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

May 28, 2015

Vice President, Operations
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1 – APPROVAL OF TIME-LIMITED AGING ANALYSIS FOR REACTOR VESSEL INTERNALS (TAC NO. MF4203)

Dear Sir or Madam:

By letter dated January 31, 2000, Entergy Operations, Inc. (the licensee), submitted its license renewal application for Arkansas Nuclear One, Unit 1 (ANO-1), which included 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. Also in its license renewal application, the licensee stated that the generic analysis in Babcock & Wilcox Owners Group License Renewal Task Force Topical Report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," demonstrated that the aging effects of the RVI components are managed adequately for the period of extended operation for a number of Babcock & Wilcox plants, including ANO-1. In April 2001, the U.S. Nuclear Regulatory Commission (NRC) staff issued NUREG-1743, "Final Safety Evaluation Report related to the License Renewal of Arkansas Nuclear One, Unit 1," which included a requirement that a plant-specific analysis be performed and submitted for review in order to demonstrate that, under loss-of-coolant accident and seismic loading, the internals have adequate ductility to absorb local strain at the regions of maximum stress intensity and that irradiation accumulated over the period of extended operation will not adversely affect deformation limits.

By letter dated May 6, 2014, the licensee submitted its plant-specific analysis constituting the updated TLAA for the loss of ductility of the RVI at ANO-1 for NRC staff review. 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 at ANO-1 is acceptable, and that the associated license renewal commitment is fulfilled. Details of the staff's evaluation are contained in the enclosed safety evaluation. A proprietary version of the safety evaluation is provided in Enclosure 1 and a redacted, non-proprietary version is provided in Enclosure 2.

Enclosure 1 to this letter contains proprietary information subject to withholding per Title 10 of the *Code of Federal Regulations* Section 2.390. When separated from Enclosure 1, this letter is DECONTROLLED.

If you have any questions, please contact me at (301) 415-1081 or by e-mail at Andrea.George@nrc.gov.

Sincerely,

Andrea E. George, Project Manager
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures:
1. Safety Evaluation (Proprietary)
2. Safety Evaluation (Non-Proprietary)

cc w/Enclosure 2: Distribution via Listserv

ENCLOSURE 2
(NON-PROPRIETARY)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TIME-LIMITED AGING ANALYSIS REPORT FOR REACTOR VESSEL INTERNALS

RELATED TO A LICENSE RENEWAL COMMITMENT

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING THE TIME-LIMITED AGING ANALYSIS REPORT FOR REACTOR VESSEL

INTERNALS RELATED TO A LICENSE RENEWAL COMMITMENT

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated January 31, 2000 (Reference 1), Entergy Operations, Inc. (the licensee), submitted its license renewal application (LRA) for Arkansas Nuclear One, Unit 1 (ANO-1), which included 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 (a material property measuring how much strain a material can withstand before breaking, measured in percent elongation) and deformation limits of the RVI. Also in its LRA, the licensee stated that the generic analysis in Babcock & Wilcox Owners Group License Renewal Task Force Topical Report (TR) BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," demonstrated that the aging effects of the RVI components are adequately managed for the period of extended operation (PEO) for a number of Babcock & Wilcox (B&W) plants, including ANO-1.

The U.S. Nuclear Regulatory Commission (NRC) staff-approved version of the report, TR BAW-2248A, dated April 2000 (Reference 2), indicates that AREVA NP, Inc. TR BAW-10008, Part 1, Revision 1, "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake," dated June 1970 (Reference 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 TR BAW-10008, Part 1, Revision 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." In the NRC staff Safety Evaluation Report (SER) for BAW-2248A dated December 9, 1999, the NRC staff included 12 LRA action items for those licensees that reference BAW-2248 in an LRA. Since the data related to the properties of irradiated stainless steel necessary to re-evaluate this TLAA through the PEO was not available at the time the NRC staff was reviewing the ANO-1 LRA, ANO-1 stated in its letter dated August 24, 2000 (Reference 4), that in response to LRA action item 12 from the NRC staff SER for BAW-2248, it would perform a plant-specific analysis to demonstrate that the RVI would meet the deformation limits during the PEO.

In May 2001, the NRC staff issued NUREG-1743, "Final Safety Evaluation Report related to the License Renewal of Arkansas Nuclear One, Unit 1" (Reference 5). In Section 3.3.2.4.2 of NUREG-1743, Item 12, the NRC staff stated that a plant-specific analysis is required to demonstrate that, under LOCA and seismic loading, the RVI have adequate ductility to absorb local strain at the regions of maximum stress intensity and that irradiation accumulated at the expiration of the renewed license will not adversely affect deformation limits.

By letter dated May 6, 2014 (Reference 6), the licensee submitted the plant-specific analysis constituting the updated TLAA regarding the loss of ductility of the RVI at ANO-1 (hereafter referred to as the ANP-3281 report). In its letter dated May 6, 2014, the licensee reaffirmed, as it had stated in its LRA, that it is implementing the generic analysis addressed in TR BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," to demonstrate that the aging effects of the RVI components are adequately managed for the PEO. The plant-specific analysis that was included in the licensee's letter dated May 6, 2014, includes a proposed modification of the conclusions of the original analysis in TR BAW-10008, Part 1, Revision 1.

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 steel components of the RVI are due to the same mechanism, which is neutron irradiation embrittlement.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 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 for the PEO.

The regulations in 10 CFR 54.3 state, in part, 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 CLB [current licensing basis].

The regulations in 10 CFR 54.21(c)(1) state that a list of TLAAs, as defined in § 54.3, must be provided. The applicant shall demonstrate that:

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.

In its review of the licensee's submittal, the NRC staff also used the guidance contained in Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," March 2001 (Reference 7), which describes methods and assumptions acceptable to the NRC staff for determining the RPV neutron fluence.

3.0 TECHNICAL EVALUATION

3.1 Licensee Evaluation

BAW-2248A is the generic topical report for aging management of B&W designed RVI. Section 4.5.2 of BAW-2248 states, in part, that

> 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.
>
> ...However as noted in Section 4.2, Examination Category B-N-3 may not be adequate to detect reduction of fracture toughness in components. A program is being implemented to manage the effects of aging due to the reduction of fracture toughness of the reactor vessel internals. This aging management program is discussed in Section 4.6. Hence, this TLAA will be resolved on a

plant-specific basis per 10 CFR 54.21(c)(1)(iii) based on the results and conclusion of the program.

The AREVA RVI topical report (Reference 3) was prepared in 1970 to analyze the RVI components to determine the magnitudes and effect of loading during a LOCA and during an earthquake. In 2010, AREVA updated Appendix E of the 1970 RVI topical report (Reference 8) to extend its applicability to a 60-year lifetime. This update identified the locations of the maximum stress intensity where a loss of ductility because of neutron irradiation would be detrimental (from Appendix E of the 1970 RVI topical report) as the core barrel flanges. However, upon reexamination of the text and stress intensity values presented in the 1970 RVI topical report, the location of highest stress intensity occurs at the core support shield (CSS) bottom flange.

In the ANP-3281 report (see Reference 6), the licensee identified that the CSS bottom flange is the region of maximum stress intensity for the RVI and this component is exposed to a projected 54 effective full power year (EFPY) neutron fluence value of **[[** **]]**. As a result, the corresponding loss of ductility (uniform elongation) due to the exposure to this neutron fluence at operating temperatures is bounded by the minimum strain of 8.6 percent as specified in Appendix A of the 1970 RVI topical report. This requirement was also addressed in Appendix E of BAW-10008, Part 1, Revision 1. The fluence of the CSS top flange would be less than the lower flange because of the increased distance from the core. Figure 5-1 of the ANP-3281 report shows that for a fluence of **[[** **]]**, the uniform elongation at both 572 degrees Fahrenheit (°F) and 752 °F temperatures for the bottom CSS flange would be expected to decrease to a value of **[[** **]]**.

Appendix E of the 1970 RVI topical report stated that material that is irradiated for 40 years of service would be expected to have a uniform elongation of 20 percent. However, according to Appendix A of the 1970 RVI topical report, the maximum allowable strain during the design basis event is 8.6 percent. In the ANP-3281 report, the licensee provided an excerpt from the 1970 RVI topical report regarding locations of highest stress intensity during limiting events:

> **[[**
>
> **]]**.

In the ANP-3281 report, as supplemented by the RAI response dated January 26, 2015, the licensee further stated that the [[

]]. Therefore, the licensee concluded that the bottom CSS flange would be evaluated within the ANP-3281 report as the region of maximum stress intensity, as required by LRA action item 12 from the NRC staff SER for TR BAW-2248A (Reference 2) and from Section 3.3.2.4.2, Item 12, of NUREG-1743 (Reference 5).

In the ANP-3281 report, with respect to the evaluation of loss of ductility of the bottom CSS flange due to neutron exposure, the licensee stated that the uniform elongation of unirradiated solution annealed (SA) Type 304 stainless steel at 600 °F is seen to only decrease slightly with increasing strain rate as shown in Figure 5-4. The licensee also stated that even at the highest tested strain rates, at 600 °F, the uniform elongation is above the 20 percent uniform elongation of irradiated material credited for 40 years in Appendix E of the 1970 RVI topical report and the 8.6 percent maximum strain specified in Appendix A of the 1970 RVI topical report. The ANP-3281 report also stated that yield strength (the amount of stress a material can withstand without permanent deformation) increases with increasing strain rate at 600 °F as shown in Figure 5-5. The ANP-3281 report further stated that, in addition to having sufficient ductility at 60 years relative to the allowables of the 1970 RVI topical report, the upper and lower CSS flanges will have greater resistance to plastic deformation at increased strain rates.

3.2 NRC Staff Evaluation

3.2.1 Evaluation of Mechanical Properties of the ANO-1 RVI Components

The NRC staff reviewed the licensee's May 6, 2014, submittal and determined that the staff's evaluation of the ANO-1 TLAA should be focused on the licensee's selection of an RVI component that is subjected to highest stress intensity during LOCA conditions. Loss of ductility occurs in stainless steel components when they are exposed to high energy neutron fluence during the PEO. In its 2010 update to the 1970 RVI topical report (Reference 8), AREVA identified that the core barrel flanges are subjected to the maximum stress intensity where a loss of ductility because of neutron irradiation would be detrimental. However, upon re-examination of the stress intensity values presented in the 1970 RVI topical report, the location of highest stress intensity occurs at the CSS bottom flange. Based on this information, the staff noted that the licensee's plant-specific TLAA evaluation (the ANP-3281 report), it included the following two RVI components: the core barrel bottom flange and CSS bottom flange. Both of these components were evaluated for the loss of ductility due to neutron radiation.

In a request for additional information (RAI) dated December 12, 2014 (Reference 9), the NRC staff requested further information related to maximum neutron fluence values and maximum stress intensity values for the core barrel bottom flange and CSS bottom flange (RAI-1 and RAI-2), as well as further information regarding the licensee's methodology, modeling, and estimation of neutron fluence for the RVI (RAI-3).

In RAI-1, the NRC staff expressed a concern regarding the value for the uniform elongation of the core barrel flange in the ANP-3281 report, namely that based on the neutron fluence value (found in MRP-189, Revision 1, Reference 10), the uniform elongation is less than the minimum required by Appendix A of the 1970 RVI technical report. In addition, the operating stress value of the core barrel flange, per Table 3-2 of MRP-189 is **[[** **]]**. Therefore, the NRC staff requested that the licensee provide a copy of Appendix A of the 1970 RVI technical report to support the staff's review. In addition, the staff requested that the licensee provide an explanation on how the core barrel would maintain its function due to **[[** **]]**. Furthermore, the staff requested that the licensee provide the maximum stress intensity value and the maximum fluence value projected for 54 EFPY at the core barrel flange during LOCA and earthquake conditions. The staff stated that the maximum stress intensity at the core barrel flanges should be considered while evaluating the functionality of the core barrel assembly.

In its RAI response dated January 26, 2015 (Reference 11), the licensee provided Appendix A of the 1970 RVI topical report to the staff for information. Additionally, the licensee stated that the maximum stress intensity for the core barrel top flange is reported in Table 1 of the 1970 RVI topical report to be **[[** **]]**, and that this stress intensity does not exceed the un-irradiated yield strength for 304 SA stainless steel at 600 °F (Appendix A, Figure A-2 and Appendix C, Table C-1 of the 1970 RVI technical report). The licensee further stated that the maximum fluence projected in the ANO-1 specific fluence analysis for the core barrel flanges at 54 EFPY is **[[** **]]**, and that this fluence value was not available during the preparation of MRP-189 Revision 1. The licensee also stated that yield strength for 304 stainless steel increases with irradiation, while the applied stress intensity will not change. Since the stress intensity at the core barrel flanges will remain below the un-irradiated yield strength, there is no plastic deformation and no impact due to irradiation induced change in ductility. Thus, the licensee concluded that the core barrel will maintain its function during LOCA and earthquake conditions.

The NRC staff reviewed the licensee's response to RAI-1 and determined that the maximum stress intensity value of the core barrel flange under **[[** **]]**. Since the core barrel flange is not subject to any plastic deformation, reduction in ductility due to neutron exposure would not result in loss of function of the core barrel flange during LOCA conditions. The NRC staff noted that the neutron fluence value of core barrel flange at ANO-1 is bounded by the neutron fluence value addressed in MRP-189, Revision 1. Therefore, the NRC staff concludes that RAI-1 is resolved.

In the ANP-3281 report, the licensee stated that the maximum stress intensity occurs at the CSS bottom flange. In comparing the neutron fluence value at this location (given in the ANP-3281 report) versus the value at the core barrel (given in the 2010 update to the 1970 RVI topical report, Reference 7), the CSS bottom flange neutron fluence is lower than that of the core barrel. Even though the fluence value at the CSS bottom flange is lower than the core barrel flange, the licensee considered that CSS bottom flange to be the most limiting component because it is exposed to the maximum stress intensity. The NRC staff noted that in Section 4.0 of the ANP-3281 report, the licensee stated that the projected neutron fluence at 54 EFPY at the CSS bottom flange for ANO-1 is **[[**

]] In RAI-2, the NRC staff requested that the licensee provide an explanation for this inconsistency. The staff noted that RAI-2 erroneously identified the CSS upper flange as the component in question, when it should have been the CSS lower flange. This mistake was corrected, the licensee responded to the question regarding the CSS lower flange, and this safety evaluation contains the correct RVI component identification.

In its RAI response dated January 26, 2015 (Reference 9), the licensee stated that at the CSS bottom flange, which is the location of maximum stress intensity, the projected 54 EFPY fluence for this location at ANO-1 is **[[** **]]**. The generic neutron fluence for screening and categorization of the CSS bottom flange is **[[** **]]**. The generic neutron fluence for screening and categorization of the CSS top flange is less than **[[** **]]**. The licensee further stated that the **[[**

]]. The licensee also stated that the CSS bottom flange neutron fluence (E > 1 MeV) used for screening and categorization was based on the B&W Owners Group preliminary fluence and temperature analyses available at the time as indicated in the flowchart for screening and categorization contained in Figure 1-2 of Reference 10, which included **[[**

]].

During its review of the licensee's submittal, as supplemented, the NRC staff noted the new plant-specific neutron fluence value for the CSS bottom flange at ANO-1 is higher than the value generically used for screening and categorization of B&W RVI components as addressed in Reference 10. However, both the original neutron fluence value and the new plant-specific neutron fluence value for the CSS bottom flange are lower than the neutron fluence values of the core barrel flanges. The stress intensity value of the CSS bottom flange, however, is higher than that of the core barrel flanges. Therefore, the NRC staff concludes that the new plant-specific fluence analysis did not change the licensee's original assessment that the CSS bottom flange is bounding for the subject TLAA evaluation. Therefore, the NRC staff

determines that RAI-2 is resolved. Details of the staff's review of the fluence evaluations are addressed in Section 3.2.2 of this safety evaluation.

Additionally, based on the deformation limits assumed in the 1970 RVI topical report (Reference 3), as updated by Reference 10, the deformation level corresponding to 2/3 of the ultimate tensile stress (S_u), is 8.6 percent plastic strain for unirradiated Type 304 SA. For unirradiated material with minimum tensile properties, this corresponded to a stress of 42 ksi. In Section 3.4 of Reference 10, AREVA stated that for irradiated material, as fluence increases, the uniform plastic strain requirement is self-limiting (i.e., the plastic strain to 2/3 S_u will always be within the uniform elongation range). For example, for Type 304 SA with fluence of 1 x 1020 n/cm^2 or greater, the amount of plastic strain is 3 percent or less. Therefore, in Section 3.4 of Reference 10, AREVA concluded that the uniform plastic strain requirement is self-limiting, which is in alignment with the original Appendix E to Reference 3.

The NRC staff reviewed Appendix A of the 1970 RVI topical report (Reference 3), which addressed the RVI deformation limits, as well as Section 5.0 of the ANP-3281 report, and concludes that if the maximum stress corresponding to 2/3 S_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. Based on the data provided in the licensee's RAI response dated January 26, 2015 (Reference 11), the NRC staff concludes that the required minimum strain value is 8.6 percent which corresponds to 2/3 S_u of the unirradiated material, or 42 ksi. **[[**

]] Based on Figure 5-2 in the ANP-3281 report, the NRC staff concludes that the CSS bottom flange, which is exposed to a neutron fluence value of **[[**

]].

Therefore, the NRC staff concludes that even though there is reduction in uniform elongation in CSS bottom flange due to exposure to neutron fluence at 54 EFPY, the resulting uniform elongation of **[[**

]]. Based on the above, the NRC staff concludes that the data provided by the licensee demonstrates that the RVI at ANO-1 will have adequate ductility to accommodate the maximum deformation that would result from the postulated LOCA combined with a seismic event over the PEO.

3.2.2 Evaluation of the Effect of Fluence on the RVI Components

Section 4.0 of the ANP-3281 report provides inputs to the licensee's evaluation of RVI components for ANO-1. Specifically, a fluence value is provided for the CSS bottom flange, which is located above the reactor core. Also in Section 4.0 of the ANP-3281 report, the licensee stated that the neutron fluence had been determined in accordance with AREVA's NRC-approved fluence analysis methodology, AREVA TR BAW-2241-A, "Fluence and Uncertainty Methodologies," dated April 30, 2006 (Reference 12). The NRC staff's safety evaluation for BAW-2241-A, which is contained in the front matter of Reference 12, documents the NRC's review and conclusion that the methods contained in BAW-2241-A adhere to the

guidance described in NRC Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," dated March 2001 (Reference 7).

The NRC staff notes, however, that the guidance contained in RG 1.190 is intended for methods used to calculate RPV neutron fluence. The guidance in RG 1.190 is not necessarily applicable to methods used to calculate neutron fluence for RVI components. For example, in the core neutron source approximation, Regulatory Position 1.2 of RG 1.190 requires[1] the use of a pin-wise source representation for peripheral fuel bundles; however, averaging is permitted for the neutron source approximation from internal fuel assemblies. This guidance is reasonable for an RPV surface neutron fluence evaluation because the peripheral fuel assemblies are the most significant contributors to the RPV surface flux. Similar guidance would not be appropriate for an RVI component located more centrally within the RPV, since the more dominant contributor to the flux would be the higher-powered central fuel assemblies.

Based on the limitation of RG 1.190 applicability, in its RAI dated December 12, 2014 (Reference 8), the NRC staff requested that the licensee provide additional justification for using the methodology in BAW-2241-A to determine the CSS fluence (RAI-3). Specifically, the NRC staff requested that the licensee: (a) address the concern discussed above, (b) discuss the uncertainty associated with the fluence estimate and explain whether the estimated uncertainty is appropriate to use the fluence value as an input to the TLAA, and (c) clarify and justify the flux synthesis methods employed in the fluence analysis.

In response to RAI-3, Item (a), the licensee clarified that the fluence values were calculated using a pin-by-pin source representation. Since the licensee used an explicit representation of the source geometry in both peripheral and central fuel assemblies, the NRC staff determined that RAI-3, Item (a) is resolved.

In response to RAI-3, Item (b), the licensee stated that "Specific uncertain was not determined for the fluence estimates at reactor vessel internals points of interest; best-estimate is considered appropriate for the application." Although RG 1.190 recommends that fluence values be expressed with an analytic uncertainty estimate, and that the estimated uncertainty be confirmed by comparison to qualification data, this guidance is generally applied for calculating shifts in reference temperature for nil-ductility transition. In contrast to this application, the licensee is correlating the fluence to changes in the tensile properties of the CSS. More specifically, the updated 54 EFPY fluence value is used to confirm that previous assessments are conservative when applied to the PEO. Since the fluence value is being used as a best-estimate parameter to confirm the conservatism of prior evaluations, the NRC staff concludes that a best-estimate fluence parameter is an appropriate value for this purpose. The

[1] The term "requires" is used, in this context, to denote that the pin-wise source representation is necessary for the method to be considered adherent to the RG. The guidance contained in RG 1.190 is not mandatory, and thus, the pin-wise source representation is not explicitly required by NRC regulations.

NRC staff also concludes, based on this consideration, that an explicit estimate of the uncertainty associated with the CSS fluence was unnecessary.

In response to RAI-3, Item (c), the licensee clarified that while the ANO-1 fluence evaluation was performed consistent with the methods described in BAW-2241-A, the evaluation also included updated modeling techniques that have improved the accuracy of the modeling. These improvements include the use of a more explicit representation of the transport problem geometry, use of more recently published nuclear data for determining the cross-sections used in the transport problem, and a finer resolution of the transport problem geometry. Although the licensee provided information justifying the applicability of the uncertainty analysis contained in BAW-2241-A, the licensee also acknowledged, in response to RAI-3, Part (b), that for RVI locations above or below the active height of the fuel, including the CSS bottom flange, there is a lack of surveillance capsules or cavity dosimetry data (qualification data). However, since the licensee justified the use of a best-estimate value in response to RAI-3, Part (b), and since the licensee has used transport modeling techniques that improve the modeling accuracy over the methods discussed in BAW-2241-A, the NRC staff concludes that the licensee's modeling techniques are acceptable for the intended application.

Based on the above, the NRC staff concludes that the licensee's estimated neutron fluence values are acceptable insofar as they support the evaluation of loss of ductility for the CSS for the PEO.

3.2.3 Disposition of the Loss of Ductility TLAA

The NRC staff notes that the licensee did not change the disposition of the TLAA for loss of ductility as addressed in the LRA (Reference 1). 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). Licensees referencing Electric Power Research Institute MRP-227-A, "Materials Reliability Program: Reactor Internals Inspection and Evaluation Guidelines," dated December 2011 (Reference 13), must monitor relevant operating experience. Therefore, if new relevant data on the ductility of irradiated stainless steel is generated through operating experience at ANO-1 or other plants, the licensee must evaluate the effect of this data on the evaluation of the loss of ductility TLAA.

4.0 CONCLUSION

The NRC staff has reviewed the licensee's basis for the update of the loss of ductility TLAA for ANO-1 as provided in the ANP-3281 report, as supplemented. The staff concludes that:

- The licensee has projected the neutron fluence for the RVI using an acceptable methodology consistent with RG 1.190.
- The licensee's evaluation of the deformation limits of BAW-10008, Part 1, Revision 1, considering the change in tensile properties of the Type 304 SA material due to irradiation, is correct.
- The licensee appropriately revised Appendix E of BAW-10008, Part 1, Revision 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 (in NUREG-1743) 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 at ANO-1 is acceptable, and the license renewal commitment documented in Section 3.3.2.4.2, Item 12, of NUREG-1743 to perform a plant-specific analysis and develop data to demonstrate that the RVI components will meet the deformation limits at the expiration of the renewed license, is fulfilled.

5.0 REFERENCES

1. Hutchinson, C. Randy, Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Arkansas Nuclear One – Unit 1, Docket No. 50-313, License No. DPR-51, License Renewal Application," dated January 21, 2000 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML003679667).

2. Babcock & Wilcox Owners Group 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., Topical Report BAW-10008, Part 1, Revision 1, "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake," June 1970.

4. Vandergrift, Jimmy D., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "License Renewal Applications RAIs (TAC No. MA8054)," dated August 24, 2000 (ADAMS Accession No. ML003746995).

5. U.S. Nuclear Regulatory Commission, NUREG-1743, "Safety Evaluation Report Related to the License Renewal of Arkansas Nuclear One, Unit 1," May 2001 (ADAMS Package Accession No. ML011640298).

6. Pyle, Stephenie L., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Time-Limited Aging Analysis Regarding Reactor Vessel Internals Loss of Ductility for Arkansas Nuclear One, Unit 1 at 60 Years," dated May 6, 2014 (portions of this document are proprietary and withheld under 10 CFR 2.390; a publicly available version is at ADAMS Accession No. ML14126A816).

7. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Reactor Pressure Vessel Neutron Fluence," dated March 2001 (ADAMS Accession No. ML010890301).

8. AREVA NP, Inc., Document No. 47-9048125-002, "Update of Irradiation Embrittlement in BAW-10008 Part 1 Rev 1," (portions of this document are proprietary and withheld under 10 CFR 2.390; a publicly available version is at ADAMS Accession No. ML12053A333).

9. George, Andrea, U.S. Nuclear Regulatory Commission, email to David Bice, Entergy Operations, Inc., "Request for Additional Information - ANO, Unit 1 - Time-Limited Aging Analysis Regarding Reactor Vessel Internals - TAC NO. MF4203," dated December 12, 2014 (transmittal email only due to enclosure containing proprietary information withheld under 10 CFR 2.390, available at ADAMS Accession No. ML14349A007).

10. Electric Power Research Institute, Report No. 1018292, "Materials Reliability Program: Screening, Categorization, and Ranking of B&W –Designed PWR Internals Component Items (MRP-189-Revision 1)," dated March 4, 2009 (non-publicly available, abstract available at http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000000001018292).

11. Pyle, Stephenie L., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Responses to Request for Additional Information, Time Limited Aging Analysis Regarding Reactor Vessel Internals Loss of Ductility for Arkansas Nuclear One, Unit 1 at 60 Years," dated January 26, 2015 (ADAMS Package Accession No. ML15028A495).

12. AREVA NP, Inc., Topical Report BAW-2241P-A, Revision 2, "Fluence and Uncertainty Methodologies," dated April 30, 2006 (ADAMS Accession No. ML073310660).

13. Electric Power Research Institute, MRP-227-A, "Materials Reliability Program: Reactor Internals Inspection and Evaluation Guidelines," dated December 2011 (ADAMS Package Accession No. ML120170453).

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Date: May 28, 2015

If you have any questions, please contact me at (301) 415-1081 or by e-mail at Andrea.George@nrc.gov.

Sincerely,

/RA/

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Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures:
1. Safety Evaluation (Proprietary)
2. Safety Evaluation (Non-Proprietary)

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