UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 19, 2015

Mr. Scott Batson
Site Vice President
Oconee Nuclear Station
Duke Energy Carolinas, LLC
7800 Rochester Highway
Seneca, SC 29672-0752

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3, ISSUANCE OF AMENDMENTS REGARDING INSPECTION PLAN FOR REACTOR VESSEL INTERNALS (TAC NOS. ME9024, ME9025, AND ME9026)

Dear Mr. Batson:

The Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment Nos. 392, 394, and 393 to Renewed Facility Operating Licenses DPR-38, DPR-47, and DPR-55, for the Oconee Nuclear Station, Units 1, 2, and 3, respectively. The amendments consist of changes to the Updated Final Safety Analysis Report (UFSAR) in response to your application dated November 8, 2010, as superseded by letter dated June 28, 2012, and as supplemented by letters dated September 1, 2011, March 28, April 18, September 27, and November 29, 2013, two letters dated March 20, 2014, letter dated April 23, 2014, and letter dated May 28, 2015.

These amendments revise the UFSAR and the licensing basis for the Oconee Nuclear Station, Units 1, 2, and 3, to approve the use of the NRC staff-approved topical report MRP-227-A, "Material Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," as the plan for the inspection of the reactor vessel internals during the period of extended operation permitted by the renewed plant licenses.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Enclosure 5 herewith contains proprietary information. When separated from Enclosure 5, this document is decontrolled.

If you have any questions, please contact me at 301-415-4032.

Sincerely,

James R. Hall

James R. Hall, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosures:
1. Amendment No. 392 to DPR-38
2. Amendment No. 394 to DPR-47
3. Amendment No. 393 to DPR-55
4. Non-Proprietary Safety Evaluation
5. Proprietary Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 392
Renewed License No. DPR-38

1. The Nuclear Regulatory Commission (the Commission) has found that:

    A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility), Renewed Facility Operating License No. DPR-38, filed by Duke Energy Carolinas, LLC (the licensee), dated November 8, 2010, as superseded by letter dated June 28, 2012, and as supplemented by letters dated September 1, 2011, March 28, April 18, September 27, and November 29, 2013, two letters dated March 20, 2014, letter dated April 23, 2014, and letter dated May 28, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

    B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

    C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

    D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

    E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 1

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Renewed Facility Operating License No. DPR-38 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 392, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Implementation Requirements

A. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

B. Upon implementation of Amendment No. 392 to adopt MRP-227-A, ONS shall incorporate in the Updated Final Safety Analysis Report (USAR) a reference to the MRP-227-A Reactor Vessel Internals ONS Inspection Plan.

C. The UFSAR change shall be implemented in the next periodic update of the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

R. Pascarelli

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-38

Date of Issuance: June 19, 2015

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 394
Renewed License No. DPR-47

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility), Renewed Facility Operating License No. DPR-47, filed by Duke Energy Carolinas, LLC (the licensee), dated November 8, 2010, as superseded by letter dated June 28, 2012, and as supplemented by letters dated September 1, 2011, March 28, April 18, September 27, and November 29, 2013, two letters dated March 20, 2014, letter dated April 23, 2014, and letter dated May 28, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

Enclosure 2

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Renewed Facility Operating License No. DPR-47 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 394, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Implementation Requirements

A. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

B. Upon implementation of Amendment No. 394 to adopt MRP-227-A, ONS shall incorporate in the Updated Final Safety Analysis Report (USAR) a reference to the MRP-227-A Reactor Vessel Internals ONS Inspection Plan.

C. The UFSAR change shall be implemented in the next periodic update of the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-47

Date of Issuance: June 19, 2015

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 393
Renewed License No. DPR-55

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility), Renewed Facility Operating License No. DPR-55, filed by Duke Energy Carolinas, LLC (the licensee), dated November 8, 2010, as superseded by letter dated June 28, 2012, and as supplemented by letters dated September 1, 2011, March 28, April 18, September 27, and November 29, 2013, two letters dated March 20, 2014, letter dated April 23, 2014, and letter dated May 28, 2015, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Renewed Facility Operating License No. DPR-55 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 393, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Implementation Requirements

   A. This license amendment is effective as of its date of issuance and shall be implemented within 90 days of issuance.

   B. Upon implementation of Amendment No. 393 to adopt MRP-227-A, ONS shall incorporate in the Updated Final Safety Analysis Report (USAR) a reference to the MRP-227-A Reactor Vessel Internals ONS Inspection Plan.

   C. The UFSAR change shall be implemented in the next periodic update of the UFSAR in accordance with 10 CFR 50.71(e).

FOR THE NUCLEAR REGULATORY COMMISSION

Robert J. Pascarelli, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-55

Date of Issuance: June 19, 2015

ATTACHMENT TO

LICENSE AMENDMENT NO. 392

RENEWED FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

LICENSE AMENDMENT NO. 394

RENEWED FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

LICENSE AMENDMENT NO. 393

RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

Replace the following pages of the Renewed Facility Operating Licenses with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
Licenses	Licenses
License No. DPR-38, page 3	License No. DPR-38, page 3
License No. DPR-47, page 3	License No. DPR-47, page 3
License No. DPR-55, page 3	License No. DPR-55, page 3

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 392 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1 (d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

   (a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

   (b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or proposing to own or operate, facilities for the generation and transmission of electricity which meets each of

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 394 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

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A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 393 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO INSPECTION PLAN FOR REACTOR VESSEL INTERNALS

AMENDMENT NO. 392 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 394 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-47

AND

AMENDMENT NO. 393 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DUKE ENERGY CAROLINAS, LLC

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

Proprietary information pursuant to
Title 10 of the *Code of Federal Regulations* (10 CFR), Section 2.390
has been redacted from this document. Redacted information is identified by blank space
enclosed within double brackets as shown here **[[ ]]**.

1.0 INTRODUCTION

By letter dated November 8, 2010 (Reference 1), as superseded by letter dated June 28, 2012 (Reference 2, the Inspection Plan), and as supplemented by letters dated September 1, 2011 (Reference 3), March 28 (Reference 4), April 18 (Reference 5), September 27 (Reference 6), and November 29, 2013 (Reference 7), two letters dated March 20, 2014 (References 8 and 9), letter dated April 23, 2014 (Reference 10), and letter dated May 28, 2015 (Reference 22), Duke Energy Carolinas, LLC (Duke Energy, the licensee), submitted a license amendment request (LAR) for the Oconee Nuclear Station Units 1, 2, and 3 (ONS). This LAR requested to revise the ONS Updated Final Safety Analysis Report (UFSAR) to include a Reactor Vessel Internals (RVI) inspection plan based on, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," (Reference 11, MRP-227-A). The RVI inspection plan was submitted with intent to meet a license renewal (LR) commitment located in Section 3.4.3.3, "Aging Management Programs for License Renewal," of NUREG-1723, "Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3," (Reference 12).

In Attachment 2 of its submittal dated June 28, 2012, the licensee committed to provide its evaluation of licensee action items 2, 4, 6, and 7, by May 31, 2014. The licensee stated that it considers the LR Commitment located in Section 3.4.2.2 of NUREG-1723 to be satisfied via the submittal of the ONS Inspection Plan (Reference 2) and the new commitments made in

Attachment 2. In its letter dated September 27, 2013, the licensee provided evaluations addressing action items 2 and 4. In its letter dated November 29, 2013, the licensee provided an evaluation addressing action item 7. In its letter dated March 20, 2014 (Reference 8), the licensee provided responses to the NRC staff's request for additional information (RAI) regarding action item 7. By letter dated March 20, 2014 (Reference 9), the licensee committed to provide the analyses addressing licensee action item 6 by May 31, 2015. By letter dated May 28, 2015 (Reference 22), the licensee revised the commitment date addressing licensee action item 6 from May 31, 2015, to May 31, 2016.

The supplemental letters dated September 1, 2011, March 28, April 18, September 27, and November 29, 2013, two letters dated March 20, 2014, letter dated April 23, 2014, and letter dated May 28, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on October 2, 2012 (77 FR 60149).

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (CFR) Part 54 addresses the requirements for plant LR. The regulations in 10 CFR Section 54.21, "Contents of application – technical information," (10 CFR 54.21) requires that each application for LR contain an integrated plant assessment (IPA) and an evaluation of time-limited aging analyses (TLAAs). The plant-specific IPA shall identify and list those structures and components subject to an aging management review (AMR) and demonstrate that the effects of aging (cracking, loss of material, loss of fracture toughness, dimensional changes, loss of preload) will be adequately managed so that their intended functions will be maintained consistent with the current licensing basis (CLB) for the period of extended operation (PEO) as required by 10 CFR 54.29(a). In addition, 10 CFR 54.22, "Contents of application – technical specifications," requires that an LR application include any technical specification changes or additions necessary to manage the effects of aging during the PEO as part of the LR application.

Structures and components subject to an aging management program (AMP) shall encompass those structures and components that (1) perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties and (2) are not subject to replacement based on a qualified life or specified time period. These structures and components are referred to as "passive" and "long-lived" structures and components, respectively.

On January 12, 2009, the Electric Power Research Institute (EPRI) submitted for NRC staff review and approval Revision 0 of MRP Report MRP-227 (Reference 13), which was intended as guidance for applicants in developing their plant-specific AMP for RVI components. The scope of components considered for inspection under the guidance of MRP-227, Revision 0, includes core support structures, which are typically denoted as Examination Category B-N-3 by Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), and those RVI components that serve an intended safety function consistent with the criteria in 10 CFR 54.4(a)(1). 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 subject to an AMR, as defined in 10 CFR 54.21(a)(1).

The NRC staff's review of the LR application for ONS was documented in NUREG-1723, of which Section 3.4.3.3 contains specific commitments related to RVI components. The ONS Inspection Plan was developed by the licensee based on MRP-227-A with the intent to meet these LR commitments. Revision 1 to the final safety evaluation (SE) regarding MRP-227, Revision 0, was issued on December 16, 2011 (Reference 14). This SE contains specific conditions on the use of the topical report and licensee action items that must be addressed by those utilizing the topical report as the basis for a submittal to the NRC. On January 9, 2012, EPRI published the NRC approved version of topical report MRP-227-A (Reference 11). MRP-227-A contains a discussion of the technical basis for the development of plant-specific AMPs for RVI components in pressurized-water reactor (PWR) vessels and also provides inspection and evaluation guidelines for PWR applicants to use in their plant-specific AMPs. MRP-227-A provides the basis for renewed license holders to develop plant-specific inspection plans to manage aging effects on RVI components, as described by their LR commitment.

Subsequent to the submittal of MRP-227 and prior to the issuance of the SE for MRP-227, NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," Final Report (Reference 15), was issued, providing new AMR line items and aging management guidance in AMP XI.M16A, "PWR Vessel Internals." License Renewal Interim Staff Guidance LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors," (Reference 16) was published on May 28, 2013, to update AMP XI.M16A for consistency with MRP-227-A.

2.1 <u>Overview of the MRP-227-A Process</u>

As the initial step in the process for developing the inspection recommendations of MRP-227-A, components were screened for eight different aging mechanisms: (1) stress-corrosion cracking (SCC); (2) irradiation-assisted stress-corrosion cracking (IASCC); (3) wear; (4) fatigue; (5) thermal aging embrittlement (TE); (6) irradiation embrittlement (IE); (7) irradiation-enhanced stress relaxation and creep; and (8) void swelling. Screening inputs included chemical composition (material grade), neutron fluence, temperature history, and representative stress levels. Components determined to be below the screening criteria for all aging mechanisms were designated category "A" while those exceeding the criteria for at least one mechanism were designated "non A." For the "non A" components, Failure Modes, Effects, and Criticality Analyses (FMECA) were then performed to categorize each component as category A, B, or C, with A being the least affected and C being the most affected. The components determined to be category A in the initial screening were also reviewed by the FMECA expert panel to confirm their category A status. Category B and C components were determined to need further evaluation and were subject to a functionality assessment using irradiated and aged material properties to determine the effects of the degradation mechanisms on functionality. As a result of the functionality assessment, each RVI component was assigned to one of four functional groups:

- 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 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. MRP-227-A specifies the scope, methods, coverage and schedule of inspections of Primary components. Initial inspection of most Primary components is required within two refueling outages (RFOs) of the start of the

PEO. For a few components, actions other than inspections are specified for aging management, such as analysis.

- 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 inspections or other 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.

- 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.

Aging management strategy development combined the results of 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.

Augmented inspections recommendations are identified for each Primary and Expansion category component. The recommendations for the Primary components also identify timelines[1] for the inspection. The inspection strategy generally employs VT-3 level visual examinations to evaluate general component condition, EVT-1 level visual examinations to identify surface breaking flaws, and VT-1 level visual examination to identify surface discontinuities such as gaps. Cracking in baffle-former bolts is monitored with ultrasonic (UT) techniques.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the ONS Inspection Plan to determine if it demonstrated that the effects of aging on the subject RVI components covered by MRP-227-A would be adequately managed so that the components' intended functions would be maintained consistent with the CLB for the PEO, in accordance with 10 CFR 54.21(a)(3). The final SE for MRP-227, Revision 0, concluded that the MRP-227, Revision 0, report, as modified by the conditions and limitations and licensee action items of the SE, provides for the development of an acceptable AMP for PWR RVI components. Therefore, the NRC staff's technical evaluation of the ONS Inspection Plan focused on determining whether the plan is consistent with the recommendations of MRP-227-A, and that it addresses the plant-specific licensee action items.

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[1] For Babcock & Wilcox (B&W) - design RVI such as ONS, inspection schedules, methods, and inspection coverage are defined in MRP-227-A, Table 4-1 for Primary category components and Table 4-4 for Expansion category components.

3.1 Industry and ONS Programs and Activities (Section 4 of the ONS Inspection Plan)

Section 4, "Industry and ONS Programs and Activities" of the ONS Inspection Plan contains pertinent ONS and industry programs and activities used for the development and implementation of MRP-227-A. It also contains discussion of specific technical issues which affect the ONS Inspection Plan based on MRP-227-A. The NRC staff's evaluation of these issues includes an evaluation of the licensee's response to the staff's RAI as provided in Reference 4.

3.1.1 TLAA for the Replacement Lower Thermal Shield (LTS) Bolts

Section 4.1.3.1, "Currently Identified TLAAs," of the ONS Inspection Plan identified transient cycle count assumptions for the replacement bolting as a TLAA. The NRC staff's RAI-3 as provided by letter dated February 11, 2013 (Reference 17), requested the licensee to confirm whether (1) the replacement bolts are the LTS bolts, (2) the MRP-227-A specified examination method, frequency, and coverage for this RVI component coexist with the management of it through TLAA, and (3) any plant-specific RVI programs (e.g., TLAAs) will replace the MRP-227-A inspections.

In Reference 4, the licensee confirmed that the replacement bolts are the LTS bolts. Regarding the transient cycle count assumptions for the LTS bolts as a TLAA, the licensee's response indicated that 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 PEO. Therefore, Duke Energy will continue to monitor and track occurrences of design transients for ONS during the PEO as part of the TLAA, and will not supersede the MRP-227-A classification of the LTS bolts as an Expansion component. Regarding RAI-3(3), the licensee stated that no plant-specific programs will replace the MRP-227-A inspections or require a deviation from MRP-227-A. The NRC staff considers this issue to be resolved because all plant-specific programs such as TLAAs are used to supplement, not to replace any MRP-227-A inspections.

3.1.2 Impact of the 2006 and 2008 Core Clamping Measurements on the MRP-227-A Inspections

Section 4.1.6, "Fuel/Baffle Interaction Investigation," of the ONS Inspection Plan stated that the MRP-227-A requirement for a one-time physical measurement of the interference fit between the plenum cover weldment rib pads and the RV flange was performed at the ONS units between 2006 and 2008. RAI-6 (Reference 17) requested the licensee discuss the impact of these measurements on the MRP-227-A specified subsequent visual (VT-3) examination on the 10-year inservice inspection (ISI) interval for the ONS Units.

In Reference 4, the licensee stated that the core clamping measurements at ONS found no evidence of wear and there was no evidence that core clamping has been degraded. Therefore, ONS will continue to monitor potential wear of the core clamping items in the plenum cover assembly and core support shield (CSS) assembly via subsequent VT-3 examinations during the 10-year ISI interval per MRP-227-A requirements. The NRC staff considers this acceptable because, to date, no indication of wear has been found in the subject RVI components and no modification of the MRP-227-A requirements is needed.

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

Section 4.2, "ONS Programs and Activities," of the ONS Inspection Plan mentioned visual examination of baffle-to-baffle (BB) and baffle-to-former (BF) bolts at each refueling outage. RAI-7 (Reference 17) requested the licensee confirm that Duke Energy will perform UT examinations for BB and BF bolts in accordance with Examination Method/Frequency of MRP-227-A, and the stated visual examinations simply supplement the UT examinations.

In Reference 4, the licensee clarified that the following examinations will be performed in accordance with MRP-227-A, Table 4-1, "B&W Plants Primary Components:" (1) the VT-3 examinations of the BF bolt and internal BB bolt locking devices, including locking welds, and (2) the UT examination of the BF bolts. These resolved the NRC staff's concern of any reduced MRP-227-A inspections related to the BB and BF bolts to credit the ONS VT-3 examinations performed on them during refueling outages. The licensee further clarified that visual examinations performed on the internal BB bolts and the BF bolts during refueling outages simply supplement the inspection requirements in MRP-227-A. The NRC staff considers this issue to be resolved.

3.1.4 ONS Defined Core Support Structure Components for ASME Code, Section XI, Examination Category B-N-3

Table 4-2, "Typical ONS Core Support Structure Components (Examination Category B-N-3) of the ONS Inspection Plan defined "typical" ONS core support structure components for Examination Category B-N-3 of the ASME Code, Section XI. RAI-8 (Reference 17) requested the licensee describe the process and effort in establishing Table 4-2 and identify the ONS core support structure components, which were inspected as B-N-3 Examination Category in the past but are not included in Table 4-2. The intent of the RAI was to ensure that all ONS core support structure components for Examination Category B-N-3 of the ASME Code, Section XI which could affect the functionality of core support are included in Table 4-2.

In Reference 4, the licensee explained that Table 4-2 of the ONS Inspection Plan is based on an AREVA inspection matrix and Duke Energy engineering judgment. It includes accessible components that Duke Energy engineering considers to serve a core support function, as well as additional components that have potential to affect the functionality of the internals. There is not a generic list of core support structures for Babcock and Wilcox (B&W)-designed internals. Since Table 4-2 contains all accessible core support components identified by Duke Energy, including even non-core support components that could affect the functionality of core support, the NRC staff accepts the ONS list of core support structure components for Examination Category B-N-3 of the ASME Code, Section XI. The NRC staff considers this issue to be resolved.

3.1.5 Operating Experience of the Replacement LTS Studs/Nuts

Section 4.2.5, "Lower Thermal Shield Replacement Studs/Nuts," of the ONS Inspection Plan mentioned that failure of the original Alloy A-286 LTS bolts in the 1980s led to the installation of replacement Alloy X-750 studs/nuts at ONS. RAI-9 (Reference 17) requested the licensee discuss the inspection results accumulated in the past 30 years at ONS for the replacement LTS bolts to support the MRP-227-A specified examination method/frequency/coverage for this RVI component.

In Reference 4, the licensee stated that limited UT inspection of the currently installed LTS closure designs has been performed in 1983 (16 Alloy X-750 studs/nuts were UT inspected at ONS, Unit 1 with no rejectable indications), 1984 (96 Alloy A-286 bolts were UT inspected at another B&W designed plant with no rejectable indications, and 1990 (24 Alloy X-750 bolts were UT inspected at another B&W designed plant with no rejectable indications). It further confirmed that none of the replacement Alloy X-750 stud/nuts at ONS, Units 2 or 3 have been UT inspected, and a VT-3 examination of the LTS closure mechanisms, including their locking mechanisms is performed every 10-year ISI interval. Since there have been no failures in the replacement LTS studs and nuts or bolts, the NRC staff determined that the plant-specific and industry operating experience supports the "Expansion" category of these bolts in MRP-227-A. The NRC staff considers this issue to be resolved.

3.1.6 MRP-227-A Examination Requirements for the Current 40-year License

Section 4.2.7, "Volumetric (UT) Examinations of Upper Core Barrel Bolts," of the ONS Inspection Plan stated that UT examinations for the lower core barrel (LCB) bolts are planned during the RVI inspections in 2012, 2013, and 2014, in compliance with the MRP-227-A examinations requirement for the LCB bolts. RAI-11 (Reference 17) requested the licensee provide similar information on MRP-227-A required examinations (visual or UT) for all other RVI components to be performed before the end of the current 40-year license for the NRC staff to assess the performance of the MRP-227-A based programs prior to approval of this inspection plan.

In Reference 4, the licensee listed all MRP-227-A required examinations and the examination results for RVI components at ONS, Unit 1 obtained before the end of the current 40-year license, stating, in part, that:

1. A one-time physical measurement (initial inspection) of the differential height of the top of the plenum rib pads to reactor vessel seating surface, with the plenum in the reactor vessel, was performed at ONS-1 in 2006 and no relevant indications (no evidence of wear) were noted. A VT-3 subsequent inspection of the plenum cover weldment rib pads, plenum cover support flange and CSS top flange was performed in 2012 and no relevant indications were noted.

2. A VT-3 inspection of accessible surfaces of 100% of the control rod guide tube (CRGT) spacer castings including each of the 4 screw locations (at every 90 degrees) was performed at ONS-1 in 2012 and no relevant indications were noted.

3. A VT-3 inspection of accessible surfaces of the CSS vent valve top and bottom retaining rings was performed at ONS-1 in 2012. A verification of the operation of each vent valve was performed through manual actuation of the valve in 2012. The jack screws and jack screw locking devices, which are not included in MRP-227-A, also received a VT-3 inspection in 2012.

   Results: No relevant indications were noted on retaining rings or locking devices. On vent valve ZW, one jack screw was shorter than the other, there was a crack-like indication on one of the jack screws, and there were signs of mechanical damage on that vent valve. Vent valve ZW was replaced with a new vent valve.

4. Four of the 120 upper core barrel (UCB) bolts had been removed in the 1980s for testing and no indications were found. Volumetric examination (UT) of the remaining 116 UCB bolts was performed at ONS-1 in 2008 and no relevant indications were found. Visual (VT-3) examination of 116 UCB bolt locking devices was performed in 2012 and no relevant indications were found.

5. Volumetric examination (UT) of 108 lower core barrel (LCB) bolts was performed at ONS-1 in 2012 and 5 crack-like indications were found. Visual (VT-3) examination of 108 LCB bolt locking devices accessible surfaces was performed in 2012. (100% inspection was achieved on 101 locking devices while 40 to 50% inspection was achieved on the other 7 due to interferences.)

   Results: one locking device had a missing weld on one side and an undersized weld on the other side.

6. Volumetric examination (UT) was performed on 860 (of 864) baffle-to-former bolts at ONS-1 in 2012. (Four bolts could not be inspected due to large welds on the locking bars.) No relevant indications were found.

7. Visual examination (VT-3) of 100% of the accessible surface within 1 inch around each flow and bolt hole in all baffle plates was performed in 2012 and no relevant indications were found.

8. Visual examination (VT-3) was performed on 864 baffle-to-former bolt locking devices and 272 internal baffle-to-baffle bolt locking devices in 2012 and no relevant indications were found.

9. One flow distributor (FD) bolt was removed at ONS-1 in 1981 and no indications were found. (A second bolt could not be removed due to high torque values.) Volumetric examination (UT) of the remaining 95 FD bolts was performed at ONS-1 in 2012. One flow distributor bolt had a crack-like indication in 2012.

   Two FD bolt locking devices were removed in 1981. Visual (VT-3) examination of the remaining 94 FD bolt locking devices was performed in 2012 and no relevant indications were found.

10. Previously one guide block was lost and its pair was removed (neither has been replaced.) Visual examination (VT-3) was performed on accessible surfaces of the 22 remaining lower grid assembly Alloy X-750 dowel-to-guide block welds at ONS-1 in 2012 and no relevant indications were found.

11. Visual examination (VT-3) was performed on 100% of the top surfaces of 52 Incore Monitoring Instrumentation Guide Tube Assembly spider castings and welds of the castings to the adjacent lower grid rib section (8 welds per casting) at ONS-1 in 2012 and no relevant indications were found. An unidentified loose part was identified under one spider casting.

The NRC staff found that only Inspections 3, 5, and 9 resulted in relevant indications, and their dispositions were: Inspection 3 led to replacement of one vent valve and Inspections 5 and 9 led to an operability assessment approved for one operating cycle. Neither would trigger expansion inspections. It should be noted that the majority of the 11 inspections were performed in 2012 in accordance with the ONS inspection plan of ONS, Unit 1 based on MRP-227-A. The fact that most of them showed no relevant indications and the ones having indications are consistent with operating experience demonstrate that the proposed ONS inspection plan based on MRP-227-A is appropriate and effective in managing aging effects of ONS, Unit 1 RVI components for the 40-year operation, and is likely to remain so throughout the period of extended operation because when time is a factor (e.g., neutron fluence, MRP-231, Rev. 1) in finalizing the ONS inspection plan, 60-year is considered. Similar MRP-227-A required inspections were planned to be performed by October 6, 2013, for ONS, Unit 2, and by July 19, 2014, for ONS, Unit 3. The NRC staff considers this issue to be resolved because the licensee has provided the requested information regarding MRP-227-A required RVI examinations that were performed before the end of the current 40-year license to support the staff's assessment of the adequacy of the ONS Inspection Plan in the early phase of its implementation. The MRP-227-A required inspections for ONS, Unit 2 were performed about eight months after the RAI-11 response was provided to the NRC. To be complete, the NRC staff audited the inspection results of ONS, Unit 2 and found them consistent with what is expected in the MRP-227-A. The disposition of the indications is also within the framework of MRP-227-A and justified.

Separately, RAI-15 (Reference 17) requested the licensee confirm whether an operability evaluation required by WCAP-17096, Rev. 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," (Reference 18) will be performed if relevant conditions are identified in the UCB or LCB bolts. This question was raised before Inspection 5 was conducted. As described above, an operability evaluation was performed to demonstrate acceptable operation for one operating cycle after cracking was found in 5 of the 108 LCB bolts. The NRC staff considers this issue resolved. Note that Duke Energy will need to perform similar operability assessments for ONS for future operating cycles.

3.1.7 Consistency of RVI Components under the ONS Inspection Plan and under MRP-227-A

Section 2.4, "ONS RV Internals Scope Background," of the ONS Inspection Plan is related to the ONS RVI components scope. The NRC staff reviewed Section 2.4, but did not find a clear definition of the scope of the components. RAI-12 (Reference 17) requested the licensee confirm that the ONS Inspection Plan includes all RVI components listed in Tables 4-1 (Primary) and 4-4 (Expansion) of MRP-227-A and those listed in Table 4-2 (Existing) of the ONS Inspection Plan.

In Reference 4, the licensee stated that the ONS Inspection Plan was developed considering all the RVI components listed in Tables 4-1 (Primary) and 4-4 (Expansion) of MRP-227-A and Table 4-2 of the ONS Inspection Plan and thus "includes" all these components. The NRC staff finds this to be acceptable because all RVI components required by MRP-227-A to be inspected are included in the ONS Inspection Plan.

3.2 ONS RVI AMP Attribute Evaluation (Section 5 of the ONS Inspection Plan)

Section 1.2, "Purpose," of Reference 14, stated, in part, that:

> The review also considered compliance with license renewal (LR) requirements in 10 CFR 54.21(a)(3) in order to allow licensees or applicants the option of adopting the aging management methodology described in MRP-227 as the basis for

> managing age-related degradation in RVI components and incorporating, by reference, the recommended guidelines into PWR Vessel Internals AMPs (or their equivalents). This option is consistent with the recommendations in AMP, XI.M16A, "PWR Vessel Internals," of the Generic Aging Lessons Learned (GALL) Report, Revision 2 (NUREG-1801, Revision 2).

The licensee's evaluation of each AMP element in Section 5, "ONS RV Internals AMP Attribute Evaluation," of the ONS Inspection Plan cited the appropriate MRP-227-A information to demonstrate that the intent of the specific GALL program element from AMP XI.M16A from LR-ISG 2011-04 (Reference 16) is met at ONS. Therefore, any PWR Vessel Internals AMP that was developed consistently with MRP-227-A, addressing all eight licensee action items appropriately, will meet the 10 program elements from AMP XI.M16A.

The ONS Inspection Plan was developed in accordance with MRP-227-A, adopting Table 4-1, "B&W Plants Primary Components," Table 4-4, "B&W Plants Expansion Components," and Table 5-1, "B&W Plants Examination Acceptance and Expansion Criteria," of MRP-227-A without any plant-specific alterations. Based on the above, and the NRC staff's evaluation documented in Sections 3.1 and 3.3 of this SE, the NRC staff agrees with the licensee's conclusion in Section 5.11, "Program Conclusion," which stated, in part, that:

> [T]he ONS RV Internals inspection plan meets the intent of the ten GALL program elements from Chapter XI, AMP XI.M1 6A from NUREG-1801, Rev. 2; this demonstrates the adequacy of managing the aging effects of the ONS RV Internals.

3.3 <u>Conditions/Limitations and Plant-Specific Actions Items Contained in MRP-227-A (Section 6 of the ONS Inspection Plan)</u>

As stated in Section 1.0 of this SE, the ONS Inspection Plan was developed by the licensee based on MRP-227-A to meet the LR commitments. The NRC staff SE on MRP-227-A (Reference 14) specified seven topical report conditions and eight licensee action items. The topical report conditions were specified to ensure that certain information was revised generically in the published MRP-227-A, and the action items were specified for each licensee to address plant-specific issues which could not be resolved generically.

3.3.1 <u>Topical Report Conditions</u>

Although the topical report conditions have been satisfied as a result of issuance of MRP-227-A, the NRC staff still issued an RAI regarding Topical Report Condition 7, which requires updating of operating experience in Appendix A of MRP-227-A periodically. RAI-1 (Reference 17) requested the licensee identify the MRP-227-A, Appendix A operating experience which was contributed by ONS, and provide ONS operating experience on RVI degradations that were not discussed in Appendix A of MRP-227-A.

In Reference 4, the licensee provided a list of ONS operating experience (or failure) which contributed to MRP-227-A, Appendix A. Several RVI component failures that were not included in MRP-227-A because they were unrelated to aging were also identified. Further, a report on the 2012 RVI inspection for ONS, Unit 1 was provided to EPRI per the MRP-227-A requirement for future updating of MRP-227-A. A summary of the age-related conditions found during this inspection (conducted after issuance of MRP-227-A) is contained in the licensee's RAI response. The NRC staff considers RAI-1 to be resolved because the licensee provided a good case study for providing ONS operating experience, contributing to the completion of MRP-227-A, Appendix A; screened out RVI operating experience unrelated to aging; and provided the 2012 and 2013 inspection results information to EPRI for future updating of MRP-227-A, Appendix A in accordance with the established process with EPRI.

3.3.2 Licensee Action Items from SE of MRP-227, Revision 0

The NRC staff's final SE of MRP-227, Revision 0 (Reference 14), contained 8 plant-specific licensee action items. The NRC staff determined that licensee action items 1, 2, 4, 6, 7, and 8 are applicable to ONS and licensee action items 3 and 5 are not applicable to ONS.

3.3.2.1 Licensee Action Item 1

Per Section 4.2.1 of Reference 14, each licensee is responsible for assessing its plant's design and operating history and demonstrating that MRP-227-A is applicable to the facility. Each 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, Combustion Engineering (CE), or B&W) which support MRP-227-A. The licensee shall also describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. Finally, the licensee shall submit this evaluation for NRC review and approval as part of its application to implement MRP-227-A.

3.3.2.1.1 Licensee Evaluation

In Section 4.1.1.1.2, "MRP-227-A Applicability to ONS," of the ONS Inspection Plan, the licensee listed the assumptions found in Section 2.4 of MRP-227-A, followed by their applicability to ONS as follows:

- 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 requirements are applicable to all operating U. S. PWR operating plants as of May 2007 for the three designs (i.e., B&W, Westinghouse, and [CE]) identified. No modifications have been made to the

> ONS RV Intemals 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 licensee further stated, in part, that:

> 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, 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.

3.3.2.1.2 NRC Staff Evaluation

This action item requires that licensees assess their plant's design and operating history and demonstrate that the approved version of MRP-227 is applicable to the facility.

The ONS Inspection Plan for RVI components is based on MRP-227-A, which groups all RVI components into four categories: Primary, Expansion, Existing, and No Additional Measures. The four different component inspection categories contain different RVI components for each nuclear steam supply system (NSSS) designs: B&W plants, Combustion Engineering (CE) plants, and Westinghouse plants. In Reference 14, the NRC staff expressed a concern that the industry's FMECAs and generic functionality analyses may not be bounding. Therefore, the NRC staff specified Licensee Action Item 1 in Reference 14, requesting each licensee "describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories."

For B&W plants, the screening process for the RVI components is documented in MRP-189, Rev. 1, "Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189-Rev. 1)," (Reference 19); FMECAs are documented in MRP-190, "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals," (Reference 20); and generic aging management strategies are documented in MRP-231, Rev. 2, "Materials Reliability Program: Aging Management Strategies for B&W Pressurized Water Reactor Internals," (Reference 21).

The NRC staff's evaluation of the ONS Inspection is focused on identification of the plant-specific information that could affect the conclusions of MRP-190 and MRP-231, Rev. 1 regarding

FMECAs and generic functionality analyses supporting MRP-227-A. MRP-190 provides a systematic, qualitative review of the B&W-designed PWR internals to identify combinations of RVI components and age-related degradation mechanisms that potentially result in degradation leading to significant risk. An expert panel was formed during the process to finalize the FMECA table documented in MRP-190. Therefore, it is clear that the development of the FMECA table is independent of the plant-specific analyses based on the plant-specific design configurations.

Some plant-specific design features were directly identified in the FMECA table (e.g., the component "Original CSS-to-Core Barrel Bolts (ANO-1, DB, ONS)," applies to only three B&W plants) and required no plant-specific verification. Further, when consolidating results into 5 risk bands (I: not significant; II: mild; III: moderate; IV: significant; and V: an immediate reevaluation of the design is necessary), only the most conservative risk was used in the FMECA table for like components. Based on this and the NRC staff's review of other assumptions stated in Section 4 of MRP-227, "FMECA", the staff agrees with the MRP-190 conclusion that, "These assumptions are either bounding or methodological, and do not require plant-specific verification for each of the B&W-designed operating units."

The nature of MRP-231, Rev. 1, which has been identified as containing proprietary information, is different. In the beginning, the report summary declared that, **[[**

]]

The staff found that generic structural analysis, engineering analysis, and operating and previous ISIs were used in functionality assessment documented in Section 2 of MRP-231, Rev. 1 to provide the basis for updating the categorization of PWR RVI components for the B&W-design: Category A (not a concern), Category B (a potential concern), and Category C (likely). Additional information was discussed in Section 3 of MRP-231, Rev. 1 to provide the basis for redefining the categorization of RVI components into Primary, Expansion, Existing, and No Additional Measures. If no additional information is available, it is expected that Category C RVI components be redefined as Primary and Category B RVI components be redefined as Expansion. For Licensee Action Item 1, NRC staff's review is focused on examining the RVI components in Table 2-8 of MRP-231, Rev. 1 to determine whether the RVI components designated as "Category B" could be reclassified as "Category C" when plant-specific information to be obtained from the licensee was used instead of the generic information documented in MRP-231, Rev. 1. Any RVI components designated as "Category C" or later as "Primary" in Table 3-8 in accordance with the new categorization need not be reviewed because they are already in the highest inspected category. According to this principle, 12 RVI components designated as "Category B" in Table 2-8 need to be reviewed (except for those that don't apply to ONS):

- CRGT Assembly
  - CRGT Spacer Castings
- Core Barrel Assembly
  - Core Barrel Cylinder (including the vertical and circumferential seam welds)

- Former plates
- External BB bolts
- Inaccessible Locking Device and Locking Weld for Core Barrel-to-Former (CB) Bolts and External BB Bolts
- Upper Thermal Shield (UTS) Bolts
- Surveillance Specimen Holder Tube (SSHT) Bolts (Crystal River Unit 3 (CR-3), Davis-Besse (DB))

- Upper Grid Assembly
  - Upper Grid Fuel Assembly Support Pads: Alloy X-750 Dowel Locking Weld (except DB)
- Lower Grid Assembly
  - Lower Grid Fuel Assembly Support Pad Component items: Pad, Pad-to-Rib Section Weld,
    - Alloy X-750 Dowel, Cap Screw, Their Locking Welds
  - X-750 Bolts for Lower Grid Shock Pads (TMI-1 only)
  - Lower Thermal Shield (LTS) Bolts
- FD Assembly
  - FD Bolts

3.3.2.1.2.1 Evaluations of Specific Components

CRGT Spacer Castings

[[ ]] this component was elevated to "Primary" in MRP-227-A. Therefore, the only potential impact from a change in screening assumptions would be the screening in of additional degradation mechanisms which might affect the type of inspection performed. Although the castings were [[ ]] the castings are already being inspected for cracking since [[ ]] is only a problem if cracking is present.

[[

]] Therefore,

the NRC staff concludes that determination of categorization of CRGT spacer castings with respect to various degradation mechanisms remains qualitative, and this determination was not based on any generic quantitative analysis from which certain parameters can be revised using ONS plant-specific information. [[

]]

[[

]] Based on the discussion above, the NRC staff concludes that the final categorization for the CRGT spacer castings is unlikely to be affected by any ONS plant-specific information.

Core Barrel Cylinder (including the vertical and circumferential seam welds)

[[

]]

[[

]] Since all B&W plants have baffle plates located closest to the core and core barrel cylinder farthest from the core, the NRC staff concludes that the final categorization for the core barrel cylinder is unlikely to be affected by any ONS plant-specific information.

Former plates

[[

]]

[[

]] Therefore, the NRC staff concludes that revision of categorization of IASCC for former plates from Category C to Category A is not sensitive to plant-specific information.

Further, since [[

]] the staff determined that it is unlikely that the ONS plant-specific void swelling analysis for former plates will show an increase of 50 percent void swelling and elevate its categorization from Category A to Category C. Therefore, the NRC staff concludes that revision of categorization of void swelling for former plates from Category C to Category A is not sensitive to plant-specific information.

It should be noted that determination of categorization of various degradation mechanisms other than IASCC and void swelling for former plates remains qualitative, and this determination was not based on any generic quantitative analysis from which certain parameters can be revised using ONS plant-specific information. In summary, the NRC staff concludes that the re-categorization of the former plates and the underlying generic analyses is not sensitive to plant-specific information.

The former plates were redefined as [[

]] Since the limited additional information provided in this section regarding the former plates is very general, the staff concludes that the final categorization for former plates is unlikely to be affected by any ONS plant-specific information.

<u>External BB bolts</u>

[[

]] The NRC staff reviewed the calculated values of the critical parameters in the two generic evaluations/analyses and determined that in each case, the margin in the results is not sufficient to tolerate possible changes caused by ONS plant-specific information while still supporting Category B categorization.

Nevertheless, the NRC staff concludes that even if the categorization of external BB bolts was changed from Category B to Category C after ONS plant-specific information was considered, it would have no impact on the applicability of MRP-227-A to ONS. This is because inaccessibility of the external BB bolts makes their final categorization as Primary not practical. Therefore, redefining them as Expansion [[ ]] is appropriate. Further, due to the relatively low neutron fluence level, the external BB bolts are less susceptible to degradation than the internal BB bolts and baffle-to-former (FB) bolts [[

]], and the condition of the external BB bolts can be inferred from the inspection results of internal BB bolts and FB bolts. Based on the above discussion and the rather general additional information provided in [[ ]] regarding the external BB bolts, the NRC staff concludes that the final categorization for the external BB bolts is unlikely to be affected by any ONS plant-specific information.

Inaccessible Locking Device and Locking Weld (CB Bolts and External BB Bolts)

The NRC staff reviewed [[ ]] and found no generic evaluation supporting the categorization of this item as Category B. Therefore, the NRC staff concludes that determination of categorization of inaccessible locking device and locking weld (CB bolts and external BB bolts) with respect to various degradation mechanisms remains qualitative, and this determination was not based on any generic quantitative analysis from which certain parameters can be revised using ONS plant-specific information.

Further, inaccessibility of this item makes its classification as Expansion [[ ]] appropriate. Due to the lower neutron fluence, the locking device and locking weld of CB bolts and external BB bolts are less susceptible to degradation than the locking device and locking weld of FB bolts and internal BB Bolts [[ ]], and the condition of this RVI component item can be inferred from the inspection results of the accessible locking device and locking weld of FB bolts and internal BB Bolts.

Based on the above discussion and the additional information provided in [[ ]] regarding the inaccessible locking device and locking weld, the staff concludes that the final categorization for the inaccessible locking device and locking weld is unlikely to be affected by any ONS plant-specific information.

UTS and LTS Bolts

The revised UTS and LTS bolts categorization in [[ ]] Therefore, the NRC staff concludes that determination of categorization of various degradation mechanisms for UTS and LTS bolts remains qualitative, and this determination was not based on any generic quantitative analysis from which certain parameters can be revised using ONS plant-specific information.

The UTS and LTS bolts were redefined as Expansion in [[ ]] consistent with the previous Category B categorization. Since no additional information was provided in this section regarding the UTS bolts, the staff concludes that the final categorization for the UTS bolts is unlikely to be affected by any ONS plant-specific information. [[ ]] revealed additional information that [[ ]] Since this additional information does not affect the Expansion category of the LTS bolts for ONS, Units 1, 2, and 3, the staff concludes that the final categorization for the LTS bolts is appropriate for the ONS units.

SSHT Bolts (CR-3, DB)

Not applicable to ONS.

Upper Grid Fuel Assembly Support Pads: Alloy X-750 Dowel Locking Weld (except DB)

The NRC staff reviewed [[
]] and found no generic evaluation supporting the categorization of this item as Category B. Therefore, the NRC staff concludes that determination of categorization of upper grid fuel assembly support pads (Alloy X-750 dowel locking weld) with respect to various degradation mechanisms remains qualitative, and this determination was not based on any generic quantitative analysis from which certain parameters can be revised using ONS plant-specific information.

[[

]] Since no additional information was provided in this section regarding this RVI component item, the NRC staff concludes that the final categorization for the upper grid fuel assembly support pads (Alloy X-750 dowel locking weld) is unlikely to be affected by any ONS plant-specific information.

Lower Grid Fuel Assembly Support Pad Component Items: Pad, Pad-to-Rib Section Weld, Alloy X-750 Dowel, Cap Screw, Their Locking Welds

The NRC staff reviewed [[
]] and found no generic evaluation supporting the categorization of this item as Category B. Therefore, the NRC staff concludes that determination of categorization of this RVI item with respect to various degradation mechanisms remains qualitative, and this determination was not based on any generic quantitative analysis from which certain parameters can be revised using ONS plant-specific information.

The lower grid fuel assembly support pad component items (pad, pad-to-rib section weld, Alloy X-750 dowel, cap screw, and their locking welds) were categorized as [[

]] Since the relationship of these items to the linked Primary component (the IMI guide tube spiders) is thus unlikely to be affected by any ONS plant-specific information, the NRC staff concludes that the final categorization for this RVI item as Expansion is unlikely to be affected by any ONS plant-specific information.

X-750 Bolts for Lower Grid Shock Pads (TMI-1 only)

Not applicable to ONS.

FD Bolts

[[

]] it is unlikely that the ONS plant-specific structural analyses for [[ ]] would show significantly smaller differences or reverse the trend. Therefore, designating [[ ]] in MRP-227-A is unlikely to be affected by the ONS plant-specific information.

The FD bolts were redefined as [[ ]] However, as determined by the NRC staff in Reference 14, the FD bolts are included in the Primary inspection category in MRP-227-A (Reference 11). Since no additional information was provided in this section regarding the FD bolts, the NRC staff concludes that the final categorization of the FD bolts is unlikely to be affected by any ONS plant-specific information.

3.3.2.1.2.2 <u>Licensee Action Item 1 - Conclusion</u>

Based on the evaluation above for the components applicable to ONS, the NRC staff determined that the component aging management recommendations in MRP-227-A for generic B&W-designed RVI components will not be affected by ONS plant-specific information. Therefore, the NRC staff concluded that the licensee's response to licensee action item 1, which confirmed that ONS meets the three basic assumptions outlined in MRP-227-A, Section 2.4, stated that ONS will continue to use a low-leakage core in the future, and any future power uprate will not have an effect on the ONS Inspection Plan, is sufficient to resolve Licensee Action Item 1 for ONS.

3.3.2.2 <u>Licensee Action Item 2</u>

Per Section 4.2.2, "PWR Vessel Internal Components Within the Scope of License Renewal," of Reference 14, this action item requires that, consistent with the requirements addressed in 10 CFR 54.4, each licensee is responsible for identifying the RVI components that are within the scope of LR for its facility. Licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4.

Section 4.2.6, "Fabrication Records Search," of the ONS Inspection Plan stated that the Alloy X-750 dowels, Type 304 screws, and their locking welds (related to plenum cover ribs for ONS, Unit 1 only) were unknown and were not screened for aging degradation mechanisms for inclusion in MRP-227. RAI-10 (Reference 17) requests the licensee confirm that these RVI components are within the scope of the LR in accordance with 10 CFR 54.4 and will be included in the licensee's response to address licensee action item 2 of the SE on MRP-227-A regarding whether Tables 4-1 and 4-2 in MRP-189, Rev. 1 have missed any RVI components that should be within the scope of the LR.

In Reference 4, the licensee response stated that these additional components that were not identified in MRP-227-A as within the scope of LR in accordance with 10 CFR 54.4. This resolved the first part of RAI-10. The second part of RAI-10 regarding whether Tables 4-1 and 4-2 in MRP-189, Rev. 1 have missed any RVI components is answered in a supplemental report, ANP-3186-P (Reference 6) to meet the commitment made in the licensee's letter dated June 28, 2012 (Reference 2). Reference 2 was submitted to address licensee action item 2, and to satisfy the RAI-13 (Reference 17) request to submit the resulting document to support the current NRC staff review.

The NRC staff has reviewed Reference 6 and found that the list of RVI components within the scope of LR, as defined by 10 CFR 54.4, was developed independently in the report. This list was compared with the RVI components in the MRP-189, Rev. 1 tables to identify the RVI components that were missed in the MRP report. They are vent valve miscellaneous locking device parts (original/stainless steel), vent valve miscellaneous locking device parts (modified/stainless steel, Alloy 600, and Alloy 718), and remnant of the SSHTs. These missed RVI components were then examined through the FMECA review; categorization to A, B, C, and D; engineering evaluation and assessment; and final categorization to Primary, Expansion, Existing, and No Additional Measures. This effort concluded that the appropriate categorization for the three identified RVI components was "No Additional Measures." In addition, the Alloy X-750 dowel and the Type 304 screws and their locking welds that fastened the plenum cover rib pads to the plenum cover weldment ribs (unique to ONS, Unit 1 only) were identified by a previous records search, and a similar effort resulted in designating them also as "No Additional Measures." Based on the report's description of the above four processing steps and the associated evaluation tables, the NRC staff determined that the process and the methodology leading to the final categorization of the ONS-specific RPV components is consistent with those of MRP-227-A and the MRP reports that support it. Therefore, the second part of RAI-10 is resolved, and the licensee's documentation in Reference 6 is sufficient to resolve licensee action item 2 for ONS.

3.3.2.3 <u>Licensee Action Item 3</u>

Not applicable to B&W design units.

3.3.2.4 <u>Licensee Action Item 4</u>

This action item requires that the B&W licensees confirm that the CSS upper flange weld was stress relieved during the original fabrication of the RPV 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 inspection category component.

In Reference 2, the licensee committed to provide this information by February 29, 2013. RAI-14 (Reference 17) requested the licensee provide it to the NRC to support the current review. In Reference 4, the licensee stated that B&W's stress relief documentation for the ONS, Unit 1 core support structure to upper flange weld showed that a thermal stress relief heat treatment was performed at **[[** **]]**. Information in the B&W approved Allis-Chalmers (the manufacturer for these RVI components) process outlines indicated that the core support structure to upper flange welds for the ONS, Units 2 and 3 RVI components were also thermally stress relieved at **[[** **]]** during original fabrication. Therefore, the NRC staff considers RAI-14 to be resolved, and the licensee's documentation in Reference 4 is sufficient to resolve licensee action item 4 for ONS.

3.3.2.5 <u>Licensee Action Item 5</u>

Not applicable to B&W design units.

3.3.2.6 Licensee Action Item 6

Licensee action item 6 requires licensees provide their justification for the continued operability of each of the inaccessible RVI components and, if necessary, provide their plan for the replacement of the components for NRC review and approval.

Consistent with the SE on MRP-227-A and the Draft SE on WCAP-17096, RAI-4 (Reference 17) informed the licensee that the NRC staff will impose a condition (Condition 1) on this SE, requiring the licensee to submit the detailed analysis, replacement schedule, or justification for some other alternative process, within one year of the initial inspection of the linked Primary components (the end of this period is the Implementation Date of this condition) for the NRC staff to determine whether review and approval are needed if the inspection results indicate aging triggering the expansion criteria in Table 5-1 of MRP-227-A for the following inaccessible RVI components:

- the core barrel cylinder and welds,
- the former plates, and
- the core barrel-to-former (CF) bolts, the internal and external BB bolts, and the locking devices for CF and external BB bolts.

The NRC staff determined that "other alternative process" mentioned above shall include justification of operation in the degraded condition on a generic or plant-specific basis. The one year time shall not be used as justification for delaying submittals which are required to support continued operation under the current regulations.

In Reference 4, the licensee stated that they agree with the imposition of the condition with the Implementation Date as stated above. However, after further review, the NRC staff determined that the date required for the submission of the information to resolve licensee action item 6 can be best handled using the Licensee Commitment Management Program via the commitment originally proposed in Reference 2. In Reference 9, the licensee requested to delay its response to licensee action item 6 by 1 year, requesting to submit the required information by May 31, 2015. In Reference 22, the licensee requested a further delay in its response to licensee action item 6, by an additional year to May 31, 2016. This delay was discussed with the NRC staff on May 6, 2015. This delay in providing the response to licensee action item 6 is found to be acceptable by the NRC staff based on the information the licensee provided as an attachment to Reference 22. This commitment will be listed in Section 3.3.3 of this SE. The NRC staff considers RAI-4 to be resolved, and finds the licensee's proposed commitment to submit the detailed analyses of the expansion components for licensee action item 6, acceptable.

3.3.2.7 Licensee Action Item 7

This action item requires the licensees of B&W, CE, and Westinghouse reactors to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI guide tube assembly spiders, CRGT assembly spacer castings, CE lower support columns, and Westinghouse lower support column bodies, or additional RVI components that may be fabricated from CASS, martensitic or precipitation hardened (PH) stainless steel, will maintain their functionality during the PEO. These analyses should also consider the possible loss of fracture toughness in these components due to TE and irradiation embrittlement (IE). 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 licensees shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227.

In Reference 7, the licensee submitted a response to licensee action item 7 in support of the NRC staff's review of the ONS Inspection Plan. The evaluation is intended to demonstrate (1) failure is unlikely and (2) the effect of failure on functionality is acceptable, based on flaw size, degraded material properties, and stress/distortion considerations for the following two CASS RVI components and one Type 15-5 PH component:

<u>CRGT Spacer Castings</u>

In Section 3.0, "CRGT Spacer Castings," of Reference 7, the licensee stated the following for CRGT spacer castings: **[[**

]]. By Reference 8, the licensee responded to RAI-1 confirming that appropriate loading was considered in the analysis to determine operational displacement and distortion for the CRGT, which included deadweight, preload, horizontal and vertical loss of coolant accident loads, safe shutdown earthquake loads, and cross flow loading. In the response to RAI-2 the licensee identified all sources of loading on the spacer casting with the **[[**

]]. The licensee's response also provides a description of the stress analysis on the CRGT assembly and all the cases that were analyzed, demonstrating that the qualitative functionality analysis of the CRGT spacer casting is supported by relevant quantitative structural analyses. The NRC staff considers the issues for this component to be resolved.

<u>IMI Guide Tube Spider Castings</u>

In Section 4.0, "IMI Guide Tube Spider Castings," of Reference 7, the licensee stated the following for IMI guide tube spider castings: **[[**

]] By Reference 8, the licensee responded to RAI-3 clarifying the load redistribution after spider leg weld separation. The licensee further explained in the response to RAI-3 that, even if all welds failed, as the NRC staff postulated, **[[**

]] Under this scenario, the guide tube will not move because it is welded to the flow distributor head. The licensee's response to RAI-4 provided justification for fracture toughness saturation estimation. The licensee's response to RAI-5 clarified the configurations used in the FIV analysis and presented details of the dynamic analysis (natural frequencies and mode shapes) for the IMI spider to support that **[[**

]]. The licensee's response to RAI-6 provided relevant stresses to support the conclusion that failure of one leg would not cause guide tube misalignment. The NRC staff considers the issues for this component to be resolved.

Vent Valve Retaining Rings

In Section 5.0, "Vent Valve Retaining Rings," of Reference 7, the licensee stated the following for the vent valve retaining rings: [[

]] By Reference 8, the licensee responded to RAI-7 clarifying that the replacement of retaining rings in 2012 was caused by apparent damage made during component installation, and is unrelated to thermal embrittlement. The licensee's response to RAI-8 provided the basis for the stated fracture toughness value.
Based on the above, the NRC staff considers the licensee's documentation in References 7 and 8 sufficient to resolve licensee action item 7 for ONS.

3.3.2.8 Licensee Action Item 8

This action item requires licensees make a submittal, for NRC review and approval, to credit their implementation of MRP-227, as amended by the SE on MRP-227-A (Reference 14), as an AMP and/or inspection program for the RVI components at their facility.

The licensee's LR application (ONS renewed license issued via Reference 12) was submitted before the issuance of the SE on MRP-227-A (Reference 14), and, in accordance with that SE, was required to submit only an AMP that addresses the 10 program elements on AMP XI.M16A of NUREG-1801, Rev. 2 (Reference 15), and an inspection plan that addresses the licensee action items. The licensee has satisfactorily addressed licensee action item 8 by submitting this inspection plan based on MRP-227-A, which also contains evaluation of the 10 program elements, for NRC staff review and approval.

3.3.3 Licensee Commitments

The NRC staff identified, as outlined in Section 3.3.2.6 of this SE, several analyses/evaluations supporting the ONS Inspection Plan, which are still under preparation and are required to be submitted to the NRC to determine whether their review and approval are needed. As discussed above, in Reference 4, the licensee stated that they agree with the imposition of a condition on this SE to submit the required information for licensee action item 6. However, after further review, the NRC staff determined that the date required for the submission of the information to resolve licensee action item 6 can be best handled using the Licensee Commitment Management Program via the commitment originally proposed in Reference 2. Therefore, the Commitment listed below is still open to ensure the licensee submits the final analyses/evaluations to the NRC staff for review.

Commitments

In Section 3.3.2.6 of the SE, the NRC staff stated a license commitment, requiring the licensee to submit the detailed analysis, replacement schedule, or justification for some other alternative process, by May 31, 2016, for NRC staff review, for the following inaccessible RVI components:

- the core barrel cylinder and welds,
- the former plates, and
- the CF bolts, the internal and external BB bolts, and the locking devices for the CF and external BB bolts.

The NRC staff determined that "other alternative process" mentioned above shall include justification of operation in the degraded condition. This commitment was originally listed in Reference 2. In Reference 9, and again in Reference 22, the licensee requested to delay its response to licensee action item 6. Reference 22 requested to delay the response to May 31, 2016. The NRC staff has no objection to the revised date, based on the information provided in Reference 22, and considering the lack of immediate impact on the inspection program.

3.4 Technical Evaluation Conclusion

The NRC staff has reviewed the ONS Inspection Plan in accordance with the guidance above and determined that it is consistent with the I&E guidelines of MRP-227-A, and, upon submittal of the information in support of the commitment, the licensee will have addressed all eight licensee action items specified in MRP-227-A appropriately. Consequently, Duke Energy meets the LR commitments listed in Table 2-1. Therefore, the NRC staff concludes that the ONS Inspection Plan is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (77 FR 60149). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Oconee Nuclear Station, Units 1, 2, and 3, "Proposed License Amendment Request for the Reactor Vessel Internals Inspection Plan," November 8, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML103140599).

2. Oconee Nuclear Station, Units 1, 2, and 3, "Proposed License Amendment Request for the Reactor Vessel Internals Inspection Plan," Supplement 1, June 28, 2012, (ADAMS Accession No. ML121870231).

3. Oconee Nuclear Station, Units 1, 2, and 3, "Reactor Vessel Internals License Amendment Request Status Update," September 1, 2011, (ADAMS Accession No. ML11251A160).

4. Oconee Nuclear Station, Units 1, 2, and 3, "Proposed License Amendment Request for the Reactor Vessel Internals Inspection Plan," Supplement 2, March 28, 2013 (ADAMS Accession No. ML13098A032).

5. Oconee Nuclear Station, Units 1, 2, and 3, "Proposed License Amendment Request for the Reactor Vessel Internals Inspection Plan," Supplement 3, April 18, 2013 (ADAMS Accession No. ML131190040).

6. Oconee Nuclear Station, Units 1, 2, and 3, "Proposed License Amendment Request for the Reactor Vessel Internals Inspection Plan," Supplement 4, September 27, 2013 (ADAMS Accession No. ML13275A297).

7. Oconee Nuclear Station, Units 1, 2, and 3, "Proposed License Amendment Request for the Reactor Vessel Internals Inspection Plan," Supplement 5, November 29, 2013 (ADAMS Accession No. ML13339A312).

8. Oconee Nuclear Station, Units 1, 2, and 3, "Proposed License Amendment Request for the Reactor Vessel Internals Inspection Plan," Supplement 6, March 20, 2014 (ADAMS Accession No. ML14087A172).

9. Oconee Nuclear Station, Units 1, 2, and 3, "Proposed License Amendment Request for the Reactor Vessel Internals Inspection Plan, Commitment Due Date Change for Action Item No. 6," March 20, 2014 (ADAMS Accession No. ML14086A018).

10. Oconee Nuclear Station, Units 1, 2, and 3, "Proposed License Amendment Request for the Reactor Vessel Internals Inspection Plan," Supplement 7, April 23, 2014 (ADAMS Package Accession No. ML15065A342)

11. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), EPRI Final Report 1022863, December, 2011 – Transmitted to NRC by MRP letter MRP-2011-036 dated January 9, 2012 (ADAMS Accession No. ML120170453).

12. NUREG-1723, Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3, March, 2000 (ADAMS Accession No. ML003695154).

13. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines, (MRP-227-Rev. 0), EPRI Final Report 1016596, December, 2008, - Transmitted to NRC by MRP letter number MRP 2009-04 dated January 12, 2009, (ADAMS Accession No. ML090160206).

14. USNRC, to Neil Wilmshurst, EPRI, dated December 16, 2011; Subject: Revision 1 to the Final Safety Evaluation of EPRI Report, Materials Reliability Program Report 1016596 (MRP-227), Revision 0, "PWR Internals Inspection and Evaluation Guidelines" (TAC NO. ME0680) (ADAMS Accession No. ML11308A770).

15. NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," Final Report, December, 2010 (ADAMS Accession No. ML103490041).

16. Final License Renewal Interim Staff Guidance LR-ISG 2011-04: Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors, May 28, 2013 (ADAMS Accession No. ML12270A436).

17. USNRC to Preston Gillespie, "Oconee Nuclear Station, Units 1, 2, and 3, Request for Additional Information Related to Reactor Vessel Internals Inspection Plan Based on MRP-227-A (TAC Nos. ME9024, ME9025, AND ME9026)." February 11, 2013. (ADAMS Accession No. ML13039A131).

18. WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements." December 2009. (ADAMS Accession No. ML101460157).

19. "Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189-Rev. 1)," EPRI Final Report 1018292, March, 2009, Proprietary (not publicly available, proprietary).

20. "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals (MRP-190)," EPRI Technical Report 1013233, November, 2006, (ADAMS Accession No. ML091910128).

21. "Materials Reliability Program: Aging Management Strategies for B&W Pressurized Water Reactor Internals (MRP-231-Rev. 2)," EPRI Final Report 1021028, December, 2010, Proprietary (not publicly available, proprietary).

22. Oconee Nuclear Station, Units 1, 2, and 3, "Proposed License Amendment Request for the Reactor Vessel Internals Inspection Plan," Action Item No. 6, May 28, 2015 (ADAMS Accession No. ML15159A452)

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Date: June 19, 2015

If you have any questions, please contact me at 301-415-4032.

Sincerely,

/RA/

James R. Hall, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosures:
1. Amendment No. 392 to DPR-38
2. Amendment No. 394 to DPR-47
3. Amendment No. 393 to DPR-55
4. Non-Proprietary Safety Evaluation
5. Proprietary Safety Evaluation

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OFFICE	NRR/LPL2-1/PM	NRR/LPL2-1/LA	NRR/EVIB/BC	OGC (NLO)	NRR/LPL2-1/BC	NRR/LPL2-1/PM
NAME	JWhited	SFigueroa	SRosenberg (JPoehler for)	BMizuno	RPascarelli	RHall
DATE	04/22/15	04/29/15	04/22/15	06/11/15	06/19/15	06/19/15

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