From: <George.Beck@exeloncorp.com>
To: <dja1@nrc.gov>, <gvc@nrc.gov>
Date: 12/16/2005 1:42:48 PM
Subject: Batch Five - 3 o f 7 PBDs due 12/19/05

Donnie/Greg,

Here are three of the seven AMP PBDs that we had indicated we would provide by Monday, 12/19 (Batch 5). This brings to 31 the number of upgraded program basis documents that we have provided for the Auditor's review. The other four from Batch 5 will be provided on Monday, 12/19.

Attached please find the following four PBDs in Word format:
PBD B.3.02 EQ, Rev. 0
PBD B.1.10 Cast Austenitic Stainless Steel, Rev. 0
PBD B.1.04 Vessel Attachment welds, Rev 0

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 four documents of Batch 5, we 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 5, and list those documents.

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

George
610-765-5631

<<PBD B.3.02 EQ Rev. 0.doc>>
<<PBD B.1.10 Cast Austenitic Stainless Steel, Rev. 0.doc>>
<<PBD B.1.04 Vessel ID Attachment welds Rev 0.doc>>

CC: <fred.polaski@exeloncorp.com>, <donald.warfel@exeloncorp.com>, <john.hufnagel@exeloncorp.com>

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MESSAGE	2128	16 December, 2005 1:42:20 PM
TEXT.htm	3298	
PBD B.3.02 EQ Rev. 0.doc	142848	
PBD B.1.10 Cast Austenitic Stainless Steel, Rev. 0.doc	134144	
PBD B.1.04 Vessel ID Attachment welds Rev 0.doc	142336	
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Expiration Date: None
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Security: Standard

PROGRAM BASIS DOCUMENT

PBD-AMP-B.3.02

Revision 0

ENVIRONMENTAL QUALIFICATION (EQ) PROGRAM

GALL PROGRAM X.E1 - ENVIRONMENTAL QUALIFICATION (EQ) PROGRAM

Prepared By: M. J. May

Reviewed By: G. J. Beck

Program Owner Review: Art Hertz

Technical Lead Approval: F. W. Polaski

Revision History:

Revision	*Prepared by:*	*Reviewed by:*	*Program Owner:*	*Approved by:*
0	M. J. May	G. J. Beck	Art Hertz	F. W. 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 Environmental Qualification (EQ) aging management program that are credited for managing effects of aging on electrical components important to safety in harsh environments 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 X 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.E1, Environmental Qualification of Electrical Components. 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:

The Nuclear Regulatory Commission (NRC) has established nuclear station environmental qualification (EQ) requirements in 10 CFR Part 50, Appendix A, Criterion 4, and 10 CFR 50.49. 10 CFR 50.49 specifically requires that an EQ program be established to demonstrate that certain electrical components located in harsh plant environments (that is, those areas of the plant that could be subject to the harsh environmental effects of a loss of coolant accident [LOCA], high energy line breaks [HELBs] or post-LOCA environment) are qualified to perform their safety function in those harsh environments after the effects of inservice aging. 10 CFR 50.49 requires that the effects of significant aging mechanisms be addressed as part of environmental qualification.

a) *All operating plants must meet the requirements of 10 CFR 50.49 for certain electrical components important to safety. 10 CFR 50.49 defines the scope of components to be included, requires the preparation and maintenance of a list of in-scope components, and*

b) *requires the preparation and maintenance of a qualification file that includes component performance specifications, electrical characteristics, and the environmental conditions to which the components could be subjected.*

c) *10 CFR 50.49(e)(5) contains provisions for aging that require, in part, consideration of all significant types of aging degradation that* *can affect component functional capability.*

d) *10 CFR 50.49(e) also requires replacement or refurbishment of components not qualified for the current license term prior to the end of designated life, unless additional life is established through ongoing qualification. 10 CFR 50.49(f) establishes four methods of demonstrating qualification for aging and accident conditions.*

e) *10 CFR 50.49(k) and (l) permit different qualification criteria to apply based on plant and component vintage. Supplemental EQ regulatory guidance for compliance with these different qualification criteria is provided in the DOR Guidelines, Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors; NUREG-0588, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment; and Regulatory Guide 1.89, Rev. 1, Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants. Compliance with 10 CFR 50.49 provides reasonable assurance that the component can perform its intended functions during accident conditions after experiencing the effects of inservice aging.*

EQ programs manage component thermal, radiation, and cyclical aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, EQ components not qualified for the current license term are to be refurbished, replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation. Aging evaluations for EQ components that specify a qualification of at least 40 years are considered time-limited aging analyses (TLAAs) for license renewal.

Under 10 CFR 54.21(c)(1)(iii), plant EQ programs, which implement the requirements of 10 CFR 50.49 (as further defined and clarified by the DOR Guidelines, NUREG-0588, and Regulatory Guide 1.89, Rev. 1), are viewed as aging management programs (AMPs) for license renewal. Reanalysis of an aging evaluation to extend the qualification of components under 10 CFR 50.49(e) is performed on a routine basis as part of an EQ program.

f) *Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). These attributes are discussed in the "EQ Component Reanalysis Attributes" section.*

This reanalysis program can be applied to EQ components now qualified for the current operating term (i.e., those components now qualified for 40 years or more). As evaluated below, this is an acceptable AMP. Thus, no further evaluation is recommended for license renewal if an applicant elects this option under 10 CFR 54.21(c)(1)(iii) to evaluate the TLAA of EQ of electric equipment.

<u>Oyster Creek:</u>

The Oyster Creek Environmental Qualification (EQ) Program is an existing program that, as described in the respective UFSAR, Section 3.11, meets the requirements of the Code of Federal Regulations, Title 10 Part 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants." The Oyster Creek EQ program has been established to demonstrate that certain electrical components located in harsh plant environments are qualified to perform their safety function in those harsh environments after the effects of inservice aging. As required by 10 CFR 50.49 the program addresses the effects of significant aging mechanisms as part of environmental qualification **(References: UFSAR Section 3.11; CC-AA-203, paragraph 1.1).**

a) The Oyster Creek Environmental Qualification (EQ) Program, as described in the respective UFSAR, Section 3.11, meets the requirements of the Code of Federal Regulations, Title 10 Part 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants." The EQ components important to safety are identified in the controlled plant Component Record List (CRL) database and site EQ documentation files (EQ Binders). The station EQ Program requirements for maintaining qualification of specific components are included in the EQ Binder for those components **(Reference: CC-AA-203, paragraphs 2.14 & 4.2).**

b) Environmental qualification auditable documentation files that include component performance specifications, electrical characteristics, and the environmental conditions to which the components could be subjected are maintained for all components within the scope of 10 CFR 50.49. The EQ auditable documentation files demonstrate the equipment qualified life and identify replacement or refurbishment of components not qualified for the current license term **(Reference: CC-AA-203, paragraph**

4.2 & 4.3).

c) In accordance with 10 CFR 50.49(e)(5) the EQ program contains provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. An aging limit (qualified life) is established for equipment subject to significant thermal, radiation and cyclic aging mechanism **(Reference: CC-MA-203-1001, Exhibit 4).**

d) Additionally, in accordance with 10 CFR 50.49(e) the Oyster Creek EQ program provides for appropriate actions, such as reanalysis or replacement of the equipment, to be taken prior to or at the end of the equipment qualified life. Reanalysis of an aging evaluation is used to extend the qualification of the component where possible. If the qualification cannot be extended by reanalysis, the component is refurbished, replaced, or re-qualified prior to exceeding the period for which the current qualification remains valid. A reanalysis is performed in a timely manner (that is, the corrective action is performed to ensure the qualified life of the component is not exceeded if the reanalysis is unsuccessful). The requirements for replacement, refurbishment, or requalification are documented in the EQ binder for that component **(Reference: CC-AA-203, paragraph 4.1.6; CC-MA-203-1001, Exhibit 4).**

e) The Oyster Creek Environmental Qualification (EQ) Program, as described in the respective UFSAR, Section 3.11, meets the requirements of the Code of Federal Regulations, Title 10 Part 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants". Equipment within the scope of the Oyster Creek has evaluated for compliance with the DOR Guidelines NUREG 0588 Category I, or 10CFR50.49 with guidance from RG 1.89. By compliance with these EQ rules the Oyster Creek Environmental Qualification (EQ) Program provides reasonable assurance that the component can perform its intended functions during accident conditions after experiencing the effects of inservice aging **(Reference: UFSAR, Section 3.11).**

The Oyster Creek EQ Program manages component thermal, radiation, and cyclical aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. The Oyster Creek EQ Program requires replacement or refurbishment of components not qualified for the current license term prior to the end of designated life, unless additional life is established through ongoing qualification. An aging limit (qualified life) is established for equipment subject to significant thermal, radiation and cyclic aging mechanism. The program is structured to qualify components using the methods described in 10 CFR 50.49(f), which include testing and analysis. An appropriate action such as replacement or refurbishment of the equipment is taken prior to or at the end of the equipment qualified life. Environmental qualification auditable documentation files that include component performance specifications, electrical characteristics, and the environmental conditions to which the components could be subjected are maintained for all components within the scope of 10 CFR 50.49 **(Reference: CC-MA-203-1001 Exhibit 4).**

EQ components that specify a qualification of at least 40 years are considered time-limited aging analyses (TLAAs) for license renewal. The Oyster Creek EQ program is the aging management program (AMP) used to manage EQ components in the period of extended operation in accordance with 10 CFR 54.21(c)(1)(iii). **(Reference: Oyster Creek LRA Section 4.4).**

f) Reanalysis of an aging evaluation to extend the qualification of components under 10 CFR 50.49(e) is performed as part of the Oyster Creek EQ program. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met) **(Reference: CC-MA-203-1001, Exhibit 4; CC-AA-203, paragraph 4.2).**

EQ Component Reanalysis Attributes

a) *Analytical Methods: The reanalysis of an aging evaluation is normally performed to extend the qualification by reducing excess conservatism incorporated in the prior evaluation. Reanalysis of an aging evaluation to extend the qualification of a component is performed on a routine basis pursuant to 10 CFR 50.49(e) as part of an EQ program. While a component life limiting condition may be due to thermal, radiation, or cyclical aging, the vast majority of component aging limits are based on thermal conditions. Conservatism may exist in aging evaluation parameters, such as the assumed ambient temperature of the component, an unrealistically low activation energy, or in the application of a component (de-energized versus energized). The reanalysis of an aging evaluation is documented according to the station's quality assurance program requirements, which requires the verification of assumptions and conclusions. As already noted, important attributes of a reanalysis include analytical methods, data collection* *and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met). These attributes are discussed below.*

b) *The analytical models used in the reanalysis of an aging evaluation are the same as those previously applied during the prior evaluation. The Arrhenius methodology is an acceptable thermal model for performing a thermal aging evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose).*

c) *For license renewal, one acceptable method of establishing the 60-year normal radiation dose is to multiply the 40-year normal radiation dose by 1.5 (that is, 60 years/40 years). The result is added to the accident radiation dose to obtain the total integrated dose for the component. For cyclical aging, a similar approach may be used. Other models may be justified on a case-by-case basis*

d) *Data collection and Reduction Methods: Reducing excess conservatism in the component service conditions (for example, temperature, radiation, cycles) used in the prior aging evaluation is the chief method used for a reanalysis. Temperature data used in an aging evaluation is to be conservative and based on plant design temperatures or on actual plant temperature data.*

e) *When used, plant temperature data can be obtained in several ways, including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, and temperature sensors on large motors (while the motor is not running).*

f) *A representative number of temperature measurements are conservatively evaluated to establish the temperatures used in an aging evaluation.*

g) *Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation, or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation.*

h) *Any changes to material activation energy values as part of a reanalysis are to be justified on a plant-specific basis.*

i) *Similar methods of reducing excess conservatism in the component service conditions used in prior aging evaluations can be used for radiation and cyclical aging.*

j) *Underlying Assumptions: EQ component aging evaluations contain* *sufficient conservatism to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the normal operating environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.*

k) *Acceptance Criteria and Corrective Actions: The reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component is to be refurbished, replaced, or requalified prior to exceeding the period for which the current qualification remains valid. A reanalysis is to be performed in a timely manner (that is, sufficient time is available to refurbish, replace, or requalify the component if the reanalysis is unsuccessful).*

Oyster Creek:

a) The reanalysis of an aging evaluation may be performed to extend the qualification by reducing excess conservatism incorporated in the prior evaluation. Reanalysis of an aging evaluation to extend the qualification of a component is performed based on 10 CFR 50.49(e) as part of an EQ program. While a component life limiting condition may be due to thermal, radiation, or cyclical aging, the vast majority of component aging limits are based on thermal conditions. Conservatisms frequently exist in aging evaluation parameters, such as the assumed ambient temperature of the component, unrealistically low activation energy, or in the assumed operation of a component. These conservatisms often can be replaced with more realistic assumptions to help extend the qualified life of a component. Such changes in assumptions and conservatism are documented in the EQ binder for the component. The reanalysis of an aging evaluation is documented according to the station's quality assurance program requirements, which requires the verification of assumptions and conclusions. The important attributes of reanalysis would include analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, and corrective actions (if acceptance criteria are not met), as discussed below **(Reference: CC-MA-203-1001, Exhibit 4)**.

b) *Analytical Methods:* The analytical models used in the reanalysis of an aging evaluation are the same as those previously applied during the prior evaluation. The Arrhenius methodology is the model used in the Oyster Creek EQ Program for performing a thermal aging evaluation.

The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose) **(Reference: CC-MA-203-1001, Exhibit 4, SRS 5 and SRS 6)**

c) For license renewal, the method used in the Oyster Creek EQ Program of establishing the 60-year normal radiation dose is to multiply the 40-year normal radiation dose by 1.5 (that is, 60 years/40 years). The result is added to the accident radiation dose to obtain the total integrated dose for the component. For cyclical aging, a similar approach may be used. Other models may be justified on a case-by-case basis. The approach is documented in each EQ binder and in the design change records **(e.g. ECR 04-00795 see reference: 4.3.2).**

d) *Data Collection and Reduction Methods*: In the Oyster Creek EQ program reducing excess conservatism in the component service conditions (for example, temperature, radiation, cycles) used in the prior aging evaluation is the chief method used for a reanalysis. The temperature data used in aging evaluations are based on plant design temperatures **(Reference: 4.3.1).** If needed, the program allows plant temperature data to be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation, or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation. Any changes to material activation energy values as part of a reanalysis are justified on a case-specific basis. Similar methods of reducing excess conservatism in the component service conditions used in prior aging evaluations can be used for radiation and cyclical aging. Any changes to assumptions or methods are justified in the component EQ binder **(Reference: CC-MA-203-1001, Exhibit 4).**

e) Temperature data used in an aging evaluation is based on plant design temperatures **(Reference: 4.3.1).** When used, the Oyster Creek EQ Program allows plant temperature data to be obtained in several ways, including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, or temperature sensors on equipment.

f) If plant data were used, the Oyster Creek Program would conservatively evaluate a representative a representative number of temperature measurements to establish the temperatures to be used in an aging evaluation.

g) If used, the program would allow plant temperature data to be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation, or (b) using the plant temperature data to verify the temperatures that are used for an evaluation. This approach would be documented in the component EQ binder.

h) Changes to material activation energy values as part of a reanalysis are justified on a case-specific basis and are documented in the component EQ binder **(Reference: CC-MA-203-1001, Exhibit 4, SRS-5).**

i) Similar methods of reducing excess conservatism in the component service conditions that were used in prior aging evaluations, can also be used for radiation and cyclical aging. Any such changes are documented in the component EQ binder **(Reference: CC-MA-203-1001, Exhibit 4, SRS 6 & SRS 7).**

j) *Underlying Assumptions:* EQ component aging evaluations contain conservatism to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the normal operating environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions **(Reference: CC-AA-203, paragraph 4.1).**

k) *Acceptance Criteria and Corrective Actions:* The reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component is refurbished, replaced, or re-qualified prior to exceeding the period for which the current qualification remains valid. A reanalysis is performed in a timely manner (that is, sufficient time is available to refurbish, replace, or re-qualify the component if the reanalysis is unsuccessful) **(Reference: CC-AA-203, paragraph 4.1).**

2.2 Overall NUREG-1801 Consistency

The Oyster Creek Environmental Qualification (EQ) Program is an existing program that is consistent with NUREG-1801 aging management program X.E1, "Environmental Qualification (EQ) of Electrical Components".

2.3 Summary of Exceptions to NUREG-1801

None. The existing Oyster Creek Environmental Qualification (EQ) Program aging management program is found to be adequate to support the extended period of operation with no exceptions.

2.4 Summary of Enhancements to NUREG-1801

None. The existing Oyster Creek Environmental Qualification (EQ) 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:

EQ programs apply to certain electrical components that are important to safety and could be exposed to harsh environment accident conditions, as defined in 10 CFR 50.49 and Regulatory Guide 1.89, Rev.1.

Oyster Creek:

The Oyster Creek Environmental Qualification Program *is an* existing program that manages the qualification of electrical components that are important to safety and could be exposed to harsh environment accident conditions, as defined in the Code of Federal Regulations, Title 10 Part 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants," and Regulatory Guide 1.89. The EQ components are identified in the controlled plant Component Record List (CRL) database and site EQ documentation files (EQ Binders). The station EQ Program requirements for maintaining qualification of specific components are included in the EQ Binder for those components. **(References: UFSAR Section 3.11.1; CC-AA-203, paragraph 4.1).**

The Oyster Creek Environmental Qualification Program manages the aging effects of thermal, radiation, and cyclical operation on electrical components that are important to safety due to exposure to harsh environments of radiation and high temperature. 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:

10 CFR 50.49 does not require actions that prevent aging effects. EQ program actions that could be viewed as preventive actions include (a) establishing the component service condition tolerance and aging limits (for example, qualified life or condition limit) and (b) where applicable, requiring specific installation, inspection, monitoring or periodic maintenance actions to maintain component aging effects within the bounds of the qualification basis.

Oyster Creek:

While not required by 10 CFR 50.49, the Oyster Creek Environmental Qualification Program includes actions that could be viewed as preventive. These actions include establishing margin in the component service condition tolerance and aging limits (qualified life) and establishing a schedule for replacement where needed. The EQ Program establishes specific installation requirements, surveillances (inspections or monitoring) or periodic maintenance actions to maintain environmental qualification of those components. These requirements are specified in the station EQ Binders. **(Reference: CC-MA-203-1001, Exhibit 4; MA-MA-716-009, paragraph 4.2.2 & 4.6.2).**

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:

EQ component qualified life is not based on condition or performance monitoring. However, pursuant to Regulatory Guide 1.89, Rev. 1, such monitoring programs are an acceptable basis to modify a qualified life through reanalysis. Monitoring or inspection of certain environmental conditions or component parameters may be used to ensure that the component is within the bounds of its qualification basis, or as a means to modify the qualified life.

Oyster Creek:

The Code of Federal Regulations, Title 10 Part 50.49, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants" does not require condition or performance monitoring to maintain the qualified life of equipment. However, the Oyster Creek EQ Program allows for monitoring or inspection of certain environmental conditions or component parameters if required for qualification. The maintenance and surveillance requirements are specified in the station EQ binders **(Reference: CC-AA-203-1001, Exhibit 4).**

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) *10 CFR 50.49 does not require the detection of aging effects for in-service components. Monitoring or inspection of certain environmental conditions or component parameters may be used to ensure that the component is within the bounds of its qualification basis, or as a means to modify the qualified life.*

Oyster Creek:

The Oyster Creek Environmental Qualification Program does not monitor for detection of aging effects through surveillance and maintenance activities. If required for qualification, surveillance or maintenance activities could be performed within the EQ Program. Such activities would be specified in the component EQ binder.

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:

10 CFR 50.49 does not require monitoring and trending of component condition or performance parameters of in-service components to manage the effects of aging. EQ program actions that could be viewed as monitoring include monitoring how long qualified components have been installed. Monitoring or inspection of certain environmental, condition, or component parameters may be used to ensure that a component is within the bounds of its qualification basis, or as a means to modify the qualification.

Oyster Creek:

The Oyster Creek EQ Program does not require monitoring and trending of component condition or performance parameters of in-service components to manage the effects of aging. If required, surveillance (monitoring and inspections) of component parameters and performance characteristics could be used to ensure that components are maintained in the qualified (EQ) status. These activities would be specified in the component EQ binder.

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) *10 CFR 50.49 acceptance criteria are that an inservice EQ component is maintained within the bounds of its qualification basis, including (a) its established qualified life and (b) continued qualification for the projected accident conditions. 10 CFR 50.49 requires refurbishment, replacement, or requalification prior to exceeding the qualified life of each installed device.*

b) *When monitoring is used to modify a component qualified life, plant-specific acceptance criteria are established based on applicable 10 CFR 50.49(f) qualification method.*

Oyster Creek:

a) The Oyster Creek EQ Program ensures that EQ components are replaced, refurbished, or requalified prior to exceeding qualified life of each installed device. The component qualified life is established in EQ evaluations, which are included in the EQ Binders. The EQ Binders contain requirements for maintenance, surveillance, and installation of equipment that must be implemented to achieve and maintain the component EQ qualified status. Component replacement (prior to or at the end of its qualified life) is performed in accordance with established component replacement schedules **(Reference: CC-AA-203, 4.2; MA-MA-716-009, paragraph 4.2.2 & 4.6.2).**

b) If monitoring is used to modify a component qualified life, plant-specific acceptance criteria are established based on applicable 10 CFR 50.49(f) qualification methods and the component qualified life is modified accordingly. The bases for requalification are maintained within the component EQ binder.

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:

a) *If an EQ component is found to be outside the bounds of its qualification basis, corrective actions are implemented in accordance with the station's corrective action program.*

b) *When unexpected adverse conditions are identified during operational or maintenance activities that affect the environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.*

c) *When an emerging industry aging issue is identified that affects the qualification of an EQ component, the affected component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.*

d) *Confirmatory actions, as needed, are implemented as part of the station's corrective action program, pursuant to 10 CFR 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 corrective actions.*

Oyster Creek:

a) Corrective actions are implemented for EQ components found to be outside the bounds of its qualification basis in accordance with the station's corrective action program **(Reference: LS-AA-125; CC-AA-203, paragraph 4.1).**

b) When unexpected adverse conditions are identified during operational or maintenance activities that affect the environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken. These actions include changing the qualification bases and conclusions, and reanalysis of the component qualified life. Evaluations are performed for EQ issues that include departure from design requirements, departure from required maintenance or surveillance actions, or deficiency in documentation that supports qualification Additionally, actions could be taken to mitigate the adverse conditions, or replace or refurbish the affected component **(Reference: CC-AA-203, paragraph 4.1).**

c) When an emerging industry issue is identified that affects the qualification of an EQ component, the affected component is evaluated for Oyster Creek applicability **(Reference: LS-AA-115; CC-AA-203, paragraph 4.1)**.

d) The Oyster Creek 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.

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:

Confirmatory actions, as needed, are implemented as part of the station's corrective action program, pursuant to 10 CFR 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.

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:

a) *EQ programs are implemented through the use of station policy, directives, and procedures. EQ programs will continue to comply with 10 CFR 50.49 throughout the renewal period, including development and maintenance of qualification documentation demonstrating reasonable assurance that a component can perform required functions during harsh accident conditions.*

b) *EQ program documents identify the applicable environmental conditions for the component locations.*

c) *EQ program qualification files are maintained at the plant site in an auditable form for the duration of the installed life of the component.*

d) *EQ program documentation is controlled under the station's quality assurance program. 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:

a) The Oyster Creek EQ Program is implemented through the use of station policy, directives, and procedures. The Oyster Creek EQ Program will continue to comply with 10 CFR 50.49 throughout the renewal period, including development and maintenance of qualification documentation demonstrating reasonable assurance that a component can perform required functions during harsh accident conditions **(Reference: CC-AA-203, paragraph 1.1)**.

b) The Oyster Creek EQ Program documents identify the applicable environmental conditions for the component locations in the EQ binders **(Reference: CC-MA-203-1001, Exhibit 4)**.

c) The Oyster Creek EQ Program qualification files are maintained at the plant site in an auditable form for the duration of the installed life of the component **(Reference: CC-AA-203, paragraph 4.1.9).**

d) EQ program documentation is controlled under the station's quality assurance program. 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 (See Element 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:

EQ programs include consideration of operating experience to modify qualification bases and conclusions, including qualified life. Compliance with 10 CFR 50.49 provides reasonable assurance that components can perform their intended functions during accident conditions after experiencing the effects of inservice aging.

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., 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 EQ components are being adequately managed by the Oyster Creek EQ Program. The following examples of operating experience provide objective evidence that the Oyster Creek EQ Programs uses industry operating experience to assess and modify if necessary qualification bases and conclusions, including qualified life.

Oyster Creek participates through corporate engineering in nuclear industry EQ users groups. These groups distribute relevant information on EQ issues throughout the industry. The Oyster Creek Operating Experience engineer reviews these notices and generates an open item in a controlled Operating Experience program database for review by the station EQ engineer. The Oyster Operating Experience program process then ensures reviews and resolutions are performed for issues that need disposition at Oyster Creek.

The Oyster Creek response to industry EQ issues is maintained with each EQ binder. For example in file OC-375 **(Reference: 4.3.3),** form SRS A-14 documents the Oyster Creek response to NRC information notices IEN 83-08, IEN 84-20, IE 91-20, IEN 9494-78. Similarly, in OC-319 for NAMCO switches the impact of NRC IEN 98-21 is addressed.

A review of the correction action program database did not reveal any occurrence where the qualified life of a component has been exceeded. This review did not indicate any adverse trend in the EQ program. A review of the program's heath reports indicates that the Oyster Creek EQ program is properly managing EQ component at Oyster Creek. This provides evidence that the Oyster Creek Program is complying with 10 CFR 50.49 and provides reasonable assurance that EQ components can perform their intended functions during accident conditions after experiencing the effects of inservice aging.

3.10 Conclusion

The Oyster Creek Oyster Creek EQ Program aging management program is credited for managing electrical components in harsh environments for the systems important to safety. The Oyster Creek Environmental Qualification (EQ) Program's elements have been evaluated against NUREG-1801 in Section 3.0. No program exceptions have been identified as discussed in Section 2.3. No program enhancements have been identified as discussed 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 Environmental Qualification (EQ) aging management program provides reasonable assurance that electrical components important to safety in harsh environments 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 10 CFR 50.49, *Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants*, Office of the Federal Register, National Archives and Records

Administration, 2005

4.2.2 DOR Guidelines, *Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors,* November 1979

4.2.3 Regulatory Guide 1.89, Rev. 1, *Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants,* U. S. Nuclear Regulatory Commission, June 1984

4.2.4 NUREG-0588, *Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,* U. S. Nuclear Regulatory Commission, July 1981

4.3 Oyster Creek Program References

4.3.1 ES-027, Engineering Standard , “Environmental Parameters-Oyster Creek NGS”, 6/13/2000.

4.3.2 ECR 04-00795, “Revise EQ Files for Licensing Renewal”, 5/5/2005

4.3.3 OC-375, “Environmental Qualification of ASCo automatic Transfer Switches”, Rev 3,

4.3.4 OC-319, “Environmental Qualification Namco Control Limit Switches Model EA740 Series”, Rev 9

5.0 TABLES

5.1 Aging Management Program Implementing Documents

Procedure Number	Procedure Title	AR No.	Status
CC-AA-203	Environmental Qualification Program	330592.45.01	ACC/ASG
CC-MA-203-1001	Environmental Qualification Engineering	330592.45.02	ACC/ASG
MA-MA-716-009	Preventive Maintenance(PM) Work Order Process	330592.45.03	ACC/ASG

5.2 Aging Management Review Results

The systems and components important to safety managed by the Oyster Creek Environmental Qualification Program are listed in the station EQ binders. The analyses used to qualify these components for the current 40 years or more have been reanalyzed for the extended period of operation. These analyses are time limited aging analyses (TLAAs) as defined by 10 CFR 54.3(a). As described in paragraph 4.4 of the Oyster Creek LRA the option selected to disposition these TLAAs in accordance with 10 CFR 54.21(c)(1)(iii) is the Oyster Creek Environmental Qualification aging management program described in this program basis document.

6.0 ATTACHMENTS

- LRA Appendix A
- LRA Appendix B

PROGRAM BASIS DOCUMENT

PBD-AMP-B.1.10

Revision 0

THERMAL AGING AND NEUTRON IRRADIATION EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL

GALL PROGRAM XI.M13 - THERMAL AGING AND NEUTRON IRRADIATION EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL

Prepared By: ______________________

Reviewed By: ______________________

Program Owner Review: ______________________

Technical Lead Approval: ______________________

Revision History:

Revision	*Prepared by:*	*Reviewed by:*	*Program Owner:*	*Approved by:*
0	*Dennis Davis*	*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 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel aging management program that are credited for managing loss of fracture toughness 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 for 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 provides a comparison of the credited Oyster Creek program with the elements of the corresponding NUREG-1801 Chapter XI program XI.M13 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel. 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>

a) *The reactor vessel internals receive a visual inspection in accordance with the American Society of Mechanical Engineers (ASME) Code Section XI, Subsection IWB, Category B-N-3. This inspection is augmented to detect the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) reactor vessel internals.*

b) *This aging management program (AMP) includes (a) identification of susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (i.e., ferrite and molybdenum contents, casting process, and operating temperature) and/or neutron irradiation embrittlement (neutron fluence), and*

c) *(b) for each "potentially susceptible" component, aging management is accomplished through either a supplemental examination of the affected component based on the neutron fluence to which the component has been exposed as part of the applicant's 10-year inservice inspection (ISI) program during the license renewal term, or a component-specific evaluation to determine its susceptibility to loss of fracture toughness.*

Oyster Creek:

a) The Oyster Creek reactor vessel internals receive a visual inspection in accordance with the American Society of Mechanical Engineers (ASME) Code Section XI, Subsection IWB, Category B-N-3. This inspection will be augmented to detect the effects of loss of fracture toughness due to thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel (CASS) reactor vessel internals determined to be susceptible to loss of fracture toughness.

b) The Oyster Creek Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel aging management program includes a component specific evaluation to (a) identify the "susceptible components" determined to be limiting from the standpoint of thermal aging susceptibility (i.e., ferrite and molybdenum contents, casting process, and operating temperature) and/or neutron irradiation embrittlement (neutron fluence), and

c) (b) for each "potentially susceptible" component, aging management is accomplished through either a supplemental examination of the affected component based on the neutron fluence to which the component has been exposed as part of the Oyster Creek's BWR Reactor Internals program during the license renewal term, or a component-specific evaluation to determine its susceptibility to loss of fracture toughness.

The component specific evaluations will be performed prior to the period of extended operation as discussed in Section 3.10. The supplemental examinations, if required, will be performed during the period of extended operation as discussed in Section 3.4.

2.2 Overall NUREG-1801 Consistency

The Oyster Creek Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel aging management program is a new program that is consistent with NUREG-1801 aging management program XI.13 Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS).

2.3 Summary of Exceptions to NUREG-1801

None. The new Oyster Creek Thermal Aging and Neutron

Irradiation Embrittlement of Cast Austenitic Stainless Steel aging management program is found to be adequate to support the extended period of operation with no exceptions.

2.4 Summary of Enhancements to NUREG-1801

None. The new Oyster Creek Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel 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:

a) *The program provides screening criteria to determine the susceptibility of CASS components to thermal aging on the basis of casting method, molybdenum content, and percent ferrite.*

b) *The screening criteria are applicable to all primary pressure boundary and reactor vessel internal components constructed from SA-351 Grades CF3, CF3A, CF8, CF8A, CF3M, CF3MA, CF8M, with service conditions above 250°C (482°F).*

c) *The screening criteria for susceptibility to thermal aging embrittlement are not applicable to niobium-containing steels; such steels require evaluation on a case-by-case basis.*

d) *For "potentially susceptible" components, the program provides for the consideration of the synergistic loss of fracture toughness due to neutron embrittlement and thermal aging embrittlement. For each such component, an applicant can implement either (a) a supplemental examination of the affected component as part of a 10-year ISI program during the license renewal term, or (b) a component-specific evaluation to determine the component's susceptibility to loss of fracture toughness.*

e) *Based on the criteria set forth in the May 19, 2000 letter from Christopher Grimes, Nuclear Regulatory Commission (NRC), to Mr. Douglas Walters, Nuclear Energy Institute (NEI), the susceptibility to thermal aging embrittlement of CASS components is determined in terms of casting method, molybdenum content, and ferrite content. For low-molybdenum content (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement.*

f) *Static-cast low-molybdenum steels with ≤20% ferrite and all centrifugal-cast low-molybdenum steels are not susceptible.*

g) *For high-molybdenum content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% ferrite and centrifugal-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement.*

h) *Static-cast high-molybdenum steels with ≤14% ferrite and centrifugal-cast high-molybdenum steels with ≤20% ferrite are not susceptible.*

i) *In the susceptibility screening method, ferrite content is calculated by using the Hull's equivalent factors (described in NUREG/CR-4513, Rev. 1) or a method producing an equivalent level of accuracy (±6% deviation between measured and calculated values).*

j) *A fracture toughness value of 255 kJ/m2 (1,450 in.-lb/in.2) at a crack depth of 2.5 mm (0.1 in.) is used to differentiate between CASS materials that are nonsusceptible and those that are potentially susceptible to thermal aging embrittlement. Extensive research data indicate that for nonsusceptible CASS materials, the saturated lower-bound fracture toughness is greater than 255 kJ/m2 (NUREG/CR-4513, Rev. 1).*

Oyster Creek:

a) The Oyster Creek Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel aging management program provides screening criteria to determine the susceptibility of CASS reactor internal components to thermal aging on the basis of casting method, molybdenum content, and percent ferrite **(Reference: ER-AB-331-101 Paragraph 4.5.2).**

b) The Oyster Creek screening criteria are applicable to the primary pressure boundary reactor vessel internal components constructed from SA-351 Grades CF3, CF3A, CF8, CF8A, CF3M, CF3MA, CF8M, with service conditions above 250°C (482°F) **(Reference: ER-AB-331-101 Paragraph 1.1).**

c) The Oyster Creek screening criteria for susceptibility to thermal aging embrittlement are not applicable to niobium-containing steels; such steels require evaluation on a case-by-case basis.

d) For "potentially susceptible" components, the Oyster Creek Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel aging management program provides for the consideration of the synergistic loss of fracture toughness due to neutron embrittlement and thermal aging embrittlement. For each such component, Oyster Creek implements either (a) a supplemental examination of the affected component as part of the BWR Reactor Internals program during the license renewal term, or (b) a component-specific evaluation to determine the component's susceptibility to loss of fracture toughness **(Reference: ER-AB-331-101 Paragraph 4.4).**

e) Based on the criteria set forth in the May 19, 2000 letter from Christopher Grimes, Nuclear Regulatory Commission (NRC), to Mr. Douglas Walters, Nuclear Energy Institute (NEI), Oyster Creek's susceptibility to thermal aging embrittlement of CASS components is determined in terms of casting method, molybdenum content, and ferrite content. For low-molybdenum content (0.5 wt.% max.) steels, only static-cast steels with >20% ferrite are potentially susceptible to thermal embrittlement **(Reference: ER-AB-331-101 Paragraph 4.5.1 and 4.6.2).**

e) Static-cast low-molybdenum steels with ≤20% ferrite and all centrifugal-cast low-molybdenum steels are not susceptible and will be excluded from the Oyster Creek Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel aging management program **(Reference: ER-AB-331-101 Paragraph 4.6.1).**

f) High-molybdenum content (2.0 to 3.0 wt.%) steels, static-cast steels with >14% *ferrite and centrifugal-cast steels with* >20% *ferrite are* potentially susceptible to thermal embrittlement and will be evaluated **(Reference: ER-AB-331-101 Paragraph 4.6.1).**

g) Static-cast high-molybdenum steels with ≤14% ferrite and centrifugal-cast high-molybdenum steels with ≤20% ferrite are not susceptible and will be excluded from the Oyster Creek Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel aging management program **(Reference: ER-AB-331-101 Paragraph 4.6.1).**

h) In the Oyster Creek susceptibility screening, ferrite content is calculated by using the Hull's equivalent factors (described in NUREG/CR-4513, Rev. 1) **(Reference: ER-AB-331-101 Paragraph 4.5.2).**

j) A fracture toughness value of 255 kJ/m2 (1,450 in.-lb/in.2) at a crack depth of 2.5 mm (0.1 in.) may be used at Oyster Creek to differentiate between CASS materials that are nonsusceptible and those that are potentially susceptible to thermal aging embrittlement as an alternative approach to the above. Extensive research data indicate that for nonsusceptible CASS materials, the saturated lower-bound fracture toughness is greater than 255 kJ/m2 (NUREG/CR-4513, Rev. 1).

The scope of thermal/neutron embrittlement applies to all CASS reactor vessel internals in the scope for license renewal. These components include the fuel support pieces, base of the CRD guide tubes, and the core spray nozzle elbows **(Reference: ER-AB-331-101 Paragraph 4.1).**

Each component will be evaluated based on its susceptibility to loss of fracture toughness. The program performs an assessment of cast austenitic stainless steel (CASS) components that may be susceptible to thermal aging/neutron embrittlement to determine whether loss of fracture toughness due to thermal aging/neutron embrittlement is occurring **(Reference: ER-AB-331-101 Paragraph 4.2).**

For susceptible components, a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions may be performed as discussed in Section 3.4. For those components where the function may be affected, a supplemental inspection will be performed as part of the station BWR Reactor Internals program. These visual inspections will be performed consistent with enhanced ASME Section XI VT-1 (EVT-1) visual inspection requirements. Enhanced visual inspection techniques will be consistent with those described in BWRVIP-03 **(Reference: ER-AB-331-101 Paragraph 4.5.3).**

The Oyster Creek Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel aging management program manages the aging effect of loss of fracture toughness for the components and environments listed in Table 5.2. The implementing documents for this aging management program are listed in Table 5.1 and are described throughout the individual program element discussions. The commitment numbers under which these implementing documents are being revised are contained within the listings in Table 5.1.

Exceptions to NUREG-1801, Element 1:

None.

Enhancements to NUREG-1801, Element 1:

None.

Comparison and Evaluation Conclusion:

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

3.1 Preventive Actions

NUREG-1801:

The program consists of evaluation and inspection and provides no guidance on methods to mitigate thermal aging and neutron irradiation embrittlement.

Oyster Creek:

The program consists of an evaluation and a supplemental inspection of susceptible components and provides no guidance on methods to mitigate thermal aging and neutron irradiation embrittlement **(Reference: ER-AB-331-101).**

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 program specifics depend on the neutron fluence and thermal embrittlement susceptibility of the component. The AMP monitors the effects of loss of fracture toughness on the intended function of the component by identifying the CASS materials that either have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV) or are determined to be susceptible to thermal aging embrittlement. For such materials, the program consists of either supplemental examination of the affected component based on the neutron fluence to which the component has been exposed, or component-specific evaluation to determine the component's susceptibility to loss of fracture toughness.

Oyster Creek:

The program will identify components that are potentially susceptible to thermal aging, as well as those exposed to neutron fluences in excess of 10^{17} n/cm^2 (E>1 MeV) **(Reference: ER-AB-331-101 Paragraph 1.1).**

Each potentially susceptible component will be evaluated to determine the susceptibility to the loss of fracture toughness, as well as the effect on the components ability to perform its function. For those components where the function may be affected, a supplemental inspection will be performed as part of the Oyster Creek BWR Reactor Internals program **(Reference: ER-AB-331-101 Paragraph 4.2 and 4.4).**

Exceptions to NUREG-1801, Element 3:

None.

<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) *For reactor vessel internal CASS components that have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV) or are determined to be susceptible to thermal embrittlement, the 10-year ISI program during the renewal period includes a supplemental inspection covering portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (i.e., ferrite and molybdenum contents, casting process, and operating temperature), neutron fluence, and cracking susceptibility (i.e., applied stress, operating temperature, and environmental conditions).*

b) *The inspection technique is capable of detecting the critical flaw size with adequate margin. The critical flaw size is determined based on the service loading condition and service-degraded material properties. One example of a supplemental examination is enhancement of the visual VT-1 examination of Section XI IWA-2210. A description of such an enhanced visual VT-1 examination could include the ability to achieve a 0.0005-in. resolution, with the conditions (e.g., lighting and surface cleanliness) of the inservice examination bounded by those used to demonstrate the resolution of the inspection technique.*

c) *Alternatively, the applicant may perform a component-specific evaluation, including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component is not required. Failure to meet this criterion requires continued use of the supplemental inspection program.*

d) *For each CASS component that has been subjected to a neutron fluence of less than 10^{17} n/cm^2 (E>1 MeV) and is potentially susceptible to thermal aging, the supplement inspection program applies; otherwise, the existing ASME Section XI inspection requirements are adequate if the components are not susceptible to thermal aging embrittlement.*

Oyster Creek:

a) For Oyster Creek reactor vessel internal CASS components that have a neutron fluence of greater than 10^{17} n/cm^2 (E>1 MeV) or are determined to be susceptible to thermal embrittlement, the Reactor Vessel Internals program during the renewal period will include a supplemental inspection covering portions of the susceptible components determined to be limiting from the standpoint of thermal aging susceptibility (i.e., ferrite and molybdenum contents, casting process, and operating temperature), neutron fluence, and cracking susceptibility (i.e., applied stress, operating temperature, and environmental conditions) **(Reference: ER-AB-331-101 Paragraph 4.5.1 and 4.5.2).**

b) The Oyster Creek inspection technique is capable of detecting the critical flaw size with adequate margin. The critical flaw size is determined based on the service loading condition and service-degraded material properties. The supplemental examination method will be an enhanced visual VT-1 examination with the ability to achieve a 0.0005-in. resolution, with the necessary plant conditions (e.g., lighting and surface cleanliness) to adequately support the resolution and inspection technique **(Reference: ER-AB-331-101 Paragraph 4.5.3).**

c) Alternatively, Oyster Creek may perform a component-specific evaluation, including a mechanical loading assessment to determine the maximum tensile loading on the component during ASME Code Level A, B, C, and D conditions. If the loading is compressive or low enough (<5 ksi) to preclude fracture, then supplemental inspection of the component will not be performed. Failure to meet this criterion will require continued use of the supplemental inspection program **(Reference: ER-AB-331-101 Paragraph 4.6.1).**

d) For each Oyster Creek CASS component that has been subjected to a neutron fluence of less than 10^{17} n/cm^2 (E>1 MeV) and is potentially susceptible to thermal aging, the supplement inspection program will apply; otherwise, the existing ASME Section XI Subsection IWB, Category B-N-3 inspection requirements will be applied if the components are not susceptible to thermal aging embrittlement **(Reference: ER-AB-331-101 Paragraph 1.1).**

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:

Inspections scheduled in accordance with IWB-2400 and reliable examination methods provide timely detection of cracks.

Oyster Creek:

Inspections will be performed for susceptible components in accordance with the Oyster Creek in-vessel visual examination procedure that incorporates the requirements of ASME Section XI Subsection IWB-2400 for additional examinations, as well as past inservice inspection experience. Evaluation of flaws will be conducted consistent with applicable and approved BWRVIP guidelines **(Reference: ER-AB-331-101 Paragraph 4.4)**.

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) *Flaws detected in CASS components are evaluated in accordance with the applicable procedures of IWB-3500. Flaw tolerance evaluation for components with ferrite content up to 25% is performed according to the principles associated with IWB-3640 procedures for submerged arc welds (SAW), disregarding the Code restriction of 20% ferrite in IWB-3641(b)(1).*

b) *Extensive research data indicate that the lower-bound fracture toughness of thermally aged CASS materials with up to 25% ferrite is similar to that for SAWs with up to 20% ferrite (Lee et al., 1997). Flaw evaluation for CASS components with >25% ferrite is performed on a case-by-case basis by using fracture toughness data provided by the applicant.*

Oyster Creek:

a) Detected flaws at Oyster Creek will be evaluated in accordance with IWB-3500. Flaw tolerance evaluation for components with ferrite content up to 25% will be performed according to the principles associated with IWB-3640 procedures for submerged arc welds (SAW), disregarding the Code restriction of 20% ferrite in IWB-3641(b)(1) **(Reference: ER-AB-331-101 Paragraph 4.6.2).**

b) Oyster Creek understands that extensive research data indicates that the lower-bound fracture toughness of thermally aged CASS materials with up to 25% ferrite is similar to that for SAWs with up to 20% ferrite (Lee et al., 1997). Flaw evaluation for CASS components with >25% ferrite will be performed on a case-by-case basis by using Oyster Creek fracture toughness data provided by Oyster Creek **(Reference: ER-AB-331-101 Paragraph 4.6.2).**

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:

Repair is performed in conformance with IWA-4000 and IWB-4000, and replacement in accordance with IWA-7000 and 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:

Repairs and replacements are performed in accordance with the requirements of ASME Section XI Subsection IWA-4000 in the 1995 Edition with the 1996 Addenda. In the 1995 Edition of ASME Section XI, Subsections IWB-4000 and IWB-7000 were deleted and the requirements were relocated in IWA-4000. Oyster Creek's corrective action process is governed by 10 CFR 50, Appendix B, and is implemented by corporate administrative procedures **(Reference: ER-AB-331-101 Paragraph 4.7)**.

Evaluations are performed for test or inspection results that do not satisfy established criteria and an Issue Report is initiated to document the concern in accordance with the station's corrective action program. The 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 prevent recurrence **(Reference: ER-AB-331-101 Paragraph 4.7)**.

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:

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 AMP was developed by using research data obtained on both laboratory-aged and service-aged materials. Based on this information, the effects of thermal aging embrittlement on the intended function of CASS components are effectively managed.

Oyster Creek:

Research data on both laboratory-aged and service-aged materials has confirmed that loss of fracture toughness could occur in some reactor vessel CASS internal components. However, as indicated in BWRVIP-47, there is no record of any reactor vessel CASS internal component failures during operation. Oyster Creek internal reactor vessel CASS components are periodically examined, but no degradation has been identified to date. Since the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel aging management program is a new program, a review of plant operating experience can not confirm at this time that loss of fracture toughness of cast austenitic stainless steel is a factor.

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.

The Oyster Creek Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel aging management program will include a component specific evaluation to assess the susceptibility for the loss of fracture toughness. This evaluation will be performed prior to the period of extended operation. A supplemental inspection will be performed for those components where loss of fracture toughness may affect function of the component, using the criteria provided in NUREG-1801 Aging Management Program, XI.M13, Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel. This inspection will ensure the integrity of the CASS components exposed to the high temperature and neutron fluence present in the reactor environment.

3.10 Conclusion

The Oyster Creek Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel aging management program is credited for managing loss of fracture toughness for the systems, components, and environments listed in Table 5.2. The Oyster Creek Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel 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 implementation of the Oyster Creek Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel aging management program will provide reasonable assurance that loss of fracture toughness 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 BWRVIP-03, *BWR Vessel Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines*

4.2.2 BWRVIP-47, *BWR Lower Plenum Inspections and Flaw Evaluation Guidelines*

4.2.3 ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* ASME Boiler and Pressure Vessel Code, 1995 Edition through the 1996 Addenda, American Society of Mechanical Engineers, New York, NY.

4.2.4 Letter from Christopher I. Grimes, U.S. Nuclear Regulatory Commission, License Renewal and Standardization Branch, to Douglas J. Walters, Nuclear Energy Institute, License Renewal Issue No. 98-0030, *Thermal Aging Embrittlement of Cast Stainless Steel Components*, May 19, 2000.

4.2.5 NUREG/CR-4513, Rev. 1, *Estimation of Fracture Toughness of Cast Stainless Steels during Thermal Aging in LWR Systems,* U.S. Nuclear Regulatory Commission, August 1994.

4.2.6 Lee, S., Kuo, P. T., Wichman, K., and Chopra, O., *Flaw Evaluation of Thermally Aged Cast Stainless Steel in Light-Water Reactor Applications*, Int. J. Pres. Ves. and Piping, 72, pp. 37-44, 1997

4.3 Oyster Creek Program References

4.3.1 ER-AB-331, Rev. 3, *BWR Rx Internals Management Program Activities*

4.3.2 ER-AB-331-101, Rev. 0, *Evaluation for Thermal Aging/Neutron Embrittlement of Reactor Internals Components*

5.0 TABLES

5.1 Aging Management Program Implementing Documents

Procedure Number	Procedure Title	Commitment No.	Status
ER-AB-331-101, Rev. 0	Evaluation for Thermal Aging/Neutron Embrittlement of Reactor Internals Components		

5.2 Aging Management Review Results

SSC Name	Structure and/or Component	Material	Environment	Aging Effect
Reactor Internals	Fuel Support Piece	CASS	Treated Water >482F	Loss of Fracture Toughness
Reactor Internals	Core Spray Line Spray Nozzle Elbows	CASS	Treated Water >482F	Loss of Fracture Toughness
Reactor Internals	Control Rod Drive Assembly (Housing and Guide Tube)	CASS	Treated Water >482F (Internal)	Loss of Fracture Toughness

6.0 ATTACHMENTS

6.1 LRA Appendix A

6.2 LRA Appendix B

PROGRAM BASIS DOCUMENT

PBD-AMP-B.1.04

Revision 0

BWR VESSEL ID ATTACHMENT WELDS

GALL PROGRAM XI.M4 - BWR VESSEL ID ATTACHMENT WELDS

Prepared By: ______________________

Reviewed By: ______________________

Program Owner Review: ______________________

Technical Lead Approval: ______________________

Revision History:

Revision	*Prepared by:*	*Reviewed by:*	*Program Owner:*	*Approved by:*
0	*Charles Micklo*	*George Beck*	*Greg Harttraft*	*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 BWR Vessel ID Attachment Welds aging management program that are credited for managing cracking due to stress corrosion cracking (SCC) and intergranular stress corrosion cracking (IGSCC) 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.M4, BWR Vessel ID Attachment Welds. 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>

a) *The program includes inspection and flaw evaluation in accordance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP)-48, and*

b) *monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 (Electric Power Research Institute [EPRI] TR-103515) to ensure the long-term integrity and safe operation of boiling water reactor (BWR) vessel inside diameter (ID) attachment welds.*

<u>Oyster Creek:</u>

a) The Oyster Creek BWR Vessel ID Attachment Welds aging management program is an existing program that incorporates the inspection and flaw evaluation recommendations of staff-approved Boiling Water Reactor Vessel and Internals Project BWRVIP-48-A, "Vessel ID Attachment Weld Inspection and Evaluation Guidelines."

b) Recommendations for monitoring and control of reactor coolant water chemistry contained in BWRVIP-130 (EPRI TR-1008192), which replaces BWRVIP-29, are implemented through the station's water chemistry procedures to ensure the long-term integrity and safe operation of boiling water reactor (BWR) vessel inside diameter (ID) attachment welds. This is discussed in section 3.2.b.

Thus, the Oyster Creek BWR Vessel ID Attachment Welds aging management program provides for identification, evaluation and mitigation by periodic examinations and water chemistry control.

Note: "-A" is used to denote NRC acceptance of the BWRVIP document.

2.2 Overall NUREG-1801 Consistency

The Oyster Creek BWR Vessel ID Attachment Welds program is an existing program that is consistent with NUREG-1801 aging management program XI.M4, BWR Vessel ID Attachment Welds with exception(s) as described in 2.3 below.

2.3 Summary of Exceptions to NUREG-1801

The Oyster Creek BWR Vessel ID Attachment Welds aging management program is found to be adequate to support the extended period of operation with the following exceptions:

- NUREG-1801 indicates that water chemistry control is in accordance with BWRVIP-29 for water chemistry in BWRs. BWRVIP-29 references the 1993 revision of EPRI TR-103515, "BWR Water Chemistry Guidelines." The Oyster Creek water chemistry programs are based on BWRVIP-130: "BWR Vessel and Internals Project BWR Water Chemistry Guidelines (EPRI TR-1008192)", which is the 2004 revision of "BWR Water Chemistry Guidelines". For justification of exceptions to the water chemistry program see the Oyster Creek Water Chemistry aging management program, PBD-AMP-B.1.02.
- Additionally, NUREG-1801 program XI.M4 references ASME Section XI, Table IWB 2500-1 (2001 edition, including the 2002 and 2003 Addenda). Oyster Creek ISI program is based on the 1995 (including 1996 Addenda) version of ASME Section XI. For justification of exceptions to the ISI program see the ASME Section XI In-service Inspection, Subsections IWB, IWC, and IWD aging management program, PBD-AMP-B.1.01.

2.4 Summary of Enhancements to NUREG-1801.

None. The existing Oyster Creek BWR Vessel ID Attachment Welds 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:

a) *The program is focused on managing the effects of cracking due to stress corrosion cracking (SCC), including intergranular stress corrosion cracking (IGSCC).*

b) *The program contains preventive measures to mitigate SCC inservice inspection (ISI) to detect cracking and monitor the effects of cracking on the intended function of the components, and repair and/or replacement, as needed, to maintain the ability to perform the intended function.*

c) *The guidelines of BWRVIP-48 include inspection recommendations and evaluation methodologies for the attachment welds between the vessel wall and vessel ID brackets that attach safety-related components to the vessel (e.g., jet pump riser braces and core-spray piping brackets). In some cases, the attachment is a simple weld; in others, it includes a weld build-up pad on the vessel. The BWRVIP-48 guidelines include information on the geometry of the vessel ID attachments, evaluate susceptible locations and safety consequence of failure, provide recommendations regarding the method, extent, and frequency of inspection, and discuss acceptable methods for evaluating the structural integrity* *significance of flaws detected during these examinations.*

Oyster Creek:

a) The Oyster Creek BWR Vessel ID Attachment Welds Program station program manages the effects of stress corrosion cracking (SCC), including intergranular stress corrosion cracking (IGSCC) to ensure long term integrity of the vessel ID attachment welds. The scope of the programs includes the steam dryer support lugs, guide rod wall bracket, feedwater sparger bracket, and surveillance sample holder bracket. **(Reference: OC-5, Inspection Plan 7.9, paragraph 3)**

b) The program includes measures to mitigate SCC by ensuring the water chemistry recommendations of BWRVIP-130 (EPRI TR-1008192) are used in the stations the water chemistry program **(Reference: OC-5, paragraph 1.3; CY-AB-120-100).** In addition, the stations' in-vessel examination programs inspect for and monitor the effects of cracking **(Reference: OC-5, Inspection Plan 7.9, paragraph 5)**. Specifically, inspections of the vessel ID brackets and their attachments to the vessel ID are currently performed under the Oyster Creek's ASME Section XI program in examination category B-N-2 ("Interior Attachments to the Reactor Vessel"). These ASME Section XI visual inspections use examination methods VT-1 and VT-3 to detect discontinuities and imperfections on the surfaces of components and to determine the general mechanical and structural condition of the component supports **(Reference: OC-5, paragraph 1.9; ER-AA-330-002)**. Repair and replacement activities, if needed, are performed in accordance with the recommendations of the appropriate BWRVIP repair/replacement guidelines and the requirements of ASME Section XI **(Reference: OC-5, paragraph 1.7; ER-AA-330-009).**

c) The program is consistent with the guidelines of BWRVIP-48-A that include inspection recommendations and evaluation methodologies for the attachment welds between the vessel wall and vessel ID brackets that attach safety-related components to the vessel. BWRVIP-48-A maintains the inspection frequency per ASME Section XI Examination Category B-N-2 and recommends more stringent inspection techniques for certain selected attachments. All of these welds are inspected using both VT-3 and Enhanced VT-1 methods **(Reference: OC-5, Inspection Plan 7.9, paragraph 4).**

The existing Oyster Creek BWR Vessel ID Attachment Welds

Program aging management program manages the aging effect of *cracking for the reactor vessel 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>

None.

<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>

a) *The BWRVIP-48 provides guidance on detection, but does not provide guidance on methods to mitigate cracking.*

b) *Maintaining high water purity reduces susceptibility to SCC or IGSCC. Reactor coolant water chemistry is monitored and maintained in accordance with the guidelines in BWRVIP-29 (EPRI TR-103515). The program description and evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Section XI.M2, "Water Chemistry".*

<u>Oyster Creek:</u>

a) The program follows the guidance for inspecting and detecting cracks provided in BWRVIP-48-A as discussed in Element 1-c above **(Reference: OC-5, Inspection Plan 7.9, paragraph 4).**

b) The Oyster Creek station mitigates the potential for SCC and IGSCC through the water chemistry program. The reactor water chemistry monitors and controls known detrimental contaminants such as chlorides, dissolved oxygen, and sulfate concentrations in accordance with the recommendations of the BWRVIP-130 **(Reference: OC-5, paragraph 1.3; CY-AB-120-100).** BWRVIP-130 (EPRI TR-1008192) replaces BWRVIP-29, the previous EPRI water chemistry standard. This exception is discussed below. The use of BWRVIP-130 is discussed in the Oyster Creek Water Chemistry aging management program, PBD-AMP-B.1.02.

Exceptions to NUREG-1801, Element 2:

NUREG-1801 indicates that water chemistry control is in accordance with BWRVIP-29 for water chemistry in BWRs. BWRVIP-29 references the 1993 revision of EPRI TR-103515, "BWR Water Chemistry Guidelines." The Oyster Creek water chemistry programs are based on BWRVIP-130: "BWR Vessel and Internals Project BWR Water Chemistry Guidelines (EPRI TR-1008192)", which is the 2004 revision of "BWR Water Chemistry Guidelines". For justification of exceptions to the water chemistry program see the Oyster Creek Water Chemistry aging management program, PBD-AMP-B.1.02.

Enhancements to NUREG-1801, Element 2:

None.

Comparison and Evaluation Conclusion:

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

3.2 Parameters Monitored or Inspected

NUREG-1801:

a) *The program monitors the effects of SCC and IGSCC on the intended function of vessel attachment welds by detection and sizing of cracks by ISI in accordance with the guidelines of approved BWRVIP-48 and the requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI, Table IWB 2500-1 (2001 including the 2002 and 2003 addenda).*

b) *An applicant may use the guidelines of BWRVIP-62 for inspection relief for vessel internal components with hydrogen water chemistry provided that such relief is submitted under the provisions of 10 CFR 50.55a and approved by the staff.*

Oyster Creek:

a) The program monitors the effects of SCC and IGSCC on the intended function of vessel attachment welds through inspection of the internal components. The inspections are performed consistent with the recommendations of BWRVIP-48-A, as well as the requirements of Section XI of the ASME Code. Specifically, inspections of the vessel ID brackets and their attachments to the vessel ID are currently performed under the Oyster Creek's ASME Section XI program in examination category B-N-2 of Table 2500-1 ("Interior Attachments to the Reactor Vessel"). The Oyster Creek ISI program is consistent with ASME Section XI, Table IWB 2500-1 (1995 addition through 1996 addenda). The Section XI program is updated periodically in accordance 10CFR 50.55a. **(Reference: OC-5, Inspection Plan 7.9; ER-AA-330-002).**

b) No inspection relief based on BWRVIP-62 has been requested by Oyster Creek.

Exceptions to NUREG-1801, Element 3:

NUREG-1801 program XI.M4 references ASME Section XI, Table IWB 2500-1 (2001 edition, including the 2002 and 2003 Addenda). Oyster Creek ISI program is based on the 1995 (including 1996 Addenda) version of ASME Section XI. For justification of exceptions to the ISI program see the ASME Section XI In-service Inspection, Subsections IWB, IWC, and IWD aging management program, PBD-AMP-B.1.01.

Enhancements to NUREG-1801, Element 3:

None.

Comparison and Evaluation Conclusion:

This element is consistent with NUREG-1801, Element 3, Preventive Actions, with exceptions as described above.

3.3 Detection of Aging Effects

NUREG-1801:

a) The extent and schedule of the inspection and test techniques prescribed by BWRVIP-48 guidelines are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of intended function. Inspection can reveal cracking.

b) Vessel ID attachment welds are inspected in accordance with the requirements of ASME Section XI, Subsection IWB, examination category B-N-2. The Section XI inspection *specifies visual VT-1 examination to detect discontinuities and imperfections on the surfaces of components and visual VT-3 examination to determine the general mechanical and structural condition of the component supports.*

c) *The inspection and evaluation guidelines of BWRVIP-48 recommend more stringent inspections for selected attachments. The guidelines recommend enhanced visual VT-1 examination of all safety-related attachments and those nonsafety-related attachments identified as being susceptible to IGSCC. Visual VT-1 examination is capable of achieving 1/32 in. resolution; the enhanced visual VT-1 examination method is capable of achieving a 1-mil wire resolution.*

d) *The nondestructive examination (NDE) techniques appropriate for inspection of BWR vessel internals and their implementation needs, including the uncertainties inherent in delivering and executing NDE techniques in a BWR, are included in BWRVIP-03.*

<u>Oyster Creek:</u>

a) Prior to each refueling outage, the necessary inspections are determined based on BWRVIP-48-A and Section XI of the ASME Code to ensure that aging effects are identified and repaired before the loss of intended function. **(Reference: OC-5, Inspection Plan 7.9, paragraph 4.2)**

b) The inspections are performed consistent with the recommendations of BWRVIP-48-A, as well as the requirements of Section XI of the ASME Code. Specifically, inspections of the vessel ID brackets and their attachments to the vessel ID are currently performed under the Oyster Creek's ASME Section XI program in examination category B-N-2 of Table 2500-1 ("Interior Attachments to the Reactor Vessel"). These ASME Section XI visual inspections use examination methods VT-1 and VT-3 to detect discontinuities and imperfections on the surfaces of components and to determine the general mechanical and structural condition of the component supports. **(Reference: OC-5, Inspection Plan 7.9, paragraph 4)**

c) The program is consistent with the recommendations of the BWRVIP-48-A. BWRVIP-48-A maintains the inspection frequency per ASME Section XI Examination Category B-N-2 and recommends more stringent inspection techniques for certain selected attachments. All of the ID attachment welds at Oyster Creek are inspected using both VT-3 and Enhanced VT-1 methods. **(Reference: OC-5, Inspection Plan 7.9, paragraph 4)**

d) The examination procedures employ Enhanced VT-1 nondestructive examination (NDE) techniques appropriate for inspection of BWR vessel internals and their implementation needs, including the uncertainties inherent in delivering and executing NDE techniques in a BWR, as defined in BWRVIP-03. **(Reference: OC-5, Inspection Plan 7.9, paragraph 4)**

Exceptions to NUREG-1801, Element 4:

NUREG-1801 program XI.M4 references ASME Section XI, Table IWB 2500-1 (2001 edition, including the 2002 and 2003 Addenda). Oyster Creek ISI program is based on the 1995 (including 1996 Addenda) version of ASME Section XI. For justification of exceptions to the ISI program see the ASME Section XI In-service Inspection, Subsections IWB, IWC, and IWD aging management program, PBD-AMP-B.1.01.

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:

a) *Inspections scheduled in accordance with IWB-2400 and approved BWRVIP-48 guidelines provide timely detection of cracks.*

b) *If flaws are detected, the scope of examination is expanded.*

Oyster Creek:

a) The inspections to be performed at each refuel outage are determined by review of BWRVIP-48-A and ASME Section XI.IWB-2400, as well as past experience. The attachment welds at Oyster Creek have been scheduled for re-inspection more frequently based on the furnace-sensitized material of the brackets and the Inconel attachment welds. Evaluation of flaws is conducted consistent with the BWRVIP guidelines. If indications are detected, then an engineering evaluation is required to disposition the indications. **(Reference: OC-5, Inspection Plan 7.9, paragraph 4.2)**

b) If one or more flaws are found during either the baseline inspection or re-inspection, all of the remaining locations of the same type are inspected during the same outage unless one can correlate the flaw to a specific event that would not affect other locations, in accordance with BWRVIP-48-A **(Reference: OC-5, Inspection Plan 7.9, paragraph 4.3)**. The initial expansion of examinations shall comply with the requirements of IWB-2430(a) or alternatives approved by the NRC. If the examinations reveal additional indications exceeding the standards of IWB-3500, then a second expansion of scope is required during the outage. This second expansion shall comply with the requirements of IWB-2430(b) or alternatives approved by the NRC. **(Reference: ER-AA-330-002, paragraphs 4.12 & 4.13)**

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

NUREG-1801 program XI.M4 references ASME Section XI, Table IWB 2500-1 (2001 edition, including the 2002 and 2003 Addenda). Oyster Creek ISI program is based on the 1995 (including 1996 Addenda) version of ASME Section XI. For justification of exceptions to the ISI program see the ASME Section XI In-service Inspection, Subsections IWB, IWC, and IWD aging management program, PBD-AMP-B.1.01.

<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, with exceptions as described above.

3.5 Acceptance Criteria

<u>NUREG-1801:</u>

a) *Any indication detected is evaluated in accordance with ASME Section XI or the staff-approved BWRVIP-48 guidelines.*

b) *Applicable and approved BWRVIP-14, BWRVIP-59, and BWRVIP-60 documents provide guidelines for evaluation of crack growth in stainless steels (SSs), nickel alloys, and low-alloy steels, respectively.*

Oyster Creek:

a) Evaluation of indications is conducted consistent with BWRVIP–48-A or Section XI of the ASME Code, as appropriate. **(Reference: OC-5, Inspection Plan, paragraph 4)**

b) Flaw evaluations for the vessel ID attachment welds, if needed, include the guidance of BWRVIP-14, BWRVIP-59, and BWRVIP-60 guidelines for evaluation of crack growth in stainless steels (SSs), nickel alloys, and low-alloy steels, respectively as applicable. **(Reference: ER-AB-331-1001, paragraph 3.1.3; and OC-5, paragraph 1.6).**

Exceptions to NUREG-1801, Element 6:

NUREG-1801 program XI.M4 references ASME Section XI, Table IWB 2500-1 (2001 edition, including the 2002 and 2003 Addenda). Oyster Creek ISI program is based on the 1995 (including 1996 Addenda) version of ASME Section XI. For justification of exceptions to the ISI program see the ASME Section XI In-service Inspection, Subsections IWB, IWC, and IWD aging management program, PBD-AMP-B.1.01.

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:

a) Repair and replacement procedures are equivalent to those requirements in the ASME Section XI. Repair is in conformance with IWB-4000 and replacement occurs according to IWB-7000.

b) As discussed in the appendix to this report, the staff finds that licensee implementation of the guidelines in BWRVIP-48, as modified, will provide an acceptable level of quality for inspection and flaw evaluation of the safety-related components addressed in accordance with 10 CFR Part 50, Appendix B, corrective actions.

Oyster Creek:

a) 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: OC-5, Inspection Plan 7.9; ER-AA-330-002, paragraph 4.12.4; ER-AA-330-009, paragraph 1.2.3).**

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. **(Reference: OC-5, paragraph 1.5; ER-AA-330-002, paragraph 4.12)**

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

NUREG-1801 program XI.M4 references ASME Section XI, Table IWB 2500-1 (2001 edition, including the 2002 and 2003 Addenda). Oyster Creek ISI program is based on the 1995 (including 1996 Addenda) version of ASME Section XI. For justification of exceptions to the ISI program see the ASME Section XI In-service Inspection, Subsections IWB, IWC, and IWD aging management program, PBD-AMP-B.1.01.

<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, with descriptions as described above.

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 that *licensee implementation of the guidelines in BWRVIP-48, as modified, will provide an acceptable level of quality for inspection and flaw evaluation of the safety-related components addressed in accordance with the 10 CFR Part 50, Appendix B, 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 due to SCC, including IGSCC, has occurred in BWR components. The program guidelines are based on evaluation of available information, including BWR inspection data and information on the elements that cause IGSCC, to determine which attachment welds may be susceptible to cracking. Implementation of the program provides reasonable assurance that crack initiation and growth will be adequately managed and the intended functions of the vessel ID attachments 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 or IGSCC has occurred in BWR components made of stainless steel and alloy steels. **(Reference: BWRVIP-48-A, paragraph 3.1.1)** BWRVIP-48-A has been developed to evaluate materials, environments, and mechanisms that can lead to a loss of intended function of the reactor vessel ID attachment welds. These evaluations determined that these ID attachment welds have the potential to develop cracking due to SCC and IGSCC. BWRVIP-48-A and ASME Section XI, Subsection IWB provide guidelines for performing inspections and flaw evaluations to manage the effects of cracking. A review of Oyster Creek operating experience shows that attachment weld flaw indications have not been reported. During several outages wear indications were reported by inspectors that were determined to be wear marks caused by interferences with the dryer assembly.

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.

The inspection requirements for reactor vessel ID attachment welds are implemented through the station Reactor Internals Program Plan, OC-5, which incorporate the requirements of Section XI of the ASME Code as well as those from BWRVIP-48-A. A visual (VT-1) inspection for examination to detect discontinuities and imperfections on the surfaces of components and visual (VT-3) examination to determine the general mechanical and structural condition of the component supports are performed in accordance with the ASME Section XI ISI program each refueling outage. To date Oyster Creek has found no indications of cracking in the attachment welds. **(Reference: OC-5, Inspection Plan 7.9)**

Demonstration that the effects of aging are effectively managed is achieved through objective evidence that demonstrates that cracking due to SCC and IGGSCC is being adequately managed for the reactor vessel ID attachment welds. The following examples of operating experience provide objective evidence that the BWR Vessel ID Attachment Welds aging management program is effective in assuring that intended function(s) will be maintained consistent with the CLB for the period of extended operation:

1) <u>ID Attachment Welds</u>

Attachment welds flaw indications have not been reported at Oyster Creek. During several outages wear indications were reported by inspectors that were determined to be wear marks caused by interferences with the dryer assembly. Corrective action reports were generated to review these indications and disposition the findings. This example provides objective evidence that the Reactor Internals Program is aggressive in detecting potentials failures and promptly evaluating them. (CAP O2000-1680)

Aging management activities similar to those for the Reactor

Vessel ID Attachment Welds have detected aging degradation with appropriate corrective actions implemented to maintain system and component intended functions, including prompt repair of degraded components prior to failure.

1) Top Guide

Attachment welds follow the processed and procedures for other Reactor Internals. For example, cracking of the Top Guide was first detected in 1994 (MNCR 94-0165). Subsequent inspections in 1996, 2000 and 2004 have tracked cracking growth (MNCR 96-0087, MNCR 96-0131, CAP-O2000-1612, CAP-O2004-3747). Flaw evaluations have been performed to ensure the top guide is acceptable for continued operation and to schedule future inspections. Monitoring of the Top Guide cracking remains a critical Reactor Internals issue. This example provides objective evidence that the Reactor Internals Program whose aging management activities are similar to those of the vessel ID attachment welds program is effective in detecting cracking and that corrective actions are taken to manage cracking before a loss of intended function occurs.

2) Shroud repairs:
In response to industry experience (e.g. SIL 572, IEN 93-79, IEN 94-42) inspection of the shroud was conducted in 1994 (R15) and found significant cracking in the core shroud circumferential weld H4. The inspection was conducted in accordance with the recommendations of the BWRVIP-76. During the same outage a shroud repair system was installed addressing all susceptible shroud circumferential welds. The repair consisted of 10 tie rods anchored to the top and bottom of the Shroud. Subsequent inspections were conducted in 1996 (R16) on the vertical welds. Some crack indications in V-9 were found and were dispositioned as acceptable (MNCR 96-0084). In 1998 (R17) additional vertical welds were inspected using both enhanced visual and UT methodologies. Again minor indications were dispositioned as acceptable (MNCR CAP O1998-1460). Inspections of vertical welds in 2000 (R18) did not find additional crack indications.

The shroud repair and inspection program provides objective evidence that aging management program for the Reactor Internals and BWR penetrations will detect cracking and take corrective action prior to the loss of intended function. This example also demonstrates industry operating experience is considered when establishing acceptance criteria and other effective elements of the aging management programs that manage the reactor vessel and internals.

There is reasonable assurance that the continued implementation of the BWR Vessel ID Attachment Welds Aging Management Program through the Reactor Internals Program will effectively detect and manage the effects of cracking from SCC and IGSCC. Appropriate guidance for reevaluation, repair or replacement is provided for locations where the calculations indicate an area will reach minimum allowable thickness before the next inspection. Periodic self-assessments of the Reactor Internals program are performed to identify the areas that need improvement to maintain the quality performance of the program.

3.10 Conclusion

The existing Oyster Creek BWR Vessel ID Attachment Welds aging management program is credited for managing cracking for the components and environments listed in Table 5.2. The Oyster Creek BWR Vessel ID Attachment Welds program 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 Vessel ID Attachment Welds aging management program provides reasonable assurance that cracking due to SSC or IGSSC 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 BWRVIP-48-A, BWR Vessels and Internal Project, Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines, November 2004

4.2.2 BWRVIP-130: BWR Vessel and Internals Project, BWR Water Chemistry Guidelines - 2004 Revision, Final Report, October 2004

4.2.3 BWRVIP-03, BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines, July 1999

4.2.4 BWRVIP-14, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals, December 1999

4.2.5 BWRVIP-59, Evaluation of Crack Growth in BWR Nickel-Base Austenitic Alloys in RPV Internals, March 2000

4.2.6 BWRVIP-60, BWR Vessel and Internals Project, Evaluation of Crack Growth in BWR Low Alloy Steel RPV Internals, July 1999

4.2.7 BWRVIP-62, BWR Vessel and Internals Project, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection, March 2000

4.3 Oyster Creek Program References

4.3.1 OC-5, Rev 0, Oyster Creek Reactors Internal Program

4.3.2 CAP O2002-1680, Dryer Assembly Procedure

4.3.3 MNCRs 94-0165, 96-0087, 96-0131; CAPs O2000-1612, O2004-3747 Cracking of Top Guide

5.0 TABLES

5.1 Aging Management Program Implementing Documents

Procedure Number	Procedure Title	Commitment Number	Status
OC-5	Reactor Internals Program Plan	330592.04.03	ACC/ASG
ER-AA-330-002	Inservice Inspection of Section XI Welds and Components	330592.04.04	ACC/ASG
ER-AA-330-009	ASME Section XI Repair/Replacement Program	330592.04.02	ACC/ASG
ER-AB-331-1001	BWR RX Internals	330592.04.01	ACC/ASG
CY-AB-120-100	Reactor Water Chemistry	330592.04.05	ACC/ASG

5.2 Aging Management Review Results

SSC Name	Structure and/or Component	Material	Environment	Aging Effect
Reactor Pressure Vessel	Vessel Shell Attachment Welds	Nickel Alloy	Treated Water (Internal)	Cracking Initiation and Growth

6.0 ATTACHMENTS

6.1 LRA Appendix A

6.2 LRA Appendix B