Duke Energy®

T. PRESTON GILLESPIE, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ONO1VP / 7800 Rochester Hwy.
Seneca, SC 29672

864-873-4478
864-873-4208 fax
T.Gillespie@duke-energy.com

November 8, 2010

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington D. C. 20555-0001

Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Site, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Proposed License Amendment Request for the Reactor Vessel Internals Inspection Plan
License Amendment Request Number 2010-06

In accordance with 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy) proposes to amend Renewed Facility Operating Licenses (FOLs) DPR-38, DPR-47, and DPR-55 for Oconee Nuclear Station (ONS), Units 1, 2, and 3. Specifically, Duke Energy requests Nuclear Regulatory Commission (NRC) review and approval for the proposed adoption of the Reactor Vessels (RV) Internals inspection plan based on the use of Materials Reliability Program (MRP) 227, Pressurized Water Reactors Internals Inspection and Evaluation Guidelines.

A letter dated June 16, 2010 was submitted to the NRC stating Duke Energy's intent to adopt MRP-227, Pressurized Water Reactors Internals Inspection and Evaluation Guidelines.

The inspection plan contains a discussion of the background of the Babcock and Wilcox (B&W) designed plant RV Internals programs, first sponsored by the utilities through the B&W Owner's Group (BWOG) and later by the Pressurized Water Reactor Owner's Group (PWROG), culminating in a submittal to the NRC through the Electric Power Research Institute (EPRI) MRP. The ONS inspection plan also contains a discussion of operational experience, time-limited aging analyses (TLAAs), and relevant existing programs.

The RV Internals Aging Management Program (AMP) includes the inspection plan and demonstrates that the program adequately manages the effects of aging for RV Internals components. It also establishes the basis for providing reasonable assurance the RV Internals components will remain functional through the license renewal period of extended operation.

Attachment 1 provides the proposed RV Internals inspection plan. The regulatory commitments associated with the RV Internals inspection plan are provided in Attachment 2.

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Duke Energy requests approval of this LAR by June 30, 2012. The ONS RV Internals inspection plan will be revised as necessary following approval by the NRC. The ONS UFSAR will be updated per 10 CFR 50.71(e) as required.

In accordance with Duke administrative procedures and the Quality Assurance Program Topical Report, these proposed changes have been reviewed and approved by the Plant Operations Review Committee. Additionally, a copy of this LAR is being sent to the State of South Carolina in accordance with 10 CFR 50.91 requirements.

Inquiries on this proposed amendment request should be directed to Kent Alter of the Oconee Regulatory Compliance Group at (864) 873-3255.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 8, 2010.

Sincerely.

T. Gillespie

T. Preston Gillespie, Jr.
Vice President
Oconee Nuclear Station

Enclosure:
1. EVALUATION OF PROPOSED CHANGE

Attachments:
1. INSPECTION PLAN FOR THE OCONEE NUCLEAR STATION UNITS 1, 2, and 3 REACTOR VESSEL INTERNALS
2. REGULATORY COMMITMENTS

bc w/enclosures and attachments:

Mr. Luis Reyes, Regional Administrator
U. S. Nuclear Regulatory Commission - Region II
Marquis One Tower
245 Peachtree Center Ave., NE, Suite 1200
Atlanta, Georgia 30303-1257

Mr. John Stang, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-8 G9A
Washington, D. C. 20555

Mr. Andy Sabisch
Senior Resident Inspector
Oconee Nuclear Site

Susan E. Jenkins, Manager,
Radioactive & Infectious Waste Management
SC Dept. of Health and Env. Control
2600 Bull St.
Columbia, SC 29201

ENCLOSURE 1

EVALUATION OF PROPOSED CHANGE

Subject: Proposed License Amendment Request for the Reactor Vessel Internals Inspection Plan

1.0 SUMMARY DESCRIPTION

In accordance with 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy) proposes to amend Renewed Facility Operating Licenses (FOLs) DPR-38, DPR-47, and DPR-55 for Oconee Nuclear Station (ONS), Units 1, 2, and 3.

The proposed License Amendment Request (LAR) provides the Reactor Vessel (RV) Internals Inspection Plan report. The LAR also provides a description of the inspection plan as it relates to the management of aging effects consistent with previous commitments. The inspection plan is based on MRP-227, Revision 0, "PWR Internals Inspection and Evaluation Guidelines" and describes using the ten Aging Management Program (AMP) elements in the current revision of NUREG-1801 "Generic Aging Lessons Learned" (GALL, Revision 1) report.

The inspection plan contains a discussion of the background of the Babcock and Wilcox (B&W) designed plant RV Internals programs, first sponsored by the utilities through the B&W Owner's Group (BWOG) and later by the Pressurized Water Reactor Owner's Group (PWROG), culminating in a submittal to the NRC through the Electric Power Research Institute (EPRI) Materials Reliability Program (MRP). The ONS inspection plan also contains a discussion of operational experience, time-limited aging analyses (TLAAs), and relevant existing programs.

The RV Internals AMP includes the inspection plan and demonstrates that the program adequately manages the effects of aging for RV Internals components and establishes the basis for providing reasonable assurance the RV Internals components will remain functional through the license renewal period of extended operation.

2.0 DETAILED BACKGROUND

The RV Internals Inspection Plan, located in Attachment 1, provides the detailed background associated with this LAR.

3.0 TECHNICAL EVALUATION

The RV Internals Inspection Plan, located in Attachment 1, provides the technical evaluation for this LAR.

4.0 REGULATORY SAFETY ANALYSIS

4.1 Significant Hazards Consideration

Pursuant to 10 CFR 50.91, Duke has made the determination that this amendment request does not involve a significant hazards consideration by applying the standards established by the NRC regulations in 10 CFR 50.92. This ensures that operation of the facility in accordance with the proposed amendment would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

   No. The proposed license amendment request provides the Reactor Vessel Internals Inspection Plan report. The report also provides a description of the inspection plan as it relates to the management of aging effects consistent with previous commitments. The inspection plan is based on MRP-227, Revision 0, "Pressurized Water Reactors Internals Inspection and Evaluation Guidelines" and describes using the ten Aging Management Program (AMP) elements in the current revision of NUREG-1801 "Generic Aging Lessons Learned" (GALL, Revision 1) report.

The inspection plan contains a discussion of the background of the Babcock and Wilcox designed plant Reactor Vessel Internals programs, first sponsored by the utilities through the Babcock and Wilcox Owner's Group and later by the Pressurized Water Reactor Owner's Group, culminating in a submittal to the Nuclear Regulatory Commission through the Electric Power Research Institute Materials Reliability Program. The inspection plan also contains a discussion of operational experience, time-limited aging analyses, and relevant existing programs.

The Reactor Vessel Internals Aging Management Program includes the inspection plan and demonstrates that the program adequately manages the effects of aging for Reactor Vessel Internals components and establishes the basis for providing reasonable assurance the Reactor Vessel Internals components will remain functional through the license renewal period of extended operation.

This license amendment request provides an inspection plan based on industry work and experiences as agreed to in Duke Energy's license renewal commitments for Reactor Vessel Internals Inspection. It is not an accident initiator; therefore, it will not increase the probability or consequences of an accident previously evaluated

2) Create the possibility of a new or different kind of accident from any accident previously evaluated.

   No. The proposed Reactor Vessel Internals Inspection Plan does not change the methods governing normal plant operation, nor are the methods utilized to respond to plant transients altered. The revised inspection plan is not an accident / event initiator. No new initiating events or transients result from the use of the Reactor Vessel Internals Inspection plan.

3) Involve a significant reduction in a margin of safety.

   No. The proposed safety limits have been preserved. The License Amendment Request requests review and approval for the Reactor Vessel Internals Inspection plan that Duke Energy committed to provide prior to commencing inspections.

4.2 Applicable Regulatory Requirements/Criteria

1. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3," NUREG-1723, March 31, 2000.

2. Letter (from D. Baxter) of Intent to adopt Materials Reliability Program 227, Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, June 16, 2010.

3. UFSAR 18.3.20, Reactor Vessel Internals Inspection

4.3 Precedent

1. Letter from Constellation Energy to Nuclear Regulatory Commission - License Renewal Aging Management Reactor Vessel Internals Program, February 27, 2009.
2. Letter from Progress Energy to Nuclear Regulatory Commission – Reactor Vessel Internals Aging Management Program Inspection Plan, September 24, 2009.

5.0 ENVIRONMENTAL CONSIDERATION

Duke Energy Carolinas, LLC, has evaluated this license amendment request against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. Duke has determined that this license amendment request meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50 that changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or that changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

(i) The amendment involves no significant hazards consideration.

As demonstrated in Section 4.1, the proposed Reactor Vessel Internals Inspection plan does not involve significant hazards consideration.

(ii) There is no significant change in the types or significant increase in the amounts of any effluent that may be released offsite.

The proposed Reactor Vessel Internals Inspection plan will not impact effluents released offsite. Therefore, there will be no significant change in the types or significant increase in the amounts of any effluents released offsite.

(iii) There is no significant increase in individual or cumulative occupational radiation exposure.

The proposed Reactor Vessel Internals Inspection plan will not have an adverse impact on occupational radiation exposure. Therefore, there will be no significant increase in individual or cumulative occupational radiation exposure resulting from this action.

6.0 REFERENCES

The RV Internals Inspection Plan, located in Attachment 1, provides the references associated with this LAR.

ATTACHMENT 1

INSPECTION PLAN FOR THE OCONEE NUCLEAR STATION UNITS 1, 2, and 3 REACTOR VESSEL INTERNALS

ANP-2951
Revision 001
October 2010

Inspection Plan for the Oconee Nuclear Station Units 1, 2, and 3 Reactor Vessel Internals

by
Sarah Davidsaver, AREVA NP, Inc.
and
Rachel Doss, Duke Energy Carolinas, LLC
David Whitaker, Duke Energy Carolinas, LLC
Dorrington Williams, Duke Energy Carolinas, LLC

Prepared for
Duke Energy Carolinas, LLC
Oconee Nuclear Station

AREVA NP, Inc.
3315 Old Forest Road
P.O. Box 10935
Lynchburg, Virginia 24506-0935

Nature of Changes

Revision	Date	Changes
000	August 2010	Initial Release
001	October 2010	Complete Rewrite

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List of Acronyms and Abbreviations

AMP – Aging Management Program

AMR – Aging Management Review

ASME – American Society of Mechanical Engineers

B&PV – Boiler and Pressure Vessel

B&W – Babcock & Wilcox

B&WOG – B&W Owners Group

CASS – Cast Austenitic Stainless Steel

CFR – Code of Federal Regulations

CLB – Current Licensing Basis

CRGT – Control Rod Guide Tube

CR-3 – Crystal River Unit 3

CSA – Core Support Assembly

CSS – Core Support Shield

Duke Energy – Duke Energy Carolinas, LLC

EdF – Électricité de France

EPRI – Electric Power Research Institute

FD – Flow Distributor

FMECA – Failure Modes, Effects, and Criticality Analysis

FSER – Final Safety Evaluation Report

GALL – Generic Aging Lessons Learned

GLRP – Generic License Renewal Program

HTH – High Temperature Heat-Treatment (Alloy X-750)

IBSP– Internals Bolting Surveillance Program

I&E Guidelines – Inspection and Evaluation Guidelines (MRP-227, Rev. 0)

IASCC – Irradiation-Assisted Stress Corrosion Cracking

IE – Irradiation Embrittlement

IGSCC – Intergranular Stress Corrosion Cracking

IMI – Incore Monitoring Instrumentation

INOS – Independent Nuclear Oversight

ISI – In-Service Inspection

ITG – Issue Task Group (EPRI)

JOBB – Joint Owners' Baffle Bolt (Program)

LCB – Lower Core Barrel

LOCA – Loss of Coolant Accident

LRA – License Renewal Application

List of Acronyms and Abbreviations (Continued)

LTS – Lower Thermal Shield

MRP – Materials Reliability Program

NDE – Non-Destructive Examination

NRC – U.S. Nuclear Regulatory Commission

OEP – Operating Experience Program

ONS – Oconee Nuclear Station

ONS-1 – Oconee Nuclear Station Unit 1

ONS-2 – Oconee Nuclear Station Unit 2

ONS-3 – Oconee Nuclear Station Unit 3

PIP – Problem Investigation Process

PWR – Pressurized Water Reactor

PWROG – Pressurized Water Reactor Owners Group

QA – Quality Assurance

RI-FG – Reactor Internals-Focus Group

RFO – Refueling Outage

RV – Reactor Vessel

SCC – Stress Corrosion Cracking

SER – Safety Evaluation Report

S/N – Serial Number

SSC – Structures, Systems, and Component

SSHT – Surveillance Specimen Holder Tube

TLAA – Time-Limited Aging Analysis

TE – Thermal Embrittlement

TJ – Technical Justification

UCB – Upper Core Barrel

UFSAR – Updated Final Safety Analysis Report

U.S. – United States

UT – Ultrasonic Testing (Nondestructive Examination Technique)

UTS – Upper Thermal Shield

VT-3 – Visual Examination

1.0 INTRODUCTION

The purpose of this report is to document the Oconee Nuclear Station (ONS) Units 1, 2, and 3 (ONS-1, ONS-2, and ONS-3) Reactor Vessel (RV) Internals inspection plan for submittal to the United States (U.S.) Nuclear Regulatory Commission (NRC). This report provides a description of the ONS RV Internals inspection plan as it relates to the management of aging effects consistent with previous commitments. The ONS RV Internals inspection plan is based on "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Rev. 0)"[1] and described using the ten Aging Management Program (AMP) elements in the current revision of NUREG-1801 "Generic Aging Lessons Learned" (GALL) report.[2]

This ONS RV Internals inspection plan contains a discussion of the background of the Babcock and Wilcox (B&W)-designed plant RV Internals programs, first sponsored by the utilities through the B&W Owner's Group (B&WOG) and later through the PWR Owner's Group (PWROG), culminating in a submittal to the NRC through the Electric Power Research Institute (EPRI) Pressurized Water Reactor (PWR) Materials Reliability Program (MRP). The ONS RV Internals inspection plan also contains a discussion of operational experience, time-limited aging analyses (TLAAs), and relevant existing ONS programs.

The ONS RV Internals AMP will include this ONS RV Internals inspection plan and will demonstrate that the program adequately manages the effects of aging for RV Internals components and establish the basis for providing reasonable assurance the RV Internals components will remain functional through the ONS license renewal period of extended operation.

2.0 BACKGROUND

2.1 ONS License Renewal Background

By letter dated July 6, 1998, Duke Energy Carolinas, LLC (Duke Energy hereafter) submitted the License Renewal Application (LRA) for ONS in accordance with Title 10, Part 54, of the Code of Federal Regulations (10 CFR 54).[3] Through the LRA, Duke Energy requested the NRC to renew the operating license for ONS-1 (license number DPR-38), ONS-2 (license number DPR-47), and ONS-3 (license number DPR-55) for a period of 20 years beyond the original expiration of midnight February 6, 2013 (ONS-1), midnight October 6, 2013 (ONS-2), and midnight July 19, 2014 (ONS-3). The renewed license was issued by the NRC on May 23, 2000.[4] The safety evaluation report (SER) NUREG-1723[5] documented the technical review of the ONS-1, ONS-2, and ONS-3 LRA by the NRC Staff.

The Renewed Facility Operating License Numbers DPR-38, DPR-47, and DPR-55 for the ONS-1, ONS-2, and ONS-3 plants were granted, as documented in NRC letter of April 10, 2000[6] which identifies the technical basis for issuing the renewed license as being set forth in NUREG-1723.

Section 4.3.11 of the LRA[3] discusses the ONS RV Internals AMP for license renewal. Per the LRA, the proposed ONS RV Internals inspection plan includes the following activities:

a) Continue the characterization of the potential aging effects that have been identified in BAW-2248[7], *Demonstration of the Management of Aging Effects for the Reactor Vessel Internals*. The scope of the characterization includes, but is not limited to, the development of key program elements to address the following aging effects: cracking, reduction of fracture toughness, and loss of closure integrity.

b) After the characterization of the potential aging effects and prior to February 6, 2013, Duke Energy will develop an appropriate monitoring and inspection program, with attributes as defined in Section 4.2 [of the LRA[3]]. This monitoring and inspection program will provide additional assurance that the RV Internals will remain functional through the period of extended operation.

Since the submittal of BAW-2248, the B&WOG (now incorporated into the PWROG) has periodically met with the NRC to discuss RV Internals aging management issues. The Joint Owners' Baffle Bolt (JOBB) program (discussed further in Section 4.1.5 of this report) was completed under the direction of the EPRI PWR MRP. In addition, the EPRI PWR MRP has taken on the industry initiative to provide inspection and evaluation (I&E) guidelines for PWR RV Internals. EPRI PWR MRP meets periodically with the NRC to provide updates. These meetings between the industry and the NRC comply with Duke Energy's commitment in the safety evaluation of BAW-2248A, as repeated in NUREG-1723 (Section 3.4.3, Action Item 4 under Action Items from Previous Staff Evaluation of BAW-2248), to participate in industry RV Internals AMP and provide updates to the NRC on a periodic basis after completion of significant milestones commencing within one year of the issuance of the renewed license.

Section 4.2.5 of NUREG-1723 identifies that the TLAAs from the LRA were reviewed. NUREG-1723 concludes the LRA identified and evaluated the TLAAs associated with RV Internals for ONS-1, ONS-2, and ONS-3 consistent with the requirements of 10 CFR 54.21. See Section 4.1.3 of this report for a further discussion of TLAAs.

Section 6 of NUREG-1723 concludes that, based on the evaluation of the application as discussed in NUREG-1723, the staff determined the requirements of 10 CFR 54.29 were met by the ONS-1, ONS-2, and ONS-3 application. The staff found reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis (CLB) for ONS-1, ONS-2, and ONS-3.

Table 2-1 summarizes the ONS RV Internals LRA commitments and their resolutions.

Table 2-1. ONS RV Internals LRA Commitment Resolutions

Commitment reference section	Commitment/Action Items	Reference section describing the fulfillment of the commitment
NUREG-1723[5], Section 3.4.3 (pages 3-97 through 3-100)	Action Items 1-3, 5-10 from the NRC Staff evaluation of BAW-2248	NUREG-1723, Section 3.4.3 (pages 3-97 through 3-100) contains Duke Energy's responses to Renewal Applicant Items and resolves the action items and states the action items have been resolved
NUREG-1723[5], Section 3.4.3 (page 3-98)	Action Item 4 from Staff Evaluation of BAW-2248 states "The applicant must commit to participation in the B&WOG RVIAMP, and any other industry programs as appropriate, to continue the investigation of potential aging effects for RVI components, and to establish monitoring and inspection programs for RVI components. The applicant shall provide the NRC with either annual reports or periodic updates (after completion of significant milestones) on the status of the RVIAMP, commencing within one year of the issuance of the renewal license."	NUREG-1723, Section 3.4.3 (page 3-98) contains Duke Energy's commitment Section 2.1 of this report describes how the commitment was fulfilled
NUREG-1723[5], Section 3.4.3.3 (page 3-114)	Combination of periodic in-service inspection required by ASME Section XI, Subsection IWB and a flaw evaluation procedure specified in IWB-3640 for "CASS Flaw Evaluation Procedure".	See Section 4.2.1 of this report for a discussion of the ONS In-Service Inspection (ISI) program Per an assessment of thermal aging and neutron embrittlement of cast austenitic stainless steel (CASS), CASS items in the B&W designed PWR internals are redundant and/or potentially able to be analyzed for functionality in the anticipated degraded conditions. Replacement of the degraded item or component is also a potential option. Thus, no fracture toughness properties would be required for fracture mechanics analyses.
NUREG-1723[5], Section 3.4.3.3 (pages 3-120 through 3-122)	Commitment to manage the aging of RV Internals with (1) In-Service Inspection Plan and (2) Oconee RV Internals Inspection (3) The final report will contain the test results from the RVIAMP and the recommended inspection program for the RV Internals.	(1) See Section 4.2.1 of this report for a discussion of the ONS ISI program (2) This commitment is fulfilled by this report (3) See Section 4.1.5 of this report
NUREG-1723[5], Section 4.2.5 (pages 4-23 through 4-25)	1. Flow-induced vibration endurance limit assumptions	The BAW-2248 evaluation of this TLAA was found to be acceptable by NUREG-1723[5], Section 4.2.5 (pages 4-23 through 4-24). No additional action by Duke Energy is required.
	2. Transient cycle count assumptions for the replacement bolting (Action Item 11 from NRC Staff evaluation of BAW-2248)	The BAW-2248 evaluation of this TLAA was found to be acceptable by NUREG-1723[5], Section 4.2.5 [pages 4-23 through 4-25). Duke Energy will continue to monitor and track occurrences of design transients.
	3. Reduction in fracture toughness (Action Item 12 from NRC Staff evaluation of BAW-2248)	An analysis has been performed for the ONS units for this TLAA (see Section 4.1.3 of this report). The analysis will be provided to the NRC to demonstrate the completion of this TLAA for the ONS units.

Commitment reference section	Commitment/Action Items	Reference section describing the fulfillment of the commitment
	4. Flaw growth acceptance	Duke Energy's response for this TLAA was found to be acceptable by NUREG-1723[5], Section 4.2.5 (pages 4-24 through 4-25). No additional action by Duke Energy is required.
ONS LRA[3], Section 4.3.11 (page 4.3-28)	Identification of activities which may be included in the ONS RV Internals AMP: (a) Continued characterization of the potential aging effects identified in BAW-2248 and (b) Develop an appropriate monitoring and inspection program	See discussion under (a) and (b) in Section 2.1 of this report

2.2 ONS RV Internals Aging Management Review/Industry Program Background

The ONS LRA was submitted in 1998 and the SER was granted in 2000; these license renewal documents predate NUREG-1801. However, this ONS RV Internals inspection plan is defined using the ten AMP elements identified in the GALL report published in 2005.[2]

The initial work performed, which supports the ONS RV Internals inspection plan, included an aging management review (AMR) documented in BAW-2248[7] that was directed by the B&WOG Generic License Renewal Program (GLRP). The NRC final safety evaluation report (FSER) of BAW-2248 was attached to the NRC's letter to the B&WOG dated December 9, 1999.[8] The NRC's letter and FSER are included in the updated BAW-2248A report.[9] The NRC identified 12 action items in the FSER to be addressed in the plant-specific LRA when incorporating BAW-2248A in a renewal application. Upon resolution of these action items, Duke Energy may rely on BAW-2248A to demonstrate there is reasonable assurance the ONS RV Internals components will perform their intended functions in accordance with the CLB.

As presented in BAW-2248A, Table 4-1, a combination of existing programs and additional work, to be identified by the "RV Internals Aging Management Program" was credited for aging management of the B&W operating plant RV Internals, including the ONS RV Internals.

An AMR was performed for the ONS units in 2001 in accordance with 10 CFR 54(a)(3).[10] The methodology Duke Energy used to ensure the ONS RV Internals components are bounded by BAW-2248A included three steps: 1) Comparison of RV Internals intended functions, 2) Comparison of RV Internals items subject to AMR, and 3) Review of ONS-specific operating history to ensure the aging effects identified in the generic report are applicable to the ONS RV Internals. This AMR is an ONS-specific application of the B&W generic AMR performed in BAW-2248A.

The additional industry work on the aging of the RV Internals, begun by the submittal of BAW-2248, culminated in the submittal of MRP-227, Rev. 0.[1] MRP-227, Rev. 0 was submitted in January of 2009 to the NRC for review and SER approval.[11] Components requiring inspections are categorized as "Primary", "Expansion", or "Existing Programs". Components not requiring augmented inspections are categorized as "No Additional Measures". The industry program is intended to provide a consistent approach to the aging management of PWR RV Internals components across the PWR fleet. For additional information about MRP-227, Rev. 0 see Section 4.1.1.1 of this report.

Table A.1 in Appendix A shows how the components identified in the BAW-2248A, Table 4-1 AMR were evaluated and characterized by the industry program. Justifications for ONS unit-specific amendments to MRP-227, Rev. 0 are found in Appendix F of this report.

2.3 ONS RV Internals AMP Intent

The ONS RV Internals AMP, which will include the ONS RV Internals inspection plan described in this report after it is approved by the NRC, utilizes a combination of prevention, mitigation, and condition monitoring. Where applicable, credit is taken for existing programs (e.g., primary water chemistry and American Society of Mechanical Engineers [ASME] Boiler & Pressure Vessel [B&PV] Section XI inspections) and mitigation projects such as lower thermal shield (LTS) bolt replacement. The ONS RV Internals inspection plan then incorporates recommendations for augmented inspections provided by industry guidelines in MRP-227, Rev. 0, as modified by amendments to MRP-227, Rev. 0 (see Appendix F of this report). Augmented inspections are in addition to the requirements of ASME B&PV Code Section XI[12]; the I&E guidelines do not reduce, alter, or otherwise affect current ASME B&PV Code Section XI inspections.

Aging degradation mechanisms that impact the RV Internals have been identified in MRP-227, Rev. 0. The overall outcome of the additional work performed by the Industry summarized in MRP-227, Rev. 0 is to ensure functionality of the RV Internals is maintained by detection of the effect of the degradation mechanism listed in Table 2-2. Therefore, this ONS RV Internals inspection plan is consistent with the industry work provided in MRP-227, Rev. 0 as modified by amendments to MRP-227, Rev. 0 (see Appendix F of this report).

Table 2-2. RV Internals Aging Degradation Mechanisms and Their Aging Effects

Aging Degradation Mechanism	Aging Effect
Stress Corrosion Cracking (SCC)	Cracking
Irradiation-Assisted Stress Corrosion Cracking (IASCC)	Cracking
Wear	Loss of Material
Fatigue	Cracking
Thermal Aging Embrittlement	Loss of Ductility and Unstable Crack Extension
Irradiation Embrittlement (IE)	Loss of Ductility and Unstable Crack Extension
Void Swelling and Irradiation Growth	Dimension Change, Distortion, and Cracking
Thermal and Irradiation-Enhanced Stress Relaxation or Irradiation-Enhanced Creep	Loss of Mechanical Closure Integrity leading to Cracking

Section 5.0 of this report uses the ten AMP elements of NUREG-1801, Rev. 1 to describe the ONS RV Internals inspection plan and AMP as required by the NRC. The ONS RV Internals AMP, which will also include this ONS RV Internals inspection plan after it is approved by the NRC, incorporates programs and activities that are credited for managing the aging effects produced by the aging degradation mechanisms listed in Table 2-2. ONS RV Internals components within the scope of BAW-2248A, the LRA, and NUREG-1723 have been considered in this ONS RV Internals inspection plan.

Table A.1 in Appendix A of this report shows how the components identified in the BAW-2248A, Table 4-1 AMR were evaluated and characterized by the industry program.

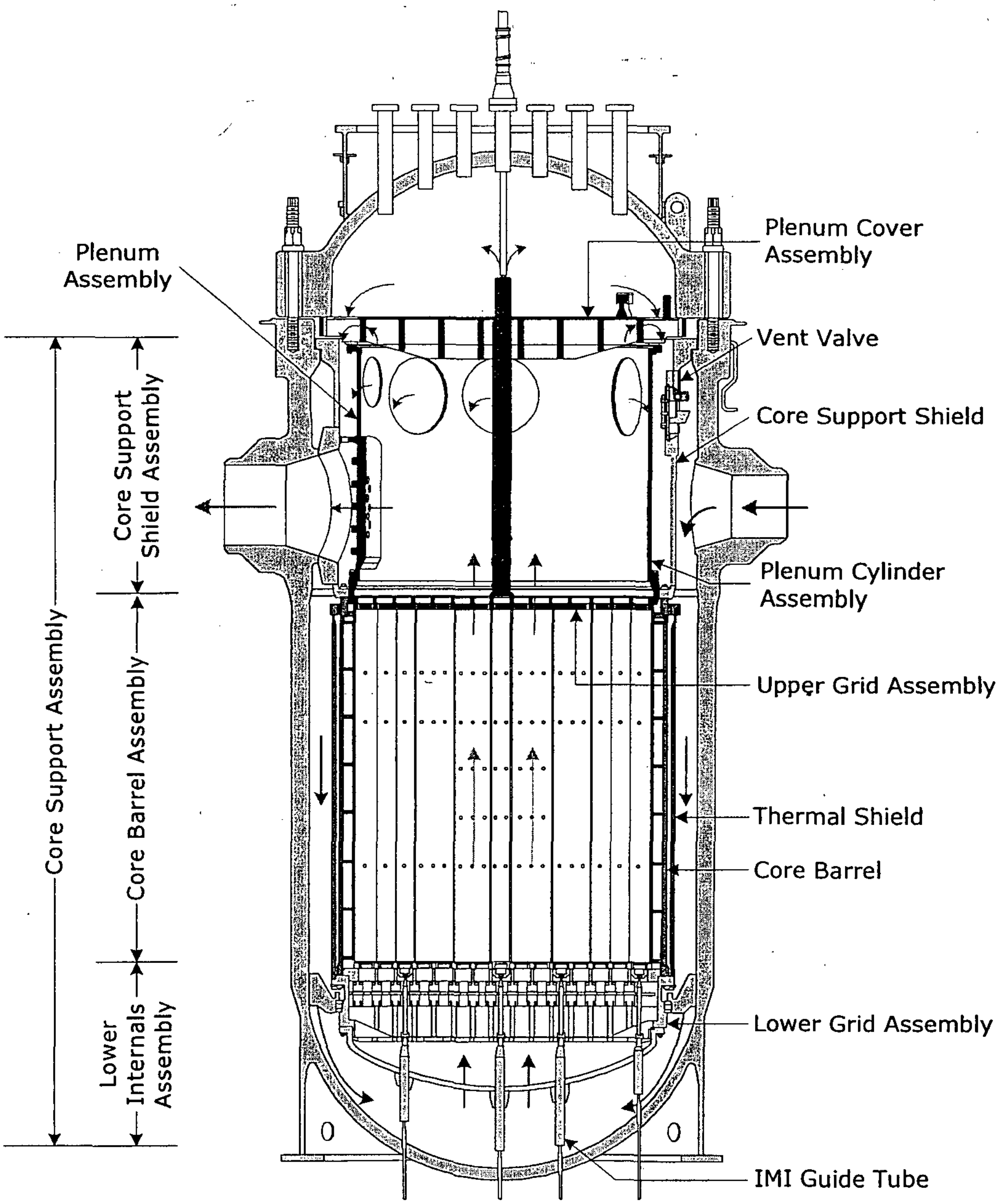
2.4 ONS RV Internals Background

The intended functions of the ONS RV Internals are as follows[5,9]:

- Support and orient the reactor core
- Support, orient, guide, and protect the control rod assemblies
- Provide a passageway to distribute the reactor coolant flow to the reactor core
- Provide a passageway to support, guide, and protect incore instrumentation
- Provide a secondary core support to limit downward displacement of core support structure
- Provide gamma and neutron shielding

The ONS RV Internals consists of two structural subassemblies that are located within the RV: the plenum assembly and the core support assembly. Duke Energy has reviewed the design and operation of the ONS RV Internals using the process described in Section 2.4 of the ONS LRA[3] and determined they are bounded by the description contained in BAW-2248A, with the exception of the thermal shield and thermal shield upper restraint. Note that the thermal shield and thermal shield upper restraint were omitted from BAW-2248A; however these items support an ONS RV Internals intended function and were found to be subject to an AMR. The thermal shield surrounds the core barrel and is constructed of austenitic stainless steel. The thermal shield upper restraint is also constructed of austenitic stainless steel. These items were included in the ONS AMR[10] and the industry work which culminated in MRP-227, Rev. 0. The general arrangement of the ONS RV Internals is shown in Figure 2-1.[1]

Figure 2-1. ONS RV Internals



2.5 RV Internals Inspection Commitment Change Letter

By letter dated June 16, 2010 to the NRC[13], Duke Energy stated its intent to revise the existing license renewal commitment to inspect the RV Internals at each Duke Energy nuclear station, including the three ONS units. The existing inspection commitments are contained in Section 18.3.20 of the ONS Updated Final Safety Analysis Report (UFSAR). The UFSAR section contains an allowance that permits Duke Energy to modify or eliminate these inspections based on industry data or other evaluations if plant-specific justification is provided to demonstrate the basis for the modification or elimination.

Duke Energy is revising its commitments for RV Internals inspections from those that currently exist in the ONS UFSAR to the inspection guidelines provided in MRP-227, Rev. 0, as approved by the NRC. This ONS RV Internals inspection plan, as documented herein, will follow MRP-227, Rev. 0. After the anticipated release of MRP-227-A, Duke Energy will review and, if needed, revise this ONS RV Internals inspection plan.

The ONS RV Internals inspection plan will also be revised as necessary following review by the NRC. Once the ONS RV Internals inspection plan is approved, the ONS UFSAR will be updated as required.

Note: This ONS RV Internals inspection plan contains several amendments to MRP-227, Rev. 0 which are discussed in Section 4.2.10 and Appendix F of this report.

3.0 PROGRAM OWNER

The Oconee Reactor and Electrical Systems (Reactor Team) and the General Office Nuclear Technical Services are responsible for maintaining and implementing the ONS RV Internals inspection plan.

4.0 INDUSTRY AND ONS PROGRAMS AND ACTIVITIES

The ONS RV Internals inspection plan is based on the ONS LRA, NUREG-1723, and evaluations supporting MRP-227, Rev. 0. The ONS RV Internals AMP is demonstrated by implementation of MRP-227, Rev. 0 methodology and the continuation of the existing programs discussed in this section.

4.1 Industry Programs and Activities

There are various industry programs and activities in which Duke Energy has been or is participating that support the aging management of the PWR RV Internals; those discussed in this section include EPRI PWR MRP activities, PWROG activities, TLAAs, the internals bolting surveillance program (IBSP), the JOBB program, and the fuel/baffle interaction investigation for the B&W-designed units. Duke Energy will continue to participate in industry activities addressing PWR RV Internals.

4.1.1 EPRI PWR MRP Activities

As part of the ONS License Renewal Program, Duke Energy made the commitment to participate in industry activities associated with the development of the standard industry guidance, which includes the EPRI PWR MRP activities which produced the guidelines and standards discussed below.

4.1.1.1 MRP-227, Rev. 0

The EPRI PWR MRP efforts have defined the required inspections and examination techniques for the RV Internals. The results of the industry recommended inspections, published in MRP-227, Rev. 0, serve as the basis for identifying any augmented inspections that are required to complete this ONS RV Internals inspection plan.

4.1.1.1.1 Development of MRP-227, Rev. 0

The MRP-227, Rev. 0 "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines"[1] were developed by a team of industry representatives who reviewed available data and industry experience to identify and prioritize I&E requirements for RV Internals. MRP-227, Rev. 0 is the culmination of the industry work that began with BAW-2248A for B&W plants. The key sequential steps in the process included the following:

- The development of screening criteria, with susceptibility levels for the eight postulated aging degradation mechanisms relevant to reactor internals and their effects;
- An initial component screening and categorization, using the susceptibility levels and FMECA (failure modes, effects, and criticality assessment) to identify the relative ranking of the components;
- Functionality assessment of degradation for components and assemblies of components; and
- Aging management strategy development combining results of the functionality assessment with component accessibility, operating experience (OE), existing evaluations, and prior examination results to determine the appropriate aging management methodology, baseline examination timing, and the need for and the timing of subsequent inspections.

Through this process, the RV Internals for all three PWR designs in the U.S. were evaluated, and appropriate recommendations for aging management actions specific to each component were provided.

MRP-227, Rev. 0 utilized the screening and ranking process to aid in the identification of required inspections for "Primary" and "Expansion" components and credits existing component inspections when they were deemed adequate.

The basic description of each classification is as follows:

- **"Primary"**

  Those PWR internals that are highly susceptible to the effects of at least one of the eight aging mechanisms were placed in the "Primary" group. The aging management requirements that are needed to ensure functionality of "Primary" components are described in these I&E guidelines. The "Primary" group also includes components which have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.

- **"Expansion"**

  Those PWR internals that are highly or moderately susceptible to the effects of at least one of the eight aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects, were placed in the "Expansion" group. The schedule for implementation of aging management requirements for "Expansion" components will depend on the findings from the examinations of the "Primary" components at individual plants.

- **"Existing Programs"**

  Those PWR internals that are susceptible to the effects of at least one of the eight aging mechanisms and for which generic and plant-specific existing AMP elements are capable of managing those effects, were placed in the "Existing Programs" group.

  Note there are no "Existing Programs" components in MRP-227, Rev. 0 for the B&W-designed PWRs.

- **"No Additional Measures"**

  Those PWR internals for which the effects of all eight aging mechanisms are below the screening criteria were placed in the "No Additional Measures" group. Additional components were placed in the "No Additional Measures" group as a result of FMECA and the functionality assessment. No further action is required by these guidelines for managing the aging of the "No Additional Measures" components.

The categorization and analysis processes used in the MRP-227, Rev. 0 approach are not intended to supersede any ASME B&PV Code Section XI requirements. Any components that are classified as removable core support structures, as defined in ASME B&PV Code Section XI IWB-2500, Examination Category B-N-3, have requirements that remain in effect and may only be altered as allowed by 10 CFR 50.55a.

The requirements of MRP-227, Rev. 0 are classified in accordance with the requirements of NEI 03-08 Guidelines.[14] For the MRP-227, Rev. 0 guidelines there are one "Mandatory", three "Needed", and one "Good Practice" elements as follows:

- **"Mandatory"**
  Each commercial U. S. PWR unit shall develop and document a PWR reactor internals aging management program (AMP) within thirty-six months following issuance of MRP-227-Rev. 0.

  MRP-227, Rev. 0 was issued in December 2008 and submitted to the NRC in January 2009. Duke Energy will fulfill this "Mandatory" element by developing the ONS RV Internals AMP by the end of December 2011.

- **"Needed"**
  Each commercial U. S. PWR unit shall implement Tables 4-1 through 4-9 and Tables 5-1 through 5-3 for the applicable design within twenty-four months following issuance of MRP-227-A.

  The applicable B&W tables contained in MRP-227, Rev. 0, Table 4-1 ("Primary"), Table 4-4 ("Expansion"), and Table 5-1 (Examination Acceptance and Expansion Criteria) are attached herein as Appendices B, C, and D. There are no "Existing Program" components in MRP-227, Rev. 0 for the B&W-designed PWRs. The ONS units have followed the MRP-227, Rev. 0 recommended inspections by the inspection activities already performed or planned as described in this ONS RV Internals inspection plan. The justification for ONS unit-specific amendments to MRP-227 Rev. 0 are discussed in Section 4.2.10 and Appendix F of this report. MRP-227, Rev. 0 has been submitted to the NRC with the ultimate goal of obtaining approval and issuance of an SER. After the release of MRP-227-A, Duke Energy will review and, if needed, revise this ONS RV Internals inspection plan. Therefore, implementation of this ONS RV Internals inspection plan will fulfill this "Needed" requirement for the three ONS units.

- **"Needed"**
  Examinations specified in these guidelines shall be conducted in accordance with the Inspection Standard MRP-228.

Inspection standards developed under MRP-228[15] will be used by Duke Energy for the augmented inspections described in this ONS RV Internals inspection plan developed in accordance with MRP-227, Rev. 0 for the three ONS units. Implementation of this ONS RV Internals inspection plan will fulfill this "Needed" requirement for the three ONS units.

- "Needed"
  Examination results that do not meet the examination acceptance criteria defined in Section 5 of the MRP-227 guidelines shall be recorded and entered in the plant corrective action program and dispositioned.

  The ONS Corrective Action Program (PIP) will be applied as discussed in Section 5.7 of this report. Implementation of this ONS RV Internals inspection plan will fulfill this "Needed" requirement for the three ONS units.

- "Good Practice"
  Each commercial U. S. PWR unit should provide a summary report of all inspections and monitoring, items requiring evaluation, and new repairs to the MRP Program Manager within 120 days of the completion of an outage during which PWR internals are examined. The MRP template should be used for this report.

  Duke Energy will make the best effort in providing summary reports to MRP of future ONS inspection activities within 120 days of the completion of an outage during which PWR RV Internals are examined. Implementation of this ONS RV Internals inspection plan will fulfill this "Good Practice" requirement for the three ONS units.

4.1.1.1.2 MRP-227, Rev. 0 Applicability to ONS

The MRP-227, Rev. 0 guidelines are based on several general assumptions that were used for the analysis in the development of MRP-227, Rev. 0. These assumptions and their applicability to ONS units are listed below:

- **30 years or less of operation with high leakage core loading patterns (fresh fuel assemblies loaded in peripheral locations) followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation.**

  The fuel management program for the three ONS units changed from a high to a low-leakage core loading pattern prior to 30 years of operation.

- **Base load operation, i.e., typically operates at fixed power levels and does not usually vary power on a calendar or load demand schedule.**

  The three ONS units operate as a base load unit.

- **No design changes beyond those identified in general industry guidance or recommended by the original vendors.**

  MRP-227, Rev. 0 states that the recommendations are applicable to all operating U.S. PWR operating plants as of May 2007 for the three designs (i.e., B&W, and Westinghouse, CE) identified. No modifications have been made to the ONS RV Internals since May 2007. Fabrication records searches have been conducted for the ONS units as described in Section 4.2.6 of this report to verify the applicability of MRP-227, Rev. 0 recommendations to the ONS units. The justifications for ONS unit-specific amendments to MRP-227, Rev. 0 from the fabrication records search results are described in Section 4.2.10 of this report.

Based on the above review, MRP-227, Rev. 0 is applicable to all three ONS units, except for the amendments identified in Section 4.2.10 and Appendix F of this report.

4.1.1.2 MRP-228, Final Report

"Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228)"[15] was developed by the EPRI PWR MRP Inspection Issue Task Group (ITG) in cooperation with the reactor internals focus group (RI-FG). These inspection standards are intended to support MRP-227, Rev. 0 to detect the effects of aging degradation mechanisms. This report provides the PWR fleet with inspection procedure requirements for the RV Internals "Primary" and "Expansion" components included in MRP-227, Rev. 0 and offers a stable mechanism for documenting the capability of the evolving inspection technology.

MRP-228, Rev. 0 contains four "Needed" and two "Good Practice" requirements, which will be followed in accordance with the requirements of the NEI 03-08 Guidelines.[14] Duke Energy will implement the MRP-228 requirements for the augmented inspections described in this ONS RV Internals inspection plan developed in accordance with MRP-227, Rev. 0 for the three ONS units.

4.1.2 PWROG Activities

As part of the ONS License Renewal Program, Duke Energy made the commitment to participate in industry activities associated with the development of the standard industry guidance, which includes the activities performed by the PWROG. The PWROG activities provide continuous industry support and a strategic plan for the aging management of the PWR RV Internals through participation in technical meetings and industry forums. Sections 4.2.7, 5.3, 5.4.5, and 5.6.2 of this report discuss past and ongoing PWROG activities.

4.1.3 Time-Limited Aging Analyses

In the ONS LRA[3], the identified three RV Internals applicable TLAAs, as listed below, were evaluated for the period of extended operation consistent with the requirements of 10 CFR 54.21.

1. Flow-induced vibration endurance limit assumptions

   The flow-induced vibration fatigue limit assumptions were increased from 10^{12} cycles for 40 years to 10^{13} cycles for 60 years. The stress values calculated were found to be less than the endurance limit, rendering the evaluation acceptable according to the requirements of 10 CFR 54.21. Therefore, this TLAA has been resolved.

2. Transient cycle count assumptions for the replacement bolting

   The ability to withstand cyclic loading without fatigue failure was evaluated using a cumulative usage factor methodology. In BAW-2248, for each utility, the number of transients accrued to date was conservatively extrapolated, and in all cases it was found that the number of design cycles would not be exceeded in the period of extended operation. The B&WOG reported that each of the participating utilities monitors occurrences of design transients and is thus managing the potential for cracking resulting from fatigue. Therefore, Duke Energy will continue to monitor and track occurrences of design transients for all three ONS units during the extended license period.

3. Reduction in fracture toughness

   The TLAA described as "reduction in fracture toughness" is related to the acceptability of the RV Internals under loss of coolant accident (LOCA) and seismic loading. BAW-2248 states that BAW-10008, Part 1, Revision 1[16] concludes "that at the end of 40 years, the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and that irradiation will not adversely affect deformation limits." BAW-2248 also states that this TLAA will be resolved on a plant-specific basis per 10 CFR 54.21 (c)(1)(iii) based on the results and conclusion of the planned B&WOG RV Internals AMP. Duke Energy has stated that appropriate action will be taken in a timely manner to ensure continued validity of the design of the ONS RV Internals. Plant-specific analysis is required to demonstrate that, under LOCA and seismic loading and with irradiation accumulated at the expiration of the period of extended operation, the RV Internals have adequate ductility to absorb local strain at the regions of maximum stress intensity and will meet the deformation limits. The applicant must provide a plan to develop data to demonstrate that the RV Internals will meet the deformation limits through the period of extended operation. Duke Energy committed to perform the plant-specific analysis.

   A bounding analysis applicable to the B&W-designed units including ONS-1, ONS-2, and ONS-3 was performed for the period of extended operation. The analysis concluded that at the end of a 60-year lifetime, the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and the irradiation will not adversely affect deformation limits. The analysis will be provided to the NRC to demonstrate the completion of this TLAA for the ONS units.

There is a fourth TLAA discussed in NUREG-1723 regarding flaw growth acceptance in accordance with the ASME B&PV Code Section XI In-Service Inspection (ISI) requirements. This TLAA is identified in BAW-2248 as requiring plant-specific evaluation. An open item (Open Item 4.2.5.3-2) was identified in the June 16, 1999 SER[17] and subsequently in a letter dated October 15, 1999[18] Duke Energy responded that no flaws have been identified in the ONS RV Internals and hence no evaluation is required. The October 15, 1999 letter closes Open Item 4.2.5.3-2.

4.1.4 Internals Bolting Surveillance Program

Starting in 1981, ultrasonic testing (UT) at several B&W units revealed the LTS bolt, upper core barrel (UCB) bolt, lower core barrel (LCB) bolt, flow distributor (FD) bolt, and surveillance specimen holder tube (SSHT) bolt locations had rejectable UT indications. Some of the bolts with rejectable UT indications were later determined to be cracked due to intergranular stress corrosion cracking (IGSCC) by laboratory examination. The failed bolts were fabricated from Alloy A-286, Condition A (ASTM A 453, Grade 660) material, except for the SSHT bolts, which were fabricated from Alloy A-286, Condition B (ASTM A 453, Grade 660) material.[19]

As a result of the noted bolt failures, utilities began replacing bolts where needed. The B&WOG initiated the IBSP (which was completed by EPRI PWR MRP) to better assess the IGSCC susceptibility of the replacement bolts. The IBSP exposed scaled down replacement bolts to simulated PWR conditions in an autoclave and an actual PWR environment inside the RV specimen tube holder at an operating B&W unit. The scaled down bolts used in the testing were manufactured from Alloy A-286, Condition A and Alloy X-750, high temperature heat-treatment (HTH) condition materials. The scaled down bolts were tested in two surface conditions, peened and un-peened.

After the completion of the IBSP tests, several peened replacement Alloy A-286 scaled down replacement bolts developed IGSCC when loaded to a high stress, while the un-peened Alloy A-286 scaled down replacement bolts were free from IGSCC when loaded to the same high stresses for the test duration of 8 ½ years. The Alloy X-750, HTH Condition scaled down replacements bolts of both peened and un-peened conditions were free from IGSCC when subjected to the same environmental and loading conditions as the Alloy A-286 bolts for the test duration of 8 ½ years.

Only the LTS bolts at the three ONS units have been replaced with Alloy X-750 HTH studs and nuts. The other locations such as UCB, LCB, upper thermal shield (UTS), and FD bolts are the original Alloy A-286 bolts at the ONS units. Most of the SSHT assemblies at the three ONS units, including the SSHT bolts, were removed from the RV Internals, and therefore no longer have an IGSCC concern.

A 2005 evaluation of the IBSP and industry experience resulted in PWROG Letter OG-06-1880, which makes recommendations for UT examinations of the high strength (Alloy X-750 or Alloy A-286) bolts in the B&W units. These recommendations were made in accordance with NEI 03-08. These UT inspection recommendations have been incorporated into MRP-227, Rev. 0 with the "Needed" recommendation being incorporated into the "Primary" category and the "Good Practice" recommendation being incorporated into the "Expansion" category. The MRP-227, Rev. 0 inspections supersede these UT inspection recommendations.

4.1.5 Joint Owners' Baffle Bolt Program

The JOBB Program stemmed from ultrasonic inspections of baffle-to-former bolts at several Électricité de France (EdF) plants. Indications were noted under the bolt head in the head-to-shank fillet radius. The bolt failures were attributed to irradiation-assisted stress corrosion cracking (IASCC). Various tasks, including non-destructive examination (NDE) inspections, temperature, fluence, loading, and chemical composition comparisons of bolts, irradiation and mechanical testing, corrosion testing, helium effect evaluation, and microstructural evaluation were used to characterize the effect of irradiation on bolting materials under the JOBB.

The JOBB is now being managed by EPRI with additional research on RV Internals material being performed under EPRI programs. The results of the JOBB program have been incorporated into EPRI PWR MRP documents through the screening criteria for IASCC, and specifically referenced in MRP-227, Rev. 0. In BAW-2248A, repeated in NUREG-1723, there is an action item to provide a final report that contains the test results from the RVIAMP and the recommended inspection program for the RV Internals. EPRI PWR MRP provides results to the NRC during meetings (see Reference 20 for an example), which fulfills this commitment.

4.1.6 Fuel/Baffle Interaction Investigation

An investigation was conducted between 2004 and 2010 on the interaction between the baffle plates and fuel assembly grid straps in the B&W units by AREVA NP and the utilities with operating B&W-designed units. Wear of fuel assembly spacer grid outer straps against RV Internals baffle plates has been observed since initial plant operation and has increased significantly with the switch from Alloy-718 to Zircaloy-4 grids in the 1980s. The "Primary" requirement from MRP-227, Rev. 0 for a one-time physical measurement of the interference fit between the plenum cover weldment rib pads and the RV flange was performed at ONS between 2006 and 2008 in order to provide data to the investigation. Recommendations from the fuel/baffle interaction investigation were entered into the Duke Energy Problem Investigation Process (PIP).

4.2 ONS Programs and Activities

ONS has a number of programs and activities that support the aging management of the RV Internals; these include the ASME B&PV Code Section XI In-Service Inspection program, primary water chemistry program, the vent valve in-service test program, implementation of low-leakage cores, LTS bolt replacement, a fabrication records search, UT examination of UCB bolts, core clamping measurements, visual examination of baffle-to-baffle and baffle-to-former bolts at each RFO, and ONS unit-specific amendments to MRP-227, Rev. 0.

4.2.1 ASME B&PV Code Section XI In-Service Inspection Requirements

The ONS ASME B&PV Code Section XI ISI requirements for examination of the RV interior, attachments, and internals are contained in ASME B&PV Code Section XI, Subsection IWB-2500-1.[12] Areas accessible during a refueling outage (RFO) of the RV interior (Examination Category B-N-1) are examined using visual examinations (VT-3) examination methods each period (approximately every 3 years). RV interior attachments (Examination Category B-N-2) within the beltline region are examined using visual VT-1 examination methods and interior attachments beyond the beltline region are examined using visual VT-3 examination methods each interval (approximately every 10 years). Removable core support structures are examined using visual VT-3 examination methods each interval (approximately every 10 years). Category B-N-1, B-N-2, and B-N-3 examinations will be performed during the next ONS ASME Section XI 10-year ISI examinations currently scheduled for the Fall 2012 RFO for ONS-1, the Fall 2013 RFO for ONS-2, and the Spring 2014 RFO for ONS-3.

The RV core guide lugs will receive a VT-1 examination in accordance with Examination Category B-N-2 during the fourth and future 10-year ISI intervals at ONS. The remnants of the flow stabilizers and the incore monitoring instrumentation (IMI) nozzles will receive a VT-3 examination in accordance with Examination Category B-N-2 during the fourth and future 10-year ISI intervals at ONS. Removable core support structures (Examination Category B-N-3), which will receive a VT-3 examination during the fourth and future 10-year ISI intervals at ONS, are listed in Table 4-1 below. Relevant conditions for these examination categories are found in ASME B&PV Code Section XI, IWB-3520.

Table 4-1. ONS Removable Core Support Structure Components (Examination Category B-N-3)

Component
Thermal Shield
Thermal Shield Upper Restraint Assemblies
Upper Thermal Shield Bolting
Remnants of Surveillance Holder Tube Structures

Component
Core Support Shield (CSS) Assembly
CSS Top Flange, including Seating Surfaces
CSS Outlet Nozzles
CSS Outlet Nozzle Sealing Surfaces
CSS Outlet Flow Deflectors
Internals Vent Valves, Retaining Rings, Guide Blocks, Jack Screws and Locking Devices
Core Support Assembly (CSA) Lifting Lugs
CSA Keyways
CSA Loss of Coolant Accident (LOCA) Bosses
Upper Core Barrel Bolting
CSA Baffle Plates
CSA Former Plates
Baffle Plate Bolting
CSA Lower Grid
CSA Lower Grid Pads
Instrument Guide Tube Spiders
Flow Distributor Bolting
Interface between Upper Former Plate and Core Barrel and Adjacent Surfaces
IMI Tubes and Guide Tubes
Flow Distributor Head
Guide Block Assemblies – Pairs
Guide Block Bolting
Shock Pad Assemblies
Shock Pad Bolting
Lower Core Barrel Bolting
Lower Thermal Shield Bolting
Lifting Lugs and Base Blocks
Plenum Cover and Ribs
Plenum Cover to Cylinder Bolted Connection
Plenum Clamping Surfaces
Plenum Cylinder to Upper Grid Bolted Connection
Plenum Assembly Keyways
Plenum Assembly Outside Surfaces
Thermocouple Guide Tube Assemblies and Attachments
Control Rod Guide Tube Assemblies (from top of plenum assembly)
Plenum LOCA Bosses and Welds
Upper Grid Assembly (including bolting and grid pads)
Control Rod Guide Tube Assemblies (from bottom of plenum assembly)

Note: The ASME B&PV Code Section XI, Category B-N-3 ISI scope is defined by the owners (utilities) of the B&W units.

4.2.2 Primary Water Chemistry Program

The ONS Primary Water Chemistry Program limits the concentration of oxygen, halogens, and sulfate species in the primary water to prevent the coolant from becoming an environment favorable to stress corrosion cracking (SCC), and therefore greatly reduces the probability of SCC and IASCC. The limits imposed by the primary water chemistry program meet the intent of the EPRI Pressurized Water Reactor Primary Water Chemistry Guidelines.[21]

4.2.3 Vent Valve In-Service Test Program

There is an existing ONS program that requires vent valve testing and visual inspection each RFO. The accessible surfaces of the vent valve are visually inspected, including the locking devices. Any observed surface irregularities on the valve body and disc seating surface are identified and evaluated. Additionally, vent valve operation is tested through manual actuation to verify that the lifting force required to fully open the vent valves does not exceed the specific limit.

4.2.4 Continuation of Use of Low-Leakage Cores

As discussed in Section 3.4.3.3 of NUREG-1723, Duke Energy will continue to use low-leakage core loading patterns, which is considered a preventative action to lessen the effects of aging on the ONS RV Internals.

4.2.5 Lower Thermal Shield Replacement Bolt

Failure of the original Alloy A-286 LTS bolts in the 1980s led to the replacement of all the original LTS bolts with replacement studs/nuts at the three ONS units. The replacement LTS nuts/studs on each unit are secured by tie plate and crimp locking cups in groups of two (two-hole design). The replacement studs are made from Alloy X-750 in the HTH condition. The compression nuts are also made of Alloy X-750 in the HTH condition. The tie plate and crimp cup are both made of Type 304 stainless steel. Note that the LTS bolts are categorized as an "Expansion" item in MRP-227, Rev. 0.[1]

4.2.6 Fabrication Records Search

Two fabrication records searches were conducted by AREVA NP for the ONS RV Internals components listed as "Primary" and "Expansion" in MRP-227, Rev. 0. The goal of the first records search was to locate the chemical composition of the CASS items and, if possible, to screen them for susceptibility to thermal aging embrittlement, consistent with the screening criteria used by the EPRI PWR MRP. The goal of the second ONS-specific record search was to obtain a detailed description of the component, obtain fabrication records (heat numbers, CMTRs, etc.), obtain a description of the anticipated degradation mechanisms, and review of operational experience.

4.2.6.1 Cast Austenitic Stainless Steel Records Search

In 2009 and 2010, a search of original fabrication records was made for several B&W CASS RV Internals items identified as susceptible to thermal aging embrittlement in MRP-227, Rev. 0. The thermal aging embrittlement susceptibility can be screened per the criteria in MRP-175[22], using Hull's equivalent factors in NUREG/CR-4513, Rev. 1[23], if chemical composition is known. The MRP-175 screening criteria for thermal aging embrittlement of CASS are identical to those in Section XI.M12 of NUREG-1801. During the screening phase for MRP-227, Rev. 0, if the estimated ferrite content did not exceed the MRP-175 screening criteria for CASS thermal aging embrittlement, the CASS item was screened out. CASS items whose chemical composition was unknown were identified as susceptible to thermal aging embrittlement in MRP-227, Rev. 0.

The records search included the CSS vent valve discs, CSS outlet nozzles (ONS-3 only), and the CRGT assembly spacer castings. The following conclusions were made from this records search and corresponding calculation of ferrite content:

1. CSS vent valve discs, CF-8 (ONS-1, ONS-2, and ONS-3)

   The CSS vent valve discs are categorized as "Primary" in MRP-227, Rev. 0. Each ONS unit has eight 14-inch CSS vent valves. The CSS vent valve discs were fabricated from Grade CF-8 castings. The chemical composition has been found for all 24 original CSS vent valves discs supplied to the three ONS units. The ferrite content for all of the original CSS vent valve discs is under the 20% screening criteria for thermal aging embrittlement of CF-8 castings.

However, ONS was also supplied with spare CSS vent valves, in addition to the originally installed CSS vent valves. The spare CSS vent valve records were not found at AREVA NP during this records search. It is known that some spare CSS vent valves were installed (i.e., replaced the original CSS vent valves) in the late 1970's or early 1980's to fix the CSS vent valve jackscrew locking devices at ONS. To ascertain that each installed CSS vent valve disc is under the screening threshold, it was recommended that the CSS vent valve disc serial number (S/N) and CSS vent valve disc heat number for the currently installed CSS vent valves be identified. The S/N and heat numbers are stamped on the CSS vent valve disc surface.

The S/N of the eight CSS vent valves discs installed in ONS-1 were recorded during the Fall 2009 RFO. The S/N of the eight CSS vent valve discs installed in ONS-2 were recorded during the Spring 2010 RFO. Similar CSS vent valve disc identification is planned for ONS-3 in the Fall 2010 RFO. Based on the S/N and heat number identification, the CSS vent valve disc ferrite content of all currently installed CSS vent valves at ONS-1 and ONS-2 are confirmed to be below the screening threshold. Based on the records search and ONS-1 and ONS-2 identification results, a similar finding is expected for ONS-3.

Therefore, a justification for an ONS unit-specific amendments to MRP-227, Rev. 0 requirements has been written (see Appendix F of this report) to recategorize all currently installed CSS vent valve discs at ONS-1 and ONS-2 from "Primary" to "No Additional Measures". No augmented inspection is required. This recategorization will apply to the ONS-3 installed vent valve discs after the ferrite is confirmed to be below the screening criteria. The existing inspection requirements for these items such as CSS Vent Valve In-Service Test Program, as discussed in Section 4.2.3 of this report, will continue to be performed.

2. CSS outlet nozzles, CF-8 (ONS-3 only)

   The two CSS outlet nozzles in the ONS-3 RV Internals are categorized as "Primary" in MRP-227, Rev. 0 and are fabricated from Grade CF-8 castings. The records search confirmed the ferrite content calculated from the chemical composition of the CSS outlet nozzles at ONS-3 is below the 20% screening criteria for thermal aging embrittlement of a CF-8 casting. Therefore, a justification for an ONS unit-specific amendment to MRP-227, Rev. 0 requirements has been written (see Appendix F of this report) to recategorize the CSS outlet nozzles at ONS-3 from "Primary" to "No Additional Measures". No augmented inspection is required. The existing Section XI ISI for the CSS outlet nozzles will continue to be performed at ONS-3. The CSS outlet nozzles at ONS-1 and ONS-2 are not fabricated from CASS and do not have a thermal aging embrittlement concern.

3. CRGT spacer castings, CF-3M (ONS-1, ONS-2, and ONS-3)

   The CRGT spacer castings are categorized as "Expansion" in MRP-227, Rev. 0. Each ONS unit contains 690 CRGT spacer castings in its RV Internals fabricated from Grade CF-3M castings. Based on the chemical compositions found, the ferrite content for most of the CRGT spacer castings at the ONS units exceeds the 14% ferrite screening criteria for thermal aging embrittlement for CF-3M casting. Therefore, it is concluded that the CRGT spacer castings should remain applicable to the ONS units and cannot be recategorized to "No Additional Measures". Due to the recategorization of its primary linked items (CSS vent valve discs and CSS outlet nozzles [ONS-3 only]) to "No Additional Measures", the CRGT spacer castings are recategorized as "Primary" for the ONS units as described in Appendix F of this report.

4.2.6.2 ONS-1 Plenum Cover Weldment Rib Pad Items

The records search for "Primary" and "Expansion" items in MRP-227, Rev. 0 also performed in 2009 and 2010 for the RV Internals for the three ONS units identified a feature unique to the ONS-1 plenum cover weldment rib pads. Each of the 32 plenum cover weldment rib pads at ONS-1 are fastened to the plenum cover ribs with two Type 304 stainless steel screws and one Alloy X-750 dowel. The Alloy X-750 dowels, Type 304 screws, and their locking welds were unknown and were not screened for aging degradation mechanisms, nor evaluated for inclusion in MRP-227, Rev. 0.

Using the MRP screening criteria and process, these items are categorized as Category "A" or "No Additional Measures". Therefore, no additional augmented inspection is required for this location. The evaluation for the ONS-1 plenum cover weldment rib pad items is documented in Appendix F of this report.

4.2.7 Volumetric (UT) Examinations of Upper Core Barrel Bolts

The most recent UCB bolt UT inspections were performed in April 2008 (ONS-1, 100%), October 2008 (ONS-2, 100%), and November 2007 (ONS-3, 100%) in accordance with a "Needed" NEI 03-08 recommendation in PWROG letter OG-07-43 (1/26/07) and also met the initial inspection requirements in MRP-227, Rev. 0. The results of the inspections at ONS-1 and ONS-2 were zero UCB bolts rejected as in previous inspections. Two UCB bolts were rejected due to a lack of back wall reflection at ONS-3; these results are identical to the previous ONS-3 UT inspection results in 1984, 1985, and 1987. Note that at ONS-1, four of the 120 UCB bolts were removed for verification and better interpretation of bolt UT signals in the 1980s. Visual, ultrasonic, and fluorescent liquid penetrant examinations were performed in the laboratory on all four bolts. The examinations found no indications confirming the on-site UT results. These four UCB bolt locations at ONS-1 are still empty.

The most recent LCB UT inspections were performed in June 1983 (ONS-1, partial), October 1983 (ONS-2, partial), and January 1987 (ONS-3, 100%) in response to the original B&W internal A-286 bolt failures. UT inspections for the LCB bolts are planned during the RV Internals inspections in 2012, 2013, and 2014, in compliance with the MRP-227, Rev. 0 recommended inspection requirement for the LCB bolts.

The AREVA NP UT examination procedure for the ONS UCB and LCB bolt examinations in 2007 and 2008 was validated by blind performance demonstration at EPRI in 2007 prior to the bolt inspections at ONS. The demonstration ensured the UT examination procedure's capability of determining the integrity of the UCB bolts. This demonstration was documented by EPRI with essential elements identified.

After the ONS UCB bolt examinations were completed in 2008, an ONS-specific technical justification (TJ) in accordance ASME B&PV Code Section V Article 14 for both the UCB and LCB bolts examinations was created for both UCB and LCB bolts by compiling existing information in one document. In addition to providing a detailed explanation of the examination process and other influential parameters important to the overall performance of the examination system, the TJ contains a description of the component, manufacturing history, flaws of interest, and operating history. Appendix E of this report provides a nonproprietary version of the UCB and LCB bolt ONS-specific TJ. It is an NEI 03-08 "Needed" requirement that TJs be created for each examination procedure in accordance with Section 2.1 of MRP-228, except for visual examinations. ONS-specific TJs are being prepared for visual and UT examination methods to be used for inspecting "Primary" and "Expansion" RV Internals components.

The evaluation criteria for the most recent ONS UCB bolt examinations were based on the stress limits for threaded structural fasteners in Subsection NG of the ASME B&PV Code. Using an analytical tool developed under PWROG PA-MSC-0350, the ONS unit-specific analysis demonstrates large margins using the most recent UCB bolt UT inspection results.

4.2.8 Core Clamping Measurements

Core clamping measurements were obtained by AREVA NP at ONS-1 (2006), ONS-2 (2008), and ONS-3 (2007) during RFOs and satisfy the MRP-227, Rev. 0 requirements for a one-time physical differential height measurement of the plenum rib pad to RV seating surface. The reason for the timing of the measurements was a concern that loss of core clamping force could be a contributor to wear between baffle plates and fuel grids (see Section 4.1.6 of this report). The measurements at the three units found no evidence of wear occurring during the service period of operation and it was concluded there was no evidence that core clamping has been degraded. Wear of the core clamping items in the plenum cover assembly and core support shield assembly will continue to be monitored via subsequent VT-3 examinations performed on the 10-year ISI interval per MRP-227, Rev. 0 requirements.

4.2.9 Visual Examination of Baffle-to-Baffle and Baffle-to-Former Bolts

In addition to the ASME B&PV Code Section XI ISI inspection requirement discussed in Section 4.2.1 of this report, Duke Energy has voluntarily performed visual inspection by underwater camera every RFO of the internal baffle-to-baffle bolts and baffle-to-former bolts at ONS units, per model work orders. The visual inspection is not required by the ASME B&PV Code and therefore is not conducted in accordance with the ASME B&PV Code. No out of design configuration baffle-to-baffle bolts and baffle-to-former bolts have been observed during these visual inspections at ONS units.

The only abnormal condition for the baffle-to-baffle and baffle-to-former bolts in the B&W units was noted from the inspection of the baffle-to-former and internal baffle-to-baffle bolts at Crystal River Unit 3 (CR-3) in 2005. Visual inspections indicated that three or four internal baffle-to-baffle bolts were not within the design configuration. The bolt heads extended beyond the baffle plate surface. This was an indication that the locking devices, and potentially the baffle-to-baffle bolts as well, had failed. A UT inspection of 100% of the baffle-to-former bolts at CR-3 was performed with no indications of broken baffle-to-former bolts. No UT inspection was performed on the internal baffle-to-baffle bolts. The abnormal baffle-to-baffle bolts at CR-3 have not been removed for laboratory examination to confirm the failures.

4.2.10 ONS Unit-Specific Amendments to MRP-227, Rev. 0 Requirements

Due to new information found during an ONS fabrication records search and the development of TJs, six amendments to MRP-227, Rev. 0 were identified. The full description of needed ONS unit-specific amendments to MRP-227, Rev. 0 requirements and their bases are provided in Appendix F of this report. A synopsis of each amendment is given in the bullets below.

- The CSS vent valve discs at ONS-1 and ONS-2 are recategorized to "No Additional Measures". No augmented inspection is required. This recategorization will apply to the ONS-3 installed vent valve discs after the ferrite is confirmed to be below the screening criteria.
- The CSS cast outlet nozzles at ONS-3 are recategorized to "No Additional Measures". No augmented inspection is required.
- The CRGT spacer castings at the three ONS units are recategorized to "Primary". The expansion link, inspection coverage, method, and examination frequency are listed in Appendix F of this report.
- The CSS vent valve disc shaft (hinge pin) is inaccessible for visual inspection. The examination method/frequency and examination coverage in MRP-227, Rev. 0 Tables 4-1 and 5-1 for the CSS vent valve disc shaft (hinge pin) at ONS-1, ONS-2, and ONS-3 are revised as described in Appendix F of this report.
- The aging effect (mechanism) in Table 4-1 of MRP-227, Rev. 0 for the UCB and LCB bolt and bolt locking devices is clarified as described in Appendix F of this report.
- The aging effect (mechanism) in Table 4-4 of MRP-227, Rev. 0 for the UTS, LTS, and FD bolt and bolt locking devices is clarified as described in Appendix F of this report.

4.3 Conclusions of Section 4.0

The ONS RV Internals inspection plan is based on the ONS LRA, NUREG-1723, and evaluations supporting MRP-227, Rev. 0. Inspections will consist of the ASME B&PV Code Section XI Examination Category B-N-3 inspections given in Table 4-1 of this report and the augmented inspections from MRP-227, Rev. 0, as modified by the amendments discussed in Appendix F of this report. Changes resulting from the NRC's review of this report and MRP-227, Rev. 0 will be incorporated as appropriate.

Table A.1 in Appendix A of this report shows how the components identified in BAW-2248A, Table 4-1 AMR were evaluated and characterized by the industry program. Past and on-going activities by the EPRI PWR MRP, B&WOG, PWROG, and Duke Energy provide the needed clarification to the level of inspection quality necessary to determine the proper examination method and frequencies. The ONS RV Internals AMP includes existing ONS programs and ASME B&PV Code Section XI inspections combined with MRP-227, Rev. 0 augmented inspections to provide reasonable assurance that the ONS RV Internals components will continue to perform their intended functions through the period of extended operation.

5.0 ONS RV INTERNALS AMP ATTRIBUTE EVALUATION

The ONS RV Internals AMP, which will include this ONS RV Internals inspection plan, utilizes a combination of prevention, mitigation, and condition monitoring. Where applicable, credit is taken for existing programs (e.g., primary water chemistry and ASME B&PV Code Section XI inspections) and mitigation projects such as LTS bolt replacement. The ONS RV Internals inspection plan then incorporates recommendations for augmented inspections provided by industry guidelines in MRP-227, Rev. 0 as applicable to ONS.

This section uses the ten AMP elements from NUREG-1801, Rev. 1 to describe the ONS RV Internals inspection plan.

5.1 AMP Element 1 – Scope of Program

The ONS RV Internals AMP is focused on managing the effects of the eight age-related degradation mechanisms given in Section 2.3 of this report and ensuring the RV Internals remain functional during the license renewal period. The ONS RV Internals AMP consists of not only the ONS RV Internals inspection plan ("Primary" and "Expansion" inspections from MRP-227, Rev. 0, as applicable to ONS, and ASME B&PV Code Section XI Examination Category B-N-3 inspections) but also credits programs such as the primary water chemistry program and the vent valve in-service test program and preventative actions such as the LTS bolt replacement. The ONS RV Internals inspection plan is focused on detecting possible degradation effects from the eight aging degradation mechanisms. The components chosen for inspection are the result of the LRA AMR refined by industry activity that culminated in MRP-227, Rev. 0.

5.1.1 ONS Scope

The ONS RV Internals consist of two basic assemblies located inside the RV[3]:

- Plenum Assembly

  The plenum assembly provides continuous guidance and protection of the control rods. In addition, the plenum assembly directs flow out of the core to the vessel outlet nozzles. The plenum assembly is removed every RFO to permit access to the fuel assemblies.

- Core Support Assembly (CSA)

  The CSA remains in place in the RV and is only removed to perform scheduled inspections of the RV interior surfaces and attachments as well as the RV Internals.

A description of the ONS RV Internals intended functions is provided in Section 2.4 of this report. Additional RV Internals details are provided in the ONS UFSAR.

5.1.2 ONS RV Internals Components Subject to an AMR

The components of the ONS RV Internals that were evaluated by the industry, culminating in MRP-227, Rev. 0, include those identified in the ONS LRA AMR and those contained in BAW-2248A. Table A.1 in Appendix A of this report is included to show how the components identified in the AMR from BAW-2248A were evaluated and characterized by the industry program resulting in MRP-227, Rev. 0. The components evaluated also include the components identified in NUREG-1801, Rev. 1.

5.1.3 Conclusion

The scope of the ONS RV Internals inspection plan includes the specific structures and components subject to an AMR and encompasses those components in the LRA AMR and those listed in Table IV.B4 of NUREG-1801, Rev. 1, thus satisfying the regulatory criteria in 10 CFR 54.

5.2 AMP Element 2 – Preventative Actions

The ONS RV Internals AMP includes the following existing programs and activities that comply with the requirements of this AMP element. Maintaining high water purity reduces susceptibility to SCC and IASCC.

Reactor coolant water chemistry is monitored and maintained in accordance with the EPRI Pressurized Water Reactor Chemistry Guidelines.[21] Additionally, Duke Energy will continue to use low-leakage core loading patterns as a preventative action. A description of the Primary Water Chemistry Control Program, reference to the commitment to low-leakage cores, and their applicability to the ONS RV Internals inspection plan is provided in the following subsections.

5.2.1 Primary Water Chemistry Control Program

The ONS Primary Water Chemistry Program, as implemented by the ONS primary water chemistry program limits the concentration of oxygen, halogens, and sulfate species in the primary water to prevent the coolant from becoming an environment favorable to SCC, and therefore effectively prevents SCC and greatly reduces the probability of IASCC. The limits imposed by the primary water chemistry program meet the intent of the Pressurized Water Reactor Chemistry Guidelines.[21] MRP-227, Rev. 0, the ONS LRA, and the GALL report credit the water chemistry AMP for mitigation of material loss due to crevice and pitting corrosion, as well as SCC.

5.2.2 Low–Leakage Cores

As discussed in Section 3.4.3.3 of NUREG-1723, Duke Energy will continue to use low-leakage core loading patterns, which are considered a preventative action to lessen the effects of aging on the ONS RV Internals.

5.2.3 Conclusion

The preventative actions for the ONS RV Internals AMP include the Primary Water Chemistry program as well as low-leakage core loading patterns, to mitigate the applicable aging effects, thus satisfying the regulatory criteria in 10 CFR 54.

5.3 AMP Element 3 – Parameters Monitored or Inspected

The ONS RV Internals inspection plan monitors for the detectable effects of the eight aging degradation mechanisms outlined in Section 2.3 of this report. The ONS RV Internals inspection plan credits, and further augments, the ASME B&PV Code Section XI, Table IWB-2500-1 inspections with the inspections listed in MRP-227, Rev. 0, Tables 4-1 and 4-4 as applicable to ONS. For "Expansion" bolts that are inaccessible, the PWROG is performing a justification by evaluation under PA-MSC-0692. TLAAs identified in NUREG-1723 for "flow-induced vibration endurance limit assumptions", "transient cycle count assumptions for replacement bolting", and "reduction in fracture toughness" have been resolved as discussed in Section 4.1.3 of this report.

The ONS RV Internals inspection plan uses UT, VT-3, and physical measurement to monitor for the detectable effects of the eight aging degradation mechanisms outlined in Section 2.3 of this report. UT is used to detect cracking in bolts. VT-3 is used to identify the conditions detailed in ASME B&PV Code Section XI, IWB-3520.

5.3.1 The ONS In-Service Inspection Program

The ONS ASME B&PV Code Section XI ISI Program (as discussed in Section 4.18 of the ONS LRA[3] and Section 4.2.1 of this report) is credited for inspection of numerous RV Internals components requiring aging management, as identified by the AMR and listed in Table A.1 in Appendix A of this report. The Examination Category B-N-3 components to be examined during the upcoming 10-year RV ISI are listed in Table 4-1 of this report.

Though some "Primary" and "Expansion" components are also listed as Examination Category B-N-3 items in the ISI Program, no "Existing Programs" were identified for B&W plants in MRP-227, Rev. 0. This was done in order to provide further guidance with respect to examination coverage and relevant conditions, as well as highlight the possible need for further evaluations. If a component receives a VT-3 inspection to satisfy both the ASME B&PV Code Section XI ISI program and MRP-227, Rev. 0 "Primary" or "Expansion" examinations, the examinations may be completed at the same time.

5.3.2 MRP-227, Rev. 0 "Primary" and "Expansion" (Augmented) Inspections

MRP-227, Rev. 0 "Primary" and "Expansion" inspections will be listed as augmented (required non-ASME B&PV Code Section XI inspections) inspections in the ONS ISI program. MRP-227, Rev. 0, Table 4-1 "B&W Plants Primary Components" and Table 4-4 "B&W Plants Expansion Components" are included as Appendices B and C of this report, respectively. In addition to listing the components requiring inspections, the Tables provide the Applicability (Plant), Effect (Mechanism), Expansion/Primary (Link), Examination Method, Frequency (for "Primary" components), and Examination Coverage. Appendix D of this report contains MRP-227, Rev. 0, Table 5-1 "B&W Plant Examination NDE Acceptance and Expansion Criteria," which provides Applicability, Examination Acceptance Criteria, Expansion Link, Expansion Criteria, and Additional Examination Acceptance Criteria for the applicable ONS components.

5.3.2.1 MRP-227, Rev. 0 "Primary" (Augmented) Inspections

MRP-227, Rev. 0 lists the following B&W "Primary" RV Internals bolts to be inspected with UT:

- UCB Bolts
- LCB Bolts
- Baffle-to-Former Bolts

MRP-227, Rev. 0 lists the following B&W "Primary" RV Internals components to be examined with a VT-3:

- Plenum Cover Weldment Rib Pads
- Plenum Cover Support Flange
- CSS Top Flange
- CSS Cast Outlet Nozzles (ONS-3 only) – ONS-3 Amendment, recategorized as "No Additional Measures" (see Appendix F of this report)
- CSS Vent Valve Disc – ONS Amendment, recategorized as "No Additional Measures" (see Appendix F of this report)
- CSS Vent Valve Top Retaining Ring
- CSS Vent Valve Bottom Retaining Ring
- UCB Bolt Locking Devices
- LCB Bolt Locking Devices
- Baffle Plates
- Baffle-to-Former Bolt Locking Devices including Locking Welds
- Internal Baffle-to-Baffle Bolt Locking Devices including Locking Welds
- Alloy X-750 Dowel-to-Guide Block Welds
- IMI Guide Tube Spiders
- IMI Guide Tube Spider-to-Lower Grid Rib Section Welds

During development of the TJ for the VT-3 visual inspection of the CSS vent valve disc shaft or hinge pin, it was discovered that the component was inaccessible and will need to be justified by evaluation with respect to the vent valve in-service test program or by replacement, as discussed in Appendix F of this report.

The following additional RV Internals component will be considered "Primary" via amendments and will be examined with a VT-3, as discussed in Appendix F of this report:

- CRGT Spacer Castings

MRP-227, Rev. 0 lists the following B&W "Primary" RV Internals components to be examined by physical measurement:

A one-time physical differential height measurement of the plenum rib pad to RV seating surface is required by MRP-227, Rev. 0. This measurement would indicate any change from the as-fabricated stacked height of the following components:

- Plenum Cover Weldment Rib Pads
- Plenum Cover Support Flange
- CSS Top Flange

As discussed in Section 4.2.7 of this report, the most recent inspections of the UCB bolts were performed in 2007 and 2008 in compliance with an NEI 03-08 "Needed" requirement from the PWROG. These UCB bolt inspections are being credited as the initial UT inspections required in MRP-227, Rev. 0, Table 4-1. The inspection results were the same as previous inspections performed in 1984 through 1987. No bolts were rejected at ONS-1 and ONS-2. ONS-3 had two bolts rejected due to a lack of back wall reflection.

The one-time differential height physical measurements for loss of core clamping of the plenum rib pads to RV seating surface were completed in 2006, 2007, and 2008 in support of the root cause for baffle-to-fuel wear. There was no evidence of loss of clamping due to wear.

Section 4.2.10 of this report provides a summary of the ONS unit-specific amendments to MRP-227, Rev. 0 and the justification for the ONS amendments to MRP-227, Rev. 0 is included in Appendix F of this report.

5.3.2.2 MRP-227, Rev. 0 "Expansion" (Augmented) Inspections

MRP-227, Rev. 0 lists the following B&W "Expansion" RV Internals bolts to be inspected with UT:

- Upper Thermal Shield (UTS) Bolts
- Lower Thermal Shield (LTS) Bolts
- Flow Distributor (FD) Bolts

MRP-227, Rev. 0 lists the following B&W "Expansion" RV Internals components to be examined with a VT-3:

- Alloy X-750 Dowel-to-Upper Fuel Assembly Support Pad Welds
- CRGT Spacer Castings – ONS Amendment, recategorized as "Primary" (see Appendix F of this report)
- Lower Fuel Assembly Support Pad
- Lower Fuel Assembly Support Pad-to Rib Section Welds
- Lower Fuel Assembly Support Pad Alloy X-750 Dowel
- Lower Fuel Assembly Support Pad Cap Screws and their Locking Welds
- Alloy X-750 Dowel-to-Lower Fuel Assembly Support Pad Welds

VT-3 examinations of the locking devices are added to the "Expansion Items" for ONS units per the amendments in Section 4.2.10 of this report.

- UTS Bolt Locking Devices
- LTS Bolt Locking Devices
- FD Bolt Locking Devices

MRP-227, Rev. 0 lists the following B&W RV Internals "Expansion" components as inaccessible:

- Core Barrel Cylinder (Including Vertical and Circumferential Seam Welds)
- Former Plates

- Baffle-to-Baffle Bolts (External)
- Core Barrel-to-Former Bolts
- External Baffle-to-Baffle Bolts Locking Devices, including Locking Welds
- Core Barrel-to-Former Bolts Locking Devices, including Locking Welds

For baffle-to-baffle bolts and core barrel-to-former bolts that are inaccessible, the PWROG is performing a justification by evaluation.

As discussed in Section 4.2.10 of this report, the justification for the ONS amendments to MRP-227, Rev. 0 is included in Appendix F of this report.

5.3.3 *Conclusion*

The parameters monitored or inspected in the ONS RV Internals inspection plan along with the TLAAs and evaluations of "Expansion" components are linked to the effects of aging on the intended functions of the particular structure and components. The inspections performed in accordance with ASME B&PV Code Section XI, the augmented inspections from MRP-227, Rev. 0 (as applicable to ONS), TLAAs, and evaluations provide reasonable assurance that the intended functions of the RV Internals discussed in Section 2.4 of this report will continue to be met through the period of extended operation.

The parameters to be monitored or inspected are the result of industry activity that satisfies the regulatory criteria listed in 10 CFR 54.21.

5.4 AMP Element 4 – Detection of Aging Effects

The methods, coverage and schedule of the ONS RV Internals inspection plan along with the other programs credited in Section 5.1 of this report are designed to ensure that aging effects will be detected in a timely manner in order to maintain the intended functions of the RV Internals components. The methods used for detection are a one-time physical measurement, ASME B&PV Code Section XI Examination Category B-N-3 VT-3 examination, augmented VT-3 examination to detect specific aging effects, augmented UT examinations to detect cracking of bolting, and justification by evaluation for inaccessible components.

The MRP-227, Rev. 0 augmented inspections, as applicable to ONS, will be entered into the ISI program as augmented inspections and the results will be included in the 90 day outage report. There is also a NEI 03-08 "Good Practice" contained in MRP-227, Rev. 0 to provide a summary of inspection results to the MRP within 120 days (see Section 4.1.1.1.1 of this report).

The visual and UT examination data will be recorded and stored. Inspection reports will also be provided by the inspection vendor. These examinations have been scheduled to be performed at ONS during the 2012, 2013, and 2014 RFOs.

5.4.1 One-Time Physical Measurement

A one-time physical differential height measurement of the plenum rib pad to RV seating surface at all three ONS units was conducted between 2006 and 2008 to ensure the clamping force holding the RV Internals was adequate, and thus rule out loss of clamping force as a possible cause of baffle plate to fuel grid wear as discussed in Section 4.1.6 of this report. There was no evidence of loss of clamping due to wear. The data collection for the one-time physical measurement was recorded by AREVA NP.

5.4.2 ASME B&PV Code Section XI Examination Category B-N-3 VT-3 Examinations

ASME B&PV Code Section XI Examination Category B-N-3 VT-3 examinations of accessible core support locations are conducted on a schedule in accordance with Table IWB 2500-1. The ASME B&PV Code Section XI VT-3 examination is intended to detect the type of general degradation condition identified in ASME B&PV Code Section XI IWB-3520.

5.4.3 Augmented VT-3 Examination to Detect Specific Aging Effects

The purpose of the augmented MRP-227, Rev. 0 "Primary" and "Expansion" VT-3 examinations is to detect aging effects of the eight aging degradation mechanisms outlined in Section 2.3 of this report. The method, coverage, schedule, and frequency of examinations are identified in MRP-227, Rev. 0, Table 4-1 and Table 4-4 and contained in Appendices B and C of this report, and for ONS unit-specific amendments to MRP-227, Rev. 0 are found in Appendix F of this report. Technical Justifications in accordance with ASME B&PV Code Section V, Article 14 Low Rigor are being prepared for each component receiving an augmented VT-3 examination. Where fitting, additional analyses are being performed to complete the TJs and demonstrate why the examination method, schedule, frequency, and coverage are appropriate.

For VT-3 examinations performed within the core barrel region, the functionality analysis (finite element model) performed to support MRP-227, Rev. 0 indicates the inspections are being performed in a timely manner to prevent any loss of intended function. Additionally, ASME B&PV Code Section XI VT-3 examinations and B&W plant OE support the timeliness of the augmented VT-3 examinations.

5.4.4 Augmented UT Examinations to Detect Cracking of Bolting

The purpose of the augmented MRP-227, Rev. 0 "Primary" and "Expansion" UT examinations is to detect cracking in bolts resulting from some of the eight identified aging degradation mechanisms outlined in Section 2.3 of this report. The method, coverage, schedule, and frequency of examinations are identified in MRP-227, Rev. 0, Table 4-1 and Table 4-4 and contained in Appendices B and C of this report. Technical Justifications in accordance with ASME B&PV Code Section V, Article 14 Low Rigor are being prepared for each component receiving an augmented UT examination. Appendix E of this report is provided as a representation of what will be contained in the TJs for the ONS units. This document, along with the EPRI letter containing the demonstration results, will make up the TJ for the UCB and LCB bolts.

The timing of the B&W RV Internals structural bolt inspections (UCB, LCB, UTS, LTS, FD and SSHT bolts, note ONS units no longer have SSHT bolts) is based on past inspections performed due to failures of Alloy A-286 bolting that occurred in the 1980s. The MRP-227, Rev. 0 inspection requirements for B&W RV Internals structural bolts are similar to the B&WOG NEI 03-08 "Needed" requirements which are presently in effect. The functionality analysis (finite element model) for the baffle-to-former bolts performed to support MRP-227, Rev. 0 indicates the inspections are being performed in a timely manner to prevent any loss of intended function.

The latest UT examinations of the UCB bolts at ONS units were completed in 2007 and 2008 (see Section 4.2.7 of this report).

5.4.5 Justification by Evaluation for Inaccessible Bolts

Augmented MRP-227, Rev. 0 "Expansion" components identified as inaccessible are listed in MRP-227, Rev. 0 Table 4-4 and contained in Appendix C of this report. Additionally, the "Primary" vent valve disc shaft or hinge pin has been identified as being inaccessible (see Appendix F of this report). Additional analyses are being performed under PWROG PA-MSC-0692 "B&W Internals MRP-227 Phase II Functionality Criteria Core Barrel Assembly Modeling" to show functionality of the core barrel-to-former bolts and baffle-to-baffle bolts for the B&W plants through extended plant operation, regardless of potential failures in the linked "Primary" components.

5.4.6 *Conclusion*

The ONS RV Internals inspection plan ensures detection of aging effects before there is loss of any structure and component intended function, including aspects such as method, frequency, coverage, and schedule.

5.5 AMP Element 5 – Monitoring and Trending

RV Internals components at ONS are monitored and the results are used to develop potential trends in RV Internals aging management concerns.

The timing of one-time physical measurement or initial inspection along with inspection intervals are based on previous inspections, OE, and expert opinion. For components within the beltline region, results of the functionality analysis performed on a typical B&W plant indicates the choice of components and timing of inspections confirms the prediction of the effects of aging and timely corrective or mitigative actions. "Expansion" components have been defined if the scope of examination and re-examination needs to be expanded beyond the "Primary" group. Additionally, reporting requirements allow the industry to monitor and trend results, thus driving preemptive industry action through notifications and updating of the MRP-227 guidelines. EPRI PWR MRP will compile the summary of results into an overall history report which will track industry progress, aid in evaluation of significant issues, identify fleet trends, and determine any needed revisions to the MRP-227 guidelines. The industry (including Duke Energy) will be updated biennially by EPRI PWR MRP; the biennial report will serve to assist the monitoring and trending for AMPs established by the industry. Duke Energy will periodically review the industry inspection results and reports provided by EPRI PWR MRP.

See Section 4.1.1 of this report for a discussion of development of the industry program.

5.5.1 Conclusion

The monitoring and trending provided in the ONS RV Internals inspection plan allows for prediction of the extent of the affects of aging and timely corrective or mitigative actions.

5.6 *AMP Element 6 – Acceptance Criteria*

Two acceptance criteria will be discussed in this section. The first is examination acceptance criteria that define relevant conditions during component examinations. The second is functionality/engineering acceptance criteria, which dispositions relevant conditions, thus ensuring the particular structure and component intended functions are maintained under all CLB design conditions during the period of extended operation.

5.6.1 Examination Acceptance Criteria

Examination acceptance criteria identify the visual examination relevant condition(s), signal-based level, or relevance of an indication that requires formal disposition for acceptability. Section 5 of MRP-227, Rev. 0 provides the examination acceptance criteria for the "Primary" and "Expansion" components; Table 5-1 of MRP-227, Rev. 0 is provided in Appendix D of this report.

In addition, the criteria for expanding the examinations from the "Primary" components to include the "Expansion" components are provided. The examination acceptance criteria include:

- Specific, descriptive relevant conditions for the visual (VT-3) examinations;
- Specific relevant indications for volumetric (UT) examination of bolting.

Additionally, TJs are being developed for the ONS RV Internals component inspections for VT-3 and UT examinations. The TJs, where appropriate, will include further guidance with respect to examination coverage and relevant conditions.

Relevant conditions requiring corrective action for ASME B&PV Code Section XI Examination Category B-N-3 VT-3 examinations of RV Internals components are detailed in ASME B&PV Code Section XI, IWB-3520.

Any detected condition that does not satisfy these examination acceptance criteria must be dispositioned. See Section 5.6.2 of this report for a discussion of functionality/engineering acceptance criteria.

5.6.2 *Functionality/Engineering Acceptance Criteria*

Based on the identified condition and supplemental examinations, if required, the disposition process results in an evaluation and determination of whether to accept the condition until the next examination or repair or replace the item. Relevant conditions identified during ASME B&PV Code Section XI Examination Category B-N-3 VT-3 examinations of RV Internals components are evaluated per ASME B&PV Code Section XI, IWB-3142. For augmented MRP-227, Rev. 0 inspections, the results of the PWROG projects discussed below will allow for the development of acceptance criteria used to determine whether to accept the condition until the next examination or repair or replace the components.

The PWROG PA-MSC-0473 developed WCAP-17096, "Reactor Internals Acceptance Criteria Methodology and Data Requirements". For each of the "Primary" and "Expansion" components listed in MRP-227, Rev. 0, WCAP-17096 outlines the type of analyses required, required evaluation procedures, data required to support analysis, logic chart illustrating evaluation path and potential disposition options, and components that can be addressed on a generic basis.

PWROG PA-MSC-0350 developed the basis for functional acceptance criteria that consisted of researching and documenting the functional (design basis) requirements for the UCB and LCB bolts. Models that represent the UCB and LCB bolted connections of the RV Internals of the ONS units were developed to evaluate and disposition inspection results.

PWROG PA-MSC-0692 will develop the basis for functional acceptance criteria which will consist of researching and documenting the functional (design basis) requirements for selected items and loading conditions that may exist during operation. The locking devices for the bolts do not require a functionality analysis. The existing core barrel assembly (CBA) supermodel developed under PWROG PA-MSC-0350 will be modified for specific differences that are anticipated to cover the variations in the B&W-designed RV Internals. This will provide unit-specific tools that can be used to analyze the condition of the CBA. The items included in the model are (a) the "Primary" items baffle-to-former bolts and baffle plates; and (b) the "Expansion" items core barrel cylinder, former plates, baffle-to-baffle bolts, core barrel-to-former bolts. A safety assessment of the consequence of failed inaccessible bolts (core barrel-to-former bolts, baffle-to-baffle bolts) and associated locking devices will also be performed. Task 4 of this PA is to "Develop Plan and Phase III Scope Definition" for future work.

5.6.3 Conclusions

Acceptance criteria for the ONS RV Internals inspection plan, against which the need for corrective action will be evaluated, will ensure the particular structure and component intended functions are maintained under all CLB design conditions during the period of extended operation.

5.7 AMP Element 7 – Corrective Actions

In accordance with 10 CFR 50, Appendix B, Duke Energy has established a corrective action program for the ONS units.

Measures are established in the ONS corrective action program to ensure conditions adverse to quality are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures ensure the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to appropriate levels of management.

5.7.1 ONS PIP Program

Duke Energy uses the PIP program and the Work Management System (WMS) to implement its Corrective Action Program per 10 CFR 50, Appendix B. The PIP program for the ONS units includes topics such as responsibilities, timeliness guidelines, action categories, reportability, and problem evaluation. The WMS and corrective maintenance function are established to identify and resolve the normal and expected degradation.

5.7.2 ONS Root Cause

Cause analysis is an essential part of an effective corrective action program. Root cause analysis prevents repetitive or similar problems by the identification and correction of specific causes of failures. The ONS cause analysis program provides a systematic approach to identify the fundamental reason or cause for a problem that has occurred and includes topics such as responsibilities, qualifications of personnel, cause analysis process, and record retention requirements. This directive is used in conjunction with the ONS PIP program on degraded conditions identified as needing a root cause analysis.

5.7.3 Operability

The ONS Operability/Functionality program applies to degraded/non-conforming conditions and unanalyzed conditions associated with structures, systems, and components (SSCs) that perform specified functions as set forth in the CLB. This directive compliments the guidance in the ONS PIP program for the resolution of degraded and/or nonconforming conditions.

5.7.4 Conclusion

Corrective actions, including root cause determination and prevention of recurrence, are timely.

5.8 AMP Element 8 – Confirmation Process

Duke Energy Topical Report Duke-1-A, "Quality Assurance Program" [24] describes the Duke Energy Quality Assurance (QA) program for the operational phase of its nuclear power plants. The Duke Energy QA program conforms to applicable regulatory requirements such as 10 CFR 50, Appendix B and to approved industry standards such as ANSI N45.2-1977 and ANSI N18.7-1976 and corresponding daughter standards or to equivalent alternatives. Duke Energy regularly reviews the status and adequacy of the QA program.

At ONS, Independent Nuclear Oversight (INOS) is assigned the QA functions of (1) assuring that an appropriate QA program is established and effectively executed; and (2) verifying, such as by checking, auditing, and inspecting, that activities affecting the safety related functions have been correctly performed.

5.8.1 Conclusion

Duke Energy's QA program ensures preventative actions are adequate and appropriate corrective actions have been completed and are effective.

5.9 AMP Element 9 – Administrative Controls

Administrative controls, including Duke Energy-specific documents used to implement this ONS RV Internals inspection plan, provide for a formal review and approval process.

5.9.1 Conclusion

Duke Energy's administrative controls provide for a formal review and approval process.

5.10 AMP Element 10 – Operating Experience

MRP-227, Rev. 0 establishes new augmented inspection guidance for the B&W units. Accordingly, there is no direct programmatic history for the bulk of these inspections. The program is based upon industry OE that is contained in MRP-227, Rev. 0, MRP-231, research data, and vendor evaluations. Development of the program relied upon the consensus review and inputs of the MRP Reactor Internals Core and Focus Groups, which include representatives from utilities, research scientists, and vendors. The MRP-227, Rev. 0 guidelines will continue to evolve as additional operating experience and augmented inspection results are gained. PWR RV Internals failures, both domestic and international, have been considered in the development of MRP-227, Rev. 0.

The ONS Operating Experience Program (OEP) defines and communicates Duke Energy's OE Program and management expectations for the use of OE information. This program also defines and communicates expectations for the receipt, evaluation, and distribution of OE information and the resolution of applicable OE items.

5.10.1 Incidents of Degradation in B&W RV Internals

Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. In B&W units, the incidents observed have been limited to the baffle-to-baffle bolts (see Section 4.2.9 of this report), RV Internals structural bolting (see Sections 4.1.4 and 4.2.7 of this report), locking devices for the vent valve jackscrews (see Section 4.2.6.1 of this report), and interaction between the baffle plates and fuel assembly grid straps (see Section 4.1.6 of this report).

5.10.2 Conclusion

Operating experience has been used in the development of the ONS RV Internals inspection plan, thus ensuring the effects of aging will be adequately managed so the structure and component intended functions described in Section 2.4 of this report will be maintained during the period of extended operation.

5.11 Program Conclusion

Section 5.0 of this report shows that the ONS RV Internals inspection plan complies with the ten AMP elements from NUREG-1801, Rev. 1; compliance with these ten AMP elements demonstrates adequacy of managing aging effects of the ONS RV Internals.

6.0 SUMMARY AND CONCLUSIONS

This report documents and provides a description of the ONS-1, ONS-2, and ONS-3 RV Internals inspection plan and how it relates to the RV Internals AMP at ONS for management of aging effects consistent with previous commitments. This ONS RV Internals inspection plan is based on MRP-227, Rev. 0, as modified by the ONS unit-specific amendments identified in Section 4.2.10 of this report. Section 5.0 of this report has demonstrates that the ONS RV Internals AMP, which contains the ONS RV Internals inspection plan, will comply with the ten AMP elements described in NUREG-1801, Rev. 1.

This ONS RV Internals inspection plan contains a discussion of the background of the B&W-designed plant RV Internals programs, including operational experience, TLAAs, and existing ONS programs.

The examinations required by ASME B&PV Code Section XI, MRP-227, Rev. 0, and ONS unit-specific amendments as described in this report have been scheduled to be performed at ONS during the 2012, 2013, and 2014 RFOs, and any relevant conditions will be documented and dispositioned in Duke Energy's corrective action program and reported to the industry.

The ONS RV Internals AMP will include this ONS RV Internals inspection plan and will demonstrate that the program adequately manages the effects of aging for RV Internals components and establishes the basis for providing reasonable assurance that the RV Internals components will remain functional through the ONS license renewal period of extended operation.

This ONS RV Internals inspection plan, along with the ONS RV Internals AMP, fulfills the approved license renewal methodology requirement to identify the most susceptible components and to inspect those components with an indication detection level commensurate with the expected degradation mechanism indication, thus meeting 10 CFR 54.

Once this ONS RV Internals inspection plan is approved, the ONS UFSAR will be updated as required.

7.0 REFERENCES

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2. NUREG-1801, Revision 1, "Generic Aging Lessons Learned (GALL) Report," Dated September 2005.

3. Letter from Duke Energy Corporation forwarding application for renewal of operating licenses for the Oconee Nuclear Station, Unit Nos. 1, 2, and 3. Requests extension of operating licenses to 20 years beyond current expiration dates. July 6, 1998. NRC Accession No.: 9807200136

4. Duke Energy Corporation, Oconee Nuclear Station Units 1, 2, and 3 Renewed Facility Operating License," May 23, 2000. NRC Accession No.: ML013510445

5. U. S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3," NUREG-1723, March 31, 2000. NRC Accession No.: ML003697717

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7. BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," July 1997. NRC Accession No.: ML003708443

8. U. S. Nuclear Regulatory Commission Letter, "Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled, 'Demonstration of the Management of Aging Effects for the Reactor Vessel Internals'," BAW-2248, July 1997. NRC Accession No.: ML993490288

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10. Duke Power Company, Oconee Nuclear Station Units 1, 2, and 3 Reactor Coolant System Aging Management Review for License Renewal, OSS-0274.00-00-0004, Revision 1, August 2001.

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12. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Rules for In-Service Inspection of Nuclear Power Plant Components, 1998 Edition with 2000 Addenda, American Society of Mechanical Engineers, New York, NY.

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14. Nuclear Energy Institute, "Guideline for the Management of Materials Issues," NEI 03-08, Revision 2, January 2010.

15. Materials Reliability Program: Inspection Standard for Reactor Internals (MRP-228). EPRI, Palo Alto, CA: 2009. 1016609.

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17. NRC letter (from D. Matthews) forwards safety evaluation report regarding licensee's July 6, 1998, application to NRC for renewal of Oconee Nuclear Stations 1, 2, and 3 operating license for additional 20 years. Open items must be resolved before NRC can make final determination on application. NRC Accession No. 9906210071.

18. Memorandum (signed by: J. Sebrosky) forwards Duke editorial comments on June 1996 SER related to license renewal application of Oconee Nuclear Station Units 1, 2, and 3. Duke intends to provide additional comments on SER of more substantial nature in separate letter. NRC Accession No.: 9910220085

19. Materials Reliability Program: Stress Corrosion Cracking of High Strength Reactor Vessel Internals Bolting in PWRs (MRP-88), EPRI, Palo Alto, CA: 2003. 1003206.

20. "Briefing Material from June 14, 2005 Mtg on Rx Vessel Internals from the EPRI MRP – HT Tang", June 14, 2005. NRC Accession No. ML051710058.

21. Pressurized Water Reactor Primary Water Chemistry Guidelines: Volume 1, Revision 5, EPRI, Palo Alto, CA: 2003. 1002884.

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23. NUREG/CR-4513, Rev. 01, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in LWR Systems," U.S. Nuclear Regulatory Commission, May 1994.

24. Duke Energy Carolinas Topical Report, Duke-1-A, "Quality Assurance Program," Revision 37, February 2010. NRC Accession No. ML100610381

APPENDIX A: BAW-2248A AMR AND INDUSTRY PROGRAM COMPARISON

Table A.1 in Appendix A shows how the components identified in the BAW-2248A, Table 4-1 AMR were evaluated and characterized by the industry program.

Table A.1. AMR and MRP-227, Rev. 0 Comparison		
Affected Parts	**Aging Effect**	**Programs That Manage Applicable Aging Effects**
PLENUM ASSEMBLY		
Upper Grid Assembly		
Upper Grid Assembly	Cracking of base metal and welds	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Category A, except for the No Additional Measures X-750 dowel locking weld MRP-189 Rev. 1 (Tables 4-1 and 4-2) and MRP-231, Rev. 1 (Tables 3-8 and 3-9)
	Reduction of Fracture Toughness of base metal and welds	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Category A, MRP-189 Rev. 1 Tables 4-1 and 4-2
Fuel Assembly Support Pads	Loss of Material	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval. MRP Category A, except for the Expansion X-750 dowel locking weld MRP-189 Rev. 1 (Tables 4-1 and 4-2) and MRP-231, Rev. 1 (Tables 3-8 and 3-9)

Table A.1. AMR and MRP-227, Rev. 0 Comparison		
Affected Parts	**Aging Effect**	**Programs That Manage Applicable Aging Effects**
PLENUM ASSEMBLY (continued)		
Plenum Rib Pads	Loss of Material	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval. MRP Primary, MRP-231, Rev. 1 (Tables 3-8 and 3-9)
Control Rod Guide Tube Assembly		
CRGT Flange to Upper Grid Screws	Loss of Closure Integrity	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Category A, MRP-189 Rev. 1 (Table 4-1)
CRGT Assembly Spacer Castings	Reduction of Fracture Toughness	RVI Aging Management Program See Section 4.6. MRP Expansion, MRP-231, Rev. 1 (Tables 3-8 and 3-10)

Table A.1. AMR and MRP-227, Rev. 0 Comparison		
Affected Parts	**Aging Effect**	**Programs That Manage Applicable Aging Effects**
CORE SUPPORT SHIELD ASSEMBLY		
Core Support Shield Cylinder Top Flange	Loss of Material	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval. MRP Primary, MRP-231, Rev. 1 (Tables 3-8 and 3-9)
Core Support Shield to Core Barrel Bolts	Cracking	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Primary, MRP-231, Rev. 1 (Tables 3-8 and 3-9)
	Reduction of Fracture Toughness	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Category A, MRP-189 Rev. 1 (Table 4-1)
	Loss of Closure Integrity	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Category A, MRP-189 Rev. 1 (Table 4-1)

Table A.1. AMR and MRP-227, Rev. 0 Comparison		
Affected Parts	**Aging Effect**	**Programs That Manage Applicable Aging Effects**
CORE SUPPORT SHIELD ASSEMBLY (continued)		
ONS-3 CSS Outlet Nozzles	Reduction of Fracture Toughness	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval Supplemented by extension of the evaluation procedures in IWB-3640 to CASS items as specified in Section 4.2 of BAW-2243A [30]. RVI Aging Management Program See Section 4.6. MRP Primary, MRP-231, Rev. 1 (Tables 3-8 and 3-9) Notes: DB's outlet nozzles are also made of casting, but were not listed because DB was not a participant of BAW-2248A. The cast outlet nozzles at ONS-3 and DB have been determined to be below the NUREG/CR-4513 Rev. 1 screening level for thermal aging after the issuance of MRP-227.

Table A.1. AMR and MRP-227, Rev. 0 Comparison		
Affected Parts	**Aging Effect**	**Programs That Manage Applicable Aging Effects**
CORE SUPPORT SHIELD ASSEMBLY (continued)		
Vent Valve Bodies	Reduction of Fracture Toughness	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval. Supplemented by: Plant Technical Specifications (ANO-1, TMI-1) Vent valve testing and inspection requirements each refueling outage Pump and Valve In-Service Test Program (ONS-1,-2,-3) Vent valve testing and inspection requirements each refueling outage Further Supplemented by extension of the evaluation procedures in IWB-3640 to CASS items as specified in Section 4.2 of BAW-2243A [30]. RVI Aging Management Program See Section 4.6. MRP Category A, MRP-189 Rev. 1 (Table 4-1) Note: The vent valve body castings have been confirmed to be below the NUREG/CR-4513 Rev. 1 screening level for thermal aging.

Table A.1. AMR and MRP-227, Rev. 0 Comparison		
Affected Parts	**Aging Effect**	**Programs That Manage Applicable Aging Effects**
CORE SUPPORT SHIELD ASSEMBLY (continued)		
Vent Valve Retaining Rings	Reduction of Fracture Toughness	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval. Supplemented by: Plant Technical Specifications (ANO-1, TMI-1) Vent valve testing and inspection requirements each refueling outage Pump and Valve In-Service Test Program (ONS-1,-2,-3) Vent valve testing and inspection requirements each refueling outage RVI Aging Management Program See Section 4.6. MRP Primary, MRP-231, Rev. 1 (Tables 3-8 and 3-9) [both top and bottom retaining rings]
Vent Valve Locking Devices (Modified)	Cracking	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval. MRP "No Additional Measures" (see Section 4.2.10), MRP-231, Rev. 1 (Table 3-9 Note 1) Supplemented by: Plant Technical Specifications (ANO-1, TMI-1) Vent valve testing and inspection requirements each refueling outage Pump and Valve In-Service Test Program (ONS-1,-2,-3) Vent valve testing and inspection requirements each refueling outage

Table A.1. AMR and MRP-227, Rev. 0 Comparison		
Affected Parts	**Aging Effect**	**Programs That Manage Applicable Aging Effects**
CORE SUPPORT SHIELD ASSEMBLY (continued)		
Vent Valve Locking Devices (Original)	Loss of Material	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval. MRP "No Additional Measures", MRP-231, Rev. 1 (Table 3-9 Note 1) Supplemented by: Plant Technical Specifications (ANO-1, TMI-1) Vent valve testing and inspection requirements each refueling outage Pump and Valve In-Service Test Program (ONS-1,-2,-3) Vent valve testing and inspection requirements each refueling outage
CORE BARREL ASSEMBLY		
Core Barrel Assembly	Cracking of base metal and welds	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Primary and Expansion, MRP-231, Rev. 1 (Tables 3-8, 3-9, and 3-10)
	Reduction of Fracture Toughness of base metal and welds	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Primary and Expansion, MRP-231, Rev. 1 (Tables 3-8, 3-9, and 3-10)

Table A.1. AMR and MRP-227, Rev. 0 Comparison		
Affected Parts	**Aging Effect**	**Programs That Manage Applicable Aging Effects**
CORE BARREL ASSEMBLY (continued)		
	Dimensional Change	RVI Aging Management Program See Section 4.6. MRP No Additional Measure, MRP-231, Rev. 1 (Tables 3-8)
Baffle-to-Baffle Bolts	Cracking	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP No Additional Measure and Expansion, MRP-231, Rev. 1 (Tables 3-8 and 3-10)
	Reduction of Fracture Toughness	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Expansion, MRP-231, Rev. 1 (Tables 3-8 and 3-10)
	Loss of Closure Integrity	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. Expansion, MRP-231, Rev. 1 (Tables 3-8 and 3-10)

Table A.1. AMR and MRP-227, Rev. 0 Comparison		
Affected Parts	**Aging Effect**	**Programs That Manage Applicable Aging Effects**
CORE BARREL ASSEMBLY (continued)		
	Dimensional Change	<u>RVI Aging Management Program</u> See Section 4.6. <u>MRP</u> No Additional Measure, MRP-231, Rev. 1 (Tables 3-8)
Baffle-to-Former Bolts	Cracking	<u>ASME Section XI ISI Programs</u> Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval <u>RVI Aging Management Program</u> See Section 4.6. <u>MRP</u> Primary, MRP-231, Rev. 1 (Tables 3-8 and 3-9)
	Reduction of Fracture Toughness	<u>ASME Section XI ISI Programs</u> Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval <u>RVI Aging Management Program</u> See Section 4.6. <u>MRP</u> Primary, MRP-231, Rev. 1 (Tables 3-8 and 3-9)

Table A.1. AMR and MRP-227, Rev. 0 Comparison		
Affected Parts	**Aging Effect**	**Programs That Manage Applicable Aging Effects**
CORE BARREL ASSEMBLY (continued)		
	Loss of Closure Integrity	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Primary, MRP-231, Rev. 1 (Tables 3-8 and 3-9)
	Dimensional Change	RVI Aging Management Program See Section 4.6. MRP No Additional Measure, MRP-231, Rev. 1 (Tables 3-8)
Lower Internals Assembly to Core Barrel Bolts	Cracking	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Primary, MRP-231, Rev. 1 (Tables 3-8 and 3-9)

Table A.1. AMR and MRP-227, Rev. 0 Comparison		
Affected Parts	**Aging Effect**	**Programs That Manage Applicable Aging Effects**
CORE BARREL ASSEMBLY (continued)		
	Reduction of Fracture Toughness	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Category A, MRP-189 Rev. 1 (Table 4-1)
	Loss of Closure Integrity	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Category A, MRP-189 Rev. 1 (Table 4-1)
Core Barrel to Thermal Shield Bolts	Cracking	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Expansion, MRP-231, Rev. 1 (Tables 3-8 and 3-10)

Table A.1. AMR and MRP-227, Rev. 0 Comparison		
Affected Parts	**Aging Effect**	**Programs That Manage Applicable Aging Effects**
CORE BARREL ASSEMBLY (continued)		
	Loss of Closure Integrity	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Category A,MRP-189 Rev. 1 (Table 4-1)
LOWER INTERNALS ASSEMBLY		
Lower Grid Assembly	Cracking of base metal and welds	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Note: The dowel locking weld for the guide blocks are "Primary" for SCC. MRP-231, Rev. 1 (Tables 3-8 and 3-9) Category A, except for the "Expansion" for the dowel locking welds for the lower grid fuel assembly support pads MRP-189 Rev. 1 (Tables 4-1 and 4-2) and MRP-231, Rev. 1 (Tables 3-8 and 3-10)

Table A.1. AMR and MRP-227, Rev. 0 Comparison		
Affected Parts	**Aging Effect**	**Programs That Manage Applicable Aging Effects**
LOWER INTERNALS ASSEMBLY (continued)		
	Reduction of Fracture Toughness of base metal and welds	<u>ASME Section XI ISI Programs</u> Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval <u>RVI Aging Management Program</u> See Section 4.6. <u>MRP</u> Category A, except for the for the "Expansion" items and welds associated with the lower grid fuel assembly support pads MRP-189 Rev. 1 (Tables 4-1 and 4-2) and MRP-231, Rev. 1 (Tables 3-8 and 3-10)
Lower Internals Assembly to Thermal Shield Bolts	Cracking	<u>ASME Section XI ISI Programs</u> Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval <u>RVI Aging Management Program</u> See Section 4.6. <u>MRP</u> Expansion, MRP-231, Rev. 1 (Tables 3-8 and 3-10)
	Loss of Closure Integrity	<u>ASME Section XI ISI Programs</u> Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval <u>RVI Aging Management Program</u> See Section 4.6. <u>MRP</u> Category A,MRP-189 Rev. 1 (Table 4-1)

Table A.1. AMR and MRP-227, Rev. 0 Comparison		
Affected Parts	**Aging Effect**	**Programs That Manage Applicable Aging Effects**
LOWER INTERNALS ASSEMBLY (continued)		
Fuel Assembly Support Pads	Loss of Material	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval. MRP Category A, MRP-189 Rev. 1 (Table 4-1) Note: The items and welds associated with the lower grid fuel assembly support pads are "Expansion" for SCC and/or irradiation embrittlement. MRP-231, Rev. 1 (Tables 3-8 and 3-10)
Guide Blocks	Loss of Material	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval. MRP Category A, MRP-189 Rev. 1 (Table 4-1) Note: The dowel locking weld for the guide blocks are "Primary" for SCC. MRP-231, Rev. 1 (Tables 3-8 and 3-9)
Shell Forging to Flow Distributor Bolts	Cracking	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Expansion, MRP-231, Rev. 1 (Tables 3-8 and 3-10)
	Loss of Closure Integrity	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP Category A, MRP-189 Rev. 1 (Table 4-1)

Table A.1. AMR and MRP-227, Rev. 0 Comparison		
Affected Parts	**Aging Effect**	**Programs That Manage Applicable Aging Effects**
LOWER INTERNALS ASSEMBLY (continued)		
Lower Grid Rib to Shell Forging Screws	Loss of Closure Integrity	ASME Section XI ISI Programs Examination Category B-N-3, Removable Core Support Structures VT-3 visual examination of accessible surfaces - Every interval RVI Aging Management Program See Section 4.6. MRP No Additional Measure, MRP-231, Rev. 1 (Tables 3-8)
Incore Guide Tube Spider Castings	Reduction of Fracture Toughness	RVI Aging Management Program See Section 4.6. MRP Primary, MRP-231, Rev. 1 (Tables 3-8 and 3-9)

APPENDIX B: TABLE 4-1 FROM MRP-227, REV. 0

Appendix B contains Table 4-1, "B&W Plants Primary Components" from MRP-227, Rev. 0. The "Primary" component inspections listed in the ONS RV Internals inspection plan are based on this table, as modified by the ONS unit-specific amendments identified in Section 4.2.10 and Appendix F of this report.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Frequency	Examination Coverage
Plenum Cover Assembly & Core Support Shield Assembly Plenum cover weldment rib pads Plenum cover support flange CSS top flange	All plants	Loss of material and associated loss of core clamping pre-load (Wear)	None	On-time physical measurement no later than two refueling outages from the beginning of the license renewal period. Perform subsequent visual (VT-3) examination on the 10-year ISI interval.	Determination of differential height of top of plenum rib pads to reactor vessel seating surface, with plenum in reactor vessel.
Core Support Shield Assembly CSS cast outlet nozzles	ONS-3, DB	Cracking (TE), including the detection of surface irregularities, such as damaged or fractured material	CRGT spacer castings	Visual (VT-3) examination during the next 10-year ISI. Subsequent examination on the 10-year ISI interval.	100% of accessible surfaces.
Core Support Shield Assembly CSS vent valve discs (Note 1)	All plants				100% of accessible surfaces.
Core Support Shield Assembly CSS vent valve top retaining ring CSS vent valve bottom retaining ring CSS vent valve disc shaft or hinge pin (Note 1)	All plants	Cracking (TE), including the detection of surface irregularities, such as damaged, fractured, or missing items	None	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible surfaces.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Frequency	Examination Coverage
Core Support Shield Assembly Upper core barrel (UCB) bolts and their locking devices	All plants	Cracking (SCC)	LCB bolts (Note 3) UTS, LTS, and FD bolts SSHT bolts (CR-3 and DB only) Lower shock pad bolts (TMI-1 only)	Volumetric examination (UT) of the bolts within two refueling outages from 1/1/2006 or next 10-year ISI interval, whichever is first. Subsequent examination to be determined after evaluation the baseline results. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts.
Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	All plants	Cracking (SCC)	UTS, LTS, and FD bolts SSHT bolts (CR-3 and DB only) Lower shock pad bolts (TMI-1 only)	Volumetric examination (UT) of the bolts during the next 10-year ISI interval from 1/1/2006. Subsequent examination to be determined after evaluation the baseline results. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts.
Core Barrel Assembly Baffle-to-former bolts	All plants	Cracking (IASCC, IE, IC/ISR/Fatigue/Wear, Overload)	Baffle-to-baffle bolts Core barrel-to-former bolts	Baseline volumetric examination (UT) no later than two refueling outages from the beginning of the license renewal period with subsequent examination after 10 to 15 additional years.	100% of accessible bolts.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Frequency	Examination Coverage
Core Barrel Assembly Baffle plates	All plants	Cracking (IE), including the detection of readily detectable cracking in the baffle plates	Core barrel cylinder (including vertical and circumferential seam welds) Former plates	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of the accessible surface within 1 inch around each flow and bolt hole.
Core Barrel Assembly Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	All plants	Cracking (IASCC, IE, Overload), including the detection of missing, non-functional, or removed locking devices or welds	Locking devices for the external baffle-to-baffle bolts and baffle-to-former bolts	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	100% of accessible baffle-to-former and internal baffle-to-baffle bolt locking devices.
Lower Grid Assembly Alloy X-750 dowel-to-guide block welds	All plants	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Alloy X-750 dowel locking welds to the upper and lower fuel assembly support pads	Initial visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examination on ten-year interval.	100% of accessible locking welds of the 24 dowel-to-guide block welds.
Incore Monitoring Instrumentation (IMI) Guide Tube Assembly IMI guide tube spiders IMI guide tube spider-to-lower grid rib section welds	All plants	Cracking (TE/IE), including the detection of fractured or missing spider arms or separation of spider arms from the lower grid rib section at the weld	CRGT spider castings Lower fuel assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds (Note: the pads, dowels, and cap screws are included because of TE/IE of the welds)	Initial visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on ten-year interval.	100% of accessible top surfaces of 52 spider castings and welds to the adjacent lower grid rib section.

Notes:

1. A verification of the operation of each vent valve shall also be performed through manual actuation of the valve. Verify that the valves are not stuck in the open position and that no abnormal degradation has occurred. Examine the valves for evidence of scratches, pitting, embedded particles, variation in coloration of the seating surfaces, cracking of lock welds and locking cups, jack screws for proper position, and wear. The frequency is defined in each unit's technical specifications or in their pump and valve in-service test programs (see Appendix A).
2. Examination acceptance criteria and expansion criteria for the B&W components are in Table 5-1 (of MRP-227, Rev. 0).
3. Expansion to LCB applies if the required Primary examination of LCB has not been performed as scheduled in this table.

APPENDIX C: TABLE 4-4 FROM MRP-227, REV. 0

Appendix C contains Table 4-4, "B&W Plants Expansion Components" from MRP-227, Rev. 0. The "Expansion" component inspections listed in the ONS RV Internals inspection plan are based on this table, as modified by the ONS unit-specific amendments identified in Section 4.2.10 and Appendix F of this report.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Upper Grid Assembly Alloy X-750 dowel-to-upper fuel assembly support pad welds	All plants (except DB)	Cracking (SCC) including the detection of separated or missing locking welds, or missing dowels	Alloy X-750 dowel-to-guide block welds	Visual (VT-3) examination.	100% of accessible dowel locking welds.
Control Rod Guide Tube Assembly CRGT spacer castings	All plants	Cracking (TE), including the detection of fractured spacers or missing screws	CSS cast outlet nozzle, CSS vent valve disks, or IMI guide tube spiders	Visual (VT-3) examination.	100% of accessible surfaces at the 4 screw locations (every 90°) (limited accessibility)
Core Barrel Assembly Upper thermal shield bolts	All plants	Cracking (SCC)	UCB and LCB bolts	Volumetric examination (UT)	100% of accessible bolts.
Core Barrel Assembly Surveillance specimen holder tube (SSHT) studs/nuts (CR-3) or bolts (DB)	CR-3, DB				
Core Barrel Assembly Core barrel cylinder (including vertical and circumferential seam welds) Former plates	All plants	Cracking (IE) including readily detectable cracking	Baffle plates	Justify by evaluation or by replacement	Inaccessible
Core Barrel Assembly Baffle-to-baffle bolts Core barrel-to-former bolts	All plants	Cracking (IASCC, IE, IC/ISR/Fatigue/Wear, Overload)	Baffle-to-former bolts	Internal baffle-to-baffle bolts: No examination requirements, Justify by evaluation or replacement	N/A
				External baffle-to-baffle bolts, Barrel-to-former bolts: No examination requirements, Justify by evaluation or replacement	Inaccessible

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Barrel Assembly Locking devices, including locking welds, for the external baffle-to-baffle bolts and core barrel-to-former bolts	All plants	Cracking (IASCC, IE)	Locking devices, including locking welds, of baffle-to-former bolts or internal baffle-to-baffle bolts	Justify by evaluation or by replacement	Inaccessible
Lower Grid Assembly Lower fuel assembly support pad items: pad, pad-to-rib sections welds, Alloy X-750 dowel, cap screw, and their locking welds (Note: the pads, dowels, and cap screws are included because of TE/IE of the welds)	All plants	Cracking (IE), including the detection of separated or missing welds, missing support pads, dowels, cap screws and locking welds, or misalignment or the support pads	IMI guide tube spiders and spider-to-lower grid rib section welds	Visual (VT-3) examination	100% of accessible pads, dowels, and cap screws, and associated welds.
Lower Grid Assembly Alloy X-750 dowel-to-lower fuel assembly support pad welds	All plants	Cracking (SCC), including the detection of separated or missing locking welds, or missing dowels	Alloy X-750 dowel-to-guide block welds	Visual (VT-3) examination	100% of accessible dowel welds.
Lower Grid Assembly Lower grid shock pad bolts	TMI-1	Cracking (SCC)	UCB and LCB bolts	Volumetric examination (UT)	100% of accessible bolts
Lower Grid Assembly Lower thermal shield (LTS) bolts **Flow Distributor Assembly** Flow distributor (FD) bolts	All plants	Cracking (SCC)	UCB and LCB bolts	Volumetric examination (UT)	100% of accessible bolts

Notes:

1. Examination acceptance criteria and expansion criteria for the B&W components are in Table 5-1 (of MRP-227, Rev. 0).

APPENDIX D: TABLE 5-1 FROM MRP-227, REV. 0

Appendix D contains Table 5-1, "B&W Plants Examination Acceptance and Expansion Criteria" from MRP-227, Rev. 0. The inspections listed in the ONS RV Internals inspection plan will use the acceptance and expansion criteria in this table, as modified by the ONS unit-specific amendments identified in Section 4.2.10 and Appendix F of this report.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Plenum Cover Assembly & Core Support Shield Assembly Plenum cover weldment rib pads Plenum cover support flange CSS top flange	All plants	One-time physical measurement. In addition, a visual (VT-3) examination is conducted for these items. The measured differential height from the top of the plenum rib pads to the vessel seating surface shall average less than 0.004 inches compared to the as-built condition. The specific relevant condition for these items is wear that may lead to a loss of function.	None	N/A	N/A
Core Support Shield Assembly CSS cast outlet nozzles	ONS-3, DB	Visual (VT-3) examination The specific relevant condition is evidence of surface irregularities, such as damaged or fractured nozzle material.	CRGT spacer castings	Confirmed evidence of relevant conditions for a single CSS cast outlet nozzle shall required that VT-3 examination be expanded to include 100% of the accessible surfaces at the 4 screw locations (at every 90°) of the CRGT spacer castings by the completion of the next refueling outage.	The specific relevant condition is evidence of fractured spacers or missing screws.
Core Support Shield Assembly CSS vent valve discs	All plants	Visual (VT-3) examination. The specific relevant condition is evidence of surface irregularities, such as damaged or fractured disc material.	CRGT spacer castings	Confirmed evidence of relevant conditions in two or more CSS vent valve discs shall require that the VT-3 examination be expanded to include 100% of the accessible surfaces at the 4 screw locations (at every 90°) of the CRGT spacer castings by the completion of the next refueling outage.	The specific relevant condition is evidence of fractured spacers or missing screws.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Support Shield Assembly CSS vent valve top retaining ring CSS vent valve bottom retaining ring CSS vent valve disc shaft or hinge pin	All plants	Visual (VT-3) examination. The specific relevant condition is evidence of damaged or fractured material, and missing items.	None	N/A	N/A

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Support Shield Assembly Upper core barrel (UCB) bolts and their locking devices	All plants	1) Volumetric (UT) examination of the UCB bolts. The examination acceptance criteria for the UT of the UCB bolts shall be established as a part of the examination technical justification. 2) Visual (VT-3) examination of the UCB bolt locking devices. The specific relevant condition for the VT-3 of the UCB bolt locking devices is evidence of broken or missing bolt locking devices.	LCB bolts (Note 2) UTS, LTS, and FD bolts SSHT bolts (CR-3 and DB only) Lower grid shock pad bolts (TMI-1 only)	1) Confirmed unacceptable indications exceeding 10% of the UCB bolts shall require that the UT examination be expanded by the completion of the next refueling outage to include: For all plants 100% of the accessible UTS, LTS, and FD bolts, Additionally for TMI-1 UT examination to include 100% of the accessible lower grid sock pad bolts. Additionally for CR-3 and DB UT examination to include 100% of the accessible SSHT bolts. 2) Confirmed evidence of relevant conditions exceeding 10% of the UCB bolt locking devices shall require that the VT-3 examination be expanded by the completion of the next refueling outage to include: For all plants 100% of the accessible UTS, LTS, and FD bolt locking devices Additionally for TMI-1 100% of the accessible lower grid shock pad bolt locking devices, Additionally for CR-3 and DB 100% of the accessible SSHT bolt locking devices	1) The acceptance criteria for the UT of the expansion bolting shall be established as part of the examination technical justification. 2) The specific relevant condition for the expansion of the VT-3 locking devices is evidence of broken or missing bolt locking devices.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	All plants	1) Volumetric (UT) examination of the LCB bolts. The examination acceptance criteria for the UT of the LCB bolts shall be established as part of the examination technical justification. 2) Visual (VT-3) examination of the LCB bolt locking devices. The specific relevant condition for the VT-3 of the LCB bolt locking devices is evidence of broken or missing bolt locking devices.	UTS, LTS, and FD bolts	1) Confirmed unacceptable indications exceeding 10% of the LCB bolts shall require that the UT examination be expanded by the completion of the next refueling outage to include: For all plants: 100% of the accessible UTS, LTS, and FD bolts Additionally for TMI-1 100% of the accessible lower grid shock pad bolts Additionally for CR-3 and DB 100% of the accessible SSHT bolts 2) Confirmed evidence of relevant conditions exceeding 10% of the LCB bolt locking devices shall require that the VT-3 examination be expanded by the completion of the next refueling outage to include: For all plants 100% of the accessible UTS, LTS, and FD bolt locking devices Additionally for TMI-1 100% of the accessible lower grid shock pad bolt locking devices, Additionally for CR-3 and DB 100% of the accessible SSHT bolt locking devices	1) The acceptance criteria for the UT of the expansion bolting shall be established as part of the examination technical justification. 2) The specific relevant condition for the expansion of the VT-3 locking devices is evidence of broken or missing bolt locking devices.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Baffle-to-former bolts	All plants	Baseline volumetric (UT) examination of the baffle-to-former bolts. The examination acceptance criteria for the UT of the baffle-to-former bolts shall be established as part of the examination technical justification.	Baffle-to-baffle bolts, Core barrel-to-former bolts	Confirmed unacceptable indications in greater than or equal to 5% (or 43) of the baffle-to-former bolts, provided that none of the unacceptable bolts are on former elevations 3, 4, and 5, or greater than 25% of the bolts on a single former plate, shall require an evaluation of the internal baffle-to-baffle bolts for the purpose of determining whether to examine or replace the internal baffle-to-baffle bolts. The evaluation may include external baffle-to-baffle bolts and core barrel-to-former bolts for the purpose of determining whether to replace them.	N/A
Core Barrel Assembly Baffle plates	All plants	Visual (VT-3) examination. The specific relevant condition is readily detectable cracking in the baffle plates.	a) Former plates b) Core barrel cylinder (including vertical and circumferential seam welds)	a and b. Confirmed cracking in multiple (2 or more) locations in the baffle plates shall require expansion, with continued operation of former plates and the core barrel cylinder justified by evaluation or by replacement by the completion of the next refueling outage.	a and b. N/A
Core Barrel Assembly Locking devices, including locking welds, of baffle-to-former bolts and internal baffle-to-baffle bolts	All plants	Visual (VT-3) examination. The specific relevant condition is missing, non-functional, or removed locking devices.	Locking devices for the external baffle-to-baffle bolts and barrel-to-former bolts	Confirmed relevant conditions in greater than or equal to 1% (or 11) of the baffle-to-former or internal baffle-to-baffle bolt locking devices shall require and evaluation of the external baffle-to-baffle and core barrel-to-former bolt locking devices for the purpose of determining continued operation or replacement.	N/A

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Lower Grid Assembly Alloy X-750 dowel-to-guide block welds	All plants	Initial visual (VT-3) examination	Alloy X-750 dowel locking welds to the upper and lower fuel assembly support pads	Confirmed evidence of relevant conditions at two or more locations shall require that the VT-3 examination be expanded to include the Alloy X-750 dowel locking welds to the upper and lower fuel assembly support pads by the completion of the next refueling outage.	The specific relevant condition is separated or missing locking weld, or missing dowel.
Incore Monitoring Instrumentation (IMI) Guide Tube Assembly IMI guide tube spiders IMI guide tube spider-to-lower grid rid section welds	All plants	Initial visual (VT-3) examination The specific relevant conditions for the IMI guide tube spiders are fractured or missing spider arms. The specific relevant conditions for the IMI spider-to-lower grid rib section welds are separated or missing welds.	a) CRGT spider castings b) Lower fuel assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds	a. Confirmed evidence of relevant conditions for two or more IMI guide tube spider locations shall require that the VT-3 examination be expanded to include 100% of the accessible surfaces at the 4 screw locations (at every 90°) of the CRGT spacer castings by the completion of the next refueling outage. b. Confirmed evidence of relevant conditions at two or more IMI guide tube spider locations or IMI guide tube spider-to-lower grid rib section welds shall require that the VT-3 examination be expanded to include lower fuel assembly support pad items by the completion of the next refueling outage.	a. For the CRGT spacer castings, the specific relevant conditions are fractured spacers or missing screws. b. For the lower fuel assembly support pad items (pads, pad-to-rib section welds, Alloy X-750 dowels, cap screws, and their locking welds), the specific relevant conditions are separated or missing welds, missing support pads, dowels, cap screws and locking welds, or misalignment of the support pads.

Notes:

1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).
2. Expansion to LCB applies if the required Primary examination of LCB has not been performed as scheduled in Table 4-1 (of MRP-227, Rev. 0).

APPENDIX E: NON-PROPRIETARY UCB AND LCB BOLT OCONEE UNIT-SPECIFIC TECHNICAL JUSTIFICATION

Appendix E contains the non-proprietary UCB and LCB Bolt Technical Justification report for the ONS units. The references and numbering for the section, figures, and tables in Appendix E are from the original TJ report.

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Engineering Information Record

Technical Justification for Upper and Lower Core Barrel Bolting Volumetric (Ultrasonic) Examinations at Oconee Nuclear Station

Table of Contents

1.0 PURPOSE

The purpose of this document is to provide a detailed explanation of the examination process, including the theory of the examination technique (as applied to reactor internals inspections), the essential variables of the procedure, other influential parameters important to the overall performance of the examination system and field experience and/or mockup demonstrations supporting the capabilities of the NDE system for volumetric (ultrasonic testing, UT) examinations of the upper and lower core barrel bolting in the Oconee Nuclear Station (ONS) Units 1, 2, and 3 reactor vessel internals.

This document is prepared in accordance with the requirements detailed in Section V, Article 14, Examination System Qualifications of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code [1].

This document is the deliverable for Task 2.1.1.4, "Update of the Beta Testing for UCB and LCB Bolts," in AREVA NP's proposal GC2008246 to Duke Energy (AREVA NP contract number A0002579). This document is the non-proprietary version of AREVA NP document 51-9112756-000.

2.0 DESCRIPTION OF COMPONENT/FLAWS TO BE EXAMINED

2.1 Upper and Lower Core Barrel Bolting Design

The five structural joint locations in the seven operating B&W 177-FA (Fuel Assembly) reactors (Oconee Unit 1 [ONS-1], Oconee Unit 2 [ONS-2], Oconee Unit 3 [ONS-3], Crystal River Unit 3 [CR-3], Arkansas Nuclear One Unit 1 [ANO-1], Davis-Besse [D-B], and Three Mile Island Unit 1 [TMI-1]) are shown in Figure 2-1. In addition to the five locations shown in Figure 2-1, a sixth structural bolting location exists at two of the units (D-B and CR-3). This is a redesigned surveillance specimen holder tube (SSHT), which is attached to the side of the thermal shield. Remnants of the original SSHTs (Figure 2-2 and Figure 2-3) and some of the bolting still exist at each of the ONS units and are also believed to exist at the remaining two units (ANO-1 and TMI-1).

Upper core barrel-to-core support shield bolting (a.k.a. upper core barrel bolts, UCB bolts) fastens the bottom flange of the core support shield to the top of the core barrel cylinder. There are a total of 120 UCB bolt locations. The UCB joint carries the entire weight of the core and majority of the weight of the reactor vessel internals. The typical UCB bolt configurations for ONS-1, ONS-2, and ONS-3 are shown in Figure 2-4. Figure 2-5 shows a video capture image of representative UCB bolts at ONS. Note that the ordering drawing allows that the bolt head design may either be a six-pointed or twelve-pointed head configuration.

The lower core barrel-to-lower grid bolting (a.k.a. lower core barrel bolts, LCB bolts) fastens the bottom of the core barrel cylinder to the lower grid assembly flange. There are a total of 108 LCB bolt locations. The LCB joint carries the weight of the core, but not the weight of the core barrel. The typical LCB configuration for ONS-1, ONS-2, and ONS-3 is given in Figure 2-6 (also shown are the replacement lower thermal shield studs/nuts). Figure 2-7 shows a video capture image of representative LCB bolts at ONS. Note that the ordering drawing allows that the bolt head design may either be a six-pointed or twelve-pointed head configuration.

Of the six structural joints in the B&W-design reactor vessel internals, only the UCB and LCB joints have a core support function and, therefore, represent any potential safety concerns (Reference [2]).

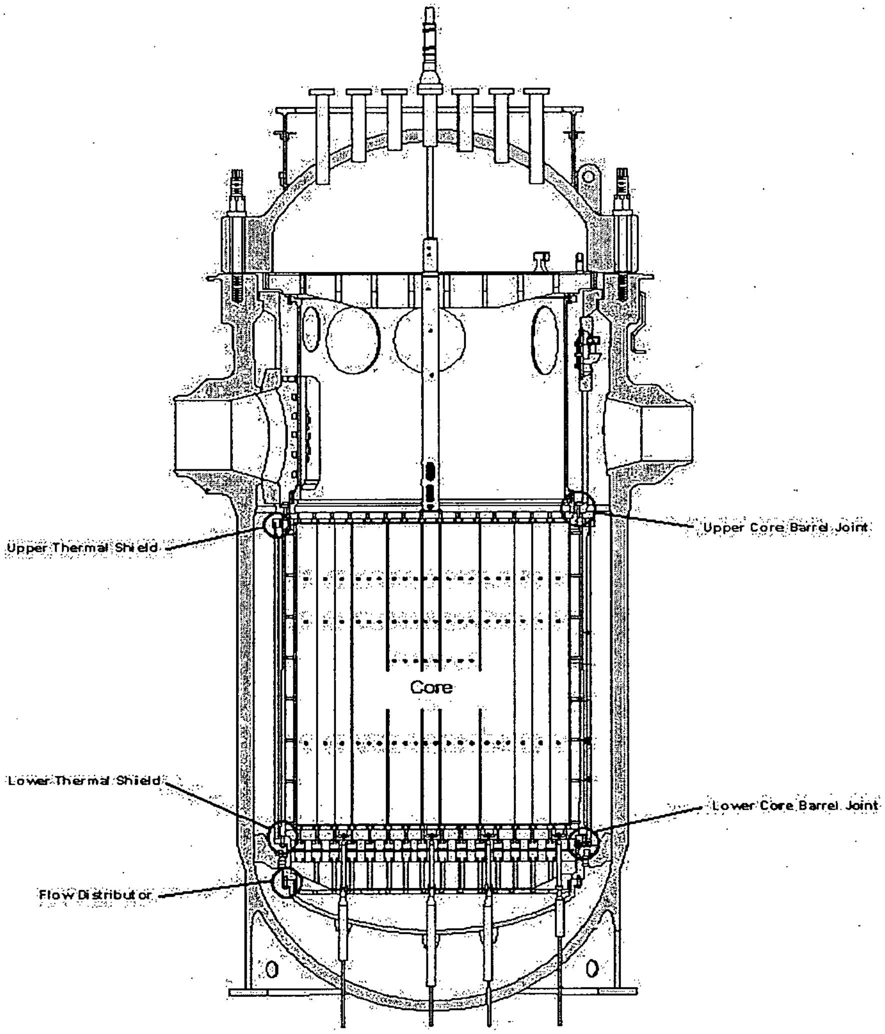

Figure 2-1: Structural Bolting Locations in B&W-Design Reactor Vessel Internals

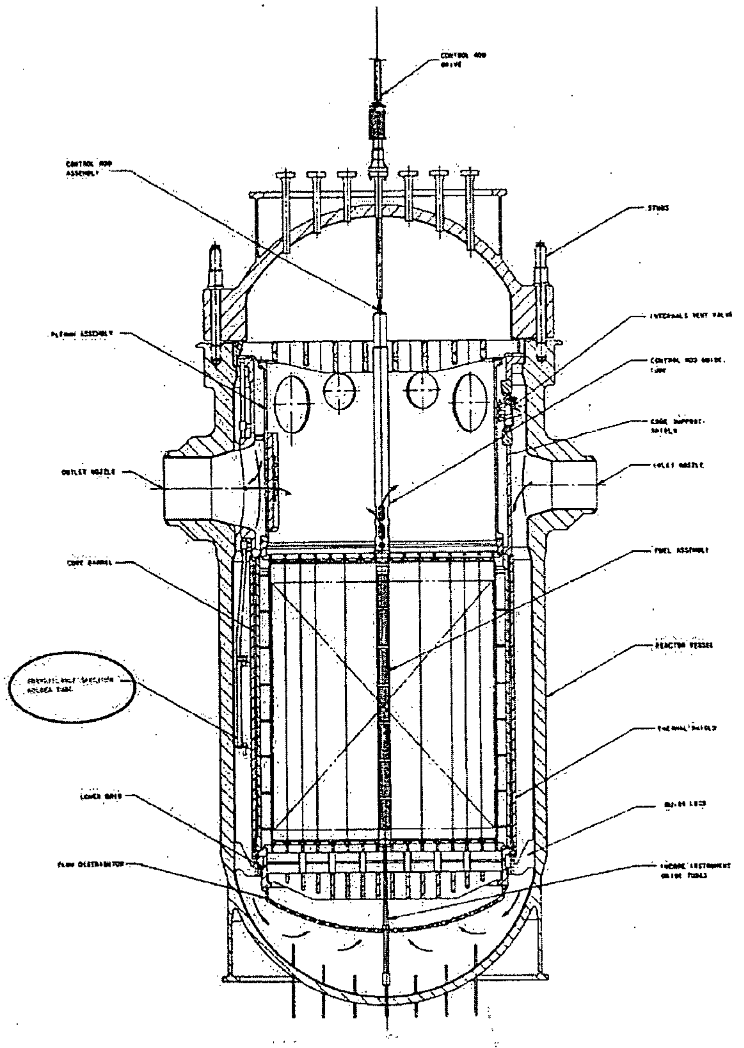

Figure 2-2: Location of Original Surveillance Specimen Holder Tube in B&W-Design Reactor Vessel Internals

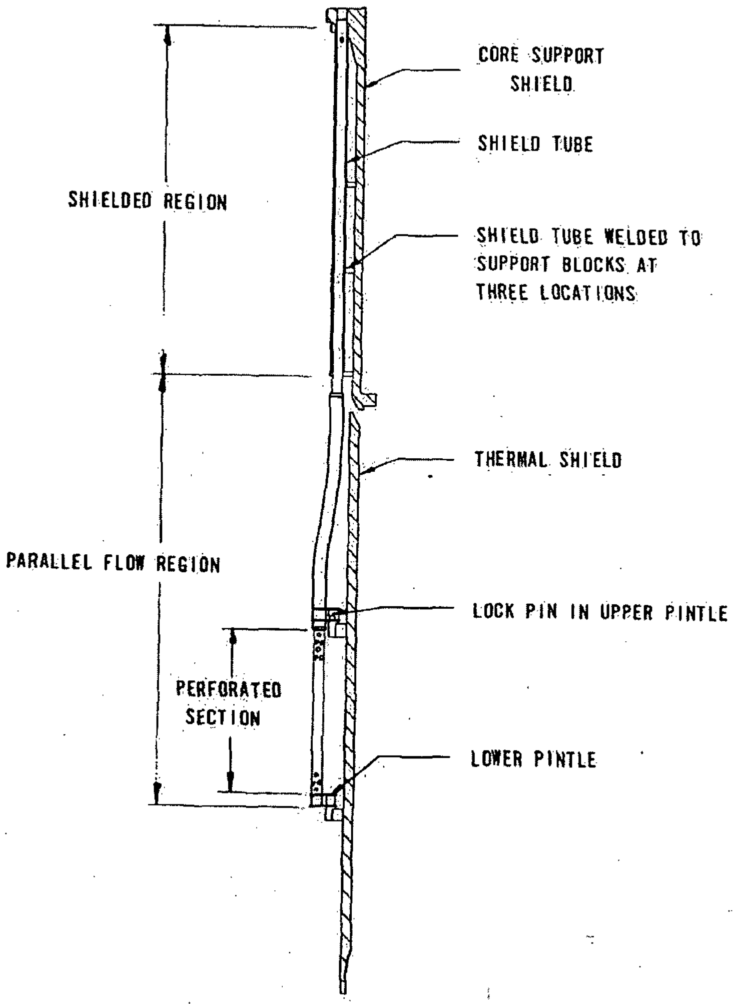

Figure 2-3: Original Surveillance Specimen Holder Tube Design in B&W-Design Reactor Vessel Internals

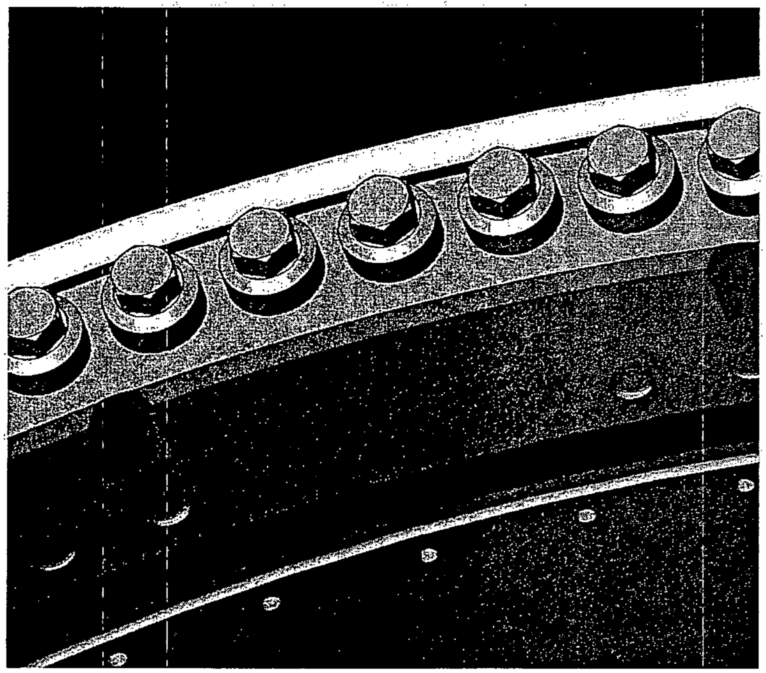

Figure 2-4: Upper Core Barrel Bolt Configuration at ONS units

(Note: locking devices not modeled; both hex bolt head and 12-point bolt head were allowed during the original fabrication)

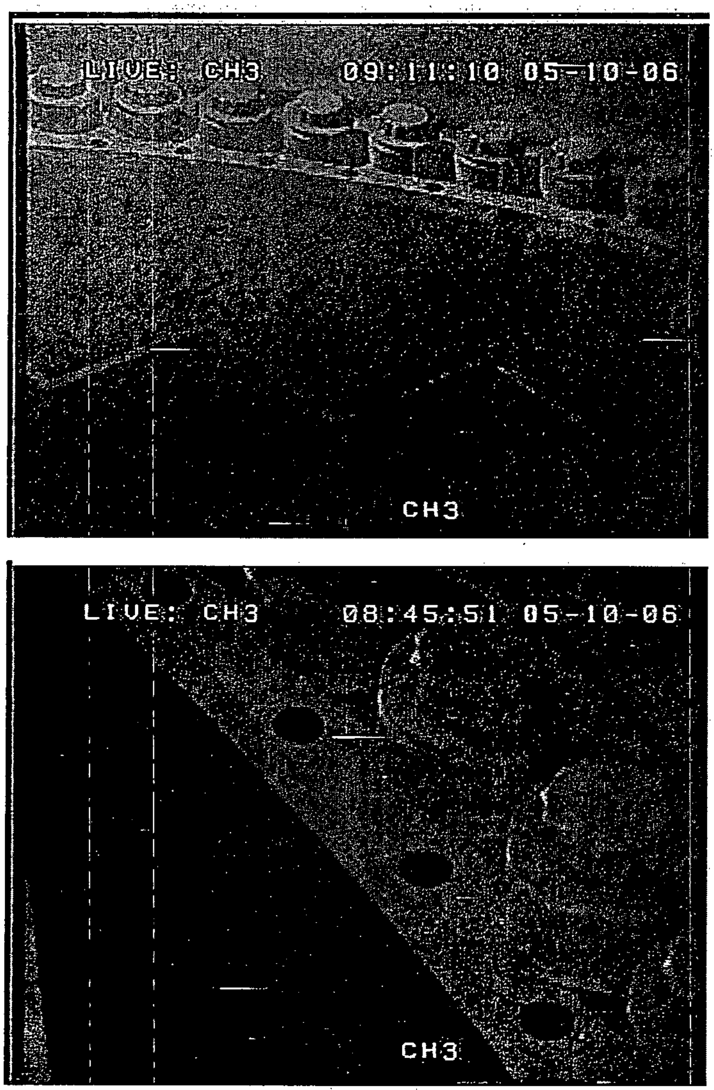

Figure 2-5: Representative Upper Core Barrel Bolt Configuration at ONS

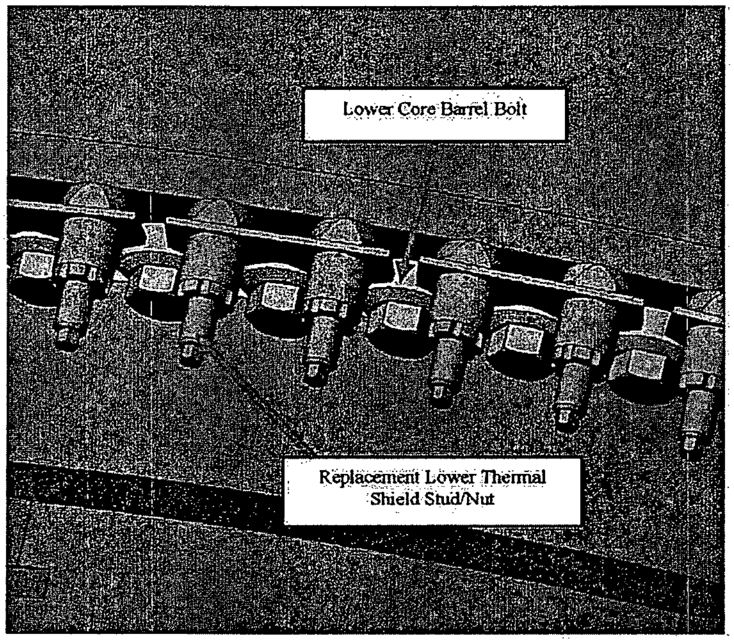

Figure 2-6: Typical Lower Core Barrel Bolt Configuration at ONS Units

(Note: locking devices not modeled; both hex bolt head and 12-point bolt head were allowed during the original fabrication)

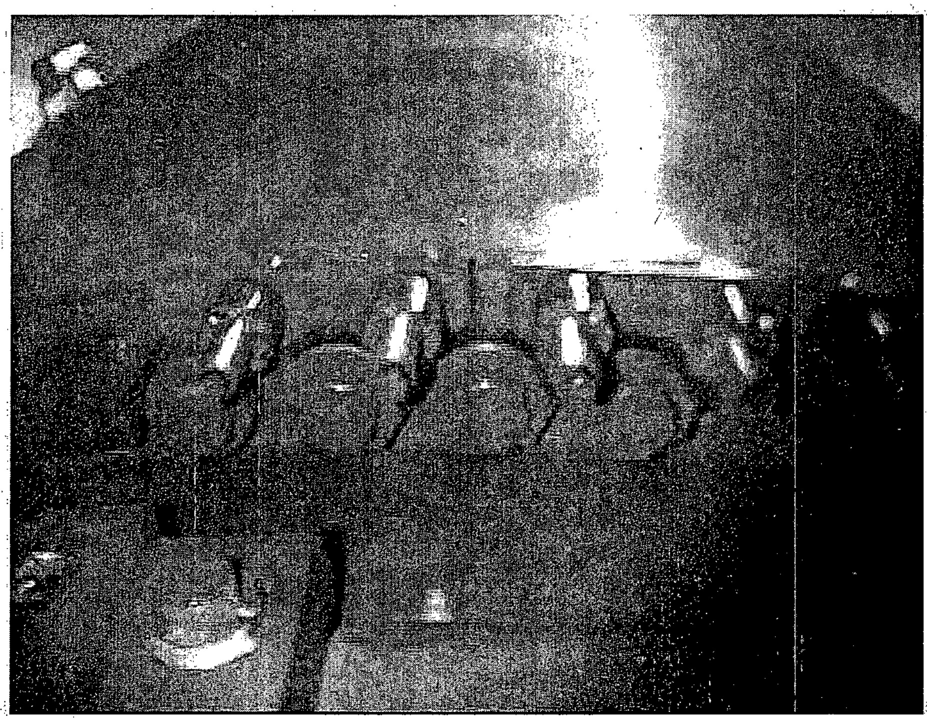

Figure 2-7: Representative Lower Core Barrel Bolt Configuration at ONS

The original Alloy A-286 UCB and LCB bolts were identical (MK256) at six of the operating B&W 177-FA units. TMI-1 is the exception, having used Alloy X-750 for all of the structural bolting locations. All the original Alloy A-286 UCB and LCB bolts were fabricated by the same process. A single heat of Alloy A-286 bar stock material was used for both UCB and LCB bolting locations at each unit; however, several heats of material were used to fabricate the required original Alloy A-286 MK-256 bolts used at the six B&W units. The distribution of bolting material heats for ONS is provided in Reference [2].

At ONS-1 only, the LCB joint also contains 12 extra Type 304 stainless steel bolts (ASTM A 193-65 Grade B8), which are located behind the 12 guide lug shock pads, and therefore they are inaccessible and not visible during inspections.

2.2 Bolting Configuration, Fabrication, and Size

All UCB and LCB bolts in operation at the three ONS units are the originally installed bolting; none have been replaced to date. ONS-1 began commercial operation on April 19, 1973; ONS-2 began commercial operation on November 11, 1973; and ONS-3 began commercial operation on September 5, 1974.

The original MK-256 UCB and LCB bolts are fabricated from ASTM A 453 Grade 660 (a.k.a. Alloy A-286), Condition A material. Condition A material was solution-annealed at 1650°F for two hours and age-hardened at 1325°F for 16 hours. All material is believed to have been 15-20% cold-worked (as inferred by practices described by the bolt fabricator) by the material supplier before bolting fabrication. Material was hot-headed, solution-annealed, and age-hardened with no passivation performed, followed by thread-rolling by the bolt fabricator. The bar stock was ultrasonically examined and 100% of the finished bolt's surface was examined using fluorescent liquid penetrant. Details of the fabrication processing steps are provided in Reference [2]. No record of fabrication flaws was found.

A summary of identified changes to the original bolting joint configurations at the ONS units is as follows:

- Four UCB bolts at ONS-1 were removed for verification and better interpretation of bolt ultrasonic signals in the 1980's (Reference [3]). At the end-of-cycle (EOC)-6 (August 1981), two bolts were removed and shipped to the B&W hot cell laboratory. At EOC-7 (June 1983), two additional bolts were removed and shipped to the B&W hot cell laboratory. Visual, ultrasonic, and fluorescent liquid penetrant examinations were performed in the laboratory on all four bolts. The examinations confirmed the on-site ultrasonic examination result in these four bolts (References [4] and [5]).

- At ONS-1, there is a missing guide block. It was identified as missing during the video inspection conducted at the EOC-7 outage in 1981 (Reference [6]). Further investigations concluded that the guide block had been missing from the CSA since the 1976 SSHT removal effort during the EOC-2 outage. Figure 2-8 shows the typical guide block configuration at ONS.

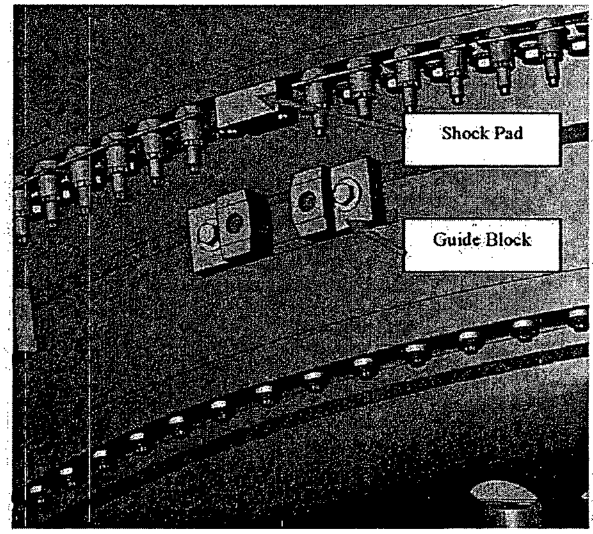

Figure 2-8: Typical Shock Pad Configuration at ONS-1, ONS-2, and ONS-3

- At ONS-2, during the EOC-5 (December 1981) outage, in addition to failed lower thermal shield bolts, the Type 304 stainless steel attachment bolts for a shock pad were found broken (Reference [7]). This shock pad was removed and has never been reinstalled. Figure 2-9 shows a video capture image of the missing shock pad at ONS-2.
- At ONS-3, two UCB bolts have shown unusual ultrasonic signals in prior examination campaigns and have conservatively been assumed to have failed (Reference [2]); however, they remain captured by the locking device and have not been removed. The most recent ultrasonic examination (11/2007) has also identified indeterminate flaw indications at these two locations (Reference [8]).
- At ONS-3, three LCB bolts have shown unusual ultrasonic signals during the 1987 examinations and have conservatively been assumed to have failed (Appendix C of Reference [9]), however, they remain captured by the locking device and have not been removed.

The UCB and LCB bolts were preloaded to various stress levels prior to service. Reference [2] contains the detailed preload information for each of the ONS units. The nominal peak stress for these two bolt locations at each of the ONS units includes the stress concentration factor (SCF) in the head-to-shank area. The yield strength values for both the UCB and LCB bolt heats of material are also provided in Reference [2].

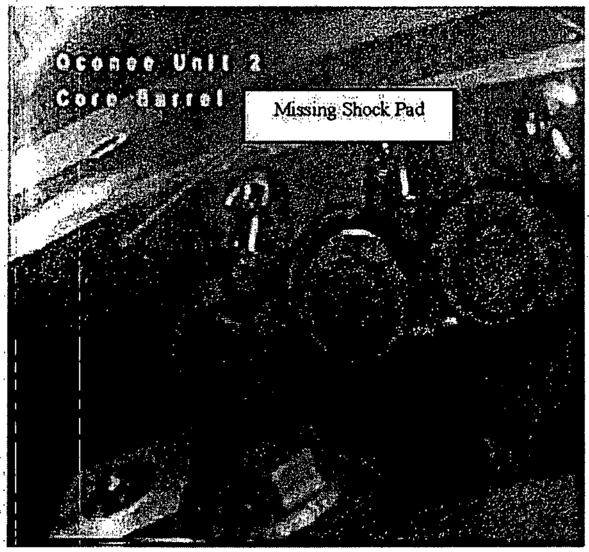

Figure 2-9: Missing Shock Pad Configuration at ONS-2

2.3 Previous Inspections and Flaws of Interest

References [9, 10] provide a summary of past inspection information for the UCB and LCB bolting locations at each of the B&W operating units.

Various rejected bolts were sent to the B&W hot cell for failure analyses. Laboratory examinations of those rejected bolts sent to the hot cell showed that all fractures were primarily caused by intergranular stress corrosion cracking (IGSCC) in or near the fillet between the head and the shank. Numerous branched intergranular cracking also intersected the fracture surfaces. Almost all fractures initiated by intergranular cracking and completed by transgranular fatigue (some bolts showing more transgranular fatigue than others). This led to some belief that corrosion fatigue also might have played a role in the failures.

The crack initiation locations coincided with the peak stress location of the head-to-shank fillet region. Bolt material and fabrication factors also contributed to the IGSCC that included chromium content on the low end of the allowable specification range, heavy cold-working, and hot-heading. The observed fractures show cracking that curves upward slightly from the fillet area into the head region, as shown schematically in Figure 2-10 and in a cross-sectional view in Figure 2-11.

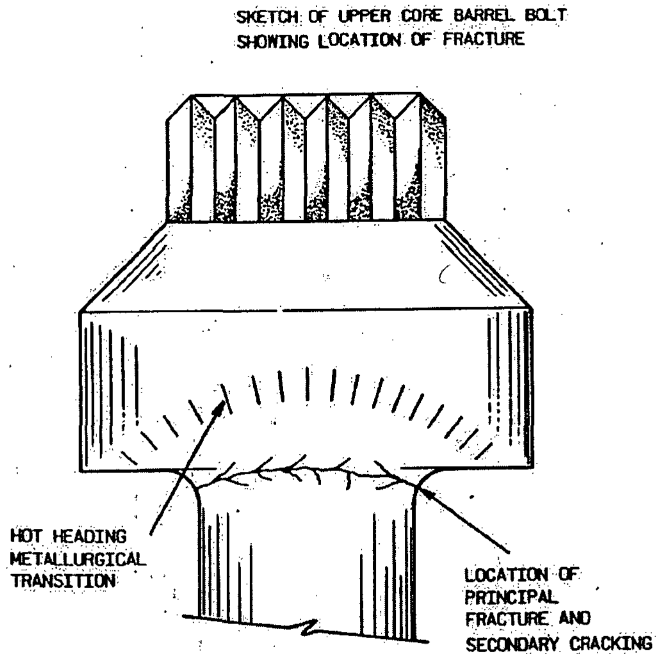

Figure 2-10: Schematic of Typical Fracture Orientation Observed on Failed Structural Bolting at B&W Units

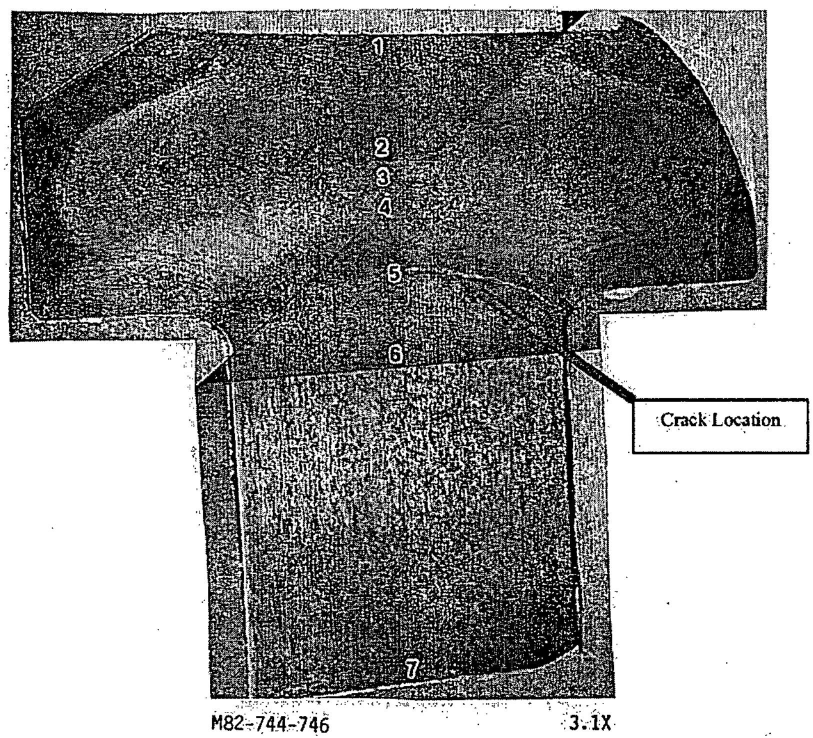

Figure 2-11: Typical Fracture Orientation Observed during Laboratory Examinations on Failed Structural Bolting at B&W Units (Note: Original Magnification was 3.1X) [Reference 7]

2.4 Material Heats Common to Other B&W Operating Units

As summarized above in Section 2.1, several heats of material were used during the manufacture of the ONS UCB and LCB bolts. These heats of material are also common with several other operating B&W-design units.

2.5 Critical Flaw Size and Crack Growth Rate for Alloy A-286 Bolting

Reference [11] summarizes a review of crack growth rates for Alloy A-286 material in a primary water environment. The information gathered indicates that a relatively fast crack growth rate would be anticipated. For such a crack growth rate, failure of an UCB or LCB bolt within an 18-month cycle of operation would easily occur. Therefore, an estimation of crack initiation time is believed to be of more importance and determination of a critical crack size is not relevant for these bolts. Assessment of operability needs to be determined by the number of bolts broken, their location, and the rate at which additional bolts will break before the following inspection period.

2.6 Minimum Number of Bolts Required for Operation

The UCB and LCB joints have large structural margins. This conservatism greatly reduces the likelihood that either the UCB or LCB joint might fail. Acceptance criteria are based on stress limits for threaded structural fasteners given in Subsection NG of the ASME B&PV Code, Reference [12].

Six conditions of removed/rejected bolts have been evaluated for each analyzed core support structures corresponding to all B&W-design operating units.

The first scenario analyzed for each unit represents currently existing conditions. The remaining scenarios are hypothetical.

The analyzed scenarios for the three ONS units, along with the number and locations for the upper and lower core barrel bolts are summarized in Reference [13].

Maximum bolt stresses are calculated and compared with their allowable values per the requirements of ASME Code Subsection NG. The results of the hypothetical cases are given in Reference [13].

3.0 OVERVIEW OF EXAMINATION SYSTEM

The examination system consists of AREVA NP procedure 54-ISI-165-11 Ultrasonic Examination of PWR Internal Bolting [14], an ultrasonic scope, coaxial cable, a round transducer applicable to the bolt size mounted in a spring-loaded fixture, a suitable liquid couplant such as borated, demineralized, or distilled water, and software to acquire an image of the screen. The UCB bolt exam is typically performed by lowering the transducer fixture on a series of poles until the transducer is seated on the bolt head. The LCB exam is typically performed by delivering the transducer to the bolt heads from below using either a crawler or a pogo stick type man-handled pole that reacts off of the deep-end floor. However, this does not limit the potential for the development of alternative transducer delivery system options in the future. The ultrasonic data collection and analysis is the same for the UCB and LCB bolts.

A transducer monitors the back-wall and bolt signals during the acquisition process. A loss of back-wall or multiple indication signals equally-spaced in time are considered recordable indications. This technique basically rejects a bolt if a flaw is recorded without regard to flaw size.

4.0 DESCRIPTION OF INFLUENTIAL PARAMETERS

The anticipated damage mechanism for UCB or LCB bolting is IGSCC, based on past failures and laboratory data.

The essential variables for the bolting exams can be found in AREVA NP procedure 54-ISI-165-11 Ultrasonic Examination of PWR Internal Bolting [14]. The essential variables include transducer size, transducer frequency, cable length, number of intermediate connectors, UT scope, and UT scope settings. Should any of these essential variables change, AREVA NP would be required to perform a new demonstration using these new essential variables.

The examiner shall be qualified at a minimum to Level II (see [15] for definition) in accordance with the AREVA NP written practice. The Level II or III shall be responsible for interpretation of the data and shall accept the results of the examination. The examiner shall have additional documented training in the techniques described within AREVA NP procedure 54-ISI-165-11 Ultrasonic Examination of PWR Internal Bolting.

5.0 DESCRIPTION OF EXAMINATION TECHNIQUES

The UCB and LCB bolts are interrogated for a direct reflection of sound from the flaw or a lack of back-wall response. The technique is not intended to determine flaw size, orientation or percent of remaining ligament in the bolt. The gain level is adjusted up and down to observe the response from the bolt.

This technique was demonstrated at EPRI on bolts representative of the UCB and LCB bolting in the ONS units. EDM notches were machined into some of the bolts. The technique used on these Alloy A-286 bolts detected the anticipated degradation.[16]

When evaluating flaws, there are several points to consider. This technique basically rejects a bolt if a flaw is recorded without regard to the size.

6.0 DESCRIPTION OF EXAMINATION MODELING

Since these examinations have been performed during the past 28 years and have recently been validated by blind performance demonstration at EPRI, no modeling is required.

7.0 DESCRIPTION OF PROCEDURE EXPERIENCE

UT examination of Upper and Lower Core Barrel Bolting has been performed at one time or another at all of the operating B&W units since 1981. These examinations were performed using earlier versions of 54-ISI-165 with the same essential variables as the current revision.

8.0 SUMMARY

An explanation of the UT examination technique for use in examining reactor vessel internals upper and lower core barrel structural bolting at ONS is given in this document. In addition, details of the bolting designs currently in use and prior field experience with the UT examinations are also provided. In conclusion, 54-ISI-165-11 is capable, as demonstrated, of determining the integrity of the Alloy A-286 upper and lower core barrel bolts in the B&W-design reactor vessel internals at ONS.

APPENDIX F: ONS UNIT-SPECIFIC AMENDMENTS TO MRP-227, REV. 0

Appendix F contains the justification for ONS unit-specific amendments to MRP-227, Rev. 0. The references and numbering for the section, figures, and tables in Appendix F are from the original report.

AREVA NP Inc.,

an AREVA and Siemens company

Engineering Information Record

Justification for ONS Unit-Specific Amendments to MRP-227-Rev. 0 Requirements

Table of Contents

1.0 BACKGROUND AND PURPOSE

In December 2008, Electric Power Research Institute (EPRI) issued Rev. 0 of the Material Reliability Program (MRP) MRP-227[1] Inspection and Evaluation (I&E) Guidelines for managing long-term aging of pressurized water reactor (PWR) reactor vessel (RV) internals in the U.S. MRP-227-Rev. 0 provides generic augmented inspection requirements for the current operating fleet of U.S. PWRs. MRP-227-Rev. 0 Section 7.3 categorizes the implementation of the augmented inspections as "Needed" in accordance with Nuclear Energy Institute (NEI) 03-08 Guidelines[2].

The generic augmented inspection requirements for the Babcock and Wilcox (B&W) PWRs are listed in MRP-227-Rev. 0, Tables 4-1, 4-4 and 5-1. MRP-227-Rev. 0 requires implementation of the internals program requirements listed in these tables within 24 months following issuance of MRP-227-A and completion of the examinations within the applicable defined inspection timelines. MRP-227-Rev. 0 recognizes that earlier implementation may be required by plant-specific regulatory commitments. Plants implementing MRP-227 prior to issuance of the U.S. NRC approved MRP-227 (MRP-227-A) version will thus implement the requirements in accordance with the current published version of the guidelines (MRP-227-Rev. 0).

Since the publication of MRP-227-Rev. 0, AREVA NP has performed additional records search for the "Primary" and "Expansion" items for Duke Energy's Oconee B&W PWR units (ONS-1, ONS-2, and ONS-3). The records search is for preparing the ONS unit-specific RV internals inspection plan and aging management program (AMP) that will comply with the MRP-227-Rev. 0 requirements.

Fabrication records for several cast austenitic stainless steel (CASS) items were not available during the screening and evaluation performed for MRP-227-Rev. 0. Therefore, these CASS items were conservatively categorized as "Primary" for thermal aging embrittlement in MRP-227-Rev. 0. The fabrication records found recently have shown that these CASS items are not susceptible to thermal aging embrittlement and therefore can be re-categorized to the "No Additional Measures" category (see Section 2). In addition, the records search also has identified a new Alloy X-750 dowel location at ONS-1 that is not generic to the B&W RV internals and not previously known. Thus, it was not listed in MRP-227-Rev. 0 (see Section 3).

Recently, it has also been discovered that the vent valve disc shaft is completely inaccessible to visual inspection. The VT-3 examination requirement was therefore incorrectly specified in MRP-227-Rev. 0 Table 4-1 and Table 5-1 for the vent valve disc shaft. In addition, it has been found that the aging degradation and VT-3 requirement in MRP-227-Rev. 0 Tables 4-1 and 4-4 for the locking devices of high-strength bolts are imprecise or inconsistent.

In light of the findings, amendments to the MRP-227-Rev. 0 requirements for the affected items are needed for the ONS units in order to meet the intent of MRP-227-Rev. 0. This document provides written technical justification for the needed amendments to the MRP-227-Rev. 0 requirements for several affected RV internals items at ONS units.

It is intended that the technical justification in this document be used to satisfy appropriate procedural processes in order to incorporate the needed amendments into the ONS unit-specific RV internals inspection plan and the AMP. At this time, it has not been decided what procedural process will be used. Regardless of the procedural process to be used, this document provides the written technical justification for the needed amendments to MRP-227-Rev. 0 for the ONS RV internals inspection plan and the AMP.

2.0 ONS CASS AMENDMENTS

Three CASS items in the B&W units are identified in MRP-227-Rev. 0 as "Primary" and "Expansion" due to the potential for thermal aging embrittlement. This section provides the ONS unit-specific CASS amendments and justification.

2.1 MRP-227-Rev. 0 Requirement

The following three CASS items in the ONS RV internals are defined in MRP-227-Rev. 0 as "Primary" or "Expansion" due to thermal aging embrittlement:

Primary

- CSS assembly vent valve disc at ONS-1, ONS-2, and ONS-3
- CSS assembly cast outlet nozzles at ONS-3

Expansion

- Control rod guide tube (CRGT) assembly spacer castings at ONS-1, ONS-2, and ONS-3

The MRP-227-Rev. 0 augmented inspection requirements for the above CASS items are listed in Table 4-1, Table 4-4, and Table 5-1 of MRP-227-Rev. 0. During the preparatory work leading to MRP-227-Rev. 0, the generic and known unit-specific B&W RV internals components were screened for all applicable aging degradation mechanisms. The screening process and results are documented in MRP-189-Rev. 1.[3] For CASS items with unknown chemical compositions, the ferrite contents were conservatively assumed to exceed the screening criterion. Due to a lack of fabrication records, the above three CASS items were conservatively screened as susceptible to thermal aging embrittlement and eventually categorized as MRP-227-Rev. 0 "Primary" or "Expansion".

There is a fourth CASS item in the B&W RV internals that also requires augmented inspection by MRP-227-Rev. 0, the spider castings for the incore instrumentation (IMI) guide tube assembly. The spider castings are categorized as "Primary" in MRP-227-Rev. 0 due to both thermal aging embrittlement (TE) *and irradiation aging embrittlement (IE)*. Therefore, the spider castings cannot be re-categorized by using the screening criteria for thermal aging embrittlement. However, "TE" in MRP-227-Rev. 0 Table 4-1 is no longer applicable to the spider castings.

2.2 Basis for CASS Amendments

Since the publication of MRP-227-Rev. 0, AREVA NP has performed a records search for the above CASS items. The records search results are documented in References 4 and 5. The records were not available during development of MRP-227 and these CASS components were therefore considered to be susceptible to thermal aging embrittlement when MRP-227-Rev. 0 was being developed.

The records search found the applicable CASS heat numbers and certified material test reports (CMTRs), which allows screening in accordance with the MRP-175[6] screening criteria. The MRP-175 screening criteria are identical to those recommended by Section XI.M12 of NUREG-1801-Rev. 1[7] for thermal aging embrittlement of CASS.

The CASS ferrite content is estimated with Hull's equivalent factors in NUREG/CR-4513, Rev. 1[8]. Because the nitrogen content is not listed in the CMTRs, nitrogen is assumed to be 0.04% as recommended by NUREG/CR-4513, Rev 1. The technical background for the CASS screening criteria is detailed in MRP-175, NUREG-1801-Rev. 1, and NUREG/CR-4513, Rev. 1.

2.3 Amendments

2.3.1 CSS Vent Valve Disc Amendment

Each ONS unit has eight vent valves installed in the CSS cylinder. Each vent valve contains a hinged vent valve disc and a vent valve body. Each vent valve can be remotely handled as a unit for removal or installation. The hinged disc includes a device for remote testing and verification of proper disc function. The vent valves are swing check valves meant to relieve pressure in the interior of the CSS assembly during a large break LOCA, preventing backpressure from reversing coolant flow through the core. During all other operating conditions the vent valves are closed and prevent bypass reactor coolant flow from the reactor vessel annulus to the core outlet region.

The vent valve discs are fabricated from Grade CF8 of ASTM A 351-65[9]. The heat numbers and CMTRs of the vent valve discs have been retrieved for all 24 original vent valves supplied to the ONS units. The heat numbers and ferrite contents based on CMTR chemical composition with Hull's equivalent factors are documented in Ref. 4. The ferrite contents for all original vent valve discs are below the 20% ferrite screening criterion for Grade CF8 material.

The ONS units were also supplied with spare vent valves and some spares were installed (i.e., replaced the original vent valves). Some vent valves were exchanged between the ONS units. Therefore, Duke Energy recently identified the S/N and heat number of currently installed vent valves at ONS-1 (Fall 2009) and ONS-2 (Spring 2010) using a remote underwater video camera. Their identification and ferrite content are documented in Ref. 5. All currently installed vent valve discs at ONS-1 and ONS-2 are below the 20% ferrite screening criterion for Grade CF8. A similar identification for the ONS-3 installed vent valve discs is planned for Fall 2010. Based on the records search and ONS-1 and ONS-2 identification results, a similar finding is expected for ONS-3.

If these records had been available during initial screening, the vent valve discs would have been categorized as Category "A" (below screening values) and no augmented inspection would have been required by MRP-227-Rev. 0. Therefore, the vent valve discs at ONS are re-categorized from the "Primary" in MRP-227-Rev. 0 to "No Additional Measures", and the augmented inspection in MRP-227-Rev. 0, Table 4-1 is not required.

It should be noted that this ONS vent valve disc amendment to the MRP-227-Rev.0 requirement does not affect the following requirements:

- Note 1 to MRP-227-Rev. 0, Table 4-1, vent valve exercising and visual inspection.
- MRP-227-Rev. 0, Table 4-1 inspection requirement for the vent valve top and bottom retaining rings, and disc shaft (hinge pin).
- ASME Section XI, Category B-N-3 VT-3 examination of the vent valves (see note below).

  Note: The ASME Section XI, Category B-N-3 inservice inspection (ISI) scope is defined by the owners (utilities) of the B&W units.

2.3.2 CSS Outlet Nozzle (ONS-3 only) Amendment

Each of the three ONS units has two outlet nozzles in the CSS cylinder. The outlet nozzles are used to direct the RCS flow out of the core to the two reactor vessel outlet nozzles. Only the CSS outlet nozzles at ONS-3 are fabricated from Grade CF8 of ASTM A 351-65. The CSS outlet nozzles at ONS-1 and ONS-2 are fabricated from wrought stainless steel and do not have a thermal aging embrittlement concern.

The heat numbers and CMTRs have been found for the two ONS-3 CSS outlet nozzles. The heat numbers and ferrite contents based on CMTR chemical composition with Hull's equivalent factors are documented in Ref. 4. The ferrite contents of both outlet nozzles are below the 20% ferrite screening criterion for Grade CF8 material.

If these records had been available during initial screening, the ONS-3 CSS outlet nozzles would have been categorized as "A" and no augmented inspection would have been required by MRP-227-Rev. 0. Therefore, the ONS-3 CSS outlet nozzles are re-categorized from the "Primary" in MRP-227-Rev. 0 to "No Additional Measures", and the augmented inspection in MRP-227-Rev. 0, Table 4-1 is not required.

It should be noted that this CSS outlet nozzle (ONS-3 only) amendment to MRP-227-Rev.0 requirement does not affect the following requirement:

- ASME Section XI, Category B-N-3 VT-3 examinations of the outlet nozzles (see note below)

  Note: The ASME Section XI, Category B-N-3 inservice inspection (ISI) scope is defined by the owners (utilities) of the B&W units.

2.3.3 CRGT Spacer Casting Amendment

Each of the three ONS units has 69 CRGT assemblies welded to the plenum cover plate and bolted to the upper grid. Inside each CRGT assembly is a brazement subassembly consisting of ten parallel horizontal spacer castings. Each spacer casting is brazed to 12 perforated vertical rod guide tubes and 4 pairs of vertical rod guide sectors, also called "C-tubes." At the ten spacer elevations, the CRGT housing pipe is drilled at four equally spaced circumferential locations to accommodate four cap screws that hold each spacer casting in place. The 10 spacer castings provide structural support to the 12 perforated vertical rod guide tubes and 4 pairs of vertical rod guide sectors within each CRGT assembly and keep them aligned with the guide tubes in the fuel assembly below.

The spacer castings are fabricated from Grade CF3M of ASTM A 351-65. The heat numbers and CMTRs have been found for most spacer castings at ONS units. A majority of these spacer castings exceed the 14% screening criterion for Grade CF3M. Therefore the CRGT spacer castings, as a group, cannot be re-categorized to the "No Additional Measures" category from their current MRP-227-Rev. 0 "Expansion" category.

The CRGT spacer castings are linked in MRP-227-Rev. 0 to the following "Primary" items:

- CSS vent valve discs,
- CSS outlet nozzles (ONS-3 only), and
- IMI guide tube spider castings.

The amendments in this document re-categorize the CSS vent valve discs and CSS outlet nozzles to "No Additional Measures", and therefore removed their link to the CRGT spacer casting. Inspecting the CSS vent valve discs and CSS outlet nozzles will not meet the "Expansion" link intent because these two items are not susceptible for thermal aging embrittlement based on the screening results.

The amendments leave the IMI guide tube spider castings as the only "Primary" link for the CRGT spacer castings at ONS units. However, the spider castings alone do not meet the intent of MRP-227-Rev. 0 to provide detection of the aging effect before it affects the CRGT spacer castings due to the following considerations:

- The CRGT spacer castings are fabricated from Grade CF3M (containing molybdenum) while the spider castings are fabricated from Grade CF8 material.
- The spider ferrite content per Hull's equivalent factors (Ref. 5) is below the 20% criterion for Grade CF8 for thermal aging embrittlement; the potential for irradiation embrittlement is what prevents the spider castings from being re-categorized to "No Additional Measures".

- The spacer castings are exposed to the core exit reactor coolant temperature while the spider castings are exposed to cold leg reactor coolant temperature.

Therefore, the CRGT spacer castings at ONS units are re-categorized from "Expansion" in MRP-227-Rev. 0 to "Primary" with the following requirements:

- Effect (Mechanism): Cracking (TE), including detection of fractured spacers or missing screws.

  Note: The effect is unchanged from MRP-227-Rev. 0, Table 4-4.

- Expansion Link: None.

  Note: The CRGT spacer casting expansion link to the spider castings in MRP-227-Rev. 0, Tables 4-1 and 5-1 is removed by this amendment.

- Examination Method/Frequency: Visual (VT-3) examination during the next 10-year ISI; subsequent examinations on the 10-year ISI interval.

  Note: VT-3 method is unchanged from MRP-227-Rev. 0, Table 4-4. Frequency is identical to MRP-227-Rev. 0, Table 4-1 for the CSS vent valve discs and CSS outlet nozzles.

- Examination Coverage: 100% of accessible surfaces at the 4 screw locations (at every 90°) (limited accessibility).

  Note: The coverage is unchanged from MRP-227-Rev. 0, Table 4-4.

3.0 PLENUM COVER WELDMENT RIB PAD ITEMS AT ONS-1 ONLY

During the records search, a feature unique to ONS-1 was identified for the plenum cover weldment rib pads. Each of the 32 plenum cover rib pads at ONS-1 is fastened to the plenum cover ribs with two Type 304 screws and one Alloy X-750 dowel. At the other ONS units, the plenum cover rib pads are welded to the plenum cover weldment ribs.

Since this unique feature at ONS-1 was unknown during the preparatory work leading to MRP-227-Rev. 0, the Alloy X-750 dowel and Type 304 screws at this location were not screened for all applicable aging degradation mechanisms, nor evaluated for inclusion in MRP-227-Rev. 0. This section provides an evaluation using the MRP criteria and process to determine if additional augmented inspection is required for this location at ONS-1.

3.1 Screening

If this feature at ONS-1 had been known, it would have been screened for all applicable aging degradation mechanisms. Therefore, this location is screened below with the same screening process and criteria that has been used for other locations and documented in MRP-189-Rev. 1.[3] The fabrication records related to this location are summarized in Section 9 of Ref. 5.

3.1.1 Alloy X-750 Dowel

The environmental parameters (temperature and neutron exposure) are the same as for the plenum cover weldment rib pads (P.1.1) in MRP-189 Rev. 1 Table 3-2. The Alloy X-750 dowel body can carry the possible shear load introduced by the relative thermal movement of the closure head and plenum cover. However, the dowels are under compressive stress at the surfaces exposed to the primary coolant. Therefore, SCC of the Alloy X-750 dowel is not a concern. The dowel locking weld is not a multiple pass weld. Applying the material and parameters against the MRP-175 criteria produces the following screening results:

- Stress Corrosion Cracking (SCC), below screening, Category "A"
- Irradiation-Assisted SCC (IASCC), below screening, Category "A"
- Irradiation Stress Relaxation and Creep (ISR/IC), below screening, Category "A"
- Wear, below screening, Category "A"
- Fatigue, below screening, Category "A"
- Thermal Aging Embrittlement, below screening, Category "A"
- Irradiation Aging Embrittlement, below screening, Category "A"
- Void Swelling, below screening, Category "A"

Therefore, the overall screening category for the Alloy X-750 dowel is Category "A" and no augmented inspection is required.

3.1.2 Alloy X-750 Dowel Locking Weld

The environmental parameters (temperature and neutron exposure) are the same as for the Alloy X-750 dowels. As documented in Ref. 5, the Alloy X-750 dowels for the ONS-1 plenum cover weldment rib pads have the same diameter as for the Alloy X-750 dowels used for the upper grid and lower grid fuel assembly support pads at ONS-1. The locking weld for the Alloy X-750 dowels for the ONS-1 plenum cover rib pads is identical to the locking weld for the Alloy X-750 dowels for the upper and lower fuel assembly support pads at ONS-1 in the following ways:

- Same Alloy 69 (INCO 69) weld metal
- Same weld process
- Same heat of weld wire
- Same weld size

Applying the material and environmental parameters against the MRP-175 criteria produces the following screening results:

- Stress Corrosion Cracking (SCC), above screening, Category "Not A"
- Irradiation-Assisted SCC (IASCC), below screening, Category "A"
- Irradiation Stress Relaxation and Creep (ISR/IC), below screening, Category "A"
- Wear, below screening, Category "A"
- Fatigue, below screening, Category "A"
- Thermal Aging Embrittlement, below screening, Category "A"
- Irradiation Aging Embrittlement, below screening, Category "A"
- Void Swelling, below screening, Category "A"

Because this locking weld used nickel-base material, it is potentially susceptible to primary water SCC (PWSCC) similar to the nickel-base locking welds used in the upper and lower grid fuel assembly support pads, and lower grid guide blocks. Therefore, the overall screening category for the Alloy X-750 dowel locking weld is Category "Not A" due to PWSCC. This locking weld is further assessed in Section 3.2 of this document.

3.1.3 Type 304 Screws

The environmental parameters (temperature and neutron exposure) are the same as for the plenum cover weldment rib pads (P.1.1) in MRP-189 Rev. 1 Table 3-2. The screws are fabricated from Type 304 austenitic stainless. The screw locking weld is not a multiple pass weld. Applying the material and parameters against the MRP-175 criteria produces the following screening results:

- Stress Corrosion Cracking (SCC), below screening, Category "A"
- Irradiation-Assisted SCC (IASCC), below screening, Category "A"
- Irradiation Stress Relaxation and Creep (ISR/IC), below screening, Category "A"
- Wear, below screening, Category "A"
- Fatigue, below screening, Category "A"
- Thermal Aging Embrittlement, below screening, Category "A"
- Irradiation Aging Embrittlement, below screening, Category "A"
- Void Swelling, below screening, Category "A"

Therefore, the overall screening category for the Type 304 screws is Category "A" and no augmented inspection is required.

3.1.4 Type 304 Screw Locking Weld

The environmental parameters (temperature and neutron exposure) are the same as for the plenum cover weldment rib pads (P.1.1) in MRP-189 Rev. 1 Table 3-2. The locking weld used Type 308 austenitic stainless weld metal. The locking weld directly welds the screw head to the rib pads without using a locking bar or a locking cup. Applying the material and parameters against the MRP-175 criteria produces the following screening results:

- Stress Corrosion Cracking (SCC), below screening, Category "A"
- Irradiation-Assisted SCC (IASCC), below screening, Category "A"
- Irradiation Stress Relaxation and Creep (ISR/IC), below screening, Category "A"
- Wear, below screening, Category "A"
- Fatigue, below screening, Category "A"
- Thermal Aging Embrittlement, below screening, Category "A"
- Irradiation Aging Embrittlement, below screening, Category "A"
- Void Swelling, below screening, Category "A"

Therefore, the overall screening category for the Type 304 screw locking weld is Category "A" and no augmented inspection is required.

3.2 Assessment of Alloy X-750 Dowel Locking Weld

This nickel-based Alloy 69 (INCO 69) locking weld is susceptible to PWSCC. Each of the 32 plenum cover weldment rib pads at ONS-1 is fastened to the plenum cover weldment ribs with two Type 304 screws and one Alloy X-750 dowel. The dowel is located in the center of each rib pad while the two screws are at the two ends of each rib pad.

The plenum cover weldment rib pads are part of the RV internals holddown stack, which provides clamping force to stabilize and significantly restrict the rigid body pendulum motion of the core support and plenum assemblies. In other words, the clamping action prevents rigid body rotation at the interface area. The clamping action does not have a direct core support safety function. Loss of clamping would undoubtedly lead to core barrel motions that would eventually lead to a reactor shut down. Due to the wear concern, the plenum cover weldment rib pads are categorized as MRP-227-Rev. 0 "Primary".

The dowel locking weld serves as a loose part prevention device. The dowel locking weld will retain the dowel if any portion of the weld is in place. The only loading on the dowel is shear, which is in the dowel body and not near the locking weld. In addition, the Alloy X-750 dowels on the rib pads are completely covered by the reactor vessel head flange during plant operation and therefore cannot back out even if the dowel locking weld is completed cracked. The Alloy X-750 dowels and Type 304 screws are Category "A" components, and are not affected by any aging degradation mechanisms. Therefore, cracking of the Alloy X-750 dowel locking weld will not affect the functionality of the plenum cover weldment ribs.

Based on the above functionality assessment, the locking weld for the ONS-1 plenum cover weldment rib pad Alloy X-750 dowel is categorized as "No Additional Measures". Therefore, no additional augmented inspection is required for this location.

Note: the following requirements for the plenum cover rib pads are not affected by the screening in Section 3.1 and assessment in Section 3.2 of this document.

- MRP-227-Rev. 0, Table 4-1, plenum cover weldment rib pads (Primary).

- ASME Section XI, Category B-N-3 VT-3 examination of the plenum cover weldment rib pads (see note below).

  Note: The ASME Section XI, Category B-N-3 inservice inspection (ISI) scope is defined by the owners (utilities) of the B&W units.

4.0 CSS VENT VALVE DISC SHAFT (HINGE PIN)

Each of the three ONS units has eight vent valves installed in the CSS cylinder. Each vent valve consists of a hinged disc, a valve body with sealing surfaces, a split-retaining ring, and jackscrews that hold the retaining rings in place to support the perimeter of the valve assembly.

4.1 MRP-227-Rev. 0 Requirement

The CSS vent valve disc shaft (also called hinge pin) is listed in MRP-227-Rev. 0 as "Primary" due to thermal aging embrittlement. The disc shaft is made of ASTM A 276, Type 431 martensitic stainless steel. The vent valve disc shaft inspection requirements in MRP-227 (Rev. 0) Table 4-1 and Table 5-1 are the following:

- Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval. Examination coverage is 100% of accessible surfaces.

4.2 Accessibility

The hinge assembly consists of a vent valve disc shaft, four flanged shaft journals (bushings) and two journal receptacles. The vent valve disc contains an integral exercise lug for remote exercising. After the vent valve disc was hinged to the vent valve body, the ends of the journal receptacles were covered with welded cover plates.[10] As illustrated in Figure 4-1, the vent valve disc shaft is completely enclosed by the integral exercise lug, four flanged shaft journals, two valve disc journal receptacles and their cover plates. Therefore, the vent valve disc shaft is inaccessible for visual inspection without disassembling the vent valve.

4.3 Amendment

The vent valve disc shaft is completely enclosed and inaccessible to visual inspection. The visual (VT-3) examination requirement was incorrectly specified in MRP-227-Rev. 0 Table 4-1 and Table 5-1 for this item. Therefore, the MRP-227-Rev. 0 Table 4-1 requirement for the vent valve disc shaft is revised to the following for the ONS units:

- Primary: The vent valve disc shaft remains "Primary" unless justified by evaluation.
- Effect (Mechanism): Cracking (TE), including the detection of surface irregularities, such as damaged, fractured material, or missing items.
- Expansion Link: None.
- Examination Method/Frequency: No examination requirements, justify by evaluation or by replacement
- Examination Coverage: Inaccessible

Note: The above changes do not apply to the CSS vent valve top and bottom retaining ring.

In addition, the "Visual (VT-3) examination" under "Examination Acceptance Criteria" in MRP-227-Rev. 0 Table 5-1 for the vent valve disc shaft is revised to "Inaccessible, justify by evaluation or by replacement."

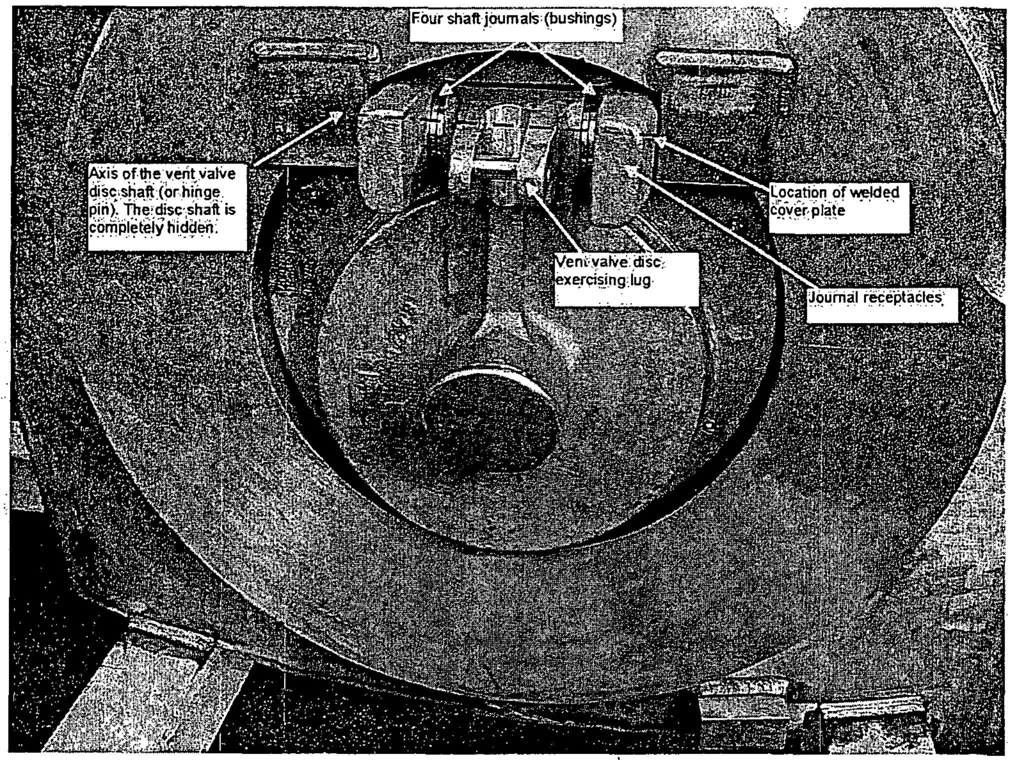

Figure 4-1: Typical CSS vent valve – outside view

The vent valve disc shaft (also called hinge pin) is completely hidden and inaccessible without disassembling the vent valve.

5.0 LOCKING DEVICES FOR HIGH-STRENGTH BOLTING

This section clarifies the description aging degradation effect and examination method in MRP-227-Rev. 0 Tables 4-1 and 4-4 for the high-strength bolts and their locking devices. The high-strength bolts inside the RV internals of ONS units are the following bolts:

- MRP-227-Rev. 0 Table 4-1, Primary
  - Upper Core Barrel (UCB) Bolts, including locking devices and locking welds
  - Lower Core Barrel (LCB) Bolts, including locking devices and locking welds
- MRP-227-Rev. 0 Table 4-4, Expansion
  - Flow Distributor (FD) Bolts, including locking devices and locking welds
  - Upper Thermal Shield (UTS) Bolts, including locking devices and locking welds
  - Lower Thermal Shield (LTS) Bolts, including locking devices and locking welds

5.1 Clarification to MRP-227-Rev. 0 Table 4-1

MRP-227-Rev. 0 Table 4-1 (Primary Items) requires UT inspection of the UCB and LCB bolts, and VT-3 inspection of their locking devices. Table 4-1 lists SCC (stress corrosion cracking) under "Effect (Mechanism)" for the UCB and LCB bolts and their locking devices.

The UCB and LCB bolts are susceptible to SCC; the bolt locking devices are not susceptible to SCC.[3] However, damaged locking devices are evidence of potentially failed UCB and LCB bolts due to SCC. Due to this consideration, MRP-227-Rev. 0 also requires VT-3 inspection of the locking devices.

Therefore, the following clarification is provided to the UCB and LCB bolts in MRP-227-Rev. 0 Table 4-1:

- The aging effect (mechanism) for bolt is cracking (SCC).
- The aging effect (mechanism) for bolt locking devices is loss of material, damaged, distorted, or missing locking devices (wear or fatigue damage by failed bolt).

5.2 Clarification to MRP-227-Rev. 0 Table 4-4

In MRP-227-Rev. 0 Table 4-4, the listing for the "Expansion" bolts linked to UCB and LCB bolts does not explicitly list the bolt locking devices or the VT-3 visual inspection of the locking devices. This error originated from MRP-231-Rev. 0[11] Table 3-10, which was the underlying source for MRP-227-Rev. 0 Table 4-4. The omission of the locking devices and VT-3 requirement in MRP-227-Rev. 0 Table 4-4 was inadvertent as MRP-227 Rev. 0 Table 5-1 clearly lists the VT-3 requirement for the locking devices of the "Expansion" bolts linked to the "Primary" UCB and LCB bolts.

Therefore, the following clarification is provided to the "Expansion" UTS, LTS, and FD bolts in MRP-227-Rev. 0 Table 4-4 that are linked to the "Primary" UCB and LCB bolts:

- Each item is revised to include: "and their locking devices."
- The aging effect (mechanism) for bolt is cracking (SCC).
- The aging effect (mechanism) for bolt locking devices is loss of material, damaged, distorted, or missing locking devices (wear or fatigue damage by failed bolt).
- Each primary link is revised to include: "and their locking devices."
- The examination method for bolts is volumetric examination (UT).

- The examination method for bolt locking devices is visual (VT-3) examination.
- The examination coverage is 100% of accessible bolts and locking devices.

6.0 SUMMARY

Amendments to MRP-227-Rev. 0 requirements for the ONS units:

1. MRP-227-Rev. 0 Primary: CSS assembly vent valve discs
   - The vent valve discs at ONS-1 and ONS-2 are re-categorized to "No Additional Measures". No augmented inspection is required.

     Note: All currently installed vent valve discs at ONS-1 and ONS-2 have been verified to be below the 20% ferrite screening criterion for Grade CF8. A similar verification for the ONS-3 installed vent valve discs is planned for Fall 2010. This amendment will apply to the ONS-3 installed vent valve discs after the ferrite is confirmed to be below the screening criteria.
2. MRP-227-Rev. 0 Primary: CSS assembly cast outlet nozzles (ONS-3 only)
   - The outlet nozzles at ONS-3 are re-categorized to "No Additional Measures". No augmented inspection is required.
3. MRP-227-Rev. 0 Expansion: CRGT spacer castings
   - The CRGT spacer castings at ONS-1, ONS-2, and ONS-3 are re-categorized to "Primary"
     - There is no "Expansion" link.
     - Examination Method/Frequency: Visual (VT-3) examination during the next 10-year ISI; subsequent examinations on the 10-year ISI interval.
     - Effect (Mechanism) and examination coverage are unchanged from MRP-227-Rev 0 Table 4-4.

   Notes:
     - The CRGT spacer casting expansion link to the IMI guide tube spider castings in MRP-227-Rev. 0, Tables 4-1 and 5-1 is removed by this amendment.
     - "TE" in MRP-227-Rev. 0 Table 4-1 is no longer applicable to the IMI guide tube spider castings. However, the spider castings remain as "Primary" due to "IE" and are linked to the "Expansion" of lower grid fuel assembly support pad items.
4. MRP-227-Rev. 0 Primary: CSS Vent Valve Disc Shaft (Hinge Pin)
   - The examination method/frequency and examination coverage in MRP-227-Rev. 0 Table 4-1 and Table 5-1 for CSS vent valve disc shaft (hinge pin) at ONS-1, ONS-2, and ONS-3 are revised to the following:
     - Examination Method/Frequency in Table 4-1: No examination requirements, justify by evaluation or by replacement
     - Examination Coverage in Table 4-1: Inaccessible
     - Examination Acceptance Criteria in Table 5-1: Inaccessible, justify by evaluation or by replacement
5. MRP-227-Rev. 0 Table 4-1, UCB and LCB Bolts
   - The aging effect (mechanism) for bolt is cracking (SCC).
   - The aging effect (mechanism) for bolt locking devices is loss of material, damaged, distorted, or missing locking devices (wear or fatigue damage by failed bolt).
6. MRP-227-Rev. 0 Table 4-4, "Expansion" UTS, LTS, and FD Bolts Linked to UCB and LCB bolts.

- Each item is revised to include: "and their locking devices."
- The aging effect (mechanism) for bolt is cracking (SCC).
- The aging effect (mechanism) for bolt locking devices is loss of material, damaged, distorted, or missing locking devices (wear or fatigue damage by failed bolt).
- Each primary link is revised to include: "and their locking devices."
- The examination method for bolts is volumetric examination (UT).
- The examination method for bolt locking devices is visual (VT-3) examination.
- The examination coverage is 100% of accessible bolts and locking devices.

ONS-1 Plenum Cover Weldment Rib Pad Items

During the records search, a feature unique to ONS-1 was identified for the plenum cover weldment rib pads. The Alloy X-750 dowels, Type 304 screws, and their locking welds were unknown and were not screened for aging degradation mechanisms, nor evaluated for inclusion in MRP-227-Rev. 0. Using the MRP screening criteria and process, these items are categorized as Category "A" or "No Additional Measures". Therefore, no additional augmented inspection is required for this location.

The MRP-227-Rev. 0 report is currently being reviewed by the U.S. NRC for a safety evaluation report (SER). If necessary, the ONS amendments in this document will be updated for compliance with the U.S. NRC approved MRP-227 (MRP-227-A) after the SER is granted.

ATTACHMENT 2

REGULATORY COMMITMENTS

The following table identifies the regulatory commitments in this document. Any other statements in this submittal represent intended or planned actions. They are provided for information purposes and are not considered to be regulatory commitments.

Commitment	Due Date
After approval of MRP-227, Duke Energy will review and, if needed, revise the ONS RV Internals inspection plan.	90 days after issuance of MRP-227-A
The ONS ISI program will be updated to include the items from the NRC-approved ONS RV Internals Inspection Plan as augmented inspections and the inspection results will be submitted.	90 day outage report