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

February 19, 2013

Mr. Preston Gillespie
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 - APPROVAL OF TIME-LIMITED AGING ANALYSIS FOR REACTOR VESSEL INTERNALS (TAC NOS. ME8436, ME8437, AND ME8438)

Dear Mr. Gillespie:

The "Application for Renewed Operating Licenses DPR-38, DPR-47, and DPR-55, for Oconee Nuclear Station, Units 1, 2, and 3," (ONS Units 1, 2, and 3 License Renewal Application (LRA)) described a time-limited aging analysis (TLAA) related to "loss of fracture toughness" of the reactor vessel internals (RVI), which is more accurately described as "loss of ductility and deformation limits" of the RVI. The original analysis was described in BAW-10008, Part 1, Revision 1 (Rev. 1), "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake." Since the data necessary to re-evaluate this TLAA through the end of the period of extended operation was not available at the time the U.S. Nuclear Regulatory Commission (NRC) staff was reviewing the ONS Units 1, 2, and 3 LRA, the licensee, Duke Energy Carolinas, LLC, made a commitment by letter to the NRC dated December 17, 1999, to perform a plant-specific analysis and develop data to demonstrate that the RVI will meet the deformation limits throughout the period of the renewed license. This commitment is described in Section 4.2.5.3 of NUREG-1723, "Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3," March 2000.

By letter dated February 20, 2012, the licensee submitted the plant-specific analysis constituting the updated TLAA for fracture toughness of RVI for NRC staff review. The plant-specific analysis includes a proposed modification of the conclusions of the original analysis in BAW-10008, Part 1, Rev.1. The NRC staff has completed its review of the licensee's plant-specific analysis and concludes that the licensee's evaluation of the TLAA for the deformation limits of the RVI is acceptable, and the associated license renewal commitment is fulfilled. Details of the staff's evaluation are contained in the enclosed Safety Evaluation.

If you have any questions, please contact me at 301-415-2901, or via email at John.Boska@nrc.gov.

Sincerely,

John P. Boska, 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

Enclosure:
As stated

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TIME-LIMITED AGING ANALYSIS FOR REACTOR VESSEL INTERNALS

DUKE ENERGY CAROLINAS, LLC

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

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

1.0 INTRODUCTION

The "Application for Renewed Operating Licenses DPR-38, DPR-47 & DPR-55, for Oconee Nuclear Station, Units 1, 2, and 3," (Reference [Ref.] 1, ONS Units 1, 2 and 3 LRA) described a time-limited aging analysis (TLAA) related to "reduction in fracture toughness" of the reactor vessel internals (RVI). The ONS Units 1, 2, and 3 LRA refers to topical report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," as containing the detailed analysis. The U.S. Nuclear Regulatory Commission (NRC)-approved version of the topical report, BAW-2248A (Ref. 2), indicates that BAW-10008, Part 1, Revision 1 (Rev. 1), "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake (Ref. 3)," documents the acceptability of the RVI under accident conditions consisting of a combination of loss of coolant accident (LOCA) and seismic loadings. The original analysis described in Appendix E to BAW-10008, Part 1, Rev.1, concluded that "at the end of 40 years, the [RVI] will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and that irradiation will not adversely affect deformation limits." Since the data related to the properties of irradiated stainless steel necessary to re-evaluate this TLAA through the end of the period of extended operation was not available at the time the NRC staff was reviewing the ONS Units 1, 2, and 3 license renewal application (LRA), the licensee, Duke Energy Carolinas, LLC, made a commitment by letter to the NRC dated December 17, 1999 (Ref. 4), to perform a plant-specific analysis and develop data to demonstrate that the RVI would meet the deformation limits at expiration of the renewed license. This commitment is described in Section 4.2.5.3 of NUREG-1723, "Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3," March 2000 (Ref. 5). The NRC staff notes that this TLAA is more accurately described as "reduction in ductility" than "reduction in fracture toughness," since the property of interest is ductility, as measured by the uniform elongation, rather than fracture toughness. The changes to both fracture toughness and ductility in stainless steels in RVI are due to the same mechanism, neutron irradiation embrittlement.

By letter dated February 20, 2012 (Ref. 6), the licensee submitted the plant-specific analysis constituting the updated TLAA for reduction in ductility of RVI for NRC staff review. The plant-specific analysis includes a proposed modification of the conclusions of the original analysis in BAW-10008, Part 1, Rev.1. The non-proprietary version of the updated analysis (Ref. 7) and the proprietary version (Ref. 8) were included in the Enclosure to Reference 6. By letter dated

November 16, 2012 (Ref. 9), the licensee responded to NRC staff requests for additional information.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Section 54.21(a)(3) requires that for each component within the scope of license renewal as defined in 10 CFR 54.4 and subject to aging management review according to the criteria of 10 CFR 54.21(a)(1)(typically described as long-lived, passive components), applicants for license renewal must demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation.

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 54, Section 54.21(c)(1) requires an evaluation of TLAAs, as defined in 10 CFR 54.3, which states that TLAAs, for the purposes of this part, are those licensee calculations and analyses that:

1) Involve systems, structures, and components within the scope of license renewal, as delineated in § 54.4(a);

2) Consider the effects of aging;

3) Involve time-limited assumptions defined by the current operating term, for example, 40 years;

4) Were determined to be relevant by the licensee in making a safety determination;

5) Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in § 54.4(b); and

6) Are contained or incorporated by reference in the current licensing basis. Section 54.21(3)b requires for each TLAA that the applicant shall demonstrate [one of the following]:

    i. The analyses remain valid for the period of extended operation;
    ii. The analyses have been projected to the end of the period of extended operation; or
    iii. The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

3.0 TECHNICAL EVALUATION

3.1 Licensee Evaluation

The licensee's reevaluation of the loss-of ductility TLAA in References 7 and 8 indicate that BAW-2248A is the generic topical report for aging management of Babcock & Wilcox (B&W) design reactor vessel internals. References 7 and 8 provide the following quote from Section 4.5.2 of BAW-2248:

BAW-10008, Part 1, Rev. 1, documents the acceptability of the reactor vessel internals under LOCA and a combination of LOCA and seismic loadings. The effect of irradiation on the material properties and deformation limits for the internals is presented in Appendix E, where it is concluded 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.

References 7 and 8 additionally indicate that BAW-2248 concluded that additional testing of the physical and mechanical property changes of irradiated material and continued surveillance of the reactor internals should be performed to provide reliable data on the irradiated properties of stainless steel.

References 7 and 8 also note that the staff's final safety evaluation (SE) of BAW-2248 identified an action item to address loss of fracture toughness on a plant-specific basis, and that resolution of this TLAA will require determination of the expected material properties at the end of the license renewal period. References 7 and 8 state that the scope of the submittal is to address the commitment in Section 3.4.2 of the NRC SE of BAW-2248 related to the conclusion for a 60-year lifetime that resolution of the TLAA will require determination of the expected material properties at the end of the license renewal period.

References 7 and 8 therefore state that the current evaluation will:

1. Update fluence of the core barrel flanges to a 60-year lifetime;
2. Examine the validity of Figure E-3 in BAW-10008;
3. Examine the "deformation limits" assumed in BAW-10008;
4. Update Appendix E to BAW-10008 to a 60-year lifetime.

Figure 2-1 of the References 7 and 8 reproduces Figure E-3 of BAW-1008, which provides curves of uniform elongation versus fluence at room temperature, at 572 °F, and at 752 °F.

References 7 and 8 provide the calculated neutron fluence at 60 years for the limiting locations for the RVI, and mechanical test data from materials irradiated in decommissioned PWRs and BWRs, and some materials irradiated in test reactors. The mechanical tests performed were normal tensile tests, slow strain rate tests (SSRT), or constant extension rate tests (CERT). The licensee intended to demonstrate that the uniform elongation values measured in the slow strain rate tests are conservative relative to uniform elongation values measure in normal tensile tests, which are conducted at a much higher strain rate. The uniform elongation results from these mechanical tests were plotted on the same graph as the curves from the original Figure E-3 of BAW-10008. This comparative plot showed that most of the test data had larger uniform elongation values than given by the lines on the graph, which the licensee stated they believe were based on irradiation of Type 304 solution annealed stainless steel (Type 304SA) in fast breeder reactors. Based on this comparison, the licensee concluded the 572 °F line is bounding for the newer measured data.

Based on their evaluation, the licensee updated the conclusion in the last paragraph of Appendix E of BAW-10008 to essentially state that the uniform elongation values (based on the data presented in References 7 and 8) corresponding to the maximum 60-year (54 EFPY) fluence in the core barrel in the region near the flanges were sufficient such that even at the end of a 60-year lifetime the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and the irradiation will not adversely affect deformation limits.

3.2 Staff Evaluation

Fluence of Core Barrel Flanges

Section 3.2 of References 7 and 8 provide the neutron flux and fluence values for the core barrel flanges. The assumptions used in determining the fluence were described, but the method used to determine the fluence was not described. Therefore, in Request for Additional Information (RAI) 1, the staff requested the licensee describe the methodology used to calculate the neutron fluence and confirm whether it is consistent with Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." In the response to NRC staff RAI 1, provided by letter dated November 16, 2012 (Ref. 9), the licensee stated that the method used was described in NRC-approved licensing topical report BAW-2241NP-A, "Fluence and Uncertainty Methodologies," Agencywide Documents Access and Management System (ADAMS) Accession No. ML073310660. Because the licensee's fluence values are (1) determined in accordance with NRC-approved methods and (2) adherent to RG 1.190 guidance, they are considered acceptable by the NRC staff.

The staff reviewed the assumptions used in calculating the neutron fluence. Specifically, the licensee assumed a high-leakage core design and used conservative capacity factor assumptions resulting in an overall capacity factor of 90 percent over the 60-year (54 EFPY) plant life. In addition, the licensee stated that the neutronics model represents an approximation of the axial material regions between the core and flange locations which would allow a higher neutron flux to reach the core barrel flanges than would be expected with a more detailed model that contained fewer approximations. The staff agrees that the capacity factor assumptions are conservative since a weighted average of the two periods would equate to a 90 percent lifetime capacity factor.

In References 7 and 8, the licensee indicated that the maximum fluence at 60 years occurs at the core barrel flanges (upper and lower). The distances from the top or bottom surface of the two flanges were provided in the proprietary version. In response to RAI 2 (Ref. 9), the licensee indicated that References 7 and 8 contained an error with regard to the location with the maximum stress intensity. The correct locations are the upper and lower core support shield flanges. The licensee provided the value of the maximum neutron fluence for the upper and lower core support shield fluences (which occurs at the lower core support shield flange).

Ductility Data for Irradiated Stainless Steel (Figure E-3 of BAW-10008)

The following information was presented in Section 3.3 of References 7 and 8:

- The Figure 2-1 data are most likely from Type 304SA irradiated in fast breeder reactors.

- A search was performed for more recent solution-annealed Type 304 stainless steel (Type 304SA) test data irradiated in the relevant range of neutron fluence.

- Data on tensile properties, including test temperature, fluence, yield strength, ultimate tensile, uniform and total elongation from materials irradiated in operating reactors and test reactors. The operating reactors included both BWRs and PWRs. The test reactors include the Halden boiling heavy water reactor, the Advanced Test Reactor, and the Japan Materials Testing Reactor. All the test reactors are light-water reactors except for the Halden reactor. Some of the data are from normal tensile tests and some of the data are from slow strain rate tests (SSRT). The uniform elongation values from the new data were plotted on a figure with the lines from Figure 2-1.

The staff reviewed the test data presented by the licensee to determine if it was irradiated and tested under conditions that are representative of the conditions that would be experienced by the core barrel flanges in the ONS 1, 2, and 3 reactors. The staff finds that the data are adequately representative because they were generally irradiated in light-water reactors, thus should be more representative than materials irradiated in fast breeder reactors, and because the new test data have accumulated neutron fluence values that are in the range of interest.

The staff noted that the majority of the uniform elongation data for Type 304SA presented in Reference 1 is from SSRT, which are performed at strain rates of around 10^{-7}/second, while standard tensile tests are performed at 10^{-2} to 10^{-4} per second. However, it is not clear how the strain rates from the different test methods relate to the strain rate that would occur in the postulated LOCA plus seismic event that is evaluated in Appendix E of Reference 4. Therefore, in RAI 2, the staff requested the licensee provide the strain rates that are assumed in the loadings evaluated in BAW-10008 Part 1, Rev. 1. The staff further requested in RAI 2 that if the strain rates for the test data are substantially different than the strain rates for the postulated event, to discuss how the test data for uniform elongation are conservative for evaluating the ductility of the RVI during the postulated event. In its response to RAI 2 (Ref. 9), the licensee noted that an error was discovered in References 7 and 8 with respect to the location of maximum stress intensity during a seismic event. The correct locations for the maximum stress intensity from Section 3.2.3.2 of BAW-10008, Part 1, Rev. 1, are the upper and lower core support shield flanges. The licensee further indicated that the neutron fluence for these locations is sufficiently low that the effect of irradiation embrittlement on the ductility of these locations is minimal. Therefore, the licensee concluded the irradiated tensile and slow strain rate tensile test data are not relevant for the upper and lower core support shield flanges. The licensee provided additional data in the form of graphs showing the effect of strain rate on the uniform elongation and the yield stress of unirradiated Type 304 stainless steel. These data covered strain rates from 10^{-5}/sec to 10^{2}/sec and show that at 600 °F, the material has greater than 20 percent uniform elongation over the entire strain range, compared to the minimum required of 8.6 percent of Appendix A of BAW-10008, Part 1, Rev. 1. Also, the yield strength increases with strain rate. Therefore, the licensee concluded Type 304 stainless steel in the upper and lower core support shield flanges will have more than adequate ductility over the entire range of strain rates, and that the original conclusions from Appendix E to BAW-10008, Part 1, Rev. 1, concerning the acceptable ductility and deformation limits for a 40-year lifetime remain valid for the 60-year lifetime for the ONS Units.

The staff was concerned that although the location of maximum stress intensity in the RVI is essentially unaffected by irradiation embrittlement, other locations with lower stress intensities but higher embrittlement may be closer to the deformation limits of BAW-10008, Part 1, Rev. 1. In a teleconference on December 12, 2012 (Ref. 10), the licensee indicated that the higher fluence locations in the RVI are subject to significantly lower stress intensities than the upper and lower core support shield flanges. The licensee also indicated that although a few RVI locations have peak stress intensities approaching the yield strength, these locations are subject to much lower fluence so irradiation effects at these locations would be insignificant. The licensee additionally indicated that the baffle plates may have stress intensities greater than the unirradiated yield strength, but due to significant irradiation hardening, the stress intensities would be below the irradiated yield strength.

By email dated January 30, 2013 (Ref. 11), the licensee provided additional information related to the core barrel flange stresses. The licensee stated that the applied stress intensity on the upper core barrel flange does not exceed the unirradiated yield strength for Type 304 stainless steel at 600 °F. The licensee further stated that [stress intensity] results for the lower core barrel flange are not included in BAW-10008, Part 1, Rev. 1 and therefore the licensee assumed the stress intensities were less than that of the upper core barrel flange, (i.e., also below the unirradiated yield strength of 304 stainless steel). The licensee also stated that since the stress intensity at the core barrel flanges is less than the unirradiated yield strength for 304 stainless steel, there is no plastic deformation and no impact due to irradiation induced change in ductility. Finally, the licensee stated that the stress intensity will not change and the yield strength for 304 stainless steel increases with irradiation; thus, the previously identified limiting locations (core barrel flanges) do not need to be evaluated in the context of the original TLAA. Based on the information from References 10 and 11, the staff considers this issue to be resolved.

Section 3.4 of References 7 and 8 reference Figure 3-11 of the report which provides uniform elongation data from Type 304SA irradiated in the EBR-II test reactor as a function of strain rate, at room temperature, at 450 °F and at 700 °F. Section 3.4 of References 7 and 8 state that uniform elongation is seen to decrease moderately with decreasing strain rate at elevated temperatures of 450 °F and 700 °F, thus the uniform elongation values from SSRT at ~10^{-7} are conservative compared to those obtained from conventional tensile tests at a strain rate typically ranging from 10^{-4} to 10^{-2}/sec. However, the staff notes that the uniform elongation data shown in Figure 3-11 are from material irradiated to approximately 1×10^{23} n/cm^2, which are much higher than the predicted end-of-life fluence for the limiting high-strain component for the ONS RVI. Additionally, the strain rates of the tests depicted in Figure 3-11 range from 3.3×10^{-5}/second to 3.3×10^{-2}/second, which do not overlap the strain rates of the SSRT. Therefore, in RAI 3, the staff requested the licensee provide further justification for the data in Figure 3-11 being representative of the behavior of Type 304SA in the ONS 1, 2, and 3 RVI, given the higher neutron fluence of the EBR-II materials and the different strain rate range of the testing. In response to RAI 3, the licensee stated that as indicated in the response to RAI 2, the locations of maximum stress intensity are the core support shield upper and lower flanges. Any decrease in ductility due to irradiation embrittlement is insignificant for these components. Therefore, the irradiated test data described in RAI 3 are no longer applicable.

Although in its responses to RAI 2 and RAI 3 the licensee stated the mechanical test data presented in References 7 and 8 are no longer relevant given the lower neutron fluence of the correct location of peak stress intensity, the staff still considers this data useful in validating that the uniform elongation versus fluence relationship presented in Figure 2-1 from BAW-10008, Part 1, Rev. 1, is conservative. The test data presented in References 7 and 8 cover a range of strain rates from 10^{-7}/sec to almost 10^{-3}/sec. Two of the data sources, representing five data points, are from standard tensile tests which have higher strain rates than SSRT or CERT tests. Further, the data on the effect of strain rate versus elongation for unirradiated Type 304 stainless steel provided in the RAI 2 response shows that uniform elongation does not change much with increasing strain rate at 600 °F, while irradiated Type 304 data on uniform elongation versus strain rate in Figure 3-11 of Reference 6 shows either an increase with strain rate (at room temperature), or approximately flat curve at higher temperatures. The data on the effect of strain rate in the RAI 2 response covers a strain range of 10^{-5}/sec to 10^{2}/sec. Therefore, unless strain rates during a design basis event are extremely high, these data can be used to assess the effect of strain rate on elongation for the RVI. Therefore, the staff concludes that the uniform elongation data in References 7 and 8 can be used to demonstrate the conservatism of the original curves for uniform elongation versus fluence presented in Figure 2-1 of BAW-10008, Part 1, Rev. 1.

The licensee indicated in Reference 11 that the non-proprietary and proprietary versions of "Update of Irradiation Embrittlement in BAW-10008 Part 1 Rev 1," would be revised in the future to reflect the response to RAI 2.

Deformation Limits Assumed in BAW-10008

The licensee summarized the deformation limits assumed in BAW-10008, Part 1, Rev. 1. Essentially, the report determined that the deformation level corresponding to 2/3 of the ultimate tensile stress (S_u), determined to be 7.6 percent plastic strain for unirradiated Type 304SA, was acceptable. For unirradiated material with minimum tensile properties, this corresponded to a stress of 42 ksi. For irradiated material, since the uniform elongation decreases as fluence increases, the amount of plastic strain corresponding to this stress level decreases as the fluence increases. The licensee states that for Type 304SA with fluence of 1×10^{20} n/cm^2 or greater, the amount of plastic strain is 3 percent or less. Therefore, the licensee concluded the uniform plastic strain requirement is self-limiting.

The staff reviewed the information presented by the licensee regarding the deformation limits. The staff agrees that provided the stress corresponding to $2/3S_u$ of the unirradiated material, or 42 ksi, is not exceeded, then the originally identified deformation limits (in terms of percent uniform elongation) will not be exceeded.

Update of Appendix E of BAW-10008

In its response to RAI 2, the licensee concluded that the original conclusions from Appendix E to BAW-10008, Part 1, Rev. 1, concerning the acceptable ductility and deformation limits for a 40-year lifetime remain valid for a 60-year lifetime for the ONS Units. The original conclusion, as provided in Section 2.2 of References 7 and 8, states:

> The maximum fluence in the core barrel in the region near flanges is much less than 10^{20} nvt (E > 1 Mev) at the end of a 40-year design lifetime. This is the region of maximum stress intensity and the region where a loss of ductility would be detrimental. As noted in figure E-3, the uniform elongation is greater than 20 for this fluence. It is concluded that even at the end of a 40-year lifetime, the internals will have adequate ductility to absorb local strain at the regions of maximum stress intensity, and the irradiation will not adversely affect deformation limits.

The response to RAI 2 provided a maximum fluence for the upper or lower core support shield flanges that is bounded by the fluence of 1×10^{20} n/cm^2 (E > 1 MeV) given in the quote above. Figure 2-1 of References 7 and 8 from BAW-10008 shows the predicted uniform elongation of Type 304SA as a function of neutron fluence. The figure has curves for room temperature, 300 °C (572 °F) and 400 °C (752 °F). Figure 3-12 of References 7 and 8 show the original curves from Figure 2-1 with the addition of the more recent experimental data. The staff notes that all of the recent experimental data lies above the 400 °C curve. The more recent data therefore validates the conservatism of the original curves. Using the uniform elongation data from the original figure, the minimum elongation at 1×10^{20} n/cm^2 (E>1 MeV) would be around 24 percent at 572 °F which is much greater than the required 7.6 percent for the RVI location with the maximum stress intensity.

Therefore, the staff finds that the data compiled by the licensee demonstrates that the reactor internals will have adequate ductility to accommodate the maximum deformation that would result from the postulated LOCA combined with a seismic event.

Disposition of the Loss of Ductility TLAA

The staff notes that the licensee did not change the disposition of the TLAA for loss of ductility. The NRC staff concluded in Reference 5 that this aging effect would be adequately managed by the RVI Aging Management Program in accordance with 10 CFR 54.31 (c)(1)(iii). The licensee by letter dated November 8, 2010 (Ref. 12) requested a license amendment to implement a RVI inspection program, and by letter dated June 28, 2012 (Ref. 13), submitted a supplement to the license amendment request including a RVI Inspection Program in accordance with "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)." Licensees referencing MRP-227-A must monitor relevant operating experience. Therefore, if new relevant data on the ductility of irradiated stainless steel is generated through operating experience at ONS or other plants, the licensee must evaluate the effect of this data on the evaluation of the loss of ductility TLAA.

4.0 CONCLUSIONS

The staff has reviewed the licensee's basis for the update of the loss of ductility TLAA for the ONS 1, 2, and 3 RVI as provided in References 7 and 8. The staff finds that:

- The licensee has projected the neutron fluence for the RVI using an acceptable methodology consistent with RG 1.190.

- The licensee compiled test data from materials irradiated in operating light-water reactors, plus the Halden test reactor. These materials were irradiated under conditions more similar to the conditions of the ONS, 1, 2, and 3 RVI and therefore should more accurately represent the behavior of the Type 304SA material in the ONS 1, 2 and 3 RVI. The newer test data, when plotted on the original graph, confirms the conservatism of the original figure E-3 from References 7 and 8.

- The licensee's evaluation of the deformation limits of BAW-10008, Part 1, Rev. 1, considering the change in tensile properties of the Type 304SA material due to irradiation, is correct.

- The licensee appropriately revised Appendix E of BAW-10008, Part 1, Rev.1 to conclude the RVI would have adequate ductility at 60 years (54 EFPY) to withstand the postulated LOCA plus seismic event.

- The disposition of the TLAA for loss of fracture toughness was not changed by this analysis. Since the NRC staff-approved disposition of this TLAA is that aging will be adequately managed in accordance with 10 CFR 54.21(c)(1)(iii), the licensee must reevaluate this TLAA if new relevant data on loss of ductility of irradiated stainless steel is generated.

Based on the above, the NRC staff concludes that the licensee's evaluation of the TLAA for loss of ductility of the RVI is acceptable, and the license renewal commitment documented in Section 4.2.5.3 of NUREG-1723 to perform a plant-specific analysis and develop data to demonstrate that the internals will meet the deformation limits at the expiration of the renewed license, is fulfilled.

5.0 REFERENCES

1. Volumes I-IV of "Application for Renewed Operating Licenses DPR-38, DPR-47 & DPR-55, for Oconee Nuclear Station, Units 1, 2, and 3." July 6, 1998 (ADAMS Legacy Accession No. 9807200148) Note, Chapter 5 containing TLAA's is available at http://www.nrc.gov/reactors/operating/licensing/renewal/applications/oconee/exhibita45.pdf

2. B&WOG License Renewal Task Force Topical Report BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," April 2000, (ADAMS Accession No. ML003708443)

3. AREVA NP Inc. Document BAW-10008, Part 1, Rev.1, "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake," June 1970.

4. Letter from M.S. Tuckman to NRC dated December 17, 1999; Subject: License Renewal Response to NRC Letter dated November 18, 1999 Oconee Nuclear Station Docket Nos. 50-269, -270, -287 (ADAMS Accession No. ML993620451).

5. 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).

6. Letter from T. P. Gillespie to NRC dated February 20, 2012, Subject: Oconee Nuclear Station (ONS), Units 1, 2, and 3 Docket Numbers 50-269, 50-270, and 50-287 - License Renewal Commitment to Submit a Time Limiting Aging Analysis for the Reactor Vessel Internals to the NRC for Review. (ADAMS Accession No. ML12053A332).

7. Areva Document No. 47-9048125-002, "Update of Irradiation Embrittlement in BAW-10008 Part 1 Rev 1," (Non-Proprietary) (ADAMS Accession No. ML12053A333

8. Areva Document No. 51-9038244, "Update of Irradiation Embrittlement in BAW-10008, Part 1, Rev. 1, Duke Energy Calculation OSC-10237 (ADAMS Accession No. ML12053A334).

9. Oconee, Units 1, 2, and 3, Enclosure 3, Response to Request for Additional Information – Proprietary, November 16, 2012 (ADAMS Accession No. ML12333A318).

10. 12/12/12 Summary of Teleconference with Duke Energy Carolinas, LLS Regarding Oconee Nuclear Station, Units 1, 2, and 3, January 28, 2013 (ADAMS Accession No. ML13024A265).

11. E-mail re: Oconee Nuclear Station, Units 1, 2, and 3, Time Limiting Aging Analysis for the Reactor Vessel Internals, TACs ME8436, ME8437, ME8438 (B. Shingleton to J. P. Boska) dated January 30, 2012 (ADAMS Accession No. ML13032A186).

12. Oconee Nuclear Site, Units 1, 2, and 3, Proposed License Amendment Request for the Reactor Vessel Internals Inspection Plan Number 2010-06, November 8, 2010 (ADAMS Accession No. ML103140599).

13. Oconee, Units 1, 2, and 3, Proposed License Amendment Request for the Reactor Vessel Internals Inspection Plan, License Amendment Request Number 2010-06 - Supplement 1, June 28, 2012 (ADAMS Accession No. ML12187A214).

Principal Contributor: J. Poehler, NRR

Date: February 19, 2013

If you have any questions, please contact me at 301-415-2901, or via email at John.Boska@nrc.gov.

Sincerely,

/RA/

John P. Boska, 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

Enclosure:
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ADAMS Accession No. ML13045A489 ***Memo dated**

OFFICE	NRR/LPL2-1/PM	NRR/LPL2-1/LA	NRR/EVIB/BC*	NRR/LPL2-1/BC	NRR/LPL2-1/PM
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DATE	2/14/13	2/15/13	2/13/13	2/19/13	2/19/13

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