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To: <dja1@nrc.gov>
Date: 02/06/2006 3:02:40 PM
Subject: PBD AMP B.1.09

Donnie,

Attached is an electronic copy of AMP PBD B.1.09. We believe this should address the draft question B.1.9-10.

The file is in Word format, password protected to prevent inadvertent modification. Please let me know is this addresses the question.

We understand that this Word file will be treated similarly to the other AMP PBDs transmitted for Auditing purposes. Thanks, Donnie.

- John.

<<PBD B.1.09 BWR Vessel Internals Rev 0.doc>>

CC: <michaelj.may@exeloncorp.com>, <George.Beck@exeloncorp.com>

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Recipients
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DJA1 (D. Ashley)

exeloncorp.com
George.Beck CC
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PROGRAM BASIS DOCUMENT

PBD-AMP-B.1.09

Revision 0

BWR VESSEL INTERNALS

GALL PROGRAM XI.M9 - BWR VESSEL INTERNALS

Prepared By: ______

Reviewed By: ______

Program Owner Review: ______

Technical Lead Approval: ______

Revision History:

Revision	*Prepared by:*	*Reviewed by:*	*Program Owner:*	*Approved by:*
0	*Michael Guthrie*	*Michael May*	*Greg Harttraft*	*Don Warfel*
Date				

Summary of Revisions:

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

TABLE OF CONTENTS

1.0 PURPOSE AND METHODOLOGY

1.1 Purpose

The purpose of this Program Basis Document is to document and evaluate those activities of the Oyster Creek BWR Vessel Internals aging management program that are credited for managing cracking initiation and growth in reactor internal components 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 also provides a comparison of the credited Oyster Creek program with the elements of the corresponding NUREG-1801 Chapter XI program XI.M9 - BWR Vessel Internals. 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>

The program includes:

(a) inspection and flaw evaluation in conformance with the guidelines of applicable and staff-approved boiling water reactor vessel and internals project (BWRVIP) documents, and

(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 internal components.

<u>Oyster Creek:</u>

The Oyster Creek BWR Vessel Internals program includes:

a) inspection and flaw evaluation, performed in conformance with the guidelines of applicable staff-approved Boiling Water Reactor Vessel and Internals Project (BWRVIP) documents, as well as the requirements of Section XI of the ASME Boiler and Pressure Vessel Code **(Reference OC-5 Program Plan, Reactor Internals Program, Inspection Plans 7.1 – 7.15)**, and

b) monitoring and control of reactor coolant water chemistry, performed in accordance with BWRVIP-130: "BWR Vessel and Internals Project BWR Water Chemistry Guidelines (EPRI TR-1008192)", which is the 2004 revision of "BWR Water

Chemistry Guidelines," **(Reference: OC-5, Program Plan, Reactor Internals Program, Paragraph 1.3).**

2.2 Overall NUREG-1801 Consistency

The Oyster Creek BWR Vessel Internals program is an existing program that is consistent with NUREG-1801 aging management program XI.M9 - BWR Vessel Internals with the exceptions and enhancements listed below.

2.3 Summary of Exceptions to NUREG-1801

NUREG 1801 indicates that water chemistry control is in accordance with BWRVIP 29 for water chemistry in BWRs. BWRVIP 29 references the 1993 revision of EPRI TR 103515, "BWR Water Chemistry Guidelines." The 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.

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 Program Enhancements

The Oyster Creek BWR Vessel Internals program will be enhanced to include inspections of the steam dryer in accordance with BWRVIP-139.

The inspections recommended in NUREG-1801 program XI.M9 for the top guide will be added to the program. The fluence at the top guide has exceeded the IASCC threshold ($5x10^{20}$ n/cm2, E>1 MeV). For those locations that have exceeded this threshold, after entering the period of extended operation ten (10) percent of the top guide locations will be inspected using enhanced visual inspection technique, EVT-1, within 12 years, with one-half of the inspections (5 percent of locations) to be completed within 6 years after entering the period of extended operation.

Oyster Creek has roll repaired two leaking CRD stub tubes. Oyster Creek is pursuing a code case within the ASME to make these roll expansion repairs permanent. Once the ASME and the NRC approve the code case, the Oyster Creek BWR Reactor Internals program will be revised to make these repairs permanent. If the code case is not approved, the program will be changed to use a permanent repair acceptable to the NRC.

The existing Oyster Creek BWR Vessel Internals program is found to be consistent with NUREG-1801, with the exceptions and enhancements described above, and is found adequate to assure that the reactor vessel internal components will continue to perform their intended functions consistent with the current licensing basis through the period of extended operation.

3.0 EVALUATIONS AND TECHNICAL BASIS

Note

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

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

3.0 Scope of Program

NUREG-1801:

The program is focused on managing the effects of cracking due to stress corrosion cracking (SCC), intergranular stress corrosion cracking (IGSCC), or irradiation assisted stress corrosion cracking (IASCC). The program contains preventive measures to mitigate SCC, IGSCC, or IASCC, inservice inspection (ISI) to 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 of BWR vessel internals.

The BWRVIP documents provide generic guidelines intended to *present the applicable inspection recommendations to assure safety function integrity of the subject safety-related reactor pressure vessel internal components. The guidelines include information on component description and function; evaluate susceptible locations and safety consequences of failure; provide recommendations for methods, extent, and frequency of inspection; discuss acceptable methods for evaluating the structural integrity significance of flaws detected during these examinations; and recommend repair and replacement procedures.*

The various applicable BWRVIP guidelines are as follows:

a) Core shroud: BWRVIPs-07, -63, and -76 provide guidelines for inspection and evaluation; BWRVIP-02, Rev. 2, provides guidelines for repair design criteria.

b) Core plate: BWRVIP-25 provides guidelines for inspection and evaluation; BWRVIP-50 provides guidelines for repair design criteria.

c) Shroud support: BWRVIP-38 provides guidelines for inspection and evaluation; BWRVIP-52 provides guidelines for repair design criteria.

d) Low-pressure coolant injection (LPCI) coupling: BWRVIP-42 provides guidelines for inspection and evaluation; BWRVIP-56 provides guidelines for repair design criteria.

e) Top guide: BWRVIP-26 provides guidelines for inspection and evaluation; BWRVIP-50 provides guidelines for repair design criteria. Additionally, for top guides with neutron fluence exceeding the IASCC threshold (5E20, E>l MeV) prior to the period of extended operation, inspect five percent (5%) of the top guide locations using enhanced visual inspection technique, EVT-1 within six years after entering the period of extended operation. An additional 5% of the top guide locations will be inspected within twelve years after entering the period of extended operation. Alternatively, if the neutron fluence for the limiting top guide location is projected to exceed the threshold for IASCC after entering the period of extended operation, inspect 5% of the top guide locations (EVT-1) within six years after the date projected for exceeding the threshold. An additional 5% of the top guide locations will be inspected within twelve years after the date projected for exceeding the threshold. The top guide inspection locations are those that have high neutron fluence exceeding the IASCC threshold. The extent of the examination and its frequency will be based on a ten percent sample of the total population, which includes all *grid beam and beam-to-beam crevice slots.*

f) Core spray: BWRVIP-18 provides guidelines for inspection and evaluation; BWRVIP-16 and 19 provides guidelines for replacement and repair design criteria, respectively.

g) Jet pump assembly: BWRVIP-41 provides guidelines for inspection and evaluation; BWRVIP-51 provides guidelines for repair design criteria.

h) Control rod drive (CRD) housing: BWRVIP-47 provides guidelines for inspection and evaluation; BWRVIP-58 provides guidelines for repair design criteria.

i) Lower plenum: BWRVIP-47 provides guidelines for inspection and evaluation; BWRVIP-57 provides guidelines for repair design criteria for instrument penetrations.

j) In addition, BWRVIP-44 provides guidelines for weld repair of nickel alloys; BWRVIP-45 provides guidelines for weldability of irradiated structural components.

Oyster Creek:

The Oyster Creek BWR Vessel Internals program manages the effects of cracking due to stress corrosion cracking (SCC), intergranular stress corrosion cracking (IGSCC), and irradiation-assisted stress corrosion cracking (IASCC). The program includes measures to prevent or mitigate SCC, IGSCC, and IASCC; inservice inspection activities (ISI) to monitor the effects of cracking on the intended function of the components; and repair and/or replacement activities as needed to maintain the ability of BWR vessel internal components to perform their intended functions.

Adhering to the requirements of the Oyster Creek BWR Water Chemistry program reduces the susceptibility of the reactor internals components to cracking due to SCC or IGSCC as described paragraph 3.2 below. The applicable and approved BWRVIP inspection and evaluation documents are used at Oyster Creek to provide inspection recommendations to assure safety function integrity of the subject safety-related reactor pressure vessel internal components. The Oyster Creek BWR Vessel Internals program uses the guidance of the BWRVIP documents for evaluation of susceptible locations. The program also utilizes the recommendations for methods, extent, and frequency of inspection and acceptable methods for evaluating the structural integrity significance of flaws detected during these examinations. The applicable BWRVIP repair documents are also used to prepare repair and/or replacement procedures.

The Oyster Creek BWR Vessel Internals program uses the following BWRVIP guidelines to prevent or mitigate cracking in the components specified below:

a) Core Shroud: Inspection and Flaw Evaluation: BWRVIP-76 and BWRVIP-104 **(Reference: OC-5, Program Plan – Reactor Internals, Inspection Plan 7.3, Core Shroud)**; Repair: BWRVIP-02 **(Reference: ER-AB-331-1001, Paragraph 3.1.2, Repair Design Guidelines)** (Note: BWRVIP-07 and BWRVIP-63 have been superseded by BWRVIP-76).

b) Core Plate: Inspection and Flaw Evaluation: BWRVIP-25 **(Reference: OC-5, Program Plan – Reactor Internals, Inspection Plan 7.4, Core Plate)**; Repair: BWRVIP-50 **(Reference: ER-AB-331-1001, Paragraph 3.1.2, Repair Design Guidelines)**.

c) Shroud Support: Inspection and Evaluation: BWRVIP-38 and BWRVIP-104 **(Reference: OC-5, Program Plan – Reactor Internals, Inspection Plan 7.5, Shroud Support)**; Repair: BWRVIP-52 **(Reference: ER-AB-331-1001, Paragraph 3.1.2, Repair Design Guidelines)**.

d) BWRVIP-42 is not applicable since no LPCI Coupling is present at Oyster Creek since it is a BWR-2 design.

e) Top Guide: Inspection and Evaluation: BWRVIP-26 **(Reference: OC-5, Program Plan – Reactor Internals, Inspection Plan 7.2, Top Guide)**; Repair: BWRVIP-50 **(Reference: ER-AB-331-1001, Paragraph 3.1.2, Repair Design Guidelines)**. The top guide inspection locations include those that have neutron fluence exceeding the IASCC threshold of 5×10^{20} n/cm^2. The OC Reactor Internals program will inspect ten (10) percent of the top guide locations after entering the period of extended operation, including all grid beam and beam-to-beam crevice slots, using enhanced visual inspection technique, EVT-1 within 12 years, with one-half of the inspections (5 percent of locations) to be completed within 6 years after entering the period of extended operation. The alternative inspection schedule suggested in NUREG 1801 XI.M9 for the top guide is not applicable to Oyster Creek, since the fluence threshold has already been exceeded for the top guide components.

f) Core Spray: Inspection and Evaluation: BWRVIP-18 **(Reference: OC-5, Program Plan – Reactor Internals, Inspection Plan 7.1, Core Spray Spargers and Annulus Piping)**; Repair: BWRVIP-19; Replacement: BWRVIP-16

(Reference: ER-AB-331-1001, Paragraph 3.1.2, Repair Design Guidelines).

g) BWRVIP-41 is not applicable since no jet pumps exist at Oyster Creek since it is a BWR-2 design.

h) Control Rod Drive housing: Inspection and Evaluation: BWRVIP-47 **(Reference: OC-5, Program Plan – Reactor Internals, Inspection Plan 7.14, Stub Tubes, CRD Housings)**; Repair: BWRVIP-58 **(Reference: ER-AB-331-1001, Paragraph 3.1.2, Repair Design Guidelines).**

i) Lower Plenum: Inspection and Evaluation: BWRVIP-47 and BWRVIP-17 **(Reference: OC-5, Program Plan – Reactor Internals, Inspection Plan 7.14, Stub Tubes, CRD Housings;** Repair: BWRVIP-57 **(Reference: ER-AB-331-1001, Paragraph 3.1.2, Repair Design Guidelines).**

j) Repair and replacement activities, if needed, are performed in accordance with ASME Section XI requirements, consistent with the recommendations of the appropriate BWRVIP repair/replacement guidelines For nickel alloy repairs: BWRVIP-44 would; for weld repairs of irradiated structural components BWRVIP-45 would be used. **(Reference: Procedure ER-AB-331, "BWR RX Internals Management Program Activities").**

k) The Oyster Creek BWR Vessel Internals program also inspects the following Steam Dryers and Feedwater Sparger to manage cracking, using ASME Section XI inspection requirements and BWRVIP guidelines where applicable (References: OC-5, Inspection Plans 7.10 and 7.12)

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

None.

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

The Oyster Creek BWR Vessel Internals program will be enhanced to include inspections of the steam dryer in accordance with BWRVIP-139.

The inspections recommended in NUREG-1801 program XI.M9 for the top guide will be added to the program. The fluence at the top guide has exceeded the IASCC threshold ($5x10^{20}$ n/cm2, E>1 MeV). For those locations that have exceeded this threshold, ten

(10) percent of the top guide locations will be inspected after entering period of extended operation using enhanced visual inspection technique, EVT-1, within 12 years, with one-half of the inspections (5 percent of locations) to be completed within 6 years after entering the period of extended operation.

Oyster Creek has roll repaired two leaking CRD stub tubes. Oyster Creek is pursuing a code case within the ASME to make these roll expansion repairs permanent. Once the ASME and the NRC approve the code case, the Oyster Creek BWR Reactor Internals program will be revised to make these repairs permanent. If the code case is not approved, the program will be changed to use a permanent repair acceptable to the NRC.

Comparison and Evaluation Conclusion:

With enhancements, this program element is consistent with NUREG-1801, Element 1, Scope of Program.

3.1 Preventive Actions

NUREG-1801:

Maintaining high water purity reduces susceptibility to cracking due 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 the evaluation and technical basis of monitoring and maintaining reactor water chemistry are presented in Chapter XI.M2, "Water Chemistry."

Oyster Creek:
The Oyster Creek water chemistry programs are based on BWRVIP-130: "BWR Vessel and Internals Project BWR Water Chemistry Guidelines", which replaces BWRVIP-29. Water chemistry monitoring and maintenance are implemented through the station's water chemistry procedures **(Reference: OC-5, Program Plan, Reactor Internals Program, paragraph 1.3).**

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", 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 program element is consistent with NUREG-1801, Element 2, Preventive Actions, with the exception noted.

3.2 Parameters Monitored or Inspected

NUREG-1801:

a. The program monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by inspection in accordance with the guidelines of applicable and approved BWRVIP documents and the requirements of the American Society of Mechanical Engineers (ASME) Code, Section XI, Table IWB 2500-1 (2001 edition[1] 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 such relief is submitted under the provisions of 10 CFR 50.55a and approved by the staff.

Oyster Creek:

a) The Oyster Creek BWR Vessel Internals program monitors the effects of cracking on the intended function of internals components through detection and sizing of cracks by inspections performed in accordance with the guidelines of applicable and approved BWRVIP documents, as well as the requirements of ASME Section XI. The inspections are performed consistent with the recommendations of applicable and approved BWRVIP guidelines (as described in paragraph 2.1 above), as well as the requirements of ASME Section XI, Table IWB 2500-1 (1995 addition through 1996 addenda). **(Reference: OC-5, Inspection Plans 7.1 – 7.15).**

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

b) The Oyster Creek plant uses hydrogen water chemistry but has not requested inspection relief under the provisions of 10 CFR 50.55a using BWRVIP-62 guidelines. If inspection relief is needed in the future, the guidance in BWRVIP-62 will be utilized and an appropriate relief request will be submitted under the provisions of 10 CFR 50.55a for staff approval.

Exceptions to NUREG-1801, Element 3:

NUREG-1801 program XI.M9 references ASME Section XI, Table IWB 2500-1 (2001 edition, including the 2002 and 2003 Addenda). The Oyster Creek ISI program is based the 1995 Edition, (including 1996 Addenda) of ASME Section XI. The program is updated in accordance with 10CFR 505.55a after each 10-year interval.

Enhancements to NUREG-1801, Element 3:

None.

Comparison and Evaluation Conclusion:

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

3.3 Detection of Aging Effects

NUREG-1801:

a) The extent and schedule of the inspection and test techniques prescribed by the applicable and approved BWRVIP guidelines are designed to maintain structural integrity and ensure that aging effects will be discovered and repaired before the loss of intended function of BWR vessel internals. Inspection can reveal cracking. Vessel internal components are inspected in accordance with the requirements of ASME Section XI, Subsection IWB, examination category B-N-2.

b) The ASME Section XI inspection specifies visual VT-1 examination to detect discontinuities and imperfections, such as cracks, corrosion, wear, or erosion, on the surfaces of components. This inspection also specifies visual VT-3 examination to determine the general mechanical and structural condition of the component supports by (i) verifying parameters, such as clearances, settings, and physical displacements, and (ii) detecting discontinuities and imperfections, such as loss of *integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion.*

c) *The applicable and approved BWRVIP guidelines recommend more stringent inspections, such as enhanced visual VT-1 examinations or ultrasonic methods of volumetric inspection, for certain selected components and locations.*

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

Oyster Creek:

a) The Oyster Creek Vessel Internals program utilizes inspections and test techniques prescribed by the applicable and approved BWRVIP guidelines to maintain structural integrity of BWR vessel internal components and to assure that aging effects will be discovered and repaired before the loss of intended function occurs. By following the BWRVIP guidelines, Oyster Creek also complies with ASME Section XI requirements. Welded core supports, welded attachments, and other reactor internal components are inspected in accordance with the requirements of ASME Section XI, Subsection IWB, examination category B-N-2 **(Reference: OC-1, Oyster Creek Inservice Inspection Program, Table 2.2-21).**

b) ASME Section XI specifies VT-1 examinations to detect surface discontinuities and imperfections, cracks, corrosion, wear, or erosion. VT-3 examinations are specified to determine the general condition of component supports by verifying parameters, such as clearances and displacements, and by detecting discontinuities and imperfections, such as loss of integrity of bolted or welded connections, or loose or missing parts, debris, corrosion, wear, or erosion. Prior to each outage, the necessary inspections are determined based on the BWRVIP guidelines and the ASME Code. The examination procedures also identify the type and location of examination (VT-1, EVT-1, VT-3) required for each component as well as the basis for the inspection (BWRVIP, ASME Code or design requirement) **(References: OC-5, Program Plan – Reactor Internals Program, Inspection Plans 7.01 – 7.15; ER-AA-300, Conduct of Inservice Inspection Activities and ER-AA-300-002, Inservice Inspection of Section XI Welds and Components).**

c) The recommendations applicable and approved BWRVIP guidelines, which often recommend more stringent inspections, are used for inspecting reactor internals. **(References: OC-5, Program Plan – paragraph 1.9, Inspection Plans 7.10 – 7.15 and Attachment B).**

d) The appropriate nondestructive examination (NDE) techniques used for inspection of BWR vessel internals, comply with requirements of BWRVIP-03, including the uncertainties inherent in delivering and executing NDE techniques in a BWR **(Reference: OC-5, Program Plan – paragraph 1.9).**

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

None.

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

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

3.4 Monitoring and Trending

<u>NUREG-1801:</u>

a) *Inspections scheduled in accordance with the applicable and approved BWRVIP guidelines provide timely detection of cracks.*

b) *The scope of examination and reinspection must be expanded beyond the baseline inspection if flaws are detected.*

<u>Oyster Creek:</u>

a) Inspections are scheduled and performed in accordance with the applicable and approved BWRVIP guidelines, and are based upon the results from previous inservice inspections.

b) If flaws are detected, the inspection scope is expanded in accordance with the requirements of ASME Section XI and the appropriate BWRVIP guideline **(References: OC-5, Program Plan, Reactor Internals Program, Inspection Plans 7.1 – 7.15; ER-AA-330-002).**

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. Any indication detected is evaluated in accordance with ASME Section XI or the applicable staff-approved BWRVIP guidelines.

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

Oyster Creek:

a) Evaluation of indications is conducted consistent with the applicable and approved BWRVIP guideline or ASME Section XI, as appropriate. **(Reference: OC-5, Program Plan, Reactor Internals Program, paragraph 1.6)**

b) Flaws are evaluated in accordance with the guidance provided in the associated BWRVIP guidelines or ASME Section XI, as appropriate for a particular component. Additional general guidelines for flaw evaluation of crack growth in stainless steels (SS), nickel alloys, and low-alloy steels is found in BWRVIP-14, BWRVIP-59, and BWRVIP-60. **(Reference: ER-AB-331-1001, paragraph 3.1.3; and OC-5, Program Plan, Reactor Internals Program, 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.

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. *Repair and replacement procedures are equivalent to those requirements in ASME Section XI. Repair and replacement is performed in conformance with the applicable and approved BWRVIP guidelines listed above.*

b. *As discussed in the appendix to this report, the staff finds that licensee implementation of the guidelines in the staff-approved BWRVIP reports 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. Repairs and replacements are performed in accordance with ASME Section XI requirements and in conformance with the applicable and staff-approved BWRVIP guidelines listed in paragraph 2.1 above. **(References: OC-5, Program Plan, Reactor Internals Program, paragraph 1.7 and ER-AB-331, BWR Reactor Internals Management Program Activities).**

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>

None.

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

None.

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

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

3.7 Confirmation Process

<u>NUREG-1801:</u>

Site quality assurance (QA) procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, the staff finds that licensee implementation of the guidelines in the staff-approved BWRVIP reports 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.

<u>Oyster Creek:</u>

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

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:

a) *Extensive cracking has been observed in core shrouds at both horizontal (Nuclear Regulatory Commission [NRC] Generic Letter [GL] 94-03) and vertical (NRC Information Notice [IN] 97-17) welds. It has affected shrouds fabricated from Type 304* *and Type 304L SS, which is generally considered to be more resistant to SCC. Weld regions are most susceptible to SCC, although it is not clear whether this is due to sensitization and/or impurities associated with the welds or the high residual stresses in the weld regions. This experience is reviewed in NRC GL 94-03 and NUREG-1544; some experiences with visual inspections are discussed in NRC IN 94-42.*

b) Both circumferential (NRC IN 88-03) and radial cracking (NRC IN 92-57) has been observed in the shroud support access hole cover made from Alloy 600.

c) Instances of cracking in core spray spargers have been reviewed in NRC Bulletin 80-13.

d) Cracking of the core plate has not been reported, but the creviced regions beneath the plate are difficult to inspect.

e) NRC IN 95-17 discusses cracking in top guides of United States and overseas BWRs. Related experience in other components is reviewed in NRC GL 94-03 and NUREG-1544. Cracking has also been observed in the top guide of a Swedish BWR.

f) Instances of cracking have occurred in the jet pump assembly (NRC Bulletin 80-07), hold-down beam (NRC IN 93-101), and jet pump riser pipe elbows (NRC IN 97-02).

g) Cracking of dry tubes has been observed at 14 or more BWRs. The cracking is intergranular and has been observed in dry tubes without apparent sensitization, suggesting that IASCC may also play a role in the cracking.

h) The program guidelines outlined in applicable and approved BWRVIP documents are based on an evaluation of available information, including BWR inspection data and information on the elements that cause SCC, IGSCC, or IASCC, to determine which components may be susceptible to cracking. Implementation of the program provides reasonable assurance that cracking will be adequately managed so the intended functions of the vessel internal components will be maintained consistent with the current licensing basis (CLB) for the period of extended operation.

<u>Oyster Creek:</u>

a) Review of industry experience has confirmed that cracking has been observed in core shrouds at both horizontal (Nuclear Regulatory Commission [NRC] Generic Letter [GL] 94-03) and vertical (NRC Information Notice [IN] 97-17) welds. It has affected shrouds fabricated from Type 304 and Type 304L SS. In 1994 Oyster Creek discovered signiifcant cracking in the core shroud H4 circumferential weld. Additional minor cracking was found in the H2 and H6 welds. The shroud inspection was comprehensive as recommendation the BWRVIP industry committee. The examinations consisted of visual examinations with cleaning and UT exams wherever practical. During the same refueling outage shroud repair hardware was installed to ensure the shroud could continue to perform its intended function. The repair consisted of 10 tie rods anchored at the top and bottom of the shroud. Subsequent inspections of the repair hardware have confirmed that the tie rods are in good condition and continue to provide reliable structural support for the shroud. Inspections of shroud vertical welds completed in1998 and 2000 have confirmed that the Reactor water chemistry program mitigation efforts have been successful, as no new crack indications have been observed. **(Reference: TDR-1211, paragraph 3.15)**

b) Both circumferential (NRC IN 88-03) and radial cracking (NRC IN 92-57) has been observed in the shroud support access hole cover made from Alloy 600 at some BWR plants. The Oyster Creek BWR-2 design does not have an access plate in the shroud support ring, therefore this operating experience for this component is not applicable to Oyster Creek

c) Instances of cracking in BWR core spray spargers have been reviewed in NRC Bulletin 80-13. In 1978 Oyster Creek identified crack indications in the Core Spray Spargers. Mechanical clamps were installed to provide structural support for known and suspected cracks in the core spray sparger. Recent inspections in 1998, 2000, 2002, and 2004 have confirmed that the repair clamps are in good condition. Inspection of the core spray piping welds has confirmed that the mitigation efforts provided by the Reactor Water Chemistry program have been successful, as no new crack indications have been found. **(Reference: TDR-1211, paragraph 3.1).**

d) Because the core plate is subject relatively high neutron fluence cracking of the core plate has been identified as a potential issue, although cracking has not been observed in a BWR to date. NRC IN 95-17 discusses cracking in top guides of United States and overseas BWRs. Related experience in other components is reviewed in NRC GL 94-03 and NUREG-1544. Ring-to-plate weld cracking was reported in GENE SIL-588 at a foreign utility.

In 1996 (1R16) Oyster Creek inspected 18 of the 36 hold down bolts (top only). The underside of the core support plate is not accessible unless major disassembly of fuel cells is performed. Core support plate wedges (lateral restraints) were designed and installed during the 17R Outage by GE. The reason for installing the wedges is because the material condition of the core support plate cannot be determined without major fuel component disassembly. Eight (8) wedges were installed between the OD of the core support plate and the I.D. of the shroud. The wedges are installed around the core support plate at the following azimuth locations: 24 degrees, 60 degrees, 96 degrees, 132 degrees, 204 degrees, 240 degrees, 276 degrees and 312 degrees. This modification not only provides horizontal support for the core plate, it also assures control rod insertion even if the hold down bolts fail. The wedges will keep the core plate aligned vertically during design basis loading conditions, even if the core support plate lifts upward a small amount. Therefore, since control rod insertion is assured by the installation of wedges, the need for inspection of the core plate to assure its integrity is not required from a nuclear safety point of view. The installation of the wedges also eliminates the need to inspect the core support plate hold-down bolts. A baseline inspection of the core support wedges using VT-3 Performed VT-3 was performed in 2000 to verify that the wedges are in place. **(Reference: TDR-1211, paragraph 3.5).**

e) NRC IN 95-17 discusses cracking in top guides of United States and overseas BWRs. Related experience in other components is reviewed in NRC GL 94-03 and NUREG-1544. Cracking has also been observed in the top guide of a Swedish BWR. In the 1991 refueling outage Oyster Creek found a cracked beam on the underside of the top guide. Additional cracked beams were discovered during 1992 and 1994. The top guide is visually inspected every outage to monitor crack growth. The results of the 2004 inspection indicate that there is no new crack growth, which indicates that the Oyster Creek BWR Reactor Water Chemistry program mitigation efforts are effective in mitigating the effects of cracking in the top guide. **(Reference: TDR-1211, paragraph 3.10).**

f) Instances of cracking have occurred in the jet pump assembly (NRC Bulletin 80-07), hold-down beam (NRC IN 93-101), and jet pump riser pipe elbows (NRC IN 97-02). The Oyster Creek

BWR-2 design does not have jet pumps; therefore this operating experience for this component is not applicable to Oyster Creek

g) Cracking of dry tubes has been observed at several BWRs. The cracking is intergranular and has been observed in dry tubes without apparent sensitization, suggesting that IASCC may also play a role in the cracking. Cracking in the instrumentation dry tubes was first observed at Oyster Creek in 1984. In 1984 eight of the 12 original dry tubes were found with cracking in the non-pressure retaining plunger shaft area, which is a crevice condition. In 1984, 2 of the 12 dry tubes (IRM-14 and IRM-18) were replaced with new redesigned non-crevice single piece plunger shaft. The other 6 cracked tubes were technically justified as acceptable for one additional cycle of operation. In 1986, the balance of 12 dry tubes was replaced with the same redesigned crevice-free single piece plunger shaft. In 1988 IRM-17 was replaced again due to a bent tip that was observed the previous outage. Inspections were performed in 1991, 1992, 1994, 1996, and 1998 with no findings. In 2000, visually inspected (VT-1) five dry tubes: (IRMs 16, 17, 18 and SRMs 21 and 24). IRM-17 was found slightly bent at the top and was dispositioned to use-as-is. Similarly dry tube inspections in 2004 (1R20) identified degradation in SRM-24, and IRM-17, and IRM-18. These incores are scheduled to be replaced in 2006 **(References: TDR-1211, paragraph 3.13; CAP #2000-1561; CAP-2004-3694).**

h) The Oyster Creek Reactor Internals program utilizing the BWRVIP guidelines to perform inspections, evaluate flaws, and made repairs where necessary. The applicable and approved BWRVIP documents are based on an evaluation of available information, including BWR inspection data and information on the elements that cause SCC, IGSCC, or IASCC, to determine which components may be susceptible to cracking. Implementation of the program provides reasonable assurance that cracking will be adequately managed so the intended functions of the vessel internal components will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. The experience at Oyster Creek with the BWR Vessel Internals program shows that the BWR Vessel Internals program is effective in managing cracking initiation and growth.

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

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

Demonstration that the effects of aging are effectively managed is achieved through objective evidence that shows that cracking initiation and growth is being adequately managed in BWR vessel internals. In addition to the examples described above the following example of operating experience provides additional objective evidence that the BWR Vessel Internals program is effective in assuring that intended function(s) will be maintained consistent with the CLB for the period of extended operation:

Oyster Creek has been inspecting the steam dryer every refueling outage for many years. Cracks were first identified on a lower bank brace in 1983. Weld repairs of the lower bank brace were made in 1983 and again in 1986. A different repair method, "stop drilling", was implemented in 1996 to mitigate the cracks. Subsequent inspections indicate these measures have been successful in arresting crack growth. In 2002 minor indications on a lifting lug and a section of the skirt were found. These indications did not affect the structural integrity of the steam dryer and were dispositioned by engineering analysis to use-as-is.

These examples provide objective evidence that cracking will be detected prior to the loss of intended functions. The BWR Reactor Internals program at Oyster Creek was recently enhanced in a few key areas based on recommendations identified during a 2003 self-assessment and as a result of an INPO BWRVIP Review Visit performed in late 2003. Two examples of these enhancements are (1) that periodic inspections have been added for the shroud lug-clevis assemblies and (2) monthly moisture carryover readings are now taken to monitor Steam Dryer performance. Oyster Creek monitors quarterly program performance and effectiveness through program health reports and periodically performs self-assessments.

A review of the operating experience of the BWR Vessel Internals program did not indicate any adverse trend in performance. Problems identified would not have caused significant impact to the safe operation of the plant, and adequate corrective actions were taken to prevent recurrence. There is sufficient confidence that the implementation of the BWR Vessel Internals program will effectively determine cracking initiation and growth. Appropriate guidance for reevaluation, repair or replacement is provided for locations where cracking initiation and growth. Periodic self-assessments of the BWR Vessel Internals program are performed to identify the areas that need improvement to maintain the quality performance of the program.

3.10 Conclusion

The Oyster Creek BWR Vessel Internals aging management program is credited for managing cracking initiation and growth for the systems, components, and environments listed in Table 5.2. The Oyster Creek BWR Vessel Internals program's elements have been evaluated against NUREG-1801 in Section 3.0. Program exceptions have been identified in Section 2.3. Program enhancements have been identified in Section 2.4. The implementing documents for this aging management program are listed in Table 5.1. The relevant operating experience has been reviewed and a demonstration of program effectiveness is provided in Section 3.10.

Based on the above, the continued implementation of the Oyster Creek BWR Vessel Internals aging management program provides reasonable assurance that cracking initiation and growth in reactor internals 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-02-A, BWR Core Shroud Repair Design Criteria, Revision 2.

4.2.2 BWRVIP-03, Reactor Pressure Vessel and Internals Examination Guidelines, Revision 6.

4.2.3 BWRVIP-14-A, Evaluation of Crack Growth in BWR Stainless Steel RPV Internals.

4.2.4 BWRVIP-16-A, Internal Core Spray Piping and Sparger Replacement Design Criteria.

4.2.5 BWRVIP-17, Roll / Expansion Repair of Control Rod Drive and In-Core Instrument Penetrations in BWR Vessels.

4.2.6 BWRVIP-18-A, BWR Core Spray Inspection and Flaw Evaluation Guidelines.

4.2.7 BWRVIP-19-A, Internal Core Spray Piping and Sparger Repair Design Criteria.

4.2.8 BWRVIP-25, BWR Core Plate Inspection and Flaw Evaluation Guidelines.

4.2.9 BWRVIP-26, BWR Top Guide Inspection and Flaw Evaluation Guidelines.

4.2.10 BWRVIP-27-A, Standby Liquid Control System/Core Spray/Core Plate Delta-P Inspection and Flaw Evaluation Guidelines.

4.2.11 BWRVIP-29, BWR Water Chemistry Guidelines – 1996 Revision.

4.2.12 BWRVIP-38, BWR Shroud Support Inspection and Flaw Evaluation Guidelines.

4.2.13 BWRVIP-44-A, Underwater Weld Repair of Nickel Alloy Reactor Vessel Internals.

4.2.14 BWRVIP-45, Weldability of Irradiated LWR Structural Components.

4.2.15 BWRVIP-47, BWR Lower Plenum Inspection and Flaw Evaluation Guidelines.

4.2.16 BWRVIP-48, Inspection and Flaw Evaluation Guidelines.

4.2.17 BWRVIP-49-A, Instrument Penetration Inspection and Flaw Evaluation Guidelines.

4.2.18 BWRVIP-50-A, Top Guide / Core Plate Repair Design Criteria.

4.2.19 BWRVIP-52-A, Shroud Support and Vessel Bracket Repair Design Criteria.

4.2.20 BWRVIP-57-A, Instrument Penetration Repair Design Criteria.

4.2.21 BWRVIP-58-A, CRC Internal Access Weld Repair.

4.2.22 BWRVIP-59, Evaluation of Crack Growth in BWR Nickel Base Austenitic Alloys in RPV Internals.

4.2.23 BWRVIP-60-A, Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment.

4.2.24 BWRVIP-62, Technical Basis for Inspection Relief for BWR Internal Components with Hydrogen Injection.

4.2.25 BWRVIP-74-A, BWR Reactor Pressure Vessel Inspection

and Flaw Evaluation Guidelines.

4.2.26 BWRVIP-76, BWR Core Shroud Inspection and Flaw Evaluation Guidelines.

4.2.27 BWRVIP-104, Evaluation and Recommendations to Address Shroud Support Cracking in BWRs.

4.2.28 BWRVIP-130, BWR Water Chemistry Guidelines, 2004 Revision.

4.2.29 BWRVIP-139, Steam Dryer Inspection and Flaw Evaluation Guideline.

4.2.30 NRC Generic Letter 94-03, Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors, U.S. Nuclear Regulatory Commission, July 25, 1994.

4.2.31 NRC Bulletin No. 80-13, *Cracking in Core Spray Spargers*, U.S. Nuclear Regulatory Commission, May 12, 1980.

4.2.32 NRC Information Notice 94-42, *Cracking in the Lower Region of the Core Shroud in Boiling Water Reactors*, U.S. Nuclear Regulatory Commission, June 7, 1994.

4.2.33 NRC Information Notice 95-17, *Reactor Vessel Top Guide and Core Plate Cracking*, U.S. Nuclear Regulatory Commission, March 10, 1995.

4.2.34 NRC Information Notice 97-17, Cracking of Vertical Welds in the Core Shroud and Degraded Repair, U.S. Nuclear Regulatory Commission, April 4, 1997.

4.3 Oyster Creek Program References

4.3.1 OC-1, Inservice Inspection Program.

4.3.2 OC-5, Program Plan, Reactor Internals Program.

4.3.3 TDR-1211, "Oyster Creek Reactor Internals Program Basis Document", 12/2003

5.0 TABLES

5.1 Aging Management Program Implementing Documents

Procedure Number	Procedure Title	Commitment No.	Status
CY-AB-120-100	Reactor Water Chemistry	330592.09.04	ACC/ASG
ER-AA-330-002	Inservice Inspection of Section XI Welds and Components	330592.09.05	ACC/ASG
ER-AB-331	BWR Rx Internals Mgmt Program Activities	330592.09.02	ACC/ASG
ER-AB-331-1001	RX Internals	330592.09.01	ACC/ASG
OC-1	Oyster Creek 10-year ISI Program	330592.09.06	ACC/ASG
OC-5	Reactor Internals Program Plan	330592.09.03	ACC/ASG

5.2 Aging Management Review Results

SSC Name	Structure and/or Component	Material	Environment	Aging Effect
Reactor Internals	Control Rod Drive Assembly (Housing and Guide Tube)	Stainless Steel	Treated Water	Cracking Initiation and Growth
Reactor Internals	Control Rod Drive Assembly (Housing and Guide Tube)	CASS	Treated Water >482F (Internal)	Cracking Initiation and Growth
Reactor Internals	Control Rod Drive Assembly (Housing and Guide Tube)	Stainless Steel	Treated Water	Cracking Initiation and Growth
Reactor Internals	Core Plate (Lower Core Grid)	Stainless Steel	Treated Water	Cracking Initiation and Growth
Reactor Internals	Core Plate (Lower Core Grid) Wedges	Nickel Alloy	Treated Water	Cracking Initiation and Growth
Reactor Internals	Core Shroud	Stainless Steel	Treated Water	Cracking Initiation and Growth
Reactor Internals	Core Shroud	Stainless Steel	Treated Water	Cracking Initiation and Growth
Reactor Internals	Core Spray Line Spray Nozzle Elbows	CASS	Treated Water >482F	Cracking Initiation and Growth
Reactor Internals	Core Spray Lines, Thermal Sleeves, Spray Rings (Sparger), and Spray Nozzles	Stainless Steel	Treated Water	Cracking Initiation and Growth
Reactor Internals	Core Spray Lines, Thermal Sleeves, Spray Rings (Sparger), and Spray Nozzles	Stainless Steel	Treated Water	Cracking Initiation and Growth
Reactor Internals	Core Spray Lines, Thermal Sleeves, Spray Rings (Sparger), and Spray Nozzles	Stainless Steel	Treated Water	Cracking Initiation and Growth
Reactor Internals	Core Spray Ring (Sparger) Repair Hardware	Stainless Steel	Treated Water	Cracking Initiation and Growth
Reactor Internals	Fuel Support Piece	CASS	Treated Water >482F	Cracking Initiation and Growth
Reactor Internals	Fuel Support Piece	CASS	Treated Water >482F	Cracking Initiation and Growth
Reactor Internals	In-core Neutron Monitor Dry Tubes, Guide Tubes, & Housings	Stainless Steel	Treated Water	Cracking Initiation and Growth
Reactor Internals	Shroud Repairs (tie rods and lug/clevis assemblies)	Stainless Steel	Treated Water	Cracking Initiation and Growth

Reactor Internals	Shroud Repairs (tie rods and lug/clevis assemblies)	Nickel Alloy	Treated Water	Cracking Initiation and Growth
Reactor Internals	Shroud Support Structure	Nickel Alloy	Treated Water	Cracking Initiation and Growth
Reactor Internals	Shroud Support Structure	Nickel Alloy	Treated Water	Cracking Initiation and Growth
Reactor Internals	Top Guide (Upper Core Grid)	Stainless Steel	Treated Water	Cracking Initiation and Growth
Reactor Internals	Vessel Steam Dryer	Stainless Steel	Steam	Cracking Initiation and Growth
Reactor Pressure Vessel	Penetrations (CRD Stub Tubes)	Stainless Steel	Treated Water (Internal)	Cracking Initiation and Growth
Reactor Pressure Vessel	Penetrations (CRD Stub Tubes)	Stainless Steel	Treated Water (Internal)	Cracking Initiation and Growth

6.0 ATTACHMENTS

6.1 LRA Appendix A

6.2 LRA Appendix B