From: <john.hufnagel@exeloncorp.com>
To: <dja1@nrc.gov>
Date: 11/17/2005 2:35:48 PM
Subject: FAC and RWCU aging management program basis documents

Donnie/Greg,

Attached please find the two subject Aging Management Program (AMP) basis documents. They are provided to you in both PDF and Word format, to facilitate review of approval signatures (PDF version) and Audit activities (Word version). This is the first set of program basis documents (PBDs) that you will receive between now and the scheduled 1/23/06 AMP Audit. We plan on transmitting to you, in the same manner as this transmittal, several additional sets of PBDs over the next several weeks.

Specifically, we plan to transmit to you four additional sets of PBDs before Christmas. The targeted delivery dates are 11/28, 12/5, 12/12 and 12/19. After these transmittals, you will have received all of the PBDs associated with your AMP audit, per your Audit plan. For planning purposes, we are projecting which PBDs you will receive in each batch, so that you can advise your reviewers as to when they can expect to receive the PBDs of interest to them. We are reviewing that schedular information and plan to e-mail you the specifics tomorrow.

It is important to note that our team is also preparing substantial backup documentation for each program, in the form of program notebooks, that will be immediately available to the reviewers when they arrive at site on January 23, 2006.

Also attached below is a PDF file containing a two-page report from the AMP/AMR Audit Access database. This report contains the partial answer to question, # 147. This question asked us to provide program basis documents. The answer to this question will be updated to reflect which documents have been provided to you, each time we provide an additional batch of PBDs.

Let us know if you have any questions or concerns on this. Thanks.

- John.

<<11-17-05 Update to AMP-147 Question.PDF>> <<RWCU - AMP-B.1.18.PDF>> <<FAC-AMP-B.1.11.PDF>> <<FINAL PBD B.1.11 Flow Accelerated Corrosion jmr.doc>> <<FINAL PBD B.1.18 Rev 0 BWR RWCU jmr.doc>>

CC: <gvc@nrc.gov>, <fred.polaski@exeloncorp.com>

Mail Envelope Properties (437CDB7B.D24 : 20 : 36132)

Subject:	FAC and RWCU aging management program basis documents
Creation Date:	11/17/2005 2:33:03 PM
From:	<john.hufnagel@exeloncorp.com>
Created By:	john.hufnagel@exeloncorp.com

Recipients
nrc.gov
OWGWPO01.HQGWDO01
DJA1 (D. Ashley)

nrc.gov
owf4_po.OWFN_DO
GVC CC (Gregory Cranston)

exeloncorp.com
fred.polaski CC

Post Office	Route
OWGWPO01.HQGWDO01	nrc.gov
owf4_po.OWFN_DO	nrc.gov
	exeloncorp.com

Files	Size	Date & Time
MESSAGE	2849	17 November, 2005 2:33:03 PM
TEXT.htm	3978	
11-17-05 Update to AMP-147 Question.PDF	652863	
RWCU - AMP-B.1.18.PDF	1147425	
FAC-AMP-B.1.11.PDF	1443115	
FINAL PBD B.1.11 Flow Accelerated Corrosion jmr.doc	161792	
FINAL PBD B.1.18 Rev 0 BWR RWCU jmr.doc	135680	
Mime.822	4856332	

Options

Expiration Date:	None
Priority:	Standard
Reply Requested:	No
Return Notification:	None
Concealed Subject:	No
Security:	Standard

ML053410352

Audit Question No.	Source:	Info Only	Status	Discipline	Date Received
AMP-147	AMP Audit	☐	Open		10/7/2005

NRC Representative:	Took Issue:	Assigned to:	Potential Submittal on Docket	Document Reference
Tran, Linh		Corsi, Lou	☐	All

Topic:

AMP Generic Question

Question

Please provide the basis document of the ten program elements review for the following aging management programs:

B.1.1 ASME Section XI Inservice Inspection, Subsections IWB, IWC and IWD
B.1.2 Water Chemistry
B.1.3 Reactor Head Closure Studs
B.1.4 BWR Vessel ID Attachment Welds
B.1.5 BWR Feedwater Nozzle
B.1.6 BWR Control Rod Drive Return Line Nozzle
B.1.7 BWR Stress Corrosion Cracking
B.1.8 BWR Penetrations
B.1.10 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS)
B.1.11 Flow-Accelerated Corrosion
B.1.13 Open-Cycle Cooling Water System
B.1.14 Closed-Cycle Cooling Water System
B.1.15 Boraflex Rack Management Program
B.1.16 Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems
B.1.17 Compressed Air Monitoring
B.1.18 BWR Reactor Water Cleanup System
B.1.19 Fire Protection
B.1.20 Fire Water System
B.1.21 Aboveground Outdoor Tanks
B.1.22Fuel Oil Chemistry
B.1.24 One-Time Inspection
B.1.25Selective Leaching of Materials
B.1.26 Buried Piping Inspection
B.1.27 ASME Section XI, Subsection IWE
B.1.28 ASME Section XI, Subsection IWF
B.1.29 10 CFR Part 50, Appendix J
B.1.30 Masonry Wall Program
B.1.31 Structures Monitoring Program
B.1.32 RG 1.27, Inspection of Water-Control Structures Associated with Nuclear Power Plants
B.1.33 Protective Coating Monitoring and Maintenance Program
B.1.34 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements
B.1.35 Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits
B.1.36 Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements
B.3.1 Metal Fatigue of Reactor Coolant Pressure Boundary
B.3.2 Environmental Qualification (EQ) Progr

Final Response

This request will be responded to in several steps. That is, batches of aging management program basis documents (PBDs) will be provided over a period of time such that when a set of these PBDs is ready for NRC review, that set will be transmitted. This way, the NRC Audit team can continue their reviews while the Oyster Creek team continues to generate the upgraded PBDs. These transmittals are being made an an ongoing activity as part of the AMP Audit.

11/17/05 Update

The initial (two) PBDs provided to the NRC were the Flow Accelerated Corrosion (FAC) and the Reactor Water Cleanup (RWCU) PBDs. These were e-mailed to NRC Project Manager Donnie Ashley, with a copy to Audit Team lead Greg Cranston, on 11/17/05. They were provided in two formats. PDF versions of the documents were provided, which included copies of the signatures of the preparer, reviewer, program owner (site) and Approver (Project Technical Lead). In addition, as requested by the NRC, Word versions were provided to facilitate the Audit review and report writing process.
- J.G. Hufnagel

Followup Actions Required

Scope/Screen Change ☐ LR Drawing Change ☐ Commitment Chang ☐ None ☐

AMR Change ☐ Program Basis Document Chang ☐ Docketed Response ☐

IR#: LRCR #:

Prepared By: Hufnagel, John

Reviewed By:

Approved By:

NRC Acceptance Date:

Prepared Date:

Reviewed Date:

Approved Date:

NRC Response Report

PROGRAM BASIS DOCUMENT

PBD-AMP-B.1.11

Revision 0

FLOW-ACCELERATED CORROSION

GALL PROGRAM XI.M17 - FLOW-ACCELERATED CORROSION

Prepared By: ______________________

Reviewed By: ______________________

Program Owner Review: ______________________

Technical Lead Approval: ______________________

Revision History:

Revision	*Prepared by:*	*Reviewed by:*	*Program Owner:*	*Approved by:*
0	*Mark A. Miller*	*K. Muggleston*	*Roger Gayley*	*Don Warfel*
Date				

Summary of Revisions:

Rev. Number	Reason for the Revision(s)
0	Initial Issue

TABLE OF CONTENTS

1.0 PURPOSE AND METHODOLOGY

1.1 Purpose

The purpose of this Program Basis Document is to document and evaluate those activities of the Oyster Creek Flow-Accelerated Corrosion aging management program that are credited for managing the loss of material due to flow-accelerated corrosion as part of Oyster Creek License Renewal to meet the requirements of the License Renewal Rule 10 CFR 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

This includes the following:

- The identification of the scope of the program;
- The evaluation of program elements against NUREG-1801;
- The review of Operating Experience to demonstrate program effectiveness;
- The identification of required program enhancements;
- The identification of Oyster Creek documents required to implement the program.

1.2 Methodology

The nuclear power plant License Renewal Rule 10 CFR 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," describes the License Renewal process and provides requirements for the contents of License Renewal Applications. 10 CFR 54.21(a)(3) states:

"For each structure and component identified in paragraph (a)(1) of this section, demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the Current Licensing Basis (CLB) for the period of extended operation."

The NRC and the industry identified 10 program elements that are useful in describing an aging management program and then demonstrating its effectiveness. These program elements are described in Appendix A.1, Section A.1.2.3 of the Standard Review Plan. NUREG-1801 uses these program elements in Section XI to describe acceptable aging management programs.

This Program Basis Document provides a comparison of the credited Oyster Creek program with the elements of the corresponding NUREG-1801 Chapter XI program XI.M17, Flow-Accelerated Corrosion. Project Level Instruction PLI-8 "Program Basis Documents" prescribes the methodology for evaluating Aging Management Programs. An evaluation of Oyster Creeks aging management program criteria or activities to those of the NUREG-1801 program elements is performed and a conclusion is reached concerning consistency for each individual program element. A demonstration of overall program effectiveness is made after all program elements are evaluated. Required program enhancements are documented. An overall determination is made as to consistency with the program description in NUREG-1801.

2.0 PROGRAM DESCRIPTION

2.1 Program Description

NUREG-1801:

The program relies on implementation of the Electric Power Research Institute (EPRI) guidelines in the Nuclear Safety Analysis Center (NSAC)-202L-R2 for an effective flow-accelerated corrosion (FAC) program. The program includes performing (a) an analysis to determine critical locations, (b) limited baseline inspections to determine the extent of thinning at these locations, and (c) follow-up inspections to confirm the predictions, or repairing or replacing components as necessary.

Oyster Creek:

The Flow-Accelerated Corrosion (FAC) program is an existing program based on the Electric Power Research Institute (EPRI) guidelines in the Nuclear Safety Analysis Center (NSAC)-202L-R2 that predicts, detects, and monitors wall thinning in piping and components due to flow-accelerated corrosion. The FAC program provides for prediction of the amount of wall thinning through analytical evaluations and periodic examinations of locations most susceptible to FAC induced loss of material. Specifically, the program includes analyses to determine critical locations, baseline inspections to determine the extent of thinning at these locations, and follow-up inspections to confirm the predictions. Repairs and replacements are performed as necessary.

2.2 Overall NUREG-1801 Consistency

The Oyster Creek Flow-Accelerated Corrosion program is an existing program that is consistent with NUREG-1801 aging management program XI.M17, Flow-Accelerated Corrosion.

2.3 Summary of Exceptions to NUREG-1801

None. The existing Oyster Creek Flow-Accelerated Corrosion aging management program is found to be adequate to support the extended period of operation with no exceptions.

2.4 Summary of Enhancements to NUREG-1801

None. The existing Oyster Creek Flow-Accelerated Corrosion aging management program is found to be adequate to support the extended period of operation with no enhancements.

3.0 EVALUATIONS AND TECHNICAL BASIS

Note

This section is organized by quoting the relevant NUREG-1801 Chapter XI program element (September 2005 version) followed by the related Oyster Creek program attributes and a conclusion of the comparison. Where applicable, the NUREG-1801 program element was separated into logical sub-elements and addressed accordingly.

Implementing procedure references are included in **( )** for information purposes. This information from the source procedure has been either directly extracted from the procedure or summarized for inclusion into this PBD.

3.1 Scope of Program

NUREG-1801:

a) *The FAC program, described by the EPRI guidelines in NSAC-202L-R2, includes procedures or administrative controls to assure that the structural integrity of all carbon steel lines containing high-energy fluids (two phase as well as single phase) is maintained.*

b) *Valve bodies retaining pressure in these high-energy systems are also covered by the program.*

c) *The FAC program was originally outlined in NUREG-1344 and was further described through the Nuclear Regulatory Commission (NRC) Generic Letter (GL) 89-08.*

d) *A program implemented in accordance with the EPRI guidelines predicts, detects, and monitors FAC in plant piping and other components, such as valve bodies, elbows and expanders. Such a program includes the following recommendations: (1) conducting an analysis to determine critical locations; (2) performing limited baseline inspections to determine the extent of thinning at these locations; and (3) performing follow-up inspections to confirm the predictions, or repairing or replacing components as necessary. NSAC-202L-R2 (April 1999) provides general guidelines for the FAC program.*

e) *To ensure that all the aging effects caused by FAC are properly managed, the program includes the use of a predictive code, such as CHECWORKS, that uses the implementation guidance of NSAC-202L-R2 to satisfy the criteria specified in 10 CFR Part 50, Appendix B, criteria for development of procedures and control of special processes.*

Oyster Creek:

a) The FAC program is based on the EPRI guidelines in NSAC-202L, R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program" and includes procedures and administrative controls to assure that the structural integrity of all carbon steel lines containing high-energy fluids (two phase as well as single phase) is maintained **(Reference: ER-AA-430 paragraph 4.1.1, 4.3.1, 4.6.1, 4.8, and 6.4; ER-AA-430-1001 paragraph 4.2.1, 4.4, 4.5, 4.9, and 6.9; ER-AA-430-1002 paragraph 4.4.4, 4.6, and 6.10).**

b) The FAC program predicts, detects, and monitors FAC in plant piping. FAC inspections of other components (e.g., valve bodies, pump casings, orifices, nozzles, etc.) are also performed. When a component cannot be inspected completely with UT due to its shape and thickness, the inspection grid for the component should be placed on the downstream piping for a minimum distance of two (2) pipe diameters from the connecting weld **(Reference: ER-AA-430-1001 paragraph 4.3.1.3.C)**. If significant thinning is found downstream of a component the component should be inspected also **(ER-AA-430-1001 Attachment 2 for valve or orifice inspection grid location).**

Work requests are reviewed to identify areas that may be accessible for the visual inspection of components **(Reference: ER-AA-430-1001 paragraph 4.2.2.2.B)**. When a component is disassembled or removed from a FAC susceptible line, and wear patterns are detected, the FAC Program Engineer or designee should visually inspect the component interior and the area adjacent to the opening to determine if FAC damage is present **(Reference: ER-AA-430-1001 paragraph 4.3.3.1.B)**. When a feedwater heater or nozzle piping is disassembled or removed for maintenance and the shell interior is accessible, the FAC Program Engineer or designee will visually inspect the shell interior to determine if thinning is present **(Reference: ER-AA-430-1002 paragraph 4.4.5)**.

c) The FAC program, which was originally outlined in NUREG-1344, is implemented at Oyster Creek as required by NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning" **(Reference: ER-AA-430 paragraph 1.1.1 and 4.10.1)**.

d) The FAC program is based on the EPRI guidelines in NSAC-202L, R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program" and includes procedures and administrative controls to predict, detect, and monitor FAC in plant piping and other components **(Reference: ER-AA-430 paragraph 4.1.1, 4.3.1, 4.6.1, 4.8, and 6.4; ER-AA-430-1001 paragraph 4.2.1, 4.4, 4.5, 4.9, and 6.9; ER-AA-430-1002 paragraph 4.4.4, 4.6, and 6.10)**.

1) Analyses to determine critical locations in piping and other components susceptible to FAC is performed utilizing CHECKWORKS runs and should consider industry experience (e.g., CHECWORKS Users Group (CHUG) Notices, Plant Event databases, Significant Operating Event Reports, Nuclear Operation Notices, Information Notices, etc.), Exelon experience/station experience, or special considerations **(Reference: ER-AA-430-1001 paragraph 4.2.1; ER-AA-430-1002 paragraph 4.2)**.

2) The Oyster Creek FAC aging management program includes inspections to determine thinning at critical locations. BASELINE INSPECTIONS are performed on new or replacement components that have not previously been involved in plant operations. INITIAL OPERATIONAL INSPECTIONS are those inspections that involve the first inspection of a component that has been in service and has not been subjected to a baseline inspection. SUBSEQUENT REINSPECTIONS are the inspection of components that have had a baseline inspection and/or an initial operational inspection **(Reference: ER-AA-430 Attachment 2; ER-AA-430-1001 Attachment 1).**

3) Follow-up, or subsequent reinspections, are performed to confirm predictions and to determine the need for repairs or replacements as necessary. Inspection results are reviewed to ensure that the examined component has adequate wall thickness remaining to be returned to service. Factors that are considered when reviewing the inspection data include initial wall thickness, counterbore obstructions and manufactured wall thickness variations. For each examined component, a verified and validated PC-based computer program, called FAC Manager, is utilized in conjunction with CHECWORKS to calculate component wear, wear rate, projected thickness, remaining life (defined as the time for the wall thickness to reach code minimum allowable based on calculated wear rate) and Next Scheduled Inspection (NSI) (defined as the future outage number immediately preceding the cycle during which the component is projected to fall below code minimum allowable wall thickness) **(Reference: ER-AA-430 paragraph 4.6; ER-AA-430-1001 paragraph 4.4; ER-AA-430-1002 paragraph 4.6).** If a component's remaining life cannot be demonstrated to be more than 1 operating cycle, then corrective action is required such as repair, replacement, or reevaluation **(Reference: ER-AA-430 paragraph 4.8; ER-AA-430-1001 paragraph 4.7).** Repairs, replacements, and reevaluations are performed in accordance with the applicable codes and station procedures. The guidance provided in EPRI Report, "Recommendations for an Effective Flow Accelerated Corrosion Program," NSAC-202L, Rev. 2 is considered in evaluating repair/replacement options **(Reference: ER-AA-430 paragraph 4.8; ER-AA-430-1001 paragraph 4.9).**

e) The FAC program includes the use of CHECWORKS that uses the implementation guidance of NSAC-202L-R2 to satisfy the criteria specified in 10 CFR Part 50, Appendix B, criteria for development of procedures and control of special processes **(Reference: ER-AA-430 paragraph 4.2)**. The original FAC susceptibility analysis is contained in Technical Data Report TDR No. 861 "Erosion/Corrosion Inspection Program for Steam, Two Phase, and Liquid Systems." The susceptible piping systems are divided into two categories: The "Modeled" category consists of piping systems, or portions of systems, that are susceptible to FAC and have a completed FAC wear rate analysis in EPRI's CHECWORKS computer code **(Reference: ER-AA-430 paragraph 4.2)**. The "Non-Modeled" category consists of piping systems, or portions of systems, that are susceptible to FAC but do not have a completed FAC wear rate analysis in the CHECWORKS computer code **(Reference: ER-AA-430 paragraph 4.3)**.

The Oyster Creek Flow-Accelerated Corrosion aging management program manages the aging effect of loss of material due to flow-accelerated corrosion for the systems, components, and environments listed in Table 5.2. The implementing documents for this aging management program are listed in Table 5.1 and are described throughout the individual program element discussions. The commitment numbers under which these implementing documents are being revised are contained within the listings in Table 5.1.

Exceptions to NUREG-1801, Element 1:

None.

Enhancements to NUREG-1801, Element 1:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 1, Scope of Program.

3.2 Preventive Actions

NUREG-1801:

The FAC program is an analysis, inspection, and verification program; thus, there is no preventive action. However, it is noted that monitoring of water chemistry to control pH and dissolved oxygen content, and selection of appropriate piping material, geometry, and hydrodynamic conditions, are effective in reducing FAC.

Oyster Creek:

No preventive or mitigative attributes are associated with the FAC program. It is noted that monitoring of water chemistry to control pH and dissolved oxygen content is effective in reducing FAC. The FAC program requires that when requested, the Unit Chemist update the FAC Program Engineer of any water treatment changes that may affect the FAC rates (e.g., water treatment amines, hydrogen water chemistry, hydrazine addition, or any other change that affects the pH or dissolved oxygen concentration) **(Reference: ER-AA-430 Attachment 1 paragraph 8.0).**

Analytical models are developed for susceptible systems suitable for modeling to quantify the potential for FAC damage. The models are developed using CHECWORKS software. The model inputs include component material, geometry effects, and subsystem (line) operating conditions (including hours of operation, flowrates, fluid temperatures, system chemistry and flow thermodynamic conditions) **(Reference: ER-AA-430 paragraph 4.2.1).** If a component's remaining life cannot be demonstrated to be more than 1 operating cycle, then repair, replacement, or reevaluation is required **(Reference: ER-AA-430 paragraph 4.8; ER-AA-430-1001 paragraph 4.7).** Repairs, replacements, and reevaluations are performed in accordance with the applicable codes and station procedures. The guidance provided in EPRI Report, "Recommendations for an Effective Flow Accelerated Corrosion Program," NSAC-202L, Rev. 2 is considered in evaluating repair/replacement options. The selection of appropriate piping material, geometry, and hydrodynamic conditions or operating parameters may be considered in evaluating repair/replacement options **(Reference: ER-AA-430 paragraph 4.8; ER-AA-430-1001 paragraph 4.9).**

Exceptions to NUREG-1801, Element 2:

None.

Enhancements to NUREG-1801, Element 2:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 2, Preventive Actions.

3.3 Parameters Monitored or Inspected

NUREG-1801:

The aging management program (AMP) monitors the effects of FAC on the intended function of piping and components by measuring wall thickness.

Oyster Creek:

The FAC program provides for inspection and monitoring of the wall thickness of piping and components to determine the effects of FAC on their intended function. BASELINE INSPECTIONS are performed on new or replacement components that have not previously been involved in plant operations. INITIAL OPERATIONAL INSPECTIONS are those inspections that involve the first inspection of a component that has been in service and has not been subjected to a baseline inspection. SUBSEQUENT REINSPECTIONS are the inspection of components that have had a baseline inspection and/or an initial operational inspection **(Reference: ER-AA-430 Attachment 2; ER-AA-430-1001 Attachment 1)**. Inspection results are reviewed to ensure that the examined component has adequate wall thickness remaining to be returned to service. Factors that are considered when reviewing the inspection data include initial wall thickness, counterbore obstructions and manufactured wall thickness variations. For each examined component, a verified and validated PC-based computer program, called FAC Manager, is utilized in conjunction with CHECWORKS to calculate component wear, wear rate, projected thickness, remaining life and Next Scheduled Inspection (NSI) **(Reference: ER-AA-430 paragraph 4.6; ER-AA-430-1001 paragraph 4.4; ER-AA-430-1002 paragraph 4.6).**

Exceptions to NUREG-1801, Element 3:

None.

Enhancements to NUREG-1801, Element 3:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 3, Parameters Monitored or Inspected.

3.4 Detection of Aging Effects

NUREG-1801:

a) *Degradation of piping and components occurs by wall thinning. The inspection program delineated in NSAC-202L-R2 consists of identification of susceptible locations as indicated by operating conditions or special considerations.*

b) *Ultrasonic and radiographic testing is used to detect wall thinning. The extent and schedule of the inspections assure detection of wall thinning before the loss of intended function.*

Oyster Creek:

a) The FAC program is based on the EPRI guidelines in NSAC-202L, R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program" and includes procedures and administrative controls to assure that the structural integrity of all carbon steel piping and components containing high-energy fluids (two phase as well as single phase) is maintained **(Reference: ER-AA-430 paragraph 4.1.1, 4.3.1, 4.6.1, 4.8, and 6.4; ER-AA-430-1001 paragraph 4.2.1, 4.4, 4.5, 4.9, and 6.9; ER-AA-430-1002 paragraph 4.4.4, 4.6, and 6.10)**. Analyses to determine critical locations in piping and components where degradation due to wall thinning can occur is performed utilizing CHECWORKS runs and should consider industry experience (e.g., CHECWORKS Users Group (CHUG) Notices, Plant Event databases, Significant Operating Event Reports, Nuclear Operation Notices, Information Notices, etc.), Exelon experience/station experience, or special considerations **(Reference: ER-AA-430-1001 paragraph 4.2.1; ER-AA-430-1002 paragraph 4.2)**.

b) BASELINE INSPECTIONS are performed on new or replacement components that have not previously been involved in plant operations. INITIAL OPERATIONAL INSPECTIONS are those inspections that involve the first inspection of a component that has been in service and has not been subjected to a baseline inspection. SUBSEQUENT REINSPECTIONS are the inspection of components that have had a baseline inspection and/or an initial operational inspection **(Reference: ER-AA-430 Attachment 2; ER-AA-430-1001 Attachment 1)**. Ultrasonic (UT) and Radiographic (RT) inspection results are reviewed to ensure that the examined component has adequate wall thickness remaining to be returned to service. Factors that are considered when reviewing the inspection data include initial wall thickness, counterbore obstructions and manufactured wall thickness variations. For each examined component, a verified and validated PC-based computer program, called FAC Manager, is utilized in conjunction with CHECWORKS to calculate component wear, wear rate, projected thickness, remaining life and Next Scheduled Inspection (NSI) to assure the detection of wall thinning before the loss of intended function **(Reference: ER-AA-430 paragraph 4.6; ER-AA-430-1001 paragraph 4.4; ER-AA-430-1002 paragraph 4.6)**.

Exceptions to NUREG-1801, Element 4:

None.

Enhancements to NUREG-1801, Element 4:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 4, Detection of Aging Effects.

3.5 Monitoring and Trending

NUREG-1801:

a) *CHECWORKS or a similar predictive code is used to predict component degradation in the systems conducive to FAC, as indicated by specific plant data, including material, hydrodynamic, and operating conditions. CHECWORKS is acceptable because it provides a bounding analysis for FAC. CHECWORKS was developed and benchmarked by using data obtained from many plants.*

b) *The inspection schedule developed by the licensee on the basis of the results of such a predictive code provides reasonable assurance that structural integrity will be maintained between inspections.*

c) *Inspection results are evaluated to determine if additional inspections are needed to assure that the extent of wall thinning is adequately determined, assure that intended function will not be lost, and identify corrective actions.*

Oyster Creek:

a) CHECWORKS is used to predict component degradation in the systems conducive to FAC and provides reasonable assurance that structural integrity will be maintained between inspections. CHECWORKS provides a bounding analysis for FAC and was developed and benchmarked by using data obtained from many plants. The model inputs include component material, geometry effects, and subsystem (line) operating conditions (including hours of operation, flowrates, fluid temperatures, system chemistry and flow thermodynamic conditions) **(Reference: ER-AA-430 paragraph 4.2.1)**. FAC susceptible piping and components not suitable for CHECWORKS modeling are qualitatively evaluated based on NSAC-202L, Rev. 2 guidance **(Reference: ER-AA-430 paragraph 4.3)**.

b) Determination of which components require inspection is accomplished by querying the approved program database. For each examined component, a verified and validated PC-based computer program, called FAC Manager, is utilized in conjunction with CHECWORKS to calculate component wear, wear rate, projected thickness, remaining life and Next Scheduled Inspection (NSI) **(Reference: ER-AA-430 paragraph 4.6; ER-AA-430-1001 paragraph 4.4; ER-AA-430-1002 paragraph 4.6)**. At a minimum, inspections are performed on all components with a next scheduled inspection (NSI) that is less than or equal to the upcoming outage. In addition, re-inspections of difficult to replace components (e.g., large bore tees or expanders) are considered one outage earlier than the calculated NSI. This reduces the risk that an unexpected inspection failure could result in outage delays **(Reference: ER-AA-430-1001 paragraph 4.2.1.4.A).**

c) Inspection results are reviewed to ensure that the examined component has adequate wall thickness remaining to be returned to service, thus assuring that intended functions are not lost. Factors that are considered when reviewing the inspection data include initial wall thickness, counterbore obstructions and manufactured wall thickness variations. For each examined component, a verified and validated PC-based computer program, called FAC Manager, is utilized in conjunction with CHECWORKS to calculate component wear, wear rate, projected thickness, remaining life and Next Scheduled Inspection (NSI) **(Reference: ER-AA-430 paragraph 4.6; ER-AA-430-1001 paragraph 4.4; ER-AA-430-1002 paragraph 4.6).** If any component has a current or projected wall thickness less than the minimum acceptable wall thickness, or if any component exhibits significant unexpected wall thinning, then additional inspections are performed to bound the area of thinning. However, such additional inspections are not required if the thinning was expected or if the thinning is unique to that component (e.g., degradation downstream of a leaking valve). Sample expansion requirements apply to all components, including Susceptible Non-Modeled Program (SNM) components. For SNM components, sample expansion criteria relating to rankings may be replaced by components selected using engineering judgment **(Reference: ER-AA-430 paragraph 4.7; ER-AA-430-1001 paragraph 4.8; ER-AA-430-1002 paragraph 4.8).** If a component's remaining life cannot be demonstrated to be more than 1 operating cycle, then corrective action is required such as repair, replacement, or reevaluation **(Reference: ER-AA-430 paragraph 4.8; ER-AA-430-1001 paragraph 4.7).** Repairs, replacements, and reevaluations are performed in accordance with the applicable codes and station procedures. The guidance provided in EPRI Report, "Recommendations for an Effective Flow Accelerated Corrosion Program," NSAC-202L, Rev. 2 is considered in evaluating repair/replacement options **(Reference: ER-AA-430 paragraph 4.8; ER-AA-430-1001 paragraph 4.9).**

Exceptions to NUREG-1801, Element 5:

None.

Enhancements to NUREG-1801, Element 5:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 5, Monitoring and Trending.

3.6 Acceptance Criteria

NUREG-1801:

a) *Inspection results are input for a predictive computer code, such as CHECWORKS, to calculate the number of refueling or operating cycles remaining before the component reaches the minimum allowable wall thickness.*

b) *If calculations indicate that an area will reach the minimum allowed thickness before the next scheduled outage, the component is to be repaired, replaced, or reevaluated.*

Oyster Creek:

a) Analytical models are developed for susceptible systems suitable for modeling to quantify the potential for FAC damage. The models are developed using CHECWORKS software. The model inputs include component material, geometry effects, and subsystem (line) operating conditions (including hours of operation, flowrates, fluid temperatures, system chemistry and flow thermodynamic conditions) **(Reference: ER-AA-430 paragraph 4.2.1)**. Inspection results are reviewed to ensure that the examined component has adequate wall thickness remaining to be returned to service. Factors that are considered when reviewing the inspection data include initial wall thickness, counterbore obstructions and manufactured wall thickness variations. For each examined component, a verified and validated PC-based computer program, called FAC Manager, is utilized in conjunction with CHECWORKS to calculate component wear, wear rate, projected thickness, remaining life and Next Scheduled Inspection (NSI) **(Reference: ER-AA-430 paragraph 4.6; ER-AA-430-1001 paragraph 4.4; ER-AA-430-1002 paragraph 4.6)**.

b) If a component's remaining life cannot be demonstrated to be more than 1 operating cycle, then the component is repaired, replaced, or reevaluated **(Reference: ER-AA-430 paragraph 4.8; ER-AA-430-1001 paragraph 4.7)**. Repairs, replacements, and reevaluations are performed in accordance with the applicable codes and station procedures. The guidance provided in EPRI Report, "Recommendations for an Effective Flow Accelerated Corrosion Program," NSAC-202L, Rev. 2 is considered in evaluating repair/replacement options **(Reference: ER-AA-430 paragraph 4.8; ER-AA-430-1001 paragraph 4.9)**.

Exceptions to NUREG-1801, Element 6:

None.

Enhancements to NUREG-1801, Element 6:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 6, Acceptance Criteria.

3.7 Corrective Actions

NUREG-1801:

a) Prior to service, components for which the acceptance criteria are not satisfied are reevaluated, repaired, or replaced. Long-term corrective actions could include adjusting operating parameters or selecting materials resistant to FAC.

b) As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing corrective actions.

Oyster Creek:

a) Inspection results are reviewed to ensure that the examined component has adequate wall thickness remaining to be returned to service. Factors that are considered when reviewing the inspection data include initial wall thickness, counterbore obstructions and manufactured wall thickness variations. For each examined component, a verified and validated PC-based computer program, called FAC Manager, is utilized in conjunction with CHECWORKS to calculate component wear,

wear rate, projected thickness, remaining life and Next Scheduled Inspection (NSI) **(Reference: ER-AA-430 paragraph 4.6; ER-AA-430-1001 paragraph 4.4; ER-AA-430-1002 paragraph 4.6).** If a component's remaining life cannot be demonstrated to be more than 1 operating cycle, then the component is repaired, replaced, or reevaluated **(Reference: ER-AA-430 paragraph 4.8; ER-AA-430-1001 paragraph 4.7).** Repairs, replacements, and reevaluations are performed in accordance with the applicable codes and station procedures. The guidance provided in EPRI Report, "Recommendations for an Effective Flow Accelerated Corrosion Program," NSAC-202L, Rev. 2 is considered in evaluating repair/replacement options. The selection of appropriate piping material, geometry, and hydrodynamic conditions or operating parameters may be considered in evaluating repair/replacement options **(Reference: ER-AA-430 paragraph 4.8; ER-AA-430-1001 paragraph 4.9).**

b) Evaluations are performed for test or inspection results that do not satisfy established criteria and an Issue Report (IR) is initiated to document the concern in accordance with the 10 CFR Part 50, Appendix B Corrective Action Program **(Reference: ER-AA-430-1001 paragraph 4.7.2; ER-AA-430-1002 paragraph 4.7).** The corrective action process ensures that the conditions adverse to quality are promptly corrected. If the deficiency is assessed to be significantly adverse to quality, the cause of the condition is determined and an action plan is developed to preclude recurrence.

<u>Exceptions to NUREG-1801, Element 7:</u>

None.

<u>Enhancements to NUREG-1801, Element 7:</u>

None.

<u>Comparison and Evaluation Conclusion:</u>

This element is consistent with NUREG-1801, Element 7, Corrective Actions.

3.8 Confirmation Process

NUREG-1801:

Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable in addressing the confirmation process and administrative controls.

Oyster Creek:

Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B.

Exceptions to NUREG-1801, Element 8:

None.

Enhancements to NUREG-1801, Element 8:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 8, Confirmation Process.

3.9 Administrative Controls

NUREG-1801:

See Item 8, above.

Oyster Creek:

See Item 8, above.

Exceptions to NUREG-1801, Element 9:

None.

Enhancements to NUREG-1801, Element 9

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 9, Administrative Controls.

3.10 Operating Experience

NUREG-1801:

Wall-thinning problems in single-phase systems have occurred in feedwater and condensate systems (NRC IE Bulletin No. 87-01; NRC Information Notices [INs] 81-28, 92-35, 95-11) and in two-phase piping in extraction steam lines (NRC INs 89-53, 97-84) and moisture separation reheater and feedwater heater drains (NRC INs 89-53, 91-18, 93-21, 97-84). Operating experience shows that the present program, when properly implemented, is effective in managing FAC in high-energy carbon steel piping and components.

Oyster Creek:

Review of industry operating experience has confirmed that wall-thinning problems in single-phase systems have occurred in feedwater and condensate systems (NRC IE Bulletin No. 87-01; NRC Information Notices [INs] 81-28, 92-35, 95-11) and in two-phase piping in extraction steam lines (NRC INs 89-53, 97-84) and moisture separation reheater and feedwater heater drains (NRC INs 89-53, 91-18, 93-21, 97-84). A review of plant operating experience at Oyster Creek shows that wall thinning has occurred in several different systems. In most cases, the existing Flow-Accelerated Corrosion aging management program has identified the wall thinning prior to leaks occurring. However, there have been some cases where leaks did occur that were not predicted. As a result of these pipe failures, the FAC program was modified to more accurately predict that wall thinning was occurring. The experience at Oyster Creek with the FAC program shows that the FAC program is effective in managing FAC in high-energy carbon steel piping and components.

Operating experience, both internal and external, is used in two ways at Oyster Creek to enhance plant programs, prevent repeat events, and prevent events that have occurred at other plants from occurring at Oyster Creek. The first way in which operating experience is used is through the Oyster Creek Operating Experience process. The Operating Experience process screens, evaluates, and acts on operating experience documents and information to prevent or mitigate the consequences of similar events. The second way is through the process for managing programs. This process requires the review of program related operating experience by the program owner.

Both of these processes review operating experience from both external and internal (also referred to as in-house) sources. External operating experience may include such things as INPO documents (e.g., SOERs, SERs, SENs, etc.), NRC documents (e.g., GLs, LERs, INs, etc.), Westinghouse documents (e.g., TBs, etc.), General Electric documents (e.g., RCSILs, SILs, TILs, etc.), and other documents (e.g., 10CFR Part 21 Reports, NERs, etc.). Internal operating experience may include such things as event investigations, trending reports, and lessons learned from in-house events as captured in program notebooks, self-assessments, and in the 10 CFR Part 50, Appendix B corrective action process.

Demonstration that the effects of aging are effectively managed is achieved through objective evidence that shows that wall thinning due to FAC is being adequately managed in piping and other components. The following examples of operating experience provide objective evidence that the FAC program is effective in assuring that intended function(s) will be maintained consistent with the CLB for the period of extended operation:

1) Inspections of the "C" Feedpump Min. Recirculation line showed that several 90-degree elbows experienced significant wear. Similar wear was found on several 45-degree elbows. As a result of these inspections, approximately 25 feet of 4" pipe, one 90-degree elbow, and three 45-degree elbows were replaced with chrome-moly material (reference CAPs O2000-1578 and O2000-1607). This example provides objective evidence that wall thinning will be detected prior to the loss of intended function, and, adequate corrective actions are taken to prevent recurrence (use of FAC resistant material).

2) During 18R inspections in 2000, two elbows on the 11th stage extraction line to the "B" IP Feedwater Heater showed severe localized degradation (reference TDR No. 943). Separate inspections of similar piping associated with the "A" IP Feedwater Heater also revealed that an active pipe degradation mechanism existed. Although engineering analysis determined that the identified conditions were acceptable for continued operation, it was concluded that the degradation in these lines was problematic and that extensive inspections were required to be performed during future outages. This example provides objective evidence that the implementation of the FAC program will effectively determine the susceptible locations of flow accelerated corrosion, predict the component degradation, and detect the wall thinning in piping and other components due to flow accelerated corrosion prior to the loss of intended function.

3) In January 2000, UT Inspections were performed on the High-Pressure (HP) Feedwater Heater (FWH) shells (reference TDR No. 943). These inspections were driven by the Point Beach FWH shell rupture event and other industry experience as described in SEN 199 and NRC Information Notice 99-19. Results of the inspections showed wall thinning on all three HP FWH shells. Two areas on the "A" HP FWH required immediate repair. Other identified degradation was evaluated and determine to be acceptable through the remainder of the operating cycle at which time further inspections and repairs were performed. This example provides objective evidence that industry operating experience is considered when establishing inspection criteria. It also demonstrates that the FAC program provides appropriate guidance for evaluation, repair or replacement for locations where the calculations indicate an area will reach minimum allowable thickness before the next scheduled inspection.

4) There have been a number of examples of steam leaks associated with flash tank and drain tank piping and attached piping. CAP O2003-1903-3 was initiated to determine why the FAC scope and inspection frequency did not prevent these failures from occurring. As documented in the CAP response, the Corporate FAC Program Manager performed an Oversight Self-Assessment of the Flow Accelerated Corrosion (FAC) Program at Oyster Creek during February 2003. Two deficiencies in the program were identified: 1) The System Susceptibility Evaluation did not meet EPRI or procedural requirements and 2) plant model input to the FAC Program

software tool, CHECWORKS, contained a number of errors and omissions. These deficiencies were identified as the primary reasons the FAC program has missed identifying components that developed leaks as a result of Flow Accelerated Corrosion. A FAC program improvement project was implemented to correct the deficiencies. The project was completed during August 2003. The program was reviewed by Exelon Nuclear Oversight (QA) during September 2003 and found to be acceptable. As a result of the improvement project, the risk of a FAC failure in an unidentified susceptible line is reduced, and FAC inspections and outage inspection costs and time will be optimized since the tools are now available to assist in selecting the right outage inspection scope. This example provides objective evidence that deficiencies in the FAC program are entered into the 10 CFR Part 50, Appendix B Corrective Action process. It also demonstrates that FAC model updates are performed as necessary to ensure that the FAC program accurately predicts wall thinning thereby maintaining intended functions.

The operating experience of the FAC program did not show any adverse trend in performance. Problems identified would not cause significant impact to the safe operation of the plant, and adequate corrective actions were taken to prevent recurrence. There is sufficient confidence that the implementation of the FAC program will effectively determine the susceptible locations of flow accelerated corrosion, predict the component degradation, and detect the wall thinning in piping and other components due to flow accelerated corrosion. Appropriate guidance for reevaluation, repair or replacement is provided for locations where the calculations indicate an area will reach minimum allowable thickness before the next inspection. Periodic self-assessments of the FAC program are performed to identify the areas that need improvement to maintain the quality performance of the program.

3.11 Conclusion

The Oyster Creek Flow-Accelerated Corrosion aging management program is credited for managing the aging effect of loss of material due to flow-accelerated corrosion for the systems, components, and environments listed in Table 5.2. The Oyster Creek FAC program's elements have been evaluated against NUREG-1801 in Section 3.0. Program exceptions have been identified in Section 2.3. Program enhancements have been identified in Section 2.4. The implementing documents and commitment numbers for this aging management program are listed in Table 5.1. The relevant operating experience has been reviewed and a demonstration of program effectiveness is provided in Section 3.10.

Based on the above, the continued implementation of the Oyster Creek Flow-Accelerated Corrosion aging management program provides reasonable assurance that the loss of material due to flow-accelerated corrosion will be adequately managed so that the intended functions of components within the scope of license renewal will be maintained during the period of extended operation.

4.0 REFERENCES

4.1 Generic to Aging Management Programs

4.1.1 10 CFR 50, Appendix B, *Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants*

4.1.2 10 CFR 54, *Requirements for Renewal of Operating Licenses for Nuclear Power Plants*

4.1.3 NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants*, Revision 1, dated September 2005

4.1.4 NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, Revision 1, dated September 2005

4.2 Industry Standards

4.2.1 EPRI NSAC-202L-R2, April, 1999, *Recommendations for an Effective Flow-Accelerated Corrosion Program*

4.2.2 EPRI TR-109623, November 21,2003, *Mechanical Series Module 15, Erosion, Corrosion and Flow-Accelerated Corrosion*

4.3 Oyster Creek Program References

4.3.1 Specification SP-1302-12-237, Revision 11, *Nuclear Safety Related Pipe Wall Thinning Inspections Specification For Oyster Creek Nuclear Generating Station Erosion/Corrosion*

4.3.2 Technical Data Report TDR No. 943, Revision 3, *OC Flow Accelerated Corrosion Inspection History*

5.0 TABLES

5.1 Aging Management Program Implementing Documents

Procedure Number	Procedure Title	Commitment No.	Status
ER-AA-430-1001	Guidelines for Flow Accelerated Corrosion Activities	00330592.11.02	ACC/ASG
ER-AA-430-1002	Feedwater Heater Shell Inspection for Detection of Flow Accelerated Corrosion	00330592.11.03	ACC/ASG
ER-AA-430	Conduct of Flow Accelerated Corrosion Activities	00330592.11.01	ACC/ASG

5.2 Aging Management Review Results

SSC Name	Structure and/or Component	Material	Environment	Aging Effect
Condensate System	Piping and fittings	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material
Condensate System	Valve Body	Cast Iron	Treated Water (Internal)	Loss of Material
Condensate System	Valve Body	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material
Feedwater System	Piping and fittings	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material
Feedwater System	Valve Body	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material
Feedwater System	Valve Body	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material
Feedwater System	Piping and fittings	Chrome Moly steels	Treated Water (Internal)	Loss of Material
Feedwater System	Piping and fittings	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material
Feedwater System	Valve Body	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material
Feedwater System	Piping and fittings	Chrome Moly steels	Treated Water (Internal)	Loss of Material
Feedwater System	Piping and fittings	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material
Feedwater System	Piping and fittings	Chrome Moly steels	Treated Water (Internal)	Loss of Material
Main Steam System	Piping and fittings	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material
Main Steam System	Piping and fittings	Carbon and low alloy steel	Steam (Internal)	Loss of Material
Main Steam System	Valve Body	Carbon and low alloy steel	Steam (Internal)	Loss of Material
Main Steam System	Valve Body	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material
Main Steam System	Valve Body	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material

SSC Name	Structure and/or Component	Material	Environment	Aging Effect
Main Steam System	Eductor	Carbon and low alloy steel	Steam (Internal)	Loss of Material
Main Steam System	Piping and fittings	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material
Main Steam System	Piping and fittings	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material
Main Steam System	Valve Body	Carbon and low alloy steel	Steam (Internal)	Loss of Material
Main Steam System	Piping and fittings	Carbon and low alloy steel	Steam (Internal)	Loss of Material
Main Steam System	Valve Body	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material
Main Steam System	Valve Body (Steam Chest)	Carbon and low alloy steel	Steam (Internal)	Loss of Material
Main Steam System	Piping and fittings	Carbon and low alloy steel	Steam (Internal)	Loss of Material
Main Steam System	Valve Body	Carbon and low alloy steel	Steam (Internal)	Loss of Material
Main Turbine and Auxiliary System	Heat Exchangers	Carbon and low alloy steel - Shell side component	Treated Water (Internal)	Loss of Material
Main Turbine and Auxiliary System	Heat Exchangers	Carbon and low alloy steel - Shell side component	Steam (Internal)	Loss of Material
Main Turbine and Auxiliary System	Piping and fittings	Carbon and low alloy steel	Steam (Internal)	Loss of Material
Main Turbine and Auxiliary System	Valve Body	Carbon and low alloy steel	Steam (Internal)	Loss of Material
Main Turbine and Auxiliary System	Valve Body	Carbon and low alloy steel	Steam (Internal)	Loss of Material
Main Turbine and Auxiliary System	Valve Body	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material
Main Turbine and Auxiliary System	Piping and fittings	Carbon and low alloy steel	Treated Water (Internal)	Loss of Material

SSC Name	Structure and/or Component	Material	Environment	Aging Effect
Main Turbine and Auxiliary System	Piping and fittings	Carbon and low alloy steel	Steam (Internal)	Loss of Material
Main Turbine and Auxiliary System	Heat Exchangers	Carbon and low alloy steel - Shell side component	Steam (Internal)	Loss of Material

Note: The FAC program has not been applied to the following systems:

- RWCU carbon steel bottom head drain line (not susceptible to FAC)
  - o Temperature: The wear factor from temperature is a bell curve that peaks around 300º F to 325º F. At the almost 500º F that this line sees, the oxide layer is more stable.
  - o Velocity: For this 2" line, operating flow of 6 GPM (ref. calc. No. C-1302-215-5360-004) provides a velocity of ~1 ft/sec. Flow rates less than 6 ft/sec do not need to be considered for flow-accelerated corrosion.
  - o Chemistry: The dissolved oxygen is the critical chemistry factor for BWR FAC. Normal Chemistry or Noble Metal Chemistry should provide negligible wall thinning for this line. The main concern for this location is at plants that have run with Moderate Hydrogen Water Chemistry for any length of time. According to the table that the EPRI CHECWORKS Users Group (CHUG) and BWRVIP are working with, OC has not run with MHWC, and therefore should not be experiencing high wear.
- Shutdown Cooling System (not susceptible to FAC)
  - o Shutdown cooling is a stand-by system that operates less than 2% of plant operating time. 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.
- Plant heating steam
  - o FAC is an applicable aging mechanism. The Periodic Inspection Program, B.2.5, will be used to manage the aging of plant heating steam piping, piping components, and piping elements subject to loss of material from flow-accelerated corrosion.

6.0 ATTACHMENTS

6.1 LRA Appendix A

6.2 LRA Appendix B

PROGRAM BASIS DOCUMENT

PBD-AMP-B.1.18

Revision 0

BWR REACTOR WATER CLEANUP SYSTEM

GALL PROGRAM XI.M25 - BWR REACTOR WATER CLEANUP SYSTEM

Prepared By: ______________________

Reviewed By: ______________________

Program Owner Review: ______________________

Technical Lead Approval: ______________________

Revision History:

Revision	*Prepared by:*	*Reviewed by:*	*Program Owner:*	*Approved by:*
0	*Mark A. Miller*	*Stu Getz*	*Greg Harttraft*	*Don Warfel*
Date				

Summary of Revisions:

Rev. Number	Reason for the Revision(s)
0	Initial Issue

TABLE OF CONTENTS

1.0 PURPOSE AND METHODOLOGY

1.1 Purpose

The purpose of this Program Basis Document is to document and evaluate those activities of the Oyster Creek BWR Reactor Water Cleanup System aging management program that are credited for managing stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) on stainless steel Reactor Water Cleanup System piping welds outboard of the second isolation valve as part of Oyster Creek License Renewal to meet the requirements of the License Renewal Rule 10 CFR 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants."

This includes the following:

- The identification of the scope of the program;
- The evaluation of program elements against NUREG-1801;
- The review of Operating Experience to demonstrate program effectiveness;
- The identification of required program enhancements;
- The identification of Oyster Creek documents required to implement the program.

1.2 Methodology

The nuclear power plant License Renewal Rule 10 CFR 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," describes the License Renewal process and provides requirements for the contents of License Renewal Applications. 10 CFR 54.21(a)(3) states:

"For each structure and component identified in paragraph (a)(1) of this section, demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the Current Licensing Basis (CLB) for the period of extended operation."

The NRC and the industry identified 10 program elements that are useful in describing an aging management program and then demonstrating its effectiveness. These program elements are described in Appendix A.1, Section A.1.2.3 of the Standard Review Plan. NUREG-1801 uses these program elements in Section XI to describe acceptable aging management programs.

This Program Basis Document provides a comparison of the credited Oyster Creek program with the elements of the corresponding NUREG-1801 Chapter XI program XI.M25, BWR Reactor Water Cleanup System. Project Level Instruction PLI-8 "Aging Management Program Reviews" prescribes the methodology for evaluating Aging Management Programs. An evaluation of Oyster Creeks aging management program criteria or activities to those of the NUREG-1801 program elements is performed and a conclusion is reached concerning consistency for the individual program element. A demonstration of overall program effectiveness is made after all program elements are evaluated. Required program enhancements are documented. An overall determination is made as to consistency with the program description in NUREG-1801.

2.0 PROGRAM DESCRIPTION

2.1 Program Description

NUREG-1801:

a) *The program includes inservice inspection (ISI) and monitoring and control of reactor coolant water chemistry to manage the effects of stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) on the intended function of austenitic stainless steel (SS) piping in the reactor water cleanup (RWCU) system.*

b) *Based on the Nuclear Regulatory Commission (NRC) criteria related to inspection guidelines for RWCU piping welds outboard of the second isolation valve, the program includes the measures delineated in NUREG-0313, Rev. 2, and NRC Generic Letter (GL) 88-01.*

c) *Coolant water chemistry is monitored and maintained in accordance with the Electric Power Research Institute (EPRI) guidelines in boiling water reactor vessel and internals project (BWRVIP)-29 (TR-103515) to minimize the potential of cracking due to SCC or IGSCC.*

Oyster Creek:

a) The BWR Reactor Water Cleanup System program describes the requirements for augmented inservice inspection (ISI) for stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) on stainless steel Reactor Water Cleanup System piping welds outboard of the second isolation valve. The program also includes water chemistry control in accordance with EPRI BWRVIP-130: "BWR Vessel and Internals Project BWR Water Chemistry Guidelines" to minimize the potential of crack initiation and growth due to SCC or IGSCC. The inservice inspection activities, and the control of reactor water chemistry, manage the aging effects in stainless steel Reactor Water Cleanup System piping welds outboard of the second isolation valve thereby maintaining the intended function of this piping.

b) The program includes inspection guidelines delineated in NUREG-0313, Rev. 2 and NRC Generic Letter (GL) 88-01 and includes the alternate measures approved by the NRC in BWRVIP-75 "BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules."

   In accordance with Generic Letter (GL) 88-01, Supplement 1, upgrades and enhancements have been implemented to the RWCU isolation valves in accordance with Generic Letter 89-10 to ensure that the valves will produce sufficient thrust to perform their design basis function, which is the isolation of containment in the event of a pipe break downstream of the valves. Based on these upgrades/enhancements, an effective Hydrogen Water Chemistry program, and the complete lack of cracking found during any of the RWCU piping weld inspections under Generic Letter 88-01, all inspection requirements for the portion of the RWCU System outboard of the second containment isolation valves have been eliminated.

c) Reactor coolant system (RCS) chemistry activities consist of preventive measures that are used to manage cracking in license renewal components exposed to reactor water and steam. RCS chemistry activities provide for monitoring and controlling RCS water chemistry using Oyster Creek procedures and processes based on BWRVIP-130, "BWR Vessel and Internals Project BWR Water Chemistry Guidelines." The BWR Water Chemistry Guidelines include information to develop proactive plant-specific water chemistry programs to minimize IGSCC. Refer to Water Chemistry Program Basis Document PBD-AMP-B.1.2 for chemistry specifics to mitigate IGSCC.

2.2 Overall NUREG-1801 Consistency

The Oyster Creek BWR Reactor Water Cleanup System program is an existing program that is consistent with NUREG-1801 aging management program XI.M25, BWR Reactor Water Cleanup System with one (1) exception as described in paragraph 2.3 below.

2.3 Summary of Exceptions to NUREG-1801

NUREG-1801 indicates that water chemistry control is in accordance with BWRVIP-29, "EPRI Report TR-103515-R1, BWR Water Chemistry Guidelines" dated 1996. The Oyster Creek water chemistry program is based on BWRVIP-130, "BWR Vessel and Internals Project BWR Water Chemistry Guidelines" dated 2004. For justification of exceptions, see Water Chemistry Program Basis Document PBD-AMP-B.1.2.

2.4 Summary of Enhancements to NUREG-1801

None. The existing Oyster Creek BWR Reactor Water Cleanup System aging management program is found to be adequate to support the extended period of operation with no enhancements.

3.0 EVALUATIONS AND TECHNICAL BASIS

Note

This section is organized by quoting the relevant NUREG-1801 Chapter XI program element (September 2005 version) followed by the related Oyster Creek program attributes and a conclusion of the comparison. Where applicable, the NUREG-1801 program element was separated into logical sub-elements and addressed accordingly.

Implementing procedure references are included in ( ) for information purposes. This information from the source procedure has been either directly extracted from the procedure or summarized for inclusion into this PBD.

3.1 Scope of Program

NUREG-1801:

Based on the NRC letter (September 15, 1995) on the screening criteria related to inspection guidelines for RWCU piping welds outboard of the second isolation valve, the program includes the measures delineated in NUREG-0313, Rev. 2, and NRC GL 88-01 to monitor SCC or IGSCC and its effects on the intended function of austenitic SS piping. The screening criteria include:

a) Satisfactory completion of all actions requested in NRC GL 89-10,

b) No detection of IGSCC in RWCU welds inboard of the second isolation valves (ongoing inspection in accordance with the guidance in NRC GL 88-01), and

c) No detection of IGSCC in RWCU welds outboard of the second isolation valves after inspecting a minimum of 10% of the susceptible piping.

No IGSCC inspection is recommended for plants that meet all the above three criteria or that meet criterion (a) and piping is made of material that is resistant to IGSCC.

Oyster Creek:

In accordance with Generic Letter (GL) 88-01, Supplement 1, upgrades and enhancements have been implemented to the RWCU isolation valves in accordance with Generic Letter 89-10 to ensure that the valves will produce sufficient thrust to perform their design basis function, which is the isolation of containment in the event of a pipe break downstream of the valves. Based on these upgrades/enhancements, an effective Hydrogen Water Chemistry program, and the complete lack of cracking found during any of the RWCU piping weld inspections under Generic Letter 88-01, all inspection requirements for the portion of the RWCU System outboard of the second containment isolation valves have been eliminated **(Reference: Letter No. 1940-00-20096 dated April 13, 2000 "Deletion of the Generic Letter 88-01 Augmented Inservice Inspection Requirements for the Reactor Water Cleanup System"; OC-2 IGSCC Inspection Program)**. Based on meeting all three criteria specified in NUREG-1801 Element 1, Scope of Program as described above, inspections of RWCU piping welds outboard of the second isolation valve are not required.

The Oyster Creek BWR Reactor Water Cleanup System aging management program is credited for managing stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) on stainless steel Reactor Water Cleanup System piping welds outboard of the second isolation valve as listed in Table 5.2. The implementing documents for this aging management program are listed in Table 5.1 and are described throughout the individual program element discussions. The commitment numbers under which these implementing documents are being revised are contained within the listings in Table 5.1.

Exceptions to NUREG-1801, Element 1:

None.

Enhancements to NUREG-1801, Element 1:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 1, Scope of Program.

3.2 Preventive Actions

NUREG-1801:

a) *The comprehensive program outlined in NUREG-0313 and NRC GL 88-01 addresses improvements in all three elements that, in combination, cause SCC or IGSCC. These elements are a susceptible (sensitized) material, a significant tensile stress, and an aggressive environment.*

b) *The program delineated in NUREG-0313 and NRC GL 88-01 includes recommendations regarding selection of materials that are resistant to sensitization, use of special processes that reduce residual tensile stresses, and monitoring and maintenance of coolant chemistry. The resistant materials are used for new and replacement components and include low-carbon grades of austenitic SS and weld metal, with a maximum carbon of 0.035 wt.% and a minimum ferrite of 7.5% in weld metal and cast austenitic stainless steel (CASS). Inconel 82 is the only commonly used nickel-base weld metal considered to be resistant to SCC; other nickel-alloys, such as Alloy 600, are evaluated on an individual basis.*

c) *Special processes are used for existing as well as new and replacement components. These processes include solution heat treatment, heat sink welding, induction heating, and mechanical stress improvement.*

d) *The program delineated in NUREG-0313 and NRC GL 88-01 varies depending on the plant- specific reactor water chemistry to mitigate SCC or IGSCC.*

Oyster Creek:

a) The BWR Reactor Water Cleanup System program describes the requirements for augmented inservice inspection (ISI) for stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) on stainless steel Reactor Water Cleanup System piping welds outboard of the second isolation valve. The program includes inspection guidelines delineated in NUREG-0313, Rev. 2 and NRC Generic Letter (GL) 88-01 and includes the alternate measures approved by the NRC in BWRVIP-75 "BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules." The program also provides for water chemistry control in accordance with EPRI BWRVIP-130: "BWR Vessel and Internals Project BWR Water Chemistry Guidelines" to minimize the potential of crack initiation and growth due to SCC or IGSCC **(References: OC-2, IGSCC Inspection Program; OC-1, ISI Program, Section 1, paragraph 2.2.1 and Section 2, paragraph 4.1.10; Water Chemistry Program Basis Document PBD-AMP-B.1.2).**

In accordance with Generic Letter (GL) 88-01, Supplement 1, upgrades and enhancements have been implemented to the RWCU isolation valves in accordance with Generic Letter 89-10 to ensure that the valves will produce sufficient thrust to perform their design basis function, which is the isolation of containment in the event of a pipe break downstream of the valves. Based on these upgrades/enhancements, an effective Hydrogen Water Chemistry program, and the complete lack of cracking found during any of the RWCU piping weld inspections under Generic Letter 88-01, all inspection requirements for the portion of the RWCU System outboard of the second containment isolation valves have been eliminated **(Reference: Letter No. 1940-00-20096 dated April 13, 2000 "Deletion of the Generic Letter 88-01 Augmented Inservice Inspection Requirements for the Reactor Water Cleanup System"; OC-2 IGSCC Inspection Program).** Based on meeting all three criteria specified in NUREG-1801 Element 1, Scope of Program as described above, inspections of RWCU piping welds outboard of the second isolation valve are not required.

b) Resistant materials used at Oyster Creek for new and replacement components include low-carbon grades of stainless steel. Weld metal carbon content is limited to 0.035 wt. % maximum with a minimum ferrite of 7.5% **(Reference: Specifications OC-IS-402585-001, SP-1302-32-027, and SP-1302-32-017).**

c) Special processes such as solution heat treatment and heat sink welding are used for existing as well as new and replacement components. Mechanical Stress Improvement Process (MSIP) or Induction Heating Stress Improvement (IHSI) has not been performed on the RWCU system welds **(Reference: Specifications OC-IS-402585-001, SP-1302-32-027, and SP-1302-32-017; OC-2 IGSCC Inspection Program).**

d) Reactor coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-130, "BWR Vessel and Internals Project BWR Water Chemistry Guidelines" to maintain high water purity to reduce susceptibility to SCC or IGSCC. The program description, evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Water Chemistry Program Basis Document PBD-AMP-B.1.2 **(Reference: Water Chemistry Program Basis Document PBD-AMP-B.1.2).**

Exceptions to NUREG-1801, Element 2:

NUREG-1801 indicates that water chemistry control is in accordance with BWRVIP-29, "EPRI Report TR-103515-R1, BWR Water Chemistry Guidelines" dated 1996. The Oyster Creek water chemistry program is based on BWRVIP-130, "BWR Vessel and Internals Project BWR Water Chemistry Guidelines" dated 2004.

Enhancements to NUREG-1801, Element 2:

None.

Comparison and Evaluation Conclusion:

This element is not consistent with NUREG-1801, Element 2, Preventive Actions. NUREG-1801 indicates that water chemistry control is in accordance with BWRVIP-29, "EPRI Report TR-103515-R1, BWR Water Chemistry Guidelines" dated 1996. The Oyster Creek water chemistry program is based on BWRVIP-130, "BWR Vessel and Internals Project BWR Water Chemistry Guidelines" dated 2004. For justification of exceptions, see Water Chemistry Program Basis Document PBD-AMP-B.1.2.

3.3 Parameters Monitored or Inspected

NUREG-1801:

The aging management program (AMP) monitors SCC or IGSCC of austenitic SS piping by detection and sizing of cracks by implementing the inspection guidelines delineated in the NRC screening criteria for the RWCU piping outboard of isolation valves. The following schedules are followed:

Schedule A: No inspection is required for plants that meet all three criteria set forth above, or if they meet only criterion (a). Piping is made of material that is resistant to IGSCC, as described above in preventive actions.

Schedule B: For plants that meet only criterion (a): Inspect at least 2% of the welds or two welds every refueling outage, whichever sample is larger.

Schedule C: For plants that do not meet criterion (a): Inspect at least 10% of the welds every refueling outage.

Oyster Creek:

In accordance with Generic Letter (GL) 88-01, Supplement 1, upgrades and enhancements have been implemented to the RWCU isolation valves in accordance with Generic Letter 89-10 to ensure that the valves will produce sufficient thrust to perform their design basis function, which is the isolation of containment in the event of a pipe break downstream of the valves. Based on these upgrades/enhancements, an effective Hydrogen Water Chemistry program, and the complete lack of cracking found during any of the RWCU piping weld inspections under Generic Letter 88-01, all inspection requirements for the portion of the RWCU System outboard of the second containment isolation valves have been eliminated **(Reference: Letter No. 1940-00-20096 dated April 13, 2000 "Deletion of the Generic Letter 88-01 Augmented**

Inservice Inspection Requirements for the Reactor Water Cleanup System"; OC-2 IGSCC Inspection Program). Based on meeting all three criteria specified in NUREG-1801 Element 1, Scope of Program as described above, inspections of RWCU piping welds outboard of the second isolation valve are not required.

Exceptions to NUREG-1801, Element 3:

None.

Enhancements to NUREG-1801, Element 3:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 3, Parameters Monitored or Inspected.

3.4 Detection of Aging Effects

NUREG-1801:

a) *The extent, method, and schedule of the inspection and test techniques delineated in the NRC inspection criteria for RWCU piping and NRC GL 88-01 are designed to maintain structural integrity and to detect aging effects before the loss of intended function of austenitic SS piping and fittings. Guidelines for the inspection schedule, methods, personnel, sample expansion, and leak detection guidelines are based on the guidelines of NRC GL 88-01.*

b) *The NRC GL 88-01 recommends that the detailed inspection procedure, components, and examination personnel be qualified by a formal program approved by the NRC. Inspection can reveal cracking and leakage of coolant.*

c) *The extent and frequency of inspections recommended by the program are based on the condition of each weld (e.g., whether the weldments were made from IGSCC-resistant material, whether a stress improvement process was applied to a weldment to reduce the residual stresses, and how the weld was repaired if it had been cracked).*

Oyster Creek:

a) The BWR Reactor Water Cleanup System program describes the requirements for augmented inservice inspection (ISI) for stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) on stainless steel Reactor Water Cleanup System piping welds outboard of the second isolation valve. The program includes inspection guidelines delineated in NUREG-0313, Rev. 2 and NRC Generic Letter (GL) 88-01 and includes the alternate measures approved by the NRC in BWRVIP-75 "BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules" **(References: OC-2 IGSCC Inspection Program; OC-1, ISI Program, Section 1, paragraph 2.2.1 and Section 2, paragraph 4.1.10).**

b) In accordance with Generic Letter (GL) 88-01, Supplement 1, upgrades and enhancements have been implemented to the RWCU isolation valves in accordance with Generic Letter 89-10 to ensure that the valves will produce sufficient thrust to perform their design basis function, which is the isolation of containment in the event of a pipe break downstream of the valves. Based on these upgrades/enhancements, an effective Hydrogen Water Chemistry program, and the complete lack of cracking found during any of the RWCU piping weld inspections under Generic Letter 88-01, all inspection requirements for the portion of the RWCU System outboard of the second containment isolation valves have been eliminated **(Reference: Letter No. 1940-00-20096 dated April 13, 2000 "Deletion of the Generic Letter 88-01 Augmented Inservice Inspection Requirements for the Reactor Water Cleanup System"; OC-2 IGSCC Inspection Program).** Based on meeting all three criteria specified in NUREG-1801 Element 1, Scope of Program as described above, inspections of RWCU piping welds outboard of the second isolation valve are not required.

c) In accordance with Generic Letter (GL) 88-01, Supplement 1, upgrades and enhancements have been implemented to the RWCU isolation valves in accordance with Generic Letter 89-10 to ensure that the valves will produce sufficient thrust to perform their design basis function, which is the isolation of containment in the event of a pipe break downstream of the valves. Based on these upgrades/enhancements, an effective Hydrogen Water Chemistry program, and the complete lack of cracking found during any of the RWCU piping weld inspections under Generic Letter 88-01, all inspection requirements for the portion of the RWCU System outboard of the second containment isolation valves have been eliminated **(Reference: Letter No. 1940-00-20096 dated April 13, 2000 "Deletion of the Generic Letter 88-01 Augmented Inservice Inspection Requirements for the Reactor Water Cleanup System"; OC-2 IGSCC Inspection Program)**. Based on meeting all three criteria specified in NUREG-1801 Element 1, Scope of Program as described above, inspections of RWCU piping welds outboard of the second isolation valve are not required.

Exceptions to NUREG-1801, Element 4:

None.

Enhancements to NUREG-1801, Element 4:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 4, Detection of Aging Effects.

3.5 Monitoring and Trending

NUREG-1801:

a) The extent and schedule for inspection in accordance with the recommendations of NRC GL 88-01 provide timely detection of cracks and leakage of coolant.

b) Based on inspection results, NRC GL 88-01 provides guidelines for additional samples of welds to be inspected when one or more cracked welds are found in a weld category.

Oyster Creek:

a) The BWR Reactor Water Cleanup System program describes the requirements for augmented inservice inspection (ISI) for stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) on stainless steel Reactor Water Cleanup System piping welds outboard of the second isolation valve. The program includes inspection guidelines delineated in NUREG-0313, Rev. 2 and NRC Generic Letter (GL) 88-01 and includes the alternate measures approved by the NRC in BWRVIP-75 "BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules" **(References: OC-2 IGSCC Inspection Program; OC-1, ISI Program, Section 1, paragraph 2.2.1 and Section 2, paragraph 4.1.10).**

b) In accordance with Generic Letter (GL) 88-01, Supplement 1, upgrades and enhancements have been implemented to the RWCU isolation valves in accordance with Generic Letter 89-10 to ensure that the valves will produce sufficient thrust to perform their design basis function, which is the isolation of containment in the event of a pipe break downstream of the valves. Based on these upgrades/enhancements, an effective Hydrogen Water Chemistry program, and the complete lack of cracking found during any of the RWCU piping weld inspections under Generic Letter 88-01, all inspection requirements for the portion of the RWCU System outboard of the second containment isolation valves have been eliminated **(Reference: Letter No. 1940-00-20096 dated April 13, 2000 "Deletion of the Generic Letter 88-01 Augmented Inservice Inspection Requirements for the Reactor Water Cleanup System"; OC-2 IGSCC Inspection Program).** Based on meeting all three criteria specified in NUREG-1801 Element 1, Scope of Program as described above, inspections of RWCU piping welds outboard of the second isolation valve are not required.

Exceptions to NUREG-1801, Element 5:

None.

Enhancements to NUREG-1801, Element 5:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 5, Monitoring and Trending.

3.6 Acceptance Criteria

NUREG-1801:

The NRC GL 88-01 recommends that any indication detected be evaluated in accordance with the requirements of ASME Section XI, Subsection IWB-3640 (2001 edition including the 2002 and 2003 Addenda).

Oyster Creek:

The BWR Reactor Water Cleanup System program describes the requirements for augmented inservice inspection (ISI) for stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) on stainless steel Reactor Water Cleanup System piping welds outboard of the second isolation valve. The program includes inspection guidelines delineated in NUREG-0313, Rev. 2 and NRC Generic Letter (GL) 88-01 and includes the alternate measures approved by the NRC in BWRVIP-75 "BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules" **(References: OC-2 IGSCC Inspection Program; OC-1, ISI Program, Section 1, paragraph 2.2.1 and Section 2, paragraph 4.1.10).**

In accordance with Generic Letter (GL) 88-01, Supplement 1, upgrades and enhancements have been implemented to the RWCU isolation valves in accordance with Generic Letter 89-10 to ensure that the valves will produce sufficient thrust to perform their design basis function, which is the isolation of containment in the event of a pipe break downstream of the valves. Based on these upgrades/enhancements, an effective Hydrogen Water Chemistry program, and the complete lack of cracking found during any of the RWCU piping weld inspections under Generic Letter 88-01, all inspection requirements for the portion of the RWCU System outboard of the second containment isolation valves have been eliminated **(Reference: Letter No. 1940-00-20096 dated April 13, 2000 "Deletion of the Generic Letter 88-01 Augmented Inservice Inspection Requirements for the Reactor Water Cleanup System"; OC-2 IGSCC Inspection Program).** Based on meeting all three criteria specified in NUREG-1801 Element 1, Scope of Program as described above, inspections of RWCU piping welds outboard of the second isolation valve are not required. Therefore, acceptance criteria for evaluation of detected indications in accordance with the requirements of ASME Section XI, Subsection IWB-3640 (2001 edition including the 2002 and 2003 Addenda) are not applicable.

Exceptions to NUREG-1801, Element 6:

None.

Enhancements to NUREG-1801, Element 6:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 6, Acceptance Criteria.

3.7 Corrective Actions

NUREG-1801:

The guidance for weld overlay repair, stress improvement, or replacement is provided in NRC GL 88-01. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the corrective actions.

Oyster Creek:

The BWR Reactor Water Cleanup System program describes the requirements for augmented inservice inspection (ISI) for stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) on stainless steel Reactor Water Cleanup System piping welds outboard of the second isolation valve. The program includes inspection guidelines delineated in NUREG-0313, Rev. 2 and NRC Generic Letter (GL) 88-01 and includes the alternate measures approved by the NRC in BWRVIP-75 "BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules" **(References: OC-2 IGSCC Inspection Program; OC-1, ISI Program, Section 1, paragraph 2.2.1 and Section 2, paragraph 4.1.10).**

In accordance with Generic Letter (GL) 88-01, Supplement 1, upgrades and enhancements have been implemented to the RWCU isolation valves in accordance with Generic Letter 89-10 to ensure that the valves will produce sufficient thrust to perform their design basis function, which is the isolation of containment in the event of a pipe break downstream of the valves. Based on these upgrades/enhancements, an effective Hydrogen Water Chemistry program, and the complete lack of cracking found during any of the RWCU piping weld inspections under Generic Letter 88-01, all inspection requirements for the portion of the RWCU System outboard of the second containment isolation valves have been eliminated **(Reference: Letter No. 1940-00-20096 dated April 13, 2000 "Deletion of the Generic Letter 88-01 Augmented Inservice Inspection Requirements for the Reactor Water Cleanup System"; OC-2 IGSCC Inspection Program).** Based on meeting all three criteria specified in NUREG-1801 Element 1, Scope of Program as described above, inspections of RWCU piping welds outboard of the second isolation valve are not required.

Any corrective actions associated with this aging management program will be addressed under the Oyster Creek 10 CFR Part 50, Appendix B quality assurance program.

<u>Exceptions to NUREG-1801, Element 7:</u>

None.

<u>Enhancements to NUREG-1801, Element 7:</u>

None.

<u>Comparison and Evaluation Conclusion:</u>

This element is consistent with NUREG-1801, Element 7, Corrective Actions.

3.8 Confirmation Process

<u>NUREG-1801:</u>

Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the confirmation process and administrative controls.

Oyster Creek:

Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B.

Exceptions to NUREG-1801, Element 8:

None.

Enhancements to NUREG-1801, Element 8:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 8, Confirmation Process.

3.9 Administrative Controls

NUREG-1801:

See Item 8, above.

Oyster Creek:

See Item 8, above.

Exceptions to NUREG-1801, Element 9:

None.

Enhancements to NUREG-1801, Element 9:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 9, Administrative Controls.

3.10 Operating Experience

NUREG-1801:

The IGSCC has occurred in small- and large-diameter boiling water reactor (BWR) piping made of austenitic stainless steels or nickel alloys. The comprehensive program outlined in NRC GL 88-01 and NUREG-0313 addresses improvements in all elements that cause SCC or IGSCC (e.g., susceptible material, significant tensile stress, and an aggressive environment) and is effective in managing IGSCC in austenitic SS piping in the RWCU system.

Oyster Creek:

Operating experience, both internal and external, is used in two ways at Oyster Creek to enhance plant programs, prevent repeat events, and prevent events that have occurred at other plants from occurring at Oyster Creek. The first way in which operating experience is used is through the Oyster Creek Operating Experience process. The Operating Experience process screens, evaluates, and acts on operating experience documents and information to prevent or mitigate the consequences of similar events. The second way is through the process for managing programs. This process requires the review of program related operating experience by the program owner.

Both of these processes review operating experience from both external and internal (also referred to as in-house) sources. External operating experience may include such things as INPO documents (e.g., SOERs, SERs, SENs, etc.), NRC documents (e.g., GLs, LERs, INs, etc.), Westinghouse documents (e.g., TBs, etc.), General Electric documents (e.g., RCSILs, SILs, TILs, etc.), and other documents (e.g., 10CFR Part 21 Reports, NERs, etc.). Internal operating experience may include such things as event investigations, trending reports, and lessons learned from in-house events as captured in program notebooks, self-assessments, and in the 10 CFR Part 50, Appendix B corrective action process.

IGSCC Program Background: (See PBD-AMP-B.1.07 BWR Stress Corrosion Cracking for specific IGSCC Program Operating Experience)

The Oyster Creek IGSCC inservice inspection program has been established for piping identified in NRC Generic Letter 88-01 and NUREG-0313, and includes the alternate measures approved by the NRC in BWRVIP-75 "BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules." The inspections and planned mitigating actions thru outage 13R for IGSCC were previously documented in GPUN Topical Report 050. Utilizing Topical Report 050 as the basis for IGSCC planning and implementation, IGSCC work scope for examinations and mitigating actions for outages 14R thru 17R were controlled thru unique specifications generated for each outage. After 17R Oyster Creek elected to follow the guidelines set forth in BWRVIP-75.

Since Generic Letter 88-01 was issued, Oyster Creek has performed numerous examinations on piping subject to the generic letter requirements. During this period, Oyster Creek has improved water chemistry, employed Hydrogen Water Chemistry (HWC), and Noble Metals Chemical Addition (NMCA) as IGSCC mitigators. In addition examination procedures have been improved and examination personnel have received training on the latest techniques for IGSCC detection and have gained years of experience in the detection and sizing of IGSCC.

Document Number OC-2, Rev. 0, "IGSCC Inspection Program (Generic Letter 88-01 and BWRVIP-75)" provides the basis and administrative controls required for the conduct of the Oyster Creek IGSCC inspection program and further addresses and clarifies Oyster Creek's commitment to the implementation of BWRVIP-75 as an alternate measure for IGSCC inspection as referenced in plant technical specifications, Section 4.3.I. In addition OC-2 includes and provides control of the database system used to manage scope and schedule examinations.

The BWRVIP-75 inspection program as delineated in OC-2 runs concurrent with the fourth Section XI ISI interval.

RWCU System Operating Experience:

Demonstration that the effects of aging are effectively managed is achieved through objective evidence that shows that stress corrosion cracking and intergranular stress corrosion cracking is being adequately managed in RWCU piping welds outboard of the second isolation valve. The following example of operating experience provides objective evidence that the BWR Reactor Water Cleanup System program is effective in assuring that intended function(s) will be maintained consistent with the CLB for the period of extended operation:

1. No indications of IGSCC have been found in RWCU, which is not stress improved. Table 2 of OC-2 includes a detailed list of RWCU System welds and summarizes the results of RWCU IGSCC inspections conducted to date. This example provides objective evidence that the implementation of the BWR Reactor Water Cleanup System aging management program has been effective in detecting stress corrosion cracking and intergranular stress corrosion cracking in RWCU piping welds outboard of the second isolation valve prior to the loss of intended functions.

3.11 Conclusion

The Oyster Creek BWR Reactor Water Cleanup System aging management program is credited for managing stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) on stainless steel Reactor Water Cleanup System piping welds outboard of the second isolation valve as listed in Table 5.2. The Oyster Creek BWR Reactor Water Cleanup System program's elements have been evaluated against NUREG-1801 in Section 3.0. Program exceptions have been identified in Section 2.3. Program enhancements have been identified in Section 2.4. The implementing documents and commitment numbers for this aging management program are listed in Table 5.1. The relevant operating experience has been reviewed and a demonstration of program effectiveness is provided in Section 3.10.

Based on the above, the continued implementation of the Oyster Creek BWR Reactor Water Cleanup System aging management program provides reasonable assurance that stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) on stainless steel Reactor Water Cleanup System piping welds outboard of the second isolation valve will be adequately managed so that the intended functions of components within the scope of license renewal will be maintained during the period of extended operation.

4.0 REFERENCES

4.1 Generic to Aging Management Programs

4.1.1 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

4.1.2 10 CFR 54, Requirements for Renewal of Operating Licenses for Nuclear Power Plants

4.1.3 NUREG-1800, *Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,* Revision 1, dated September 2005

4.1.4 NUREG-1801, *Generic Aging Lessons Learned (GALL) Report*, Revision 1, dated September 2005

4.2 Industry Standards

4.2.1 NRC Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping, U.S. Nuclear Regulatory Commission," January 25, 1988; Supplement 1, February 4, 1992

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

4.2.3 BWRVIP-75, "BWR Vessel and Internals Project Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedule," October 1999

4.2.4 Letter J. R. Strosnider (NRC) to C. Terry (BWRVIP Chairman) dated September 15, 2000 "Safety Evaluation of the "BWRVIP Vessel and Internals Project, BWR Vessel and Internals Project, Technical Basis for Revisions to Generic Letter 88-01 Inspection Schedules (BWRVIP-75)," EPRI Report TR-113932, October 1999 (TAC NO. MA5012)"

4.2.5 ASME Boiler and Pressure Vessel Code, Section XI, "Rules for In-service Inspection of Nuclear Plant Components," 1995 Edition through 1996 Addendum

4.3 Oyster Creek Program References

4.3.1 Document Number OC-2, Rev. 0, "IGSCC Inspection Program (Generic Letter 88-01 and BWRVIP-75)"

4.3.2 Document Number OC-1, Rev. 1 "ISI Program Plan, Oyster Creek Nuclear Generating Station, Fourth Interval"

4.3.3 OC-IS-402585-001, Rev. 5, "Installation Specification for Repair and Replacement of Reactor Coolant Systems Piping Oyster Creek Nuclear Generating Station"

4.3.4 SP-1302-32-027, Rev. 2, "Material Conformance Specification for Procurement of Nuclear Grade Stainless Steel Components for Isolation Condenser System, Reactor Water Cleanup System, and Reactor Closure Head Pipe Modifications"

4.3.5 SP-1302-32-017, Rev. 1, "Technical Specification for Manufacture of Austenitic Stainless Steel Pipe, Fittings, and Spool Pieces for Oyster Creek Nuclear Generating Station"

4.3.6 GU 3E-000-A3-002 sh. 2, "Isometric Composite Clean-up Demineralizer IGSCC Weld Identification"

4.3.7 GU 3E-000-A3-002, sh. 4 and 5, "Isometric Composite Various Systems IGSCC Weld History"

5.0 TABLES

5.1 Aging Management Program Implementing Documents

Procedure Number	Procedure Title	Commitment No.	Status
OC-1	ISI Program Plan Fourth Ten-Year Inspection Interval	00330592.18.01	ACC/ASG
OC-2	IGSCC Inspection Program Fourth Ten-Year Interval	00330592.18.02	ACC/ASG
ER-AA-330-002	Inservice Inspection of Section XI Welds and Components	00330592.18.03	ACC/ASG
ER-AA-330-009	ASME Section XI Repair-Replacement Program	00330592.18.04	ACC/ASG

5.2 Aging Management Review Results

SSC Name	Structure and/or Component	Material	Environment	Aging Effect
Reactor Water Cleanup System	Piping and fittings	Stainless Steel	Treated Water (Internal)	Cracking Initiation and Growth

Note: Includes 4" nominal pipe size and larger RWCU System piping outboard of the second isolation valves exposed to coolant temperature above 140 Deg F during power operation. See IGSCC weld identification isometric composite drawing GU 3E-000-A3-002 sh. 2 for the extent of this program boundary as it relates to the RWCU system.

6.0 ATTACHMENTS

6.1 LRA Appendix A

6.2 LRA Appendix B