage the aging effects of loss of material and buildup of deposits in license renewal components that are exposed to fuel oil.

Program activities assure that contaminant levels are maintained at acceptable levels in fuel oil for systems within the scope of license renewal. A biocide is added to the fuel oil storage tanks during each new fuel delivery. Fuel oil sampling and analysis are performed in accordance with procedures. Emergency diesel generator fuel oil analysis acceptance criteria are contained in the Technical Specifications and are based on the requirements of ASTM D975. Diesel fuel oil storage tanks are periodically cleaned and inspected for evidence of internal corrosion.

NUREG-1801 Consistency

With enhancements, the fuel oil chemistry aging management program is consistent with the ten elements of aging management program XI.M30, "Fuel Oil Chemistry," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 discusses the need to mitigate corrosion of fuel oil storage tanks. Corrosion mitigation activities are not performed for the Quad Cities Unit 1 underground fuel oil storage tank because it is constructed of fiberglass.

NUREG-1801 indicates that ASTM D1796 standard should be used to analyze fuel oil for water and sediment. The Dresden and Quad Cities programs use ASTM D2709 as specified by ASTM D975 for analysis of grades 1-D and 2-D fuel used at the stations.

NUREG-1801 indicates that ASTM D2276 (Modified), which provides for field monitoring, should be used to analyze fuel oil for particulate content. The Dresden and Quad Cities programs use ASTM D5452 as the preferred method of analysis.

NUREG-1801 indicates that a modified ASTM D2276 test with a filter pore size of 3.0 μm should be utilized for detection of particulate in fuel oil. Dresden and Quad Cities particulate tests utilize filters with a pore size of 0.8 μm instead of 3.0 μm because 0.8 μm filters provide conservative results. The use of 0.8 μm filters is consistent with use of ASTM D5452.

NUREG-1801 discusses the need to add stabilizers and corrosion inhibitors to diesel fuel oil. Quad Cities does not add stabilizers because grade 1-D low sulfur fuel oil is used and stored fuel is periodically sampled and analyzed for quality. Dresden and Quad Cities do not add corrosion inhibitors because fuel oil storage tank bottoms are periodically sampled and analyzed for corrosion products in accordance with ASTM D4057 and ASTM D2709. Dresden and Quad Cities employ sample techniques and particulate contamination detection methods that identify fuel degradation or the presence of corrosion products at an early stage.

NUREG-1801 indicates that fuel oil tanks should be sampled for water, biological activity, and particulate on a periodic basis and that multilevel sampling of tanks should be performed. The Dresden and Quad Cities emergency diesel generator fuel oil day tanks do not have the capability of being sampled. As an alternative, Dresden and Quad Cities sample for water and sediment from the bottom of the associated storage tanks quarterly and particulate from the fuel oil transfer pump discharge line on a monthly basis in accordance with approved procedures. Dresden and Quad Cities do not perform multilevel sampling of other fuel oil day tanks (isolation condenser makeup pump [Dresden only], fire pump, and station blackout) because the tanks are small and experience a high turnover of fuel due to routine diesel engine operations. Additionally, ASTM D4057, Table 4, “Spot Sampling Requirements,” indicates that multilevel sampling is not required for tanks with a capacity less than 42,000 gallons. The fuel oil storage tanks and day tanks at Dresden and Quad Cities are 15,000 gallons or smaller.

NUREG-1801 indicates that fuel oil sample results should be monitored and trended. At Dresden, the results of analysis of new fuel oil are reviewed for acceptability, but are not trended. In the event the quantitative oil acceptance criteria in plant procedures are approached or exceeded the fuel oil is restored to within limits or an action request or condition report is initiated.

Enhancements

- The fuel oil chemistry program will provide for inspection of the fuel oil storage tank interiors for corrosion during the regularly scheduled tank cleanings and the performance of engineering evaluations in the event corrosion of the tank interiors is found.

The enhancement is scheduled for implementation at Dresden and Quad Cities prior to the period of extended operation.

Operating Experience

The fuel oil chemistry aging management program has proven to be effective in identifying and correcting abnormal conditions in a timely manner. Dresden and Quad Cities have experienced a small number of events where plugging of drain lines in fuel oil system low points was caused by sediment buildup. These events did not affect the ability of the associated diesel generator or fire pump to perform its intended functions. Quad Cities experienced plugging of both fuel filters on one diesel generator in 1998 for an indeterminate reason.

Conclusion

The fuel oil chemistry aging management program provides reasonable assurance that the loss of material aging effects are adequately managed so that the intended functions of components exposed to fuel oil within the scope of license renewal are maintained during the period of extended operation.

B.1.22 Reactor Vessel Surveillance

Description

The reactor vessel surveillance aging management program provides management of irradiation embrittlement of the reactor pressure vessel through testing that monitors reactor vessel beltline materials. The program is implemented through station procedures that conform to the requirements of 10 CFR 50, Appendix H, “Reactor Vessel Material Surveillance Program Requirements.” Neutron embrittlement for the period of extended operation is predicted utilizing chemistry tables and Position 1.3 limitations as described in Regulatory Guide 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials.”

The program will be consistent with the BWRVIP-78, "Integrated Surveillance Program," (ISP.) The program will provide for capsule testing consistent with BWRVIP-78, and saving withdrawn capsules for future reconstitution. Additionally the program will be consistent with BWRVIP-86, “BWR Integrated Surveillance Program Implementation Plan.” The ISP will increase technical quality and make most effective use of existing BWR surveillance capsules and data.

The enhanced program provides for capsule testing consistent with BWRVIP-78 and BWRVIP-86. Existing capsules not included in the current ISP will be maintained in the vessels as a contingency for the future.

The BWRVIP-78 report described the technical basis related to material selection and testing on which the proposed BWRVIP ISP was constructed. The report principally addressed the methodology established to identify existing plant-specific surveillance capsules and surveillance capsules from the Supplemental Surveillance Program initiated by the Boiling Water Reactors Owners’ Group in the late 1980s, which contain important surveillance materials for inclusion within the ISP.

The BWRVIP-86 report was submitted to follow up on the material presented in the BWRVIP-78 report by establishing specific guidelines for ISP implementation. The BWRVIP-86 report addressed determination of ISP surveillance capsule withdrawal and testing dates, information on ISP project administration, additional information on neutron fluence determination issues, additional information on data utilization and sharing, and information on licensing aspects of ISP implementation.

NUREG-1801 Consistency

With enhancements the aging management program for reactor vessel surveillance is consistent with the aging management program XI.M31, "Reactor Vessel Surveillance," described in NUREG-1801.

Enhancements

- The program will provide for performance of reactor beltline material surveillance consistent with BWR ISP, BWRVIP-78, BWRVIP-86, the NRC SER for the ISP and associated RAIs.

Enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

Material capsules are withdrawn and testing is performed in accordance with the program requirements. Program updating is performed as new industry standards are developed. No adverse conditions have been identified.

Conclusion

The reactor vessel surveillance aging management program provides reasonable assurance that the aging effects of irradiation embrittlement are adequately managed so that the intended functions of reactor vessel components within the scope of license renewal are maintained during the period of extended operation.

B.1.23 One-Time Inspection

Description

The one-time inspection aging management program provides one-time inspections that manage aging effects of identified components within the scope of license renewal. The purpose of the program is to determine if a specified aging effect is occurring. If the aging effect is occurring, the program provides for evaluation of the effect it will have on the ability of affected components to perform their intended functions for the period of extended operation, and appropriate corrective action. The program is implemented through station procedures.

The program includes the following one-time inspections:

One-Time Inspections Identified in Section 3 "Further Evaluation" Discussions

- Inspect a sample of Class I piping less than four inch NPS exposed to reactor coolant for cracking.
- Inspect a sample of torus saddle Lubrite baseplates for galvanic corrosion, wear, and lockup to confirm the condition of the inaccessible drywell radial beam Lubrite baseplates.
- Inspect a sample of spent fuel pool cooling and demineralizer system (Dresden only) components for corrosion in stagnant locations to verify effective water chemistry control.

One-Time Inspections Identified in Section 3 "Further Evaluation" Discussions, and Also Applicable to Non-NUREG-1801 Components

- Inspect a sample of piping exposed to the containment atmosphere (safety relief valve discharge piping and HPCI turbine exhaust sample locations) for loss of material.
- Inspect a sample of condensate and torus water components for corrosion in stagnant locations to verify effective water chemistry control.
- Inspect a sample of compressed gas system piping components for corrosion and a sample of compressed gas system flexible hoses for elastomer degradation.
- Inspect a sample of lower sections of carbon steel fuel oil and lubricating oil tanks for reduced thickness.
- Inspect a sample of fuel oil and lubricating oil piping and components for corrosion.

- Inspect a sample of main control room ventilation, emergency diesel generator ventilation, SBO building ventilation, reactor building ventilation and standby gas treatment system components for loss of material.

One-Time Inspections Not Identified in Section 3 "Further Evaluation" Discussions

These inspections are for component, material, environment, and aging effect combinations that are not addressed in NUREG-1801, and for which one-time inspections are not identified under "Further Evaluations" in LRA Section 3. Therefore, brief descriptions of the inspections are included here:

- Inspect a sample of stainless steel standby liquid control (SBLC) system components not in the reactor coolant pressure boundary section of the SBLC system for cracking, to verify effective water chemistry control.

  Aging of SBLC system components not in the reactor coolant pressure boundary section of SBLC system relies on monitoring and control of SBLC makeup water chemistry based on the Water Chemistry (B.1.2) aging management program. The effectiveness of the water chemistry program will be verified by a one-time VT-3 inspection of a Quad Cities SBLC pump casing and a Dresden SBLC pump discharge valve.

- Inspect a sample of HPCI turbine lubricating oil hoses for age-related degradation.

  A sample of lubricating oil hoses from each facility will be removed and inspected for evidence of elastomer degradation due to aging.

- Inspect a sample of non-safety related vents and drains including their valves and associated piping, for age-related degradation leading to a loss of structural integrity.

  A sample from each facility will be visually inspected for evidence of degradation due to aging.

- Inspect a sample of 10 CFR 54.4(a)(2) components for corrosion for which the component, material, environment, aging effect, or their combination is not specifically identified in NUREG-1801.

  Some components are in scope of license renewal only because they meet the 10 CFR 54.4(a)(2) criterion for

  > …systems, structures, or components whose failure could prevent satisfactory accomplishment of any of the [critical, in-scope] functions identified in [other sections of the license renewal rule].

  Some of these components may be susceptible to corrosion, which may affect the intended in-scope function. However, the component, material, environment, aging effect, or their combination may not be specifically identified in NUREG-1801.

A review of the 10 CFR 54.4(a)(2) system component materials, and their environments, found examples of industry component type-material-environment combinations, which are not addressed by NUREG-1801, and include components whose failure might affect intended functions of safety related systems in the scope of license renewal. The one-time inspection procedure contains a list material-environment pairs from which a sample will be selected for inspection. The program provides a one-time, internal, visual inspection; for general, crevice, galvanic, and pitting corrosion as appropriate; on at least one component for each of the material-environment pairs. Some of these systems met the 10 CFR 54.4(a)(2) criterion at only one of the two plants. In that case the sample location or locations will be chosen at the plant where the criterion is met.

NUREG-1801 Consistency

The one-time inspection aging management program is scheduled for implementation prior to the period of extended operation. Program activities are consistent with the ten elements of aging program XI.M32, "One-Time Inspection," specified in NUREG-1801.

Operating Experience

The one-time inspection program is new. Therefore, no programmatic operating experience is available. The program is scheduled for implementation prior to the period of extended operation.

Conclusion

The one-time inspection management program provides reasonable assurance that the aging effects are adequately managed so that the intended functions of components within the scope of license renewal that are covered by this program are maintained during the period of extended operation.

B.1.24 Selective Leaching of Materials

Description

The selective leaching of materials aging management program consists of numerous one-time inspections to determine if selective leaching of materials is occurring. The scope of the program includes susceptible components within the scope of license renewal that are exposed to chemically treated water, demineralized water, raw water and ground water, and moist ventilation and gas environments. Susceptible component materials are gray cast iron, copper alloys with less than 85% copper, aluminum bronze, and Muntz metal.

The one-time inspection program includes visual inspection or other appropriate examination methods of components of the different susceptible materials selected from each applicable environment. The purpose of the program is to determine if loss of material due to selective leaching is occurring. If selective leaching is occurring, the program provides for evaluation as to the effect it will have on the ability of the affected components to perform their intended function for the period of extended operation, and the need to expand the sample of components to be tested.

NUREG-1801 Consistency

The selective leaching of materials aging management program is a new program. The program is scheduled for implementation prior to the period of extended operation. Program activities are consistent with the ten elements of aging program XI.M33, "Selective Leaching of Materials," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that the selective leaching of materials aging management program includes a one-time hardness measurement of a selected set of components. NUREG-1801 also indicates that the hardness test provides the basis for determination that the intended functions are not lost due to selective leaching induced loss of material during the period of extended operation.

The Dresden and Quad Cities programs provide for visual examination in lieu of hardness testing. Hardness testing provides definitive results only if baseline values are available for comparison purposes. Because materials in general have an acceptable ASTM hardness range rather than a specific value, definitive results may not be possible using hardness measurements. Components that exhibit visual indications of selective leaching will receive further examination or evaluation, which may include nondestructive testing or other examinations that provide definitive results regarding the presence of selective leaching.

Operating Experience

The selective leaching of materials aging management program is new. Therefore, no programmatic operating experience is available. The program is scheduled for implementation prior to the period of extended operation.

Conclusion

The selective leaching of materials aging management program will provide reasonable assurance that selective leaching aging effects are adequately managed so that the intended functions of components within the scope of license renewal are maintained during the period of extended operation.

B.1.25 Buried Piping and Tanks Inspection

Description

The buried piping and tanks inspection program for aging management consists of preventive and condition monitoring measures to manage loss of material due to corrosion from external environments for buried piping and tanks in the scope of license renewal.

Inspection of buried components uncovered during maintenance cannot be relied upon as the sole method for providing effective aging management because operating experience has demonstrated that buried piping or tanks are not likely to be uncovered during yard excavation activities due to their locations and depths of the specific routings. Therefore, program activities as enhanced include the use of piping and component coatings and wrappings, periodic pressure testing, buried tank leakage checks, inspections of buried tank internal surfaces, and inspections of the ground above buried tanks and piping. Enhanced activities also include a one-time visual inspection of the external surface of a buried piping section, one-time internal UT's of buried steel tanks, and a one-time internal UT of the bottom of an outdoor aluminum storage tank.

NUREG-1801 Consistency

With enhancements the buried piping and tanks inspection program for aging management is consistent with the ten elements of aging management program XI.M34, "Buried Piping and Tanks Inspection," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that buried piping and tanks are inspected when they are excavated during maintenance. NUREG-1801 also indicates that because the inspection frequency is plant-specific and also depends on plant operating experience, the inspection frequency requires further evaluation. Inspections of Dresden and Quad Cities buried components uncovered due to maintenance cannot be relied upon as the sole method for providing effective aging management because uncovering of piping or tanks during maintenance is not likely. Therefore, the program as enhanced includes the use of piping and component coatings and wrappings, periodic pressure testing, buried tank leakage checks, inspections of buried tank internal surfaces, and inspections of the ground above buried tanks and piping. It also includes one-time internal UT's of buried steel tanks, a one-time internal UT of the bottom of an outdoor aluminum storage tank, and a one-time visual inspection of the external surface of a buried piping section.

Enhancements

- The program will provide for the performance of one-time internal UT's of buried steel tanks at Dresden and Quad Cities.
- The program will provide for the performance of a one-time UT of the bottom of an outdoor aluminum storage tank at either Dresden or Quad Cities.
- The program will provide for periodic leakage checks of the Quad Cities buried carbon steel fuel oil storage tanks.
- The program will provide for a one-time visual inspection of the external surface of a section of buried ductile iron fire main piping (including a mechanical joint) at Dresden and Quad Cities.

Enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

Dresden and Quad Cities operating history shows no aging degradation induced leakage of buried fuel oil storage tanks or transfer piping. However, other buried piping has had leakage that has required repair. Some examples are:

- Underground piping to the demineralized water storage tank at Dresden developed a leak in 1985 and required excavation and repair.
- Six underground fire main leaks required excavation and repair at Dresden between 1988 and 1997. In none of these cases was the intended function of the fire protection system lost.

Conclusion

The buried piping and tanks inspection program for aging management provides reasonable assurance that the loss of material aging effects are adequately managed so that the intended functions of buried piping and tanks within the scope of license renewal are maintained during the period of extended operation.

B.1.26 ASME Section XI, Subsection IWE

Description

The ASME Section XI, Subsection IWE aging management program provides for primary containment inspections for loss of material. The program includes visual examination and limited surface or volumetric examination, when augmented examination is required. It is implemented through station plans and procedures and covers steel containment shells and their integral attachments; containment hatches and airlocks; seals, gaskets and moisture barriers; and pressure-retaining bolting.

The Quad Cities program complies with Subsection IWE for steel containments (Class MC) of ASME Section XI, 1992 Edition including 1992 Addenda.

The Dresden program utilizes a relief request. The relief request permits utilization of the 1998 Edition of Subsection IWE of ASME Section XI in its entirety instead of the 1992 Edition and Addenda.

Additionally, the protective coatings for these components are monitored as discussed in the Protective Coating Monitoring and Maintenance Program (B.1.32) aging management program. However, no relief from ASME Section XI, Subsection IWE has been requested as a result of implementing the Protective Coating Monitoring and Maintenance Program (B.1.32).

NUREG-1801 Consistency

The ASME Section XI, Subsection IWE aging management program is consistent with the ten elements of aging management program XI.S1, "ASME Section XI, Subsection IWE," specified in NUREG-1801 with the following exceptions.

Exceptions to NUREG-1801

NUREG-1801 indicates that ASME Section XI, Subsection IWE and the additional requirements specified in 10 CFR 50.55a(b)(2) constitute an existing mandated program applicable to managing aging of a steel containment. The NUREG-1801 evaluation covers both the 1992 Edition with the 1992 Addenda and the 1995 Edition with the 1996 Addenda of ASME Section XI, Subsection IWE, as approved in 10 CFR 50.55a. The Dresden program utilizes a relief request. The Dresden program is based on the 1998 Edition of Subsection IWE of ASME Section XI as provided for in Relief Request MCR-02.

NUREG-1801 indicates that pressure retaining weld visual examinations and pressure retaining dissimilar metal welds surface examinations are optional. These requirements are not part of the Dresden program because it is based on the 1998 Edition of Subsection IWE of ASME Section XI as provided for in Relief Request MCR-02.

NUREG-1801 indicates that bolt preload is checked by either a torque or tension test. The Dresden and Quad Cities programs do not provide for checking of bolt preload by either torque or tension test because acceptance is based on Appendix J testing of associated bolted components and general visual examination. This practice is consistent with Dresden Relief Request MCR-02 and Quad Cities Relief Request CR-24.

NUREG-1801 indicates that the program provides for examination of seals, gaskets and moisture barriers by visual methods prescribed in ASME Section XI, Subsection IWE. The Dresden program uses Relief Request MCR-02 and the Quad Cities program uses Relief Request CR-21 as the basis for not routinely inspecting seals and gaskets, and the extent of surface examination of moisture barriers. Aging management program 10 CFR Part 50, Appendix J (B.1.28) provides for monitoring of seals and gaskets. Seals and gaskets are inspected only when sealed or gasketed components are disassembled for maintenance. Moisture barriers, which are accessible, are examined for tears, cracks or other damage that would allow intrusion of moisture, using general visual criteria.

Operating Experience

The operating experience of the inservice inspection (ISI) programs at Dresden and Quad Cities, which includes ASME Section XI, Subsection IWE aging management program activities, has not shown any adverse trend of program performance. Periodic self-assessments of the ISI programs have been performed to identify the areas that need improvement to maintain program quality.

Inspections were conducted on Dresden Unit 3 drywell in response to NRC Generic Letter (GL) 87-05, “Request for Additional Information Assessment of Licensee Measures to Mitigate and or Identify Degradation of Mark I Drywells,” and Information Notice 86-99, “Degradation of Steel Containments,” which addressed the potential for corrosion of boiling water reactor (BWR) Mark I steel drywells in the “sand pocket region.” The results of these inspections and analysis of the results concluded that ultrasonic examinations showed evidence of no apparent corrosion of liner in the "sand pocket region." The conclusions were found to also apply to Dresden Unit 2 and both Quad Cities units, as Dresden Unit 3 conditions were determined to be bounding based on more occasions of moisture in its sand pocket region. Dresden Unit 3 has experienced leakage from the drywell sand pocket drains during refueling outages in 1997 and 2000. As a result, an augmented UT inspection of the Unit 3 drywell sand pocket area is scheduled for the second half of 2002.

Conclusion

The ASME Section XI, Subsection IWE aging management program provides reasonable assurance that the loss of material aging effects are adequately managed so that the intended functions of primary containment components are maintained during the period of extended operation.

B.1.27 ASME Section XI, Subsection IWF

Description

The ASME Section XI, Subsection IWF aging management program provides for visual examination of component and piping supports within the scope of license renewal for loss of material and loss of mechanical function aging effects. The program is implemented through station procedures, which provide for visual examination of inservice inspection Class 1, 2, and 3 supports in accordance with the requirements of ASME Section XI, Subsection IWF, 1989 Edition and Code Case N-491-1.

NUREG-1801 Consistency

With enhancements the ASME Section XI, Subsection IWF aging management program is consistent with the ten elements of aging management program XI.S3, "ASME Section XI, Subsection IWF," specified in NUREG-1801.

Enhancements

- The program will provide for inspection of Class MC component supports consistent with NUREG-1801, Chapter III, Section B1.3.

Enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

The operating experience of the inservice inspection (ISI) programs at Dresden and Quad Cities, which include ASME Section XI, Subsection IWF aging management program activities, has not shown any adverse trend of program performance. Periodic self-assessments of the ISI programs have been performed to identify the areas that need improvement to maintain program quality.

Conclusion

The aging management program provides reasonable assurance that the loss of material and loss of mechanical function aging effects are adequately managed so that the intended functions of component and piping supports within the scope of license renewal are maintained during the period of extended operation.

B.1.28 10 CFR Part 50, Appendix J

Description

The 10 CFR Part 50, Appendix J aging management program provides for aging management of pressure boundary degradation due to loss of material in the primary containment and various systems penetrating primary containment. The program also manages changes in material properties of gaskets, o-rings, and packing materials for the primary containment pressure boundary access points.

The program consists of tests performed in accordance with the regulations and guidance provided in 10 CFR 50 Appendix J, “Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors,” Option B, Regulatory Guide 1.163, “Performance-Based Containment Leak-Testing Program,” NEI 94-01, “Industry Guideline for Implementing Performance-Based Options of 10 CFR Part 50, Appendix J,” ANSI/ANS 56.8, “Containment System Leakage Testing Requirements,” and station procedures. Containment leak rate tests are performed to assure that leakage through the primary containment and systems and components penetrating primary containment does not exceed allowable leakage limits specified in the Technical Specifications. An integrated leak rate test (ILRT) is performed during a period of reactor shutdown at the frequency specified in 10 CFR Part 50, Appendix J, Option B. Local leak rate tests (LLRT) are performed on isolation valves and containment access penetrations at frequencies that comply with the requirements of 10 CFR 50 Appendix J, Option B.

NUREG-1801 Consistency

The 10 CFR Part 50, Appendix J aging management program is consistent with the ten elements of aging management program XI.S4, "10 CFR Part 50, Appendix J," specified in NUREG-1801.

Operating Experience

The industry has found that the 10 CFR Part 50, Appendix J testing program has been effective in maintaining the pressure integrity of the containment boundaries, including identification of leakage within the various systems’ pressure boundaries.

The Dresden and Quad Cities facilities have demonstrated experience in effectively maintaining the integrity of the containment boundaries as evidenced by the selection of Option B of 10 CFR 50 Appendix J leakage testing requirements. Both stations have experienced "as found" LLRT results in excess of individual containment penetration administrative limits. Evaluations were performed and corrective actions were taken to restore the individual penetration leakage rates to within the established administrative leakage limits in accordance with the Appendix J testing program.

Conclusion

The 10 CFR Part 50, Appendix J aging management program provides reasonable assurance that the loss of material and changes in material properties aging effects are adequately managed so that the intended functions of primary containment components within the scope of license renewal are maintained during the period of extended operation.

B.1.29 Masonry Wall Program

Description

The masonry wall program, which is part of the structures monitoring program, is based on guidance provided in I. E. Bulletin 80-11, “Masonry Wall Design," and Information Notice 87-67, “Lessons Learned from Regional Inspections of Licensee Actions in Response to I. E. Bulletin 80-11," and is implemented through station procedures. The program provides for inspections of masonry walls within the scope of license renewal for cracking.

The program includes all masonry walls that perform intended functions in accordance with 10 CFR 54.4, and provides for management of aging effects so that the established evaluation basis for each masonry wall within the scope of license renewal remains valid through the period of extended operation.

NUREG-1801 Consistency

The masonry wall program is consistent with the ten elements of aging management program XI.S5, "Masonry Wall Program," specified in NUREG-1801.

Operating Experience

The masonry wall program has provided for detection of cracks, and other minor aging effects in masonry walls. The corrective action process has ensured timely repair in order to prevent continued degradation. Maintenance history revealed minor degradation of masonry block walls. In response to I. E. Bulletin 80-11, “Masonry Wall Design," and Information Notice 87-67, “Lessons Learned from Regional Inspections of Licensee Actions in Response to I. E. Bulletin 80-11," various actions were taken. Actions included program enhancements, follow-up inspections to substantiate masonry wall analyses and classifications, and the development of procedures for tracking and recording changes to the walls. These actions addressed all concerns raised by I. E. Bulletin 80-11 and Information Notice 87-67, namely unanalyzed conditions, improper assumptions, improper classification, and lack of procedural controls. Operating history shows that the program was and continues to be assessed for its effectiveness based on program specific corrective actions that addressed issues such as inspection schedules and program database discrepancies.

Conclusion

The masonry wall program provides reasonable assurance that the aging effects of cracking are adequately managed so that the intended functions of masonry walls within the scope of license renewal are maintained during the period of extended operation.

B.1.30 Structures Monitoring Program

Description

The structures monitoring program provides for aging management of various structures and external surfaces of mechanical components within the scope of license renewal. The program, which was developed for structures monitoring under 10 CFR 50.65, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” is based on the guidance in Regulatory Guide 1.160 Revision 2, “Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” and NUMARC 93-01 Revision 2, “Industry Guidelines for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” and implemented through procedures. The program is not credited for managing protective coatings.

The program will provide for visual inspections of structures and components not included in the ASME Section XI, Subsection IWF (B.1.27) aging management program.

NUREG-1801 Consistency

With enhancements the structures monitoring aging management program is consistent with the ten elements of aging management program XI.S6, "Structures Monitoring Program," specified in NUREG-1801.

Enhancements

- The program will provide for inspections of structural steel components in secondary containment, flood barriers, electrical panels and racks, junction boxes, instrument racks and panels, offsite power structural components and their foundations, and the Quad Cities discharge canal weir as part of the ultimate heat sink.

- The program will provide for periodic reviews of chemistry data on below-grade water to confirm that the environment remains non-aggressive for the license renewal term for the aging mechanisms of corrosion of embedded steel and aggressive chemical attack of concrete.

- The program will provide for inspection of a sample of non-insulated indoor piping external surfaces at locations immediately adjacent to periodically inspected piping supports.

- Program procedures will reference specific insulation inspection criteria for existing cold weather preparation and inspection procedures for outdoor insulation, and establish new inspections for various indoor area piping and equipment insulation.

- The program will provide for inspection parameter specificity for non-structural joints, roofing, grout pads and isolation gaps.

- The program will extend inspection criteria to the structural steel, concrete, masonry walls, equipment foundations, and component support sections of the program to provide consistency with NUREG-1801 component supports.

Enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

Roof leaks were detected and corrective actions taken for the Dresden turbine building and main control room and for the Quad Cities reactor building and turbine building. Minor degradation of concrete has been detected such as cracks with water stains, pitting, and leaching for various structures including the Dresden reactor building and crib house. Similar degradation has been detected in the Quad Cities reactor building and circulating water intake bays. The degradation was evaluated and dispositioned in accordance with the corrective action process.

Cracks and small gaps were detected in elastomer seals at both Dresden and Quad Cities. Most of the degraded conditions were attributed to man-made occurrences. None were determined to be significant.

Damage and degradation of insulation has been observed and repaired.

Conclusion

The structures monitoring program for aging management provides reasonable assurance that the aging effects are adequately managed so that the intended functions of structures within the scope of license renewal are maintained during the period of extended operation.

B.1.31 RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants

Description

The RG 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants," aging management program is part of the structures monitoring program and consists of procedures that provide for condition monitoring of structural steel elements and concrete. With enhancements the program provides for visual inspections of structural steel and concrete components within the scope of license renewal that are in the Unit 1 and Unit 2 and 3 crib houses at Dresden, and the Unit 1 and 2 crib house and discharge canal weir structure supporting the ultimate heat sink at Quad Cities. The program provides for aging management of concrete and structural steel elements exposed to raw water and aging management of concrete not exposed to raw water, and is based on Regulatory Guide 1.127, Revision 1.

NUREG-1801 Consistency

With enhancements the RG 1.127 aging management program is consistent with the ten elements of aging management program XI.S7, "RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants," specified in NUREG-1801.

Enhancements

- The program will provide for monitoring of crib house concrete walls and slabs with an opposing side in contact with river water and the Quad Cities discharge canal weir.
- Procedures will be revised to emphasize inspecting for structural integrity of concrete and steel components and identify specific types of components to be inspected.

Enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

Operating history of crib houses at Dresden and Quad Cities indicates that structural components are not experiencing any significant degradation. Minor degradation of concrete has been detected such as cracks with water stains, pitting, and leaching. These types of degradation were evaluated and dispositioned.

The effective use of the corrective action process has provided significant quantitative and qualitative data on performance, extent of degradation, and effects of operating and environmental conditions ensuring timely identification and correction of degraded conditions. The program has been assessed for its effectiveness based on program specific corrective actions that addressed issues such as inspection schedules and program database discrepancies.

Conclusion

The RG 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants," aging management program provides reasonable assurance that the aging effects are adequately managed so that the intended functions of concrete and structural steel components in water control structures within the scope of license renewal are maintained during the period of extended operation.

B.1.32 Protective Coating Monitoring and Maintenance Program

Description

The protective coating monitoring and maintenance program provides for aging management of Service Level I coatings inside primary containment. Service Level I coatings are used in areas where the coating failure could adversely affect the operation of post-accident fluid systems and thereby impair safe shutdown.

The program provides for visual inspections to identify any condition that adversely affects the ability of the coating film to function as intended. It is implemented through procedures based on the technical and quality requirements of Regulatory Guide 1.54, Revision 0, “Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants,” and ANSI N101 4-1972, "Quality Assurance for Protective Coatings Applied to Nuclear Facilities," and the guidance provided in EPRI TR-109937, "Guidelines on Nuclear Safety-Related Coating."

NUREG-1801 Consistency

With enhancements the protective coating monitoring and maintenance program is consistent with the ten elements of aging management program XI.S8, "Protective Coating Monitoring and Maintenance Program," specified in NUREG-1801.

Enhancements

- Procedure revisions will provide for thorough visual inspections of Service Level I coatings near sumps or screens associated with the emergency core cooling system.
- Procedure revisions will provide for pre-inspection reviews of previous reports so that trends can be identified.
- Procedure revisions will provide for analysis of suspected reasons for coating failure.

Enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

Examinations of the Dresden internal drywell accessible steel surfaces during refueling outages revealed the original coatings were acceptable other than exhibiting minor surface rust, paint flaking and discoloration. No significant degradation has been identified in the corrective action process records.

The internal surfaces of the torus for each of the Dresden units were re-coated with an epoxy coating in the late 1980’s during refuel outages D2R11 and D3R10. Surveillance of the coated torus internal surfaces during refueling outages has resulted in local coating repairs. A review of past inspections of the torus shells indicates the majority of the problems have been attributed to blistering of coating in small areas, localized pitting, and mechanical damage. Since the application of the epoxy protective coating on the internal surfaces, torus wall thinning has not been an issue.

Inspections of drywell steel at Quad Cities have not identified any significant coating or corrosion problems requiring repair of the torus. In 1994 the Quad Cities Unit 1 torus internal surface corrosion was removed and the base metal was re-coated. During the subsequent refueling outage the torus shell immersion area was inspected. Coating deficiencies, such as mechanical damage, burrs and projections were identified and repaired.

Minor local repairs to the coating on the inside of the Quad Cities Unit 2 torus were performed in March 1974. Inspections are conducted each outage, with local coating repairs performed as required.

Conclusion

The protective coating monitoring and maintenance program provides reasonable assurance that aging effects are adequately managed so that the intended functions of Service Level 1 coatings inside primary containment are maintained during the period of extended operation.

B.1.33 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements

Description

The aging management program for electrical cables and connections not subject to 10 CFR 50.49 environmental qualification requirements manages cables and connections within the scope of license renewal that are subject to an adverse environment. It also identifies and manages cables and connections subject to an adverse localized environment. The aging management program for electrical cables and connections not subject to 10 CFR 50.49 environmental qualification requirements is a new program.

An adverse localized environment is a condition in a limited plant area that is significantly more severe than the specified service environment for a subject cable or connection. An adverse variation in environment is significant if it could appreciably increase the rate of aging of a component or have an immediate adverse effect on operability.

Cables and connections subject to an adverse environment are managed by inspection of a sample of these components. Selected cables and connections from accessible areas are inspected and represent, with reasonable assurance, the cables and connections in adverse environments. They are inspected for signs of accelerated age-related degradation. Additional inspections, repair or replacement are initiated as appropriate.

Samples of cables and connections found to be located in adverse localized areas will be inspected prior to the period of extended operation, with an inspection frequency of at least once every 10 years. The scope of this program includes inspections of power, control and instrumentation cables and connections located in adverse localized areas, including the cables used in instrumentation circuits that are sensitive to reduction in conductor insulation resistance.

NUREG-1801 Consistency

The aging management program for electrical cables and connections not subject to 10 CFR 50.49 environmental qualification requirements is a new program. The program is scheduled for implementation prior to the period of extended operation. Program activities are consistent with the ten elements of aging program XI.E1, "Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements," specified in NUREG-1801.

Operating Experience

This program is new. Therefore, no programmatic operating experience is available. However, existing activities provide for inspection of butyl rubber insulated environmentally qualified cables in heater bays to assess aging of cable insulation. These cables are in an adverse environment. No adverse trends indicative of premature aging of cables have been identified. Cable failures, when identified, are subject to the station corrective action program. Operating experience does not indicate the presence of localized adverse environment or premature aging of cable insulation.

Conclusion

The aging management program for electrical cables and connections not subject to 10 CFR 50.49 environmental qualification requirements provides reasonable assurance that aging effects are adequately managed so that the intended functions of these types of cables and connections are maintained during the period of extended operation.

B.1.34 Metal Fatigue of Reactor Coolant Pressure Boundary

Description

The aging management program for metal fatigue of reactor coolant pressure boundary provides for monitoring fatigue stress cycles to ensure that the design fatigue usage factor limit is not exceeded.

NUREG-1801 Consistency

With enhancements the metal fatigue of reactor coolant pressure boundary aging management program is consistent with the ten elements of aging management program X.M1, "Metal Fatigue of Reactor Coolant Pressure Boundary," specified in NUREG-1801.

Enhancements

- The program will use the EPRI-licensed FatiguePro® cycle counting and fatigue usage factor tracking computer program. The computer program provides for calculation of stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles and automated calculation and tracking of fatigue cumulative usage factors.

- The program will provide for calculating and tracking of the cumulative usage factors for bounding locations for the reactor pressure vessel, Class I piping, the torus, torus vents, and torus attached piping and penetrations.

- The program will provide for tracking of fatigue stress cycles for the Dresden isolation condenser.

The enhancements are scheduled for implementation prior to the period of extended operation.

Operating Experience

The reactor vessel cycle counting programs have been revised to incorporate changes in design basis analysis cycles. The changes were made because certain types of operating events were found to be more frequent than anticipated in the original design. Others were found to be less frequent. The changes reduced the assumed design basis number of the less-frequent events and increased the assumed number of the more-frequent events.

In response to NRC concerns that early-life operating cycles at some units had caused fatigue usage factors to increase at a greater rate than anticipated in the design analyses the industry sponsored the development of the FatiguePro® computer program. The program is designed to ensure that the code limits are not exceeded for the remainder of each unit's licensed life and provides for incorporation of operating experience.

Conclusion

The aging management program for metal fatigue of reactor coolant pressure boundary provides reasonable assurance that the thermal and pressure transient aging effects are adequately managed so that the intended functions of pressure boundary components within the scope of license renewal that are covered by this program are maintained during the period of extended operation.

B.1.35 Environmental Qualification (EQ) of Electrical Components

Description

The environmental qualification (EQ) of electrical components aging management program is implemented through station procedures and predefined tasks. The Dresden and Quad Cities EQ program complies with 10 CFR 50.49, “Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants.” All EQ equipment is included within the scope of license renewal. The program provides for maintenance of the qualified life for electrical equipment important to safety within the scope of 10 CFR 50.49. Program activities establish, demonstrate, and document the level of qualification, qualified configuration, maintenance, surveillance and replacement requirements necessary to meet 10 CFR 50.49. Qualified life is determined for equipment within the scope of the EQ program and appropriate actions such as replacement or refurbishment are taken prior to or at the end of the qualified life of the equipment so that the aging limit is not exceeded.

NUREG-1801 Consistency

The environmental qualification (EQ) of electrical components aging management program is consistent with the ten elements of aging management program X.E1, “Environmental Qualification (EQ) of Electrical Components,” specified in NUREG-1801.

Operating Experience

The environmental qualification (EQ) of electrical components aging management program provides for consideration of operating experience to reconcile qualification bases and conclusions, including the equipment qualified life. Operating experience and system, equipment or component related information, as reported through NRC Bulletins, Notices, Circulars, Generic Letters and Part 21 Notifications, are evaluated for applicability. The evaluations are documented and corrective actions are identified.

The EQ Program requires inclusion of changes mandated by closure of Generic Safety Issue 168 (GSI-168), “Environmental Qualification of Electrical Equipment.” The NRC letter to NEI of June 2, 1998 states that no additional information is required to address GSI-168 in a license renewal application at this time.

Operating experience has demonstrated that the program manages aging as required by 10 CFR 50.49. When problems have been identified through industry or plant-specific experience, corrective actions have been taken to prevent recurrence.

Conclusion

The environmental qualification (EQ) of electrical components aging management program provides reasonable assurance that aging effects are adequately managed so that the intended functions of components within the scope of 10 CFR 50.49 are maintained during the period of extended operation.

B.1.36 Boraflex Monitoring (Quad Cities Only)

Description

The Quad Cities Boraflex monitoring program is based on EPRI TR-108761, "A Synopsis of the Technology Developed to Address the Boraflex Degradation Issue." (Note: The Boraflex monitoring aging management program is not applicable to Dresden because the station utilizes Boral as the neutron absorbing material in the spent fuel racks rather than Boraflex.)

The Quad Cities Boraflex monitoring program consists of condition monitoring activities based on the maintenance rule and implemented at a predefined frequency. Station procedures provide for inspection testing and analysis of the Boraflex neutron absorbing capability to assure that the 5% subcriticality margin is maintained. Degradation monitoring is accomplished by obtaining a computer-generated (RACKLIFE) value of boron loss, which is evaluated against the acceptance criteria. The evaluation is performed every two years. The RACKLIFE program was validated through neutron attenuation testing (blackness testing), and boron areal density testing using the BADGER device. Between evaluations, spent fuel pool silica levels are monitored and adverse trends identified through chemistry programs.

NUREG-1801 Consistency

The Boraflex monitoring aging management program for Quad Cities is consistent with the ten elements of aging management program XI.M22, "Boraflex Monitoring," specified in NUREG-1801.

Operating Experience

The Quad Cities Boraflex monitoring aging management program has been effective at determining boron loss. To date, a boron loss of approximately one percent has been identified and trended. A review of the program resulted in updating the RACKLIFE program every two years to be consistent with maintenance rule requirements.

Conclusion

The Quad Cities Boraflex monitoring aging management program provides reasonable assurance that the aging effects are adequately managed so that the 5% subcriticality margin of the Boraflex is maintained during the period of extended operation.

B.2 PLANT-SPECIFIC AGING MANAGEMENT PROGRAMS

B.2.1 Corrective Action Program

Description

The applicable elements of the Exelon Quality Assurance Program (QAP), implemented at Dresden and Quad Cities, were used for those aspects of the Aging Management Review process that affect the quality of safety related structures, systems, and components. The Exelon QAP is based on the 18 point criteria set forth in 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." For non-safety related structures and components subject to an AMR, the same existing 10 CFR Part 50, Appendix B based QA program was used to address the elements of corrective actions, confirmation process, and administrative controls. The QAP addresses all aspects of quality assurance at Dresden and Quad Cities and is described in Exelon Generation Company, LLC, QATR EGC-1A, Revision 69, which endorses ASME NQA-1-1989 as the quality standard.

The elements of the QAP most pertinent to aging management programs credited for license renewal are corrective action, confirmation process, and administrative controls. The confirmation process ensures that corrective actions to prevent recurrence of significant conditions are adequately completed and are effective. Administrative controls provide a review and approval process.

NUREG-1801 describes how a license renewal applicant may rely on the existing requirements in 10 CFR Part 50, Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to satisfy these program elements/attributes. Branch Technical Position IQMB-1 describes an acceptable process for implementing the corrective actions, confirmation process, and administrative controls elements of aging management programs for license renewal. The requirements of the corrective action, confirmatory process, and administrative control elements will be applied to all structures, systems and components (SSC) within the scope of license renewal.

A review of the elements that make up a sound AMP provided in NUREG-1800 have been used to evaluate the Exelon QAP. The results are provided below. The same model was used to evaluate each of the Aging Management Programs credited for license renewal provided in Appendix B of this application.

Evaluation and Technical Basis

(1) Scope of Program: The Exelon program applies without limitation to safety related or ASME code structures, systems, components, and activities. It also applies to certain non-safety related structures, systems, components and activities to a degree consistent with their importance to safety.

The plant corrective action process applies to all plant systems, structures and components (both safety related and non-safety related) within the scope of license renewal. Administrative controls are in place for existing aging management programs and activities. Administrative controls will also be applied to new and enhanced programs and activities as they are implemented. As a minimum, these programs and activities are or will be performed in accordance with written procedures that are or will be reviewed and approved in accordance with the QAP.

(2) Preventive Actions: The corrective action program provides a means to correct conditions identified as being adverse to quality. There are no preventive or mitigative attributes specifically credited for the corrective action program.

(3) Parameters Monitored/Inspected: No specific parameters are inspected or monitored as part of the corrective action program. Generally, when parameters inspected or monitored by other plant programs indicate a condition adverse to quality, the corrective action program provides a means to correct the identified condition.

(4) Detection of Aging Effects: Detection of aging effects is not part of the corrective action program. The corrective action program provides a means to address aging effects identified by other aging management activities.

(5) Monitoring and Trending: The corrective action process is monitored to ensure that corrective actions taken are timely. Significant and non-significant conditions are trended. Significant conditions that are adverse to quality are monitored and require formal cause determination and corrective actions to prevent recurrence. The corrective actions to prevent recurrence are monitored for effectiveness.

(6) Acceptance Criteria: The corrective action program does not include specific acceptance criteria for in scope components. Generally, when the acceptance criteria of other aging management activities are not met, the corrective action program provides a means to ensure appropriate corrective actions are taken.

(7) Corrective Actions: Corrective action is initiated following the identification of conditions adverse to quality, and is documented. The corrective action program is described in Chapter 16 of the QAP. The various components of the corrective action program provide for timely actions, including determination of the cause of the condition and corrective action taken to preclude recurrence for significant conditions adverse to quality. Condition reports are analyzed for adverse trends. Identified adverse trends are reported to the appropriate manager and documented on a condition report.

(8) Confirmation Process: Condition reports are reviewed by supervisors. Operations shift management is contacted as necessary to discuss potential operability or regulatory reportability of the condition. Items determined to be significant conditions adverse to quality are reported to the appropriate levels of management. An effectiveness review is completed for root cause analysis corrective actions to prevent recurrence.

(9) Administrative Controls: Activities affecting quality are prescribed by documented instructions, procedures, drawings, or specifications of a type appropriate to the circumstances and are accomplished in accordance with these instructions, procedures, drawings or specifications. They contain appropriate acceptance criteria and documentation requirements for determining whether important activities have been satisfactorily accomplished. The document control process is described in Chapter 6 of the QAP.

(10) Operating Experience: The corrective action program provides for evaluation of aging effects and significant operating events and requires that reasonable actions be taken to enhance programs and activities to prevent future occurrences. The other aging management programs described in the Dresden and Quad Cities LRA Appendix B provide examples of the corrective action program being used to address and correct aging related conditions adverse to quality.

Conclusion

Exelon has established a quality assurance program that is implemented at Dresden and Quad Cities. The QAP is based on the criteria set forth in 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The QAP addresses all aspects of quality assurance at Dresden and Quad Cities.

Based on the application of industry standards and Dresden and Quad Cities operating experience, there is reasonable assurance that the corrective action program will adequately manage the aging effects so that intended functions will be maintained consistent with the current licensing basis for the period of extended operation.

B.2.2 Periodic Inspection of Non-EQ, Non-Segregated Electrical Bus Ducts

Description

This program inspects the non-segregated bus ducts that connect the reserve auxiliary transformers to the 4160V ESF buses. They are normally energized, and therefore, the bus duct insulation material will experience temperature rise due to energization, which may cause age-related degradation during the extended period of operation. These bus ducts are in scope of license renewal but are not subject to 10 CFR 50.49 environmental qualification. These non-EQ, non-segregated bus ducts will therefore be inspected periodically during the period of extended operation. This inspection program considers the technical information and guidance provided in IEEE Standard P1205, "IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations," SAND 96-0344, "Aging Management Guidelines for Commercial Nuclear Power Plants – Electrical Cable and Terminations," and EPRI-109619, "Guideline for the Management of Adverse Localized Equipment Environments."

The non-segregated bus duct internal components and materials are visually inspected under station inspection procedures for signs of aging degradation that indicate possible loss of insulating function. Repair or rework is initiated as required to maintain the operating functions of the bus ducts.

This is a new program and will be implemented prior to the period of extended operation.

Evaluation and Technical Basis

(1) Scope of Activity: This inspection program applies to the normally-energized non-segregated bus ducts within the scope of license renewal, not subject to the environmental qualification requirements of 10 CFR 50.49, which can be affected by elevated temperatures prior to the end of the extended period of operation.

(2) Preventive Actions: This is an inspection program only. This program does not prevent or mitigate aging degradation.

(3) Parameters Monitored/Inspected: Accessible normally-energized non-segregated bus duct internal components are visually inspected for insulation material surface anomalies, such as embrittlement, discoloration, cracking, chipping, or surface contamination.

(4) Detection of Aging Effects: Normally-energized non-segregated bus ducts that connect the reserve auxiliary transformers to the 4160V ESF buses are inspected at least once every 10 years for material surface anomalies which are precursors to any onset of insulation failure due to temperature or radiation degradation. Experience has shown that aging degradation is a slow process. This frequency is therefore adequate to preclude age-related failures of the conductor insulation.

(5) Monitoring and Trending: Trending is not included in this activity because the parameters inspected are difficult to quantify. The 10-year inspection frequency will however provide at least 2 data points within 20 years, which will permit some characterization of the rate of degradation.

(6) Acceptance Criteria: The accessible normally-energized non-segregated bus ducts are to be free from unacceptable visual indications. Unacceptable visual indications are duct insulation material surface anomalies which suggest that bus duct insulation degradation exists, which if left unmanaged, could lead to a loss of the intended function, as determined by an engineering evaluation.

(7) Corrective Actions: Further investigation and evaluation are performed when the acceptance criterion is not met in order to ensure that the intended functions will be maintained consistent with the current licensing basis. Corrective actions may include but are not limited to increased inspection frequency, replacement, or rework of the affected bus duct insulation components.

(8) Confirmation Process: Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B.

(9) Administrative Controls: See Item 8, above.

(10) Operating Experience: No age-related bus duct insulation failures that would indicate aging to be a concern at Dresden or Quad Cities have been identified. However, industry experience indicates that high temperatures may cause degradation of electrical insulation materials. Some visual surface indications of high-temperature degradation in bus duct electrical insulation, such as color changes or surface cracking, have been observed in the industry.

Conclusion

The inspection program for the non-EQ, non-segregated electrical bus ducts that connect the reserve auxiliary transformers to the 4160V ESF buses will provide reasonable assurance that the intended function of the non-EQ, non-segregated bus ducts will be maintained consistent with the current licensing basis for the period of extended operation.

B.2.3 Periodic Inspection of Ventilation System Elastomers

Description

NUREG-1801 Sections V.B1, V.B2, VII.F1, VII.F3, and VII.F4 state that ventilation system elastomers used for flexible boots, access door seals, and filter seals are susceptible to hardening and loss of strength, and loss of material aging effects. The NUREG-1801 aging management program (AMP) column for these sections states that a plant-specific aging management program is to be evaluated.

The improved program for periodic inspection of ventilation system elastomers provides routine inspection of certain elastomers in ventilation systems in accordance with plant procedures and predefined tasks.

Dresden and Quad Cities perform routine inspections of elastomer seals and flexible boots installed in the standby gas treatment and main control room ventilation systems in accordance with approved plant procedures and predefined tasks. Dresden currently performs partial inspections of the reactor building ventilation and the station blackout and emergency diesel generator building ventilation system elastomer seals. Access door seals are inspected during the periodic damper inspections.

The improved program includes inspections for cracking, loss of material, or other evidence of aging of all flexible boots, access door seals and gaskets, filter seals and gaskets, and RTV silicone used as a duct sealant, in the components of these systems that are within the scope of license renewal. The improved program tests seals for hardening if evidence of aging is found.

Improvements are scheduled for implementation prior to the period of extended operation.

Evaluation and Technical Basis

(1) Scope of Activity: The program inspects elastomers utilized in ventilation systems within the scope of license renewal, including flexible boots, access door seals, filter seals, and RTV used as a duct sealant. These elastomers prevent external leakage and bypass of HEPA and carbon filters. These inspections apply to the standby gas treatment system and ventilation systems within the scope of license renewal; that is, to the main control room ventilation, station blackout diesel generator building ventilation, Dresden reactor building ventilation, and Quad Cities emergency diesel generator building ventilation systems.

Exelon may elect to periodically replace certain ventilation system elastomer and RTV seals, instead of inspecting them. Periodic replacement will be evaluated on a case-by-case basis.

(2) Preventive Actions: The ventilation system elastomer inspections do not provide any preventative actions. The inspections provide condition monitoring to detect degradation prior to a loss of function.

(3) Parameters Monitored/Inspected: Visual inspections of elastomers used in ventilation systems determine if aging degradation is occurring. Flexible boots, access door seals and gaskets, filter seals and gaskets, and RTV used as a duct sealant are inspected to ensure they are free of cracking, loss of material, and damage. The seals will be tested for hardening if cracking or loss of material is noted. For the standby gas treatment and main control room ventilation systems, the results of the elastomer inspections are verified by the performance of system leakage tests and filter efficiency tests.

(4) Detection of Aging Effects: The elastomer inspections are performed at intervals sufficient to detect aging prior to the equipment failing a leakage test or filter efficiency test. The seals will be inspected for hardening if cracking or loss of material is observed. See Item 10, below.

(5) Monitoring and Trending: The conditions of the elastomers used in ventilation systems are monitored, but not trended. Flexible boots, filter seals, and access door seals and gaskets are repaired or replaced if damage or deterioration is detected.

(6) Acceptance Criteria: Elastomers are inspected for cracking, loss of material, and damage. The seals will be inspected for hardening if cracking or loss of material is observed. The seals are repaired or replaced if a degraded condition is found. Surveillance tests of the standby gas treatment and main control room ventilation systems ensure that system leakage meets the requirements of the current licensing basis.

(7) Corrective Actions: If cracking, loss of material, or damage are observed; or if a hardness test does not meet the acceptance criterion, a condition report is initiated to document the concern in accordance with plant administrative procedures. The corrective actions program ensures that conditions adverse to quality are promptly corrected. If the deficiency is found to be significantly adverse to quality, the cause of the condition is determined and an action plan is developed to preclude recurrence.

(8) Confirmation Process: Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B.

(9) Administrative Controls: See Item 8, above.

(10) Operating Experience: Dresden and Quad Cities have experienced leaks in ventilation systems due to deterioration of or damage to elastomers, including flexible boots and access door seals and gaskets. The leaks were found and corrected in a timely manner and did not result in a loss of function of the ventilation system train.

Conclusion

Inspections of ventilation system elastomers will be enhanced to include all flexible boots, access door seals and gaskets, and filter seals and gaskets of systems within the scope of license renewal. This enhancement will provide assurance that the ventilation system elastomers are routinely inspected for deterioration and damage, and will adequately manage the elastomer aging effects. The program provides reasonable assurance that intended functions are maintained consistent with the current licensing basis during the period of extended operation.

B.2.4 Periodic Testing of Drywell and Torus Spray Nozzles

Description

The periodic tests of drywell and torus spray nozzles address a NUREG-1801 Section V.D2.5 concern that flow orifices and spray nozzles in the drywell and torus spray subsystems are subject to plugging by rust from carbon steel piping components, and therefore a plant-specific aging management program is to be evaluated.

The Dresden and Quad Cities drywell and torus spray nozzles are bronze. There are no carbon steel flow orifices in the system piping, within the scope of license renewal. However, upstream carbon steel piping is subject to possible general corrosion. These periodic tests use approved plant procedures to verify that the drywell and torus spray nozzles are free from plugging that could result from corrosion product buildup from upstream sources.

Evaluation and Technical Basis

(1) Scope of Activity: The tests include the drywell and torus spray nozzles.

(2) Preventive Actions: The spray nozzle tests do not provide any preventative actions. The spray nozzle tests provide condition monitoring to detect the degradation prior to a loss of function.

(3) Parameters Monitored/Inspected: The flow tests demonstrate that the drywell and torus spray nozzles are not blocked by debris or corrosion products, and thereby demonstrate that the nozzles are available to provide the drywell and torus steam quenching functions. Drywell nozzles are tested with compressed air. Torus nozzles are tested with water. Test procedures require that flow be demonstrated through each individual nozzle.

(4) Detection of Aging Effects: The periodic drywell and torus spray nozzle flow tests detect plugging by corrosion products from the degradation of carbon steel piping and fittings.

(5) Monitoring and Trending: The results of the spray nozzle tests are monitored but are not trended. If flow to a nozzle is blocked or restricted the degraded condition is evaluated and corrective actions are taken to restore normal flow.

(6) Acceptance Criteria: The test procedures contain acceptance criteria that require that flow be observed from and through each individual drywell and torus spray nozzle. The acceptance criteria provide assurance that flow to the drywell and torus spray headers and spray nozzles is not blocked or restricted.

(7) Corrective Actions: If a test shows that flow to a nozzle is blocked or restricted, a condition report is initiated to document the concern in accordance with plant administrative procedures. The degraded condition is evaluated and corrective actions are taken to restore normal flow. The corrective actions program ensures that conditions adverse to quality are promptly corrected. If the deficiency is found to be significantly adverse to quality, the cause of the condition is determined and an action plan is developed to preclude recurrence.

(8) Confirmation Process: Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B.

(9) Administrative Controls: See Item 8, above.

(10) Operating Experience: Dresden has not detected any degradation of the drywell and torus spray headers or spray nozzles. Quad Cities has experienced two events in which foreign material was found in the spray nozzles. In 1998, small amounts of rust were found in some nozzles after a flow test. However, the small amounts of rust found did not pose a blockage problem. In 2000, a 1” X 3” block of wood was found lodged in a spray nozzle subsequent to a spray test, but this was a foreign material exclusion problem unrelated to aging. No rust was found in the spray nozzles during the 2000 test.

The Dresden and Quad Cities operating experience demonstrates that the periodic flow tests effectively manage drywell and torus spray header and spray nozzle plugging by corrosion products, so that the intended function of providing a quenching spray will be maintained during the period of extended operation.

Conclusion

The periodic drywell and torus spray nozzle flow tests effectively manage drywell and torus spray header and spray nozzle plugging by corrosion products. The program provides reasonable assurance that intended functions are maintained consistent with the current licensing basis during the period of extended operation.

B.2.5 Lubricating Oil Monitoring Activities

Description

The lubricating oil monitoring activities manage loss of material and cracking in lubricating oil heat exchangers in the scope of license renewal. These activities include measures to minimize corrosion and to mitigate loss of material and cracking in heat exchangers by monitoring lubricating oil properties. Sampling, testing, and trending verify lubricating oil properties and ensure that the intended functions of the heat exchangers are not lost. Oil analysis permits identification of specific wear mechanisms, contamination, and oil degradation within operating machinery.

The activities manage physical and chemical properties in lubricating oil. The complete aging management program for lubricating oil heat exchangers also includes secondary-side (heat sink) chemistry controls, performance monitoring, and inspections. Those portions of the lubricating oil heat exchanger management program are described in:

- Section B.1.14, Closed-Cycle Cooling Water System, for the diesel generator and station blackout diesel generator oil coolers; and in
- Section B.2.6, Heat Exchanger Test and Inspection Activities, for the HPCI oil coolers.

Evaluation and Technical Basis

(1) Scope of Activity: The following lubricating oil heat exchangers are subject to this program:

- Dresden Unit 2 and 3 HPCI lubricating oil coolers
- Dresden Unit 2, 3 and 2/3 diesel generator lubricating oil coolers
- Dresden Units 2 and 3 station blackout (SBO) diesel generator lubricating oil coolers
- Quad Cities Unit 1 and 2 HPCI lubricating oil coolers
- Quad Cities Unit 1, 2, and 1/2 diesel generator lubricating oil coolers
- Quad Cities Unit 1 and 2 station blackout (SBO) diesel generator lubricating oil coolers

(2) Preventive Actions: Monitoring and control of oil impurities and properties mitigates loss of material and cracking in lubricating oil systems.

(3) Parameters Monitored/Inspected: The program includes specifications for known oil degradation indicators and degradation characteristics, sampling and analysis frequencies, and corrective actions for control of lubricating oil properties. Lubricating oil physical properties are tested to standard ASTM and ISO methods, for the applicable oil type, to provide accurate quantitative numbers with repeatable results. Samples are taken monthly for emergency diesel generators, and quarterly for HPCI and SBO diesel generators. Surveillance testing and operational surveillances verify proper heat exchanger performance to support associated system operability.

Oil is analyzed for indications of degraded chemical and physical properties depending on oil type and type of service. Analyses include:

- Chemical parameters and viscosity, total acid number, total base number, rotary bomb oxidation test, water demulsability, particle count, fuel and combustion by-products, sediment, water, anti-foaming characteristics, whole particle counting, air release and emission spectrum.

Normal, alert, and fault levels for oil chemical and physical properties, wear metals, contaminants, and additives are established for the specific oil type and application.

(4) Detection of Aging Effects: Monitoring activities maintain lubricating oil properties within predefined limits to both mitigate and detect the effects of aging. Oil analysis has become an accurate method for identifying specific wear mechanisms, contamination, and oil degradation characteristics within operating machinery. The program includes normal, alert, and fault action levels for oil chemical and physical properties, wear metals, contaminants, and additives, for the specific oil type and application. Increased impurities and degradation of oil properties indicate degradation of materials in lubricating oil systems. Monitoring of the diagnostic parameters indicates degradation due to aging effects prior to loss of intended function.

Samples are taken monthly for emergency diesel generators, and quarterly for HPCI and SBO diesel generators. Sampling frequency is increased if plant and equipment operating conditions indicate a need to do so.

(5) Monitoring and Trending: See Items 3 and 4, above for parameters and frequencies. The lubricating oil analysis results are evaluated for acceptability, and are trended and evaluated using computer software and a database.

(6) Acceptance Criteria: Normal, alert, and fault levels have been established for the various chemical and physical properties, wear metals, additives, and contaminant levels based on information from oil manufacturers, equipment manufacturers, and industry guidelines, for the specific oil type and application. The program maintains contaminant and parameter limits within the application-specific limits. The procedures outline potential actions to be taken at alert and fault levels, and actions can be chosen based on the level of deviation. Aging effects or unacceptable results are evaluated and appropriate corrective actions are taken.

(7) Corrective Actions: Lubricating oil chemical and physical test results or contaminants outside the allowable limits are returned to the acceptable range within reasonable time periods as identified in industry guidelines. Evaluations are performed for test or inspection results that do not satisfy established criteria and a condition report is initiated to document the concern in accordance with plant administrative procedures. The corrective actions program ensures that conditions adverse to quality are promptly corrected. If the deficiency is found to be significantly adverse to quality, the cause of the condition is determined and an action plan is developed to preclude recurrence.

(8) Confirmation Process: Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B.

(9) Administrative Controls: See Item 8, above.

(10) Operating Experience: The overall effectiveness of lubricating oil monitoring activities is indicated by the Dresden and Quad Cities operating experience. Lubricating oil sampling and analysis have detected particulate or water contamination (or both) in lubricating oil systems. In some cases these events resulted in systems being declared inoperable until repaired, and until the oil was flushed or replaced. Operating experience has produced procedure and program changes, which have improved the effectiveness of lubricating oil testing and inspection activities.

Conclusion

The lubricating oil preventive, inspection, and testing activities mitigate, detect, monitor, and trend the effects of loss of material and cracking in lubricating oil coolers. The program provides reasonable assurance that intended functions are maintained consistent with the current licensing basis during the period of extended operation.

B.2.6 Heat Exchanger Test and Inspection Activities

Description

The heat exchanger test and inspection activities provide condition monitoring, inspection, and performance testing. The activities manage loss of material, cracking, and buildup of deposits in heat exchangers in the scope of license renewal that are not tested and inspected by the Open-Cycle Cooling Water System (B.1.13) and Closed-Cycle Cooling Water System (B.1.14) aging management programs. The augmentation activities identified in NUREG-1801, lines IV.C1.4-a and IV.C1.4-b to manage loss of material and cracking for the Dresden isolation condensers are included in this aging management program.

The heat exchanger test and inspection activities are new and will be implemented prior to the period of extended operation.

Surveillance testing verifies that heat exchanger performance is adequate to support system operability requirements. Inspections and in-service nondestructive examinations (ISI, NDE) detect aging effects. Tests and inspections, and monitoring and trending of test results, confirm that aging effects are managed and that system and component functions are maintained.

The isolation condenser test and inspection augmentation activities detect cracking due to stress corrosion cracking or cyclic loading, and detect loss of material due to pitting and crevice corrosion. These augmentation activities are not part of the ISI program, but verify that the ISI program is effective for ensuring that significant degradation is not occurring, and therefore that the intended function of the isolation condenser is maintained during the extended period of operation. These augmentation activities consist of temperature and radioactivity monitoring of the shell-side (cooling) water, and eddy current testing of tubes.

The inservice inspection, water chemistry management and lubricating oil management activities applied to the heat exchangers in the scope of this aging management program are described in the following program evaluations.

- The inservice inspection is described in the ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD (B.1.1) aging management program.
- Management of water chemistry is described in the Water Chemistry (B.1.2) aging management program.
- Management of oil properties for the HPCI lubricating oil coolers is described in the Lubricating Oil Monitoring Activities (B.2.5) plant-specific aging management program.

Evaluation and Technical Basis

(1) Scope of Activity: The following heat exchangers are subject to these test and inspection activities:

- Dresden Unit 2 and 3 HPCI lubricating oil coolers
- Dresden Unit 2 and 3 HPCI gland seal condensers
- Dresden main control room air handling unit heat exchanger
- Dresden Unit 2 and 3 isolation condensers
- Quad Cities Unit 1 and 2 HPCI lubricating oil coolers
- Quad Cities Unit 1 and 2 HPCI gland seal condensers
- Quad Cities main control room air handling unit heat exchanger
- Quad Cities Unit 1 and 2 battery/station blackout room HVAC heat exchangers

(2) Preventive Actions: These heat exchanger test and inspection activities do not provide any preventive actions. These activities provide condition monitoring to detect degradation prior to a loss of function.

(3) Parameters Monitored/Inspected: Performance tests verify system operability by verifying proper fluid flows, or temperatures, or differential pressures during system operation under load. Inspection activities monitor the effects of corrosion and buildup of deposits. Periodic inspections and NDE tests may consist of visual inspections, eddy current testing, and ultrasonic tests or radiography to detect loss of material, cracking, or buildup of deposits. Temperature and radioactivity are monitored in the Dresden isolation condenser shell side water. Temperature monitoring of the Dresden isolation condenser is accomplished through proceduralized operational checks. Radioactivity monitoring of the Dresden isolation condenser is through periodic sampling.

(4) Detection of Aging Effects: Loss of material, cracking, or buildup of deposits would result in degradation of heat exchanger or system performance. The frequency and extent of inspections and testing ensure detection of aging effects before the loss of intended function of the heat exchanger or associated system. Heat removal capabilities are generally verified by performing system operability testing. Heat exchanger inspections are generally conducted at 10-year intervals or less. Shorter intervals are based on industry guidelines or plant operating experience. Eddy current testing is to be performed at least once every 10 years, and the procedure provides for increasing the inspection frequency based on the indications.

After initial inspection, subsequent inspection frequencies will be based on the as-found condition of the equipment. The inspection and testing intervals may be adjusted on the basis of the results of the reliability analysis, type of service, frequency of operation, or age of components and systems.

Temperature and radioactivity monitoring of the Dresden isolation condenser shell side water will detect tube-to-shell-side leaks.

(5) Monitoring and Trending: Heat transfer testing results are documented in plant test procedures, and trended and reviewed by the appropriate group. Isolation condenser temperatures are recorded in the operator’s surveillance log. Radiation monitoring of the isolation condensers is conducted by procedure.

(6) Acceptance Criteria: Specific acceptance criteria are provided in the inspection or test procedures, as required to ensure continued system and component operability. System functional testing must confirm the systems' ability to meet minimum Technical Specification requirements. EPRI guidance is used to determine allowable percent wall loss and plugging criteria, and for projections of remaining life. Indications of degradation are evaluated to determine if the material condition will maintain the system intended function prior to returning the system to operable status. Engineering evaluations are performed if evidence of aging is found, and determine if there is a need to alter fluid chemistry, replace tubing, or impose an additional aging management activity.

(7) Corrective Actions: See Item 6, above. Evaluations are performed for test or inspection results that do not satisfy established criteria and a condition report is initiated to document the concern in accordance with the corrective action program. The corrective action program ensures that conditions adverse to quality are promptly corrected. If the deficiency is found to be significantly adverse to quality, the cause of the condition is determined and an action plan is developed to preclude recurrence.

(8) Confirmation Process: Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B.

(9) Administrative Controls: See Item 8, above.

(10) Operating Experience: This is a new aging management program. No program operating experience exists at this time. However, similar controls implemented for the GL 89-13, “Service Water System Problems Affecting Safety-Related Equipment,” program have been effective in detecting aging effects in heat exchangers. Instances of loss of material, cracking, and buildup of deposits in heat exchangers have been detected in Dresden and Quad Cities heat exchangers prior to loss of system intended functions.

Conclusion

The method, extent and schedule of these heat exchanger inspection and test activities provide reasonable assurance that loss of material, cracking, and buildup of deposits in heat exchangers would be detected prior to loss of their intended function. The program provides reasonable assurance that intended functions are maintained consistent with the current licensing basis during the period of extended operation.

B.2.7 Generator Stator Water Chemistry Activities (Quad Cities Only)

Description

Quad Cities generator stator water chemistry activities manage aging by monitoring and controlling water chemistry per established procedures.

Evaluation and Technical Basis

(1) Scope of Activity: Stator cooling water is continuously monitored for purity by a conductivity cell. The conductivity cell annunciates an alarm if conductivity increases to a predetermined limit. Water chemistry is maintained in accordance with General Electric guidelines for stator cooling water systems. The system is equipped with a resin bed that continuously filters a portion of the system flow.

(2) Preventive Actions: Stress corrosion cracking (SCC) of stator cooling water components is unlikely as contaminants are maintained at very low levels, in accordance with General Electric guidelines, and the system is normally operated at temperatures less than 140°F. Quad Cities procedures provide a feed-and-bleed operation if the dissolved oxygen concentration approaches predetermined limits.

(3) Parameters Monitored/Inspected: Periodic sampling is not necessary as water conductivity is continuously monitored to ensure purity. Quad Cities procedures provide for periodically logging of water chemistry data.

(4) Detection of Aging Effects: This program mitigates loss of material aging effects. It is not credited for detection of aging effects.

(5) Monitoring and Trending: See item 3, above.

(6) Acceptance Criteria: See item 2, above.

(7) Corrective Actions: Corrective actions are taken for water chemistry parameters outside the General Electric guidelines in order to restore chemistry within acceptable parameters. Evaluations are performed for test results that do not satisfy established criteria, and a condition report is initiated to document the concern in accordance with plant administrative procedures. The corrective action program ensures that conditions adverse to quality are promptly corrected. If the condition is significantly adverse to quality, the cause of the condition is determined and an action plan is developed to preclude recurrence.

(8) Confirmation Process: Stator cooling water is continuously analyzed. Additional sampling is therefore not required following corrective actions. Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B.

(9) Administrative Controls: See Item 8, above.

(10) Operating Experience: No age-related degradation of stator cooling water system components, within the scope of license renewal, has been observed. The current water chemistry activities have proven effective in managing aging of the stator cooling water system components. The current chemistry activities therefore require no enhancements.

Conclusion

The Quad Cities generator stator water chemistry program provides reasonable assurance that intended functions of the Quad Cities stator cooling water system components within the scope of license renewal are maintained consistent with the current licensing basis during the period of extended operation.

APPENDIX C

COMMODITY GROUPS (Not Used)

APPENDIX D

TECHNICAL SPECIFICATION CHANGES

(Not Used)

APPENDIX E

ENVIRONMENTAL INFORMATION FOR DRESDEN NUCLEAR POWER STATION

(Provided under separate cover)

APPENDIX F

ENVIRONMENTAL INFORMATION FOR QUAD CITIES NUCLEAR POWER STATION

(Provided under separate cover)