ATTACHMENT 1

INSPECTION PLAN FOR THE OCONEE NUCLEAR STATION UNITS 1, 2, and 3
REACTOR VESSEL INTERNALS (ANP-2951, REVISION 2)

ANP-2951
Revision 002
March 2012

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

AMP/Application to Implement MRP-227-A

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Nature of Changes

Revision	Date	Changes
000	August 2010	Initial Release
001	October 2010	Complete Rewrite
002	March 2012	Revised to include the incorporation of MRP-227, Rev. 0 SER and MRP-227-A
		Addition of Section 6.0 and renumbering

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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

CE – Combustion Engineering

CFR – Code of Federal Regulations

CLB – Current Licensing Basis

CRGT – Control Rod Guide Tube

CSA – Core Support Assembly

CSS – Core Support Shield

Duke Energy – Duke Energy Carolinas, LLC

EPRI – Electric Power Research Institute

FD – Flow Distributor

FMECA – Failure Modes, Effects, and Criticality Analysis

FOL – Facility Operating License

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-A)

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)

LAR – License Amendment Request

LCB – Lower Core Barrel

LOCA – Loss of Coolant Accident

LR – License Renewal

List of Acronyms and Abbreviations (Continued)

LRA – License Renewal Application

LRAAI – License Renewal Applicant Action Items

LTS – Lower Thermal Shield

MRP – Materials Reliability Program

MUR – Measurement Uncertainty Recapture

NDE – Non-Destructive Examination

NRC – U.S. Nuclear Regulatory Commission

OE – Operating Experience

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 Owner's Group

PWSCC – Primary Water Stress Corrosion Cracking

QA – Quality Assurance

RI-FG – Reactor Internals-Focus Group

RIS – Regulatory Issue Summary

RFO – Refueling Outage

RV – Reactor Vessel

SCC – Stress Corrosion Cracking

SER – Safety Evaluation Report

SSC – Structures, Systems, and Component

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 (MRP): Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines (MRP-227-A)"[1] and described using the ten Aging Management Program (AMP) elements in Chapter XI, AMP XI.M16A of the 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) PWR MRP of MRP-227, Rev. 0.[3] The NRC reviewed and approved MRP-227, Rev. 0; the documentation was first provided in a June 2011 safety evaluation report (SER)[4] and later in a revised December 2011 SER.[21] The December 2011 SER is included in the updated MRP-227-A report. The ONS RV Internals inspection plan also contains a discussion of operational experience, time-limited aging analyses (TLAAs), and relevant programs and activities.

The ONS program to manage the aging of the RV Internals 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 the RV Internals components will remain functional through the ONS license renewal (LR) 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).[5] 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.[6] The SER NUREG-1723[7] documented the technical review of the ONS-1, ONS-2, and ONS-3 LRA by the NRC Staff.

The Renewed Facility Operating License (FOL) 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[8] which identifies the technical basis for issuing the renewed licenses as being set forth in NUREG-1723.

Section 4.3.11 of the LRA[5] discusses the ONS RV Internals AMP for LR. Per the LRA, Duke Energy proposed an ONS RV Internals AMP which may include the following activities:

a) Continue the characterization of the potential aging effects that have been identified in BAW-2248[11], *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[5]]. 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-2248, 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 the B&WOG RV Internals AMP and any other industry programs as appropriate, 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 in accordance with 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 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 of this report 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[7], Section 3.4.3 (pages 3-97 through 3-100)	Renewal Applicant 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 Action Items and resolves the action items and states the action items have been resolved.
NUREG-1723[7], Section 3.4.3 (page 3-98)	Renewal Applicant 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 renewed 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[7], Section 3.4.3.3 (page 3-114)	Combination of periodic in-service inspection required by ASME B&PV Code 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 RV 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.[9] See Section 6.6 of this report for a discussion of MRP-227-A Applicant/Licensee Action Item #7 which discusses analyses of CASS RV Internals items.
NUREG-1723[7], 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[7], 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[7], 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[7], Section 4.2.5 (page 4-24). Duke Energy will continue to monitor and track occurrences of design transients for the ONS units.

Commitment reference section	Commitment/Action Items	Reference section describing the fulfillment of the commitment
	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 was provided to the NRC by Duke Energy on February 20, 2012.[10]
	4. Flaw growth acceptance	Duke Energy's response for this TLAA was found to be acceptable by NUREG-1723[7], Section 4.2.5 (pages 4-24 through 4-25). No additional action by Duke Energy is required.
ONS LRA[5], 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 LR 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 2010.[2] Additionally, a Regulatory Issue Summary (RIS) was issued by the NRC in July 2011[19] to inform addressees of updated NRC procedures for LRA reviews or the review of certain licensee submittals related to commitments made in the process of receiving a renewed license for PWR RV Internals. Due to the previously granted renewed license, the ONS units are deemed a Category "A" facility (see Section 2.5.4 of this report).

The initial work performed, which supports the ONS RV Internals inspection plan, included an aging management review (AMR) documented in BAW-2248[11] 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.[12] The NRC's letter and FSER are included in the updated BAW-2248A report.[13] The NRC identified 12 license renewal applicant action items (LRAAIs) 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. 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.[3] MRP-227, Rev. 0 was submitted in January of 2009 to the NRC for review and SER approval.[14] In June 2011, the SER for MRP-227, Rev. 0 was issued by the NRC[4], in December 2011 the SER was revised[21], and in December 2011 MRP-227-A was issued by the EPRI PWR MRP.[1] Components requiring augmented inspections are grouped as "Primary", "Expansion", or "Existing Programs". Components not requiring augmented inspections are grouped as "No Additional Measures". The industry program is intended to provide a consistent approach to the aging management of RV Internals components

across the PWR fleet. For additional information about MRP-227, Rev. 0 see Section 4.1.1.1 of this report and for additional information about the SER to MRP-227, Rev. 0 and MRP-227-A see Section 4.1.1.1.3 of this report.

2.3 Intent of the ONS Program to Manage the Aging of the RV Internals

The program to manage the aging of the ONS RV Internals, 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] Code Section XI[15] 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 I&E guidelines in MRP-227-A. Augmented inspections are in addition to the requirements of ASME B&PV Code Section XI[15]; the I&E guidelines do not reduce, alter, or otherwise affect current ASME B&PV Code Section XI examinations.

Aging degradation mechanisms that impact the RV Internals have been identified in MRP-227-A. The overall outcome of the additional work performed by the industry summarized in MRP-227-A is to ensure functionality of the RV Internals is maintained by detection of the effects of the aging degradation mechanisms listed in Table 2-2 of this report. Therefore, this ONS RV Internals inspection plan is consistent with the industry work provided in MRP-227-A.

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 (TE)	Loss of Ductility, Toughness, and Unstable Crack Extension
Irradiation Embrittlement (IE)	Loss of Ductility, Toughness, 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 given in Chapter XI, AMP XI.M16A of NUREG-1801, Rev. 2 to describe the ONS RV Internals inspection plan and program to manage the aging of the ONS RV Internals as required by the NRC. The program to manage the aging of the ONS RV Internals, which will 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 of this report. ONS RV Internals components within the scope of BAW-2248A, the LRA AMR, and NUREG-1723 have been considered in this ONS RV Internals inspection plan.

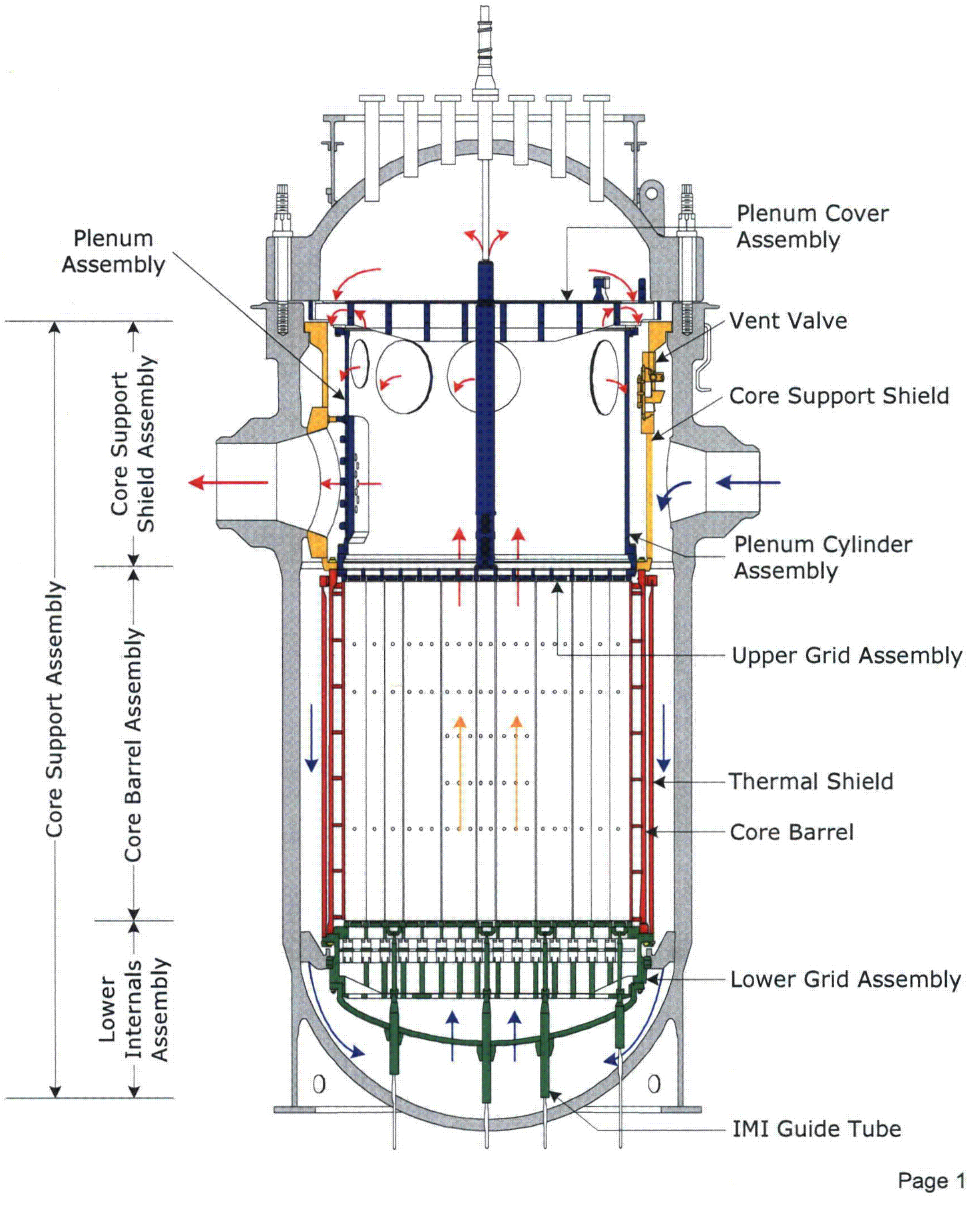
2.4 ONS RV Internals Scope Background

The intended functions of the ONS RV Internals are as follows[7,13]:

- 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 (CSA). 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[5] and determined they are bounded by the description contained in BAW-2248, 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-2248; 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 and the industry work which culminated in MRP-227-A. The general arrangement of the B&W RV Internals is shown in Figure 2-1 of this report.[1]

Figure 2-1. General B&W RV Internals Arrangement



2.5 Background on ONS RV Internals Inspection Plan Submittal

Three letters and a license amendment request (LAR) have been submitted to the NRC concerning Duke Energy's proposed ONS RV Internals inspection plan. Additionally, the NRC has issued a RIS with the purpose of facilitating a predictable and consistent method for reviewing the AMPs and inspection plans for PWR plants that have received renewed operating licenses. These letters, LAR, and RIS are discussed in chronological order in Sections 2.5.1 through 2.5.5 of this report.

2.5.1 Letter of Intent to Adopt MRP-227

By letter dated June 16, 2010 to the NRC[16], Duke Energy stated its intent to revise the existing LR 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 as approved by the NRC. This ONS RV Internals inspection plan, as documented herein, is based on MRP-227-A. Once the appropriate revision of the ONS RV Internals inspection plan is approved, the ONS UFSAR will be updated as required.

2.5.2 Letter for Commitment Date Change for License Amendment Request Submittal

By letter dated October 20, 2010 to the NRC[17], Duke Energy advised the NRC of a revision to an existing LR commitment date to submit the LAR for inspecting the RV Internals at ONS. In the letter dated June 16, 2010 (see Section 2.5.1 of this report), Duke Energy committed to submit its revised RV Internals inspection plan for ONS later in 2010 and at least two years prior to the planned inspection. The planned inspection for ONS-1 is tentatively scheduled to begin on October 25, 2012. The October 20, 2010 letter requested that the commitment date to provide the RV Internals inspection plan be extended from October 25, 2010 to November 30, 2010.

2.5.3 Previously Submitted License Amendment Request

In November 2010, Duke Energy proposed to amend Renewed FOLs DPR-38, DPR-47, and DPR-55 for ONS-1, ONS-2, and ONS-3. Specifically, Duke Energy requested the NRC to review and approve the proposed adoption of the RV Internals inspection plan based on the use of MRP-227, Rev. 0 via LAR, LAR Number 2010-06.[18] In the LAR, two regulatory commitments were identified. The first stated that after the approval of MRP-227, Duke Energy would review and, if needed, revise the ONS RV Internals inspection plan. The due date for this regulatory commitment is 90 days after issuance of MRP-227-A. The ONS RV Internals inspection plan contained within this revision of this report and the corresponding submittal to the NRC for review and approval fulfills this regulatory commitment.

The second regulatory commitment made in the November 2010 LAR submittal was to update the ONS ISI program to include the items from the NRC-approved ONS RV Internals inspection plan as augmented inservice inspections and submit the inspection results in the 90 day outage report. This regulatory commitment remains.

2.5.4 Regulatory Issue Summary (RIS) 2011-07

On July 21, 2011, the NRC issued RIS 2011-07 entitled "License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management".[19] The RIS was issued to inform addressees of updated NRC procedures for LRA reviews or the review of certain licensee submittals related to commitments made in the process of receiving a renewed license. The RIS also provides information to licensees with respect to how to meet their existing LR commitments related to RV Internals AMPs and acceptable changes to existing LR commitments in order to account for the recent issue of the SER to MRP-227, Rev. 0 and MRP-227-A.

The NRC granted Duke Energy a renewed license for ONS based on various commitments; one of the commitments made was to submit an inspection plan for RV Internals components to the NRC for review and approval as described in Section 2.1 of this report. With the issuance of the SER for MRP-227, the NRC has updated their internal guidance on implementation of this RV Internals component AMP guidance relative to LR

for PWR plants on a plant-specific basis. Four categories of plants, "A" through "D", have been created to reflect the licensee's status relative to LR approval and, for those plants with renewed licenses, whether the licensee has or has not submitted its inspection plan to the NRC for review and approval. ONS is a Category "A" facility; Category "A" is described as "plants with renewed licenses that have already submitted an AMP/inspection plan based on MRP-227, Revision 0 to comply with an existing commitment."

NRC expectations for the Category "A" plants are that licensees may withdraw their current AMP/inspection plan submittals and provide new/revised commitments to resubmit AMPs/inspection plans in accordance with MRP-227-A no later than October 1, 2012 and that future submittals should address all information required by MRP-227-A. Section 2.5.5 of this report describes Duke Energy's intent for resubmittal of the previously submitted LAR described in Section 2.5.3 of this report.

2.5.5 Letter for RV Internals License Amendment Request Status Update

By letter dated September 1, 2011 to the NRC[20], Duke Energy submitted further clarification of their intent to revise the November 2010 LAR. The subject letter references a July 25, 2011 conference call between Duke Energy and the NRC to discuss the status of the LAR submitted in November 2010 (LAR Number 2010-06). As an action from the conference call, Duke Energy committed to submit a revision to the LAR 90 days following the issuance of the revised MRP-227-A report, but no later than October 1, 2012. The ONS RV Internals inspection plan contained within this revision of this report and the corresponding submittal to the NRC for review and approval fulfills this commitment.

3.0 PROGRAM OWNER

The Oconee Reactor Engineering and Duke Energy Corporate Programs groups are responsible for establishing, maintaining, and implementing the ONS RV Internals inspection plan.

4.0 INDUSTRY AND ONS PROGRAMS AND ACTIVITIES

This section contains pertinent ONS and industry programs and activities used for the development and implementation of MRP-227-A, the ONS RV Internals inspection plan, and the program to manage the aging of the ONS RV Internals.

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. 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 as well as provided training for applying MRP-227.

4.1.1.1 MRP-227

The EPRI PWR MRP efforts have defined the inspection and examination techniques for the RV Internals. The I&E guidelines were initially published in MRP-227, Rev. 0 and submitted to the NRC for review and approval in January 2009.[14] An SER for MRP-227 was issued by the NRC in June 2011[4] and was revised in December 2011.[21] MRP-227-A contains the revised SER and incorporates the NRC's recommended changes. The results of these efforts serve as the basis for identifying any augmented inspections required by the ONS RV Internals inspection plan.

4.1.1.1.1 Development of MRP-227

The MRP-227 "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" 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-A[1] is the result of the industry work and NRC review and approval 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 analysis) 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-A utilizes the screening and ranking process to aid in the identification of required inspections for "Primary" and "Expansion" components and credits existing component aging management activities when they are deemed adequate.

The basic description of each group 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 [MRP-227] 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-A 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-A approach are not intended to supersede any ASME B&PV Code Section XI requirements.

The guidelines in MRP-227-A are classified in accordance with the NEI 03-08 Guidelines.[22] For the MRP-227-A guidelines there are one "Mandatory", five "Needed", and zero "Good Practice" elements as follows:

- **"Mandatory"**
  Each commercial U. S. PWR unit shall develop and document a program for management of aging of reactor internals components within thirty-six months following issuance of MRP-227-Rev. 0 (that is, no later than December 31, 2011).

  As described in Section 2.5.3 of this report, an LAR which documented an earlier revision of the ONS RV Internals inspection plan (based on MRP-227, Rev. 0) was previously submitted to the NRC in November 2010. This November 2010 documentation plus the additional documentation contained within this report contains the information necessary to fulfill this "Mandatory" element.

- **"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-A, Table 4-1 ("Primary"), Table 4-4 ("Expansion"), and Table 5-1 (Examination Acceptance and Expansion Criteria) are attached herein as Appendices A, B, and C. There are no "Existing Program" components in MRP-227-A for the B&W-designed PWRs. The ONS RV Internals inspection plan implements the applicable B&W tables contained in MRP-227-A, thus fulfilling this "Needed" element. Implementation is achieved by performance of inspections of applicable components within the timeframe specified in the guidance provided in the applicable tables of MRP-227-A.

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

  The examinations specified in the ONS RV Internals inspection plan will be conducted in accordance with MRP-228[23]. Reference 23 is the most recent revision of MRP-228 at the time of this report; the revision in effect at the time of the inspections will be utilized for future inspections. The ONS RV Internals inspection plan fulfills this "Needed" element for the three ONS units.

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

  The ONS Corrective Action Program (Problem Investigation Process [PIP] program) will be applied as discussed in Section 5.7 of this report. The ONS RV Internals inspection plan fulfills this "Needed" element for the three ONS units.

- **"Needed"**
  Each commercial U. S. PWR unit shall 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 within the scope of MRP-227 are examined.

  Duke Energy will provide a summary report of the inspections and monitoring, items requiring evaluation, and new repairs within the scope of MRP-227 to the MRP Program Manager within 120 days of the completion of an outage during which ONS RV Internals within the scope of MRP-227 are examined. The ONS RV Internals inspection plan fulfills this "Needed" element for the three ONS units.

- **"Needed"**
  If an engineering evaluation is used to disposition an examination result that does not meet the examination acceptance criteria in Section 5 [of MRP-227-A], this engineering evaluation shall be conducted in accordance with an NRC-approved evaluation methodology.

  Duke Energy will evaluate any examination results that do not meet the examination acceptance criteria in Section 5 [of MRP-227-A] in accordance with an NRC-approved methodology. The ONS RV Internals inspection plan fulfills this "Needed" element for ONS.

4.1.1.1.2 MRP-227-A Applicability to ONS

The MRP-227-A guidelines are based on several general assumptions that were used for the analysis in the development of MRP-227-A, found in Section 2.4 of MRP-227-A. These assumptions and their applicability to the ONS units are listed below:

- **30 years 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 each operate as a base load unit.

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

  MRP-227-A 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, Westinghouse, and Combustion Engineering [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 for material verification.

The MRP-227-A I&E guidelines scope does not ensure the satisfaction of every plant-specific LR or power uprate commitment; plant-specific commitments remain the responsibility of the owner. Duke Energy has submitted an application in September 2011[24] for a measurement uncertainty recapture (MUR) power uprate. Within this application, as required, the affect of the MUR power uprate on MRP-227 was documented.

ONS is planning to implement a MUR power uprate in 2013. An assessment of the above evaluations of the MRP-227 assumptions was performed. It is concluded that the three evaluations of the MRP-227 assumptions described in this section of the report will be unaffected by the MUR power uprate. In addition, the power uprate was assessed for its affect on the MRP-227, Rev. 0 RV Internals I&E guidelines; it was determined that the I&E guidelines will not be affected by the MUR power uprate.

Based on the above review, MRP-227-A is applicable to all three ONS units, with or without the planned MUR power uprate.

4.1.1.1.3 MRP-227, Rev. 0 Safety Evaluation Report and MRP-227-A

By letter dated January 12, 2009, EPRI submitted MRP-227, Rev. 0 for NRC review and approval.[14] In June 2011, the NRC issued Rev. 0 of the SER for MRP-227.[4] Revision 1 of the SER was issued in December 2011.[21] The NRC staff reviewed MRP-227, Rev. 0 to determine whether its guidance will provide reasonable assurance that the I&E of the subject RV Internals components will maintain their intended function during the period of extended operation. The review also considered compliance with LR requirements (10 CFR 54.21(a)(3)) in order to allow licensees or applicants the option of adopting the AMP methodology described in MRP-227 as the basis for managing age-related degradation in RV Internals components and incorporating, by reference, the recommended guidelines into PWR RV Internals AMPs (or their equivalent).

During their review of MRP-227, Rev. 0, the NRC issued four sets of requests for additional information (RAIs) that addressed technical issues. In addition, a draft version of the NRC's MRP-227 SER was posted for public comment on April 11, 2011, for 30 days. All comments received during the public comment period were reviewed and considered during the development of the final SER.

Section 3.0 of the SER documents the NRC's evaluation and findings pertaining to the adequacy of the MRP's AMP recommendations. In particular, Section 3.0 documents NRC concerns with MRP-227 and the basis for limitations and conditions being placed on the use of MRP-227 as well as licensee/applicant action items that shall be addressed by applicants/licensees who choose to implement the NRC-approved version of MRP-227. Section 4.0 of the SER summarizes the limitations and conditions and the applicant/licensee action items. Section 5.0 of the SER provides the conclusions resulting from the SER.

Based on its review, the NRC identified some issues and concerns in Section 3.0 of the SER that were not adequately resolved regarding the implementation of MRP-227. Some of the NRC's issues that are not adequately resolved and remaining concerns are related to conditions and limitations on the use of MRP-227. Seven conditions and limitations address deficiencies in the AMP defined by MRP-227, Rev. 0 and are identified in Section 4.1 of the SER. In addition, some of the NRC's issues and concerns that were not adequately resolved are related to applicant/licensee action items related to the use of MRP-227. Eight plant-specific action items address topics related to the implementation of MRP-227 that could not be effectively addressed on a generic basis in MRP-227, Rev. 0 and are identified in Section 4.2 of the SER. Although Section 4.1 and 4.2 of the SER describe the conditions and limitations and the plant-specific action items identified by the NRC, Section 3.0 more fully describes all concerns and shall be considered during any update to MRP-227 to comport with the SER. In addition, the reexamination frequency for "Primary" inspection category components shall be on a maximum 10-year interval, unless a plant-specific analysis providing justification for extended examination frequency is submitted and approved by the NRC.

MRP-227-A incorporates any needed changes as well as the SER itself. The topical report conditions/limitations and applicant/licensee action items contained in the SER and MRP-227-A applicable to ONS are addressed in Section 6.0 of this report as delineated in Table 4-1 of this report.

Table 4-1. Topical Report Conditions/Limitations and Applicant/Licensee Action Items Applicable to ONS

Item	Inspection Plan Location
Topical Report Condition/Limitation 1	Not Applicable to ONS
Topical Report Condition/Limitation 2	Not Applicable to ONS
Topical Report Condition/Limitation 3	Section 6.1
Topical Report Condition/Limitation 4	Section 6.1
Topical Report Condition/Limitation 5	Section 6.1
Topical Report Condition/Limitation 6	Section 6.1
Topical Report Condition/Limitation 7	Not Applicable to ONS
Applicant/Licensee Action Item 1	Section 6.2
Applicant/Licensee Action Item 2	Section 6.3
Applicant/Licensee Action Item 3	Not Applicable to ONS
Applicant/Licensee Action Item 4	Section 6.4
Applicant/Licensee Action Item 5	Not Applicable to ONS
Applicant/Licensee Action Item 6	Section 6.5
Applicant/Licensee Action Item 7	Section 6.6
Applicant/Licensee Action Item 8	Section 6.7

4.1.1.1.4 MRP-228

"Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228)"[23] 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 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 and offers a stable mechanism for documenting the capability of the evolving inspection technology.

MRP-228 contains four "Needed" and two "Good Practice" requirements, which will be followed in accordance with the elements of the NEI 03-08 Guidelines.[22] Duke Energy will ensure the MRP-228 requirements for the augmented inspections described in this ONS RV Internals inspection plan are met.

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 appropriate 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.

The PWROG developed WCAP-17096, "Reactor Internals Acceptance Criteria Methodology and Data Requirements." For each of the "Primary" and "Expansion" components listed in MRP-227, WCAP-17096 outlines the type of analyses required, required evaluation procedures, data required to support analysis, logic charts illustrating evaluation path and potential disposition options, and components that can be addressed on a generic basis. WCAP-17096 has been reviewed by the NRC and the NRC has issued requests for additional information (RAIs). No SER for WCAP-17096 has been issued at the date of this report.

The PWROG developed the basis for design functionality acceptance criteria consisting of researching and documenting the functional (design basis) requirements for selected items and loading conditions that may exist during operation. The existing core barrel assembly (CBA) finite element analysis model developed under a PWROG project was modified for specific differences that are anticipated to cover the variations in the B&W-designed RV Internals. This provided unit-specific tools to analyze the condition of the CBA. The components included in the model are the "Primary" and "Expansion" items in MRP-227. In addition to the above, the

PWROG is working on and proposing B&W generic tasks that are determined to be desirable in preparation of performing the inspection.

4.1.3 Time-Limited Aging Analyses

This section contains discussions of both currently identified TLAAs as well consideration of future TLAAs for the ONS RV Internals.

4.1.3.1 Currently Identified TLAAs

In the ONS LRA[5], three RV Internals applicable TLAAs were identified, as listed below. These TLAAs were evaluated in Reference 11 and summarized in NUREG-1723, Section 4.2.5, 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 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; this analysis concluded that at the end of a 60-year lifetime, the internals will have adequate ductility to absorb local strain and the irradiation will not adversely affect deformation limits. Duke Energy provided this analysis to the NRC on February 20, 2012.[10]

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[25] and subsequently in a letter dated October 15, 1999[26] Duke Energy responded that no flaws have been identified in the ONS RV Internals and hence no evaluation is required. This response closes Open Item 4.2.5.3-2.

4.1.3.2 Consideration of Future TLAAs

Duke Energy has made scheduled commitments based on the SER to MRP-227, Rev. 0 (as contained in MRP-227-A). Section 6.5 of this report describes the resolution to Applicant/Licensee Action Item 6 from the SER concerning the justification of acceptability of inaccessible components. The inaccessible items (core barrel cylinder including vertical and circumferential seam welds, former plates, external baffle-to-baffle bolts and their locking devices, the core barrel-to-former bolts and their locking devices) and the core barrel assembly internal baffle-to-baffle bolts need to be evaluated or a schedule for replacement of the components needs to be provided to the NRC; if any of the evaluations meet the definition in 10 CFR 54.3, they may be considered TLAAs. Section 6.5 of this report provides scheduling information for this Applicant/Licensee Action Item.

Section 6.6 of this report describes the resolution to Applicant/Licensee Action Item 7 from the SER concerning plant-specific analyses to demonstrate RV Internals components fabricated from CASS, martensitic stainless steel, or precipitation-hardened stainless steel will maintain their functionality during the period of extended operation. These potential analyses must also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. If any of the analyses meet the definition in 10 CFR 54.3, they may be considered TLAAs. Section 6.6 of this report provides scheduling information for this Applicant/Licensee Action Item.

4.1.4 Internals Bolting Surveillance Program

Starting in 1981, ultrasonic testing (UT) at several B&W units revealed that multiple RV Internals bolts had rejectable UT indications. The failure mechanism was determined to be intergranular stress corrosion cracking (IGSCC). The failed bolts were predominantly fabricated from Alloy A-286, Condition A (ASTM A 453, Grade 660) material.[27]

As a result of the noted bolt failures, utilities began replacing bolts where needed. The B&WOG initiated the IBSP (which was completed by the EPRI PWR MRP) to better assess the IGSCC susceptibility of the replacement bolts. The IBSP exposed replacement bolts to simulated PWR conditions in a laboratory autoclave and an actual PWR environment inside an operating PWR unit. The 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.

At the completion of the IBSP tests, several peened replacement Alloy A-286 replacement bolts were observed to have developed IGSCC when loaded to a high stress, while the un-peened Alloy A-286 replacement bolts were free from IGSCC when loaded to similarly high stresses for the test duration of 8 ½ years in the reactor capsules. The Alloy X-750, HTH Condition 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. The majority of each SSHT assembly at the three ONS units, including the SSHT bolts, was removed from the RV Internals, and therefore no longer have an IGSCC concern.

A 2005 evaluation of the IBSP and industry experience resulted in a PWROG letter, 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 examination recommendations have been incorporated into MRP-227-A with the "Needed" recommendation being incorporated into the "Primary" category and the "Good Practice" recommendation being incorporated into the "Expansion" category except for the FD bolts which are categorized as "Primary" in MRP-227-A. The MRP-227-A examinations supersede these UT examination recommendations.

4.1.5 Joint Owners' Baffle Bolt Program

The JOBB Program stemmed from UT examinations of baffle-to-former bolts at several French 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) examinations, irradiation and mechanical testing, corrosion testing, and microstructural evaluation were used to characterize the effect of irradiation on bolting materials under the JOBB program.

The JOBB program 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, and specifically referenced in MRP-227-A. In BAW-2248A, and 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 28 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 by AREVA and the utilities with similarly designed operating units. The results of the most recent investigation identified several apparent and contributing causes. The team recommended high level scoping-feasibility studies by AREVA to address the proposed preventative actions. The "Primary" requirement from MRP-227-A 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 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 ISI program, the 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, and visual examination of baffle-to-baffle and baffle-to-former bolts at each RFO. In addition Duke has contracted and is contracting with AREVA to develop analyses in preparation for the ONS inspections to aid in resolution of indications in a timely manner, determine risk, and develop contingencies.

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.[15] Areas accessible during a refueling outage (RFO) of the RV interior (Examination Category B-N-1) are examined using visual examination (VT-3) methods each period. 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. Core support structures are examined using visual VT-3 examination methods each interval (Examination Category B-N-3). 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 (including welds) 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. Typical 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 4-2 of this report. Relevant conditions for these examination categories are found in ASME B&PV Code Section XI, IWB-3520.

Table 4-2. Typical ONS Core Support Structure Components (Examination Category B-N-3)

Typical Components
Thermal Shield
Thermal Shield Upper Restraint Assemblies

Typical Components
Upper Thermal Shield Bolting
Remnants of Surveillance Specimen Holder Tube Structures
Core Support Shield (CSS) Assembly
CSS Top Flange, including Seating Surfaces
CSS Outlet Nozzles
CSS Outlet Nozzle Sealing Surfaces
CSS 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 (Baffle-to-Baffle and Baffle-to-Former 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 Studs/Nuts
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 (Top and Bottom)
Plenum LOCA Bosses and Welds
Upper Grid Assembly (including Bolting and Grid Pads)

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 help prevent the coolant from becoming an environment favorable to stress corrosion cracking (SCC), and therefore greatly reduces the probability of SCC and IASCC.

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. 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 Studs/Nuts

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 studs/nuts at each unit are secured by tie plate and crimp locking cups in groups of two (two-hole design). The replacement studs/nuts are made from 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 studs/nuts and their locking devices are categorized as "Expansion" items in MRP-227-A.[1]

4.2.6 Fabrication Records Search

Two fabrication records searches were conducted by AREVA for the ONS RV Internals components. The goals of the records searches were to locate the chemical composition of the CASS items in MRP-227, Rev. 0, and to obtain a detailed description of the "Primary" and "Expansion" components in the MRP-227, Rev. 0, including the function, fabrication records, OE, and the anticipated degradation mechanisms. The chemical composition of the CASS items was used to, if possible, screen for susceptibility to thermal aging embrittlement, consistent with the screening criteria used by the EPRI PWR MRP.

During the records search, a feature unique to the ONS-1 plenum cover weldment rib pads was identified. 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. 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.

4.2.7 Volumetric (UT) Examinations of Upper Core Barrel Bolts

The most recent UCB bolt UT examinations 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 from the PWROG that also met the initial examination requirements in MRP-227-A Table 4-1. The results of the examinations at ONS-1 and ONS-2 were zero UCB bolts rejected as in previous examinations. 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 examination 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 examinations 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 Alloy A-286 bolt failures.

UT examinations for the LCB bolts are planned during the RV Internals inspections in 2012, 2013, and 2014, in compliance with the MRP-227-A examination requirement for the LCB bolts.

The AREVA UT examination procedure for the ONS UCB bolt examinations in 2007 and 2008 was validated by blind performance demonstration at EPRI prior to the bolt examinations 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, 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 D of this report provides a nonproprietary version of the current revision 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 a PWROG project, the ONS unit-specific analysis demonstrates large margins using the most recent UCB bolt UT examination results.

4.2.8 Core Clamping Measurements

Core clamping measurements were obtained by AREVA at ONS-1 (2006), ONS-2 (2008), and ONS-3 (2007) during RFOs that satisfy the MRP-227-A Table 4-1 requirements for a one-time physical differential height measurement of the plenum rib pad to RV seating surface. 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. Potential wear of the core clamping items in the plenum cover assembly and CSS assembly will continue to be monitored via subsequent VT-3 examinations performed on the 10-year ISI interval per MRP-227-A 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 in recent RFOs of the internal baffle-to-baffle bolts and baffle-to-former bolts at ONS units. 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 internal baffle-to-baffle bolts or baffle-to-former bolts have been observed during these visual inspections at the ONS units.

The only known abnormal condition for the internal baffle-to-baffle bolts and baffle-to-former bolts in the B&W units was noted from the inspection of the baffle-to-former bolts and internal baffle-to-baffle bolts at another operating B&W unit. Visual inspections indicated that some 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 this unit 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 have not been removed for laboratory examination to confirm the failures.

4.3 Conclusions of Section 4.0

This section contains pertinent ONS and industry programs and activities used for the development and implementation of MRP-227-A, the ONS RV Internals inspection plan, and the program to manage the aging of the ONS RV Internals.

The inspections will consist of the ASME B&PV Code Section XI Examination Category B-N-3 examinations (typical components given in Table 4-2 of this report) and the augmented inspections from MRP-227-A. Changes resulting from the NRC's review of this report will be incorporated as appropriate.

Past and on-going activities by the EPRI PWR MRP, B&WOG, PWROG, Duke Energy and interaction with the NRC provide the needed clarification to the level of inspection quality necessary to determine the proper examination method and frequencies. The program to manage the aging of the ONS RV Internals includes existing ONS programs and ASME B&PV Code Section XI inspections combined with MRP-227-A 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 program to manage the aging of the ONS RV Internals, which will include this ONS RV Internals inspection plan, is based on the ten GALL program elements from Chapter XI, AMP XI.M16A from NUREG-1801, Rev. 2. Each subsection within this section quotes one GALL program element (in italics), provides information on how Duke Energy is meeting that program element for the ONS units, and provides a conclusion for that program element.

> *"This program relies on implementation of the Electric Power Research Institute (EPRI) Report No. 1016596 (MRP-227) and EPRI Report No. 1016609 (MRP-228) to manage the aging effects on the reactor vessel internal (RVI) components.*
>
> *This program is used to manage the effects of age-related degradation mechanisms that are applicable in general to the PWR RVI components at the facility. These aging effects include (a) various forms of cracking, including stress corrosion cracking (SCC), which also encompasses primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), or cracking due to fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness due to either thermal aging or neutron irradiation embrittlement; (d) changes in dimension due to void swelling; and (e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.*
>
> *The program applies the guidance in MRP-227 for inspecting, evaluating, and, if applicable, dispositioning non-conforming RVI components at the facility. The program conforms to the definition of a sampling-based condition monitoring program, as defined by the Branch Technical Position RSLB-1, with periodic examinations and other inspections of highly-affected internals locations. These examinations provide reasonable assurance that the effects of age-related degradation mechanisms will be managed during the period of extended operation. The program includes expanding periodic examinations and other inspections if the extent of the degradation effects exceeds the expected levels.*
>
> *The MRP-227 guidance for selecting RVI components for inclusion in the inspection sample is based on a four-step ranking process. Through this process, the reactor internals for all three PWR designs were assigned to one of the following four groups: Primary, Expansion, Existing Programs, and No Additional Measures components. Definitions of each group are provided in GALL Chapter IX.B.*
>
> *The result of this four-step sample selection process is a set of Primary Internals Component locations for each of the three plant designs that are expected to show the leading indications of the degradation effects, with another set of Expansion Internals Component locations that are specified to expand the sample should the indications be more severe than anticipated. The degradation effects in a third set of internals locations are deemed to be adequately managed by Existing Programs, such as ASME Code, Section XI,11 Examination Category B-N-3 examinations of core support structures. A fourth set of internals locations are deemed to require no additional measures. As a result, the program typically identifies 5 to 15% of the RVI locations as Primary Component locations for inspections, with another 7 to 10% of the RVI locations to be inspected as Expansion Components, as warranted by the evaluation of the inspection results. Another 5 to 15% of the internals locations are covered by Existing Programs, with the remainder requiring no additional measures. This process thus uses appropriate component functionality criteria, age-related degradation susceptibility criteria, and failure consequence criteria to identify the components that will be inspected under the program in a manner that conforms to the sampling criteria for sampling-based condition monitoring programs in Section A.1.2.3.4 of NRC Branch Position RLSB-1. Consequently, the sample selection process is adequate to assure that the intended function(s) of the PWR reactor internal components are maintained during the period of extended operation.*
>
> *The program's use of visual examination methods in MRP-227 for detection of relevant conditions (and the absence of relevant conditions as a visual examination acceptance criterion) is consistent with the ASME Code, Section XI rules for visual examination. However, the program's adoption of the MRP-227 guidance for visual examinations goes beyond the ASME Code, Section XI visual examination criteria*

because additional guidance is incorporated into MRP-227 to clarify how the particular visual examination methods will be used to detect relevant conditions and describes in more detail how the visual techniques relate to the specific RVI components and how to detect their applicable age-related degradation effects.

The technical basis for detecting relevant conditions using volumetric ultrasonic testing (UT) inspection techniques can be found in MRP-228, where the review of existing bolting UT examination technical justifications has demonstrated the indication detection capability of at least two vendors, and where vendor technical justification is a requirement prior to any additional bolting examinations. Specifically, the capability of program's UT volumetric methods to detect loss of integrity of PWR internals bolts, pins, and fasteners, such as baffle-former bolting in B&W and Westinghouse units, has been well demonstrated by operating experience. In addition, the program's adoption of the MRP-227 guidance and process incorporates the UT criteria in MRP-228, which calls for the technical justifications that are needed for volumetric examination method demonstrations, required by the ASME Code, Section V.

The program also includes future industry operating experience as incorporated in periodic revisions to MRP-227. The program thus provides reasonable assurance for the long-term integrity and safe operation of reactor internals in all commercial operating U.S. PWR nuclear power plants.

Age-related degradation in the reactor internals is managed through an integrated program. Specific features of the integrated program are listed in the following ten program elements. Degradation due to changes in material properties (e.g., loss of fracture toughness) was considered in the determination of inspection recommendations and is managed by the requirement to use appropriately degraded properties in the evaluation of identified defects. The integrated program is implemented by the applicant through an inspection plan that is submitted to the NRC for review and approval with the application for license renewal."

5.1 AMP Element 1 – Scope of Program

"The scope of the program includes all RVI components at the Oconee Nuclear Station, Units 1, 2, and 2, which are built to a B&W NSSS design. The scope of the program applies the methodology and guidance in the most recently NRC-endorsed version of MRP-227, which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of safety-related internals in commercial operating U.S. PWR nuclear power plants designed by B&W, CE, and Westinghouse. The scope of components considered for inspection under MRP-227 guidance includes core support structures (typically denoted as Examination Category B-N-3 by the ASME Code, Section XI), those RVI components that serve an intended license renewal safety function pursuant to criteria in 10 CFR 54.4(a)(1), and other RVI components whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii). The scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation, because these components are not typically within the scope of the components that are required to be subject to an aging management review (AMR), as defined by the criteria set in 10 CFR 54.21(a)(1). The scope of the program also does not include welded attachments to the internal surface of the reactor vessel because these components are considered to be ASME Code Class 1 appurtenances to the reactor vessel and are adequately managed in accordance with an applicant's AMP that corresponds to GALL AMP XI.M1, "ASME Code, Section XI Inservice Inspection, Subsections IWB, IWC, and IWD."

The scope of the program includes the response bases to applicable license renewal applicant action items (LRAAIs) on the MRP-227 methodology, and any additional programs, actions, or activities that are discussed in these LRAAI responses and credited for aging management of the applicant's RVI components. The LRAAIs are identified in the staff's safety evaluation on MRP-227 and include applicable action items on meeting those assumptions that formed the basis of the MRP's augmented inspection and flaw evaluation methodology (as discussed in Section 2.4 of MRP-227), and NSSS vendor-

specific or plant-specific LRAAIs as well. The responses to the LRAAIs on MRP-227 are provided in Appendix C of the LRA.

The guidance in MRP-227 specifies applicability limitations to base-loaded plants and the fuel loading management assumptions upon which the functionality analyses were based. These limitations and assumptions require a determination of applicability by the applicant for each reactor and are covered in Section 2.4 of MRP-227."

5.1.1 ONS Scope

See Section 2.4 of this report for a discussion of the scope of the components considered for inspection for the program to manage the aging of the ONS RV Internals and Section 4.1.1.1.2 of this report for applicability of MRP-227-A to the ONS units. The components of the ONS RV Internals that were evaluated by the industry, resulting in MRP-227-A, include those identified in the ONS LRA AMR. The components evaluated include the components identified in NUREG-1801, Rev. 2. TLAAs are also included in the program to manage the aging of the ONS RV Internals and are discussed in Section 4.1.3 of this report.

5.1.2 Conclusion

The components evaluated for the scope of the ONS RV Internals inspection plan include those listed in the LRA AMR and in Table IV.B4 of NUREG-1801, Rev. 2. The intent of the first GALL program element from Chapter XI, AMP XI.M16A from NUREG-1801, Rev. 2 is met as described in Section 5.1.1 of this report.

5.2 AMP Element 2 – Preventative Actions

"The guidance in MRP-227 relies on PWR water chemistry control to prevent or mitigate aging effects that can be induced by corrosive aging mechanisms (e.g., loss of material induced by general, pitting corrosion, crevice corrosion, or stress corrosion cracking or any of its forms [SCC, PWSCC, or IASCC]). Reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program. The program description, evaluation, and technical basis of water chemistry are presented in GALL AMP XI.M2, 'Water Chemistry.'"

5.2.1 ONS Preventative Actions

See Section 4.2.2 of this report for a discussion of the ONS Primary Water Chemistry Program. Additionally, see Section 4.2.4 of this report for a discussion of the use of low-leakage cores which Duke Energy considers to also be a preventative action for the ONS units.

5.2.2 Conclusion

The intent of the second GALL program element from Chapter XI, AMP XI.M16A from NUREG-1801, Rev. 2 is met as described in Section 5.2.1 of this report.

5.3 AMP Element 3 – Parameters Monitored/Inspected

"The program manages the following age-related degradation effects and mechanisms that are applicable in general to the RVI components at the facility: (a) cracking induced by SCC, PWSCC, IASCC, or fatigue/cyclical loading; (b) loss of material induced by wear; (c) loss of fracture toughness induced by either thermal aging or neutron irradiation embrittlement; (d) changes in dimension due to void swelling and irradiation growth, distortion, or deflection; and (e) loss of preload caused by thermal and irradiation-enhanced stress relaxation or creep. For the management of cracking, the program monitors for evidence of surface breaking linear discontinuities if a visual inspection technique is used as the non-destruction examination (NDE) method, or for relevant flaw presentation signals if a volumetric UT method is used as the NDE method. For the management of loss of material, the program monitors for gross or abnormal surface conditions that may be indicative of loss of material occurring in the components. For the management of loss of preload, the program monitors for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed, or pinned connections. The program does not directly monitor for loss of fracture toughness that is induced by thermal aging or

neutron irradiation embrittlement, or by void swelling and irradiation growth; instead, the impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the components and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected in the components and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation under the MRP-227 guidance or ASME Code, Section XI requirements. The program uses physical measurements to monitor for any dimensional changes due to void swelling, irradiation growth, distortion, or deflection.

Specifically, the program implements the parameters monitored/inspected criteria for B&W designed Primary Components in Table 4-1 of MRP-227. Additionally, the program implements the parameters monitored/inspected criteria for B&W designed Expansion Components in Table 4-4 of MRP-227. The parameters monitored/inspected for Existing Program Components follow the bases for referenced Existing Programs, such as the requirements for ASME Code Class RVI components in ASME Code, Section XI, Table IWB-2500-1, Examination Categories B-N-3, as implemented through the applicant's ASME Code, Section XI program, or the recommended program for inspecting Westinghouse-designed flux thimble tubes in GALL AMP XI.M37, "Flux Thimble Tube Inspection." No inspections, except for those specified in ASME Code, Section XI, are required for components that are identified as requiring "No Additional Measures," in accordance with the analyses reported in MRP-227."

5.3.1 ONS Parameters Monitored/Inspected

The ONS ASME B&PV Code Section XI ISI Program Examination Category B-N-3 is discussed in Section 4.2.1 of this report and the ONS vent valve testing and inspection program is discussed in Section 4.2.3 of this report. Tables 4-1 and 4-4 of MRP-227-A are included in this report as Appendices A and B. The inspections listed in MRP-227-A Tables 4-1 and 4-4 that are applicable to the ONS units will be included as augmented inservice inspections in the ONS ISI program. There are no "Existing Program" components for B&W-designed RV Internals in MRP-227-A.

5.3.1.1 MRP-227-A "Primary" (Augmented Inservice) Inspections for ONS

MRP-227-A lists the following ONS "Primary" RV Internals bolts to be examined with UT:

- UCB Bolts
- LCB Bolts
- Baffle-to-Former Bolts
- Flow Distributor (FD) Bolts

MRP-227-A lists the following ONS "Primary" RV Internals components to be examined with a VT-3:

- Plenum Cover Weldment Rib Pads
- Plenum Cover Support Flange
- CSS Top Flange
- CRGT Spacer Castings
- 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
- FD Bolt Locking Devices
- Alloy X-750 Dowel-to-Guide Block Welds
- IMI Guide Tube Spiders
- IMI Guide Tube Spider-to-Lower Grid Rib Section Welds

MRP-227-A lists the following ONS "Primary" RV Internals components to be examined by physical measurement:

A one-time physical differential height measurement of the top of the plenum rib pad to RV seating surface (with plenum in the RV) is required by MRP-227-A. 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

5.3.1.2 MRP-227-A "Expansion" (Augmented Inservice) Inspections for ONS

MRP-227-A lists the following ONS "Expansion" RV Internals bolts or studs/nuts to be inspected with UT:

- Upper Thermal Shield (UTS) Bolts
- Lower Thermal Shield (LTS) Studs/Nuts

MRP-227-A lists the following ONS "Expansion" RV Internals components to be examined with a VT-3:

- Alloy X-750 Dowel-to-Upper Grid Fuel Assembly Support Pad Welds
- UTS Bolt Locking Devices
- Lower Grid Fuel Assembly Support Pad
- Lower Grid Fuel Assembly Support Pad-to Rib Section Welds
- Lower Grid Fuel Assembly Support Pad Alloy X-750 Dowel
- Lower Grid Fuel Assembly Support Pad Cap Screws and their Locking Welds
- Alloy X-750 Dowel-to-Lower Grid Fuel Assembly Support Pad Welds
- LTS Studs/Nuts Locking Devices

MRP-227-A lists the following ONS "Expansion" RV Internals 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 Bolt Locking Devices, including Locking Welds
- Core Barrel-to-Former Bolt Locking Devices, including Locking Welds

MRP-227-A lists the following ONS "Expansion" RV Internals components as "an acceptable examination technique currently not available":

- Baffle-to-Baffle Bolts (Internal)

5.3.2 Conclusion

The program to manage the aging of the ONS RV Internals includes the components monitored/inspected and their associated guidance as contained in Tables 4-1 and 4-4 of MRP-227-A. The program to manage the aging of the ONS RV Internals also contains the ASME B&PV Code Section XI Examination Category B-N-3 inspections and the vent valve testing and inspection program. The intent of the third GALL program element from Chapter XI, AMP XI.M16A from NUREG-1801, Rev. 2 is met as described in Section 5.3.1 of this report.

5.4 AMP Element 4 – Detection of Aging Effects

"The detection of aging effects is covered in two places: (a) the guidance in Section 4 of MRP-227 provides an introductory discussion and justification of the examination methods selected for detecting the aging effects of interest; and (b) standards for examination methods, procedures, and personnel are provided in a companion document, MRP-228. In all cases, well-established methods were selected. These methods include volumetric UT examination methods for detecting flaws in bolting, physical measurements for detecting changes in dimension, and various visual (VT-3, VT-1, and EVT-1) examinations for detecting effects ranging from general conditions to detection and sizing of surface-breaking discontinuities. Surface examinations may also be used as an alternative to visual examinations for detection and sizing of surface-breaking discontinuities.

Cracking caused by SCC, IASCC, and fatigue is monitored/inspected by either VT-1 or EVT-1 examination (for internals other than bolting) or by volumetric UT examination (bolting). The VT-3 visual methods may be applied for the detection of cracking only when the flaw tolerance of the component or affected assembly, as evaluated for reduced fracture toughness properties, is known and has been shown to be tolerant of easily detected large flaws, even under reduced fracture toughness conditions. In addition, VT-3 examinations are used to monitor/inspect for loss of material induced by wear and for general aging conditions, such as gross distortion caused by void swelling and irradiation growth or by gross effects of loss of preload caused by thermal and irradiation-enhanced stress relaxation and creep.

In addition, the program adopts the recommended guidance in MRP-227 for defining the Expansion criteria that need to be applied to inspections of Primary Components and Expansion Components. As a result, inspections performed on the RVI components are performed consistent with the inspection frequency and sampling bases for Primary Components, Existing Requirement Components, and Expansion Components in MRP-227, which have been demonstrated to be in conformance with the inspection criteria, sampling basis criteria, and sample Expansion criteria in Section A.1.2.3.4 of NRC Branch Position RLSB-1.

Specifically, the program implements the parameters monitored/inspected criteria and bases for inspecting the relevant parameter conditions for B&W designed Primary Components in Table 4-1 of MRP-227 and for B&W designed Expansion Components in Table 4-4 of MRP-227.

The program is supplemented by the following plant-specific Primary Component and Expansion Component inspections for the program (as applicable): for the ONS program, there are no supplemental Primary or Expansion components.

In addition, in some cases (as defined in MRP-227), physical measurements are used as supplemental techniques to manage for the gross effects of wear, loss of preload due to stress relaxation, or for changes in dimension due to void swelling, deflection or distortion. The physical measurements methods applied in accordance with this program include the one-time measurement of the differential height of the top of the plenum rib pads to RV seating surface, with plenum in the RV."

5.4.1 ONS Detection of Aging Effects

The purpose of the augmented MRP-227-A "Primary" and "Expansion" inspections is to detect aging effects of the eight aging degradation mechanisms outlined in Section 2.3 of this report. Physical measurements, VT-3.

visual inspections, and UT inspections are the methods by which the effects of aging are detected in the ONS RV Internals inspection plan.

Physical measurements are used to detect gross effects of wear (i.e., loss of material). A one-time physical differential height measurement of the plenum rib pad to RV seating surface at all three ONS units was conducted as described in Section 4.2.8 of this report.

VT-3 inspections are used to detect cracking caused by SCC, IASCC, and fatigue and loss of material induced by wear and general aging conditions such as gross distortion caused by void swelling and irradiation growth or by gross effects of loss of preload caused by thermal and irradiation-enhanced stress relaxation and creep. VT-3 inspections are also used to detect the effects of thermal and irradiation embrittlement through observations of cracking and/or fracture in these items. Where fitting, additional evaluations are being performed to demonstrate why the examination method, schedule, frequency, and coverage are appropriate.

Cracking caused by SCC, IASCC, and fatigue is monitored/inspected by volumetric UT examination for bolting. As described in Section 4.2.7 of this report, some UT examinations of MRP-227-A components have been performed.

Technical Justifications in accordance with ASME B&PV Code Section V, Article 14 are being prepared for each component receiving an augmented UT and VT-3 examination. Appendix D of this report is provided as a representation of what will be contained in the TJs for the ONS units.

As required by the SER for MRP-227, Rev. 0 and described in Section 6.5 of this report, the inaccessible and uninspectable components will be addressed.

5.4.2 Conclusion

The intent of the fourth GALL program element from Chapter XI, AMP XI.M16A from NUREG-1801, Rev. 2 is met as described in Section 5.4.1 of this report with the clarification that, where fitting, additional evaluations are being performed to demonstrate why the examination methods are appropriate.

5.5 AMP Element 5 – Monitoring and Trending

"The methods for monitoring, recording, evaluating, and trending the data that result from the program's inspections are given in Section 6 of MRP-227 and its subsections. The evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications. The examinations and re-examinations required by the MRP-227 guidance, together with the requirements specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide timely detection, reporting, and corrective actions with respect to the effects of the age-related degradation mechanisms within the scope of the program. The extent of the examinations, beginning with the sample of susceptible PWR internals component locations identified as Primary Component locations, with the potential for inclusion of Expansion Component locations if the effects are greater than anticipated, plus the continuation of the Existing Programs activities, such as the ASME Code, Section XI, Examination Category B-N-3 examinations for core support structures, provides a high degree of confidence in the total program."

5.5.1 ONS Monitoring and Trending

Timely detection of aging effects is ensured by ONS's OE program (see Section 5.10) and the frequency of ASME B&PV Code Section XI Examination Category B-N-3 inspections and the augmented inspections detailed in MRP-227-A.

WCAP-17096, developed under the PWROG and submitted to the NRC, will provide direction on how evaluations of indications will be dispositioned after an SER and approved version are issued. Duke Energy will follow the NRC-accepted methodologies for evaluations of indications.

The summary of the results of the industry MRP-227-A augmented inspections will be compiled into an overall industry report which will track industry progress, aid in evaluation of significant issues,

identification of fleet trends and determination of any needed revisions to these guidelines. The industry report will be updated approximately every two years for the benefit of the fleet, the NRC, the PWROG and other industry stakeholders. This biennial report will serve to assist in review of OE and required monitoring and trending for AMPs established by the industry. In order to ensure completeness and consistency of reporting, EPRI MRP will provide a template listing the requested information. Duke Energy will continue to participate in industry programs as well as provide results as required by MRP-227-A to assist the industry in monitoring and trending.

5.5.2 Conclusion

The intent of the fifth GALL program element from Chapter XI, AMP XI.M16A from NUREG-1801, Rev. 2 is met as described in Section 5.5.1 of this report

5.6 AMP Element 6 – Acceptance Criteria

"Section 5 of MRP-227 provides specific examination acceptance criteria for the Primary and Expansion Component examinations. For components addressed by examinations referenced to ASME Code, Section XI, the IWB-3500 acceptance criteria apply. For other components covered by Existing Programs, the examination acceptance criteria are described within the Existing Program reference document.

The guidance in MRP-227 contains three types of examination acceptance criteria:

- *For visual examination (and surface examination as an alternative to visual examination), the examination acceptance criterion is the absence of any of the specific, descriptive relevant conditions; in addition, there are requirements to record and disposition surface breaking indications that are detected and sized for length by VT-1/EVT-1 examinations;*
- *For volumetric examination, the examination acceptance criterion is the capability for reliable detection of indications in bolting, as demonstrated in the examination Technical Justification; in addition, there are requirements for system-level assessment of bolted or pinned assemblies with unacceptable volumetric (UT) examination indications that exceed specified limits; and*
- *For physical measurements, the examination acceptance criterion for the acceptable tolerance in the measured differential height from the top of the plenum rib pads to the vessel seating surface in B&W plants are given in Table 5-1 of MRP-227. The acceptance criterion for physical measurements performed on the height limits of the Westinghouse-designed hold-down springs are a license renewal applicant action item for Westinghouse PWR applicants only."*

5.6.1 ONS Acceptance Criteria

5.6.1.1 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-A provides the examination acceptance criteria for the "Primary" and "Expansion" components; Table 5-1 of MRP-227-A is provided in Appendix C 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, relevant conditions for the visual (VT-3) examinations; and
- Acceptance criteria established as part of the TJ for UT examinations.

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.

5.6.1.2 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-A inspections, the results of the PWROG projects discussed in Section 4.1.2 of this report as well as ONS-specific projects as detailed in Section 6.0 of this report 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.

5.6.2 Conclusion

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. The intent of the sixth GALL program element from Chapter XI, AMP XI.M16A from NUREG-1801, Rev. 2 is met as described in Section 5.6.1 of this report

5.7 AMP Element 7 – Corrective Actions

"Corrective actions following the detection of unacceptable conditions are fundamentally provided for in each plant's corrective action program. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The disposition will ensure that design basis functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events. Examples of methodologies that can be used to analytically disposition unacceptable conditions are found in the ASME Code, Section XI or in Section 6 of MRP-227. Section 6 of MRP-227 describes the options that are available for disposition of detected conditions that exceed the examination acceptance criteria of Section 5 of the report. These include engineering evaluation methods, as well as supplementary examinations to further characterize the detected condition, or the alternative of component repair and replacement procedures. The latter are subject to the requirements of the ASME Code, Section XI. The implementation of the guidance in MRP-227, plus the implementation of any ASME Code requirements, provides an acceptable level of aging management of safety-related components addressed in accordance with the corrective actions of 10 CFR Part 50, Appendix B or its equivalent, as applicable.

Other alternative corrective action bases may be used to disposition relevant conditions if they have been previously approved or endorsed by the NRC. Examples of previously NRC-endorsed alternative corrective actions bases include those corrective actions bases for Westinghouse-design RVI components that are defined in Tables 4-1, 4-2, 4-3, 4-4, 4-5, 4-6, 4-7 and 4-8 of Westinghouse Report No. WCAP-14577-Rev. 1-A, or for B&W-designed RVI components in B&W Report No. BAW-2248. Westinghouse Report No. WCAP-14577-Rev. 1-A was endorsed for use in an NRC SE to the Westinghouse Owners Group, dated February 10, 2001. B&W Report No. BAW-2248 was endorsed for use in an SE to Framatome Technologies on behalf of the B&W Owners Group, dated

December 9, 1999. Alternative corrective action bases not approved or endorsed by the NRC will be submitted for NRC approval prior to their implementation."

5.7.1 ONS Corrective Actions

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 normal and expected degradation.

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 program is used in conjunction with the ONS PIP program on degraded conditions identified as needing a root cause analysis.

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 program compliments the guidance in the ONS PIP program for the resolution of degraded and/or nonconforming conditions.

5.7.2 Conclusion

Corrective actions, including root cause determination and prevention of recurrence, are timely. The intent of the seventh GALL program element from Chapter XI, AMP XI.M16A from NUREG-1801, Rev. 2 is met as described in Section 5.7.1 of this report

5.8 AMP Element 8 – Confirmation Process

"Site quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. As discussed in the Appendix for GALL, the staff finds the requirements of 10 CFR Part 50, Appendix B acceptable to address the confirmation process and administrative controls."

5.8.1 ONS Confirmation Process

Duke Energy Topical Report Duke-1-A, "Quality Assurance Program" [29] 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) provides support and leadership to the general office and nuclear sites with QA program audits, performance assessment, procurement quality, supplier verification and QA.

5.8.2 Conclusion

Duke Energy's QA program ensures preventative actions are adequate and appropriate corrective actions have been completed and are effective. The intent of the eighth GALL program element from Chapter XI, AMP XI.M16A from NUREG-1801, Rev. 2 is met as described in Section 5.8.1 of this report.

5.9 AMP Element 9 – Administrative Controls

"The administrative controls for such programs, including their implementing procedures and review and approval processes, are under existing site 10 CFR 50 Appendix B Quality Assurance Programs, or their equivalent, as applicable. Such a program is thus expected to be established with a sufficient level of documentation and administrative controls to ensure long-term implementation."

5.9.1 ONS Administrative Controls

The administrative controls for the program to manage the aging of the ONS RV Internals, including their implementing procedures and review and approval processes, are under existing site 10 CFR 50 Appendix B QA programs, or their equivalent, as applicable. Such a program is thus expected to be established with a sufficient level of documentation and administrative controls to ensure effective long-term implementation.

5.9.2 Conclusion

Duke Energy's administrative controls provide for a formal review and approval process. The intent of the ninth GALL program element from Chapter XI, AMP XI.M16A from NUREG-1801, Rev. 2 is met as described in Section 5.9.1 of this report.

5.10 AMP Element 10 – Operating Experience

"Relatively few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. A summary of observations to date is provided in Appendix A of MRP-227-A. The applicant is expected to review subsequent operating experience for impact on its program or to participate in industry initiatives that perform this function.

The application of the MRP-227 guidance will establish a considerable amount of operating experience over the next few years. Section 7 of MRP-227 describes the reporting requirements for these applications, and the plan for evaluating the accumulated additional operating experience."

5.10.1 ONS Operating Experience

Relatively few incidents of PWR RV Internals aging degradation have been reported in operating U.S. commercial PWR plants; a summary of some of the observations can be found in Appendix A of MRP-227-A. Duke Energy will review subsequent OE for impact on its program and will participate in industry initiatives that perform this function.

The application of the MRP-227-A guidance will establish a considerable amount of OE over the next few years at the ONS units and throughout the industry. Section 7 of MRP-227-A describes the reporting requirements for these applications, and the plan for evaluating this additional OE.

The ONS OE 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.

5.10.2 Conclusion

Operating experience has been used in the development of the ONS RV Internals inspection plan and the MRP-227-A guidelines. Augmented inspection results will provide additional OE to the MRP-227-A guidelines. The intent of the tenth GALL program element from Chapter XI, AMP XI.M16A from NUREG-1801, Rev. 2 is met as described in Section 5.10.1 of this report

5.11 Program Conclusion

Section 5.0 of this report shows that the ONS RV Internals inspection plan meets the intent of the ten GALL program elements from Chapter XI, AMP XI.M16A from NUREG-1801, Rev. 2; this demonstrates the adequacy of managing the aging effects of the ONS RV Internals.

6.0 CONDITIONS/LIMITATIONS AND PLANT-SPECIFIC ACTIONS ITEMS CONTAINED IN MRP-227-A

Seven conditions and limitations are placed on the use of MRP-227-A as well as eight licensee/applicant action items that shall be addressed by applicants/licensees who choose to implement MRP-227-A. The following subsections describe those conditions/limitations and plant-specific actions which are applicable to ONS. The applicable conditions/limitations and plant-specific actions are delineated in Table 4-1 of this report and taken from Revision 1 to the SER for MRP-227, Rev. 0.[4]

Table 6-1 of this report gives a summary of the status of the Topic Report Conditions/Limitations and Applicant/Licensee Actions contained in MRP-227-A for the ONS units. Items not applicable to the ONS units are identified as such.

Table 6-1. Summary of Status of Topical Report Conditions/Limitations and Plant-Specific Actions Contained in MRP-227-A for ONS

Item	Inspection Plan Location	Status in this Report
Topical Report Condition/Limitation 1	Not Applicable to ONS units	Not Applicable to ONS
Topical Report Condition/Limitation 2	Not Applicable to ONS units	Not Applicable to ONS
Topical Report Condition/Limitation 3	Section 6.1	Fulfilled for the ONS units by publication of MRP-227-A
Topical Report Condition/Limitation 4	Section 6.1	Fulfilled for the ONS units by publication of MRP-227-A
Topical Report Condition/Limitation 5	Section 6.1	Fulfilled for the ONS units by publication of MRP-227-A
Topical Report Condition/Limitation 6	Section 6.1	Fulfilled for the ONS units by publication of MRP-227-A
Topical Report Condition/Limitation 7	Not Applicable to ONS units	Not Applicable to ONS
Applicant/Licensee Action Item 1	Section 6.2	Fulfilled for the ONS units by evaluation within this report
Applicant/Licensee Action Item 2	Section 6.3	A schedule of when this information will be submitted to the NRC will be submitted to the NRC by July 1, 2012.
Applicant/Licensee Action Item 3	Not Applicable to ONS units	Not Applicable to ONS
Applicant/Licensee Action Item 4	Section 6.4	A schedule of when this information will be submitted to the NRC will be submitted to the NRC by July 1, 2012.
Applicant/Licensee Action Item 5	Not Applicable to ONS units	Not Applicable to ONS
Applicant/Licensee Action Item 6	Section 6.5	A schedule of when this information will be submitted to the NRC will be submitted to the NRC by July 1, 2012.
Applicant/Licensee Action Item 7	Section 6.6	A schedule of when this information will be submitted to the NRC will be submitted to the NRC by July 1, 2012.
Applicant/Licensee Action Item 8	Section 6.7	Fulfilled for the ONS units by this report

6.1 Topical Report Conditions/Limitations Applicable to the ONS Units

Topical Report Conditions/Limitations applicable to the ONS units are identified in Table 6-1 of this report. These topical report conditions/limitations are fulfilled for the ONS units by making the required changes prior to the publication of MRP-227-A.

6.2 Applicant/Licensee Action Item 1

This action item is described in Section 3.2.5.1 and Section 4.2.1 in Revision 1 to the SER for MRP-227, Rev. 0 and summarized within this section.

6.2.1 Discussion of Requirement

As addressed in Section 3.2.5.1 of the SER, each applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RV Internals components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227.

6.2.2 Resolution

The FMECA for the B&W plants is performed in MRP-190[30] and contains six assumptions and observations in Section 4 of the report. The assumptions are either bounding or methodological, and, as described in MRP-190, do not require plant-specific verification for the three ONS units.

The functionality analysis for the B&W plants is performed in MRP-229[31] and contains eight limitations and assumptions in Section 2.4.1. Seven of the limitations and assumptions do not relate to the plant design or operating history at the ONS units. A discussion of the assessment of the planned MUR power uprate at the three ONS units is given in Section 4.1.1.1.2 of this report.

A discussion of MRP-227-A applicability to ONS is in Section 4.1.1.1.2 of this report.

Therefore, this application/licensee action item is considered fulfilled for the ONS units.

6.3 Applicant/Licensee Action Item 2

This action item is described in Section 3.2.5.2 and Section 4.2.2 in Revision 1 to the SER for MRP-227, Rev. 0 and summarized within this section.

6.3.1 Discussion of Requirement

As discussed in Section 3.2.5.2 of the SER, consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RV Internals components are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RV Internals components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RV Internals components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by the SE, when submitting its plant-specific AMP. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the period of extended operation.

6.3.2 Resolution

A comparison between the RV Internals components within the scope of LR for ONS and the information in Tables 4-1 and 4-2 in MRP-189, Revision 1[32] will be performed and any necessary modification to the program

will be made based on the results of the comparison. A schedule of when this information will be submitted to the NRC will be submitted by July 1, 2012.

6.4 Applicant/Licensee Action Item 4

This action item is described in Section 3.2.5.4 and Section 4.2.4 in Revision 1 to the SER for MRP-227, Rev. 0 and summarized within this section.

6.4.1 Discussion of Requirement

As discussed in Section 3.2.5.4 of the SER, the B&W applicants/licensees shall confirm that the core support structure upper flange weld was stress relieved during the original fabrication of the RV in order to confirm the applicability of MRP-227, as approved by the NRC, to their facility. If the upper flange weld has not been stress relieved, then this component shall be inspected as a "Primary" examination category component. If necessary, the examination methods and frequency for non-stress relieved B&W core support structure upper flange welds shall be consistent with the recommendations in MRP-227, as approved by the NRC, for the Westinghouse and CE upper core support barrel welds. The examination coverage for this B&W flange weld shall conform to the staff's imposed criteria as described in Sections 3.3.1 and 4.3.1 of the SER. The applicant's/licensee's resolution of this plant-specific action item shall be submitted to the NRC for review and approval.

6.4.2 Resolution

A records search to confirm that the core support structure upper flange weld was stress relieved during original fabrication of the RV will be performed and documented. If confirmation cannot be made, the examination methods and frequency for the potentially non-stress relieved ONS core support structure upper flange welds will be consistent with the recommendations in MRP-227, as approved by the NRC, for the Westinghouse and CE upper core support barrel welds. A schedule of when this information will be submitted to the NRC will be submitted by July 1, 2012.

6.5 Applicant/Licensee Action Item 6

This action item is described in Section 3.3.6 and Section 4.2.6 in Revision 1 to the SER for MRP-227, Rev. 0 and summarized within this section.

6.5.1 Discussion of Requirement

As addressed in Section 3.3.6 in the SER, MRP-227 does not propose to inspect the following inaccessible components: the B&W core barrel cylinders (including vertical and circumferential seam welds), B&W former plates, B&W external baffle-to-baffle bolts and their locking devices, B&W core barrel-to-former bolts and their locking devices, and B&W core barrel assembly internal baffle-to-baffle bolts. The MRP also identified that although the B&W core barrel assembly internal baffle-to-baffle bolts are accessible, the bolts are non-inspectable using currently available examination techniques. Applicants/licensees shall justify the acceptability of these components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components. As part of their application to implement the approved version of MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible components and, if necessary, provide their plan for the replacement of the components for NRC review and approval.

6.5.2 Resolution

ONS will justify the acceptability of inaccessible and non-inspectable components (core barrel cylinder including vertical and circumferential seam welds, former plates, external baffle-to-baffle bolts and their locking devices, core barrel-to-former bolts and their locking devices, and internal baffle-to-baffle bolts) for continued operation through the period of extended operation by performing an evaluation or by proposing a schedule for replacement

of the components. A schedule of when this information will be submitted to the NRC will be submitted by July 1, 2012.

6.6 Applicant/Licensee Action Item 7

This action item is described in Section 3.3.7 and Section 4.2.7 in Revision 1 to the SER for MRP-227, Rev. 0 and summarized within this section.

6.6.1 Discussion of Requirement

As discussed in Section 3.3.7 of the SER, the applicants/licensees of B&W reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI guide tube assembly spiders and CRGT spacer castings will maintain their functionality during the period of extended operation or for additional RV Internals components that may be fabricated from CASS, martensitic stainless steel, or precipitation-hardened stainless steel materials. These analyses shall also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The requirement may not apply to components that were previously evaluated as not requiring aging management during development of MRP-227. That is, the requirement would apply to components fabricated from susceptible materials for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/Licensee Action Item 2. The plant-specific analysis shall be consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. The applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227.

6.6.2 Resolution

ONS will develop a plant-specific analysis to demonstrate that the IMI guide tube assembly spiders, CRGT spacer castings, and additional RV Internals components that may be fabricated from CASS, martensitic stainless steel, or precipitation hardened stainless steel materials (e.g., CSS vent valve top and bottom retaining rings) will maintain their functionality during the period of extended operation. The analysis will consider the possible loss of fracture toughness in these components due to thermal embrittlement (TE) and/or IE and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The ONS-specific analysis will be consistent with ONS's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. A schedule of when this information will be submitted to the NRC will be submitted by July 1, 2012.

6.7 Applicant/Licensee Action Item 8

This action item is described in Section 3.5.1 and Section 4.2.8 in Revision 1 to the SER for MRP-227, Rev. 0 and summarized within this section.

6.7.1 Discussion of Requirement

As addressed in Section 3.5.1 in the SER, applicants/licensees shall make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by the SER, as an AMP for the RV Internals components at their facility. This submittal shall include the information identified in Section 3.5.1 of this SE.

6.7.2 Resolution

As described in Section 3.5.1 of the SER, since ONS's licensing basis contains a commitment to submit a PWR RV Internals AMP and/or inspection program, as discussed in Section 2.1 of this report, Duke is making a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by the SER. Items 1 and 2 in Section 3.5.1 of the SER are included in this inspection plan. Items 3, 4, and 5 from Section 3.5.1 of the SER are not required for the ONS units because the LAR was submitted prior to issuance of the SER.

1. An AMP for the facility that addresses the ten program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A.

   The ten program elements as defined in NUREG-1801, Revision 2 are addressed in Section 5.0 of this report.

2. To ensure the MRP-227 program and the plant-specific action items will be carried out by applicants/licensees, applicants/licensees are to submit an inspection plan which addresses the identified plant-specific action items for staff review and approval consistent with the licensing basis for the plant. If an applicant/licensee plans to implement an AMP which deviates from the guidance provided in MRP-227, as approved by the NRC, the applicant/licensee shall identify where their program deviates from the recommendations of MRP-227, as approved by the NRC, and shall provide a justification for any deviation which includes a consideration of how the deviation affects both "Primary" and "Expansion" examination category components.

   The ONS RV Internals inspection plan is contained within this report. The plant-specific action items are addressed in Section 6.0 of this report as described in Table 4-1 of this report.

Therefore, this application/licensee action item is considered fulfilled for the ONS units.

7.0 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-A. Section 5.0 of this report demonstrates that the ONS RV Internals AMP, which contains the ONS RV Internals inspection plan, will meet the intent of the ten AMP elements described in NUREG-1801, Rev. 2, Chapter XI, AMP XI.M16A. Section 6.0 of this report has demonstrated compliance with the topical report conditions/limitations and application/licensee plant-specific action items from MRP-227-A as described.

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 and MRP-227-A 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 LR period of extended operation.

This ONS RV Internals inspection plan, along with the ONS RV Internals AMP, fulfills the approved LR 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.

The ONS UFSAR will be updated as required.

8.0 REFERENCES

1. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), EPRI, Palo Alto, CA: 2011. 1022863.

2. NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," Dated December 2010.

3. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227, Rev. 0), EPRI, Palo Alto, CA: 2008. 1016596.

4. Final Safety Evaluation of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (TAC No. ME0680), NRC Accession No. ML111600498, June 22, 2011.

5. 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

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

7. 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

8. SECY-00-0081, "Oconee Nuclear Station, Units 1, 2, and 3 – Renewal of Full-Power Operating License," April 10, 2000. NRC Accession No.: ML003693303

9. Materials Reliability Program: Thermal Aging and Neutron Embrittlement Assessment of Cast Austenitic Stainless Steels and Stainless Steel Welds in PWR Internals (MRP-276). EPRI, Palo Alto, CA: 2010. 1020959.

10. Letter (from T. Preston Gillespie, Jr.) "Duke Energy Carolinas, LLC, Oconee Nuclear Station, Units 1, 2, and 3, Docket Numbers 50-269, 50-270, and 50-287, License Renewal Commitment to Submit a Time Limited Aging Analysis for the Reactor Vessel Internals to the NRC for Review," February 20, 2012. NRC Accession No. ML12053A332.

11. BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," July 1997.

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13. BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," April 2000. NRC Accession No.: ML003708443

14. Letter (from Christian Larsen) Report Transmittal: Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0), EPRI Palo Alto, CA, 2008, 1016596 (NRC Accession No. ML090160204), January 2009.

15. 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.

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

17. Letter (from T. Preston Gillespie, Jr.) "Commitment Date Change for Submittal of License Amendment to Adopt Material Reliability Program 227, Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," October 20, 2010.

18. Letter (from T. Preston Gillespie, Jr.) "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," November 8, 2010, NRC Accession No. ML103140599.

19. Regulatory Issue Summary 2011-07, "License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management," July 21, 2011.

20. Letter (from T. Preston Gillespie) "Reactor Internals License Amendment Request Status Update, License Amendment Request (LAR) No. 2010-06," September 1, 2011, NRC Accession No. ML11251A160.

21. Revision 1 to the Final Safety Evaluation of Electric Power Research Institute (EPRI) Report, Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines" (TAC No. ME0680), NRC Accession No. ML11308A770, December 16, 2011.

22. Nuclear Energy Institute, "Guideline for the Management of Materials Issues," NEI 03-08, Revision 2, January 2010. NRC Accession No. ML102880028.

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

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25. 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.

26. 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

27. J.B. Hall, S. Fyfitch, and K.E. Moore, "Laboratory and Operating Experience with Alloy A-286 and Alloy X-750 RV Internals Bolting Stress Corrosion Cracking," 11th International Conference on Environmental Degradation of Materials in Nuclear Systems, Stevenson, WA, August 10-14, 2003.

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

29. Duke Energy Carolinas Topical Report, Duke-1-A, "Quality Assurance Program," Amendment 39, October 2011. NRC Accession No. ML11304A135.

30. Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals (MRP-190). EPRI, Palo Alto, CA: 2006. 1013233.

31. Materials Reliability Program: Functionality Analysis for B&W Representative PWR Internals (MRP-229-Rev. 3). EPRI, Palo Alto, CA: 2010. 1022402.

32. Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189-Rev. 1). EPRI, Palo Alto, CA: 2009. 1018292.

APPENDIX A: TABLE 4-1 FROM MRP-227-A

Appendix A contains Table 4-1, "B&W Plants Primary Components" from MRP-227-A. The "Primary" component examinations listed in the ONS RV Internals inspection plan are based on this table.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	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	One-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. See Figure 4-1.
Control Rod Guide Tube Assembly CRGT spacer castings	All plants	Cracking (TE), including the detection of fractured spacers or missing screws	None	Visual (VT-3) examination during the next 10-year ISI. Subsequent examinations on the 10-year ISI interval.	Accessible surfaces at each of the 4 screw locations (at every 90°) of 100% of the CRGT spacer castings (limited accessibility). See Figure 4-5.
Core Support Shield Assembly CSS vent valve top retaining ring CSS vent valve bottom retaining ring (Note 1)	All plants	Cracking (TE), including the detection of surface irregularities, such as damaged, fractured material, 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 (see BAW-2248A, page 4.3 and Table 4-1). See Figure 4-11.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Core Support Shield Assembly Upper core barrel (UCB) bolts and their locking devices	All plants	Bolts: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UTS bolts and LTS studs/nuts or bolts and their locking devices SSHT studs/nuts or bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (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 on the 10-year ISI interval unless an evaluation of the baseline results submitted for NRC staff approval justifies a longer interval between examinations. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts and their locking devices. (Note 3) See Figure 4-7.
Core Barrel Assembly Lower core barrel (LCB) bolts and their locking devices	All plants	Bolt: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UTS bolts and LTS studs/nuts or bolts and their locking devices SSHT studs/nuts or bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	Volumetric examination (UT) of the bolts during the next 10-year ISI interval from 1/1/2006. Subsequent examination on the 10-year ISI interval unless an evaluation of the baseline results submitted for NRC staff approval justifies a longer interval between examinations. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts and their locking devices (Note 3) See Figure 4-8.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Core Barrel Assembly Baffle-to-former bolts	All plants	Cracking (IASCC, IE, Overload) (Note 4)	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 additional years.	100% of accessible bolts. (Note 3) See Figure 4-2.
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. See Figure 4-2.
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, including locking welds, for the external baffle-to-baffle bolts and Core barrel-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. (Note 3) See Figure 4-2.

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 2)	Examination Method/Frequency (Note 2)	Examination Coverage
Flow Distributor Assembly Flow distributor (FD) bolts and their locking devices	All plants	Bolt: Cracking (SCC) Locking Devices: Loss of material, damaged or distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UTS bolts and LTS studs/nuts or bolts and their locking devices. SSHT studs/nuts or bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	Volumetric examination (UT) of the bolts during the next 10-year ISI interval from 1/1/2006. Subsequent examination on the 10-year ISI interval unless an evaluation of the baseline results, submitted for NRC staff approval, justifies a longer interval between examinations. Visual (VT-3) examination of bolt locking devices on the 10-year ISI interval.	100% of accessible bolts and their locking devices. (Note 3) See Figure 4-8.
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 grid fuel assembly support pads.	Initial visual (VT-3) examination no later than two refueling outages from the beginning of the license renewal period. Subsequent examinations on the 10-year ISI interval.	Accessible surfaces of 100% of the 24 dowel-to-guide block welds. See Figure 4-4.
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, Cracking (IE), including separation of spider arms from the lower grid rib section at the weld	Lower grid 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 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 the 10-year ISI interval.	100% of top surfaces of 52 spider castings and welds to the adjacent lower grid rib section. See Figures 4-3 and 4-6.

Notes to Table 4-1 [of MRP-227-A}:

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, leakage 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 inservice test programs (see BAW-2248A, page 4-3 and Table 4-1[18]).
2. Examination acceptance criteria and expansion criteria for the B&W components are in Table 5-1 [of MRP-227-A}.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 5-1 [of MRP-227-A}, must be examined for inspection credit.
4. The primary aging degradation mechanisms for loss of joint tightness for this item are IC and ISR. Fatigue and Wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to IC/ISR. Bolt stress relaxation cannot readily be inspected by NDE. Only bolt cracking is inspected by UT inspection. The effect of loss of joint tightness on the functionality will be addressed by analysis of the core barrel assembly, which will be performed to address Applicant/Licensee action item 6 in the SE [27].

APPENDIX B: TABLE 4-4 FROM MRP-227-A

Appendix B contains Table 4-4, "B&W Plants Expansion Components" from MRP-227-A. The "Expansion" component examinations listed in the ONS RV Internals inspection plan are based on this table.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Upper Grid Assembly Alloy X-750 dowel-to-upper grid 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. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of 100% of the dowel locking welds. See Figure 4-6 (i.e., these are similar to the lower grid fuel assembly support pads).
Core Barrel Assembly Upper thermal shield (UTS) bolts and their locking devices	All plants	Bolt or Stud/Nut: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UCB, LCB or FD bolts and their locking devices	Bolt or Stud/Nut: Volumetric examination (UT). Locking Devices: Visual (VT-3) examination Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible bolts or studs/nuts and their locking devices (Note 2). See Figure 4-7.
Core Barrel Assembly Surveillance specimen holder tube (SSHT) studs/nuts (CR-3) or bolts (DB) and their locking devices	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	No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Barrel Assembly Baffle-to-baffle bolts Core barrel-to-former bolts	All plants	Cracking (IASCC, IE, Overload) (Note 3)	Baffle-to-former bolts	Internal baffle-to-baffle bolts: No examination requirements, Justify by evaluation or by replacement.	An acceptable examination technique currently not available. See Figure 4-2.
				External baffle-to-baffle bolts, core barrel-to-former bolts: No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.
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	No examination requirements. Justify by evaluation or by replacement.	Inaccessible. See Figure 4-2.
Lower Grid Assembly Lower grid 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 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 of the support pads	IMI guide tube spiders and spider-to-lower grid rib section welds	Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of the pads, dowels, and cap screws, and associated welds in 100% of the lower grid fuel assembly support pads. See Figure 4-6.

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Lower Grid Assembly Alloy X-750 dowel-to-lower grid 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. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	Accessible surfaces of 100% of the support pad dowel locking welds. See Figure 4-6.
Lower Grid Assembly Lower grid shock pad bolts and their locking devices	TMI-1	Bolts: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UCB, LCB or FD bolts and their locking devices	Bolt: Volumetric examination (UT). Locking Devices: Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible bolts and their locking devices. (Note 2) See Figure 4-4.
Lower Grid Assembly Lower thermal shield (LTS) bolts (ANO-1, DB and TMI-1) or studs/nuts (ONS, CR-3) and their locking devices	All plants	Bolts or Studs/Nuts: Cracking (SCC) Locking Devices: Loss of material, damaged, distorted or missing locking devices (Wear or Fatigue damage by failed bolts).	UCB,LCB or FD bolts and their locking devices	Bolt or Stud/Nut: Volumetric examination (UT). Locking Devices: Visual (VT-3) examination. Subsequent examinations on the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections.	100% of accessible bolts and their locking devices. (Note 2) See Figure 4-8.

Notes to Table 4-4 [of MRP-227-A]:

1. Examination acceptance criteria and expansion criteria for the B&W components are in Table 5-1 [of MRP-227-A}.
2. A minimum of 75% of the total population (examined + unexamined) must be examined for inspection credit.
3. The primary aging degradation mechanisms for loss of joint tightness for these items are IC and ISR. Fatigue and Wear, which can also lead to cracking, are secondary aging degradation mechanisms after significant stress relaxation and loss of preload has occurred due to IC/ISR. Bolt stress relaxation cannot readily be inspected by NDE. Only bolt cracking could be inspected by UT inspection if it were possible for these bolts. Therefore, the effects of loss of joint tightness and/or cracking on the functionality of these bolts relative to the entire core barrel assembly will be addressed by analysis of the core barrel assembly, which will be performed to address Applicant/Licensee action item 6 of the SE [27 of MRP-227-A].

APPENDIX C: TABLE 5-1 FROM MRP-227-A

Appendix C contains Table 5-1, "B&W Plants Examination Acceptance and Expansion Criteria" from MRP-227-A. The inspections listed in the ONS RV Internals inspection plan will use the acceptance and expansion criteria in this table.

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 vent valve top retaining ring CSS vent valve bottom retaining ring	All plants	Visual (VT-3) examination. The specific relevant condition is evidence of damaged or fractured retaining ring material, and missing items.	None	N/A	N/A
Control Rod Guide Tube Assembly CRGT spacer castings	All plants	The specific relevant condition for the VT-3 of the CRGT spacer castings is evidence of fractured spacers or missing screws.	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 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.	UTS bolts and LTS bolts or studs/nuts and their locking devices SSHT studs/nuts or bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (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 bolts and 100% of the accessible LTS bolts or studs/nuts, *Additionally for TMI-1* 100% of the accessible lower grid shock pad bolts, *Additionally for CR-3 and DB* 100% of the accessible SSHT studs/nuts or 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 bolt and 100% of the accessible LTS bolt or stud/nut 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 or stud/nut locking devices.	1) The examination 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 VT-3 of the expansion 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 bolts and LTS bolts or studs/nuts and their locking devices SSHT studs/nuts or bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	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 bolts and 100% of the accessible LTS bolts or studs/nuts *Additionally for TMI-1* 100% of the accessible lower grid shock pad bolts, *Additionally for CR-3 and DB* 100% of the accessible SSHT studs/nuts or 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 bolts and 100% of the accessible LTS bolt or stud/nut 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 stud/nut or bolt locking devices.	1) The examination 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 VT-3 of the expansion 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 baffle 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

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
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, including locking welds.	Locking devices, including locking welds, for the external baffle-to-baffle bolts and core 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, including locking welds, shall require an 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
Lower Grid Assembly Alloy X-750 dowel-to-guide block welds	All plants	Initial visual (VT-3) examination. The specific relevant condition is separated or missing locking weld, or missing dowel.	Alloy X-750 dowel locking welds to the upper and lower grid 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 grid fuel assembly support pads by the completion of the next refueling outage.	The specific relevant condition for the VT-3 of the expansion dowel locking weld is separated or missing locking weld, or missing dowel.

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Flow Distributor Assembly Flow distributor (FD) bolts and their locking devices	All plants	1) Volumetric (UT) examination of the FD bolts. The examination acceptance criteria for the UT of the FD bolts shall be established as part of the examination technical justification. 2) Visual (VT-3) examination of the FD bolt locking devices. The specific relevant condition for the VT-3 of the FD bolt locking devices is evidence of broken or missing bolt locking devices.	UTS bolts and LTS bolts or studs/nuts and their locking devices SSHT studs/nuts or bolts and their locking devices (CR-3 and DB only) Lower grid shock pad bolts and their locking devices (TMI-1 only)	1) Confirmed unacceptable indications exceeding 10% of the FD 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 bolts and 100% of the accessible LTS bolts or studs/nuts *Additionally for TMI-1* 100% of the accessible lower grid shock pad bolts, *Additionally for CR-3 and DB* 100% of the accessible SSHT studs/nuts or bolts. 2) Confirmed evidence of relevant conditions exceeding 10% of the FD bolt locking devices shall require tha the VT-3 examination be expanded by the completion of the next refueling outage to include: *For all plants* 100% of the accessible UTS bolts and 100% of the accessible LTS bol or stud/nut 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 stud/nut or bolt locking devices.	1) The examination 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 VT-3 of the expansion 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
Incore Monitoring Instrumentation (IMI) Guide Tube Assembly IMI guide tube spiders IMI guide tube spider-to-lower grid rib 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.	Lower fuel grid assembly support pad items: pad, pad-to-rib section welds, Alloy X-750 dowel, cap screw, and their locking welds	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.	The specific relevant conditions for the VT-3 of the lower grid fuel assembly support pad items (pads, pad-to-rib section welds, Alloy X-750 dowels, cap screws, and their locking welds) are separated or missing welds, missing support pads, dowels, cap screws and locking welds, or misalignment of the support pads.

Notes to Table 5-1 [of MPR-227-A]:

1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).

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

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

20004-018 (10/18/2010)

AREVA

AREVA NP Inc.

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

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.[Reference 1] Subarticle 1441 requires that a Technical Justification (TJ) be prepared prior to Nondestructive Examinations (NDE).

The purpose of this document is to provide a detailed explanation of the examination process, including the theo 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, an 3 reactor vessel internals.

This document is the deliverable for Task 2.1, "2011-2012 MRP-227 AREVA Engineering Work for Oconee," i AREVA NP's proposal 2010002104 to Duke Energy (AREVA contract number L500096).

2.0 DESCRIPTION OF THE COMPONENT/FLAWS TO BE EXAMINED

2.1 Upper and Lower Core Barrel Bolting Design

The five high-strength bolting joint locations common to 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 high-strength 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. The hardware covered by this TJ is limite to the examination of the upper and lower core barrel bolts. 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 high-strength bolting 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]).

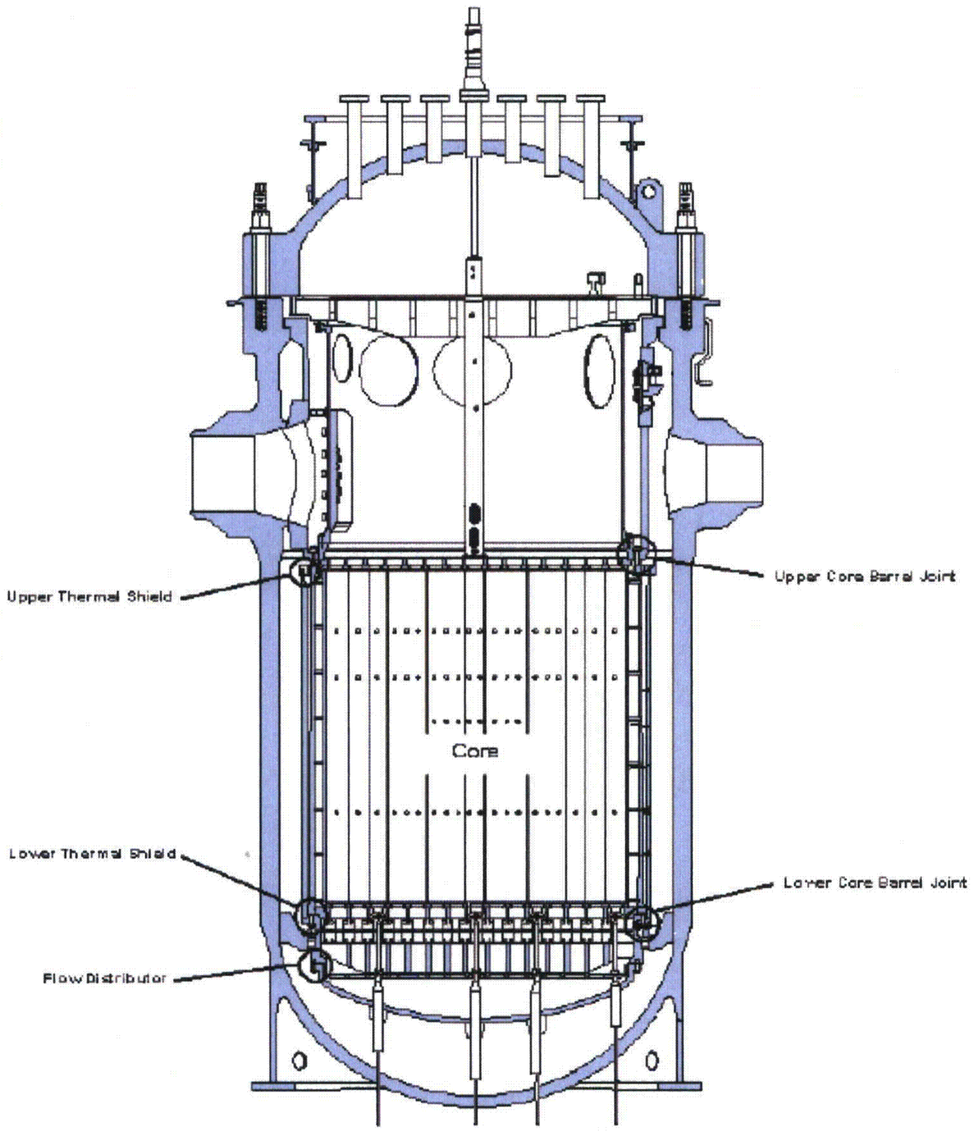

Figure 2-1: High-Strength Bolting Locations in B&W-Design Reactor Vessel Internals

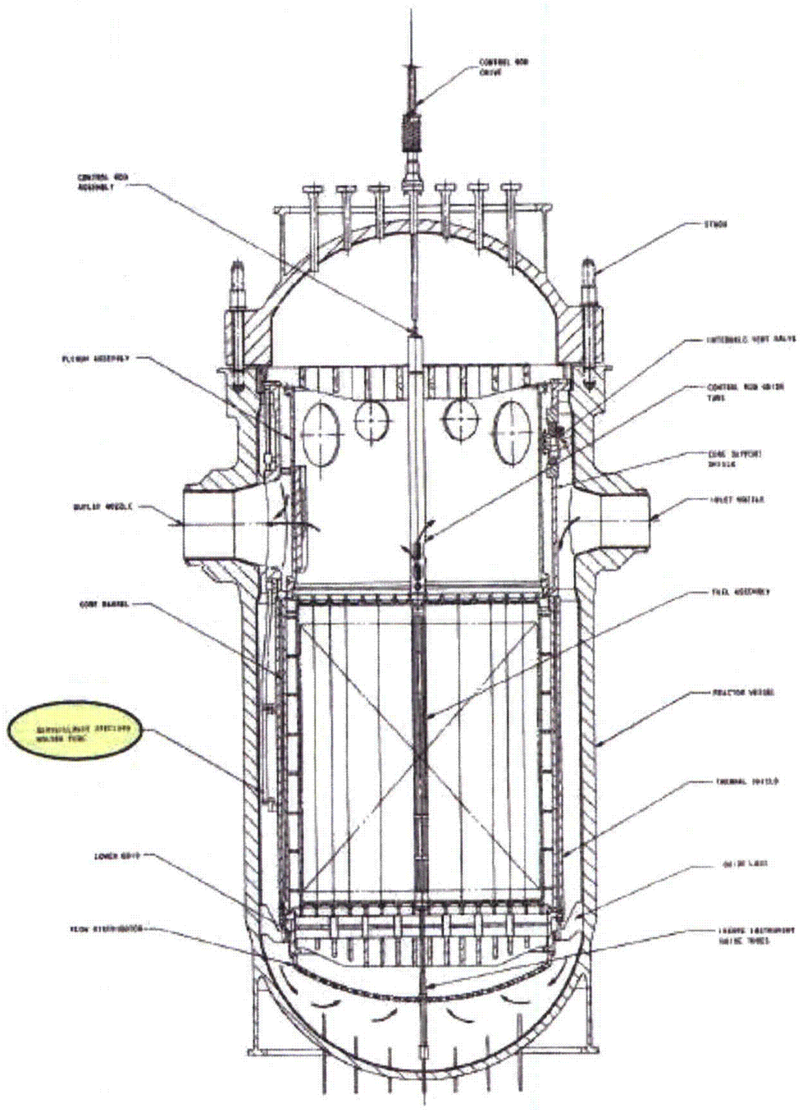

Figure 2-2: Location of Original Surveillance Specimen Holder Tube in B&W-Design Reactor Vessel Internals

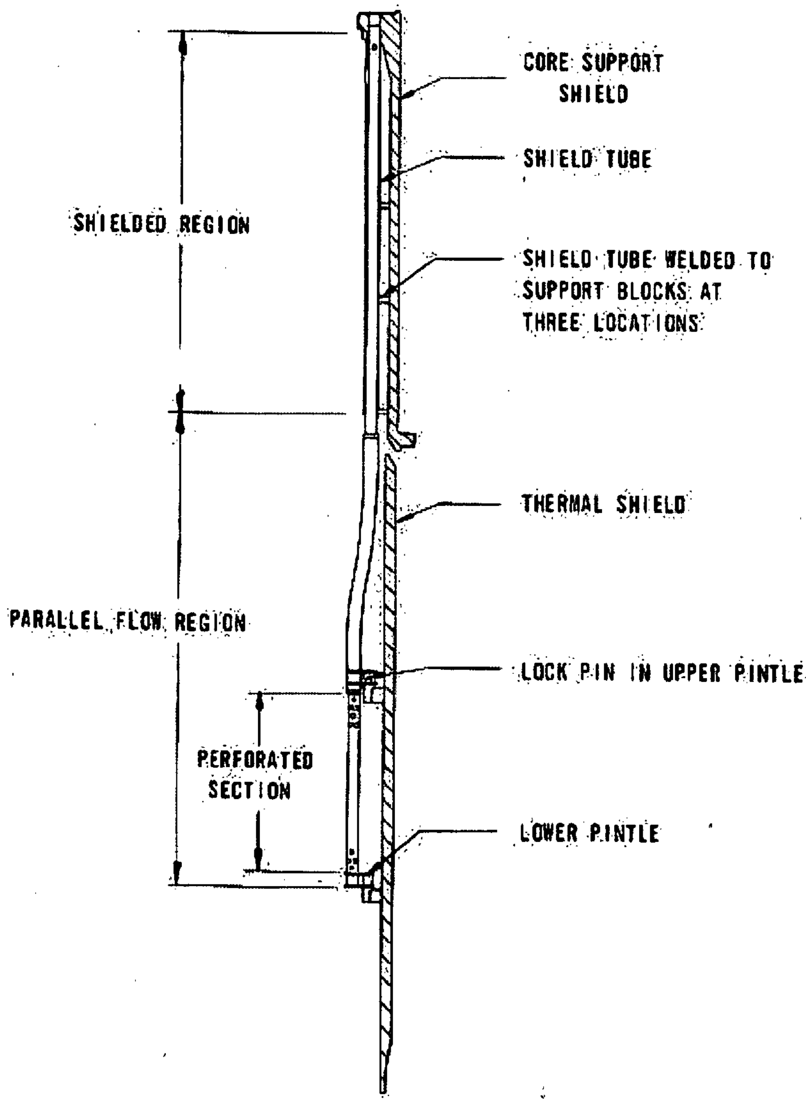

Figure 2-3: Original Surveillance Specimen Holder Tube Design in B&W-Design Reactor Vessel Internals

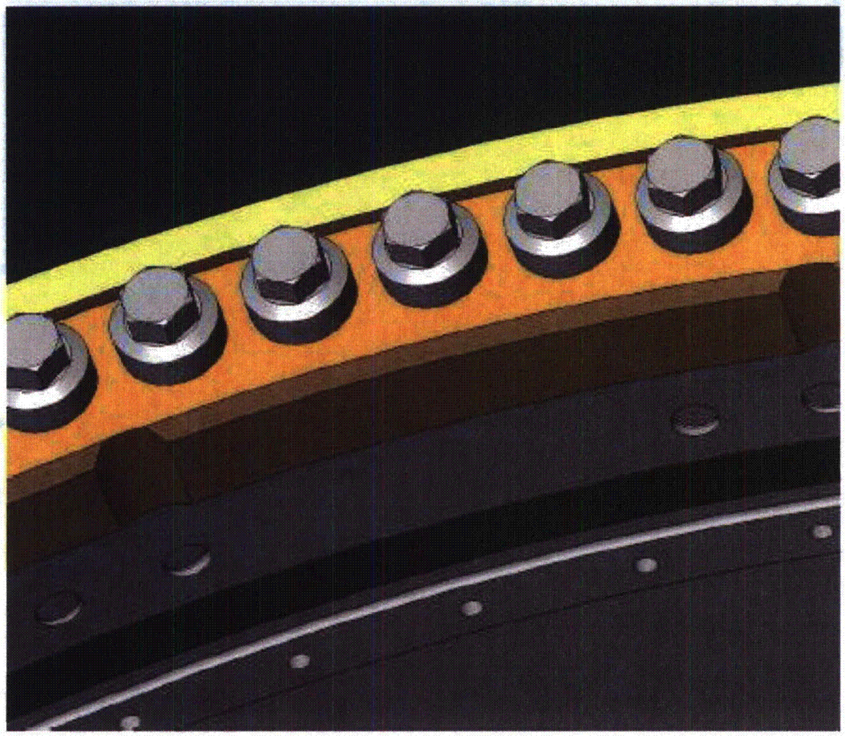

Figure 2-4: Upper Core Barrel Bolt Configuration at ONS units

(Note: Locking Devices Not Modeled)

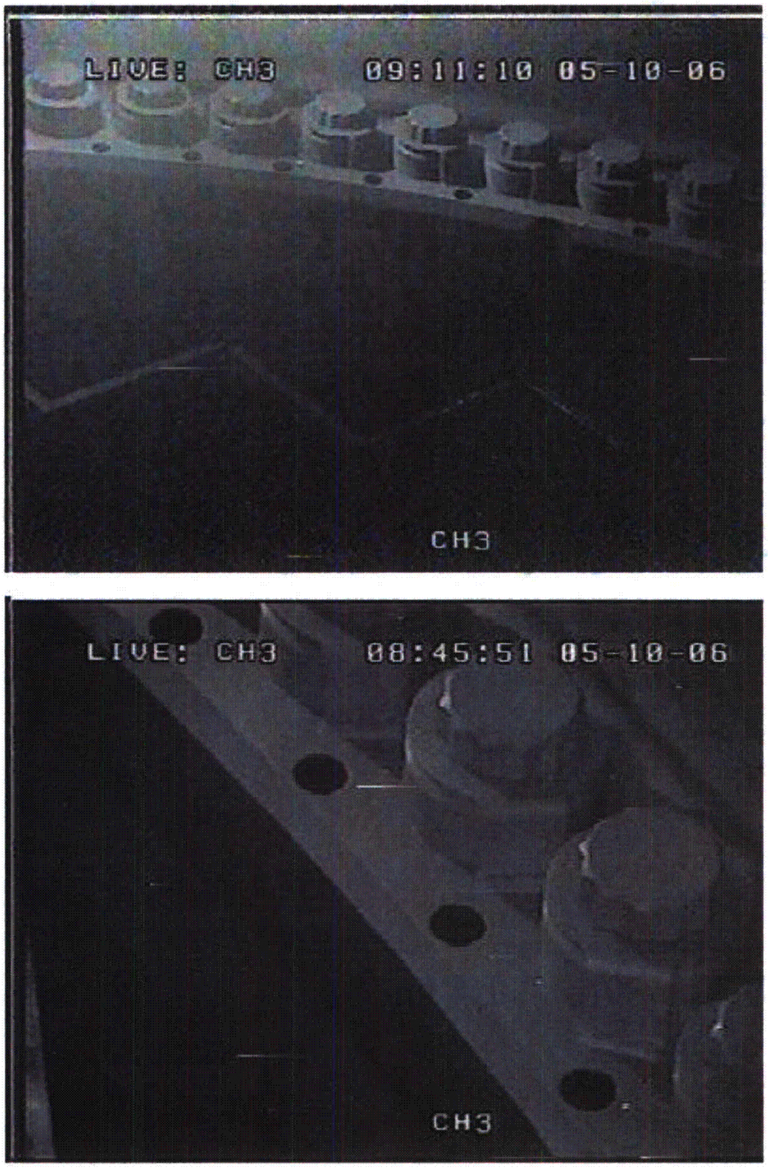

Figure 2-5: Representative Upper Core Barrel Bolt Configuration at ONS

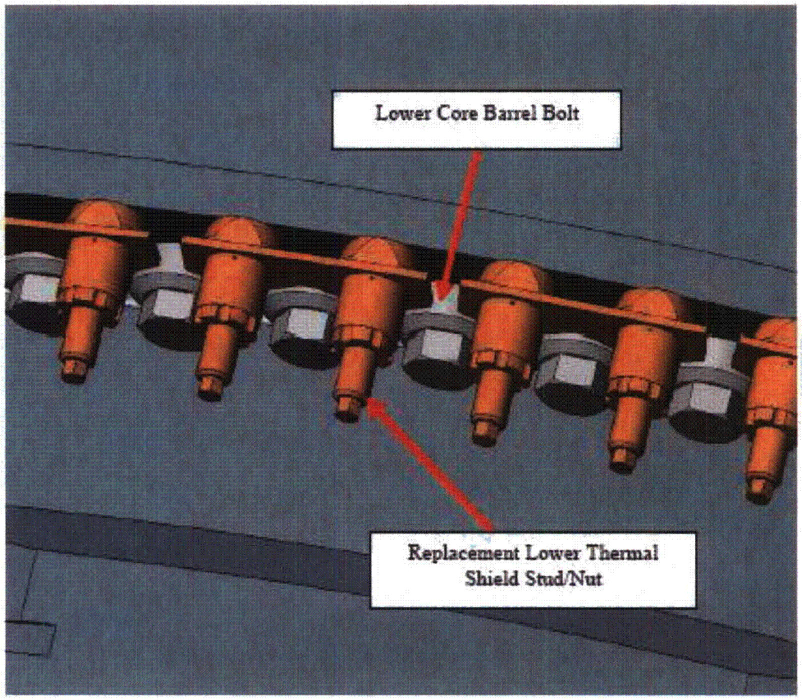

Figure 2-6: Typical Lower Core Barrel Bolt Configuration at ONS-1, ONS-2, and ONS-3

(Note: Lower Core Barrel Bolting Locking Devices Not Modeled)

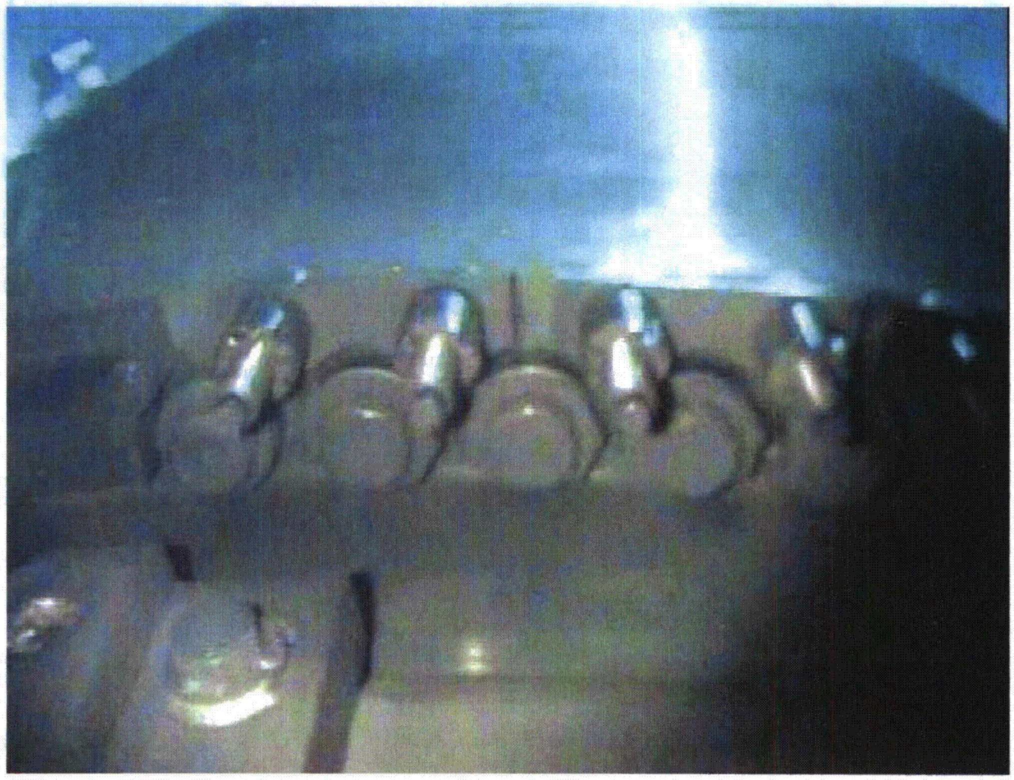

Figure 2-7: Representative Lower Core Barrel Bolt Configuration at ONS

The original Alloy A-286 UCB and LCB bolts were identical (i.e., having the same Mark number, 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 high-strength 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 MK-115 Type 304 (Carpenter Technology Steel heat number 810006) 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 that there were no flaw indications present 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.

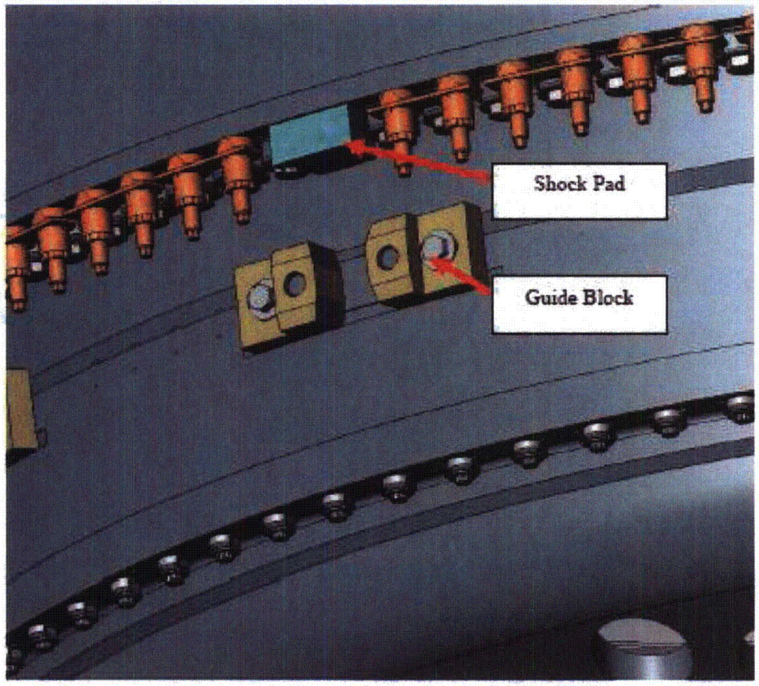

Figure 2-8: Typical Shock Pad Configuration at ONS-1, ONS-2, and ONS-3

- 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 the 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 (inspections dated 04/1984, 09/1985, and 01/1987) 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 Inspection and Flaws of Interest

Reference [10] provides a summary of past inspection information for the UCB and LCB bolting locations at each of the B&W operating units. It should be noted that an additional UCB inspection at Crystal River Unit 3 was performed, which was not summarized in Reference [10].

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.

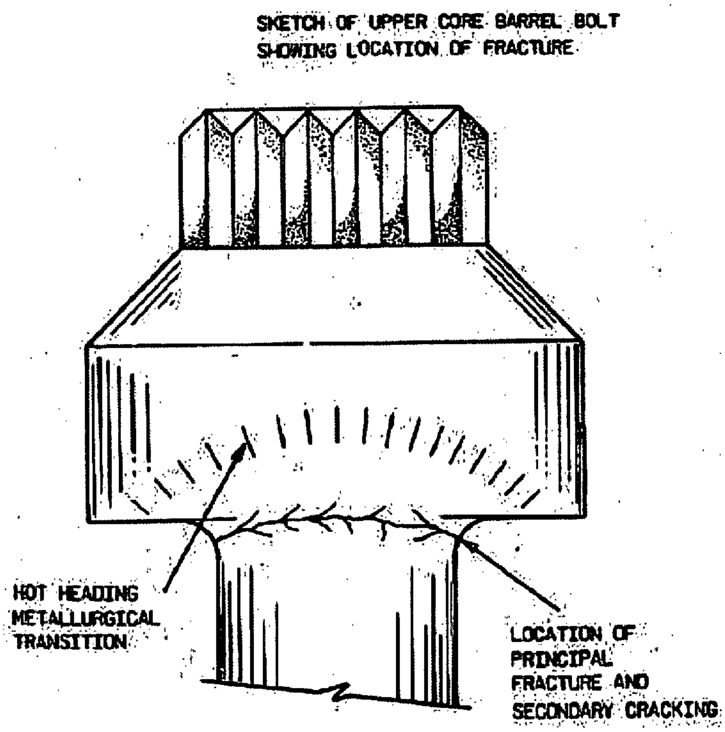

Figure 2-10: Schematic of Typical Fracture Orientation Observed on Failed High-Strength Bolting at B&W Units

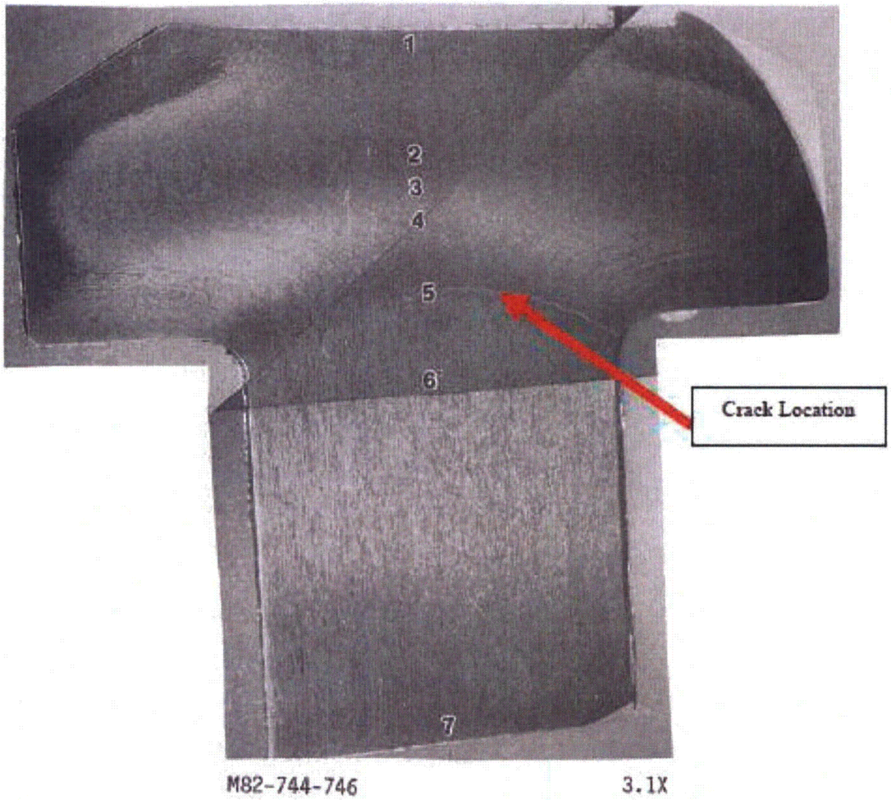

Figure 2-11: Typical Fracture Orientation Observed during Laboratory Examinations on Failed High-Strength Bolting at B&W Units (Note: Original Magnification was 3.1X) [Reference 7]

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. Such an assessment has recently been performed for the results of the LCB bolt UT inspection performed at ANO-1.[Reference 12] This assessment included a Weibull assessment of future LCB bolting SCC failure rates based on past UT results and bolt fabrication information from other B&W units, a structural analysis of potential additional LCB failures against the acceptability criteria developed in 2008 (see Section 2.6 for these acceptance criteria), an assessment of the consequence of LCB bolting failure on safe shutdown of the unit, and an assessment of the adequacy of the UT inspection results and interpretations.

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 [13].

Six conditions of removed/rejected bolts have been evaluated for each analyzed core support structure 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 [14].

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 [14].

3.0 OVERVIEW OF EXAMINATION SYSTEM

The examination system consists of AREVA NP procedure 54-ISI-165-11 Ultrasonic Examination of PWR Internal Bolting [Reference15], 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 GE Ultradoc 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.[Reference 15] 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 Reference [16] 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 nine different 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.[Reference 17]

When evaluating flaws, there are several points to consider. The 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 numerous times over approximately the past 30 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 high-strength 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.

ATTACHMENT 2

REGULATORY COMMITMENT TABLE

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
1. Schedule for Submittal of Applicant/Licensee Action Item 2 from Revision 1 of the NRC SER (for MRP-227, Revision 0).	May 31, 2013
2. Schedule for Submittal of Applicant/Licensee Action Items 4 from Revision 1 of the NRC SER (for MRP-227, Revision 0).	February 29, 2013
3. Schedule for Submittal of Applicant/Licensee Action Items 6 from Revision 1 of the NRC SER (for MRP-227, Revision 0).	May 31, 2014
4. Schedule for Submittal of Applicant/Licensee Action Items 7 from Revision 1 of the NRC SER (for MRP-227, Revision 0).	November 29, 2013