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

February 13, 2017

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

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1 – STAFF ASSESSMENT REGARDING PROGRAM PLAN FOR AGING MANAGEMENT FOR REACTOR VESSEL INTERNALS (CAC NO. MF4201)

Dear Sir or Madam:

By letter dated May 20, 2014 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML14141A553), Entergy Operations, Inc. (Entergy, the licensee) submitted an aging management program (AMP) for Arkansas Nuclear One, Unit 1 (ANO-1) reactor vessel internals (RVI). The submittal was supplemented by letters dated February 10, September 28, and December 30, 2015; and May 24, 2016 (ADAMS Accession Nos. ML15043A102, ML15278A022, ML16004A183, and ML16147A330, respectively). The ANO-1 RVI AMP was developed based on the U.S. Nuclear Regulatory Commission (NRC)-approved topical report Material Reliability Program (MRP)-227-A, "Material Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." The RVI AMP was submitted to fulfill a required action that originated from license renewal activities, the provisions of which are identified in ANO-1 License Condition 2.C(3) and described in the ANO-1 Updated Final Safety Analysis Report (UFSAR), Section 16.1.5, "Reactor Vessel Internals Aging Management." ANO-1 UFSAR Section 16.1.5 states, in part, that the licensee would submit the RVI AMP to the NRC staff. Therefore, this provision of the required action was fulfilled upon submittal of the RVI AMP on May 20, 2014.

The NRC staff has completed its review of the ANO-1 RVI AMP, and concludes that it is acceptable because it is consistent with the inspection and evaluation guidelines of MRP-227-A. The licensee has adequately addressed all eight action items specified in MRP-227-A.

The NRC staff's approval of the ANO-1 RVI AMP does not reduce, alter, or otherwise affect the current American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI inservice inspection requirements, or any ANO-1 licensing basis requirements related to inservice inspection of structures, systems, and components. The staff notes that Section 7, "Implementation Requirements," of MRP-227-A, requires that the NRC be notified of any deviations from the "needed" requirements.

NOTICE: Enclosure 1 to this letter contains Proprietary Information. Upon separation from Enclosure 2, this letter is DECONTROLLED.

The NRC's staff assessment (SA) of the ANO-1 RVI AMP is enclosed. The NRC staff has determined that the SA contains proprietary information pursuant to Title 10 of the *Code of Federal Regulations*, Section 2.390, "Public inspections, exemptions, requests for withholding." Proprietary information (Enclosure 2) is indicated by **bold** text enclosed within **[[double brackets]]**. Accordingly, the NRC staff has also prepared a redacted publicly available, non-proprietary version of the SA.

If you have any questions concerning this matter, please contact the Project Manager, Tom Wengert, at (301) 415-4037, or via e-mail at Thomas.Wengert@nrc.gov.

Sincerely,

RJPascarelli

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

Docket No. 50-313

Enclosure:
1. Proprietary Staff Assessment
2. Non-Proprietary Staff Assessment

cc w/o Enclosure 1: Distribution via Listserv

ENCLOSURE 2
(NON-PROPRIETARY)

STAFF ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE AGING MANAGEMENT PROGRAM

FOR REACTOR VESSEL INTERNALS

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

UNITED STATES
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STAFF ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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FOR REACTOR VESSEL INTERNALS

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated May 20, 2014 (Reference 1), Entergy Operations, Inc. (Entergy, the licensee), submitted an aging management program (AMP) for the Arkansas Nuclear One, Unit 1 (ANO-1), reactor vessel internals (RVI). The submittal was supplemented by letters dated February 10, September 28, and December 30, 2015; and May 24, 2016 (References 2, 3, 4, and 5, respectively). The ANO-1 RVI AMP was developed based on the U.S. Nuclear Regulatory Commission (NRC)-approved Electric Power Research Institute (EPRI) topical report, Materials Reliability Program (MRP)-227-A, "Material Reliability Program: Pressurized Water Reactor [PWR] Internals Inspection and Evaluation Guidelines" (Reference 6). The RVI AMP was submitted to fulfill a required action that originated from license renewal activities, the provisions of which are identified in ANO-1 License Condition 2.C(3) and described in the ANO-1 Updated Final Safety Analysis Report (UFSAR), Section 16.1.5, "Reactor Vessel Internals Aging Management." This UFSAR Section was originally provided as part of the ANO-1 license renewal application (Reference 7). On May 2001, the NRC staff issued NUREG-1743, "Safety Evaluation Report Related to the License Renewal of Arkansas Nuclear One, Unit 1" (Reference 8), documenting its findings related to compliance with the regulatory requirements for license renewal addressed below. The ANO-1 UFSAR, Section 16.1.5 states, among other things, that Entergy would submit the RVI AMP to the NRC on, or about, the end of the initial 40-year operating license term, corresponding to May 20, 2014. Therefore, this required action was fulfilled upon submittal of the RVI AMP to the NRC on May 20, 2014.

By letter dated June 22, 2011 (Reference 9), the NRC issued the first version of its safety evaluation (SE) for Revision 0 of the MRP-227 report (Reference 10). On July 21, 2011, the NRC issued Regulatory Issue Summary 2011-07, "License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management" (Reference 11), to provide guidance to PWR license renewal applicants and renewed license holders for the submittal of plant-specific AMPs for the RVI components. On December 16, 2011, the NRC issued Revision 1 of its SE for the MRP-227 report, which is included in MRP-227-A (Reference 6).

Portions of the February 10, 2015, September 28, 2015, December 30, 2015, and May 24, 2016, letters contain proprietary information and, therefore, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 2.390, "Public inspections, exemptions, requests for withholding," those portions have been withheld from public disclosure. In addition, the September 28, 2015, submittal of AREVA Document ANP-3417NP, Revision 0, "MRP-227-A Applicant/Licensee Action Item #7 was superseded by Revision 1 of that document, submitted by letter dated May 24, 2016.

2.0 REGULATORY EVALUATION

Part 54 of 10 CFR addresses the requirements for plant license renewal. Section 54.21 of 10 CFR, "Contents of application-technical information," requires that each application for license renewal contain an integrated plant assessment (IPA). 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 will be adequately managed so that their intended functions will be maintained consistent with the current licensing basis for the period of extended operation (PEO), as required by 10 CFR 54.29(a). Structures and components subject to an AMR are defined in 10 CFR 54.4, "Scope." These structures and components are generally referred to as "passive" and "long-lived" structures and components.

The NRC staff's final SE for MRP-227 specifies seven generic conditions for the topical report and eight action items that must be addressed on a plant-specific basis by those utilizing the topical report as the basis for an RVI AMP submittal to the NRC. On January 9, 2012, EPRI issued the NRC-approved version of the topical report, MRP-227-A (Reference 6), which incorporates Revision 1 of the final SE. MRP-227-A addresses the seven generic conditions established in the SE and provides the technical basis for the development of plant-specific AMPs for managing the effects of aging on RVI components. MRP-227-A also provides specific inspection and evaluation guidelines for PWR license renewal applicants and renewed license holders to use in their plant-specific AMPs. The aging management activities described in MRP-227-A are intended for use by licensees in meeting the conditions of the license renewal commitments related to aging management of the RVI components.

The scope of components considered for inspection under the guidance of MRP-227-A 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). The scope also includes RVI components that serve an intended safety function consistent with the criteria in 10 CFR 54.4(a)(1), and any nonsafety-related RVI components whose failure could impact the intended functions of a safety-related component that was included under 10 CFR 54.4(a)(1) and serve an intended function as defined in 10 CFR 54.4(b). 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).

In December 2010, the NRC published NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report – Final Report" (Reference 12), providing new generic AMR line items and generic AMP criteria in Chapter XI.M16A, "PWR Vessel Internals" (GALL Report AMP XI.M16A). GALL Report AMP XI.M16A was based on expectations for the guidance to be provided in MRP-227-A. Since the GALL Report, Revision 2, was published prior to the issuance of the SE for MRP-227-A, the NRC published license renewal Interim Staff Guidance (ISG) in LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal

Components for Pressurized Water Reactors" (Reference 13), which modified the criteria of GALL Report AMP XI.M16A to be consistent with MRP-227-A.

3.0 TECHNICAL EVALUATION

The licensee's May 20, 2014, RVI AMP submittal for ANO-1 provided the following information for NRC staff review and approval:

- A description of the RVI AMP attributes, based on the ten AMP elements described in the GALL Report AMP XI.M16A. This is addressed in Section 3.1 of this staff assessment (SA).
- A description of the RVI AMP inspection plan and evaluation criteria, based on MRP-227-A. This is addressed in Section 3.2 of this SA.
- A discussion of the operating experience (OE) for the ANO-1 RVI components. This is addressed in Section 3.3 of this SA.
- A discussion of the plant-specific applicability of MRP-227-A to ANO-1 based on the licensee's responses to the eight action items – hereafter designated Action Item 1 through Action Item 8 – as established in Section 4.2 of the MRP-227-A SE. This is addressed in Section 3.4 of this SA.

As part of its review, the NRC staff issued a request for additional information (RAI), dated December 12, 2014 (Reference 14) to address specific technical issues. Specific RAI questions are identified non-sequentially in this SA using the same number as that applied in the outgoing RAI correspondence (Reference 14) (i.e., RAI-8, RAI-11, etc.). The licensee provided responses to the staff's RAI questions in its supplemental correspondence dated February 10, September 28, and December 30, 2015; and May 24, 2016 (References. 2, 3, 4, and 5, respectively).

3.1 Reactor Vessel Internals Aging Management Program Attributes

3.1.1 Licensee Evaluation of Program Attributes

In Section 2.3 of the ANO-1 RVI AMP submittal, the licensee evaluated each of the ten AMP Program Elements against the corresponding elements in GALL Report AMP XI.M16A. The licensee determined that its RVI AMP for ANO-1 is consistent with GALL Report AMP XI.M16A.

3.1.2 NRC Staff Assessment of Program Attributes

The NRC staff reviewed the licensee's AMP against the ten elements of the revised version of GALL Report AMP XI.M16A, as provided in LR-ISG-2011-04. The staff determined that the ten elements of the ANO-1 RVI AMP are consistent with the ten elements described in LR-ISG-2011-04. Therefore, the staff finds the licensee's implementation of the ten AMP elements acceptable for ANO-1.

3.2 Reactor Vessel Internals Inspection Plan and Evaluation Criteria

3.2.1 Licensee Evaluation of Inspection Plan and Evaluation Criteria

The licensee's inspection plan for managing the effects of aging on the ANO-1 RVI components is described in Sections 2, 4, and 5 of its RVI AMP submittal and is based on the implementation of MRP-227-A. The licensee also referenced the MRP companion document MRP-228, "Materials Reliability Program: Inspection Standard for PWR Internals," which contains requirements specific to the inspection methodologies involved, as well as requirements for qualification of the non-destructive examination (NDE) systems used to perform these inspections.

The licensee stated that the ANO-1 RVI AMP will not deviate from the MRP-227-A inspection and evaluation guidelines, and the RVI component inspections will be performed in accordance with MRP-227-A. The licensee referenced the applicable Babcock & Wilcox (B&W) plant inspection tables contained in Tables 4-1 and 4-4 of MRP-227-A for the Primary and Expansion components, which are included in the licensee's RVI AMP submittal as Tables 5-1 and 5-2 for the Primary and Expansion components, respectively. In addition to the MRP-227-A Primary and Expansion component inspections, the ANO-1 RVI AMP submittal also includes more recent Existing Program inspection criteria for B&W vent valve locking device components in Table 5-3. This table is not included in MRP-227-A, but was later added to B&W plants' RVI AMPs based on interim industry guidance.

The applicable RVI component examination acceptance and expansion criteria for B&W plants are contained in Table 5-1 of MRP-227-A. This table is included as Table 5-4 of the ANO-1 RVI AMP submittal. The licensee stated that the methodology used to perform engineering evaluations to determine the acceptability of a detected condition shall be conducted in accordance with an NRC-approved methodology.

3.2.2 NRC Staff Assessment of Inspection Plan and Evaluation Criteria

The MRP-227-A inspection and evaluation guidelines consider the effects of eight age-related degradation mechanisms on the integrity of the RVI components. The eight age-related degradation mechanisms are: stress corrosion cracking (SCC), irradiation-assisted stress corrosion cracking (IASCC), fatigue, irradiation embrittlement (IE), thermal embrittlement (TE), wear, void swelling, and irradiation-assisted stress relaxation (ISR). The MRP-227-A inspection guidelines prescribe RVI component examinations to detect the various aging effects that are associated with the eight degradation mechanisms. The aging effects that are addressed by these examinations include cracking (due to SCC, IASCC, and/or fatigue), loss of material (due to wear), component distortion (due to void swelling), loss of fracture toughness (due to TE and/or IE), and loss of bolt preload (due to ISR).

MRP-227-A prescribes inspections of PWR RVI components based on a categorization of the components using the results of the MRP's Failure Modes, Effects, and Criticality Analyses (FMECA) for PWR internals. The FMECA for B&W plants are documented in EPRI MRP topical report MRP-189, Revision 1, "Material Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items," March 2009 (EPRI proprietary information, Reference 15). The public version of the FMECA B&W plants is provided in MRP topical report MRP-190 (Reference 16). Topical report MRP-189, Revision 1, screened the RVI components for susceptibility to the eight degradation mechanisms and

evaluated the degree of tolerance of the components' safety-related function to the associated aging effects. The FMECA results were applied to the RVI components to determine their final categorization to establish the necessary inspection and evaluation criteria for aging management. Based on this categorization for each of the three PWR designs, MRP-227-A places the RVI components into one of four functional groups: Primary components, Expansion components, Existing Programs components, and No Additional Measures components. Each of these functional groups is defined in Section 3.3.1 of MRP-227-A, and the specific inspection requirements are tabulated in Chapter 4 of MRP-227-A. Components in the No Additional Measures group require no augmented inspection activity because they were either generically determined to be not susceptible to any of the eight age-related degradation mechanisms, or the effect of an aging mechanism was determined to have negligible impact on safety function, per the FMECA

For B&W plants, the inspection guidelines for Primary and Expansion components are provided in MRP-227-A Tables 4-1 and 4-4, respectively. The NRC staff reviewed the licensee's Primary and Expansion component inspection criteria, provided in Tables 5-1 and 5-2 of the ANO-1 RVI AMP submittal, and determined that they are generally the same as those specified in Tables 4-1 and 4-4 in MRP-227-A, with one exception pertaining to the relocation of "Note 1" criteria for inservice testing and examinations of the core support shield vent valves from Table 5-1 (as compared to Table 4-1 of MRP-227-A) to Table 5-3 of the ANO-1 RVI AMP.

MRP-227-A does not identify any Existing Program components and associated inspection guidelines for B&W plants. However, the licensee identified that after the publication of MRP-227-A, AREVA, Inc. issued subsequent industry guidance in AREVA Letter No. AREVA-13-01501, "Recommended Examination of Vent Valve Locking Devices for B&W Nuclear Units," which specified VT-3 visual examinations of core support shield assembly vent valve locking devices. The licensee stated that these visual examinations were recommended for inclusion in B&W RVI AMPs as Existing Program component inspections based on recent industry OE with degradation of the vent valve locking device parts. These Existing Program component inspections are specified in Table 5-3 of the ANO-1 RVI AMP. In RAI-8 (Reference 14) the NRC staff requested that the licensee confirm whether the only modification to Table 5-1 of the AMP is the relocation of "Note 1" to Table 5-3. The staff also requested in RAI-10 (Reference 14) that the licensee identify the plant Technical Specification (TS) or inservice testing (IST) program requirements for the additional testing and examination of the vent valve components, which are specified in "Note 1" of Table 5-3 of the ANO-1 RVI AMP.

In its response to RAI-8 by letter dated February 10, 2015 (Reference 2), the licensee confirmed that the interim guidance provided to B&W plants in May 2013 by AREVA, Inc. included the recommendation to relocate "Note 1" of Table 4-1 of MRP-227-A (Table 5-1 of the ANO-1 RVI AMP) to the new Existing Programs table provided in Table 5-3 of the ANO-1 RVI AMP. In its response to RAI-10 by letter dated February 10, 2015, the licensee stated that the vent valve testing and examination provisions specified in Note 1 of Table 5-3 are implemented under the ANO-1 IST program. The NRC staff determined that the licensee's RAI responses adequately address these vent valve inspections, and they are an appropriate enhancement to MRP-227-A.

Table 4-1 of MRP-227-A, specifies that the initial volumetric examination by ultrasonic testing (UT) of the core support shield assembly upper core barrel bolts shall be performed within two refueling outages from January 1, 2006, or the next 10-year inservice inspection (ISI) interval, whichever is first. Table 5-1 of the ANO-1 AMP states that the UT examination of the upper

core barrel bolts is to be performed during Refueling Outage 26 in fall 2016. In RAI-11 (Reference 14), the NRC staff requested that the licensee clarify whether the Refueling Outage 26 upper core barrel bolt UT examination is an initial examination or a subsequent examination. If this initial UT examination has already been performed, per MRP-227-A, Table 4-1, the staff requested that the licensee discuss the examination results for these items.

In its response to RAI-11 by letter dated February 10, 2015, the licensee confirmed that the upper core barrel bolt UT examination scheduled for Refueling Outage 26 is a subsequent examination. The licensee stated that the initial UT examination was performed during the fall 2008 outage per the MRP-227-A Table 4-1 inspection guidelines, which were still under development at that time. The licensee stated that the 2008 UT examination of the upper core barrel bolts revealed **[[ ]]** rejectable indications. The NRC staff determined that the licensee's 2008 UT examination of the upper core barrel bolts provides adequate assurance of upper core barrel bolting integrity until the next scheduled UT examination for Refueling Outage 26 (fall 2016).

Table 4-1 of MRP-227-A, specifies that the initial UT examination of the lower core barrel bolts and flow distributor assembly bolts shall be performed during the next 10-year ISI interval from January 1, 2006. Table 5-1 of the ANO-1 AMP states that these UT examinations are to be performed during Refueling Outage 26. In RAI-12 (Reference 14), the NRC staff requested that the licensee state whether the initial UT examination of these bolts has already been completed prior to entering the PEO on May 20, 2014, or address whether the upcoming Refueling Outage 26 examination satisfies the initial UT requirements for these items, per MRP-227-A Table 4-1. If the initial UT examination of these bolts had already been performed, the staff requested that the licensee discuss the examination results for these items.

In its response to RAI-12 by letter dated February 10, 2015 (Reference 2), the licensee stated that the UT examination of the lower core barrel bolts scheduled for Refueling Outage 26 is a subsequent examination because the initial UT examination was performed during Refueling Outage 21 in 2008. The licensee stated that UT examination of the flow distributor assembly bolts scheduled for Refueling Outage 26 is the initial UT examination. The licensee explained that the schedule for these initial UT examinations satisfies the initial inspection schedule requirements of MRP-227-A Table 4-1 in both cases. The licensee stated that the 2008 UT examination of the 108 original Alloy A-286 lower core barrel bolts revealed **[[ ]]** rejectable indications. The licensee's evaluation of these **[[ ]]** indications is addressed in its response to RAI-1 and RAI-2, regarding plant-specific OE related to cracking in RVI bolting. This plant-specific OE and associated corrective actions are discussed in Section 3.3 of this SA. The NRC staff confirmed that these flow distributor and lower core barrel bolt inspections are being correctly implemented in accordance with the MRP-227-A, Table 4-1, Primary component inspection guidelines.

The NRC staff also reviewed the licensee's RVI examination acceptance and expansion criteria provided in Table 5-4 of the ANO-1 RVI AMP submittal and determined that they are the same as those specified in MRP-227-A, Table 5-1. Therefore, the staff determined that the licensee's RVI examination acceptance and expansion criteria are acceptable because they are consistent with MRP-227-A.

3.2.3 NRC Staff Conclusion for Inspection Plan and Evaluation Criteria

Based on its review of the ANO-1 RVI AMP inspection plan and evaluation criteria, the NRC staff finds that the ANO-1 RVI AMP will implement all Primary and Expansion component inspections in accordance with MRP-227-A, Tables 4-1 and 4-4, respectively, and it will implement the examination acceptance and expansion criteria in accordance with MRP-227-A, Table 5-1. Therefore, the staff finds the licensee's plan to implement the MRP-227-A inspection and evaluation guidelines acceptable.

3.3 <u>Plant-Specific Reactor Vessel Internals Operating Experience</u>

3.3.1 Licensee Evaluation of Operating Experience

In Section 2.3.10, "NUREG-1801/AMP Program Element 10: Operating Experience," of the ANO-1 RVI AMP submittal, the licensee provided a discussion of industry and plant-specific RVI OE. The licensee stated that plant-specific RVI component degradation OE includes cracking of the thermal shield bolting and core barrel bolting fabricated from Alloy A-286; these failures were attributed to intergranular SCC (IGSCC) and were not detected by visual examinations. The licensee also noted that recent RVI inspections at Oconee Nuclear Station, Unit 1 identified a failed vent valve jack screw. The licensee stated that, other than these failures, its review of OE did not identify any new aging degradation issues related to the ANO-1 RVI.

3.3.2 NRC Staff Assessment of Operating Experience

Based on its review of the licensee's discussion of plant-specific RVI OE, the NRC staff determined that additional information would be needed regarding the findings and corrective actions associated with RVI component degradation that occurred at ANO-1. The staff noted that Appendix A of MRP-227-A, discusses industry OE related to aging degradation of PWR RVI components, including B&W plants. In RAI-1 (Reference 14), the staff requested that the licensee provide the following information regarding this OE:

(a) Identify the MRP-227-A, Appendix A, OE that was contributed by ANO-1.

(b) Provide any ANO-1 plant-specific OE relevant to age-related degradation of RPV internal components that was not discussed in Appendix A of MRP-227-A.

In its response to RAI-1, Part (a) (RAI-1(a)) by letter dated February 10, 2015 (Reference 2), the licensee identified all RVI component degradation OE that occurred at ANO-1. Considering the numerous failures of thermal shield and core barrel bolting fabricated from Alloy A-286 that occurred at B&W units in the 1980s, the licensee provided a table summarizing the IGSCC indications found during these earlier UT examinations at ANO-1 and associated bolting replacement corrective action. This table also identifies more recent indications of cracking (and thus **[[ ]]** failure) in **[[ ]]** lower core barrel bolts found during UT examinations performed in 2008. Based on its review of this data, the NRC staff determined that the licensee performed comprehensive bolting replacement corrective actions in the 1980s by replacing **[[ ]]** bolting in the upper core barrel and lower thermal shield, which had exhibited bolting degradation at that time. Subsequent UT examinations performed for the

upper core barrel and lower thermal shield regions with replaced bolting have identified **[[ ]]** failed bolts since the 1980s replacement activity.

Therefore, the NRC staff determined that the MRP-227-A, Table 4-1, UT examinations for these primary components will provide adequate assurance of bolting integrity during the PEO for the upper core barrel and lower thermal shield regions.

The **[[ ]]** failed lower core barrel bolts found as a result of UT examinations in 2008 were all original bolts since no lower core barrel bolts have been replaced at ANO-1 to date, and **[[ ]]** lower core barrel bolt failures had been found during earlier UT examinations. The licensee provided more detailed information regarding these 2008 lower core barrel bolt inspection results and its associated structural integrity evaluation for this bolting in its response to RAI-2 discussed below.

The licensee stated that the original locking devices for the four core support shield vent valves near the outlet nozzles were replaced with redesigned locking devices to correct the vibration issues that were identified for the original locking device design. The NRC staff determined that the licensee's previous vent valve replacement activities at ANO-1 and inspection of the vent valve locking devices, per Table 5-3 of the ANO-1 RVI AMP submittal, is acceptable for aging management of these components.

In its response to RAI-1(b) by letter dated February 10, 2015 (Reference 2), the licensee stated that other than the bolting and vent valve locking device issues, described in response to RAI-1(a), there are no other plant-specific RVI aging degradation issues, and this considers the industry RVI degradation OE for B&W units addressed in MRP-227-A, Appendix A. Based on its review of this information, the NRC staff determined that currently, the only relevant plant-specific OE related to aging degradation of RVI components at ANO-1 is the reported IGSCC of core barrel and thermal shield bolting.

In RAI-2 (Reference 14), the NRC staff requested additional information related to the licensee's findings of IGSCC in the ANO-1 thermal shield bolting and core barrel bolting fabricated from Alloy A-286. RAI-2(a) concerns the identification of plant-specific RVI OE and was resolved based on its review of the licensee's response to RAI-1 above. Therefore, the staff's assessment of the licensee's actions related to the ANO-1 bolting degradation starts with RAI-2(b).

<u>RAI-2(b)</u>: The NRC staff requested that the licensee provide the details regarding the inspections, which discovered IGSCC in the ANO-1 thermal shield and core barrel bolting. In its response to RAI-2(b) by letter dated February 10, 2015 (Reference 2), the licensee cited its response to RAI-1(a). The staff confirmed that the licensee's response to RAI-1(a) provides this information. Based on its review of the licensee's discussion of upper core barrel and lower thermal shield bolting replacement activity performed in the 1980s, the staff determined that the latest 2008 UT indications of **[[ ]]** failed lower core barrel bolts is currently the only plant-specific RVI component degradation OE of concern for ANO-1. The staff's assessment of the licensee's corrective actions for the 2008 lower core barrel bolting indications is addressed below, based on review of the licensee's responses to parts (c) and (d) of RAI-2. It should be noted that parts (e), (f), and (g) of RAI-2 specifically pertain to bolting replacement corrective actions for the 1980s inspection results and were resolved based on the staff's review of the licensee's response to RAI-1. Therefore, the licensee responses to parts (e), (f) and (g) of RAI-2 will be not be repeated for this assessment.

RAI-2(c): The NRC staff requested that the licensee identify the specific AMPs under which the 2008 bolting UT inspections were conducted. The licensee identified that the 2008 core barrel bolting inspections were performed in response to the Nuclear Energy Institute (NEI) guidelines in NEI 03-08, "Guideline for the Management of Materials Issues," Revision 2, dated January 2010, which were considered during the development of MRP-227, Revision 0. The staff determined that the NEI 03-08 protocol was appropriate for any RVI component inspections that occurred prior to the issuance of MRP-227-A.

RAI-2(d): The NRC staff requested that the licensee discuss any operability evaluations that were performed for the cracked bolting, including consideration of inspection results, assumptions made for future bolt failures, or minimum bolting pattern analyses. In its response to RAI-2(d) (Reference 2), the licensee described its evaluation of the lower core barrel bolts, considering the 2008 UT indications of **[[** **]]** lower core barrel bolt failures due to IGSCC. The licensee's evaluation consisted of the following:

- probabilistic analysis of potential future lower core barrel bolt IGSCC based on the past UT results and bolt condition;

- structural analysis of the impact of potential additional lower core barrel bolt failures on compliance with the structural margins specified in the **[[** **]]**.

The licensee stated that it has not replaced any of the **[[** **]]** lower core barrel bolts with IGSCC indications, however the next UT examination of these items is scheduled for Refueling Outage 26 (fall 2016). The licensee noted that its review of the 2008 lower core barrel bolt UT examination results determined that **[[** **]]**. However, the licensee emphasized that **[[** **]]**.

The licensee's response to RAI 2(d) described some of the aspects of its probabilistic analysis for projecting future lower core barrel bolt failures, and addressed how the results of this analysis demonstrate that the probability of having an unacceptable bolt failure pattern—specifically, one that does not satisfy the **[[** **]]** structural margins—is extremely low. The licensee determined that its bolting evaluation conservatively demonstrates that the ANO-1 lower core barrel bolting will remain functional through the end of the Fuel Cycle 26, thereby indicating that performance of next UT examination during Refueling Outage 26 (fall 2016), per Table 4-1 of MRP-227-A, is appropriate.

The NRC staff determined that the licensee's structural analysis of the lower core barrel bolting based on the 2008 IGSCC indications, and projections of bolting IGSCC between 2008 and 2016, provides reasonable assurance that the functionality of the lower core barrel bolting will be maintained until the next UT examinations scheduled for the fall 2016 refueling outage, per MRP-227-A, Table 4-1. However, the staff's determination regarding the licensee's analyses is only applicable to the ANO-1 RVI AMP, specifically considering that plant-specific implementation of MRP-227-A provides for the subsequent lower core barrel bolting UT

examination during the fall 2016 refueling outage. The staff's assessment of this information shall in no way be construed as a generic endorsement of licensees' use of probabilistic methods for projecting future SCC in susceptible high strength RVI bolting. The staff noted that these lower core barrel bolting indications do not require expansion of these UT examinations to the linked thermal shield bolting components, based on the MRP-227-A, Table 5-1 expansion criteria, because the number of reported bolts with unacceptable indications is less than 10 percent of the lower core barrel bolt population at ANO-1. Therefore, the staff determined that the licensee's implementation of the MRP-227-A, Table 4-1, Primary component inspection guidelines for the lower core barrel bolting will provide adequate assurance of lower core barrel bolting integrity and functionality during the PEO.

3.3.3 NRC Staff Conclusion for RVI Operating Experience

The NRC staff determined that the licensee performed the necessary corrective actions to address the root cause of all plant-specific RVI component degradation as discussed above. Furthermore, considering the licensee's corrective actions for the RVI component degradation, the licensee adequately demonstrated that its plant-specific implementation of the MRP-227-A guidelines will provide adequate assurance of RVI component functionality during the PEO.

3.4 Licensee Action Items

Section 4.2, "Plant-Specific Action Items," of the SE for MRP-227-A, provides eight action items that must be addressed on a plant-specific basis by PWR licensees and license renewal applicants submitting RVI AMPs for NRC staff review and approval. These action items concern topics related to the implementation of MRP-227-A, which could not be effectively addressed on a generic basis in MRP-227-A. The licensee addressed these eight action items in Section 5.0 of the ANO-1 RVI AMP.

3.4.1 Action Item 1 – Plant-Specific Applicability of FMECA Assumptions for MRP-227-A

As addressed in Section 4.2.1 of the SE for MRP-227-A, Action Item 1 specifies that each licensee is responsible for demonstrating that MRP-227-A is applicable to its facility, taking into consideration the assumptions regarding plant design and operating history made in the FMECA for reactors of their design. Licensees shall also address any plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection criteria. The licensee shall submit this evaluation for NRC review and approval as part of its application to implement MRP-227-A.

3.4.1.1 Licensee Evaluation of Action Item 1

In Section 5.1 of the ANO-1 RVI AMP submittal, the licensee addressed the three general assumptions of MRP-227-A, Section 2.4 that were used to develop the MRP-227-A inspection guidelines, and described how ANO-1 is bounded by the three general assumptions.

Assumption 1: MRP-227-A assumes a maximum of 30 years of operation with high neutron leakage core loading patterns followed by implementation of a low neutron leakage fuel management strategy for the remainder of the 60-year extended license term. The licensee confirmed that ANO-1 core management practices satisfy this assumption.

Assumption 2: MRP-227-A assumes base load operation, which refers to plant operation at fixed power levels and does not involve variations in power level on a calendar or load demand schedule. The licensee confirmed that ANO-1 operates as a base load unit.

Assumption 3: MRP-227-A assumes no plant-specific design changes beyond those identified in general industry guidance or recommended by the original vendors since May 2007. The licensee stated that no RVI component modifications have been implemented at ANO-1 beyond those identified in general industry guidance or recommend by the vendor, B&W, since May 2007. The licensee described its plant-specific RVI component design distinctions (relative to the generic RVI component design assumed in MRP-189, Revision 1) in Section 3.2 of the ANO-1 RVI AMP submittal.

3.4.1.2 NRC Staff Assessment of Action Item 1

The NRC staff determined that the licensee adequately addressed how ANO-1 satisfies the three general assumptions of MRP-227-A, Section 2.4, regarding plant design and operating history that were used as the basis for the FMECA and functionality analyses. Specifically, the licensee provided the necessary confirmation that that it switched to a low-leakage core loading pattern prior to 30 calendar years of operation, has always operated as a base load unit, and has no plant-specific design changes since May 2007 beyond those identified in general industry guidance or recommended by the original vendor. Therefore, the staff determined that the licensee's evaluation of these three general assumptions is acceptable.

With respect to the plant-specific RVI component design distinctions at ANO-1, the NRC staff reviewed the information provided by the licensee in Section 3.2 of the ANO-1 RVI AMP submittal and determined that this evaluation is specifically applicable to Action Item 2. The ANO-1 RVI component design distinctions fall into one of two categories:

Category (1): ANO-1 RVI components within the scope of license renewal that are not included in the generic RVI component listings for B&W plants in Table 4-1 and 4-2 in MRP-189, Revision 1, and therefore require plant-specific evaluation for aging management.

Category (2): ANO-1 RVI components within the scope of license renewal that are included in the generic RVI component listings in Table 4-1 and 4-2 in MRP-189, Revision 1 and were thus evaluated to determine their final inspection category and inspection criteria under MRP-227-A. However these plant-specific RVI components are fabricated from a different type of material than that analyzed in MRP-189, Revision 1. Therefore, they require plant-specific evaluation to determine whether the difference in plant-specific material type affects the result of their screening for aging degradation and FMECA, and thus the potential need for changes to their inspection criteria, relative to that identified in MRP-227-A.

These two categories of RVI component design distinctions are, hereafter, referred to as Category (1) and (2) plant-specific RVI components in this SA. The NRC staff's review of these two categories of plant-specific RVI component design distinctions is documented in Section 3.4.2 of this SA for Action Item 2.

3.4.1.3 NRC Staff Conclusion for Action Item 1

Based on its review of the licensee's evaluation of the three general assumptions of MRP-227-A, Section 2.4, for addressing Action Item 1, the NRC staff determined that the

licensee has adequately demonstrated that the FMECA assumptions of MRP-227-A are applicable to ANO-1. Therefore, the staff determined that the licensee has satisfied the criteria of Action Item 1 for ANO-1.

3.4.2 Action Item 2 – RVI Components within the Scope of License Renewal

As addressed in Section 4.2.2 of the SE for MRP-227-A, Action Item 2 specifies that licensees for B&W plants shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and identify whether this table contains all of the RVI components that are within the scope of license renewal for their facilities in accordance with 10 CFR 54.4. If these tables do not include all the RVI components that are within the scope of license renewal, licensees shall identify the missing component(s) and propose any necessary AMP modifications to address plant-specific inspection criteria for the missing components.

3.4.2.1 Licensee Evaluation of Action Item 2

In Section 5.2 of the ANO-1 RVI AMP, the licensee identified three additional components that were not included in MRP-189, Revision 1, Tables 4-1 and 4-2. These RVI components, which the licensee identifies as the ANO-1 "orphan components" are:

- Reactor Vessel Level Monitoring System Probe Supports;
- Remaining Portions of the Surveillance Specimen Holder Tube Assemblies; and
- Thermal Shield and Thermal Shield Upper Restraint Welds.

Section 5.2 of the ANO-1 RVI AMP states that the above orphan components will initially receive VT-3 visual examinations, and they will undergo a future evaluation to determine the need for additional aging management requirements.

3.4.2.2 NRC Staff Assessment of Action Item 2

The NRC staff reviewed the licensee's description of the three ANO-1 orphan components that are not included in the generic RVI component listings for B&W plants in Tables 4-1 and 4-2 of MRP-189, Revision 1. These are Category (1) plant-specific RVI components, as defined above. Section 3.2 of the AMP also lists other ANO-1 RVI components that are included in Table 4-1 and 4-2 in MRP-189, Revision 1, but are fabricated from a different type of material than that analyzed in MRP-189, Revision 1. These are Category (2) plant-specific RVI components.

For Category (1) components, the NRC staff requested in RAI-5 (Reference 14) that the licensee confirm whether the three ANO-1 orphan components are the only plant-specific RVI components within the scope of license renewal that are not listed in Tables 4-1 and 4-2 of MRP-189, Revision 1, and requiring plant-specific evaluation per Action Item 2.

For Category (2) components, the NRC staff requested in RAI-3 (Reference 14) that the licensee identify and evaluate any ANO-1 RVI components fabricated from four materials that are known to be particularly susceptible to one or more of the eight aging mechanisms:

(1) Nickel base alloys: Inconel 600 Base Metals and Weld Metals of Alloy 82/182 Alloy X-750;

(2) Type 347 Stainless Steel (SS);
(3) Type 17-4 and 15-5 Precipitation Hardened Stainless Steel (PHSS); and
(4) Type 431 SS.

In its letter dated December 30, 2015 (Reference 4), the licensee provided a proprietary vendor report for addressing Action Item 2, AREVA Inc. Report ANP-3418P, Revision 0, "Arkansas Nuclear One, Unit 1 Reactor Vessel Internals License Renewal Scope and MRP-189, Revision 1 Comparison (MRP-227-A Action Item 2) Licensing Report." This report is, hereafter, referred to as the ANO-1 Action Item 2 Report in this SA. The NRC staff reviewed the Action Item 2 Report to determine whether it adequately evaluated the Category (1) and Category (2) plant-specific RVI components for addressing RAI-5 and RAI-3, respectively.

Action Item 2 Evaluation of Category (1) Plant-Specific RVI Components

The licensee's Action Item 2 Report compares all ANO-1 RVI components and material types that are within the scope of license renewal with the generic B&W RVI components and material types. Based on this comparison, the licensee identified a number of RVI components that are not included in the generic RVI component listings in MRP-189, Revision 1, Tables 4-1 and 4-2. These ANO-1 RVI components are:

1. Plenum Cover Welds;
2. Plenum Cylinder and Reinforcing Plate and Round Bar Welds;
3. Tie Plate and Weld for Replacement Upper Core Barrel Bolting;
4. Miscellaneous Locking Device Parts and Associated Welds for the Vent Valves;
5. Tie Plate Weld for Replacement Lower Thermal Shield Bolting;
6. Reactor Vessel Level Monitoring System Probe Supports, Including Welds;
7. Remaining Surveillance Specimen Holder Tube Assemblies, Including Welds; and
8. Thermal Shield and Thermal Shield Upper Restraint Welds.

(Note: Components 6, 7, and 8 are the three orphan components identified in Section 5.2 of the ANO-1 RVI AMP.)

The licensee performed a plant-specific evaluation of these components to determine whether plant-specific modifications to the ANO-1 RVI AMP are needed to ensure adequate aging management.

For the eight components listed above, the NRC staff reviewed the licensee's RVI component evaluation to determine whether it is consistent with the generic RVI component screening and FMECA methodology described in MRP-189, Revision 1. The staff confirmed that the subcomponent-level items and welds for the above components were correctly screened for the eight aging mechanisms based on material type and a conservative evaluation of their screening parameters, consistent with the MRP-189, Revision 1 methodology. The staff also confirmed that the licensee performed a plant-specific FMECA of these components that is consistent with the generic FMECA process documented in MRP-189, Revision 1. The staff determined that the licensee's plant-specific FMECA correctly identified the susceptible RVI components that require a detailed engineering evaluation for plant-specific aging management criteria, based on the analysis of risk and safety consequences associated with the screened-in component degradation mechanism(s). The staff noted that, of the eight plant-specific RVI

components listed above, the three susceptible plant-specific RVI components that require detailed evaluation for aging management include:

1. Replacement Upper Core Barrel Bolts Tie Plate;
2. Subcomponents of the Original and Modified Vent Valve Locking Devices; and
3. Hex Head Cap Screw in the Remaining Surveillance Specimen Holder Tube Assemblies.

The NRC staff also verified that no plant-specific aging management activity is needed for the other five ANO-1 RVI components that were not generically addressed in MRP-189, Revision 1, because the licensee demonstrated that these components had acceptable plant-specific FMECA results. Specifically, aging degradation would be either non-existent or insignificant considering the risk and safety consequences associated with the screened-in degradation mechanism. Therefore, the staff determined that the licensee's plant-specific FMECA for identifying the three susceptible RVI components is acceptable.

For the three susceptible RVI components identified above, the NRC staff reviewed the licensee's engineering evaluation in Section 3.4.4 of the Action Item 2 Report to determine whether the licensee adequately addressed the potential impact of the screened-in aging mechanisms on the functionality of these components. The results of this evaluation indicate the following:

1. <u>Replacement Upper Core Barrel Bolts Tie Plate</u>: The licensee performed a functionality evaluation for determining whether the Primary component inspection criteria in MRP-227-A, Table 4-1 (ANO-1 RVI AMP Table 5-1), are adequate for detecting aging effects associated with this component item. The licensee's functionality evaluation determined that the Primary component inspection criteria for the upper core barrel bolts locking devices and associated examination acceptance criteria in MRP-227-A, Table 5-1 (ANO-1 RVI AMP Table 5-4), are adequate for detecting the effects associated with the screened-in aging degradation mechanism for the tie plate. The NRC staff verified that the licensee's functionality evaluation of this item demonstrated that the VT-3 visual examination of the upper core barrel bolts locking mechanism to look for evidence of loss of material, damaged, distorted, or missing locking device components, per the Primary component inspection criteria, is sufficient for ensuring that aging degradation of the tie plate is not occurring. Therefore the staff determined that the licensee's analysis of the replacement upper core barrel bolts tie plate is acceptable, and no modification to the ANO-1 RVI AMP is required to manage the effects of aging for this component item.

2. <u>Subcomponents of the Original and Modified Vent Valve Locking Devices</u>: The licensee identified that the vent valve locking devices (both original and modified designs) are included as an existing program component, per Table 5-3 of the ANO-1 RVI AMP, and require VT-3 visual examinations of 100 percent of accessible surfaces to look for wear in the original vent valve locking devices and cracking due to SCC for the modified vent valve locking devices. However, the licensee noted that Table 5-3, VT-3 visual examinations of the locking devices do not specifically address the **[[** **]]**, within the ANO-1 original vent valve locking devices. Based on this concern, the licensee performed an evaluation for addressing the effects of this aging mechanism on locking device functionality. This functionality evaluation took into consideration the

MRP-227-A Primary component inspection criteria for vent valve retaining rings and the additional vent valve operability testing and inspection requirements, specified in Note 1 of ANO-1 AMP, Table 5-3. Note 1 of Table 5-3, specifies that a verification of the proper operation and position of the vent valves shall be performed along with visual inspections of locking device subcomponents to look for cracking, wear, and proper positioning. The NRC staff verified that the licensee's functionality evaluation demonstrated that the Primary component VT-3 visual examinations of the vent valve retaining rings and the additional vent valve operability testing and inspection requirements specified in Note 1 of Table 5-3 would provide adequate assurance of vent valve locking device functionality during the PEO, considering the effects of **[[** **]]**. Therefore, the staff determined that the licensee's analysis of the subcomponents of the original and modified vent valve locking devices is acceptable, and no modification to the ANO-1 RVI AMP is required to manage the effects of aging for these items.

3. <u>Cap Screw in the Remaining Surveillance Specimen Holder Tube Assemblies</u>: The function of the cap screw is to **[[** **]]**. In addition, the licensee stated that the **[[** **]]**. The licensee performed an evaluation of this subcomponent item for addressing the effects of the screened-in degradation mechanism, ISR, on cap screw functionality. The licensee's evaluation determined that ISR of the cap screw **[[** **]]**. The staff confirmed that the licensee's evaluation adequately demonstrated that even considering **[[** **]]**. The staff verified the licensee's determination that the underlying function of **[[** **]]**. Thus, the NRC staff determined that the licensee's analysis of the cap screw in the remaining surveillance specimen holder tube assemblies is acceptable, and no modification to the ANO-1 RVI AMP is required to manage the effects of aging for this item.

<u>Action Item 2 Evaluation of Category (2) RVI Components</u>

The licensee's Action Item 2 Report identified a number of ANO-1 RVI components within the scope of license renewal that are included in the generic listing of B&W RVI components in Tables 4-1 and 4-2 of MRP-189, Revision 1, but are of a different type of material than that analyzed in MRP-189, Revision 1. Section 3.3 of the licensee's Action Item 2 Report evaluated the difference in plant-specific material type for each of these ANO-1 RVI components by screening these plant-specific component materials for the eight aging degradation mechanisms, consistent with the screening process of MRP-189, Revision 1. The licensee's Action Item 2 Report determined that no additional aging degradation mechanisms are screened in as a result of these plant-specific differences in RVI component material type and thus, no plant-specific modifications to the ANO-1 RVI AMP inspection guidelines are necessary for these components, relative to those specified in MRP-227-A.

The NRC staff reviewed the differences between the plant-specific RVI component material types and the generic RVI component materials analyzed in the MRP-189, Revision 1, and verified that these differences would not result in any additional aging degradation mechanisms being screened in for these component materials. The staff also determined that there are no ANO-1 RVI components fabricated from the four particularly susceptible material types addressed in RAI-3 that were not already analyzed, as such, in MRP-189, Revision 1. Therefore, the staff determined that the MRP-227-A inspection criteria remain applicable to the ANO-1 RVI components that are of a different type of material than that analyzed in MRP-189, Revision 1, and the ANO-1 RVI AMP will provide for acceptable aging management and adequate assurance RVI component integrity during the PEO. Accordingly, the staff determined that Action Item 2 is resolved for these ANO-1 RVI components.

3.4.2.3 NRC Staff Conclusion for Action Item 2

Based on its review of the licensee's Action Item 2 Report and associated RAI responses, the NRC staff determined that the licensee adequately identified and evaluated all plant-specific RVI components that are within the scope of license renewal at ANO-1 relative to the generic RVI component listings in Tables 4-1 and 4-2 in MRP-189, Revision 1. For the Categories (1) and (2) plant-specific RVI components, the staff determined that the licensee adequately demonstrated that the ANO-1 RVI AMP, through its implementation of the MRP-227-A guidelines and additional existing program inspection criteria for the vent valve locking devices, will provide for adequate aging management during the PEO. Therefore, the licensee has satisfied the criteria of Action Item 2 for ANO-1.

3.4.3 Action Item 3 – Evaluation of the Adequacy of Plant-Specific Existing Programs

Action Item 3 is only applicable to the RVI for Westinghouse and Combustion Engineering plants. Since ANO-1 is a B&W plant, this action item requires no evaluation from the licensee.

3.4.4 Action Item 4 – B&W Core Support Structure Upper Flange Stress Relief

As addressed in Section 4.2.4 of the MRP-227-A SE, Action Item 4 specifies that licensees for B&W plants shall confirm that the core support structure upper flange weld was stress relieved during the original fabrication of the RVI. If the upper flange weld has not been stress relieved, Action Item 4 specifies that this component shall be inspected as a Primary component. The licensee's resolution of this plant-specific action item shall be submitted to the NRC for review and approval as part of its application to implement MRP-227-A.

3.4.4.1 Licensee Evaluation of Action Item 4

In Section 5.4 of the ANO-1 RVI AMP submittal, the licensee stated that it performed a review of the fabrication records for the RVI components and determined that the upper flange weld was stress relieved during the fabrication process. Accordingly, the licensee determined that this review satisfies the requirements of Action Item 4.

3.4.4.2 NRC Staff Assessment and Conclusion for Action Item 4

The NRC staff determined that the licensee's confirmation that the core support structure upper flange weld was stress relieved, is sufficient to confirm the applicability of MRP-227-A to ANO-1

and will provide adequate assurance of component functionality during the PEO. Accordingly, the staff determined that the licensee has satisfied Action Item 4 for ANO-1.

3.4.5 Action Item 5 – Application of Physical Measurements for Combustion Engineering and Westinghouse RVI Components

Action Item 5 specifies the identification of plant-specific physical measurement acceptance criteria only for Westinghouse and Combustion Engineering RVI components. Generic acceptance criteria for the one-time physical measurement for the plenum cover assembly at B&W plants are sufficiently identified in the examination acceptance criteria specified in Table 5-1 of MRP-227-A, which is included as Table 5-4 of the ANO-1 RVI AMP. Since ANO-1 is a B&W plant, this action item requires no plant-specific action by the licensee.

3.4.6 Action Item 6 – Evaluation of Inaccessible B&W Components

As addressed in Section 4.2.6 of the MRP-227-A SE, Action Item 6 specifies that licensees for B&W plants shall justify the acceptability of RVI components that are either inaccessible for inspection or not inspectable using the currently available examination techniques. Specifically, the licensee shall provide either a functionality evaluation of these RVI components, or propose a schedule for component replacement. The licensee's component evaluation or schedule for replacement shall be submitted to the NRC for review and approval as part of its application to implement MRP-227-A. The provisions of Action Item 6 are applicable to the following inaccessible or non-inspectable B&W RVI components:

- Core Barrel Assembly – Cylinders (Including Vertical and Circumferential Seam Welds);
- Core Barrel Assembly – Former Plates;
- Core Barrel Assembly – External Baffle-to-Baffle Bolts and their Locking Devices;
- Core Barrel Assembly – Barrel-to-Former Bolts and their Locking Devices; and
- Core Barrel Assembly – Internal Baffle-to-Baffle Bolts.

3.4.6.1 Licensee Evaluation of Action Item 6

In Section 5.6 of the ANO-1 RVI AMP submittal, the licensee stated that ANO-1 will justify the acceptability of the above inaccessible and non-inspectable RVI components for continued operation through the PEO by performing an evaluation, proposing a schedule for replacement, or justification for some other alternative process for these components. The licensee stated that the evaluation, schedule for replacement, or alternative justification will be submitted to the NRC by the end of 1 year from the initial inspection of the linked Primary component items if the inspection results indicate aging, which is the implementation date for this condition.

3.4.6.2 NRC Staff Evaluation of Action Item 6

The NRC staff identified that the above inaccessible RVI components (including the non-inspectable internal baffle-to-baffle bolts) are categorized as Expansion components, per the MRP-227-A, Table 4-4, inspection guidelines. However, no inspection requirements are identified in MRP-227-A, Table 4-4, since these components are inaccessible for examination. Therefore, in its SE for MRP-227-A, the staff included Action Item 6 to ensure that licensees either perform a plant-specific functionality evaluation of these inaccessible RVI components or propose a schedule for the replacement of these components. It should be emphasized

that Action Item 6 has no stipulation on the inspection results for any linked Primary components—the inaccessible RVI component evaluations (or proposed replacement activities), required by Action Item 6 shall be submitted to the NRC staff for review and approval as part of their plant-specific AMP to implement MRP-227-A.

For the above inaccessible components, the licensee provided a regulatory commitment to submit an evaluation, schedule for replacement, or justification for some other alternative process to the NRC by the end of 1 year from the initial inspection of the linked primary component items, if these inspections indicate aging that meets the expansion criteria for the linked primary components. Since this action item specifically requires that licensees submit the Action Item 6 information to the NRC staff for review and approval as part of their plant-specific AMP to implement MRP-227-A—without any stipulation on inspection results for linked Primary components—the NRC staff requested in RAI-14 (Reference 14) that the licensee provide the information required to satisfy Action Item 6.

In its response to RAI-14 by letter dated February 10, 2015 (Reference 2), the licensee stated that work is ongoing to develop analyses of the above inaccessible components to demonstrate that component functionality will be maintained during the PEO. The licensee stated that these functionality evaluations will be submitted by the end of 2016. The licensee revised its regulatory commitment associated with Action Item 6 to specify that Entergy will provide the evaluation of these inaccessible components to the NRC staff by December 31, 2016.

The NRC staff reviewed the licensee's response to RAI-14, and its revised commitment to provide the inaccessible component analyses described therein by December 31, 2016, per the criteria of Action Item 6. The staff confirmed that a functionality analysis of the same type of inaccessible components has recently been performed for other B&W units. In addition, by letter dated December 21, 2016 (Reference 18), the licensee submitted its Action Item 6 Report, in fulfillment of its regulatory commitment that was provided in its February 10, 2015, response to RAI-14(a). Based on its initial assessment of the Action Item 6 Report, the staff confirmed that the regulatory commitment to submit the inaccessible RVI component analytical evaluations to the NRC was fulfilled as of the date of the submittal. That staff also confirmed that the content of the licensee's Action Item 6 Report does not invalidate its findings regarding the acceptability of the ANO-1 RVI AMP for meeting the original ANO-1 UFSAR, Section 16.1.5, requirements for aging management of the ANO-1 RVI components.

The NRC staff plans to perform a more detailed review of the Action Item 6 Report as a separate action to assess whether the analytical evaluations provide reasonable assurance of functionality for the subject ANO-1 inaccessible components for the PEO. The staff's assessment of these analytical evaluations will not affect the licensee's implementation of the ANO-1 RVI AMP for meeting the AN0-1 UFSAR, Section 16.1.5, requirements.

3.4.7 Action Item 7 – Plant-Specific Evaluation of CASS Materials

As addressed in Section 4.2.7 of the MRP-227-A SE, Action Item 7 specifies that licensees shall develop plant-specific analyses to demonstrate that B&W RVI components fabricated from cast austenitic stainless steel (CASS), martensitic SS, or PHSS will maintain their functionality during the PEO, considering possible loss of fracture toughness in these components due to TE and IE. The licensee shall include this plant-specific analysis as part of its submittal to implement MRP-227-A.

3.4.7.1 Licensee Evaluation of Action Item 7

In Section 5.7 of the ANO-1 RVI AMP submittal, the licensee listed the following ANO-1 RVI components that are fabricated from CASS and PHSS materials:

<u>CASS Materials:</u>

- Control Rod Guide Tube (CRGT) Assembly - Spacer Castings (American Society for Testing and Materials (ASTM) Specification A351, Grade CF-3M CASS);
- Core Support Shield Assembly - Vent Valve Bodies (ASTM A351, Grade CF-8 CASS); and
- Incore Monitoring Instrumentation (IMI) Guide Tube Assembly - Spider Castings (ASTM A351, Grade CF-8 CASS).

<u>PHSS Materials:</u>

- Core Support Shield Assembly - Vent Valve Retaining Rings (Type 15-5 PHSS).

The licensee noted that the CRGT Assembly Spacer Castings, Instrument Guide Assembly - Spider Castings, and Core Support Shield Assembly - Vent Valve Retaining Rings are all Primary components per MRP-227-A.

The licensee identified the applicable aging effect for the above items as a reduction in fracture toughness by either TE, IE, or a combination of the two. The licensee stated that an analysis of the effects of reduction in fracture toughness due to TE and/or IE on these RVI components will be performed. The licensee provided regulatory commitments to complete the Action Item 7 evaluations, 12 months prior to the second refueling outage after entering the PEO.

3.4.7.2 NRC Staff Assessment of Action Item 7

The NRC staff determined the licensee must address the provisions of Action Item 7 for the ANO-1 RVI AMP. Therefore, in RAI-15(a) (Reference 14), the staff requested that the licensee provide plant-specific analyses to demonstrate that the above CASS and PHSS RVI components will maintain their functionality during the period of extended operation, considering possible loss of fracture toughness due to IE and TE.

In its response to RAI-15(a) by letter dated May 24, 2016 (Reference 5), the licensee provided its plant-specific analysis of the above CASS and PHSS components. The licensee's plant-specific analysis of these components is documented in the proprietary AREVA report ANP-3417P, Revision 1, "MRP-227-A Applicant/Licensee Action Item 7 Analysis for Arkansas Nuclear One Unit 1." This report is hereafter referred to as the ANO-1 Action Item 7 Report in this SA.

The ANO-1 Action Item 7 Report states that the following four RVI components were identified as requiring further evaluation based on material type and plant-specific screening results for TE and IE:

- CRGT Assembly - Spacer Castings (Grade CF-3M CASS)
  - Screened as potentially susceptible to TE, but not IE
  - Primary components, per MRP-227-A, Table 4-1;
- IMI Guide Tube Assembly - Spider Castings (Grade CF-8 CASS)
  - Screened as potentially susceptible to IE, but not TE
  - Primary components, per MRP-227-A, Table 4-1;
- Core Support Shield Assembly - Vent Valve Retaining Rings (Type 15-5 PHSS)
  - Screened as potentially susceptible to TE, but not IE
  - Primary components, per MRP-227-A, Table 4-1; and
- Select Original Vent Valve Locking Device Parts (Type 431 martensitic SS)
  - Screened as potentially susceptible to TE, but not IE
  - No inspections specified in MRP-227-A.

The NRC staff noted that this list includes three of the four components listed in Section 5.7 of the ANO-1 RVI AMP submittal, plus select original vent valve locking device parts that were determined to be martensitic SS. The CASS core support shield assembly vent valve bodies were determined to not require further evaluation based on the following assessment.

The Action Item 7 Report indicates that the vent valve bodies were identified as being fabricated from CASS; however, they were not evaluated based on the assumption that the vent valve bodies have a ferrite content of 20 percent or less. For the B&W fleet, all originally installed vent valve bodies are ASTM A351 Grade CF8 castings. The licensee's Action Item 7 Report provided some preliminary information confirming that the vent valve bodies at ANO-1 are below the 20 percent screening threshold for TE of unirradiated Grade CF8 CASS material. However, the licensee stated that this cannot be definitively confirmed unless the specific material data for the vent valve bodies currently installed at ANO-1 are known, based on the component serial numbers and heat numbers. Therefore, in its RAI response for Action Item 7, by letter dated May 24, 2016, the licensee provided a regulatory commitment to locate component serial numbers and heat numbers that are stamped on the vent valve bodies currently installed at ANO-1 when the core barrel assembly is removed during the initial MRP-227-A inspections. This regulatory commitment states:

> Entergy commits to record the serial numbers and heat numbers stamped on the vent valve bodies currently installed in the ANO-1 RV internals when the core barrel assembly is removed during the initial MRP-227 inspections.

The scheduled completion date for this commitment is the end of the fall 2016 refueling outage—Refueling Outage 26. The licensee stated that this information will support confirmation that the vent valve bodies at ANO-1 have a ferrite content of 20 percent or less.

The NRC staff determined that the licensee's preliminary information, indicating that the vent valve bodies at ANO-1 have a ferrite content below the 20 percent screening threshold for TE of

unirradiated Grade CF8 CASS material, is sufficient to support the licensee's commitment to make a final determination of this, based on confirmation of heat-specific material property data. The staff also determined that that there is adequate assurance that vent valve body integrity and functionality will be maintained for the PEO through the implementation of the AMP, Table 5-3, Note 1 requirements for vent valve operability testing and visual examinations, which are implemented under the licensee's IST program. Therefore, based on its review of this other information, the staff determined that the licensee's commitment is acceptable. The NRC staff finds that reasonable controls for the implementation of the above regulatory commitment are best provided through the licensee's administrative processes, including its commitment management program. The above regulatory commitment does not warrant the creation of regulatory requirements, and is not relied on for the approval of the ANO-1 RVI AMP.

For the four susceptible components identified as requiring further evaluation, per Action Item 7, the licensee performed evaluations to demonstrate that these components will remain functional for the PEO. This evaluation was performed using the following methodology:

- Identify appropriate inputs for the evaluation, **[[** **]]**.
- Utilize available information to determine if failure is likely or unlikely to occur.
- Determine effect of failure on component functionality for the PEO.

The NRC staff's review of the licensee's Action Item 7 evaluation for these four components is documented below.

<u>CRGT Spacer Castings</u>

The CRGT spacer castings are Primary components per MRP-227-A, Table 4-1. The inspection guidelines specify that these components are to receive VT-3 visual examinations of the accessible surfaces at each of the four screw locations for 100 percent of the spacer castings every 10-Year ISI interval during the PEO.

The licensee identified that the function of the CRGT spacer castings is to provide structural support and alignment of the control rod guide sectors within each CRGT assembly, for ensuring the required operation of the control rod assemblies (CRAs) in the event of a reactor trip or control rod movement signal. The licensee stated that CRAs travel through the path provided by the spacer casting brazement subassembly into the fuel assemblies. The licensee noted that CRGT spacer castings do not have a core support function; however, they do have a safety function relative to control rod alignment, insertion, and control of reactivity. Therefore, degradation of the spacer castings could potentially impede plant shutdown capability by hindering the insertion of the control rods into the core.

<u>NRC Staff Assessment</u>

Per the MRP-189, Revision 1, FMECA process underlying MRP-227-A, the NRC staff verified that the CRGT spacer castings are not screened as susceptible to service-induced flaws (i.e., IASCC, SCC, or fatigue). Therefore, the staff determined that the licensee's consideration of only the potential for **[[** **]]** to be valid. The

staff determined that licensee's review of available Certified Material Test Reports (CMTRs) for the spacer castings indicating that [[ ]] provides reasonable assurance that the ANO-1 spacer castings had [[ ]].

The NRC staff noted that the licensee evaluated the ANO-1 CRGT spacer castings as susceptible to TE, consistent with MRP-189, Revision 1. Furthermore the staff identified that the licensee conservatively evaluated the fracture toughness of the spacer castings as [[ ]] for determining their susceptibility to failure. Therefore, the staff determined that the licensee's evaluation of the degraded material properties for the spacer castings is acceptable.

The NRC staff verified the licensee's statements that VT-3 visual examinations of the CRGT spacer castings at two B&W plants, per the MRP-227-A inspection guidelines, with 100 percent coverage of accessible surfaces, revealed no indications of aging degradation. The staff also noted that no failures of these CASS materials due to embrittlement have been reported in the industry. The staff noted that this favorable OE confirms the licensee's statement that the [[ ]]. The staff also determined that the licensee provided sufficient evidence to support its determination that [[ ]]. Based on its review of this information, the staff was able confirm that, [[ ]]. Therefore, the staff determined that considering the reduction in fracture toughness due to TE, there is reasonable assurance that failure of the ANO-1 CRGT spacer castings is unlikely during the PEO.

The licensee described its functionality evaluation of the CRGT spacer castings, which addressed the effects of an unlikely, but postulated failure. The NRC staff noted that the functionality evaluation considered the amount of distortion allowed that will still permit the control rod to pass freely through the spacer casting brazement sub-assembly. The conclusion of the licensee's analysis was [[ ]]. The staff reviewed the functionality evaluation and determined that it adequately demonstrates that there are [[ ]] that would be expected to result in a loss of functionality for the spacer castings.

The licensee also identified that drop-time testing of the CRAs is performed at the beginning of each cycle per (TS) surveillance requirements (SRs). The licensee stated that when unusual control rod drop-times are encountered, the root cause must be investigated. The licensee noted that to date, slow trip times have always been linked to other issues unrelated to RVI component integrity. The NRC staff reviewed this information and confirmed that the TS SRs for control rod drop testing will provide additional evidence of integrity and functionality for the spacer castings during the PEO.

Based on its review of the licensee's evaluation of the CRGT spacer castings, the NRC staff determined that the licensee adequately demonstrated that the spacer castings are expected to remain functional for the PEO, considering their reduction in fracture toughness due to TE. Therefore, the staff determined that the licensee's implementation of the VT-3 visual

examinations, per the MRP-227-A inspection guidelines, will provide adequate assurance of CRGT spacer casting integrity and functionality during the PEO.

IMI Guide Tube Spider Castings

The IMI guide tube spider castings are Primary components per MRP-227-A, Table 4-1. The inspection guidelines specify that these components are to receive VT-3 visual examinations of 100 percent of the top surfaces of 52 spider castings and welds to the adjacent lower grid rib section every 10-Year ISI Interval during the PEO.

The licensee identified that the function of the IMI guide tube spider castings is to provide lateral restraint for the IMI guide tube, and the function of the spider fillet welds is to hold the IMI guide tube spider castings in place. The licensee stated that the IMI guide tube provides the continuous protected guide path for the IMI from its entry into the RPV through the RPV instrumentation nozzles to the entrance into the fuel assembly instrument guide tube. The licensee noted that loss of function of the IMI guide path would result in a sufficient misalignment at the fuel assembly instrument tube entrance to prohibit entry of the incore instrument. The licensee also indicated that failure of the guide path could result in wear of the IMI sheath due to flow-induced vibration, and therefore, would be considered a loss of function.

NRC Staff Assessment

Per the MRP-189, Revision 1, FMECA process underlying MRP-227-A, the NRC staff verified that the IMI guide tube spider castings are not screened as susceptible to service-induced flaws (i.e., IASCC, SCC, or fatigue). Therefore, the staff determined that the licensee's consideration of only the potential for **[[** **]]** to be valid. The staff determined that licensee's review of available CMTRs for the spider castings indicating that **[[** **]]** provides reasonable assurance that the ANO-1 spider castings had **[[** **]]**.

The NRC staff noted that the IMI guide tube spider castings are generically screened as potentially susceptible to both TE and IE, per the MRP-189, Revision 1 FMECA underlying MRP-227-A. However, the licensee stated in its Action Item 7 Report that they are susceptible only to IE, and not TE, based on the TE screening criteria established for non-irradiated CASS materials in an NRC Letter from Christopher Grimes, Director of the Division of Regulatory Improvement Programs, Office of Nuclear Reactor Regulation, "License Renewal Issue No. 98-0030, Thermal Aging Embrittlement of Cast Austenitic Stainless Steel Components," dated May 19, 2000 (Reference 17, also referred to as the C. Grimes Letter). The staff determined that the TE screening criteria established for non-irradiated CASS materials in the C. Grimes Letter may only be applied to the ANO-1 IMI guide tube spider castings to screen out susceptibility to TE if the projected neutron irradiation damage to the material is less than or equal to 1 displacement per atom (dpa). The licensee reported a peak neutron exposure for these castings corresponding to the **[[** **]]**.

Accordingly, the NRC staff determined that this projected neutron irradiation damage is too high to validate the licensee's use of the C. Grimes Letter to screen out TE for the ANO-1 spider castings.

The licensee also provided information indicating that the **[[**

]]. In addition, the licensee cited the NRC's fracture toughness criterion of 255 kilojoules per square meter (kJ/m^2), as set forth in the C. Grimes Letter (for non-irradiated CASS), as the basis for differentiating between non-significant and potentially significant reduction in fracture toughness for CASS materials. With respect to the loss of fracture toughness for Grade CF-8 CASS materials undergoing IE, the licensee cited a joint Boiling Water Reactor Vessel and Internals Project (BWRVIP)/MRP Working Group study on CASS that evaluated the fracture toughness for irradiated Grade CF-8 materials as a function of neutron exposure. The licensee stated that the data indicates that the reduction in fracture toughness for CF-8 CASS does not reach 255 kJ/m^2 until approximately 3.3 dpa. In addition, the licensee cited BWRVIP/MRP Working Group calculations of the driving force for flaw propagation in RVI components, which show that large preexisting flaws would not be expected to propagate if the material were at the minimum allowable fracture toughness of 255 kJ/m^2.

The NRC staff reviewed the licensee's fracture toughness information for the irradiated Grade CF-8 IMI guide tube spider castings and determined that the BWRVIP/MRP Working Group data on the fracture toughness for irradiated Grade CF-8 materials as a function of neutron exposure (as cited in the licensee's Action Item 7 Report) has not yet been validated by the staff for application to Grade CF-8 CASS PWR RVI components for determining adequate fracture toughness. Furthermore, the staff determined that the C. Grimes Letter fracture toughness acceptance criterion of 255 kJ/m^2 is only specified for CASS materials subject to TE and not IE, and considering the projected neutron irradiation damage for these components, the staff determined that there is currently no basis to support the licensee's use of the C. Grimes Letter to screen out TE for the ANO-1 spider castings. Therefore, the staff determined that it cannot verify licensee's determination that the reduction in fracture toughness for the IMI guide tube spider castings during the PEO "is not considered significant." Thus, the staff finds that a significant reduction in fracture toughness due to the combined effects of TE and IE must be considered.

Notwithstanding the above, the NRC staff noted that the generic FMECA for these components in MRP-189, Revision 1, considered TE and IE as the applicable aging mechanisms, and the MRP-227-A, Table 4-1 inspection guidelines, specify that the VT-3 visual examinations are to look for evidence of cracking due to TE and IE, which includes the detection of fractured, missing, or separated spider arms. Therefore, for the purpose of its assessment of spider casting functionality, the staff assumes that the Grade CF-8 spider casting material may be potentially susceptible to fracture, consistent with the generic FMECA in MRP-189, Revision 1.

The NRC staff reviewed the licensee's evaluation of the two other key fracture analysis criteria to determine the likelihood of IMI guide tube spider casting failure due to reduced fracture toughness from TE and IE:

1. It is not likely **[[**
   ]].

2. Based on its structural evaluation of the spider castings to determine the necessary driving force for **[[** **]]**, the licensee provided sufficient information for demonstrating that the stresses in the castings are **[[**

]]. The staff confirmed the licensee's statement that the [[ ]]. Since the Action Item 7 RVI component evaluations are based on elastic-plastic fracture mechanics, the staff verified the licensee's statement that crack driving force [[ ]].

Based on its review of these criteria, the NRC staff determined that there is reasonable assurance that failure of the IMI guide tube spider castings due to propagation of [[ ]] would be unlikely during the PEO.

In addition, the NRC staff reviewed the licensee's functionality evaluation of the spider castings. The staff determined that this evaluation provides reasonable assurance that [[ ]].

Based on its review of the licensee's evaluation of the IMI guide tube spider castings, the NRC staff determined that the licensee adequately demonstrated that these castings are expected to remain functional for the PEO, considering their reduction in fracture toughness due to TE and IE. Therefore, the staff determined that the licensee's implementation of the VT-3 visual examinations, per the MRP-227-A inspection guidelines, will provide adequate assurance of IMI guide tube spider casting integrity and functionality during the PEO.

<u>Core Support Shield Vent Valve Retaining Rings</u>

The core support shield vent valve retaining rings are Primary components, per MRP-227-A, Table 4-1. The inspection guidelines specify that these components are to receive VT-3 visual examinations of 100 percent of accessible surfaces every 10-Year ISI Interval during the PEO.

The licensee stated that ANO-1 has eight vent valves installed in the core support shield cylinder. Each vent valve is mounted in a vent valve nozzle that is welded into the cylinder. The licensee stated that the vent valve is normally closed, but in the event of a reactor pressure vessel inlet pipe rupture, the valve will open to permit steam generated in the core to flow directly to the break, thereby permitting the core to be flooded and adequately cooled. The licensee identified that each valve assembly has two retaining rings that are fabricated from Type 15-5 PHSS. The function of the retaining rings is to retain the vent valve body in the vent valve nozzle. The licensee stated that the consequence of failure of a retaining ring is a potential loss of support for the valve body. The licensee identified that there is no known cracking of the vent valve retaining rings at domestic B&W plants.

<u>NRC Staff Assessment</u>

Per the MRP-189, Revision 1, FMECA process underlying MRP-227-A, the NRC staff verified that the vent valve retaining rings are not screened as susceptible to service-induced flaws (i.e., IASCC, SCC, or fatigue). Therefore, the staff determined that the licensee's consideration of only the potential for [[ ]] to be valid. The staff determined that the licensee's review of available CMTRs for the vent valve retaining rings indicating that [[ ]] provides reasonable assurance that the

ANO-1 vent valve retaining rings had **[[** **]]**.

The NRC staff noted that the licensee evaluated the ANO-1 vent valve retaining rings as susceptible TE, consistent with MRP-189, Revision 1. The licensee estimated that the fracture toughness of this Type 15-5 PHSS vent valve retaining ring material, **[[** **]]**. Based on its review of this **[[** **]]** fracture toughness estimate, the staff could not validate the licensee's use of it for the evaluation of vent valve retaining ring functionality during the PEO because the licensee has not cited any NRC publication or other open literature as a basis. However, the staff noted that the generic FMECA for these components in MRP-189, Revision 1, considered TE as the applicable aging mechanism, and the MRP-227-A, Table 4-1, inspection guidelines, specify that the VT-3 visual examinations are to look for evidence of cracking due to TE, which includes the detection of surface irregularities, such as fractured material, or missing pieces. Therefore, for the purpose of its assessment of retaining ring functionality, the staff assumes that the Type 15-5 PHSS retaining ring material may be potentially susceptible to fracture, consistent with the generic FMECA in MRP-189, Revision 1.

The licensee identified that the stresses sustained by the retaining rings during **[[** **]]**. Based on its review of the information regarding the improbability of **[[** **]]** and the lack of any known instances of cracking for the retaining rings, the NRC staff determined that the licensee's evaluation provides reasonable assurance that significant failure of the retaining rings due to propagation of **[[** **]]** is not likely during the PEO.

In addition, the NRC staff reviewed the licensee's functionality evaluation of the vent valve retaining rings. The staff determined that this evaluation provides reasonable assurance that, in the unlikely event that fracture of the retaining rings does occur, it is not expected to impair the function of the vent valve to relieve pressure in the interior of the core support shield assembly during a large break loss of coolant accident.

Based on its review of the licensee's evaluation of the vent valve retaining rings, the NRC staff determined that the licensee adequately demonstrated that they are expected to remain functional for the PEO, considering their reduction in fracture toughness due to TE. Therefore, the staff determined that the licensee's implementation of the VT-3 visual examinations, per the MRP-227-A inspection guidelines, will provide adequate assurance of vent valve retaining ring integrity and functionality during the PEO.

<u>Vent Valve Locking Device Parts</u>

The licensee noted that the vent valve locking devices were screened and evaluated in its response to Action Item 2. The licensee identified the locking device **[[** **]]** as martensitic SS subcomponent items that are evaluated under Action Item 2, since they were not included in the generic RVI component listings in Tables 4-1 and 4-2 of MRP-189, Revision 1. The licensee identified that the projected neutron fluence for the vent valve assembly is **[[** **]]**. Therefore, the licensee determined that **[[** **]]**.

<u>NRC Staff Assessment</u>

The NRC staff's review of the licensee's functionality evaluation of the vent valve locking devices, considering the susceptibility of the locking device **[[** **]]**, is addressed in Section 3.4.2.2 of this SA for Action Item 2. For Action Item 7, the staff verified that these locking device parts **[[** **]]**. The staff verified that the licensee's evaluation of locking device functionality in the Action 7 Report is consistent with the information provided for Action Item 2, considering the **[[** **]]**. As documented in Section 3.4.2.2 of this SA, the staff determined that the licensee's locking device functionality evaluation demonstrated that the RVI AMP will provide adequate assurance of locking device functionality for the PEO, taking into consideration the **[[** **]]**.

3.4.7.3 NRC Staff Conclusion for Action Item 7

The NRC staff determined that the licensee has adequately evaluated all plant-specific CASS, PHSS, and martensitic SS RVI components for ANO-1, taking into consideration their potential reduction in fracture toughness due to TE and IE. The NRC staff also determined that the licensee has demonstrated that the ANO-1 RVI AMP will provide adequate assurance of functionality for these RVI component during the PEO. Therefore, the staff determined that the licensee has satisfied the criteria of Action Item 7 for ANO-1.

3.4.8 Action Item 8 – Submittal of RVI AMP Information for NRC Staff Review and Approval

This action item requires licensees to submit a plant-specific RVI AMP for NRC review and approval to credit their implementation of MRP-227-A guidelines for aging management of the RVI components. This submittal shall include the information identified in Section 3.5.1 of the MRP-227-A SE. Section 3.5.1 of the MRP-227-A SE specifies that a licensee's RVI AMP submittal must include the following items:

1. An AMP that addresses the ten program elements defined in Revision 2 of the GALL Report AMP XI.M16A (Reference 12).

2. An RVI AMP inspection plan that addresses the plant-specific action items. If a licensee's AMP deviates from the MRP-227-A guidelines, the licensee shall identify where its program deviates from MRP-227-A and provides a justification for any deviation.

3.4.8.1 Licensee Evaluation of Action Item 8

Section 5.8 of the ANO-1 RVI AMP submittal addresses the two plant-specific RVI AMP information requirements identified in Section 3.5.1 of the MRP-227-A SE. For AMP information Item 1, the licensee stated that the ANO-1 RVI AMP addresses the ten program elements defined in Revision 2 of the GALL Report AMP XI.M16A (Reference 12). For AMP information Item 2, the licensee stated that the ANO-1 RVI AMP addresses the plant-specific action items, and it does not deviate from the MRP-227-A guidelines.

3.4.8.2 NRC Staff Assessment of Action Item 8

The NRC staff determined that the licensee satisfied Item 1 of Action Item 8 because it provided an AMP that addresses the ten elements of the revised version of GALL Report AMP XI.M16A in LR-ISG-2011-04 (Reference 13). The staff determined that the licensee satisfied Item 2 of Action Item 8 because it addressed all plant-specific action items. The staff also determined that the ANO-1 RVI AMP does not deviate from the MRP-227-A guidelines. Therefore, the staff determined that the licensee has satisfied the criteria of Action Item 8 for ANO-1.

4.0 CONCLUSION

The NRC staff has reviewed the ANO-1 RVI AMP, and concludes that it is acceptable because it is consistent with the MRP-227-A inspection and evaluation guidelines for RVI components. The licensee has adequately addressed all eight action items established in Section 4.2 of the MRP-227-A SE.

As noted in Section 3.4.6 of this SA, the NRC staff plans to perform a more detailed review of the Action Item 6 Report as a separate action to assess whether the analytical evaluations provide reasonable assurance of functionality for the subject ANO-1 inaccessible components for the PEO. The staff's assessment of these analytical evaluations for Action Item 6 will not affect the licensee's implementation of the ANO-1 RVI AMP for meeting the ANO-1 UFSAR, Section 16.1.5, requirements.

The NRC staff's approval of the ANO-1 RVI AMP does not reduce, alter, or otherwise affect the current ASME Code, Section XI ISI requirements, or any other ANO-1 licensing basis requirements related to the ISI of structures, systems, and components. The staff notes that Section 7, "Implementation Requirements," of MRP-227-A, requires that the NRC be notified of any deviations from the "needed" requirements.

5.0 REFERENCES

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3. Pyle, S. L., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Evaluations Described Response to Request for Additional Information 15(a), Reactor Vessel Internals Aging Management Program Plan, Arkansas Nuclear One, Unit 1," dated September 28, 2015 (ADAMS Package Accession No. ML15278A022).

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10. Electric Power Research Institute, letter to U.S. Nuclear Regulatory Commission, "Report Transmittal; Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)," dated January 12, 2009 (ADAMS Package Accession No. ML090160212).

11. U.S. Nuclear Regulatory Commission, Regulatory Issue Summary 2011-07, "License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management," dated July 21, 2011 (ADAMS Accession No. ML111990086).

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Principal Contributor: C. Sydnor

Date: February 13, 2017

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1 – STAFF ASSESSMENT REGARDING PROGRAM PLAN FOR AGING MANAGEMENT FOR REACTOR VESSEL INTERNALS (CAC NO. MF4201) DATED FEBRUARY 13, 2017.

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