From: <john.hufnagel@exeloncorp.com>
To: <dja1@nrc.gov>, <gvc@nrc.gov>
Date: 12/09/2005 3:03:41 PM
Subject: Batch Four - 7 of 9 PBDs due 12/12/05

Donnie/Greg,

Here are seven of the nine AMP PBDs that we had indicated we would provide by Monday, 12/12.(Batch 4). This brings to 26 the number of upgraded program basis documents that we have provided for the Auditors review. The other two from Batch 4 will be provided on Monday, 12/12.

Attached please find the following seven PBDs in Word format: PBD B.3.01, Metal Fatigue; PBD B.1.36, Electrical (E-3); PBD B.1.13, Open Cycle Cooling Water; PBD B.1.06, BWR CRD return nozzle; PBD B.1.03, Reactor Head Closure Studs; PBD B.1.30, Masonry Walls and PBD B.1.05, Feedwater Nozzles.

Note that these Word files have been "write" protected to prevent inadvertent revisions to the files. This should not preclude viewing, copying, pasting, etc. Let us know if there are any problems.

As you know, these are being provided in response to AMP Audit question AMP-147. When we transmit the final two documents of Batch 4, I will include an updated answer to Audit question AMP-147, which will then reflect that we have provided all AMP basis documents up through Batch 4, and list those documents.

Please let me know if there are any questions/problems.

- John.

<<PBD B.3.01 Metal Fatigue Rev 0.doc>> <<PBD B.1.36 E-3 Rev 0.doc>> <<PBD B.1.13 Rev.0.doc>> <<PBD B.1.06 CRD Return Line Rev 0.doc>> <<PBD B.1.03 Rx Head Studs Rev 0.doc>> <<PBD B.1.30 Masonry Walls Rev. 0.doc>> <<PBD B.1.05 Feedwater Nozzles Rev 0 .doc>>

CC: <fred.polaski@exeloncorp.com>

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Created By: john.hufnagel@exeloncorp.com

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nrc.gov
owf4_po.OWFN_DO
GVC (Gregory Cranston)

exeloncorp.com
fred.polaski CC

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TEXT.htm	3554	
PBD B.3.01 Metal Fatigue Rev 0.doc		505344
PBD B.1.36 E-3 Rev 0.doc	140800	
PBD B.1.13 Rev.0.doc	238080	
PBD B.1.06 CRD Return Line Rev 0.doc		128512
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PBD B.1.30 Masonry Walls Rev. 0.doc		122880
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PROGRAM BASIS DOCUMENT

PBD-AMP-B.3.01

Revision 0

METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

GALL PROGRAM X.M1
METAL FATIGUE OF REACTOR COOLANT PRESSURE BOUNDARY

Prepared By: M. J. May

Reviewed By: S. C. Getz

Program Owner Review: D. P. Olszewski

Technical Lead Approval: D. B. Warfel

Revision History:

Revision	*Prepared by:*	*Reviewed by:*	*Program Owner:*	*Approved by:*
0	*M. J. May*	S. C. Getz	D. P. Olszewski	*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 Metal Fatigue of Reactor Coolant Pressure Boundary aging management program that are credited for managing the effects of fatigue 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; and
- 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 X program X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary. Project Level Instruction PLI-8, "Program Basis Documents," prescribes the methodology for evaluating Aging Management Programs. An evaluation of the Oyster Creek 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

<u>NUREG-1801:</u>

a) *In order not to exceed the design limit on fatigue usage, the aging management program (AMP) monitors and tracks the number of critical thermal and pressure transients for the selected reactor coolant system components.*

b) *The AMP addresses the effects of the coolant environment on component fatigue life by assessing the impact of the reactor coolant environment on a sample of critical components for the plant. Examples of critical components are identified in NUREG/CR-6260.*

c) *The sample of critical components can be evaluated by applying environmental life correction factors to the existing ASME Code fatigue analyses. Formulae for calculating the environmental life correction factors are contained in NUREG/CR-6583 for carbon and low-alloy steels and in NUREG/CR-5704 for austenitic stainless steels.*

d) *As evaluated below, this is an acceptable option for managing metal fatigue for the reactor coolant pressure boundary, considering environmental effects. Thus, no further evaluation is recommended for license renewal if the applicant selects this option under 10 CFR 54.21(c)(1)(iii) to evaluate metal fatigue for the reactor coolant pressure boundary.*

<u>Oyster Creek:</u>

a) In order not to exceed the design limit on fatigue usage, the existing Oyster Creek Metal Fatigue of Reactor Coolant Pressure Boundary (MFRCPB) aging management program monitors and tracks the number of critical thermal and pressure transients for the selected reactor coolant system components and compares the cycle counts to those assumed in the analysis of record. Cycle counting is managed at Oyster Creek through the Oyster Creek Transient / Cycle Fatigue Monitoring Program. The current MFRCPB program records thermal cycles on the Oyster Creek Transient / Cycle Summary Log, which is updated periodically. Plant operational data recorded by the Thermal Performance Engineer at Oyster Creek is periodically reviewed along with control room operator logs to determine if a transient meets the definition for a particular cycle type. All transients that meet the cycle type definition are added to the summary log.

The program will be enhanced to broaden the scope to additional locations in the Class I pressure boundary[1] and to some containment components and structures, and to update implementation methods. Changes to the program include the installation of the FatiguePro® fatigue monitoring software program that will monitor thermal cycles and fatigue usage that will be computed for selected bounding components. The program requires periodic assessment to ensure that cyclic limits and fatigue usage are not exceeded. The FatiguePro® fatigue management program counts fatigue stress cycles and tracks fatigue usage factors for selected components. The program consists of analytical methods to determine stress cycles and fatigue usage factors from operating cycles, automated counting of operating cycles, and automated calculation and tracking of the resulting fatigue cumulative usage factors (CUFs). The program tracks bounding locations in the reactor pressure vessel (RPV) Class I piping, isolation condenser and attached piping, and containment Torus and attached piping, in order to manage fatigue in these components and structures. FatiguePro® is an EPRI-licensed computerized data acquisition, recording and tracking program that has been customized for Oyster Creek use.

Oyster Creek "Class I" piping systems were originally designed to the requirements of USAS B31.1. The bounding set of

1 : "Class I" at Oyster Creek is not ASME III Class 1. At Oyster Creek "Class I" does not mean the set of structures, systems, and components with an ASME Section III Class 1 analysis, nor does it necessarily mean those with a seismic analysis. "Class I" structures and equipment are those whose failure could cause a significant release of radioactivity or which are vital for a safe shutdown of the plant (and for removal of decay heat, at Oyster Creek)

locations for the Oyster Creek FatiguePro® program were determined as follows:

1. By screening all existing ASME Section III reactor vessel and ASME Section III Class 1 piping fatigue analyses to identify locations with a design basis 40-year CUF greater than 0.4. All locations that exceed this CUF have been incorporated into FatiguePro® or dispositioned by monitoring other, more-controlling locations within the same system. Using a fatigue usage criterion of 0.4 to select components to monitor provides margin to the acceptance limit when extrapolating the calculated 40-years fatigue usage to 60 years **(Reference 4.3.16, paragraph 3.1).**
2. By screening all usage factor locations of the Mark I Containment Plant Unique Analyses for the torus, torus vents, attached piping, and Torus penetrations to identify locations with a design basis 40-year CUF greater than 0.4. All appropriate locations were included in the FatiguePro® program or dispositioned by monitoring other, more-controlling locations within the same system **(Reference 4.3.16, paragraph 3.4).**
3. By screening all usage factor locations of the isolation condenser analyses for the isolation condenser and attached piping to identify locations with a design basis 40-year CUF greater than 0.4. All appropriate locations were included in the FatiguePro® program or dispositioned by monitoring other, more-controlling locations within the same system **(Reference 4.3.16, paragraph 3.5).**
4. Since B31.1 rules were used for the original unmodified piping systems, the governing analyses for these systems did not include an explicit fatigue design basis and no CUFs were calculated. For those "Class I" piping systems that are potentially exposed to significant thermal cycling, fatigue analyses were developed to establish 40-year design basis CUFs, for bounding locations, to establish a valid fatigue design basis so that all "Class I" systems can be monitored, on a CUF basis, in FatiguePro® **(Reference 4.3.16, paragraph 3.3).**

FatiguePro monitors CUF for the selected locations in one of two ways: **(Reference 4.3.16, paragraph 1.0)**

1. Stress-Based Fatigue Monitoring: Stress-based fatigue (SBF) monitoring consists of computing a "real time" stress history for a given component from actual temperature,

pressure, and flow histories. CUF is then computed from the computed stress history using appropriate techniques. SBF methodology provides the most extensive, refined fatigue analysis for a component, and provides structural margin assessment without the need for categorizing and counting plant transients. SBF monitoring is intended for those high-fatigue components where a more refined approach is necessary to show long-term structural acceptability.

2. Cycle-Based Fatigue Monitoring: Cycle-based fatigue (CBF) monitoring consists of a two-step process: (a) automated cycle counting, and (b) CUF computation based on the counted cycles:

   a. Automated Cycle Counting: The FatiguePro® automated cycle counting (ACC) module categorizes and counts plant transients. The ACC module counts each transient that is defined in the plant licensing basis based on the mechanistic process or sequence of events experienced by the plant (as determined from the monitored plant instruments). This approach is conservative because it assumes each actual transient has a severity equal to that assumed in the design basis. The unique severity of any transient identified by FatiguePro® is also captured for each monitored component. All transients defined in the design basis and the plant Technical Specifications are identified and considered for implementation in the ACC module. Any additional system-specific transients that are experienced by the Class I piping and the Mark I Containment systems, which contribute significantly to the calculated CUF, are also monitored by FatiguePro®.

   b. CUF Computation: Cycle-based fatigue computation calculates fatigue directly from counted transients and parameters, as determined by the ACC module, for the RPV, Mark I Containment and attached piping, isolation condenser and attached piping, and Class I piping components. Limiting components in these systems are selected for monitoring that bound or represent all other components. CUF is computed using a design-basis fatigue calculation where the fatigue table from the governing fatigue calculation is used as a basis, but actual numbers of cycles are substituted for assumed design basis numbers of cycles. This methodology is intended for components where long-term structural acceptability can readily be shown based on cycle counts alone. Inclusion of Class I piping and the Mark I Containment systems into the fatigue management program provides a complete structural assessment of the Class I pressure boundary and all related fatigue boundaries throughout the plant.

b) Effects of the Coolant Environment on Component Fatigue Life: To address the effects of the coolant environment on component fatigue life plant-specific calculations have been performed for the locations identified in NUREG/CR-6260 for older-vintage GE BWR plants. The six locations are:

1. Reactor Vessel (Lower Head to Shell Transition)
2. Feedwater Nozzle
3. Recirculation System (SDC Return Line Tee), including the RPV recirculation inlet and outlet nozzles
4. Core Spray System (Nozzle and Safe End)
5. Isolation Condenser Return to Shutdown Cooling
6. Limiting Feedwater Line Location

c) Detailed environmental fatigue calculations have been performed for Oyster Creek using the appropriate environmental life correction factors (F_{en}) relationships from NUREG/CR-6583 for carbon and low-alloy steels, and from NUREG/CR-5704 for stainless steels, as appropriate for the material at each location. The calculations determined either a conservative bounding F_{en} multiplier for the total CUF for each component, or an appropriate F_{en} multiplier for each individual load pair in the governing fatigue calculation for each component, so that an overall CUF multiplier for environmental effects was determined for each location. For the locations listed above the CUF values that include the effects of the coolant environment on component fatigue life will be used as the basis for monitoring cumulative fatigue as shown in Table 5.5 **(Reference 4.3.8).**

d) As described in paragraph 4.3 of the Oyster Creek LRA the fatigue analyses for RCPB components are time limited aging analyses (TLAAs) as defined by 10 CFR 54.3(a). The option selected to disposition these TLAAs in accordance with 10 CFR 54.21(c)(1)(iii) is the MFRCPB aging management program described in this program basis document

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2.2 Overall NUREG-1801 Consistency

The Oyster Creek MFRCPB program is an existing program that with the enhancements described in paragraph 2.4 is consistent with NUREG-1801 aging management program X.M1, Metal Fatigue of Reactor Coolant Pressure Boundary

2.3 Summary of Exceptions to NUREG-1801

None. The enhanced Oyster Creek MFRCPB is found to be adequate to support the extended period of operation with no exceptions.

2.4 Summary of Enhancements to NUREG-1801

The program will be enhanced to use the EPRI-licensed FatiguePro® cycle counting and fatigue usage factor tracking computer program. The computer program provides for calculation of stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles and automated calculation and tracking of fatigue cumulative usage factors.

The program will provide for calculating and tracking of the cumulative usage factors for bounding locations for the reactor pressure vessel, Class I piping, the torus, torus vents, torus attached piping and penetrations, and the isolation condenser. The monitoring sample will include those locations where the predicted 40-year cumulative fatigue usage had been predicted to be 0.4 or greater, including the locations specified in NUREG/CR-6260, when applicable to Oyster Creek.

3.0 EVALUATIONS AND TECHNICAL BASIS

Note

This section is organized by quoting the relevant NUREG-1801 Chapter X 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.

3.0 Scope of Program

NUREG-1801:

The program includes preventive measures to mitigate fatigue cracking of metal components of the reactor coolant pressure boundary caused by anticipated cyclic strains in the material.

Oyster Creek:

The Oyster Creek MFRCPB aging management program includes preventive measures to mitigate fatigue cracking of metal components of the reactor coolant pressure boundary caused by anticipated cyclic strains in the material. The program provides automated cycle counting and fatigue cumulative usage factor (CUF) tracking activities that monitor critical components of the reactor vessel and Class I piping reactor coolant pressure boundary, isolation condenser, and the primary containment locations.

Bounding locations for monitoring these components were determined by an evaluation of the ASME Section III fatigue analyses of the reactor vessel and isolation condenser, newly-generated piping fatigue analyses for the "Class I" piping, and the Mark I Containment Plant Unique Analyses.

Determination of Scope: Oyster Creek "Class I" piping systems were originally designed to the requirements of USAS B31.1. The bounding set of locations for the Oyster Creek FatiguePro® program were determined as follows:

a. By screening all existing ASME Section III reactor vessel and ASME Section III Class 1 piping fatigue analyses to identify all locations with a design basis 40-year CUF greater than 0.4. All locations that exceed this CUF have been incorporated into FatiguePro® or dispositioned by monitoring other, more-controlling locations within the same system. Using a fatigue usage criterion of 0.4 to select components to monitor provides margin to the acceptance limit when extrapolating the calculated fatigue usage for 40 years to 60 years **(Reference 4.3.16, paragraph 3.1).**

b. By screening all usage factor locations of the Mark I Containment Plant Unique Analyses for the torus, torus vents, attached piping, and Torus penetrations with a design basis 40-year CUF greater than 0.4. All appropriate locations were included in the FatiguePro® program or dispositioned by monitoring other, more-controlling locations within the same system **(Reference 4.3.16, paragraph 3.4).**

c. By screening all usage factor locations of the isolation condenser analyses for the isolation condenser and attached piping with a design basis 40-year CUF greater than 0.4. All appropriate locations were included in the FatiguePro® program or dispositioned by monitoring other, more-controlling locations within the same system **(Reference 4.3.16, paragraph 3.5).**

d. Since B31.1 rules were used for the original unmodified piping systems, the governing analyses for these systems did not include an explicit fatigue design basis and no CUFs were calculated. For those "Class I" piping systems that are potentially exposed to significant thermal cycling, fatigue analyses were developed to establish 40-year design basis CUFs, for bounding locations, to establish a valid fatigue design basis so that all "Class I" systems can be monitored, on a CUF basis, in FatiguePro® **(Reference 4.3.16, paragraph 3.3)**

The result of the above screening process results in a list of locations that bound the controlling fatigue factors and will be monitored by FatiguePro®. This list is provided in Table 5.4. Table 5.4 also lists the predicted 60-year cumulative usage factor for these locations. In some cases the predicted 60-year usage factor is less than 0.4, but included based on past experience or engineering judgment that these points may be of interest in the future. The basis for the usage factors for the components listed in Table 5.4 is documented in the fatigue calculations that are listed in Section 4.3 of the Oyster Creek LRA.

The MFRCPB program will provide evaluations of CUF or cycle count tracking results that do not satisfy acceptance criteria, and will define actions to be taken. The MFRCPB program defines actions to be taken when cycle counts or usage factors acceptance limits are projected to be exceeded, as described in element 7 below **(References: 4.3.16; ER-AA-470, paragraph 4.4).**

The Oyster Creek Metal Fatigue of Reactor Pressure Coolant Pressure Boundary aging management program manages the aging effect of fatigue cracking due to cyclic strain for critical components in the reactor vessel and Class I piping reactor coolant pressure boundary, isolation condenser and attached piping, the containment Torus and attached piping for materials 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.

<u>Exceptions to NUREG-1801, Element 1:</u>

None.

<u>Enhancements to NUREG-1801, Element 1:</u>

The program will be enhanced to use the EPRI-licensed FatiguePro® cycle counting and fatigue usage factor tracking computer program. The computer program provides for calculation of stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles and automated calculation and tracking of fatigue cumulative usage factors.

The program will provide for calculating and tracking of the cumulative usage factors for bounding locations for the reactor pressure vessel, Class I piping, the torus, torus vents, torus attached piping and penetrations, and the isolation condenser. The monitoring sample will include those locations where the predicted 40-year cumulative fatigue usage had been predicted to be 0.4 or greater, including the locations specified in NUREG/CR-6260, when applicable to Oyster Creek.

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

This element is consistent with NUREG-1801, Element 1, Scope of Program with the enhancements described above.

3.1 Preventive Actions

NUREG-1801:

Maintaining the fatigue usage factor below the design code limit and considering the effect of the reactor water environment, as described under the program description, will provide adequate margin against fatigue cracking of reactor coolant system components due to anticipated cyclic strains.

Oyster Creek:

The enhanced MFRCPB program continuously monitors plant operational events, calculates usage factors for all monitored locations, and compares the accumulated data to allowable values. Bounding fatigue usage locations are tabulated inTable 5.4. The locations listed include those identified in NUREG/CR-6260. The MFRCPB aging management program monitors and trends fatigue CUF and allows corrective measures to be implemented in time to ensure that structural margins required by Codes used in the original plant design are maintained throughout the operating life of the plant. This process ensures the fatigue usage factor is maintained below the design code limit. The calculated fatigue usage factors consider effects of the reactor water environment by performing the fatigue analyses as described in element 5, Monitoring and Trending, of this program basis document. The MFRCPB program will provide adequate margin against fatigue cracking of reactor coolant system components due to anticipated cyclic strains. **(References: 4.3.16; ER-AA-470, paragraph 4.4)**

Exceptions to NUREG-1801, Element 2:

None.

Enhancements to NUREG-1801, Element 2:

The program will be enhanced to use the EPRI-licensed FatiguePro® cycle counting and fatigue usage factor tracking computer program. The computer program provides for calculation of stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles and automated calculation and tracking of fatigue cumulative usage factors.

The program will provide for calculating and tracking of the cumulative usage factors for bounding locations for the reactor pressure vessel, Class I piping, the torus, torus vents, torus attached piping and penetrations, and the isolation condenser. The monitoring sample will include those locations where the predicted 40-year cumulative fatigue usage had been predicted to be 0.4 or greater, including the locations specified in NUREG/CR-6260, when applicable to Oyster Creek.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 2, Preventive Actions with the enhancements described above.

3.2 Parameters Monitored or Inspected

NUREG-1801:

a) *The program monitors all plant transients that cause cyclic strains, which are significant contributors to the fatigue usage factor. The number of plant transients that cause significant fatigue usage for each critical reactor coolant pressure boundary component is to be monitored.*

b) *Alternatively, more detailed local monitoring of the plant transient may be used to compute the actual fatigue usage for each transient.*

Oyster Creek:

a) The current MFRCPB program records thermal cycles on the Oyster Creek Transient / Cycle Summary Log, which is updated periodically. Plant operational data recorded by the Thermal Performance Engineer at Oyster Creek is periodically reviewed along with control room operator logs to determine if a transient meets the definition for a particular cycle type. All transients that meet the cycle type definition are added to the summary log.

   The plant corrective action program is also reviewed to provide input from various plant transients that may have occurred since the last summary update.

   The program will be enhanced to use FatiguePro® to monitor thermal cycles and transients as a basis for monitoring fatigue. The results of the Oyster Creek Transient / Cycle Summary were used to establish a baseline. Not all of the plant transients needed for FatiguePro® have been tracked by the Oyster Creek Transient / Cycle Summary Log. The cycle count for several transients for some components and systems had to be added, including those for the Isolation Condenser, EMRV, and Shutdown Cooling. Plant records from first operation were reviewed to determine the number of cycles incurred for each event from first operation to present. Records searches included operator logs, Event reports, LERs, plant surveillance test reports, monthly and annual reports and plant performance data. This data was combined with the Oyster Creek Transient/ Cycle Summary Log to form the baseline history for FatiguePro®. **(Reference 4.3.17, 4.3.18).**

The enhanced MFRCPB program monitors plant transients that contribute to the fatigue usage factor for the reactor vessel and each of the selected components in the "Class I" piping, isolation condenser and attached piping, and Torus structure and attached piping included in the program scope. The enhanced MFRCPB aging management program consists of automated cycle counting and fatigue usage factor (CUF) tracking activities that monitor the number of transients for critical components of the reactor vessel and "Class I" piping reactor coolant pressure boundary, isolation condenser, and the primary containment locations. The transients monitored in the enhanced MFRCPB program are listed in Table 5.3. The transients in Table 5.3 are the same transients that are described in Table 4.3.1-1 of the Oyster Creek LRA **(Reference: 4.3.7).**

b) For components subject to complex thermal cycle events a detailed approach of fatigue monitoring is employed. FatiguePro® monitors CUF for the selected locations in one of two ways: **(Reference 4.3.16, paragraph 1.0)**

1. *Stress-Based Fatigue Monitoring*: Stress-based fatigue (SBF) monitoring is intended for those high-fatigue components where a more refined approach is necessary to show long-term structural acceptability. Stress-based fatigue monitoring consists of computing a "real time" stress history for a given component from actual temperature, pressure, and flow histories. CUF is then computed from the computed stress history using appropriate techniques. SBF methodology provides the most extensive, refined fatigue analysis for a component, and provides structural margin assessment without the need for categorizing and counting plant transients. This method is used for the monitoring the feedwater nozzle, because of the complexity of the thermal cycles experienced by the feedwater nozzle and because of a history feedwater nozzle cracking experienced in the

industry.

2. Cycle-Based Fatigue Monitoring: The remainder of the locations monitored by FatiguePro® employ a cycle-based approach. Cycle-based fatigue (CBF) monitoring consists of a two-step process: (a) automated cycle counting, and (b) CUF computation based on the counted cycles:

   (a) Automated Cycle Counting: The FatiguePro® automated cycle counting (ACC) module categorizes and counts plant transients. The ACC module counts each transient that is defined in the plant licensing basis based on the mechanistic process or sequence of events experienced by the plant (as determined from the monitored plant instruments). This approach is conservative because it assumes each actual transient has a severity equal to that assumed in the design basis. The unique severity of any transient identified by FatiguePro® is also captured for each monitored component. All transients defined in the design basis and the plant Technical Specifications are identified and considered for implementation in the ACC module. Any additional system-specific transients that are experienced by the Class I piping and the Mark I Containment systems, which contribute significantly to the calculated CUF, are also monitored by FatiguePro®.

   (b) CUF Computation: Cycle-based fatigue computation calculates fatigue directly from counted transients and parameters, as determined by the ACC module, for the RPV, Mark I Containment and attached piping, isolation condenser and attached piping, and Class I piping components. Limiting components in these systems are selected for monitoring that bound or represent all other components. CUF is computed using a design-basis fatigue calculation where the fatigue table from the governing fatigue calculation is used as a basis, but actual numbers of cycles are substituted for assumed design basis numbers of cycles. This methodology is intended for components where long-term structural acceptability can readily be shown based on cycle counts alone. Inclusion of Class the Mark I Containment systems and I piping into the fatigue management program provides a complete structural assessment of the Class I pressure boundary and all related fatigue boundaries throughout the plant.

<u>Exceptions to NUREG-1801, Element 3:</u>

None.

<u>Enhancements to NUREG-1801, Element 3:</u>

The program will be enhanced to use the EPRI-licensed FatiguePro® cycle counting and fatigue usage factor tracking computer program. The computer program provides for calculation of stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles and automated calculation and tracking of fatigue cumulative usage factors.

The program will provide for calculating and tracking of the cumulative usage factors for bounding locations for the reactor pressure vessel, Class I piping, the torus, torus vents, torus attached piping and penetrations, and the isolation condenser. The monitoring sample will include those locations where the predicted 40-year cumulative fatigue usage had been predicted to be 0.4 or greater, including the locations specified in NUREG/CR-6260, when applicable to Oyster Creek.

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

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

Detection of Aging Effects

<u>NUREG-1801:</u>

The program provides for periodic update of the fatigue usage calculations.

<u>Oyster Creek:</u>

The enhanced MFRCPB provides for periodic update of the fatigue usage calculations by continuously monitoring plant operational events. The program calculates usage factors for all monitored locations, and compares the accumulated data to allowable values. The responsible plant engineer collects the data approximately monthly. The data is reviewed to identify the need for any corrective actions. Semi-annually a report is generated for management that summarizes the transients and changes to fatigue usage incurred during the previous reporting period.

(Reference: ER-AA-470, paragraphs Section 4.4, 4.6)

Exceptions to NUREG-1801, Element 4:

None.

Enhancements to NUREG-1801, Element 4:

The program will be enhanced to use the EPRI-licensed FatiguePro® cycle counting and fatigue usage factor tracking computer program. The computer program provides for calculation of stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles and automated calculation and tracking of fatigue cumulative usage factors.

The program will provide for calculating and tracking of the cumulative usage factors for bounding locations for the reactor pressure vessel, Class I piping, the torus, torus vents, torus attached piping and penetrations, and the isolation condenser. The monitoring sample will include those locations where the predicted 40-year cumulative fatigue usage had been predicted to be 0.4 or greater, including the locations specified in NUREG/CR-6260, when applicable to Oyster Creek.

Comparison and Evaluation Conclusion:

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

3.3 Detection of Aging Effects

NUREG-1801:

The program provides for periodic update of the fatigue usage calculations.

Oyster Creek:

The enhanced MFRCPB provides for periodic update of the fatigue usage calculations by continuously monitoring plant operational events. The program calculates usage factors for all monitored locations, and compares the accumulated data to allowable values. The responsible plant engineer collects the data approximately monthly. The data is reviewed to identify the need for any corrective actions. Semi-annually a report is generated for management that summarizes the transients and changes to fatigue usage incurred during the previous reporting period. **(Reference: ER-AA-470, paragraphs Section 4.4, 4.6)**

Exceptions to NUREG-1801, Element 4:

None.

Enhancements to NUREG-1801, Element 4:

The program will be enhanced to use the EPRI-licensed FatiguePro® cycle counting and fatigue usage factor tracking computer program. The computer program provides for calculation of stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles and automated calculation and tracking of fatigue cumulative usage factors.

The program will provide for calculating and tracking of the cumulative usage factors for bounding locations for the reactor pressure vessel, Class I piping, the torus, torus vents, torus attached piping and penetrations, and the isolation condenser. The monitoring sample will include those locations where the predicted 40-year cumulative fatigue usage had been predicted to be 0.4 or greater, including the locations specified in NUREG/CR-6260, when applicable to Oyster Creek.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 4, Detection of Aging Effects with the enhancements described

3.4 Monitoring and Trending

NUREG-1801:

The program monitors a sample of high fatigue usage locations. This sample is to include the locations identified in NUREG/CR-6260, as minimum, or propose alternatives based on plant configuration.

Oyster Creek:

The fatigue management program consists of automated cycle counting and fatigue usage factor (CUF) tracking activities that monitor a sample of high fatigue usage factor components of the reactor vessel and Class I piping reactor coolant pressure boundary, isolation condenser, and the primary containment locations, including locations defined in NUREG/CR-6260. **(Reference: 4.3.16; 4.3.1; 4.3.8)**

Bounding locations for monitoring these components were determined by an evaluation of the ASME Section III fatigue analyses of the reactor vessel and isolation condenser, newly-generated piping fatigue analyses for the "Class I" piping, and the Mark I Containment Plant Unique Analyses.

Determination of Scope: Oyster Creek "Class I" piping systems were originally designed to the requirements of USAS B31.1. The sample of high fatigue usage factor locations for the Oyster Creek FatiguePro® FatiguePro® program were determined as follows:

a. By screening all existing ASME Section III reactor vessel and ASME Section III Class 1 piping fatigue analyses to identify all locations with a design basis 40-year CUF greater than 0.4. All locations that exceed this CUF have been incorporated into FatiguePro® or dispositioned by monitoring other, more-controlling locations within the same system. Using a fatigue usage criterion of 0.4 to select components to monitor provides margin to the acceptance limit when extrapolating the calculated fatigue usage for 40 years to 60 years **(Reference 4.3.16, paragraph 3.1)**.

b. By screening all usage factor locations of the Mark I Containment Plant Unique Analyses for the torus, torus vents, attached piping, and Torus penetrations with a design basis 40-year CUF greater than 0.4. All appropriate locations were included in the FatiguePro® program or dispositioned by monitoring other, more-controlling locations within the same system **(Reference 4.3.16, paragraph 3.4).**

c. By screening all usage factor locations of the isolation condenser analyses for the isolation condenser and attached piping with a design basis 40-year CUF greater than 0.4. All appropriate locations were included in the FatiguePro® program or dispositioned by monitoring other, more-controlling locations within the same system **(Reference 4.3.16, paragraph 3.5).**

d. Since B31.1 rules were used for the original unmodified piping systems, the governing analyses for these systems did not include an explicit fatigue design basis and no CUFs were calculated. For those "Class I" piping systems that are potentially exposed to significant thermal cycling, fatigue analyses were developed to establish 40-year design basis CUFs, for bounding locations, to establish a valid fatigue design basis so that all "Class I" systems can be monitored, on a CUF basis, in FatiguePro® **(Reference 4.3.16, paragraph 3.3)**.

Plant-specific calculations have been performed for all the locations identified in NUREG/CR-6260 for older-vintage GE BWR plant. The six locations are:

- Reactor Vessel (Lower Head to Shell Transition)
- Feedwater Nozzle
- Recirculation System (SDC Return Line Tee), including the RPV recirculation inlet and outlet nozzles
- Core Spray System (Nozzle and Safe End)
- Isolation Condenser Return to Shutdown Cooling
- Limiting Feedwater Line Location

Exelon has performed detailed environmental fatigue calculations using the appropriate environmentally affected fatigue (F_{en}) relationships from NUREG/CR-6583 for carbon and low-alloy steels, and from NUREG/CR-5704 for stainless steels, as appropriate for the material at each location. The calculations determined either a conservative bounding F_{en} multiplier for the total CUF for each component, or an appropriate F_{en} multiplier for each individual load pair in the governing fatigue calculation for each component, so that an overall CUF multiplier for environmental effects was determined for each location. Table 5.5 provides the results of these environmental fatigue calculations.

Table 5.4 lists the bounding locations of those components that will be monitored by FatiguePro®, including those locations where environmental effects have been analyzed. The basis for the usage factors listed in the table is documented in the calculations listed in Section 4.3 of this document **(Reference 4.3.8)**.

<u>Exceptions to NUREG-1801, Element 5:</u>

None.

<u>Enhancements to NUREG-1801, Element 5:</u>

The program will be enhanced to use the EPRI-licensed FatiguePro® cycle counting and fatigue usage factor tracking computer program. The computer program provides for calculation of stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles and automated calculation and tracking of fatigue cumulative usage factors.

The program will provide for calculating and tracking of the cumulative usage factors for bounding locations for the reactor pressure vessel, Class I piping, the torus, torus vents, torus attached piping and penetrations, and the isolation condenser. The monitoring sample will include those locations where the predicted 40-year cumulative fatigue usage had been predicted to be 0.4 or greater, including the locations specified in NUREG/CR-6260, when applicable to Oyster Creek.

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

This element is consistent with NUREG-1801, Element 5, Monitoring and Trending with the enhancements described above.

3.5 Acceptance Criteria

<u>NUREG-1801:</u>

The acceptance criteria involves maintaining the fatigue usage below the design code limit considering environmental fatigue effects as described under the program description.

<u>Oyster Creek:</u>

The Oyster Creek MFRCPB aging management program continuously monitors plant operational events, calculates fatigue usage factors for all monitored locations and are compared to allowable values. The allowable fatigue usage values include the environmental effects of the reactor coolant for the locations described in element 5 above and are listed in Table 5.5. The responsible plant engineer collects the data approximately monthly. The data is reviewed to identify the need for any corrective actions. The acceptance criterion consists of maintaining the fatigue usage below the appropriate design Code allowable limit. This acceptance criterion will ensure that all original structural margins considered in the plant design are maintained throughout the operating period, and will thereby prevent loss of the intended function **(Reference: ER-AA-470, paragraph 4.4)**

Exceptions to NUREG-1801, Element 6:

None.

Enhancements to NUREG-1801, Element 6:

The program will be enhanced to use the EPRI-licensed FatiguePro® cycle counting and fatigue usage factor tracking computer program. The computer program provides for calculation of stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles and automated calculation and tracking of fatigue cumulative usage factors.

The program will provide for calculating and tracking of the cumulative usage factors for bounding locations for the reactor pressure vessel, Class I piping, the torus, torus vents, torus attached piping and penetrations, and the isolation condenser. The monitoring sample will include those locations where the predicted 40-year cumulative fatigue usage had been predicted to be 0.4 or greater, including the locations specified in NUREG/CR-6260, when applicable to Oyster Creek. **(Reference: ER-AA-470, paragraphs Section 4.4, 4.6)**

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 6, Acceptance Criteria, with the enhancements described above.

3.6 Corrective Actions

NUREG-1801:

a) *The program provides for corrective actions to prevent the usage factor from exceeding the design code limit during the* *period of extended operation.*

b) *Acceptable corrective actions include repair of the component, replacement of the component, and a more rigorous analysis of the component to demonstrate that the design code limit will not be exceeded during the extended period of operation.*

c) *For programs that monitor a sample of high fatigue usage locations, corrective actions include a review of additional affected reactor coolant pressure boundary locations. 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.*

<u>Oyster Creek:</u>

a) The Oyster Creek MFRCPB program will provide for evaluation of CUF or cycle count tracking results that do not satisfy acceptance criteria, and will define actions to be taken. If the current fatigue usage for any location increases dramatically or if the number of cycles approaches a limit, then engineering is notified and an engineering evaluation is performed.

The corrective action program will govern events, values, or results for which the FatiguePro® program does not define actions. Evaluations are performed for test or inspection results that do not satisfy established criteria and a issue report is initiated to document the concern in accordance with plant administrative procedures. **(Reference ER-AA-470, paragraph 4.4).**

b) Possible corrective actions include repair of the component, replacement of the component, and performing a more refined fatigue analysis. Acceptance criteria and actions, and required schedules for completion of actions, will be designed to ensure that CUF limits will not be exceeded **(Reference ER-AA-470, paragraph 4.4).**

c) An engineering evaluation will be made of results that do not satisfy acceptance criteria. The engineering evaluation may result in additional locations being added to the sample of high fatigue usage locations monitored by the program to ensure that CUF limits will not be exceeded in any location. The 10 CFR Part 50, Appendix B corrective action program 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 repetition **(Reference ER-AA-470, paragraph 4.4).**

Exceptions to NUREG-1801, Element 7:

None.

Enhancements to NUREG-1801, Element 7:

The program will be enhanced to use the EPRI-licensed FatiguePro® cycle counting and fatigue usage factor tracking computer program. The computer program provides for calculation of stress cycles and fatigue usage factors from operating cycles, automated counting of fatigue stress cycles and automated calculation and tracking of fatigue cumulative usage factors.

The program will provide for calculating and tracking of the cumulative usage factors for bounding locations for the reactor pressure vessel, Class I piping, the torus, torus vents, torus attached piping and penetrations, and the isolation condenser. The monitoring sample will include those locations where the predicted 40-year cumulative fatigue usage had been predicted to be 0.4 or greater, including the locations specified in NUREG/CR-6260, when applicable to Oyster Creek.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 7, Corrective Actions with the enhancements described above.

3.7 Confirmation Process

NUREG-1801:

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

3.8 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.

Operating Experience

NUREG-1801:

The program reviews industry experience regarding fatigue cracking. Applicable experience with fatigue cracking is to be considered in selecting the monitored locations.

Oyster Creek:

In September of 1997 the Nine Mile Point Unit 1 Nuclear Power Plant experienced through wall tube failures in all Emergency Condenser tube bundles. Subsequent failure investigation and root cause analysis concluded that thermal fatigue and corrosion were the failure mechanisms, resulting of partially exposing the tubes to a steam and condensate environment during standby mode for long period periods of time. Because the Oyster creek and NMP Isolation condenser were designed essentially the same, Oyster Creek personnel took action to investigate for similar problems at Oyster Creek. Personnel at Oyster Creek suspected a similar problem might exist because high condenser shell side water temperatures were observed. Inspection and testing of the condenser also found tube failures in the Oyster Creek Isolation Condenser. The condenser tubes were replaced in 1998 and 2000 and isolation valve leakage corrected. Since these repairs were made, high temperatures or signs of tube leakage have not been observed. Because of the operating experience with Isolation Condenser fatigue at both NMP-1 and Oyster Creek the MFRCPB program will monitor locations on the Oyster Creek Isolation Condenser for fatigue usage, despite predicted fatigue for the isolation condenser are less than the 0.4 screening criterion. This provides objective evidence that Oyster Creek Oyster Creek reviews industry experience with regard to fatigue issues, considers industry experience when selecting locations for monitoring fatigue, and takes corrective action before the loss of intended function. **(Reference: CAP O1997-0787; OC-MD-H037-001).**

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.), General Electric documents (e.g., RICSILs, 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 demonstrates that fatigue to thermal cycles is being adequately managed in the RCPB and other components. The following examples of operating experience provide additional objective evidence that the Metal Fatigue of Reactor Coolant Pressure Boundary (MFRCPB) aging management program is effective in assuring that intended function(s) will be maintained consistent with the CLB for the period of extended operation.

In 1998, after the initiation of the current Thermal Cycle Tracking program, it became apparent that the number of cycles for some plant transients had exceeded the number assumed in the original RPV fatigue analysis. Oyster Creek took action in 1998 to re-assess the design cycle limits for several transients to demonstrate that the fatigue usage for the reactor vessel components remained within the fatigue usage allowable established by the original vessel specification. This example provides objective evidence that the Oyster MFRCPB aging management program review plant operating data and takes corrective action to manage the effects of metal fatigue of the reactor coolant pressure boundary.
(Reference: C-1302-221-5310-018; GPUN Deviation report 94-005)

In 2000 during a plant shutdown to cold conditions the cooldown rate inadvertently increased to 111F/hr, which exceeded the rate of 100/F/hr allowed in the plant's Technical Specifications and the cooldown rate assumed in the reactor pressure vessel fatigue analysis for a normal cooldown. A CAP and MNCR were generated to evaluate the impact on cumulative fatigue. An engineering evaluation was performed, which determined that cooldown event was bounded by the 300F/hr thermal cycle design basis assumed in the reactor vessel fatigue analysis. One of the action items from the CAP was to update the Transient/Thermal Cycle Log to reflect a change to the number of cooldown cycles design basis thermal cycles experienced. This example provides objective evidence that the MFRCPB monitors plant events and records significant thermal transients. This action ensures fatigue is adequately monitored and allows corrective action to be taken before the loss of intended function. **(Reference: CAP O2000-1915, MNCR O2000-1915).**

The operating experience of the MFRCPB has not demonstrated 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 MFRCPB program will effectively determine the fatigue usage due to thermal cycling for bounding locations before fatigue usage approaches acceptance limits. Appropriate guidance for reevaluation, repair or replacement is provided in case the predicted fatigue usage indicates a location will approach the fatigue acceptance limit. The correction action program will ensure the quality performance of the program is maintained.

3.9 Conclusion

The Oyster Creek Metal Fatigue of Reactor Coolant Pressure Boundary aging management program is credited for managing the aging effect of fatigue due to thermal cycling for the reactor pressure vessel, reactor coolant pressure boundary and other components. The Oyster Creek Metal Fatigue of Reactor Coolant Pressure Boundary program elements have been evaluated against NUREG-1801 in Section 3.0. No program exceptions have been identified as stated in Section 2.3. Program enhancements have been identified in Section 2.4. The implementing procedures 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 Metal Fatigue of Reactor Coolant Pressure Boundary aging management program provides reasonable assurance that metal fatigue in the reactor coolant pressure boundary due to thermal cycling 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 NUREG/CR-5704 (ANL-98/31), "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels." April 1999.

4.2.2 NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components." March 1995.

4.2.3 N85

4.2.4 UREG/CR-6583 (ANL-97/18), "Effects of LWR Coolant Environments on Fatigue Design Curves of Carbon and Low-Alloy Steels." March 1998Oyster Creek Program References

4.3 Fatigue Analysis Reports

4.3.1 SIA Report, SI Calculation No. OC-05Q-303, "Cycle-based Fatigue Development for RPV, Torus, and Isolation Condenser Locations", 6/30/2005.

4.3.2 SIA Report, SI Calculation No. OC-05Q-304, "Fatigue Evaluation for Feedwater Piping.", 6/29/2005.

4.3.3 SIA Report, SI Calculation No. OC-05Q-305, "Recirculation

Loop E and Attached Branch Piping Fatigue Analysis", 6/30/2005.

4.3.4 SIA Report, SI Calculation No. OC-05Q-306, "Reactor Water Clean Up (RWCU) Piping Fatigue Analysis", 6/29/2005.

4.3.5 SIA Report, SI Calc. No. OC-05Q-307, "Feedwater Nozzle Green's Functions," 7/20/2005

4.3.6 SIA Report, SI Calc. No. OC-05Q-308 "Development of Feedwater Nozzle Stratification Table", 7/20/2005

4.3.7 SIA Report, SI Calculation No. OC-05Q-311, 40-Year and 60-Year Cycle Projections, 6/24/2005.

4.3.8 SIA Report, SI Calculation No. OC-05Q-314, "Environmental Fatigue Calculations for RPV Locations", 6/30/2005.

4.3.9 SIA Report, SI Calculation No. OC-05Q-315 "Development of Recirculation Inlet Transients from Isolation Condenser Operation", 6/28/2005.

4.3.10 SIA Report, SI Calculation No. OC-05Q-316, "Recirc Outlet Nozzle Finite Element Model", 6/29/2205.

4.3.11 SIA Report, SI Calculation No. OC-05Q-317, "Recirculation Outlet Nozzle Green's Functions", 6/29/2005.

4.3.12 SIA Report, SI Calculation No. OC-05Q-318, " Fatigue Analysis of Recirculation Outlet Nozzle, 6/30/2005"

4.3.13 SIA Report, SI Calculation No. OC-05Q-320, "RPV Basin Seal Skirt Re-Evaluation", 6/28/2005.

4.3.14 SIA Report, SI Report SIR-04-150, Rev 1, "Cycle Counting and Cycle-Based Fatigue Report for the Transient and Fatigue Monitoring System for the Oyster Creek Generating Station," Sept. 2005.

4.3.15 SIA Report, SI Report SIR-04-149, Rev 1, "Transfer Function and System Logic Report for the Transient and Fatigue Monitoring System for the OC Feedwater Nozzles," September 2005.

4.3.16 SIA Report, SI Report SIR04-092, " Report on System Review and Recommendations for a Transient and Fatigue Monitoring System at the Oyster Creek Generating

Station", S Rev 0, September 2004.

4.3.17 Sargent & Lundy LLC Project No. 11324-016, "Oyster Creek Station – Unit 1 Operational Records Review for License Renewal," December 21, 2004.

4.3.18 SIA Report, SI Calculation No. OC-05Q-324, " FatiguePro® Historical Baselining", 10/21/2005.

4.3.19 **TABLES**

4.4 Aging Management Program Implementing Documents

Procedure Number	Procedure Title	Commitment No.	Status
ER-AA-470	Fatigue and Transient Monitoring Program	330592.44.01	ACC/ASG

Aging Management Review Results

The systems and components listed in Table 5.2 are subject to cumulative fatigue. The analyses that predict cumulative fatigue for the components, materials, and environments listed in Table 5.2 are time limited aging analyses (TLAAs) as defined by 10 CFR 54.3(a). As described in paragraph 4.3 of the Oyster Creek LRA the option selected to disposition these TLAAs in accordance with 10 CFR 54.21(c)(1)(iii) is the Metal Fatigue of Reactor Coolant Pressure Boundary aging management program described in this program basis document.

Table 5.2

SSC Name	Structure and/or	Material	Environment	Aging Effect
Component Supports Commodity	Supports for ASME Class	Carbon and low alloy	Indoor Air	Cumulative Fatigue Damage (TLAA)
Containment Spray System	Piping and fittings	Carbon and low alloy	Treated Water <140F	Cumulative Fatigue Damage (TLAA)
Containment Spray System	Piping and fittings	Stainless Steel	Treated Water <140F	Cumulative Fatigue Damage (TLAA)
Containment Vacuum Breakers	Piping and fittings	Stainless Steel	Containment	Cumulative Fatigue Damage (TLAA)
Containment Vacuum Breakers	Valve Body (Vacuum	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Containment Vacuum Breakers	Valve Body	Stainless Steel	Containment	Cumulative Fatigue Damage (TLAA)
Containment Vacuum Breakers	Expansion Joint	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Containment Vacuum Breakers	Piping and fittings	Carbon and low alloy	Indoor Air (Internal)	Cumulative Fatigue Damage (TLAA)
Containment Vacuum Breakers	Piping and fittings	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Containment Vacuum Breakers	Valve Body	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Containment Vacuum Breakers	Expansion Joint	Stainless Steel	Containment	Cumulative Fatigue Damage (TLAA)
Containment Vacuum Breakers	Valve Body	Stainless Steel	Treated Water <140F	Cumulative Fatigue Damage (TLAA)
Containment Vacuum Breakers	Piping and fittings	Stainless Steel	Treated Water <140F	Cumulative Fatigue Damage (TLAA)
Control Rod Drive System	Closure bolting	Alloy Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Control Rod Drive System	Valve Body	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Control Rod Drive System	Closure bolting	Alloy Steel	Containment	Cumulative Fatigue Damage (TLAA)
Control Rod Drive System	Closure bolting	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Control Rod Drive System	Piping and fittings	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Control Rod Drive System	Piping and fittings	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Control Rod Drive System	Closure bolting	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Control Rod Drive System	Valve Body	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Control Rod Drive System	Closure bolting	High Strength Alloy	Containment	Cumulative Fatigue Damage (TLAA)
Core Spray System	Valve Body	CASS	Treated Water	Cumulative Fatigue Damage (TLAA)
Core Spray System	Valve Body	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Core Spray System	Piping and fittings	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Core Spray System	Closure bolting	Alloy Steel	Containment	Cumulative Fatigue Damage (TLAA)
Core Spray System	Closure bolting	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Core Spray System	Closure bolting	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Core Spray System	Piping and fittings	Carbon and low alloy	Treated Water <140F	Cumulative Fatigue Damage (TLAA)
Core Spray System	Piping and fittings	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Core Spray System	Closure bolting	Alloy Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)

Feedwater System	Piping and fittings	Stainless Steel	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Feedwater System	Closure bolting	Carbon and low alloy steel	Containment Atmosphere (External)	Cumulative Fatigue Damage (TLAA)
Feedwater System	Closure bolting	Carbon and low alloy steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Feedwater System	Piping and fittings	Stainless Steel	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Feedwater System	Piping and fittings	Carbon and low alloy steel	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Feedwater System	Piping and fittings	Chrome Moly steels	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Feedwater System	Piping and fittings	Chrome Moly steels	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Feedwater System	Valve Body	Carbon and low alloy steel	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Feedwater System	Valve Body	Stainless Steel	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Feedwater System	Piping and fittings	Chrome Moly steels	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Feedwater System	Piping and fittings	Carbon and low alloy steel	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Feedwater System	Piping and fittings	Carbon and low alloy steel	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Feedwater System	Piping and fittings	Stainless Steel	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Isolation Condenser System	Closure bolting	Stainless Steel	Containment Atmosphere (External)	Cumulative Fatigue Damage (TLAA)
Isolation Condenser System	Piping and fittings	Stainless Steel	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Isolation Condenser System	Piping and fittings	Stainless Steel	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Isolation Condenser System	Piping and fittings	Stainless Steel	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Isolation Condenser System	Valve Body	Stainless Steel	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Isolation Condenser System	Valve Body	CASS	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Isolation Condenser System	Thermowell	Stainless Steel	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Isolation Condenser System	Heat Exchangers (isolation condenser)	Stainless Steel (Tubes and Tube Side	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Isolation Condenser System	Closure bolting	Stainless Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Isolation Condenser System	Closure bolting	Alloy Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Isolation Condenser System	Piping and fittings	Stainless Steel	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Isolation Condenser System	Closure bolting	Carbon and low alloy steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Isolation Condenser System	Piping and fittings	Carbon and low alloy steel	Treated Water <140F (Internal)	Cumulative Fatigue Damage (TLAA)
Isolation Condenser System	Valve Body	Stainless Steel	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Isolation Condenser System	Valve Body	CASS	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Main Steam System	Condensing chamber	Stainless Steel	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Main Steam System	Condensing chamber	Stainless Steel	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Main Steam System	Piping and fittings	Stainless Steel	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Main Steam System	Valve Body	Carbon and low alloy steel	Treated Water (Internal)	Cumulative Fatigue Damage (TLAA)
Main Steam System	Valve Body	Carbon and low alloy steel	Steam (Internal)	Cumulative Fatigue Damage (TLAA)

Main Steam System	Closure bolting	Alloy Steel	Containment	Cumulative Fatigue Damage (TLAA)
Main Steam System	Piping and fittings	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Main Steam System	Piping and fittings	Carbon and low alloy	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Main Steam System	Piping and fittings	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Main Steam System	Piping and fittings	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Main Steam System	Piping and fittings	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Main Steam System	Expansion Joint	Stainless Steel	Containment	Cumulative Fatigue Damage (TLAA)
Main Steam System	Piping and fittings	Carbon and low alloy	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Main Steam System	Closure bolting	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Main Steam System	Closure bolting	Alloy Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Main Steam System	Piping and fittings	Carbon and low alloy	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Main Steam System	Closure bolting	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Noble Metals Monitoring System	Closure bolting	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Noble Metals Monitoring System	Closure bolting	Alloy Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Noble Metals Monitoring System	Piping and fittings	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Nuclear Boiler Instrumentation	Valve Body	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Nuclear Boiler Instrumentation	Closure bolting	Alloy Steel	Containment	Cumulative Fatigue Damage (TLAA)
Nuclear Boiler Instrumentation	Closure bolting	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Nuclear Boiler Instrumentation	Closure bolting	Stainless Steel	Containment	Cumulative Fatigue Damage (TLAA)
Nuclear Boiler Instrumentation	Piping and fittings	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Nuclear Boiler Instrumentation	Piping and fittings	Stainless Steel	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Nuclear Boiler Instrumentation	Closure bolting	Stainless Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Nuclear Boiler Instrumentation	Closure bolting	Alloy Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Nuclear Boiler Instrumentation	Condensing chamber	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Nuclear Boiler Instrumentation	Closure bolting	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Post-Accident Sampling System	Piping and fittings	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Post-Accident Sampling System	Closure bolting	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Post-Accident Sampling System	Closure bolting	Alloy Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Post-Accident Sampling System	Closure bolting	Stainless Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Post-Accident Sampling System	Closure bolting	Stainless Steel	Containment	Cumulative Fatigue Damage (TLAA)
Post-Accident Sampling System	Piping and fittings	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Post-Accident Sampling System	Valve Body	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Post-Accident Sampling System	Valve Body	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)

Primary Containment	Suppression Chamber	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Primary Containment	Downcomers	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Primary Containment	Suppression Chamber	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Primary Containment	Suppression Chamber	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Primary Containment	Suppression Chamber	Carbon and low alloy	Treated Water <140F	Cumulative Fatigue Damage (TLAA)
Primary Containment	Suppression Chamber	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Primary Containment	Suppression Chamber	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Primary Containment	Downcomers	Carbon and low alloy	Treated Water < 140F	Cumulative Fatigue Damage (TLAA)
Primary Containment	Vent line bellows	Stainless Steel	Containment	Cumulative Fatigue Damage (TLAA)
Primary Containment	Vent line bellows	Stainless Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Primary Containment	Drywell Penetration	Stainless Steel,	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Primary Containment	Drywell Penetration	Stainless Steel,	Containment	Cumulative Fatigue Damage (TLAA)
Primary Containment	Suppression Chamber	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Primary Containment	Suppression Chamber	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Primary Containment	Suppression Chamber	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Primary Containment	Suppression Chamber	Carbon and low alloy	Treated Water <140F	Cumulative Fatigue Damage (TLAA)
Primary Containment	Suppression Chamber	Carbon and low alloy	Treated Water <140F	Cumulative Fatigue Damage (TLAA)
Primary Containment	Vent line, and Vent Header	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Primary Containment	Suppression Chamber	Carbon and low alloy	Treated Water <140F	Cumulative Fatigue Damage (TLAA)
Primary Containment	Vent line, and Vent Header	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Reactor Head Cooling System	Piping and fittings	Stainless Steel	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Reactor Head Cooling System	Piping and fittings	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Head Cooling System	Closure bolting	Alloy Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Reactor Head Cooling System	Restricting Orifice	Stainless Steel	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Reactor Head Cooling System	Valve Body	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Head Cooling System	Valve Body	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Head Cooling System	Valve Body	CASS	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Head Cooling System	Closure bolting	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Reactor Head Cooling System	Closure bolting	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Reactor Head Cooling System	Closure bolting	Alloy Steel	Containment	Cumulative Fatigue Damage (TLAA)
Reactor Head Cooling System	Restricting Orifice	Stainless Steel	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Nozzle Safe Ends	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Nozzles (Main Steam)	Carbon and low alloy	Steam (Internal)	Cumulative Fatigue Damage (TLAA)

Reactor Pressure Vessel	Penetrations (CRD Stub	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Nozzles (CRD Return)	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Top Head Closure Studs	High Strength Alloy	Containment	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Top Head Enclosure	Carbon and low alloy	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Nozzle Safe Ends (Core	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Nozzle (Bottom head	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Nozzles (Core Spray)	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Vessel Bottom Head	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Nozzles (Isolation	Carbon and low alloy	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Penetrations (CRD Stub	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Nozzles (Recirculation	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Penetrations (Standby	Nickel Alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Penetrations	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Vessel Shell (Upper, upper	Carbon and low alloy	Treated Water >482F	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Vessel Shell Flange	Carbon and low alloy	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Top Head Flange	Carbon and low alloy	Steam (Internal)	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Penetrations	Nickel Alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Nozzle Safe Ends	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	Nozzles (Feedwater)	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Pressure Vessel	RPV Support Skirt and	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Reactor Recirculation System	Closure bolting	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Reactor Recirculation System	Thermowell	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Recirculation System	Closure bolting	Alloy Steel	Containment	Cumulative Fatigue Damage (TLAA)
Reactor Recirculation System	Flow Element	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Recirculation System	Piping and fittings	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Recirculation System	Pump Casing	CASS	Treated Water >482F	Cumulative Fatigue Damage (TLAA)
Reactor Recirculation System	Valve Body	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Recirculation System	Valve Body	CASS	Treated Water >482F	Cumulative Fatigue Damage (TLAA)
Reactor Water Cleanup System	Valve Body	CASS	Treated Water >482F	Cumulative Fatigue Damage (TLAA)
Reactor Water Cleanup System	Piping and fittings	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Water Cleanup System	Closure bolting	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Reactor Water Cleanup System	Closure bolting	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Reactor Water Cleanup System	Closure bolting	Alloy Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)

Reactor Water Cleanup System	Closure bolting	Alloy Steel	Containment	Cumulative Fatigue Damage (TLAA)
Reactor Water Cleanup System	Valve Body	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Water Cleanup System	Piping and fittings	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Water Cleanup System	Closure bolting	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Reactor Water Cleanup System	Closure bolting	Alloy Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Reactor Water Cleanup System	Piping and fittings	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Reactor Water Cleanup System	Piping and fittings	Stainless Steel	Treated Water <140F	Cumulative Fatigue Damage (TLAA)
Shutdown Cooling System	Piping and fittings	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Shutdown Cooling System	Piping and fittings	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Shutdown Cooling System	Piping and fittings	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Shutdown Cooling System	Valve Body	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Shutdown Cooling System	Piping and fittings	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Shutdown Cooling System	Closure bolting	Stainless Steel	Containment	Cumulative Fatigue Damage (TLAA)
Shutdown Cooling System	Closure bolting	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Shutdown Cooling System	Closure bolting	Alloy Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Shutdown Cooling System	Flow Element	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Shutdown Cooling System	Flow Element	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Shutdown Cooling System	Valve Body	Carbon and low alloy	Treated Water	Cumulative Fatigue Damage (TLAA)
Standby Liquid Control System	Closure bolting	Alloy Steel	Containment	Cumulative Fatigue Damage (TLAA)
Standby Liquid Control System	Closure bolting	Carbon and low alloy	Containment	Cumulative Fatigue Damage (TLAA)
Standby Liquid Control System	Closure bolting	Alloy Steel	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Standby Liquid Control System	Closure bolting	Carbon and low alloy	Indoor Air (External)	Cumulative Fatigue Damage (TLAA)
Standby Liquid Control System	Valve Body	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)
Standby Liquid Control System	Flow Element	Stainless Steel	Treated Water <140F	Cumulative Fatigue Damage (TLAA)
Standby Liquid Control System	Piping and fittings	Stainless Steel	Treated Water	Cumulative Fatigue Damage (TLAA)

List of Transients counted in FatiguePro®

Transient	Included in Table 5.2-2 of UFSAR?	Cycles as of 6/30/05	60-Year Projected Number of Cycles	Design Analyzed Number of Cycles
Vessel Head Removal and Reinstallation (Bolt up/Unbolt)		24	36	80
Design Pressure Test (Leak Test at Operating Pressure)		24	37	130
Heatup = Normal Startup (100°F/hr)	Y	217	272	240
Turbine Roll and Increase to Rated Power		185	246	240
Cooldown = Normal Shutdown (100°F/hr)	Y	217	272	240
Hot Standby (Feedwater Cycling)	Y (1)	178	226	400
300°F/hr Emergency Cooldown		0	1	5
Safety Relief Valve (EMRV) Blowdown		0	1	1
SCRAM	Y (2)	140	155	200
Turbine Trip		38	43	40
Loss of Feedwater Heaters		4	9	80
Interruption of Feedwater Flow		1	2	80
Overpressure to 1,250 psig		0	1	1
Overpressure to 1,375 psig		0	1	1
Hydrostatic Pressure Test (Code Hydro Test to 1,563 psig)		1	1	3
Core Spray Injections	Y	2	3	10
EMRV 'A' Actuation		124	167	450
EMRV 'B' Actuation		68	115	450
EMRV 'C' Actuation		81	120	450
EMRV 'D' Actuation		131	188	450
EMRV 'E' Actuation		70	108	450
Shutdown Cooling 'A' Operation		21	20	Note 3
Shutdown Cooling 'B' Operation		22	21	Note 3
Shutdown Cooling 'C' Operation		182	248	Note 3,4
Emergency Condenser 'A' Actuation		350	482	1500
Emergency Condenser 'B' Actuation		388	520	1500
Unisolation of an Isolated Recirculation Loop		12	50	6,500

Notes:

1. **Includes "Temperature Change 240°F" event.**
2. **Includes "Loss of Drive Coolant" event.**
3. **Not specified in original design of system.**
4. **Loop C is by procedure used first for the initial operation of (Shutdown Cooling) SDC during a shutdown. The initial operation of SDC experiences the greatest temperature change and therefore is the bounding fatigue event for SDC. Subsequent operation of the other loops produces insignificant temperature changes. The remaining number operations of SDC through the period of extended operation are conservatively assumed to only occur with loop C.**

4.5 List of Locations Monitored

LOCATION MONITORED	CUF (Note 1)	MONITORING BASIS
Reactor Pressure Vessel Locations		
Recirculation Outlet Nozzle	0.978	CBF
Recirculation Inlet Nozzle	0.1554	CBF
Feedwater Nozzle (Nozzle Forging)	0.8433	SBF
Core Spray Nozzle (Nozzle Forging)	0.0129	CBF

Core Spray Nozzle Safe End	0.0072	CBF
Support Skirt (Transition Taper Top)	0.710	CBF
Basin Seal Skirt to Vessel Flange Junction	0.270	CBF
Closure Region Bolts	0.196	CBF
Bottom Head (Vessel-Head Junction)	0.0042	CBF
Oyster Creek "Class I" Piping Limiting Locations		
Feedwater line	0.178	CBF
RWCU line	0.1918	CBF
Recirculation line	0.493	CBF
Isolation Condenser Locations		
Tubing	0.625	CBF
Tube sheet	0.595	CBF
Nozzle Junction	0.581	CBF
Tube-to-Tube sheet weld	0.714	CBF
Primary Containment Locations		
Downcomer/Vent Header Intersection	0.896	CBF
Vent Line/Drywell Intersection	0.524	CBF
Torus Shell	0.706	CBF
Torus Shell at SRV-Supporting Ring Girder	0.683	CBF
EMRV-Supporting Ring Girders	0.683	CBF
EMRV Piping Penetration On Vent Pipe	0.492	CBF
Vent Header Ring Collar	0.510	CBF
Nozzle: Drywell to Torus Vacuum Relief (Torus End)	0.90	CBF
Attached Piping: Vacuum Relief	0.434	CBF
Penetrations: Isolation Condenser	0.743	CBF

Note 1: CUF values shown are calculations for 60 years. The values for the NUREG/CR-6260 locations include environmental effects as noted in the table above.

4.6 Environmental Fatigue Results for Oyster Creek NUREG/CR-6260 Components (Taken from LRA Section 4)

NUREG/CR-6260 Location	Equivalent OCGS Location	Material	60-Year Fatigue Usage Factor (1)	60-Year Fatigue Usage Factor with Environmental Effects(2)
Reactor Vessel (Lower Head to Shell Transition)	Reactor Vessel (Vessel-Head Junction)	Low Alloy Steel	0.0004	0.0042
Feedwater Nozzle	Feedwater Nozzle	Low Alloy Steel	0.3889	0.8433
Recirculation System (RHR Return Line Tee and the RPV inlet and outlet nozzles)	Isolation Condenser Return Line Tee into SDC Line RPV inlet nozzle RPV outlet nozzle	Stainless Steel Low Alloy Steel Low Alloy Steel	0.1205 0.0151 0.1832	0.43 0.1554 0.978
Core Spray System (Nozzle and Safe End)	Core Spray Nozzle Core Spray Nozzle Safe End	Low Alloy Steel Stainless Steel	0.0013 0.0006	0.0129 0.0072

Residual Heat Removal Line (Tapered Transition)	Bounded by Isolation Condenser Return Line Tee Location Above	Stainless Steel	N/A	N/A
Feedwater Line (Feedwater/RCIC Tee Connection)	Limiting Feedwater Line Location	Carbon Steel	0.0245	0.0767

Notes:

1. Revised fatigue usage factors were computed for all of the NUREG/CR-6260 components based on projected cycles for 60 years of plant operation and updated ASME Code fatigue methodology.
2. Environmental fatigue usage was computed using the methodology of NUREG/CR-6583 (for carbon/low alloy steels) and NUREG/CR-5704 (for stainless steels), as appropriate for the material for each location.

ATTACHMENTS

4.7 LRA Appendix A

4.8 LRA Appendix B

PROGRAM BASIS DOCUMENT

PBD-AMP-B.1.36

Revision 0

INACCESSIBLE MEDIUM-VOLTAGE CABLES NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

GALL PROGRAM XI.E3 - INACCESSIBLE MEDIUM-VOLTAGE CABLES NOT SUBJECT TO 10 CFR 50.49 ENVIRONMENTAL QUALIFICATION REQUIREMENTS

Prepared By: ______________________

Reviewed By: ______________________

Program Owner Review: ______________________

Technical Lead Approval: ______________________

Revision History:

Revision	*Prepared by:*	*Reviewed by:*	*Program Owner:*	*Approved by:*
0	Deb Spamer	Kevin Muggleston	Raj Pruthi	*Fred Polaski*
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 Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program that are credited for managing cable insulation degradation, due to circuit energization greater than 25% of the time while in wetted conditions, 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; and
- 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 also provides a comparison of the credited Oyster Creek program with the elements of the corresponding NUREG-1801 Chapter XI program XI.E3, Inaccessible Medium Voltage Cables Not Subject to Environmental Qualification Requirements. Project Level Instruction PLI-8 "Program Basis Documents" prescribes the methodology for evaluating Aging Management Programs. An evaluation of Oyster Creek's 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

<u>NUREG-1801:</u>

Most electrical cables in nuclear power plants are located in dry environments. However, some cables may be exposed to condensation and wetting in inaccessible locations, such as conduits, cable trenches, cable troughs, duct banks, underground vaults or direct buried installations. When an energized medium-voltage cable (2 kV to 35 kV) is exposed to wet conditions for which it is not designed, water treeing or a decrease in the dielectric strength of the conductor insulation can occur. This can potentially lead to electrical failure.

a) *The purpose of the aging management program described herein is to provide reasonable assurance that the intended functions of inaccessible medium-voltage cables that are not subject to the environmental qualification requirements of 10 CFR 50.49 and are exposed to adverse localized environments caused by moisture while energized will be maintained consistent with the current licensing basis through the period of extended operation. An adverse localized environment is a condition in a limited plant area that is* *significantly more severe than the specified service environment for the cable. An adverse variation in environment is significant if it could appreciably increase the rate of aging of a component or have an immediate adverse effect on operability. This program considers the technical information and guidance provided in NUREG/CR-5643, IEEE Std. P1205, SAND96-0344, and EPRI TR-109619.*

b) *In this aging management program periodic actions are taken to prevent cables from being exposed to significant moisture, such as inspecting for water collection in cable manholes, and draining water, as needed. The above actions are not sufficient to assure that water is not trapped elsewhere in the raceways. For example, if duct bank conduit has low points in the routing, there could be potential for long-term submergence at these low points. In addition, concrete raceways may crack due to soil settling over a long period of time and manhole covers may not be watertight. Additionally, in certain areas, the water table is high in seasonal cycles and therefore, the raceways may get refilled soon after purging. Furthermore, potential uncertainties exist with water trees even when duct banks are sloped with the intention to minimize water accumulation. Experience has shown that insulation degradation may occur if the cables are exposed to 100 percent relative humidity. The above periodic actions are necessary to minimize the potential for insulation degradation. In addition to above periodic actions, in-scope, medium-voltage cables exposed to significant moisture and significant voltage are tested to provide an indication of the condition of the conductor insulation. The specific type of test performed will be determined prior to the initial test, and is to be a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2, or other testing that is state-of-the-art at the time the test is performed.*

c) *As stated in NUREG/CR-5643, "The major concern with cables is the performance of aged cable when it is exposed to accident conditions." The statement of considerations for the final license renewal rule (60 Fed. Reg. 22477) states, "The major concern is that failures of deteriorated cable systems (cables, connections, and penetrations) might be induced during accident conditions." Since they are not subject to the environmental qualification requirements of 10 CFR 50.49, the* *electrical cables covered by this aging management program are either not exposed to harsh accident conditions or are not required to remain functional during or following an accident to which they are exposed.*

<u>Oyster Creek:</u>

The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program manages inaccessible medium-voltage cables that are exposed to significant moisture simultaneously with significant voltage.

Significant moisture is defined as periodic exposures to moisture that lasts more than a few days (e.g., cable in standing water). Periodic exposures to moisture that last less than a few days (i.e., normal rain and drain) are not significant. Significant voltage exposure is defined as being subjected to system voltage (2.4 kV, 4.16kV, 13.8 kV or 34.5 kV) for more than twenty-five percent of the time.

a) The purpose of this aging management program is to provide reasonable assurance that intended functions of inaccessible medium voltage cables that are exposed to localized adverse environments caused by moisture while energized, will be maintained consistent with the current licensing basis, through the period of extended operation. Oyster Creek has 2.4 kV, 4.16 kV, 13.8 kV and 34.5 kV medium voltage cable installations. Because of Oyster Creek's history of medium voltage cable failures (see Section 3.10), 2.4 kV, 4.16 kV, 13.8 kV and 34.5 kV system circuits are conservatively assumed to have the potential to be exposed to significant moisture conditions. Further, the 2.4 kV, 4.16 kV, 13.8 kV and 34.5 kV system circuits are conservatively assumed to be energized more than 25% of the time. Consequently, 2.4 kV, 4.16 kV, 13.8 kV and 34.5 kV system circuits are included in the scope of the Inaccessible Medium Voltage Cables Not Subject To 10CFR 50.49 Environmental Qualification Requirements Program. Inclusion of the 13.8 kV system circuits in this program reflects the scope expansion of the Station Blackout System electrical commodities. Inclusion of the 34.5 kV system circuits in this program reflects a change in scope for reconciliation of this new aging management program from the draft January 2005 GALL to the approved September 2005 GALL. In addition, these medium-voltage cable circuits will be tested using a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2, or other testing that is state-of-the-art at the time the test is performed. Cable testing will be performed at least once every 10 years and the frequency of testing will be adjusted depending on the results obtained. The first tests will be completed prior to the period of the extend operation.

b) In this aging management program periodic actions are taken to prevent cables from being exposed to significant moisture. This program will inspect manholes, conduits and sumps associated with the 2.4 kV, 4.16 kV, 13.8 kV and 34.5 kV system circuits for water collection so that draining or other corrective actions can be taken. Inspections for water collection will be performed at least once every 2 years and the frequency of testing will be adjusted based on the results obtained. The first inspections will be completed prior to the period of extended operation.

c) This new aging management program (AMP) has been conservatively applied to Oyster Creek non-EQ and EQ 2.4 kV, 4.16 kV, 13.8 kV and 34.5 kV inaccessible cables within the scope of license renewal. By definition, non-EQ cables are either not exposed to harsh accident conditions or are not required to remain functional during or following an accident to which they are exposed. However, because EQ and non-EQ cables are both included in this aging management program, the scope of this program provides reasonable assurance that degraded cables will be identified and remedied such that cable failures will not be induced during accident conditions.

2.2 Overall NUREG-1801 Consistency

The Oyster Creek Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements is a new program that is consistent with NUREG-1801 aging management program XI.E1, Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements.

2.3 Summary of Exceptions to NUREG-1801

None. The new Oyster Creek Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements 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 new Oyster Creek Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements is found to be adequate to support the extended period of operation with no enhancements.

3.0 EVALUATIONS AND TECHNICAL BASIS

<u>**Note**</u>

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.

3.0 Scope of Program

<u>NUREG-1801:</u>

This program applies to inaccessible (e.g., in conduit or direct buried) medium-voltage cables within the scope of license renewal that are exposed to significant moisture simultaneously with significant voltage. Significant moisture is defined as periodic exposures to moisture that last more than a few days (e.g., cable in standing water). Periodic exposures to moisture that last less than a few days (i.e., normal rain and drain) are not significant. Significant voltage exposure is defined as being subjected to system voltage for more than twenty-five percent of the time. The moisture and voltage exposures described as significant in these definitions, which are based on operating experience and engineering judgment, are not significant for medium-voltage cables that are designed for these conditions (e.g., continuous wetting and continuous energization is not significant for submarine *cables).*

Oyster Creek:

The Inaccessible Medium-Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program manages inaccessible medium-voltage cables that are exposed to significant moisture simultaneously with significant voltage. Significant moisture is defined as periodic exposures to moisture that lasts more than a few days (e.g., cable in standing water). Periodic exposures to moisture that last less than a few days (i.e., normal rain and drain) are not significant. Significant voltage exposure is defined as being subjected to system voltage for more than twenty-five percent of the time.

Oyster Creek has 2.4 kV, 4.16 kV, 13.8 kV and 34.5 kV medium voltage cable installations. Because of Oyster Creek's history of medium voltage cable failures, 2.4 kV, 4.16 kV, 13.8 kV and 34.5 kV system circuits are conservatively assumed to have the potential to be exposed to significant moisture conditions. Further, the 2.4 kV, 4.16 kV, 13.8 kV and 34.5 kV system circuits are conservatively assumed to be energized more than 25% of the time. Consequently, 2.4 kV, 4.16 kV, 13.8 kV and 34.5 kV system circuits are included in the scope of the Inaccessible Medium Voltage Cables Not Subject To 10CFR 50.49 Environmental Qualification Requirements Program. Inclusion of the 13.8 kV system circuits in this program reflects the scope expansion of the Station Blackout System electrical commodities. Inclusion of the 34.5 kV system circuits in this program reflects a change in scope for reconciliation of this new aging management program from the draft January 2005 GALL to the approved September 2005 GALL.

The Oyster Creek Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program manages cable insulation degradation, due to circuit energization greater than 25% of the time while in wetted conditions, for the commodity groups and environments listed in Table 5.2. The future 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 new implementing documents are being created 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.

3.1 Preventive Actions

NUREG-1801:

Periodic actions are taken to prevent cables from being exposed to significant moisture, such as inspecting for water collection in cable manholes, and draining water, as needed.

Oyster Creek:

Inspections for water collection in the manholes, conduits and sumps containing medium voltage cables within the scope of this program will be performed. These inspections will be performed at least every two years with the frequency of inspection being adjusted based on the inspection results obtained. Inspections will begin at the GALL recommended two-year frequency based on Oyster Creek's operating experience (see Section 3.10) that does not indicate a trend or recurrence of water accumulation or cable submergence in manholes, vaults or sumps. As part of these inspections, water will be drained and other corrective actions taken, as appropriate.

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.

3.2 Parameters Monitored or Inspected

NUREG-1801:

In-scope, medium-voltage cables exposed to significant moisture and significant voltage are tested to provide an indication of the condition of the conductor insulation. The specific type of test performed will be determined prior to the initial test, and is to be a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2, or other testing that is state-of-the-art at the time the test is performed.

Oyster Creek:

This program will test Oyster Creek's in-scope medium voltage cables to provide an indication of the condition of the conductor insulation. The specific type of test performed will be determined prior to the initial test, and will be a proven test for detecting deterioration of the insulation system due to wetting, such as power factor, partial discharge, or polarization index, as described in EPRI TR-103834-P1-2, or other testing that is state-of-the-art at the time the test is performed. Testing will be performed every 10 years and the frequency of testing will be adjusted depending on the results obtained.

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.

3.3 Detection of Aging Effects

NUREG-1801:

Medium-voltage cables exposed to significant moisture and significant voltage that are within the scope of this program are tested at least once every 10 years. This is an adequate period to preclude failures of the conductor insulation since experience has shown that aging degradation is a slow process. A 10 year testing interval will provide two data points during a 20-year period, which can be used to characterize the degradation rate. The first tests for license renewal are to be completed before the period of extended operation.

The inspection for water collection should be performed based on actual plant experience with water accumulation in the manhole. However, the inspection frequency should be at least once every two years. The first inspection for license renewal is to be completed before the period of extended operation.

Oyster Creek:

The Oyster Creek inaccessible medium-voltage cables assumed to be exposed to significant moisture and significant voltage will be tested at least once every 10 years. The first tests for license renewal will be completed prior to the period of extended operation.

Inspections for water collection in the manholes, conduits and sumps associated with the medium voltage cable circuits within the scope of this program will be performed at least every 2 years and the frequency of testing will be adjusted based on the results obtained. Water will be drained and other corrective actions taken, as appropriate. The first inspection will be completed prior to the period of extended operation.

Exceptions to NUREG-1801, Element 4:

None.

Enhancements to NUREG-1801, Element 4:

None.

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

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

3.4 Monitoring and Trending

<u>NUREG-1801:</u>

Trending actions are not included as part of this program because the ability to trend results is dependent on the specific type of method chosen. However, results that are trendable provide additional information on the rate of degradation.

<u>Oyster Creek:</u>

Test results that are trendable may be trended to provide additional information on the rate of cable degradation.

<u>Exceptions to NUREG-1801, Element 5:</u>

None.

<u>Enhancements to NUREG-1801, Element 5:</u>

None.

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

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

3.5 Acceptance Criteria

<u>NUREG-1801:</u>

The acceptance criteria for each test is defined by the specific type of test performed and the specific cable tested

<u>Oyster Creek:</u>

The acceptance criteria for the testing of the cables within the scope of this program will be defined by the specific type of test chosen. Acceptance criteria will be incorporated into controlled station procedures.

The acceptance criteria for the manhole, conduit and sump inspection for the cable circuits within the scope of this program will be defined such that cables within the scope of this program are not in a significant moisture environment (exposures that last more than a few days; e.g., cable(s) in standing water). Acceptance criteria will be incorporated into controlled station procedures or work orders.

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.

3.6 Corrective Actions

NUREG-1801:

An engineering evaluation is performed when the test acceptance criteria are not met in order to ensure that the intended functions of the electrical cables can be maintained consistent with the current licensing basis. Such an evaluation is to consider the significance of the test results, the operability of the component, the reportability of the event, the extent of the concern, the potential root causes for not meeting the test acceptance criteria, the corrective actions required, and the likelihood of recurrence. When an unacceptable condition or situation is identified, a determination is made as to whether the same condition or situation is applicable to other inaccessible, in-scope, medium-voltage cables. 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:

Unacceptable cable test results and unacceptable manhole, conduit and sump inspections will be subject to an engineering evaluation under the corrective action process. The evaluation will consider the significance of the test/inspection results, the operability of the component, the reportability of the event, the extent of the concern, the potential root causes for not meeting the test acceptance criteria, the corrective actions required, and the likelihood of recurrence. Corrective actions such as draining manholes or cable replacement will be implemented when test results do not meet the acceptance criteria. The requirements of 10 CFR Part 50, Appendix B, will be implemented to address corrective actions.

Oyster Creek's corrective action process is governed by 10 CFR 50, Appendix B and is implemented by corporate administrative procedures. The corrective action process generically applies to Oyster Creek activities, even when not specifically invoked by a procedure line item. **(Reference: LS-AA-125)**

<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.

3.7 Confirmation Process

<u>NUREG-1801:</u>

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.

<u>Oyster Creek:</u>

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.

<u>Exceptions to NUREG-1801, Element 8:</u>

None.

Enhancements to NUREG-1801, Element 8:

None.

Comparison and Evaluation Conclusion:

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

3.8 Administrative Controls

NUREG-1801:

As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the 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 9:

None.

Enhancements to NUREG-1801, Element 9

None.

Comparison and Evaluation Conclusion:

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

3.9 Operating Experience

NUREG-1801:

Operating experience has shown that cross linked polyethylene (XLPE) or high molecular weight polyethylene (HMWPE) insulation materials are most susceptible to water tree formation. The *formation and growth of water trees varies directly with operating voltage. Water treeing is much less prevalent in 4kV cables than those operated at 13 or 33kV. Also, minimizing exposure to moisture minimizes the potential for the development of water treeing. As additional operating experience is obtained, lessons learned can be used to adjust the program, as needed.*

Oyster Creek:

As noted in NUREG-1801, industry operating experience has confirmed that insulation degradation can occur in inaccessible medium voltage cables that are exposed to significant moisture and voltage, see EPRI TR-103834-P1-2, Effects of Moisture on the Life of Power Plant Cables and NRC Information Notice 2002-12, Submerged Safety-Related Electrical Cables. A review of plant operating experience at Oyster Creek shows that insulation degradation can occur in inaccessible medium voltage cables that are exposed to significant moisture and voltage and has occurred in several different cable installations. In some cases, the existing Oyster Creek Medium Voltage Cable Test Program (Engineering Evaluation 0164-97, dated May 15, 1997) has identified the occurrence of insulation degradation, prior to cable failure. However, there have been some cases where cable failures did occur that were not predicted. As a result of subsequent failures, the existing test program was modified to more accurately predict the occurrence of insulation degradation and more aggressively replace cables at risk of failure. The experience with the existing Oyster Creek Medium Voltage Cable Test Program shows that Oyster Creek has the experience and knowledge to implement the new Oyster Creek Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program effectively so as to successfully manage cable insulation degradation, due to circuit energization greater than 25% of the time while in wetted conditions.

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.), 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 cable insulation degradation, due to circuit energization greater than 25% of the time while in wetted conditions is being adequately managed in the existing Oyster Creek Medium Voltage Cable Test Program and the new Oyster Creek Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program. The following examples of operating experience provide objective evidence that the new Oyster Creek Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program will be effective in assuring that intended function(s) will be maintained consistent with the CLB for the period of extended operation:

1. Several instances of degradation of inaccessible medium voltage cable in adverse localized environments have been occurred at Oyster Creek.
   a. Generic Oyster Creek History:
      A cable test program was initiated in 1991 due to repetitive failures of original plant cables that were GE Vulkene cables. The GE Vulkene cables contain XLPE (cross linked polyethylene) insulation that was used for all medium voltage applications at Oyster Creek. The historical records show that the Vulkene cable failures were predominantly due to water immersion, or due to surge damage from

lightning. Under the cable test program the applied test voltage values were high enough to identify degraded cables by trending leakage current from a step voltage hi-potential test to allow scheduled replacements. The test voltages were also sufficient to stress the cables at a level that severely degraded cables would fail under test and be replaced as corrective maintenance. The test program succeeded in identifying degraded Vulkene cables (by trending leakage values) and limiting in-service failures of Vulkene cables. Replacement cables at OC have been EPR (ethylene-propylene) compact conductor cables. Initial installations were Anaconda. Second generation replacements were Cablec (improved version of Anaconda). Recent installations are Okonite. Okonite does not possess the cable construction issues (such as drain wires) that were associated with the Anaconda/Cablec cables. Anaconda cables have proven to be unreliable due to poor manufacturing process. The construction was not homogeneous, and there was poor adhesion of the semi-conducting jacket material. The manufacturing process had made it susceptible to water induced failures. Additionally, predictability of failure has been poor for this type (EPR) of cables. All Anaconda cables that are in wetted environment have either been replaced or have been scheduled for replacement.

Oyster Creek's operating experience has been with 4.16 kV cables. Eleven in-service failures in 4.16 kV circuits have occurred. Five failures resulted from water intrusion. Four failures resulted from manufacturing defects. Two failures resulted from a single lightning strike. The majority of those failures occurred in EPR-insulated "UniShield" cables manufactured by Anaconda before 1985. The current medium voltage cable testing program includes 2.4 kV and 4.16 kV cables. Testing has been performed for all of these circuits. Thirty-eight (38) of these circuits have been tested with a newer test methodology, an on-line partial discharge methodology, by DTE Energy. The remaining circuits have been tested via step voltage methodologies. To date, Oyster Creek has replaced approximately 75% of its original 4.16 kV cables.

b. CAP O2003-1000 evaluated a failure of the 4.16 kV C phase cable feeding the 1C emergency bus. This CAP was selected for inclusion in this program basis document because it represents the most recent and comprehensive analysis and subsequent, still applicable, corrective action implementation. Other CAPs on cable failures were not included because the programmatic issues, as described in O2003-1000, represent the current cable testing program at Oyster Creek.

The failed 4.16 kV cable feeding the 1C emergency bus was manufactured by Anaconda. A cause evaluation was performed and specific and generic corrective actions were implemented. The failed cable was replaced. It was identified that this cable had not been given its due priority for replacement because it had been incorrectly identified as a different type of cable than the type(s) of cables that had previously failed at Oyster Creek. Failure analysis was performed on the failed cable. The analysis confirmed the expected failure cause as exposure to a wetted environment aggravated by non-uniform thickness of the insulation shield (manufacturing defect). More extensive 4.16 kV cable configuration (e.g., insulation type, manufacturer, date installed, failure and replacement history) documentation was created. The cable configuration information was used for extent of condition evaluation and cable replacement and testing decisions and prioritization. Extent of condition for cables subject to the same failure was determined. Anaconda cable with the same design as this failed cable were identified for replacement with a cable with improved cable design. A decision tree was developed and implemented for proactive cable replacements to preclude future failures. As stated above, all Anaconda cables, of this vintage, that are in wetted environments have either been replaced or have been scheduled for replacement. Additional condition monitoring testing (beyond the current implementation of DC-hi-pot testing) was pursued and implemented. DTE Energy testing has since been implemented for most 4.16 kV cables.

DTE Energy testing provides an on-line cable assessment and is based upon detection of partial discharge (PD) at the system operating voltage. Partial discharge occurs within imperfections of the cable system resulting from aging, such as thermal, mechanical and electrical stresses. DTE Energy methodology consists of recording partial discharge and other emissions from the deteriorated components of a cable system (cable, splices, terminations) while in service and analyzing them to assess the condition of the cable. The assessment is performed by placing sensors over the cables. The cable remains in service and energized. Data is acquired and recorded, then analyzed to assess the condition of the cable sections and accessories. The methodology works with equal degree of success with the EPR and XLPE type of cable insulation materials. Following evaluation, a decision will be made as to the effectiveness of this testing method and whether a commitment will be made for its long term use. The current inspection program will remain in effect until replaced by the Inaccessible Medium Voltage Cables Not Subject To 10 CFR 50.49 Environmental Qualification Requirements aging management program.

c. CAP O2004-0313 evaluated a "HI-HI Level" Alarm for the Dilution Structure cable vault. The Dilution Structure cable vault is an environment in which inaccessible medium voltage cables are located and could be subject to wetting. This CAP documents an alarm condition that resulted from a pumping malfunction. The implemented corrective action restored pump-out capability (i.e., the discharge path was restored). This event did not impact associated cable/equipment operability. This CAP presents objective evidence that Oyster Creek operating experience has not identified standing water in manholes/vaults as recurrent issue or issue of consequence or significance.

d. An August 2005 Oyster Creek work order, C2010972, performed an inspection of a manhole that is in-scope for license renewal. The inspection was performed to determine if there was any moisture/water present, potentially subjecting inaccessible medium voltage cables to a wetted environment. Only a small amount of water was observed in the manhole (<1/2 inch) which does not subject the contained cables to wetted conditions since all cables run on racks along the walls.

It was additionally observed that the support structures are satisfactory with no visible signs of rusting or degradation.

The evaluations and subsequent actions associated with these CAPs provide objective evidence that Oyster Creek's operating experience does not show a current adverse trend with respect to cable insulation performance. The problems encountered have been evaluated under the corrective action process and corrective actions have been implemented preventing future impact to the safe operation of the plant. There is confidence that the implementation of this new Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program will effectively manage age related insulation degradation due to adverse environmental conditions (i.e., circuit energization greater than 25% of the time while in wetted conditions). Oyster Creek's operating experience and corrective actions to date support implementation of this aging management program in alignment with NUREG-1801.

2. There is no plant specific operating experience with an existing aging management program, beyond the above demonstrated use of the corrective action process and implementation of the existing Oyster Creek Medium Voltage Cable Test Program, since this is a new aging management program.

3. Industry operating experience has demonstrated that adverse localized environments (i.e., circuit energization greater than 25% of the time while in wetted conditions) can degrade the insulation of inaccessible medium voltage cables potentially impacting electrical continuity. The EPRI TR-1003664, Medium-Voltage Cables in Nuclear Plant Applications – State of Industry and Conditioning Monitoring, October 2003 report and NRC Information Notice 2002-12: Submerged Safety-Related Electrical Cables provide documentation of industry experience and analysis of this aging effect. The phenomenon is reflected in the program elements as described in NUREG-1801, Chapter XI program XI.E3.

The operating experience associated with the Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program does not show a current adverse trend in performance. The problems identified and evaluated in CAP O2003-1000 have appropriate corrective actions taken and in place to prevent recurrence.

There is confidence that the implementation of the new Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program will effectively determine insulation degradation in a timely manner so that replacement/repair can occur prior to failure.

3.10 Conclusion

The Oyster Creek Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program is credited for managing cable insulation degradation, due to circuit energization greater than 25% of the time while in wetted conditions, for the systems, components, and environments listed in Table 5.2. The Oyster Creek Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements 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 future implementing documents 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 implementation of the new Oyster Creek Inaccessible Medium Voltage Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements aging management program provides reasonable assurance that insulation degradation, due to circuit energization greater than 25% of the time while in wetted conditions 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

4.1.5 10 CFR 50.49, *Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants*

4.1.6 NUREG/CR-5643, *Insights Gained From Aging Research,* U. S. Nuclear Regulatory Commission, March 1992

4.1.7 NRC Information Notice 2002-12, Submerged Safety-Related Electrical Cables

4.2 Industry Standards

4.2.1 EPRI TR-109619, *Guideline for the Management of Adverse Localized Equipment Environments*, Electric Power Research Institute, Palo Alto, CA, June 1999

4.2.2 IEEE Std. P1205-2000, *IEEE Guide for Assessing, Monitoring and Mitigating Aging Effects on Class 1E Equipment Used in Nuclear Power Generating Stations*

4.2.3 SAND96-0344, *Aging Management Guideline for Commercial Nuclear Power Plants - Electrical Cable and Terminations*, prepared by Sandia National Laboratories for the U.S. Department of Energy, September 1996

4.2.4 EPRI TR-1003057, License Renewal Electrical Handbook, December 2001

4.2.5 EPRI TR-1003664, Medium-Voltage Cables in Nuclear Plant Applications – State of Industry and Conditioning Monitoring, October 2003

4.3 Oyster Creek Program References

4.3.1 AmerGen response to RAI 2.5.1.19-1, October 12, 2005

4.3.2 Oyster Creek Engineering Evaluation 0164-97, Oyster Creek Medium Voltage Cable Test Program, May 15, 1997

4.3.3 "White Paper," Appendix F, on Assessment of Underground Cable Failures, submitted to the state of New Jersey, Bureau of Nuclear Engineering, 7/25/05

4.3.4 Passport AR330592, Assignment 36, Subassignments 01, 02, and 03

4.3.5 Oyster Creek AMP Audit Questions

1. AMP-099
2. AMP-133
3. AMP-137
4. AMP-143

5.0 TABLES

5.1 Aging Management Program Implementing Documents

Procedure Number	Procedure Title	Commitment No.	Status
New Procedure ER-OC-xxx	Testing of Inaccessible Medium-Voltage Cables	AR # 00330592.36.01	ACC/ASG
New Routine Task RXXXXXXX	Perform Cable Tests per New Procedure ER-OC-xxx	AR # 00330592.36.02	ACC/ASG
New Routine Task RXXXXXXX	New WO to Perform Manhole Inspections	AR # 00330592.36.03	ACC/ASG

5.2 Aging Management Review Results

SSC Name	Structure and/or Component	Material	Environment	Aging Effect
Electrical Commodity Groups	Insulated inaccessible medium-voltage cables	Various organic polymers (e.g., EPR, XLPE, PVC, ETFE)	Adverse Localized Environment (External)	Localized damage and breakdown of insulation leading to electrical failure/ moisture intrusion, water trees
Station Blackout System - Electrical Commodities	Insulated Inaccessible Medium-Voltage Cables	Various organic polymers (e.g., EPR)	Adverse Localized Environment (Electrical Only) (External)	Localized damage and breakdown of insulation leading to electrical failure / moisture infusion, water trees

6.0 ATTACHMENTS

6.1 LRA Appendix A

6.2 LRA Appendix B

PROGRAM BASIS DOCUMENT

PBD-AMP-B.1.13

Revision 0

OPEN CYCLE COOLING WATER SYSTEM

GALL PROGRAM XI.M20 - OPEN CYCLE COOLING WATER SYSTEM

Prepared By: ______________________

Reviewed By: ______________________

Program Owner Review: ______________________

Technical Lead Approval: ______________________

Revision History:

Revision	*Prepared by:*	*Reviewed by:*	*Program Owner:*	*Approved by:*
0	*S. Rafferty-Czincila*	*Mark Miller*	*Roger Gayley*	*Don Warfel*
Date				

Summary of Revisions:

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

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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 Open Cycle Cooling Water aging management program that are credited for managing aging of piping, piping components, piping elements and heat exchangers that are included in the scope of license renewal for loss of material and reduction of heat transfer, caused by the buildup of deposits, and are exposed to raw water - salt water 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.M20. Project Level Instruction PLI-8 "Program Basis Documents" prescribes the methodology for evaluating Aging Management Programs. An evaluation of Oyster Creek's 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:

a) *The program relies on implementation of the recommendations of the Nuclear Regulatory Commission (NRC) Generic Letter (GL) 89-13 to ensure that the effects of aging on the open-cycle cooling water (OCCW) (or service water) system will be managed for the extended period of operation.*

b) *The program includes surveillance and control techniques to manage aging effects caused by biofouling, corrosion, erosion, protective coating failures, and silting in the OCCW system or structures and components serviced by the OCCW system.*

Oyster Creek:

a) The GL 89-13 activities provide for management of aging effects in raw water cooling systems through tests and inspections per the guidelines of NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety Related Equipment." System and component testing, visual inspections, NDE (RT, UT, and/or ECT-Eddy Current Testing), and chemical treatment are conducted to ensure that aging effects are managed such that system and component intended functions and integrity are maintained. Removal of existing outer coatings on piping is performed as required prior to NDE.

b) The Oyster Creek Open-Cycle Cooling Water System (OCCWS) aging management program (AMP) primarily consists of station GL 89-13 activities that include chemical and biocide injection, system testing, periodic inspections and NDE. The program includes surveillance and control techniques to manage aging effects caused by biofouling, corrosion, erosion, protective coating failures, and silting in the OCCW system or structures and components serviced by the OCCW system. Other activities include station maintenance inspections, component preventive maintenance (PM), plant surveillance testing, ISI, and inspections. These activities provide for management of loss of material (without credit for protective coatings) and buildup of deposit (including fouling from biological, corrosion product, and external sources) aging effects where applicable in system components exposed to a raw water environment.

Corporate and station procedures provide instructions and controls for preventive actions through raw water chemistry control (chemical and biocide injection), performance monitoring through station testing, and condition-monitoring and leak detection through inspection and testing of Oyster Creek raw water systems in the scope of license renewal. The Oyster Creek Inservice Pressure Testing Program provides for periodic leakage detection of aboveground and buried piping and components as well as inspection of aboveground piping and components. **(Reference: OC-4)**

OCCWS AMP testing and inspections at Oyster Creek have detected buildup of deposit and loss of material aging effects in raw water system components prior to loss of system intended functions. GL 89-13 program assessments have been performed, and corrective actions have been implemented.

For heat exchangers, an aging management program that uses multiple attributes is considered necessary to effectively address all aging effects. These AMP activities provide input into a total program that includes primary and secondary operating fluid chemistry controls, performance monitoring and inspections of all heat exchangers in the scope of license renewal at Oyster Creek to manage loss of material, and buildup of deposit where applicable as outlined in EPRI Report "Non Class 1 Mechanical Implementation Guideline and Mechanical Tools, 2001.1003056, Rev. 3, Appendix G Heat Exchangers," Sandia National Laboratory Report SAND93-7070

UC-523, "Aging Management Guideline for Commercial Nuclear Power Plants – Heat Exchangers," and ASME OM-S/G-2000, Part 21, "Inservice Performance Testing of Heat Exchangers in Light-Water Reactor Power Plants."

OCCWS AMP activities are also used for managing portions of raw water systems that have been identified as meeting Criteria 10CFR54(a)(2) for "non-safety related/safety related interactions" (NSR/SR) and either meet GALL line criteria or are non-GALL with a component intended function of "structural integrity (attached)." The AMP activities are credited for managing loss of material aging degradation to control fluid interaction with safety related components for components with the intended function of "leakage boundary (spatial)" to preserve the intended function of "structural integrity (attached). Because the AMP activities are credited for managing loss of material aging degradation, 'flow blockage' need not be evaluated for NSR/SR components.

2.2 Overall NUREG-1801 Consistency

The Oyster Creek Open Cycle Cooling Water System aging management program is an existing program that is consistent with NUREG-1801 aging management program XI.M20.

2.3 Summary of Exceptions to NUREG-1801

None. The existing Oyster Creek Open Cycle Cooling Water System 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

The Open-Cycle Cooling Water aging management program will be enhanced to include:

- Volumetric inspections, for piping that has been replaced, at a minimum of 4 aboveground locations every 4 years based on the observed and anticipated performance of the new pipe.
- Specificity on inspection of heat exchangers for loss of material due to general, pitting, crevice, galvanic and microbiologically influenced corrosion in the RBCCW, TBCCW and Containment Spray preventative maintenance tasks.

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.

3.0 Scope of Program

NUREG-1801:

a) *The program addresses the aging effects of material loss and fouling due to micro- or macro-organisms and various corrosion mechanisms.*

b) *Because the characteristics of the service water system may be specific to each facility, the OCCW system is defined as a system or systems that transfer heat from safety-related systems, structures, and components (SSC) to the ultimate heat sink (UHS). If an intermediate system is used between the safety-related SSCs and the system rejecting heat to the UHS, that intermediate system performs the function of a service water system and is thus included in the scope of recommendations of NRC GL 89-13.*

c) *The guidelines of NRC GL 89-13 include (a) surveillance and control of biofouling; (b) a test program to verify heat transfer capabilities; (c) routine inspection and a maintenance program to ensure that corrosion, erosion, protective coating failure, silting, and biofouling cannot degrade the performance of safety-related systems serviced by OCCW; (d) a system walk down inspection to ensure compliance with the licensing basis; and (e) a review of maintenance, operating, and training practices and procedures.*

Oyster Creek:

a) The Oyster Creek OCCWS AMP activities provide for management of loss of material and buildup of deposit aging effects where applicable in components exposed to raw cooling water. Components are evaluated, within the GALL piping, flow orifices, pumps and valves component groupings, for open cycle cooling water system components required for license renewal. **(Reference: PM00120M, PM00184M, PM00209M, PM00189M, PM24104I, PM24105I, PM00116M, PM00118M, PM53207M, PMVT0005, PMVT0006, PMVT0016, 607.4.007, 607.4.008, 642.4.001, 322, 309.1.1, 309.1, 607.4.016, 607.4.017, 641.4.001, TDR-829, SP-1302-12-261, TDR-1063, C-1302-241-E120-109, ER-AA-2030, ER-AA-340, CY-OC-120-1106 & 326)**

b) The OCCW AMP includes components in the station Emergency Service Water (ESW) and the Service Water (SW) systems. The Oyster Creek License Renewal systems of Chlorination and Roof Drains and Overboard Discharge are included for aging management as part of this program. The components of the above systems that are in scope and require aging management are part of the ESW and SW plant systems. However due to the license renewal boundary designations these portions of the systems were classified as the Chlorination and Roof Drains and Overboard Discharge license renewal systems. The components of the Chlorination and Roof Drains and Overboard Discharge license renewal systems will be managed as part of the OCCW aging management activities for the ESW and SW plant systems.

   The RBCCW and Containment Spray heat exchangers are included for the intended function of pressure boundary and heat transfer. The TBCCW heat exchangers are included for a leakage boundary function only. The current GL 89-13 program at Oyster Creek only includes the ESW system and Containment Spray heat exchangers. This commitment will not change, however attributes of the GL 89-13 guidance will be implemented in the SW system and RBCCW and TBCCW heat exchangers as part of the License Renewal Open Cycle Cooling Water aging management program.

c) The guidelines of NRC GL 89-13 include (a) surveillance and control of biofouling; (b) a test program to verify heat transfer capabilities; (c) routine inspection and a maintenance program to ensure that corrosion, erosion, protective coating failure, silting, and biofouling cannot degrade the performance of safety-related systems serviced by OCCW; (d) a system walk down inspection to ensure compliance with the licensing basis; and (e) a review of maintenance, operating, and training practices and procedures.

(a) Surveillance and control of biofouling is accomplished with continuous chlorination and periodically checking chlorine concentration in the ESW and SW systems. **(Reference: CY-OC-120-1106 and 326)**

(b) Oyster Creek has a test program to verify heat transfer capabilities. This is accomplished by performing heat transfer testing, operations monitoring of differential pressure across the TBCCW & RBCCW heat exchangers, as an indication of fouling, and cleaning schedules that are within the guidance of GL 89-13 requirements. **(Reference: PM24104I, PM24105I, PM00120M, PM00184M, PM00209M PM00189M, PM00116M, PM00118M, 322, 642.4.001 and 309.1)**

(c) Routine inspection and a maintenance program to ensure that corrosion, erosion, protective coating failure, silting, and biofouling cannot degrade the performance of safety-related systems serviced by OCCW.

The RBCCW & TBCCW heat exchangers are cleaned and inspected annually. The Containment Spray heat exchangers are cleaned and inspected every 3 years. The PM's for these cleaning and inspections will be enhanced to state inspection for loss of material due to general, pitting, crevice, galvanic and microbiologically influenced corrosion. **(Reference: PM00116M, PM00118M, PM00120M, PM00184M, PM00209M and PM00189M)**

Additionally, Operations monitors' differential pressure across the TBCCW & RBCCW heat exchangers when the systems are in service. The procedures for monitoring the systems state that cleaning of the heat exchangers is to be scheduled if the differential pressure exceeds the set limit. This, along with the annual cleaning, ensures adequate heat transfer capability. **(Reference: 322 and 309.1)**

Containment Spray heat exchangers are heat transfer tested annually to ensure adequate heat transfer capability. **(Reference: PM24104I and PM24105I)** ESW & SW System testing is performed quarterly. **(Reference: 607.4.016, 607.4.017 and 641.4.001)** During this IST Pump test, flow and system parameters are measured and trended. ESW & SW Pipe UT inspections are performed every 2 yrs in original piping and will be performed every 4 yrs on piping that has been replaced. **(Reference: TDR-829 and SP-1302-12-261)**

(d) System walkdowns are performed in accordance with the system manager walkdown procedure, which requires the system manager to look at System parameters normal (temperature/ pressure/ DP /flow /water levels) and Material condition of components, supports, etc. **(Reference: ER-AA-2030)**

(e) Maintenance, operations, and training practices and procedures are controlled in accordance with administrative programs that meet requirements of 10CFR50 Appendix B. These activities meet the guidelines outlined in NRC GL 89-13.

OCCW systems' activities include chemistry controls, performance monitoring and periodic inspections in order to control biofouling, verify heat transfer, monitor degradation, and ensure compliance with the current licensing bases.

The internal environment of any buried portions of the ESW and SW systems is included in this program. The external condition of the buried portions of the ESW and SW systems is included in the B.1.26, "Buried Piping and Tanks Inspection Program."

The Oyster Creek Open Cycle Cooling Water aging management program manages the aging effect of loss of material and reduction of heat transfer 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.

<u>Exceptions to NUREG-1801, Element 1:</u>

None.

<u>Enhancements to NUREG-1801, Element 1:</u>

The Open-Cycle Cooling Water aging management program will be enhanced to include:

- Volumetric inspections, for piping that has been replaced, at a minimum of 4 aboveground locations every 4 years based on the observed and anticipated performance of the new pipe.
- Specificity on inspection of heat exchangers for loss of material due to general, pitting, crevice, galvanic and microbiologically influenced corrosion in the RBCCW, TBCCW and Containment Spray preventative maintenance tasks.

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

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

3.1 Preventive Actions

<u>NUREG-1801:</u>

The system components are constructed of appropriate materials and lined or coated to protect the underlying metal surfaces from being exposed to aggressive cooling water environments. Implementation of NRC GL 89-13 includes a condition and performance monitoring program; control or preventive measures, such as chemical treatment, whenever the potential for biological fouling species exists; or flushing of infrequently used systems. Treatment with chemicals mitigates microbiologically-influenced corrosion (MIC) and buildup of macroscopic biological fouling species, such as blue mussels, oysters, or clams. Periodic flushing of the system removes accumulations of biofouling agents, corrosion products, and silt.

<u>Oyster Creek:</u>

The system components are constructed of appropriate materials and lined or coated to protect the underlying metal surfaces from being exposed to aggressive cooling water environments. Implementation of NRC GL 89-13 includes a condition and performance monitoring program; control or preventive measures, such as chemical treatment, whenever the potential for biological fouling species exists; or flushing of infrequently used systems if required. Treatment with chemicals mitigates microbiologically-influenced corrosion (MIC) and buildup of macroscopic biological fouling species, such as blue mussels, oysters, or clams. **(Reference: CY-OC-120-1106 and 326)**

Flushing of infrequently used systems is not required because all of the heat exchangers are periodically cleaned and inspected for the accumulation of biofouling agents, corrosion products and silt. Additionally, the ESW and SW systems are inservice for normal plant operations or systems testing on frequent bases that flushing is not necessary. The RBCCW & TBCCW heat exchangers are cleaned and inspected annually. **(Reference: PM00120M, PM00184M & PM00209M PM00189M)**
The Containment Spray heat exchangers are cleaned and inspected every 3 years. **(Reference: PM00116M & PM00118M)**
The ESW & SW System testing is performed quarterly, which ensures adequate flow through the systems. **(Reference: 607.4.016, 607.4.017, 641.4.001)**

The presence of internal linings for corrosion protection is conservatively not credited. Degradation of internal coatings can contribute to potential downstream flow blockage. However NUREG-1801 Table IX.F under the aging mechanism of "fouling" states that reduction of system flow rate is considered active and thus not in the purview of license renewal. Therefore credit is not being taken for internal coating inspections.

Condition and performance monitoring is accomplished by the performance of station in service tests that have specific acceptance criteria which will ensure that conditions such as biological fouling and deterioration from aggressive cooling water environments are detected. This acceptance criterion includes attributes such as differential pressure, flows and system pressures. These tests are performed on a frequency that will detect deterioration prior to the loss of system intended function. **(Reference: 607.4.016, 607.4.017, 641.4.001, 607.4.007, 607.4.008, 309.1, 309.1.1, 642.4.001 and 322)**

Control or preventive measures such as chlorination treatment mitigates microbiologically influenced corrosion (MIC) and the buildup of macroscopic biological fouling species, such as blue mussels, oysters, or clams in the ESW & SW systems. **(Reference: CY-OC-120-1106 and 326)** Chemistry control of the RBCCW & TBCCW heat exchangers, closed-cycle side, is conducted as part of the Closed-Cycle Cooling water aging management program B.1.14.

OCCWS AMP activities provide preventive actions through the use of chlorination to mitigate aging mechanisms that lead to buildup of deposits. **(Reference: CY-OC-120-1106 and 326)**

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.

3.2 Parameters Monitored or Inspected

NUREG-1801:

a) *Adverse effects on system or component performance are caused by accumulations of biofouling agents, corrosion products, and silt. Cleanliness and material integrity of piping, components, heat exchangers, elastomers, and their internal linings or coatings (when applicable) that are part of the OCCW system or that are cooled by the OCCW system are periodically inspected, monitored, or tested to ensure heat transfer capabilities.*

b) *The program ensures (a) removal of accumulations of biofouling agents, corrosion products, and silt, and (b) detection of defective protective coatings and corroded OCCW system piping and components that could adversely affect performance of their intended safety functions.*

Oyster Creek:

a) The Oyster Creek OCCWS AMP procedures direct the inspection and testing for monitoring of aging degradation and ensuring heat exchanger capabilities in raw water system components in the scope of license renewal. Procedures and work orders direct the visual and volumetric inspection of components for detection of degradation. **(Reference: PM00116M, PM00118M, PM00120M, PM00184M, PM00209M, PM00189M, PMVT0005, PMVT0006, PMVT0016, TDR-829, SP-1302-12-261)** The elastomers identified in the ESW & SW systems that are exposed to Raw Water-Slat Water are managed as part of the Periodic Inspection Program (B.2.5).

b) The program ensures (a) removal of accumulations of biofouling agents, corrosion products, and silt, and (b) detection of defective protective coatings and corroded OCCW system piping and components that could adversely affect performance of their intended safety functions.

(a) Removal of accumulations of biofouling agents, corrosion products, and silt are performed via cleaning and inspections. The RBCCW & TBCCW heat exchangers are cleaned and inspected annually. **(Reference: PM00120M, PM00184M & PM00209M PM00189M)** The Containment Spray heat exchangers are cleaned and inspected every 3 years. **(Reference: PM00116M & PM00118M)** Additionally the ESW & SW systems are tested for operability on a quarterly basis, which would ensure that biofouling agents, corrosion products and silt build up, is not preventing the systems from performing their intended functions. **(Reference: 607.4.016, 607.4.017, 641.4.001)**

(b) Detection of defective protective coatings and corroded OCCW system piping and components that could adversely affect performance of their intended safety functions is accomplished by ESW & SW operability testing which is performed quarterly. **(Reference: 607.4.016, 607.4.017, 641.4.001)**

The Oyster Creek Underground Piping Program Description and Status planning document, which is part of the Oyster Creek Buried Pipe aging management program, and the Pipe Integrity Inspection Program direct UT inspections be performed on various aboveground ESW and SW. **(Reference:** PBD-AMP-B.1.26 and TDR-829, respectively) Results of the inspections are documented in the a "Pipe Integrity Inspection Program" Technical Data Report **(Reference: TDR-829)** and continually analyzed to determine susceptible locations for future inspections. Additionally, these inspections are representative of the same internal environments and failure mechanisms present in the ESW and SW buried piping. Therefore the condition of the buried ESW and SW piping are bounded by the inspections performed on the aboveground piping. Based on the "Pipe Integrity Inspection Program" Technical Data Report's **(Reference: TDR 829)** projections, inspections and monitoring a minimum of 10 UT inspections are performed every 2 years on aboveground ESW and SW piping that is original to the plant design. Additionally, a minimum of 4 UT inspections will be performed every 4 years on aboveground piping that has been replaced with the same coatings and materials as new buried ESW and SW piping. These inspections will provide a representative sample of the condition of all underground ESW and SW piping.

The Containment Spray heat exchanger's heat transfer capabilities, as required to support system operability requirements, are tested under station procedures and work orders **(Reference: 607.4.107 and PM24104I & PM24105I)** and data from the heat transfer testing is evaluated by station calculations. **(Reference: C-1302-241-E120-109 and TDR 1063)** Results of the test are evaluated by engineering in an action request (AR) eval off of the preventive maintenance work order that performed the heat transfer test. **(Reference: PM24104I and PM24105I)**

The RBCCW heat exchangers are opened every year. **(Reference: PM00189M and PM00209M)** These activities inspect the heat exchangers for macrofouling and coating degradation. The heat exchangers are cleaned as required and any degraded coatings are replaced. Anodes are weighed for trending and replaced yearly. Operations monitor performance via system differential pressure and initiates cleanings based on this trending. **(Reference: 322 and 642.4.001)**

The TBCCW heat exchanger inspections and cleaning are performed based on operations differential pressure monitoring. Inspections assure that the material conditions of the heat exchangers are acceptable. **(Reference: 309.1 and PM00120M, PM00184M)**

The Containment Spray heat exchangers meet the guidelines outlined in NRC GL 89-13, EPRI Tools, Sandia Report, and ASME OM guidelines. These activities provide reasonable assurance that cleanliness and material integrity of components in the scope of license renewal at Oyster Creek will be maintained for the period of extended operation. **(Reference:**

PM24104I, PM24105I, PM00116M and PM00118M)

Exceptions to NUREG-1801, Element 3:

None.

Enhancements to NUREG-1801, Element 3:

The Open-Cycle Cooling Water aging management program will be enhanced to include:

- Volumetric inspections, for piping that has been replaced, at a minimum of 4 aboveground locations every 4 years based on the observed and anticipated performance of the new pipe.
- Specificity on inspection of heat exchangers for loss of material due to general, pitting, crevice, galvanic and microbiologically influenced corrosion in the RBCCW, TBCCW and Containment Spray preventative maintenance tasks.

Comparison and Evaluation Conclusion:

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

3.3 Detection of Aging Effects

NUREG-1801

Inspections for biofouling, damaged coatings, and degraded material condition are conducted. Visual inspections are typically performed; however, nondestructive testing, such as ultrasonic testing, eddy current testing, and heat transfer capability testing, are effective methods to measure surface condition and the extent of wall thinning associated with the service water system piping and components, when determined necessary.

Oyster Creek:

Oyster Creek OCCWS AMP activities as described in corporate and station procedures, provide for detection of aging effects prior to loss of intended functions. Loss of material aging mechanism is detected through visual inspections and NDE.

The Containment Spray heat exchangers are eddy current tested every 3 years to inspect for loss of material. **(Reference: PM00116M and PM00118M)** The RBCCW heat exchanger tubes are titanium and therefore are not susceptible to loss of material per industry standards (EPRI Mechanical Tools Appendices G). The TBCCW heat exchangers are inspected annually for loss of material. **(Reference: PM00120M and PM00184M)**

Buildup of deposit aging mechanism is detected using a combination of system and component performance testing and component visual inspections for detecting fouling, silting and corrosion product buildup. Submerged portions of pumps are not inspected. However, the results of pump performance testing replaces the pumps prior to the failure of any components. **(Reference: 309.1.1, 322, 607.4.016, 607.4.017, 642.4.001 and 641.4.001)**

The ESW and SW Systems are performance tested for acceptable flow rates, operating temperatures and operating pressures. **(Reference: 607.4.007, 607.4.008, 607.4.016, 607.4.017, and 641.4.001)** Oyster Creek "Safety Related Specification for Pipe Integrity Inspection Program" and "Pipe Integrity Inspection Program" direct UT inspections be performed on various aboveground ESW and SW piping including inspections performed at the intake. **(Reference: SP-1302-12-261 and TDR 829)** Results of the inspections are documented in the Pipe Integrity Inspection Program and continually analyzed to determine susceptible locations for future inspections. Additionally, these inspections are representative of the same internal environments and failure mechanisms present in the ESW and SW buried piping. Therefore the condition of the buried ESW and SW piping are bounded by the inspections performed on the aboveground piping. Based on the Pipe Integrity Inspection Program projections, inspections and monitoring a minimum of 10 UT inspections will be performed every 2 years on aboveground ESW and SW piping that is original to the plant design. **(Reference: SP-1302-12-261 and TDR 829)** Additionally, a minimum of 4 UT inspections will be performed every 4 years on aboveground piping that has been replaced with the same coatings and materials as new buried ESW and SW piping. These inspections will provide a representative sample of the condition of all underground ESW and SW piping.

Flow measurements contained in plant specific procedures provide indications for detection of macrofouling as an aging effect in Oyster Creek raw water system piping and components. **(Reference: 309.1.1, 322, 607.4.007, 607.4.008, 607.4.016,**

607.4.017, 642.4.001 and 641.4.001) Periodic component inspections and inspections during component maintenance are also used for detecting loss of material and buildup of deposit aging effects. **(Reference: PM00116M, PM00118M, PM00120M, PM00184M, PM00209M and PM00189M)** Heat exchanger performance is monitored through system flowrates and primary and/or secondary temperature values and trends as required to support associated system operability. Performance testing and inspections monitor both process sides and surfaces of the heat exchangers. **(Reference: 309.1.1, 322, 607.4.007. 607.4.008, 607.4.016, 607.4.017, 642.4.001 and 641.4.001)**

Exceptions to NUREG-1801, Element 4:

None.

Enhancements to NUREG-1801, Element 4:

The Open-Cycle Cooling Water aging management program will be enhanced to include:

- Volumetric inspections, for piping that has been replaced, at a minimum of 4 aboveground locations every 4 years based on the observed and anticipated performance of the new pipe.
- Specificity on inspection of heat exchangers for loss of material due to general, pitting, crevice, galvanic and microbiologically influenced corrosion in the RBCCW, TBCCW and Containment Spray preventative maintenance tasks.

Comparison and Evaluation Conclusion:

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

3.4 Monitoring and Trending

NUREG-1801:

a) *Inspection scope, method (e.g., visual or nondestructive examination [NDE]), and testing frequencies are in accordance with the utility commitments under NRC GL 89-13.*

b) *Testing and inspections are done annually and during refueling outages.*

c) *Inspections or nondestructive testing will determine the extent of biofouling, the condition of the surface coating, the magnitude of localized pitting, and the amount of MIC, if applicable.*

d) *Heat transfer testing results are documented in plant test procedures and are trended and reviewed by the appropriate group.*

Oyster Creek:

a) Inspection scope, method (e.g., visual or nondestructive examination [NDE]), and testing frequencies are in accordance with the utility commitments under NRC GL 89-13.

Periodic OCCW systems' performance tests and visual inspections provide for timely detection of loss of material and buildup of deposit where applicable. The periodicity of the testing and inspections are based on previous findings and are continually adjusted accordingly. NDE tests consist of eddy current testing, UTs or RTs to detect loss of material aging effect where applicable.

- The Inspection Scope is: ESW & Service Water systems, TBCCW, RBCCW & Containment Spray heat exchangers.
- TBCCW & RBCCW Heat Exchangers are inspected annually. **(Reference: PM00120M, PM00184M, PM00209M and PM00189M)**
- Operations monitors' differential pressure across the TBCCW & RBCCW heat exchangers when the systems are in service. The procedures state that cleaning of the heat exchangers is to be scheduled if the differential pressure exceeds the set limit. **(Reference: 309.1 and 322)** This, along with the annual cleaning, ensures adequate heat transfer capability.
- Containment Spray Heat Exchangers are heat transfer tested annually. **(Reference: PM24104I and PM24105I)**
- Containment Spray Heat Exchangers inspections are performed every 3 yrs. **(Reference: PM00116M and PM00118M)**
- ESW & SW System testing is performed quarterly. During this IST Pump test, flow and system parameters are measured and trended. **(Reference: 607.4.016, 607.4.017, and 641.4.001)**
- ESW & SW Pipe UT inspections are performed every 2 yrs in original piping and will be performed every 4 yrs on piping that has been replaced. **(Reference: SP-1302-12-261 and**

TDR 829)

b) Based on the above schedules the OCCW heat exchangers are all either tested or inspected every year to ensure heat transfer capability in accordance with GL 89-13 requirements. Additionally the ESW and SW systems are performance tested quarterly to ensure that the service water system will perform its intended function. Therefore each system in scope of the OCCW AMP are tested and/or inspected, as applicable, annually.

- TBCCW & RBCCW Htxr's are inspected annually. **(Reference: PM00120M, PM00184M & PM00209M PM00189M)**
- Operations monitors' differential pressure across the TBCCW & RBCCW heat exchangers when the systems are in service. **(Reference: 309.1 and 322)** The procedures state that cleaning of the heat exchangers is to be scheduled if the differential pressure exceeds the set limit. This, along with the annual cleaning, ensures adequate heat transfer capability.
- Containment Spray Heat Exchangers are heat transfer tested annually. **(Reference: PM24104I and PM24105I)**
- Containment Spray Heat Exchangers inspections are performed every 3 yrs **(Reference: PM00116M & PM00118M)**
- ESW & SW System testing is performed quarterly. During this IST Pump test, flow and system parameters are measured and trended. **(Reference: 607.4.016, 607.4.017, and 641.4.001)**
- ESW Pipe UT inspections - Committed every 2 yrs old, 4 yrs new pipe
- SW Pipe UT inspections - Committed every 2 yrs old, 4 yrs new pipe

c) Inspections or nondestructive testing will determine the extent of biofouling, the condition of the surface coating, the magnitude of localized pitting, and the amount of MIC, if applicable, to ensure the pressure retaining intended function. Oyster Creek "Safety Related Specification for Pipe Integrity Inspection Program" and "Pipe Integrity Inspection Program" direct UT inspections be performed on various aboveground ESW and SW piping including inspections performed at the intake every 2 years. The results of the inspections are documented in the

TDR and continually analyzed to determine susceptible locations for future inspections. (Reference: SP-1302-12-261 and TDR 829)

ISI procedures for the Class 3 portions of raw water systems consist of VT-2 inspections of aboveground installations and leakage evaluations of buried piping for detecting thru-wall corrosion in accordance with the Oyster Creek ISI program. **(Reference: OC-1 Program Plan)**

The RBCCW heat exchangers are opened every year. **(Reference: PM00189M and PM00209M)** Anodes are weighed for trending and replaced yearly. Operations monitor performance via system differential pressure and initiates cleanings based on this trending. **(Reference: 322 and 642.4.001)**

The TBCCW heat exchanger inspections and cleaning are performed based on operations differential pressure monitoring. Inspections assure that the material conditions of the heat exchangers are acceptable. **(Reference: 309.1, PM00120M and PM00184M)**

The OCCWS procedures outline the requirements to ensure that the testing and inspection activities have been performed and the results have been documented and sent to the appropriate station personnel for trending and analysis. Available differential pressure or flow to the heat exchangers is used to determine the extent of blockage/fouling in the system.

Station procedures provide for the periodic monitoring and trending of system performance and notification of both the system engineer and unit supervisor of any inspection deficiencies. Documentation results are maintained in accordance with ASME Section XI, IWP-6000. ISI documentation facilitates comparison with previous and subsequent inspection results and is also maintained in accordance with ASME Section XI, IWP-6000.

d) Heat transfer testing results are documented in plant test procedures and are trended and reviewed by the appropriate group. (Reference: PM24104I and PM24105I)

Station procedures provide for the documentation and periodic monitoring and trending of system performance and notification of both the system engineer and unit supervisor of any inspection deficiencies. Documentation results are maintained in accordance with ASME Section XI, IWP-6000. ISI documentation facilitates comparison with previous and subsequent inspection results and is also maintained in accordance with ASME Section XI, IWP-6000.

Exceptions to NUREG-1801, Element 5:

None.

Enhancements to NUREG-1801, Element 5:

The Open-Cycle Cooling Water aging management program will be enhanced to include:

- Volumetric inspections, for piping that has been replaced, at a minimum of 4 aboveground locations every 4 years based on the observed and anticipated performance of the new pipe.
- Specificity on inspection of heat exchangers for loss of material due to general, pitting, crevice, galvanic and microbiologically influenced corrosion in the RBCCW, TBCCW and Containment Spray preventative maintenance tasks.

Comparison and Evaluation Conclusion:

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

3.5 Acceptance Criteria

NUREG-1801:

a) Biofouling is removed or reduced as part of the surveillance and control process. The program for managing biofouling and aggressive cooling water environments for OCCW systems is preventive.

b) Acceptance criteria are based on effective cleaning of biological fouling organisms and maintenance of protective coatings or linings are emphasized.

Oyster Creek:

a) Biofouling is removed or reduced as part of the chemistry surveillance and control process. The program for managing biofouling and aggressive cooling water environments for

OCCW systems is preventive. **(Reference: CY-OC-120-1106 and 326)**

b) Engineering evaluations performed as part of OCCWS AMP procedures identify the presence of aging degradation, including loss of material and buildup of deposit aging effects; and are used to confirm the component's ability to perform their intended functions. Specific acceptance criteria are provided in the specific inspection or test procedures for ensuring continued system and component operability. Cleaning of component internal surfaces of biofouling, silt and corrosion products during periodic inspections or component maintenance is confirmed as part of the testing and inspection processes. **(Reference: CY-OC-120-1106, PM00120M, PM00184M, PM00209M PM00189M, PM00116M, PM00118M, 607.4.007, 607.4.008, 642.4.001, 322, 309.1, 641.4.001, 607.4.016, 607.4.017)** Any relevant conditions detected during performance testing or inspection activities are evaluated in accordance with program requirements. Flow rates must meet minimum Tech Spec requirements (if applicable) and confirm the systems' ability to provide the required flowrate and pressure. Station procedures require that any recordable indications be evaluated and appropriate corrective actions implemented. **(Reference: 607.4.007, 607.4.008, 642.4.001, 641.4.001)**

The Oyster Creek OCCW activities provide for management of loss of material (without credit for protective coatings) and buildup of deposit (including fouling from biological, corrosion product, and external sources) aging effects where applicable in system components exposed to a raw water environment.

These criteria provide reasonable assurance that components in the scope of license renewal at Oyster Creek will remain capable of performing their intended function for the period of extended operation.

<u>Exceptions to NUREG-1801, Element 6:</u>

None.

<u>Enhancements to NUREG-1801, Element 6:</u>

None.

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

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

3.6 Corrective Actions

<u>NUREG-1801:</u>

Evaluations are performed for test or inspection results that do not satisfy established acceptance criteria and a problem or condition report is initiated to document the concern in accordance with plant administrative procedures. The corrective actions program 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 repetition. 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.

<u>Oyster Creek:</u>
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 plant administrative procedures. The 10 CFR Part 50, Appendix B corrective action program 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. **(Reference: CY-OC-120-1106, PM00120M, PM00184M, PM00209M PM00189M, PM00116M, PM00118M, 607.4.007, 607.4.008, 642.4.001, 322, 309.1, 641.4.001, 607.4.016, 607.4.017)**

<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.

3.7 Confirmation Process

<u>NUREG-1801:</u>

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 to address the confirmation process and administrative controls.

<u>Oyster Creek:</u>

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.

<u>Exceptions to NUREG-1801, Element 8:</u>

None.

<u>Enhancements to NUREG-1801, Element 8:</u>

None.

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

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

3.8 Administrative Controls

<u>NUREG-1801:</u>

See Item 8, above.

<u>Oyster Creek:</u>

See Item 8, above.

<u>Exceptions to NUREG-1801, Element 9:</u>

None.

<u>Enhancements to NUREG-1801, Element 9</u>

None.

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

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

3.9 Operating Experience

NUREG-1801:

Significant microbiologically-influenced corrosion (NRC Information Notice [IN] 85-30), failure of protective coatings (NRC IN 85-24), and fouling (NRC IN 81-21, IN 86-96) have been observed in a number of heat exchangers. The guidance of NRC GL 89-13 has been implemented for approximately 10 years and has been effective in managing aging effects due to biofouling, corrosion, erosion, protective coating failures, and silting in structures and components serviced by OCCW systems.

Oyster Creek:
Review of industry operating experience has confirmed that significant microbiologically influenced corrosion (NRC Information Notice [IN] 85-30), failure of protective coatings (NRC IN 85-24), and fouling (NRC IN 81-21, IN 86-96) have been observed in a number of heat exchangers. The guidance of NRC GL 89-13 has been implemented for approximately 10 years and has been effective in managing aging effects due to biofouling, corrosion, erosion, protective coating failures, and sitting in structures and components serviced by OCCW systems. A review of plant operating experience at Oyster Creek shows that loss of material has occurred in several different systems. In most cases, the existing Open Cycle Cooling Water aging management program has identified the loss of material 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 Open Cycle Cooling Water program was modified to more accurately predict future loss of material. The experience at Oyster Creek with the Open Cycle Cooling Water program shows that the Open Cycle Cooling Water program is effective in managing loss of material and reduction of heat transfer in piping, piping components, piping elements and heat exchangers that are exposed to raw water - salt water.

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 Open Cycle Cooling Water program successfully manages the effects of aging is achieved through objective evidence that shows that loss of material and reduction of heat transfer is being adequately managed in piping and other components. The following examples of operating experience provide objective evidence that the Open Cycle Cooling Water program is effective in assuring that intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation:

1. During inspections of the RBCCW heat exchangers in 2003, shells of crab, mussels and some fish were discovered. However there were no living organisms. This example provides objective evidence that the chemical treatment of the RBCCW heat exchangers is effective in controlling of living organism that could become detrimental to the system.

2. The RBCCW heat exchanger tubes were replaced with Titanium. This material is very resistant to corrosion/erosion in a salt-water environment. This example provides objective evidence that the program is taking measures to preclude future degradation by replacing the tubes with an improved material.

3. The injection lines of the Chlorination system that supplies SW, ESW & Circulating water were hydrostatically tested in 1993 and there were no signs of degradation. This example provides

objective evidence that the program is effective in managing loss of material and maintaining the pressure boundary intended function of the systems.

4. The following examples provide objective evidence that loss of material will be detected prior to the loss of system intended function and adequate corrective actions are taken to prevent reoccurrence.

   a) The Oyster Creek Open Cycle Cooling Water program identified and reviewed several ESW pipe leaks and wall thinning events. A common failure mechanism (local wall thinning due to salt-water corrosion) was identified. The results were entered into the corrective action process and an operability evaluation was performed in 2003. The operability evaluation also included an evaluation of the effect of the failure mechanism on the SSC safety functions including functional thresholds and methods for detection of leaks for each of the safety functions. Additionally, the corrective action process problem resolution response included the development of an inspection plan " Topical Report 140 - ESW and Service Water System Plan ". Some of the plan's goals are to prioritize modifications and inspections based on risk and consequence of a leak, modify piping segments that pose a high risk and consequence and cannot be reasonably inspected, modify piping to allow system flexibility for future repairs and inspect piping to ensure disposition/repair prior to failure. The plan captures existing analysis, past action and future action for ESW and SW pipe.

   b) In August 2001, portions of ESW System 2 were again internally inspected by a robotic video camera, during the repair activities of the underground leak near the Startup Transformers. Results of the video inspection show no significant whole scale degradation, except for the small area that developed a leak.

   c) In 1993, the 30" Overboard Discharge line developed a significant leak in an area just down stream of an elbow located approximately 10 feet from the line discharge point at the discharge canal. The root cause was attributed to flow-impingement on this downstream side of the elbow. The impingement is due to excessive velocities at this point, which wore away the internal coating and then allowed corrosion. A repair was performed in 1994, by replacing an approximate 10-foot length of the line up to the discharge point. In 1997, this line again developed a leak near the railroad airlock. In 2000, Oyster Creek rehabilitated this 900-foot long line with CIPP (Cure In Place Pipe). As a result the internal material is a fiberglass polymer that is excellent at resisting degradation due to salt water.

d) In 2004 50% of the buried ESW and 10% of the buried SW piping has been replaced with new pipe and an improved coating system. Additionally, the other 50% of the buried ESW piping is scheduled, as well as an additionally 10% of the SW buried piping, to be replaced in 2006.

5. The following examples provide objective evidence that program updates are performed as necessary to ensure program enhancements are being made to improve the assessment of loss of material thereby maintaining system intended function.

   a) Since 1986, Oyster Creek has performed up to 27 inspections on the SW & ESW systems during each operating cycle. Three types of inspections are normally performed. These are: Ultrasonic Testing (UT), Visual Examination, or Remote Robotic Video Camera Inspection. Inspection results are summarized in TDR 829, "Pipe Integrity Inspection Program" and Topical Report 116, "Oyster Creek Underground Piping Program Description and Status" which are updated after every Refueling Outage. As a result of Oyster Creek's inspection history the following conclusions have been made: 1-The mechanism results in pinhole leaks that develop gradually, 2-The internal failure mechanism seems to develop randomly along the piping system, 3-The majority of the leaks have occurred from the inside. Therefore, this review ensures that the internal inspections provide a significant level of assurance.

   b) In 1997 TR-116 performed system evaluations to determine piping with high risk of developing leaks and high consequences should leaks occur. It prioritized systems and underground lines based on the evaluations and operating history. Inspections and tests were performed on high priority systems, and inspections at targets of opportunity for the medium and low priority systems. Analysis was performed to determine generic failure mechanisms and corrective actions.

c) After 19R in 2002 it was recognized that the methodology Oyster Creek was using as part of the Pipe Integrity Inspection Program did not accurately provide accurate basis for determining whether internal coating was degraded and experiencing corrosion. In addition, the methodology did not accurately determine corrosion rates and project service life. As a result, Oyster Creek improved their pipe integrity inspection program to allow for better monitoring and detection as well as predicted service life. The current methodology states that if piping with a documented thinnest pipe wall thickness of less than 87.5% of nominal wall then that pipe will be concluded to have failed (87.5% is based on pipe manufacturer fabrication tolerance of +/- 12.5%). The service life of a component will be based on either actual corrosion rates (if available) or a conservative service life based on a rate of 20 mpy (Actual observed corrosion rates for ESW and SW range from 4-19 mpy.) In addition a bounding service life based on 33 mpy will be determined.

d) An operability evaluation (OC-2003-OE-0013) was performed in 2003 for ESW and a common failure mechanism (local wall thinning due to salt-water corrosion) was identified. OC-2003-OE-0013 also included an evaluation of the effect of the failure mechanism on the SSC safety functions including functional thresholds and methods for detection of leaks for each of the safety functions.

The operating experience of the OCCWS Program has shown that the program has been effective in identifying and managing aging effects, including erosion and corrosion, blockage due to silt buildup, microbiological growth, mussel growth, and microbiologically influenced corrosion, prior to loss of the system intended function. Therefore, there is reasonable assurance that aging effects will be managed by the continued implementation of the OCCWS Program such that SSCs within the scope of License Renewal will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation. For heat exchangers, the OCCWS AMP activities provide input into a comprehensive aging management program that uses multiple attributes to effectively address all aging effects. The OCCW System aging management program has detected loss of material, provided chemical treatment of heat exchangers to control living organisms so not to be detrimental to the system, updated the program to improve monitoring and performance assessment of loss of material and entered deficiencies in the OCCW program into the 10 CFR Part 50, Appendix B Corrective Action process. Periodic self-assessments of the OCCW program are performed to identify the areas that need improvement to maintain the quality performance of the program.

3. 11 Conclusion

The Oyster Creek Open Cycle Cooling Water aging management program manages the aging effect of loss of material and reduction of heat transfer for the systems, components, and environments listed in Table 5.2. The Oyster Creek Open Cycle Cooling Water program's elements have been evaluated against NUREG-1801 in Section 3.0. No program exceptions were 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 Open Cycle Cooling Water aging management program provides reasonable assurance that the loss of material due and reduction of heat transfer 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 89-13, *"Service Water System Problems Affecting Safety Related Equipment*

4.2.2 NRC Generic Letter 89-13, Supplement 1 *"Service Water System Problems Affecting Safety Related Equipment*

4.2.3 NRC Information Notice IN 85-30, *Microbiologically Induced Corrosion of Containment Service Water System*

4.2.4 NRC Information Notice IN 85-24, *Failures of Protective Coatings in Pipes and Heat Exchangers*

4.2.5 NRC Information Notice IN 81-21, *Potential Loss of Direct Access to Ultimate Heat Sink*

4.2.6 NRC Information Notice IN 86-96, *Heat Exchanger Fouling Can Cause Inadequate Operability of Service Water Systems.*

4.3 Oyster Creek Program References

4.3.1 OC-1, *Oyster Creek ISI program (OC-1 Program Plan)*

4.3.2 OC-4, *Program Plan Oyster Creek Generating Station Inservice Pressure Testing Program Fourth Inspection Interval*

4.3.3 TDR-1063, *Evaluation of Heat Transfer Capability of Safety Related Heat Exchangers*

4.3.4 C-1302-241-E120-109, *Containment Spray Heat Exchanger Performance Eval*

5.0 TABLES

5.1 Aging Management *see References* Documents

Procedure Number	Procedure Title	Commitment No.	Status

PM00120M	Inspect, clean & replace anodes in the TBCCW HX	330592.13.04	ACC/ASG
PM00189M	Inspect, clean & replace anodes in the RBCCW HX	330592.13.05	ACC/ASG
PM00209M	Inspect, clean & replace anodes in the RBCCW HX	330592.13.06	ACC/ASG
PM53207M	ESW SYSTEM I VIDEO INSPECTION @ THE INTAKE	330592.13.07	ACC/ASG
PM00118M	Anode Replacement and clean as required 1-3 and 1-4 containment spray heat exchanger	330592.13.37	ACC/ASG
PM00184M	Clean HX and Replace Anodes in the Heat Exchanger	330592.13.38	ACC/ASG
PM00116M	Anode Replacement and clean as required 1-1 and 1-2 containment spray heat exchanger	330592.13.46	ACC/ASG
PMVT0016	PERFORM VT-2 INSPECTION OF PIPE AND COMPONENTS IN SERVICE WATER SYSTEM	330592.13.01	ACC/ASG
OC-1	Oyster Creek ISI program (OC-1 Program Plan)	330592.13.02	ACC/ASG
OC-4	Program Plan Oyster Creek Generating Station Inservice Pressure Testing Program Fourth Inspection Interval	330592.13.03	ACC/ASG

PMVT0006	PERFORM VT-2 INSPECTION OF PIPE AND COMPONENTS IN EMERGENCY SERVICE WATER SYSTEM 2 (1240/D)	330592.13.09	ACC/ASG
607.4.007	Containment Spray and Emergency Service Water System 1 Pump Operability Test	330592.13.47	ACC/ASG
607.4.008	Containment Spray and Emergency Service Water System 2 Pump Operability Test	330592.13.15	ACC/ASG
PMVT0005	VT-2 INSP OF PIPE/COMPS IN ESW SYS 1 (1240/D)	330592.13.16	ACC/ASG
326	Chlorination System	330592.13.18	ACC/ASG
CY-OC-120-1106	Domestic Water and NJPDES-Related System sample and analysis schedule	330592.13.48	
309.1.1	Turbine Building Closed Cooling Water Routine Evolutions	330592.13.21	ACC/ASG
309.1	Turbine Building Closed Cooling Water System	330592.13.49	ACC/ASG
322	Service Water System	330592.13.50	ACC/ASG
607.4.017	Containment Spray and Emergency Service Water Pump System 2 Operability and Quarterly Inservice Test	330592.13.22	ACC/ASG
ER-AA-340	GL-89-13 Program Implementing Procedure	330592.13.26	ACC/ASG

607.4.016	Containment Spray and Emergency Service Water System I Pump Operability and Quarterly Inservice Test	330592.13.29	ACC/ASG
642.4.001	RBCCW Inservice Test	330592.13.30	ACC/ASG
641.4.001	Service Water Pump Operability and In-Service Test	330592.13.51	ACC/ASG
ER-AA-330	Conduct of Inservice Inspection Activities	330592.13.31	ACC/ASG
ER-AA-330-001	Section XI Pressure Testing	330592.13.32	ACC/ASG
Topical Report 140	Emergency Service Water and Service Water Piping Plan	330592.13.35	ACC/ASG
PM24104I	Setup for Cont. Spray Sys I HX Perf. Monitoring	330592.13.52	ACC/ASG
PM24105I	Setup for Cont. Spray Sys II HX Perf. Monitoring	330592.13.53	ACC/ASG
ER-AA-2030	Conduct of Plant Engineering Manual	330592.13.54	ACC/ASG
SP-1302-12-261	Safety Related Specification for Pipe Integrity Inspection Program	330592.13.55	ACC/ASG
TDR-829	Inspection History of OCNGS Pipe Integrity Program	330592.13.39	ACC/ASG

5.2 Aging Management Review Results

SSC Name	Structure and/or Component	Material	Environment	Aging Effect
Chlorination System	Valve Body	Cast Iron	Raw Water – Salt Water (Internal)	Loss of Material
Chlorination System	Piping and fittings	Cast Iron	Raw Water – Salt Water (Internal)	Loss of Material
Chlorination System	Valve Body	Carbon and low alloy steel	Raw Water – Salt Water (Internal)	Loss of Material
Chlorination System	Piping and fittings	Carbon and low alloy steel	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Flow Element	Nickel Alloy	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Heat Exchangers (Containment Spray)	Titanium (Tubes)	Raw Water – Salt Water (Internal)	Reduction of Heat Transfer
Emergency Service Water System	Pump Casing (ESW Pumps)	Stainless Steel	Raw Water – Salt Water	Loss of Material
Emergency Service Water System	Piping and fittings	Carbon and low alloy steel	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Restricting Orifice	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Restricting Orifice	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Piping and fittings	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Valve Body	Carbon and low alloy steel	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Heat Exchangers (Containment Spray)	Copper Alloy (Tube Side Components)	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Sight Glasses	Bronze (Body)	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Piping and fittings	Bronze	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Piping and fittings	Carbon and low alloy steel	Raw Water – Salt Water (Internal)	Loss of Material

Emergency Service Water System	Pump Casing (HTXR Drain Pumps)	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Thermowell	Carbon and low alloy steel	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Valve Body	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Piping and fittings	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Piping and fittings	Brass	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Valve Body	Carbon and low alloy steel	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Valve Body	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Valve Body	Brass	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Heat Exchangers (Containment Spray)	Aluminum Bronze (tubesheet)	Raw Water – Salt Water (Internal)	Loss of Material
Emergency Service Water System	Heat Exchangers (Containment Spray)	Titanium (Tubes)	Raw Water – Salt Water (Internal)	Loss of Material
Roof Drains and Overboard Discharge	Piping and fittings	Bronze	Raw Water – Salt Water (Internal)	Loss of Material
Roof Drains and Overboard Discharge	Piping and fittings	Carbon and low alloy steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Pump Casing (Service Water Pumps)	Bronze (Bowl Assembly)	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Piping and fittings	Carbon and low alloy steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Pump Casing (Service Water Pumps)	Cast Iron (Discharge Head and Bowl Assembly)	Raw Water – Salt Water (External)	Loss of Material
Service Water System	Piping and fittings	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Pump Casing (Service Water Pumps)	Stainless Steel (Column Pipe)	Raw Water – Salt Water (External)	Loss of Material
Service Water System	Pump Casing (Service Water Pumps)	Bronze (Bowl Assembly)	Raw Water – Salt Water (External)	Loss of Material
Service Water System	Strainer Body	Copper Alloy	Raw Water – Salt Water (Internal)	Loss of Material

Service Water System	Valve Body	Carbon and low alloy steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Piping and fittings	Brass	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Pump Casing (Service Water Pumps)	Stainless Steel (Column Pipe)	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Valve Body	Cast Iron	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Piping and fittings	Bronze	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Piping and fittings	Bronze	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Heat Exchangers (TBCCW)	Carbon and low alloy steel (Tube Side Components)	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Sample Chamber	Titanium	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Heat Exchangers (RBCCW)	Titanium (Tubes)	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Heat Exchangers (RBCCW)	Titanium (Tube Sheet)	Raw Water – Salt Water (External)	Loss of Material
Service Water System	Heat Exchangers (RBCCW)	Titanium (Tubes)	Raw Water – Salt Water (Internal)	Reduction of Heat Transfer
Service Water System	Rotameter	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Restricting Orifice	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Restricting Orifice	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Pump Casing (Rad Monitor Sample Pump)	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Eductor	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Gauge Snubber	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Valve Body	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Piping and fittings	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material

Service Water System	Valve Body	Carbon and low alloy steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Valve Body	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Valve Body	Copper Alloy	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Sight Glasses	Stainless Steel (Body)	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Sight Glasses	Copper Alloy (Body)	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Pump Casing (Service Water Pumps)	Cast Iron (Discharge Head and Bowl Assembly)	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Thermowell	Carbon and low alloy steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Thermowell	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Strainer Body	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Piping and fittings	Brass	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Flow Element	Stainless Steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Heat Exchangers (RBCCW)	Carbon Steel (Tube Side Components)	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Piping and fittings	Carbon and low alloy steel	Raw Water – Salt Water (Internal)	Loss of Material
Service Water System	Valve Body	Copper Alloy	Raw Water – Salt Water (Internal)	Loss of Material

6.0 ATTACHMENTS

6.1 LRA Appendix A

6.2 LRA Appendix B

PROGRAM BASIS DOCUMENT

PBD-AMP-B.1.06

Revision 0

BWR CONTROL ROD DRIVE RETURN LINE NOZZLE

GALL PROGRAM XI.M6 - BWR CONTROL ROD DRIVE RETURN LINE NOZZLE

Prepared By: ____________________

Reviewed By: ____________________

Program Owner Review: ____________________

Technical Lead Approval: ____________________

Revision History:

Revision	*Prepared by:*	*Reviewed by:*	*Program Owner:*	*Approved by:*
0	*Mark Miller*	*George Beck*	*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 Control Rod Drive Return Line (CRDRL) Nozzle aging management program that are credited for managing cracking in the CRDRL nozzle 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.M6, BWR Control Rod Drive Return Line Nozzle. Project Level Instruction PLI-8 "Program Basis Documents" prescribes the methodology for evaluating Aging Management Programs. An evaluation of Oyster Creek's 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

<u>NUREG-1801:</u>

This program includes

a) *enhanced inservice inspection (ISI) in conformance with the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWB, Table IWB 2500-1 (2001 edition*[1] *including the 2002 and 2003 Addenda) and the recommendations of NUREG-0619, and*

b) *system modifications and maintenance programs to mitigate cracking.*

c) *The program specifies periodic liquid penetrant and ultrasonic inspection of critical regions of the boiling water reactor (BWR) control rod drive return line (CRDRL) nozzle.*

<u>Oyster Creek:</u>

a) Control rod drive return line (CRDRL) nozzle monitoring at Oyster Creek is in conformance with Subsection IWB, Table IWB 2500-1 of the 1995 Edition through 1996 Addendum of ASME Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components," augmented by inspections performed in accordance with NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle" as described

[1] An applicant may reply on a different version of the ASME Code, but should justify such use. An applicant may wish to refer to the SOC for the an update of 10CFR § 50.55a to justify use of a more recent edition of the Code.

in item c) below **(Reference: Document Number OC-1, Section 1, paragraphs 1.6 and 2.2.2; Document Number OC-1, Section 1, Table 7.0-1, Examination Category B-D; Document Number OC-1, Section 2, paragraph 4.1.11; Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*" dated October 4, 1994).**

b) Modification to the CRDRL nozzle thermal sleeve was made at Oyster Creek in response to NUREG-0619 to mitigate or prevent thermally induced fatigue cracking **(Reference: Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*" dated October 4, 1994; MPR Associates, Inc "*Design Report of CRD Return Nozzle Thermal Sleeve Modification,*" dated September 1977).**

c) Inspection of the CRDRL nozzle is performed in accordance with the Oyster Creek Generating Station ISI Program Plan for the Fourth Ten-Year Inspection Interval. The examination of the CRDRL nozzle as identified in NUREG-0619 is considered an augmented inspection program at Oyster Creek and includes alternate examination requirements using ultrasonic (UT) examination techniques at a frequency of at least once per each 10-years (120 months). The periodic dye penetrant (PT) examination of portions of the blend radius and bore areas has been eliminated, except that a verification examination using a PT method is required if the UT examination results indicate the presence of a flaw exceeding the ASME Code allowable crack size **(Reference: Document Number OC-1, Section 1, paragraph 2.2.2; Document Number OC-1, Section 1, Table 7.0-1, Examination Category B-D; Document Number OC-1, Section 2, paragraph 4.1.11; Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*" dated October 4, 1994).**

2.2 Overall NUREG-1801 Consistency

The Oyster Creek BWR CRDRL Nozzle aging management program is an existing program that is consistent with NUREG-1801 aging management program XI.M6, BWR Control Rod Drive Return Line Nozzle with exceptions as described in 2.3 below.

2.3 Summary of Exceptions to NUREG-1801

NUREG-1801, XI.M6, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

NUREG-1801, XI.M6, specifies dye penetrant testing of the CRD return line nozzle. The Oyster Creek augmented ISI program for the CRD return line nozzle includes ultrasonic examination (UT) testing in lieu of dye penetrant testing (PT). Oyster Creek requested and received relief from the NRC to perform ultrasonic examination (UT) testing in lieu of the periodic PT testing requirements specified in NUREG 0619. A verification examination using a PT method is still required if the UT examination results indicate the presence of a flaw exceeding the ASME Code allowable crack size.

NUREG-1801, XI.M6, specifies that any detected crack be ground out. Oyster Creek procedures allow a crack that is found unacceptable under IWB-3400 and IWB-3500 to be evaluated under ASME XI, IWB-3600 or repaired by an NRC approved procedure.

2.4 Summary of Enhancements to NUREG-1801

None. The existing Oyster Creek BWR CRDRL Nozzle 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.

3.0 Scope of Program

NUREG-1801:

The program includes systems modifications, enhanced ISI, and maintenance programs to monitor the effects of cracking on the intended function of CRDRL nozzles.

Oyster Creek:

In response to NUREG-0619, modifications were made to the CRDRL nozzle thermal sleeve at Oyster Creek to mitigate or prevent thermally induced fatigue cracking **(Reference: Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*" dated October 4, 1994; MPR Associates, Inc "*Design Report of CRD Return Nozzle Thermal Sleeve Modification,*" dated September 1977)**. The Oyster Creek CRDRL nozzle aging management program also includes inspections of the CRDRL nozzle in accordance with Section XI of the ASME Code, augmented by inspections performed in accordance with NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle" **(Reference: Document Number OC-1, Section 1, paragraph 2.2.2; Document Number OC-1, Section 1, Table 7.0-1, Examination Category B-D; Document Number OC-1, Section 2, paragraph 4.1.11; Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*" dated October 4, 1994).**

The Oyster Creek BWR CRDRL nozzle aging management program manages the aging effect of cracking initiation and growth for the CRD nozzle and thermal sleeves 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.

3.1 Preventive Actions

NUREG-1801:

a) Mitigation occurs by system modifications, such as rerouting the CRDRL to a system that connects to the reactor vessel.

b) For some classes of BWRs, or those that can prove satisfactory system operation, mitigation is also accomplished by confirmation of proper return flow capability, two-pump operation, and cutting and capping the CRDRL nozzle without rerouting.

Oyster Creek:

a) In response to NUREG-0619, modifications were made to the CRDRL nozzle thermal sleeve at Oyster Creek to mitigate or prevent thermally induced fatigue cracking **(Reference: Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*" dated October 4, 1994; MPR Associates, Inc "*Design Report of CRD Return Nozzle Thermal Sleeve Modification,*" dated September 1977).**

b) At Oyster Creek, mitigation is not accomplished by confirmation of proper return flow capability, two-pump operation, or cutting and capping the CRDRL nozzle without rerouting. The Oyster

Creek CRDRL nozzle aging management program includes inspections of the CRDRL nozzle in accordance with Section XI of the ASME Code, augmented by inspections performed in accordance with NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle" **(Reference: Document Number OC-1, Section 1, paragraph 2.2.2; Document Number OC-1, Section 1, Table 7.0-1, Examination Category B-D; Document Number OC-1, Section 2, paragraph 4.1.11; Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*" dated October 4, 1994).** Additionally, in response to NUREG-0619, modifications were made to the CRDRL nozzle thermal sleeve at Oyster Creek to mitigate or prevent thermally induced fatigue cracking **(Reference: Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*" dated October 4, 1994; MPR Associates, Inc "*Design Report of CRD Return Nozzle Thermal Sleeve Modification,*" dated September 1977).**

<u>Exceptions to NUREG-1801, Element 2:</u>

None.

<u>Enhancements to NUREG-1801, Element 2:</u>

None.

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

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

3.2 Parameters Monitored or Inspected

<u>NUREG-1801:</u>

The aging management program (AMP) monitors the effects of cracking on the intended function of the CRDRL nozzles by detecting and sizing cracks by ISI in accordance with Table IWB 2500-1 and NUREG-0619.

<u>Oyster Creek:</u>

The Oyster Creek CRDRL Nozzle aging management program monitors the effects of cracking on the intended function of the

CRDRL nozzles by detecting and sizing cracks by ISI in accordance with Subsection IWB, Table IWB 2500-1 of the 1995 Edition through 1996 Addendum of ASME Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components," augmented by inspections performed in accordance with NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle" **(Reference: Document Number OC-1, Section 1, paragraphs 1.6 and 2.2.2; Document Number OC-1, Section 1, Table 7.0-1, Examination Category B-D; Document Number OC-1, Section 2, paragraph 4.1.11; Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*" dated October 4, 1994).**

<u>Exceptions to NUREG-1801, Element 3:</u>

NUREG-1801, XI.M6, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

NUREG-1801, XI.M6, specifies dye penetrant testing of the CRD return line nozzle. The Oyster Creek augmented ISI program for the CRD return line nozzle includes ultrasonic examination (UT) testing in lieu of dye penetrant testing (PT). Oyster Creek requested and received relief from the NRC to perform ultrasonic examination (UT) testing in lieu of the periodic PT testing requirements specified in NUREG 0619. A verification examination using a PT method is still required if the UT examination results indicate the presence of a flaw exceeding the ASME Code allowable crack size.

<u>Enhancements to NUREG-1801, Element 3:</u>

None.

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

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

3.3 Detection of Aging Effects

NUREG-1801:

a) The extent and schedule of inspection, as delineated in NUREG-0619, assures detection of cracks before the loss of intended function of the CRDRL nozzles.

b) Inspection recommendations include liquid penetrant testing (PT) of CRDRL nozzle blend radius and bore regions and the reactor vessel wall area beneath the nozzle, return-flow-capacity demonstration, CRD-system-performance testing, and ultrasonic inspection of welded connections in the rerouted line. The inspection is to include base metal to a distance of one-pipe-wall thickness or 0.5 in., whichever is greater, on both sides of the weld.

Oyster Creek:

a) Inspection of the CRDRL nozzle is performed in accordance with the Oyster Creek Generating Station ISI Program Plan for the Fourth Ten-Year Inspection Interval. The examination of the CRDRL nozzle as identified in NUREG-0619 is considered an augmented inspection program at Oyster Creek and includes alternate examination requirements using ultrasonic (UT) examination techniques at a frequency of at least once per each 10-years (120 months) **(Reference: Document Number OC-1, Section 1, paragraph 2.2.2; Document Number OC-1, Section 1, Table 7.0-1, Examination Category B-D; Document Number OC-1, Section 2, paragraph 4.1.11; Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*" dated October 4, 1994).**

b) The examination of the CRDRL nozzle as identified in NUREG-0619 is considered an augmented inspection program at Oyster Creek and includes alternate examination requirements using ultrasonic (UT) examination techniques on the nozzle blend radius and bore regions and the reactor vessel wall area beneath the nozzle. The inspection includes base metal to a distance of one-pipe-wall thickness or 0.5 in., whichever is greater, on both sides of the weld. The periodic dye penetrant (PT) examination of portions of the blend radius and bore areas has been eliminated, except that a verification examination using a PT method is required if the UT examination results indicate the presence of a flaw exceeding the ASME Code allowable crack size. The Oyster Creek CRDRL Nozzle aging management program activities do not include return-flow-capacity demonstration, CRD-system-performance testing, or ultrasonic inspection of welded connections in the rerouted line **(Reference: Document Number OC-1, Section 1, paragraph 2.2.2; Document Number OC-1, Section 1, Table 7.0-1, Examination Category B-D; Document Number OC-1, Section 2, paragraph 4.1.11; Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*" dated October 4, 1994).**

Exceptions to NUREG-1801, Element 4:

NUREG-1801, XI.M6, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

NUREG-1801, XI.M6, specifies dye penetrant testing of the CRD return line nozzle. The Oyster Creek augmented ISI program for the CRD return line nozzle includes ultrasonic examination (UT) testing in lieu of dye penetrant testing (PT). Oyster Creek requested and received relief from the NRC to perform ultrasonic examination (UT) testing in lieu of the periodic PT testing requirements specified in NUREG 0619. A verification examination using a PT method is still required if the UT examination results indicate the presence of a flaw exceeding the ASME Code allowable crack size.

Enhancements to NUREG-1801, Element 4:

None.

Comparison and Evaluation Conclusion:

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

3.4 Monitoring and Trending

NUREG-1801:

The inspection schedule of NUREG-0619 provides timely detection of cracks.

Oyster Creek:

The CRDRL nozzle is included in the Oyster Creek ISI program plan under Category B-D, "Full Penetration Welds of Nozzles in Vessels," consistent with the requirements of Table IWB 2500-1. Periodic CRDRL nozzle inspections are performed using UT techniques at least once every 10-year period (120 months) in accordance with NUREG-0619 **(Reference: Document Number OC-1, Section 1, paragraph 2.2.2; Document Number OC-1, Section 1, Table 7.0-1, Examination Category B-D; Document Number OC-1, Section 2, paragraph 4.1.11; Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*" dated October 4, 1994).**

Exceptions to NUREG-1801, Element 5:

NUREG-1801, XI.M6, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

NUREG-1801, XI.M6, specifies dye penetrant testing of the CRD return line nozzle. The Oyster Creek augmented ISI program for the CRD return line nozzle includes ultrasonic examination (UT) testing in lieu of dye penetrant testing (PT). Oyster Creek requested and received relief from the NRC to perform ultrasonic examination (UT) testing in lieu of the periodic PT testing requirements specified in NUREG 0619. A verification examination using a PT method is still required if the UT examination results indicate the presence of a flaw exceeding the ASME Code allowable crack size.

Enhancements to NUREG-1801, Element 5:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 5, Monitoring and Trending with exceptions as described above.

3.5 Acceptance Criteria

NUREG-1801:

a) *Any cracking is evaluated in accordance with IWB-3100 by comparing inspection results with the acceptance standards of IWB-3400 and IWB-3500.*

b) *All cracks found in the CRDRL nozzles are to be removed by grinding.*

Oyster Creek:

a) OC procedures follow ASME Section XI for evaluating flaw indications. Flaw indications are evaluated in accordance with the guidelines of ASME Section XI IWB-3100, using the acceptance standards of IWB-3512 as directed by IWB-3410 **(Reference: Procedure ER-AA-330-002, paragraph 4.10; ASME Section XI, Table IWB-2500-1, Examination Category B-D, Item Nos. B3.90 and B3.100).**

b) Repairs such as grinding are performed only if analytical evaluations show that the flaw does not meet the acceptance criteria of IWB-3600 **(Reference: Procedure ER-AA-330-002, paragraph 4.12.4).**

Exceptions to NUREG-1801, Element 6:

NUREG-1801, XI.M6, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

NUREG-1801, XI.M6, specifies that any detected crack be ground out. Oyster Creek procedures allow a crack that is found unacceptable under IWB-3400 and IWB-3500 to be evaluated under ASME XI, IWB-3600 or repaired by an NRC approved procedure.

Enhancements to NUREG-1801, Element 6:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 6, Acceptance Criteria with exceptions as described above.

3.6 Corrective Actions

NUREG-1801:

Repair is performed in conformance with IWB-4000 and replacement in accordance with IWB-7000. 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:
If the flaw exceeds the requirements of IWB-3600, repair and replacement is performed consistent with the requirements of ASME Section XI Subsection IWA-4000. In the 1995 edition of ASME Section XI Sections IWB-4000 and IWB-7000 have been deleted and their requirements placed in IWA-4000 **(Reference: Procedure ER-AA-330-002, paragraph 4.12.4)**.

Evaluations are performed for examination results that do not satisfy the acceptance standards of IWB-3500 and an Issue Report is initiated to document the concern in accordance with plant administrative procedures **(Reference: Procedure ER-AA-330-002, paragraph 4.12)**. The 10 CFR Part 50, Appendix B corrective action process ensures that 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.

Exceptions to NUREG-1801, Element 7:

NUREG-1801, XI.M6, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda. The current Oyster

Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

NUREG-1801, XI.M6, specifies that any detected crack be ground out. Oyster Creek procedures allow a crack that is found unacceptable under IWB-3400 and IWB-3500 to be evaluated under ASME XI, IWB-3600 or repaired by an NRC approved procedure.

Enhancements to NUREG-1801, Element 7:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 7, Corrective Actions with exceptions as described above.

3.7 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 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.

3.8 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.

3.9 Operating Experience

NUREG-1801:

Cracking has occurred in several BWR plants (NUREG-0619 and Information Notice 2004-08). The present AMP has been implemented for nearly 25 years and has been found to be effective in managing the effect of cracking on the intended function of CRDRL nozzles.

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.), 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.

BWR CRD return line nozzle inspections are implemented through the station ISI program plan, which incorporates the requirements of the ASME Code. Augmented inspections are performed in accordance with NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle".

Cracking has occurred in several BWR plants (NUREG-0619 and Information Notice 2004-08). The Oyster Creek NUREG-0619 BWR CRDRL Nozzle aging management program has been implemented for over 25 years and has been effective in managing the effect of cracking on the intended function of CRDRL nozzles. In 1977, in response to industry experience (NUREG-0619), the CRD return line thermal sleeve was removed and a dye penetrant examination was performed on the inside diameter of the CRD return nozzle. No indication of cracking was observed and the thermal sleeve was replaced with a modified design to divert the cold CRD return flow away from the nozzle. The nozzle was most recently inspected using UT techniques in 2002. No signs of cracking were found. In addition to the NUREG-0619 inspections, the CRDRL nozzle thermal sleeve is visually inspected every other outage as part of the Oyster Creek OC-5 Reactor Internals Program. This VT-1 exam has not identified any cracking or shown evidence of bypass flow. This provides objective evidence that the thermal sleeve modification and inspection schedule has been effective in preventing thermal fatigue cracking in the CRD return line nozzle.

The existing BWR control rod drive return line nozzle aging management activities will provide for the timely detection of aging degradation prior to loss of system or component intended functions for the period of extended operation.

3.10 Conclusion

The Oyster Creek BWR CRDRL Nozzle aging management program is credited for managing cracking for the components, materials, and environments listed in Table 5.2. The Oyster Creek BWR CRDRL Nozzle program's elements have been evaluated against NUREG-1801 in Section 3.0. Program exceptions have been identified in Section 2.3. There are no program enhancements required as 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 CRDRL Nozzle aging management program provides reasonable assurance that aging effects 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 NUREG-0619, "*BWR Feedwater Nozzle and Control Rod*

Drive Return Line Nozzle Cracking"

4.3 Oyster Creek Program References

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

4.3.2 Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*" dated October 4, 1994

4.3.3 MPR Associates, Inc "*Design Report of CRD Return Nozzle Thermal Sleeve Modification," dated September 1977*

5.0 TABLES

5.1 Aging Management Program Implementing Documents

Procedure Number	Procedure Title	Commitment Number	Status
ER-AA-330-002	Inservice Inspection of Section XI Welds and Components	00330592.06.0 1	ACC/ASG
ER-AA-330-009	ASME Section XI Repair/Replacement Program	00330592.06.0 2	ACC/ASG
OC-1	ISI Program Plan Fourth Ten-Year Inspection Interval	00330592.06.0 3	ACC/ASG
OC-3	ISI Repair/Replacement Program	00330592.06.0 4	ACC/ASG
ER-AA-330	Conduct of Inservice Inspection Activities	00330592.06.0 5	ACC/ASG

5.2 Aging Management Review Results

SSC Name	Structure and/or Component	Material	Environment	Aging Effect
Reactor Pressure Vessel	Nozzles (CRD Return)	Carbon and low alloy steel (with stainless steel cladding)	Treated Water (Internal)	Cracking Initiation and Growth
Reactor Pressure Vessel	Nozzle Thermal Sleeves (CRD Return Line)	Stainless Steel	Treated Water	Cracking Initiation and Growth

6.0 ATTACHMENTS

6.1 LRA Appendix A

6.2 LRA Appendix B

PROGRAM BASIS DOCUMENT

PBD-AMP-B.1.03

Revision 0

REACTOR HEAD CLOSURE STUDS

GALL PROGRAM XI.M3 – REACTOR HEAD CLOSURE STUDS

Prepared By: M. J. May

Reviewed By: G. J. Beck

Program Owner Review: G. F. Harttraft

Technical Lead Approval: D. B. Warfel

Revision History:

Revision	*Prepared by:*	*Reviewed by:*	*Program Owner:*	*Approved by:*
0	M. J. May	G. J. Beck	G. F. Harttraft	D. B. 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 Reactor Head Closure Studs aging management program that are credited for managing cracking, loss of material due to wear and the effects of coolant leakage on the reactor head closure studs 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; and
- 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.M3, Reactor Closure Head Studs. Project Level Instruction PLI-8 "Program Basis Documents" prescribes the methodology for evaluating Aging Management Programs. An evaluation of Oyster Creek's 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:

This program includes:

a) *inservice inspection (ISI) in conformance with the requirements of the American Society of Mechanical Engineers (ASME), Code, Section XI, Subsection IWB (2001 edition[1] including the 2002 and 2003 Addenda), Table IWB 2500-1, and*

b) *preventive measures to mitigate cracking.*

Oyster Creek:

a) The reactor head closure studs program is an existing program that performs inservice inspections of the closure head nuts, washers, and bushings in accordance with ASME Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components," Table IWB 2500-1. The Inservice inspection program established for the fourth ten-year inspection interval (years 2002 – 2012 or end of License in 2009) are in accordance with ASME Section XI 1995 Edition through 1996 Addendum. **(Reference: OC-1 paragraphs 1.2,1.3, and 1.6)**

b) The program includes preventive measures to mitigate cracking, which include coating the studs, washer, and nuts and controlling the use of lubricates. The reactor head closure

[1] An applicant may rely on a different version of the ASME Code, but should justify such use. An applicant may wish to refer to the SOC for an update of 10 CFR § 50.55a to justify use of a more recent edition of the Code.

studs, nuts, washers, and bushings at Oyster Creek are not metal-plated, and have been parkerized (manganese phosphate surface treatment) **(Reference 4.3.2).** As specified in Reg Guide 1.65 only approved lubricants are used. Neolube No. 1, or an Engineering/Controlled Material Program approved equivalent, is applied to the nuts and threads and all bearing surfaces of the nuts and washers prior to reactor vessel head replacement. **(Reference: 2400-SMM-3221.01, paragraphs 6.2.30, 6.4.35.3)**

2.2 Overall NUREG-1801 Consistency

The Oyster Creek Reactor Head Closure Studs program is an existing program that is consistent with NUREG-1801 aging management program XI.M3, Reactor Closure Head Studs.

2.3 Summary of Exceptions to NUREG-1801

NUREG-1801, XI.M3, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

2.4 Summary of Enhancements to NUREG-1801

None. The existing Oyster Creek Reactor Head Closure Studs program 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.

3.0 Scope of Program

NUREG-1801:

The program includes

a) *ISI to detect cracking due to stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC), loss of material due to wear, and coolant leakage from reactor vessel closure stud bolting for both boiling water reactors (BWRs) and pressurized water reactors (PWRs);*

b) *preventive measures of NRC Regulatory Guide 1.65 to mitigate cracking.*

c) *The program is applicable to closure studs and nuts constructed from materials with a maximum tensile strength limited to less than 1,172 MPa (170 ksi) (Nuclear Regulatory Commission [NRC] Regulatory Guide [RG] 1.65).*

Oyster Creek:

a) The reactor head closure studs are inspected as part of the Oyster Creek ISI program in accordance with ASME Section XI, Subsection IWB. The scope of inspection activities includes the reactor head closure studs (in place or removed), nuts, washers, and bushings. The inspections monitor for cracking due to stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC), loss of material due to wear, and coolant leakage **(Reference: OC-1, Section 3, Table 2.2-8).**

b) The reactor head closure studs and nuts used at Oyster Creek include the preventive measures described in RG 1.65, which include the of approved corrosion inhibitors and lubricants. The reactor head closure studs, nuts, washers, and bushings at Oyster Creek are not metal-plated, and have been parkerized (manganese phosphate surface treatment). In accordance with Reg Guide 1.65 only approved lubricates are used. Neolube No. 1, or an Engineering/Controlled Material Program approved equivalent, is applied to the nuts and threads and all bearing surfaces of the nuts and washers prior to reactor vessel head replacement. **(Reference: 4.3.2, 2400-SMM-3221.01)**

c) The Oyster Creek reactor head closure studs are constructed of ASME SA-193 GR. AISI 4340 material, which has a maximum tensile strength of less than 170 ksi, which complies with Regulatory Guide 1.65. **(Reference: 4.3.2; 4.3.3)**

The Oyster Creek Closure Reactor Head Studs aging management program manages the aging effect of cracking due to stress corrosion cracking (SCC) or intergranular stress corrosion cracking (IGSCC) and loss of material due to wear 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.

<u>Exceptions to NUREG-1801, Element 1:</u>

NUREG-1801, XI.M3, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

<u>Enhancements to NUREG-1801, Element 1:</u>

None.

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

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

3.1 Preventive Actions

NUREG-1801:

a) *Preventive measures include avoiding the use of metal-plated stud bolting to prevent degradation due to corrosion or hydrogen embrittlement, and to*

b) *use manganese phosphate or other acceptable surface treatments and stable lubricants (RG 1.65). Implementation of these mitigation measures is can reduce SCC or IGSCC, thus making this program effective.*

Oyster Creek:

a) The Oyster Creek Reactor Closure Head Studs program does not use metal-plated studs and bolts as a measure to prevent corrosion or hydrogen embrittlement **(Reference 4.3.2).**

b) To help the program to be more effective and help reduce the effects SCC or IGSCC the reactor head closure studs, nuts, washers, and bushings at Oyster Creek have been parkerized (manganese phosphate surface treatment) as recommended in Reg Guide 1.65. **(Reference 4.3.2).** Additionally, Neolube No. 1, or an Engineering/Controlled Material Program approved equivalent, is applied to the nuts and threads and all bearing surfaces of the nuts and washers prior to reactor vessel head replacement **(Reference 4.3.2; 2400-SMM-3221.01).**

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.

3.2 Parameters Monitored/Inspected

NUREG-1801:

The ASME Section XI ISI program detects and sizes cracks, detects loss of material, and detects coolant leakage by following the examination and inspection requirements specified in Table IWB-2500-1.

<u>Oyster Creek:</u>

The Oyster Creek Reactor Closure Head Studs program implements ASME Section XI inspection requirements through the ISI program plan. The inspections monitor for cracking, loss of material and coolant leakage. The extent and schedule for inspecting the reactor head closure studs, nuts, washers, and bushings are consistent with those specified in Table IWB-2500-1 for category B-G-1 components **(Reference OC-1, Section 3, Table 2.2-8).**

<u>Exceptions to NUREG-1801, Element 3:</u>

NUREG-1801, XI.M3, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

<u>Enhancements to NUREG-1801, Element 3:</u>

None.

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

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

3.3 Detection of Aging Effects

<u>NUREG-1801:</u>

a) *The extent and schedule of the inspection and test techniques prescribed by the program are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of intended function of the component. Inspection can reveal cracking, loss of material due to corrosion* *or wear, and leakage of coolant.*

b) *The program uses visual, surface, and volumetric examinations in accordance with the general requirements of Subsection IWA-2000. Surface examination uses magnetic particle, liquid penetration, or eddy current examinations to indicate the presence of surface discontinuities and flaws. Volumetric examination uses radiographic or ultrasonic examinations to indicate the presence of discontinuities or flaws throughout the volume of material. Visual VT-2 examination detects evidence of leakage from pressure-retaining components, as required during the system pressure test.*

c) *Components are examined and tested as specified in Table IWB-2500-1.*

d) *Examination category B-G-1 for pressure-retaining bolting greater than 2 in. diameter in reactor vessels specifies volumetric examination of studs in place, from the top of the nut to the bottom of the flange hole, and surface and volumetric examination of studs when removed.*

e) *Also specified are volumetric examination of flange threads and visual VT-1 examination of surfaces of nuts, washers, and bushings.*

f) *Examination category B-P for all pressure-retaining components specifies visual VT-2 examination of all pressure-retaining boundary components during the system leakage test and the system hydrostatic test.*

Oyster Creek:

a) The extent and schedule of the inspections specified in the Reactor Head Closure Studs program are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of intended function of the component. The ISI inspections look for signs of crack initiation and growth, as well as for loss of material due to corrosion or wear, and for signs of leakage of coolant. The Oyster Creek Reactor Closure Head Studs program implements ASME Section XI inspection requirements through the ISI program plan. **(Reference: OC-1, paragraphs 1.1, 1.2, 1.3).**

b) The program uses visual, surface, and volumetric examinations in accordance with the general requirements of Subsection IWA-E Section XI Subsection IWA-2000. The Reactor Head Studs program was developed in accordance with the requirements detailed in the 1995 Edition, 1996 Addenda, of the ASME Boiler and Pressure Vessel Code, Section XI, Division 1, Subsections IWA, IWB, Mandatory Appendices, and Inspection Program B of IWA-2432 **(Reference: OC-1, paragraph 2.1).**

ASME Section XI allows for a number of examinations methods to be used for volumetric, surface and visual inspections. The current practice at Oyster Creek is to perform volumetric exams using ultrasonic technology (UT) and VT-1 to detect cracks in the head studs. For surface examinations of the studs that are removed magnetic particle exam procedures are employed. During system pressure tests, VT-2 visual techniques are employed to monitor for coolant leakage **(References; OC-1, Section 3, Table 2.2-8 and Table 2.2-24).**

c) The reactor head closure stud bolting components are examined and tested as specified in Table IWB-2500-1 for B-G-1 components. **(References; OC-1, Section 3, Table 2.2-8).**

d) The reactor head closure stud bolting components are examined and tested as specified in Table IWB-2500-1 for B-G-1 components, Pressure, "Retaining Bolting Greater than 2 Inches in Diameter". As specified in ASME Section XI Table 2500-1for B-G-1 components, the closure studs are inspected volumetrically in place, and by volumetric and surface examinations when removed. The threaded region of the stud in the flange is also inspected volumetrically from the top of the nut to the bottom of the flange **(Reference; OC-1, Section 3, Table 2.2-8).**

e) The flange threads receive a volumetric examination and the surfaces of nuts, washers, and bushings are inspected using a VT-1 examination **(Reference; OC-1, Section 3, Table 2.2-8).**

f) All pressure-retaining boundary components in Examination Category B-P receive a visual VT-2 examination during the system leakage test and the system hydrostatic test. **(Reference; OC-1, Section 3, Table 2.2-24).**

<u>Exceptions to NUREG-1801, Element 4:</u>

NUREG-1801, XI.M3, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

Enhancements to NUREG-1801, Element 4:

None.

Comparison and Evaluation Conclusion:

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

3.4 Monitoring and Trending

NUREG-1801:

The Inspection schedule of IWB-2400, and the extent and frequency of IWB-2500-1 provide timely detection of cracks, loss of material, and leakage.

Oyster Creek:

The required examinations in each examination category for Class 1 components, subject to examination per Section XI, Subsection IWB, shall be completed during the inspection interval in accordance with Table IWB-2412-1. **(Reference: OC-1, Section 3, paragraph 1.3**). The extent and frequency of the inspections are performed in accordance with Table 2500-1 for B-G-1 components. The inspection schedule of the head studs provides for timely detection of cracks, loss of material, or coolant leakage.

Exceptions to NUREG-1801, Element 5:

NUREG-1801, XI.M3, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

Enhancements to NUREG-1801, Element 5:

None.

Comparison and Evaluation Conclusion:

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

3.5 Acceptance Criteria

NUREG-1801:

Any indication or relevant condition of degradation in closure stud bolting is evaluated in accordance with IWB-3100 by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500.

Oyster Creek:

Indications and relevant conditions detected during examinations are evaluated in accordance with ASME Section XI Subsection IWB-3100, for Class 1 components by comparing ISI results with the acceptance standards of IWB-3400 and IWB-3500 **(Reference: ER-AA-330-002, paragraph 4.10; OC-1 Section 3 paragraph 2.3.2)**. Specifically, Flaw indications or relevant conditions are evaluated in accordance with IWB-3515 or IWB-3517 as indicated in table IWB-2500-1 and Table 3410-1 of ASME Section XI. If the component qualifies as acceptable for continued service, the areas containing such flaw indications or relevant conditions shall be reexamined during the next three inspection periods **(Reference: OC-1, Section 3, paragraph 2.3.2).**

Exceptions to NUREG-1801, Element 6:

NUREG-1801, XI.M3, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

Enhancements to NUREG-1801, Element 6:

None.

Comparison and Evaluation Conclusion:

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

3.6 Corrective Actions

NUREG-1801:

a) *Repair and replacement are performed in conformance with the requirements of IWB-400 and IWB-7000, respectively, and the material and inspection guidance of RG 1.65.*

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 corrective actions.*

Oyster Creek:

a) Repair and replacement is performed consistent with the requirements of ASME Section XI Subsection IWA-4000 **(Reference: OC-3, paragraph 4.1.2; ER-AA-330-009, paragraph 4.12)**. In the 1995 edition of ASME Section XI Sections IWB-4000 and IWB-7000 have been deleted and their requirements placed in IWA-4000. If required, replacement reactor head stud components will comply with the guidance in Reg Guide 1.65. Oyster Creek currently maintains the original reactor head closure studs. 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 station's corrective action program.

b) Evaluations are performed for examination results that do not satisfy the acceptance standards of IWB-3500 and an Issue Report is initiated to document the concern in accordance with plant administrative procedures **(Reference: Procedure ER-AA-330-002, paragraph 4.12)**. The 10 CFR Part 50, Appendix B corrective action process ensures that 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.

Exceptions to NUREG-1801, Element 7:

NUREG-1801, XI.M3, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

Enhancements to NUREG-1801, Element 7:

None.

Comparison and Evaluation Conclusion:

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

3.7 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 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.

3.8 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.

3.9 Operating Experience

NUREG-1801:

The SCC has occurred in BWR pressure vessel head studs (Stoller 1991). The aging management program (AMP) has provisions regarding inspection techniques and evaluation, material specifications, corrosion prevention, and other aspects of reactor pressure vessel head stud cracking. Implementation of the program provides reasonable assurance that the effects of cracking due to SCC or IGSCC and loss of material due to wear will be adequately managed so that the intended functions of the reactor head closure studs and bolts will be maintained consistent with the current licensing basis (CLB) for the period of extended operation.

Oyster Creek:

Review of industry operating experience has confirmed that cracking due to SCC has occurred in reactor head studs **(Reference 4.2.2, Stoller Report).** A review of plant operating experience at Oyster Creek shows that cracking of the head studs from SCC or IGSCC and loss of material due to wear has not occurred. The existing Reactor Head Closure Studs aging management program has identified some occurrence of mechanical damage due to mishandling. The experience at Oyster Creek with the Reactor Head Closure Studs program shows that the program is effective in managing cracking due to SCC or IGSCC and loss of material due to wear.

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.), 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.

Reactor head stud component inspections are implemented through the station ISI program plan, which incorporates the requirements of the ASME Code, Section XI. The Oyster Creek is currently in its fourth ISI inspection interval. In the history of the Oyster Creek ISI program no evidence of head stud cracking has been found. The operating experience for these components indicates that nicks, scratches, gouges, and thread damage have occurred due to maintenance activities during refueling outages. This normal wear type of damage was determined to be acceptable for continued service. There have been no deficiencies attributed to distortion/plastic deformation from stress relaxation or loss of material due to mechanical wear **(References ISI Post 1R19 Outage Report Summary Report (NIS-1), Oyster Creek Generating Station, Feb 20, 2003; ISI Post 1R20 Outage Report Summary Report (NIS-1), Oyster Creek Generating Station, January 22, 2005).**

While removing the reactor vessel head during disassembly in 2002 (1R19) the reactor head became caught on the alignment guides. Following removal of the head , all head studs, the alignment guides and stud holes in the reactor head were inspected for damage. One of the guides was replaced because of significant gouges. Threads on one of the studs were evaluated. After clean up the studs were evaluated to be acceptable for continued use. The holes and studs were cleaned up and determined to be suitable for continued use **(Reference: CAP No O2002-1544).**

These examples provide objective evidence that the effects of cracking due to SCC or IGSCC and loss of material due to wear will be detected prior to the loss of intended function and adequate corrective actions are taken to prevent recurrence.
The operating experience of the Reactor Closure Head Studs 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 Reactor Closure Head Studs program will effectively manage degradation of head stud components due to cracking and loss of material due to corrosion or wear. Appropriate guidance for reevaluation, repair or replacement is provided for head stud components that indicate degradation due to cracking or loss of material due to corrosion or wear that may exceed acceptance limits prior to the next inspection. Periodic self-assessments of the Reactor Closure Head Studs program are performed to identify the areas that need improvement to maintain the quality performance of the program.

3.10 Conclusion.

The Oyster Creek Reactor Head Closure Studs aging management program is credited for managing cracking for the systems, components, and environments listed in Table 5.2. The Oyster Creek Reactor Head Closure Studs program's elements have been evaluated against NUREG-1801 in Section 3.0. Program exceptions have been identified in Section 2.3. There are no program enhancements that have been identified in Section 2.4. The implementing documents 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 Reactor Head Closure Studs aging management program provides reasonable assurance that the effects of cracking due to SCC or IGSCC and loss of material due to wear 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 Reg Guide 1.65

4.2.2 Stoller, S. M., *Reactor Head Closure Stud Cracking, Material Toughness Outside FSAR - SCC in Thread Roots*, Nuclear Power Experience, BWR-2, III, 58, 1991, p. 30.

4.3 Oyster Creek Program References

4.3.1 OC-1, Oyster Creek ISI Program Plan

4.3.2 Drawing CE 232-573, "Stud, Nut, Washer, & Bushing Details", Revision 10.

4.3.3 Test report, Crucible Steel Company, Piece Number 573-01, code Number G-385.

5.0 TABLES

5.1 Aging Management Program Implementing Documents

Procedure Number	Procedure Title	Commitment Number	Status
OC-1	Inservice Inspection of Section XI Welds and Components	00330592.03.03	ACC/ASG
OC-3	Section XI Repair/Replacement Program	00330592.03.05	ACC/ASG
2400-SMM-3221.01	Reactor Vessel Head Removal and Replacement Rev. 12	00330592.03.04	ACC/ASG
ER-AA-330	Conduct of Inservice Inspection Activities	00330592.03.06	ACC/ASG
ER-AA-330-002	Inservice Inspection of Section XI Welds and Components	00330592.03.02	ACC/ASG
ER-AA-330-009	AME Section XI Repair/Replacement Program	00330592.03.01	ACC/ASG

5.2 Aging Management Review Results

SSC Name	Structure and/or Component	Material	Environment	Aging Effect
Reactor Pressure Vessel	Top Head Closure Studs and Nuts	High Strength Alloy Steel	Containment Atmosphere	Cracking Initiation and Growth

6.0 ATTACHMENTS

6.1 LRA Appendix A

6.2 LRA Appendix B

PROGRAM BASIS DOCUMENT

PBD-AMP-B.1.30

Revision 0

MASONRY WALL PROGRAM

GALL PROGRAM XI.S5 - MASONRY WALL PROGRAM

Prepared By: ____________________

Reviewed By: ____________________

Program Owner Review: ____________________

Technical Lead Approval: ____________________

Revision History:

Revision	*Prepared by:*	*Reviewed by:*	*Program Owner:*	*Approved by:*
0	*A. Ouaou*	*L. Corsi*	*S. Niogi*	*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 Masonry Wall aging management program that are credited for managing cracking of masonry walls 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; and
- The identification of Oyster Creek documents required implementing 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 also provides a comparison of the credited Oyster Creek program with the elements of the corresponding NUREG-1801 Chapter XI program XI.S5. Project Level Instruction PLI-8 "Program Basis Documents" prescribes the methodology for evaluating Aging Management Programs. An evaluation of Oyster Creek's 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:

a) *Nuclear Regulatory Commission (NRC) IE Bulletin (IEB) 80-11, "Masonry Wall Design," and NRC Information Notice (IN) 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11," constitute an acceptable basis for a masonry wall aging management program (AMP).*

b) *The objective of the masonry wall program is to manage aging effects so that the evaluation basis established for each masonry wall within the scope of license renewal remains valid through the period of extended operation. Since the issuance of NRC IEB 80-11 and NRC IN 87-67, the NRC promulgated 10 CFR 50.65, the Maintenance Rule. Masonry walls may be inspected as part of the Structures Monitoring Program (XI.S6) conducted for the Maintenance Rule, provided the ten attributes described below are incorporated.*

c) *Important elements in the evaluation of many masonry walls during the NRC IEB 80-11 program included (1) installation of steel edge supports to provide a sound technical basis for boundary conditions used in seismic analysis and (2) installation of steel bracing to ensure containment of unreinforced masonry walls during a seismic event. Consequently, in addition to the development of cracks in the masonry walls, loss of function of the structural steel supports and bracing would also invalidate the evaluation basis.*

The attributes of an acceptable Masonry Wall Program are described below.

Oyster Creek:

a) The Oyster Creek Masonry Wall Program was developed to satisfy the requirements of the Nuclear Regulatory Commission (NRC) IE Bulletin (IEB) 80-11, "Masonry Wall Design," The program implementation documents consider operating experience from NRC Information Notice (IN) 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11,"

b) The objective of the Masonry Wall Program is to manage cracking so that the design and evaluation basis established for each masonry wall within the scope of license renewal remains valid through the period of extended operation. The program is a part of the Oyster Structures Monitoring Program developed to satisfy the requirements of 10 CFR 50.65, the Maintenance Rule. The Oyster Creek Structures Monitoring aging management program incorporates the ten attributes of the Masonry Wall Program described below.

c) As required by NRC IE Bulletin 80-11, Oyster Creek identified safety-related masonry walls and masonry walls whose failure during a seismic event could adversely impact a safety function. Corrective actions, including (1) installation of steel edge supports to establish technical basis for assumed boundary condition in the seismic analysis modeling, (2) installation of steel edge supports and bracings, (3) removal of the unnecessary masonry walls, (3) removal of excess equipment loads from walls, and (4) repair visible cracks on both sides of the wall (**Reference: 4.3.9, 4.3.10**). The masonry wall structural supports are monitored under the Oyster Creek Structures Monitoring aging management program to ensure that a loss of support function does not occur. Masonry walls are monitored as specified in this program to ensure that cracking is detected and corrected before a loss of an intended function.

2.2 Overall NUREG-1801 Consistency

The Oyster Creek Masonry Wall Program is an existing program that is consistent with NUREG-1801 aging management program XI.S5.

2.3 Summary of Exceptions to NUREG-1801

None. The existing Oyster Creek Masonry Wall 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 Masonry Wall 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.

3.0 Scope of Program

NUREG-1801:

The scope includes all masonry walls identified as performing intended functions in accordance with 10 CFR 54.4.

Oyster Creek:

The scope of the program includes all masonry walls identified during the scoping and screening process as performing intended functions in accordance with 10 CFR 54.4 (**Reference: 125.6 paragraph 2.0**).

The Oyster Creek Masonry Wall aging management program manages the aging effect of cracking of masonry walls in 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.

3.1 Preventive Actions

NUREG-1801:

No specific preventive actions are required.

Oyster Creek:

This program specifies no preventive actions. The program is a condition monitoring that utilizes inspections to identify aging effects prior to a loss of an intended function.

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.

3.2 Parameters Monitored or Inspected

NUREG-1801:

The primary parameter monitored is wall cracking that could potentially invalidate the evaluation basis.

Oyster Creek:

Masonry walls are monitored for cracks that could impact structural integrity of the wall or potentially invalidate the structural evaluation basis **(Reference: 125.6 paragraph 7.0)**

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.

3.3 Detection of Aging Effects

NUREG-1801:

a) *Visual examination of the masonry walls by qualified inspection personnel is sufficient. The frequency of inspection is selected to ensure there is no loss of intended function between inspections.*

b) *The inspection frequency may vary from wall to wall, depending on the significance of cracking in the evaluation basis*

c) *Unreinforced masonry walls that have not been contained by bracing warrant the most frequent inspection, because the development of cracks may invalidate the existing evaluation basis.*

Oyster Creek:

a) Masonry walls are visually inspected by qualified personnel who are required to have a B.S. degree and/or Professional Engineer license, and a minimum of five years experience working on the building structures of Nuclear Power Plants **(Reference: 125.6 paragraph 4.2)**. Inspection frequency is every 4 years consistent with implementation of 10 CFR 50.65, Maintenance Rule, requirements and industry practices. This frequency is adequate to provide reasonable assurance that a loss of an intended function, due to age related degradation, will not occur between inspections **(Reference: 125.6 paragraph 2.0)**.

b) The general inspection frequency for all structures, including masonry walls, is 4 years. The program contains provisions for more frequent inspection for masonry walls categorized (a) (1) in accordance with 10 CFR 50.65 Paragraph (a)(1). This paragraph applies to (1) those masonry walls, that are degraded to the extent that the masonry wall may not meet its design basis or (2) the masonry wall has been degraded to the extent that, if the degradation were allowed to continue uncorrected until next normally scheduled assessment, the masonry wall may not meet its design basis **(Reference: 125.6 paragraph 7.2)**

c) Unreinforced masonry walls were evaluated, braced, and repaired, as required, to satisfy the requirements of IE Bulletin 80-11. The program does not distinguish between monitoring frequency of reinforced and unreinforced walls because it would be difficult to distinguish between the two through visual inspection. However cracks that would invalidate design basis can be detected through visual inspection regardless whether the masonry wall is reinforced or unreinforced.

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.

3.4 Monitoring and Trending

NUREG-1801:

Trending is not required. Monitoring is achieved by periodic examination for cracking.

Oyster Creek:

a) The program requires periodic monitoring of masonry walls for cracking. There are no specific requirements for trending the results. However data, such as a crack length and width may be trended for walls that require monitoring in accordance with the requirements of 10 CFR 50.65 Paragraph (a)(1).

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.

3.5 Acceptance Criteria

NUREG-1801:

a) *For each masonry wall, the extent of observed cracking of masonry and degradation of steel edge supports and bracing is not to invalidate the evaluation basis.*

b) *Corrective actions are taken if the extent of cracking and steel degradation is sufficient to invalidate the evaluation basis.*

c) *An option is to develop a new evaluation basis that accounts for the degraded condition of the wall (i.e., acceptance by further evaluation).*

Oyster Creek:

a) Masonry walls are inspected by a qualified, experienced engineer for cracking. Observed cracking is documented and evaluated by the engineer considering the extent of cracking and potential impact on the design or evaluation basis of the wall. Steel edge supports and bracing are also inspected by the qualified, experienced engineer for age related degradations in accordance with the Oyster Creek Structures Monitoring aging management program. Observed age related degradations are documented and evaluated for potential impact on the ability of the support or brace to provide structural support to the wall consistent with its design basis (**Procedure: 125.6 paragraph 4.0, 7.2**).

b) Corrective actions are initiated in accordance with the Oyster Creek corrective action process if observed cracking and degradation of steel edge supports and bracing has the potential of impacting the structural integrity of the wall. The specific corrective action taken depends on the extent of the degraded condition and may include additional or more frequent inspections, performing a more detailed evaluation, or initiating a modification to repair the degraded condition. The objective of the corrective action is to ensure the masonry wall maintains the intended function consistent with its design basis **(Reference: 125.6 paragraph 4.0, 7.2)**.

c) Any engineering evaluation performed to determine acceptability of the degraded condition accounts for observed age related degradations. If the evaluation concludes that the degraded condition meets the design basis of the wall, then the evaluation becomes new evaluation basis for the wall. Otherwise, a modification is initiated to repair the degraded condition.

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.

3.6 Corrective Actions

NUREG-1801:

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:

Evaluations are performed for inspection results that do not satisfy established criteria and an Issue Report (IR) is initiated to document the concern in accordance with 10 CFR 50, Appendix B, Corrective Action Program. The corrective action process ensures that 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 repetition.

Exceptions to NUREG-1801, Element 7:

None.

Enhancements to NUREG-1801, Element 7:

None.

Comparison and Evaluation Conclusion:

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

3.7 Confirmation Process

NUREG-1801:

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.

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.

3.8 Administrative Controls

NUREG-1801:

As discussed in the appendix to this report, the staff finds the requirements of 10 CFR Part 50, Appendix B, acceptable to address the administrative controls.

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.

3.9 Operating Experience

NUREG-1801:

Since 1980, masonry walls that perform an intended function have been systematically identified through licensee programs in response to NRC IEB 80-11, USI A-46, and 10 CFR 50.48. NRC IN 87-67 documented lessons learned from the NRC IEB 80-11 program, and provided recommendations for administrative controls and periodic inspection to ensure that the evaluation basis for each safety-significant masonry wall is maintained. Whether conducted as a stand-alone program or as part of structures monitoring for MR, a masonry wall AMP that incorporates the recommendations delineated in NRC IN 87-67 should ensure that the intended functions of all masonry walls within the scope of license renewal are maintained for the period of extended operation.

Oyster Creek

Oyster Creek masonry walls that perform an intended function under 10 CFR 54.4 have been systematically identified in accordance with the scoping and screening methodology described in the LRA. The walls include walls identified in response to IEB 80-11, USI A-46, and those that perform 10 CFR 54.48 intended function. Review of industry operating experience has confirmed that cracking of masonry walls has occurred in the past (NRC IE Bulletin No. 80-11; NRC Information Notices [IN] 87-67. In response to NRC I.E. Bulletin 80-11, "Masonry Wall Design", and

Information Notice 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to I.E. Bulletin 80-11" various actions were taken. Actions included modifications of some walls and removal of others, program enhancements, follow-up inspections to substantiate masonry walls analyses and classifications, and the development of procedures for tracking and recording changes to the walls. These actions addressed all concerns raised by I.E. Bulletin 80-11 and I.N. 87-67, namely unanalyzed conditions, improper assumptions, improper classification, and lack of procedural controls.

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.), 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 cracking is being adequately managed in Masonry Wall Program. The following examples of operating experience provide objective evidence that the Masonry Wall Program is effective in assuring that intended function(s) will be maintained consistent with the CLB for the period of extended operation:

1. CAP No. O2003-0038 was issued to document and evaluate degraded and missing mortar between masonry blocks on a fire barrier masonry wall in the lower cable spreading room. The

degraded and missing mortar was evaluated by the qualified structural engineer and the fire protection engineer. The structural engineer concluded that the degraded and missing mortar does not adversely impact the masonry wall structural integrity. The fire protection engineer concluded that the observed condition of the fire barrier masonry wall does not render the barrier inoperable. Action Request #A2052336 was generated to repair the masonry wall and restore it to its design condition.

2. CAP No. O02002-0065 was issued to document and evaluate small cracks in a fire barrier masonry wall in the turbine building. The cracks were evaluated by the qualified structural engineer and the fire protection engineer. The structural engineer concluded that the small cracks have no impact on the structural integrity of the masonry wall. The fire protection engineer determined that the cracks do not impact the fire barrier intended function of the masonry wall.

The operating experience of the Masonry Wall 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 Masonry Wall Program will effectively determine cracking. Appropriate guidance for reevaluation, repair or replacement is provided for locations where cracking has occurred. Periodic self-assessments of the Masonry Wall Program are performed to identify the areas that need improvement to maintain the quality performance of the program.

3.10 Oyster Creek

The Oyster Creek Masonry Wall aging management program is credited for managing cracking of masonry walls in environments listed in Table 5.2. The Oyster Creek Masonry Wall 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 document for this aging management program is 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 Masonry Wall aging management program provides reasonable assurance that cracking will be adequately managed so that the intended functions of masonry walls 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

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

4.2.1 NRC IE. Bulletin 80-11, "Masonry Wall Design", May 8, 1980.

4.2.2 NRC Information Notice 87-67, "Lessons Learned from Regional Inspections of Licensee Actions in Response to I.E. Bulletin 80-11", December 31, 1987.

4.2.3 10 CFR Part 50, Section 65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", Office of the Federal Register, National Archives and Records Administration, 2000.

4.2.4 NRC Regulatory Guide 1.160, Revision 2, Monitoring the Effectiveness of Maintenance At Nuclear Power Plants U.S. Nuclear Regulatory Commission, March 1997.

4.2.5 NUMARC 93-01, Revision 2, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", Nuclear Management and Resources Council, Inc.

4.2.6 ACI 201, "Guide for Making a Condition Survey of Concrete in Service", ACI Journal Proceeding, Vol. 65, No. 11, November 1968.

4.2.7 ACI 531-79, "Concrete Masonry Structures, Design and Construction".

4.2.8 Unresolved Safety Issue (USA) A-46 (Generic Letter 87-02), Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors.

4.2.9 GPU Letter to NRC dated August 28, 1985, Oyster Creek Nuclear Generating Station Docket No. 50-219, IE Bulletin 80-11.

4.2.10 GPU Letter to NRC dated September 19, 1980, Oyster Creek Nuclear Generating Station Docket No. 50-219, I.E. Bulletin 80-11

4.3 Oyster Creek Program References

4.3.1 "Building Structure Monitoring Plan", procedure 125.6, Revision 3.

4.3.2 TDR 242, "Reevaluation of Safety-Related Concrete Masonry Walls, I.E. Bulletin 80-11"

4.3.3 TDR 830, Oyster Creek Masonry Block Wall Walkdowns to Verify IEB 80-11 Response

4.3.4 Mod 509.01-1, "Oyster Creek Nuclear Generating Station Boundary Supports of Masonry Walls"

4.3.5 SP 1302-53-007, "I.E. Bulletin 80-11 Block Wall Engineering Support during Construction"

4.3.6 NRC Letter dated December 23, 1985, "IE Bulletin 80-11, Masonry Wall Design"

4.3.7 UFSAR Section 3.8.4.1.9, Masonry Walls

4.3.8 LS-AA-125, "Corrective Action Program (CAP) Procedure"

4.3.9 NUREG-1382 Safety Evaluation Report Related to the full-term operating license for Oyster Creek Nuclear Generating Stations Docket No. 50-219.

4.3.10 NRC Letter dated December 23, 1985, which transmitted the Safety Evaluation by the Office of Nuclear Reactor Regulation Relating to IE Bulletin 80-11, Masonry Wall

Design GPU Nuclear Corporation Jersey Central Power and Light Company Oyster Creek Nuclear Generating Station Docket No. 50-219.

4.3.11 NRC Inspection Report No. 50-219/86-09, dated October 6, 1986.

4.3.12 Specification SP 1302-53-007, I.E. Bulletin 80-11 Block Wall Engineering Support during Construction.

4.3.13 NRC Q&A AMP-004, AMP Audit question on Masonry Walls in scope of 10 CFR 54.4

5.0 TABLES

5.1 Aging Management Program Implementing Documents

Procedure Number	Procedure Title	Commitment No.	Status
125.6	Building Structure Monitoring Plan	330592.30.01	ACC/ASG

5.2 Aging Management Review Results

SSC Name	Structure and/or Component	Material	Environment	Aging Effect
Reactor Building	Masonry Block Walls	Masonry	Indoor Air	Cracking
Turbine Building	Masonry block walls	Masonry	Indoor Air	Cracking
Office Building	Masonry block walls	Masonry	Indoor Air	Cracking
Exhaust Tunnel[1]	Masonry block wall	Masonry	Indoor Air	Cracking
Exhaust Tunnel[1]	Masonry block wall	Masonry	Outdoor Air	Cracking

[1] As discussed with NRC Staff during the AMP audit, this wall is monitored under the Masonry Wall Program even though it is not required to be in scope of 10 CFR 54.4 (a)(1), a (2), or a (3) (ref. 4.3.13)

6.0 ATTACHMENTS

6.1 LRA Appendix A

6.2 LRA Appendix B

PROGRAM BASIS DOCUMENT

PBD-AMP-B.1.05

Revision 0

BWR FEEDWATER NOZZLE

GALL PROGRAM XI.M5 - BWR FEEDWATER NOZZLE

Prepared By: ______________________

Reviewed By: ______________________

Program Owner Review: ______________________

Technical Lead Approval: ______________________

Revision History:

Revision	*Prepared by:*	*Reviewed by:*	*Program Owner:*	*Approved by:*
0	*George Beck*	*Mike May*	*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 Feedwater Nozzle aging management program that are credited for managing cracking 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; and
- 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 also provides a comparison of the credited Oyster Creek program with the elements of the corresponding NUREG-1801 Chapter XI program XI.M5, BWR Feedwater Nozzle. Project Level Instruction PLI-8 "Program Basis Documents" prescribes the methodology for evaluating Aging Management Programs. An evaluation of Oyster Creek's 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

<u>NUREG-1801:</u>

This program includes

a) *Enhanced inservice inspection (ISI) in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWB, Table IWB 2500-1 (2001 edition[1] including the 2002 and 2003 Addenda) and the recommendation of General Electric (GE) NE-523-A71-0594, and*

b) *System modifications to mitigate cracking.*

c) *The program specifies periodic ultrasonic inspection of critical regions of the boiling water reactor (BWR) feedwater nozzle.*

<u>Oyster Creek:</u>

a) The BWR Feedwater Nozzle aging management program is an existing program that provides for monitoring of feedwater nozzles for cracking through station procedures based on the 1995 Edition through 1996 Addendum of ASME Section XI, Subsection IWB, Table IWB 2500-1. Inspection of the feedwater nozzles is performed in accordance with the Oyster Creek Generating Station ISI Program Plan for the Fourth Ten-Year Inspection Interval. The program specifies periodic ultrasonic inspections of critical regions of the feedwater

[1] An applicant may rely on a different version of the ASME Code, but should justify such use. An applicant may wish to refer to the SOC for an update of 10 CFR § 50.55a to justify use of a more recent edition of the Code.

nozzles. The current program is based on NUREG-0619 as modified by the relief provided in reference 4.3.3 to allow UT examinations of the Feedwater nozzles. The inspections are performed at intervals not exceeding 10 years. **(Reference: Document Number OC-1, Section 1, paragraphs 1.6 and 2.2.2; Document Number OC-1, Section 1, Table 7.0-1, Examination Category B-D; Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*", dated October 4, 1994).**

The Oyster Creek Feedwater Nozzle aging management program will be enhanced to implement the recommendations of the BWR Owners Group Licensing Topical Report General Electric (GE) NE-523-A71-0594. These enhancements will be implemented prior to entering the period of extended operation **Reference 4.3.5 - AR#: 00330592.05).**

b) Modification to the feedwater nozzles was made at Oyster Creek in response to NUREG-0619 to mitigate or prevent thermally induced fatigue cracking **(Reference: Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*", dated October 4, 1994).**

c) The program specifies periodic ultrasonic inspection of critical regions of the boiling water reactor (BWR) feedwater nozzles at a frequency of at least once per each 10-years (120 months). **(Reference: Document Number OC-1, Section 1, paragraph 2.2.2; Document Number OC-1, Section 1, Table 7.0-1, Examination Category B-D).**

2.2 Overall NUREG-1801 Consistency

The Oyster Creek BWR Feedwater Nozzle is an existing program that is consistent with NUREG-1801 aging management program XI.M5, BWR Feedwater Nozzle with exceptions and enhancements as described in 2.3 and 2.4 below.

2.3 Summary of Exceptions to NUREG-1801

NUREG-1801, XI.M5, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda and the recommendation of GE NE-523-A71-0594. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

2.4 Summary of Enhancements to NUREG-1801

The Oyster Creek Feedwater Nozzle aging management program will be enhanced to implement the recommendations of the BWR Owners Group Licensing Topical Report General Electric (GE) NE-523-A71-0594. These enhancements will be implemented prior to entering the period of extended operation.

3.0 EVALUATIONS AND TECHNICAL BASIS

<u>Note</u>

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.

3.0 Scope of Program

<u>NUREG-1801:</u>

The program includes enhanced ISI to monitor the effects of cracking on the intended function of the component, and systems modifications to mitigate cracking.

<u>Oyster Creek:</u>

The Oyster Creek BWR Feedwater Nozzle aging management program includes UT examinations inspections of the feedwater nozzles to monitor the effects of cracking in accordance with Section XI of the ASME Code using Performance Demonstration Initiative (PDI) technology **(Reference: Document Number OC-1, Section 1, paragraph 2.2.2; Document Number OC-1, Section 1,**

Table 7.0-1, Examination Category B-D; Document Number OC-1, Section 2, paragraph 4.1.11; Section 3, Table 2.2-5; Reference ER-AA-335-1000)

The Oyster Creek Feedwater Nozzle aging management program will be enhanced to implement the recommendations of the BWR Owners Group Licensing Topical Report General Electric (GE) NE-523-A71-0594. These enhancements will be implemented prior to entering the period of extended operation **(Reference: 4.3.5 AR#: 00330592.05).**

In response to NUREG-0619, modifications were made to the feedwater nozzles at Oyster Creek to mitigate or prevent thermally induced fatigue cracking **(Reference: Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*", dated October 4, 1994).**

The Oyster Creek BWR Feedwater Nozzle aging management program manages the aging effect of cracking 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.

<u>Exceptions to NUREG-1801, Element 1:</u>

NUREG-1801, XI.M5, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda and the recommendation of GE NE-523-A71-0594. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

<u>Enhancements to NUREG-1801, Element 1:</u>

The Oyster Creek Feedwater Nozzle aging management program will be enhanced to implement the recommendations of the BWR Owners Group Licensing Topical Report General Electric (GE) NE-

523-A71-0594. These enhancements will be implemented prior to entering the period of extended operation.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 1, Scope of Program with the enhancements as described above.

3.1 Preventive Actions

NUREG-1801:

a) *Mitigation occurs by systems modifications, such as removal of stainless steel cladding and installation of improved spargers.*

b) *Mitigation is also accomplished by changes to plant-operating procedures, such as improved feedwater control and rerouting of the reactor water cleanup system, to decrease the magnitude and frequency of temperature fluctuations.*

Oyster Creek:

a) In response to NUREG-0619 issues, modifications were made to the feedwater nozzles at Oyster Creek to mitigate or prevent thermally induced fatigue cracking. A new improved feedwater sparger and thermal sleeve design was installed and the cladding was removed from nozzle blend radius in 1977. **(Reference: Letter P. F. McKee (NRC) to J. J. Barton (GPU), "*Evaluation of the Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751)*" dated October 4, 1994; Transitional Operating Report, Narrative Summary of Operating Experience January 1, 1977 through December 31, 1977).**

b) At Oyster Creek changes were made to feedwater flow control system to improve system performance and reduce temperature fluctuations during low power operation **(Reference: DRF 88021, Modification Design Description for Main Feedwater Line Block Valve Addition).** Rerouting of the reactor water cleanup system was not performed at Oyster Creek.

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.

3.2 Parameters Monitored or Inspected

NUREG-1801:

The aging management program (AMP) monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by ISI in accordance with ASME Section XI, Subsection IWB and the recommendation of GE NE-523-A71-0594.

Oyster Creek:

The Oyster Creek BWR Feedwater Nozzle aging management program monitors the effects of cracking on the intended function of the feedwater nozzles by detecting and sizing cracks by ISI in accordance with Subsection IWB, Table IWB 2500-1 of the 1995 Edition through 1996 Addendum of ASME Section XI "Rules for Inservice Inspection of Nuclear Power Plant Components. The Oyster Creek Feedwater Nozzle aging management program will be enhanced to implement the recommendations of the BWR Owners Group Licensing Topical Report General Electric (GE) NE-523-A71-0594. These enhancements will be implemented prior to entering the period of extended operation. **(Reference: Document Number OC-1, Section 1, paragraphs 1.6 and 2.2.2; Document Number OC-1, Section 1, Table 7.0-1, Examination Category B-D; Document Number OC-1, Section 2, paragraph 4.1.11; Reference 4.3.5 - AR#: 00330592.05).**

Exceptions to NUREG-1801, Element 3:

NUREG-1801, XI.M5, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda and the recommendation of GE NE-523-A71-0594. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

Enhancements to NUREG-1801, Element 3:

The Oyster Creek Feedwater Nozzle aging management program will be enhanced to implement the recommendations of the BWR Owners Group Licensing Topical Report General Electric (GE) NE-523-A71-0594. These enhancements will be implemented prior to entering the period of extended operation.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 3, Parameters Monitored or Inspected with the enhancements as described above.

3.3 Detection of Aging Effects

NUREG-1801:

The extent and schedule of the inspection prescribed by the program are designed to ensure that aging effects will be discovered and repaired before the loss of intended function of the component. Inspection can reveal cracking.

a) GE NE-523-A71-0594 specifies ultrasonic testing (UT) of specific regions of the blend radius and bore. The UT examination techniques and personnel qualifications are in accordance with the guidelines of GE NE-523-A71-0594.

b) Based on the inspection method and techniques and plant-specific fracture mechanics assessments, the inspection schedule is in accordance with Table 6-1 of GE NE-523-A71-0594.

c) Leakage monitoring may be used to modify the inspection interval.

Oyster Creek:

a) Inspection of the feedwater nozzles is performed in accordance with the Oyster Creek Generating Station ISI Program Plan for the Fourth Ten-Year Inspection Interval. The program includes ultrasonic (UT) examination techniques and personal qualifications. The Oyster Creek Feedwater Nozzle aging management program will be enhanced to implement the recommendations of the BWR Owners Group Licensing Topical Report General Electric (GE) NE-523-A71-0594, including UT examination techniques and personnel qualifications. These enhancements will be implemented prior to entering the period of extended operation. **(Reference: Document Number OC-1, Section 1, paragraph 2.2.2; Document Number OC-1, Section 1, Table 7.0-1, Examination Category B-D; Document Number OC-1, Section 2, paragraph 4.1.11).**

b) The Oyster Creek Feedwater Nozzle aging management program will be enhanced to implement the recommendations of the BWR Owners Group Licensing Topical Report General Electric (GE) NE-523-A71-0594. Based on the inspection method and techniques and plant specific fracture mechanics assessments, the inspection schedule will be in accordance with Table 6-1 of GE NE-523-A71-0594. These enhancements will be implemented prior to entering the period of extended operation. **(Reference: Document Number OC-1, Section 1, paragraph 2.2.2; Document Number OC-1, Section 1, Table 7.0-1, Examination Category B-D; Document Number OC-1, Section 2, paragraph 4.1.11).**

c) BWR Owners Group Licensing Topical Report General Electric (GE) NE-523-A71-0594 notes that for plants that have thermal sleeve bypass leakage detection systems leakage data obtained from these systems can be used to further enhance the argument used to establish inspection frequency **(Reference 4.2.2. Section 5.5).** Oyster Creek does not have a thermal sleeve bypass leakage detection system and the inspection interval has not been modified based on leakage data.

Exceptions to NUREG-1801, Element 4:

NUREG-1801, XI.M5, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda and the recommendation of GE NE-523-A71-0594. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

Enhancements to NUREG-1801, Element 4:

The Oyster Creek Feedwater Nozzle aging management program will be enhanced to implement the recommendations of the BWR Owners Group Licensing Topical Report General Electric (GE) NE-523-A71-0594. These enhancements will be implemented prior to entering the period of extended operation.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, *Element* 4, *Detection of Aging Effects with the enhancements as described above.*

3.4 Monitoring and Trending

NUREG-1801:

Inspections scheduled in accordance with GE NE-523-A71-0594 provide timely detection of cracks.

Oyster Creek:

a) The feedwater nozzles are included in the Oyster Creek ISI program plan under Category B-D, "Full Penetration Welds of Nozzles in Vessels," consistent with the requirements of Table IWB 2500-1. Periodic feedwater nozzle inspections are performed using UT techniques at least once every 10-year period (120 months). The Oyster Creek Feedwater Nozzle aging management program inspection schedule will be enhanced to implement the recommendations of the BWR Owners Group Licensing Topical Report General Electric (GE) NE-523-A71-0594 to provide timely detection of cracks. These enhancements will be implemented prior to entering the period of extended operation **(Reference: Document Number OC-1, Section 1, paragraph 2.2.2; Document Number OC-1, Section 1, Table 7.0-1, Examination Category B-D; Document Number OC-1, Section 2, paragraph 4.1.11; Reference 4.3.5 -AR#: 00330592.05).**

Exceptions to NUREG-1801, Element 5:

NUREG-1801, XI.M5, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda and the recommendation of GE NE-523-A71-0594. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

Enhancements to NUREG-1801, Element 5:

The Oyster Creek Feedwater Nozzle aging management program will be enhanced to implement the recommendations of the BWR Owners Group Licensing Topical Report General Electric (GE) NE-523-A71-0594. These enhancements will be implemented prior to entering the period of extended operation.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 5, Monitoring and Trending with the enhancements as described above.

3.5 Acceptance Criteria

NUREG-1801:

Any cracking is evaluated in accordance with IWB-3100 by comparing inspection results with the acceptance standards of IWB-3400 and IWB-3500.

Oyster Creek:

OC procedures follow ASME Section XI for evaluating flaw indications. Flaw indications are evaluated in accordance with the guidelines of ASME Section XI IWB-3100, using the acceptance standards of IWB-3512 as directed by IWB-3410 **(Reference: Procedure ER-AA-330-002, paragraph 4.10; ASME Section XI, Table IWB-2500-1, Examination Category B-D, Item Nos. B3.90 and B3.100).**

Exceptions to NUREG-1801, Element 6:

NUREG-1801, XI.M5, specifies the 2001 ASME Section XI B&PV Code, including the 2002 and 2003 Addenda and the recommendation of GE NE-523-A71-0594. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

Enhancements to NUREG-1801, Element 6:

None.

Comparison and Evaluation Conclusion:

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

3.6 Corrective Actions

NUREG-1801:

Repair is performed in conformance with IWB-4000 and replacement in accordance with IWB-7000. 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:
If the flaw exceeds the requirements of IWB-3600, repair and replacement is performed consistent with the requirements of ASME Section XI Subsection IWA-4000. In the 1995 edition of ASME Section XI Sections IWB-4000 and IWB-7000 have been deleted and their requirements placed in IWA-4000 **(Reference: Procedure ER-AA-330-002, paragraph 4.12.4; Procedure ER-AA-330-009, paragraph 1.2.3).**

Evaluations are performed for examination results that do not satisfy the acceptance standards of IWB-3500 and an Issue Report is initiated to document the concern in accordance with plant administrative procedures **(Reference: Procedure ER-AA-330-002, paragraph 4.12).** The 10 CFR Part 50, Appendix B corrective action process ensures that 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.

Exceptions to NUREG-1801, Element 7:

NUREG-1801, XI.M5, specifies the 2001 ASME Section XI B&PV

Code, including the 2002 and 2003 Addenda and the recommendation of GE NE-523-A71-0594. The current Oyster Creek ISI Program Plan for the fourth ten-year inspection interval effective from October 15, 2002 through October 14, 2012, approved per 10CFR50.55a, is based on the 1995 ASME Section XI B&PV Code, including 1996 addenda. The next 120-month inspection interval for Oyster Creek will incorporate the requirements specified in the version of the ASME Code incorporated into 10 CFR 50.55a twelve months before the start of the inspection interval.

Enhancements to NUREG-1801, Element 7:

None.

Comparison and Evaluation Conclusion:

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

3.7 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 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.

3.8 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.

3.9 Operating Experience

NUREG-1801:

Cracking has occurred in several BWR plants (NUREG-0619, NRC Generic Letter 81-11). This AMP has been implemented for nearly 25 years and has been found to be effective in managing the effect of cracking on the intended function of feedwater nozzles.

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.), 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.

BWR Feedwater Nozzle inspections have been implemented through the station ISI program plan, which incorporates the requirements of the ASME Code. Augmented inspections were performed in accordance with NUREG-0619, "BWR Feedwater Nozzle and Control Rod Return Line Nozzle " as modified in accordance with reference 4.3.3.

Demonstration that the effects of aging are effectively managed is achieved through objective evidence that shows that cracking is being adequately managed in feedwater nozzles. The following examples of operating experience provide objective evidence that the BWR Feedwater Nozzle aging management program is effective in assuring that intended function(s) will be maintained consistent with the CLB for the period of extended operation.

Oyster Creek inspected the feedwater nozzles in 1977 in response to industry experience. Cracks were found in the nozzles and repaired. To minimize thermal cycling and fatigue induced cracking, the thermal sleeves were modified with a piston type design. Subsequent inspections, the most recent in 2000, have found no indications in the feedwater nozzles. This provides objective evidence that the modifications have been effective in mitigating the effects of thermal fatigue on the feedwater nozzles.

As outlined in Reference 4.2.2, "Since 1980 significant field experience without the presence of additional fatigue cracking in feedwater and CRD return line nozzles has been accumulated." The operating experience at Oyster Creek is consistent with the experience throughout the industry. For greater than 25 years the Oyster Creek BWR Feedwater Nozzle aging management program has been effective in managing the effect of cracking on the intended function of the feedwater nozzles.

The BWR Feedwater Nozzle aging management activities will provide timely detection of aging degradation prior to loss of system or component functions for the period of extended operation.

3.10 Conclusion

The Oyster Creek BWR Feedwater Nozzle aging management program is credited for managing cracking for the systems, components, and environments listed in Table 5.2. The Oyster Creek BWR Feedwater Nozzle 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 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 Feedwater Nozzle aging management program provides reasonable assurance that cracking 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 NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*

4.2.2 GE NE-523-A71-0594, Revision 1, *Alternate BWR Feedwater Nozzle Inspection Program*

4.3 Oyster Creek Program References

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

4.3.2 Document Number OC-3, *ISI Repair/Replacement Program, Oyster Creek Nuclear Generating Station*

4.3.3 Letter P. F. McKee (NRC) to J. J. Barton (GPU), *Evaluation of Request for Relief From NUREG-0619 for Oyster Creek Nuclear Generating Station (TAC No. M85751),* Dated October 4, 1994

4.3.4 GE NE-B13-02064-00-21, *Safety Assessment of Feedwater Spargers in Light of 18R IVVI Findings, Oyster Creek Nuclear Generating Station,* November 2000

4.3.5 Assignment Report, AR#: 00330592.05, Subject/Description: A.1.05 Commitment (BWR Feedwater Nozzle)

4.3.6 ER-AA-335-1000, *Nondestructive Examination (NDE) Program, Revision 3*

5.0 TABLES

5.1 Aging Management Program Implementing Documents

Procedure Number	Procedure Title	Commitment No.	Status
ER-AA-330-002	Inservice Inspection of Section XI Welds and Components	330592.05.02	ACC/ASG
ER-AA-330-009	ASME Section XI Repair-Replacement Program	330592.05.01	ACC/ASG
OC-1	ISI Program Plan Fourth Ten-Year Inspection Interval	330592.05.03	ACC/ASG
OC-3	Section XI Repair/Replacement Program	330592.05.04	ACC/ASG

5.2 Aging Management Review Results

SSC Name	Structure and/or Component	Material	Environment	Aging Effect
Reactor Pressure Vessel	Nozzles (Feedwater)	Carbon and low alloy steel	Treated Water (Internal)	Cracking Initiation and Growth
Reactor Pressure Vessel	Nozzle Thermal Sleeves (Feedwater Nozzle)	Nickel Alloy	Treated Water	Cracking Initiation and Growth

6.0 ATTACHMENTS

6.1 LRA Appendix A

6.2 LRA Appendix B