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

October 26, 2017

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

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 2 – STAFF ASSESSMENT REGARDING PROGRAM PLAN FOR AGING MANAGEMENT FOR REACTOR VESSEL INTERNALS (CAC NO. MF8155; EPID L-2016-LRO-0001)

Dear Sir or Madam:

By letter dated July 18, 2016 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML16202A168), Entergy Operations, Inc. (the licensee) submitted a reactor vessel internals (RVI) aging management program (AMP) for Arkansas Nuclear One, Unit 2 (ANO-2). The submittal was supplemented by a letter dated September 6, 2017 (ADAMS Accession No. ML17251A758). The ANO-2 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 license renewal Commitment Nos. 15 and 19, the provisions of which are described in the ANO-2 Safety Analysis Report (SAR), Sections 18.1.23 and 18.1.24, respectively. ANO-2 SAR Sections 18.1.23 and 18.1.24 state, in part, that the vessel internals program will be initiated prior to the period of extended operation.

The NRC staff has completed its review of the ANO-2 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 assessment of the ANO-2 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-2 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.

The NRC's staff assessment of the ANO-2 RVI AMP is enclosed. If you have any questions concerning this matter, please contact the NRC Project Manager, Thomas Wengert, at (301) 415-4037, or via e-mail at Thomas.Wengert@nrc.gov.

Sincerely,

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

Docket No. 50-368

Enclosure:
Staff Assessment

cc: Distribution via Listserv

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

STAFF ASSESSMENT BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REACTOR VESSEL INTERNALS AGING MANAGEMENT PROGRAM PLAN

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE – UNIT 2

DOCKET NO. 50-368

1.0 INTRODUCTION

By letter dated July 18, 2016 (Reference 1), as supplemented by letter dated September 6, 2017 (Reference 2), Entergy Operations, Inc. (the licensee) submitted a reactor vessel internals (RVI) aging management program (AMP) for Arkansas Nuclear One, Unit 2 (ANO-2). The RVI AMP was submitted to fulfill license renewal (LR) Commitment Nos. 15 and 19, the provisions of which are described in the ANO-2 Safety Analysis Report (SAR) (Reference 3), Section 18.1.23, "Reactor Vessel Internals Cast Austenitic Stainless Steel (CASS) Program," and Section 18.1.24, "Reactor Vessel Internals Stainless Steel Plates, Forgings, Welds, and Bolting Program," respectively. These two LR commitments were submitted by letter dated September 10, 2004 (Reference 4), in support of the ANO-2 license renewal application (LRA) (Reference 5). The ANO-2 LRA was reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC) staff in NUREG-1828 (Reference 6), "Safety Evaluation Report Related to the License Renewal of the Arkansas Nuclear One, Unit 2." Appendix A, "Commitments for License Renewal" of NUREG-1828 provides the commitments, their implementation schedule, and source references.

The ANO-2 RVI AMP was developed based on the NRC-approved version of Electric Power Research Institute (EPRI) topical report, Materials Reliability Program (MRP)-227-A, "Material Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" (Reference 7). By letter dated June 22, 2011 (Reference 8), the NRC issued the first version of its safety evaluation (SE) for Revision 0 of MRP-227 (Reference 9). On July 21, 2011, the NRC issued Regulatory Issue Summary 2011-07, "License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management" (Reference 10), to provide guidance to pressurized water reactor (PWR) LR applicants and renewed license holders for the submittal of plant-specific AMPs for RVI components. On December 16, 2011, the NRC issued Revision 1 of its SE (Reference 11) for MRP-227, which is included in MRP-227-A (Reference 7).

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," addresses the requirements for plant license renewal. Section 54.21, "Contents of application: Technical information," of 10 CFR 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."

Section 2.F, "Updated Final Safety Analysis Report Supplement," of Renewed Facility Operating License No. NPF-6 (Reference 12) specifies that the future activities described in the ANO-2 SAR Supplement be completed prior to entering the PEO. The ANO-2 SAR Supplement is located in Appendix A of the LRA (Reference 5), as amended by the applicable licensee letters. License renewal Commitment Nos. 15 and 19 are related to the implementation of the activities described in ANO-2 SAR Supplement Sections A.2.1.23, "Reactor Vessel Internals Cast Austenitic Stainless Steel (CASS) Program," and A.2.1.24, "Reactor Vessel Internals Stainless Steel Plates, Forgings, Welds, and Bolting Program," respectively.

The NRC staff's final SE (Reference 11) 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 MRP-227 as the basis for an RVI AMP submittal to the NRC. On January 9, 2012, EPRI issued the NRC-approved (e.g., "-A") version of the topical report, MRP-227-A (Reference 7), 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 LR 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 LR 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 (Reference 13) 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 14), providing new generic AMR line items and generic AMP criteria in Chapter XI.M16A, "PWR Vessel Internals" (GALL Report AMP XI.M16A). The GALL Report AMP XI.M16A was developed with expectations for 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 LR Interim Staff Guidance (ISG) in

LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors" (Reference 15), which modified the criteria of GALL Report AMP XI.M16A to be consistent with MRP-227-A.

3.0 TECHNICAL EVALUATION

The licensee's letter dated July 18, 2016 (Reference 1) includes an attachment entitled "PWR Internals Aging Management Program Plan for Arkansas Nuclear One, Unit 2." This attachment contains the RVI AMP elements and supporting information. The attachment will be referred to as the RVI Plan in this NRC staff assessment to avoid confusion when discussing the RVI AMP. The ANO-2 RVI Plan contains six sections that are organized in the following manner:

- Section 1.0: Introduction
- Section 2.0: Aging Management Approach
- Section 3.0: ANO-2 Reactor Vessel Internals Design and Operating Experience
- Section 4.0: Examination and Acceptance and Expansion Criteria
- Section 5.0: Responses to NRC Safety Evaluation Applicant/Licensee Action Items
- Section 6.0: References

The information within these six sections is provided for NRC staff review and approval. Sections 1.0 and 6.0 do not contain specific technical information that affects the review and approval of the ANO-2 RVI AMP. Therefore, the focus of the NRC staff's review will be on Sections 2.0 through 5.0.

3.1 Age-Related Degradation and Aging Management Strategy

3.1.1 Licensee Evaluation of Age-Related Degradation and Aging Management Strategy

Section 2.1, "Mechanisms of Age-Related Degradation in PWR Internals," of the ANO-2 RVI Plan states that the age-related degradation mechanisms that impact the ANO-2 RVI are documented in the LRA AMR tables (Reference 5). Specifically, LRA Table 3.1.2-2, "Reactor Vessel Internals (Combustion), Summary of Aging Management," lists the plant-specific AMR line items for the RVI. The age-related degradation mechanisms use during the initial screening of the RVI components are described in the following ANO-2 submittal subsections:

- 2.1.1: Stress Corrosion Cracking
- 2.1.2: Irradiation-Assisted Stress Corrosion Cracking
- 2.1.3: Wear
- 2.1.4: Fatigue
- 2.1.5: Thermal Aging Embrittlement
- 2.1.6: Irradiation Embrittlement
- 2.1.7: Void Swelling and Irradiation Growth
- 2.1.8: Thermal and Irradiation-Enhanced Stress Relaxation or Creep

Section 2.2, "Aging Management Strategy," of the ANO-2 RVI Plan states that the guidelines provided in MRP-227-A were used in the development of the ANO-2 RVI AMP. Section 2.2 also provides a high level overview of the screening and ranking processes used to identify RVI components for inspection and the approach used to categorize components.

3.1.2 NRC Staff Assessment of Age-Related Degradation and Aging Management Strategy

The NRC staff's assessment of the ANO-2 AMR line items for the RVI is documented in NUREG-1828 (Reference 6), Section 3.1.2.3.2, "Reactor Vessel Internals." The NRC staff confirmed that the eight aging mechanisms identified in Section 2.1 of the ANO-2 RVI Plan are consistent with those identified in MRP-227-A. The NRC staff also confirmed that the aging effects associated with the aging mechanisms are appropriately identified in LRA Table 3.1.2-2.

The NRC staff reviewed and confirmed that the screening and ranking processes used to identify RVI components for inspection are consistent with the guidance provided in MRP-227-A. The NRC staff also confirmed that the criteria used to categorize components as "Primary," "Expansion," "Existing Programs," and "No Additional Measures" in the ANO-2 submittal is consistent with MRP-227-A. The NRC staff finds the information in the ANO-2 RVI Plan related to the identification of age related degradation mechanisms and aging management strategy acceptable.

3.2 Reactor Vessel Internals Aging Management Program Attributes

3.2.1 Licensee Evaluation of Reactor Vessel Internals Aging Management Program Attributes

Section 2.3, "ANO-2 Reactor Vessel Internals Aging Management Program Attributes," of the ANO-2 RVI Plan states that the ANO-2 RVI AMP is compliant with the 10 elements of GALL Report AMP XI.M16A. The licensee provided its evaluation of the 10 AMP elements against the corresponding elements in GALL Report AMP XI.M16A. The licensee also states that each of the 10 ANO-2 RVI AMP elements is consistent with GALL Report AMP XI.M16A (Reference 14), Appendix A of NUREG-1828 (Reference 6) and ANO-2 SAR (Reference 3).

3.2.2 NRC Staff Assessment of Reactor Vessel Internals Aging Management Program Attributes

The NRC staff reviewed the licensee's AMP against the 10 elements of the GALL Report AMP XI.M16A, as revised by LR-ISG-2011-04 (Reference 15). The NRC staff determined that the 10 program elements of the ANO-2 RVI AMP are consistent with the 10 program elements described in LR-ISG-2011-04. Therefore, the NRC staff finds the licensee's implementation of the 10 AMP elements acceptable for ANO-2.

3.3 Reactor Vessel Internals Design

3.3.1 Licensee Evaluation of ANO-2 Reactor Vessel Internal Design

Section 3.0, "ANO-2 Reactor Vessel Internals Design and Operating Experience," of the ANO-2 RVI Plan focuses primarily on the RVI design. ANO-2 is a two-loop Combustion Engineering (CE) designed nuclear steam supply system. Sections 3.1 through 3.6 of the ANO-2 RVI Plan describe the ANO-2 core support structure, core shroud assembly, flow skirt, upper guide structure assembly, in-core instrumentation (ICI) support system, and general design distinctions. Section 3.7, "ANO-2 Unit Operating Experience," addresses the replacement of zirconium alloy thimble tubes.

3.3.2 NRC Staff Assessment of ANO-2 Reactor Vessel Internal Design

The NRC staff reviewed the licensee's description of the ANO-2 RVI design against the ANO-2 SAR (Reference 3) Section 4.2.2, "Reactor Internals." The NRC staff noted that the core shroud is fabricated from two vertical assemblies that are welded at the half height of the assembly. The NRC staff also noted that the core shroud assembly is completely welded and does not utilize bolts. The NRC staff finds the licensee's description of the ANO-2 RVI consistent with its SAR. The replacement of the zirconium alloy ICI thimble tubes is reviewed in Section 3.4.2 of this staff assessment. The NRC staff finds the description of the ANO-2 RVI design consistent with its SAR and, therefore, acceptable.

3.4 Operating Experience

3.4.1 Licensee Evaluation of ANO-2 Reactor Vessel Internals Operating Experience

Section 2.3.10.1, "ANO-2 Operating Experience," and Section 3.7, "ANO-2 Unit Operating Experience," of the ANO-2 RVI Plan discuss the operating experience (OE) relevant to the age related degradation of the ANO-2 RVI. Section 2.3.10.1 states that extensive industry and ANO-2 OE has been reviewed during the development of the RVI AMP. It also states that OE relevant to aging has been compiled into the ANO-2 OE program (Reference 16). The licensee stated that it submitted a survey of RVI inspection results in support of a Pressurized Water Reactor Owners Group (PWROG) project, which were compiled in Westinghouse Commercial Atomic Power (WCAP) report WCAP-17435-NP (Reference 17). Section 3.7 of the ANO-2 RVI AMP submittal states that zirconium alloy thimble tubes were replaced at ANO-2 due to irradiation induced growth.

3.4.2 NRC Staff Assessment of ANO-2 Reactor Vessel Internals Operating Experience

The NRC staff reviewed WCAP-17435-NP, Revision 1 (Reference 18), "Results of the Reactor Internals Operating Experience Survey Conducted under PWROG Project Authorization PA-MSC-0568." The NRC staff noted that the licensee was a participant in this PWROG project; however, WCAP-17435-NP, Table 3-3, "List of Plants that Responded to the Survey," indicates that ANO-2 did not provide plant-specific OE for the survey. The only recordable indication reported in WCAP-17435-NP, Revision 1, for a CE plant was associated with the core support plate. The NRC staff also reviewed the biennial reports for the MRP-227-A reactor internals inspection results, MRP 2014-009, "Biennial Report of Recent MRP-227-A Reactor Internals Inspection Results (Project 694)" (Reference 19), and MRP 2016-008, "Biennial Report of MRP-227-A Reactor Internals Inspection Results" (Reference 20). MRP 2016-008, provides the inspection results for the only CE plant to perform MRP-227-A inspections to date. The CE inspection results did not reveal any indications or conditions that required action. Therefore, there have been no inspections of CE expansion components to date.

By letter dated May 9, 2017 (Reference 21), the NRC staff issued Request for Additional Information (RAI)-3 requesting that the licensee provide a summary of any ANO-2 plant-specific OE relevant to age-related degradation of RVI components, including the plant-specific OE contained in Appendix A of MRP-227-A. The RAI also requested that the licensee describe the actions taken to address this OE.

By letter dated September 6, 2017 (Reference 2), the licensee responded to RAI-3 by providing a summary of its plant-specific OE relevant to age-related degradation of RVI components. The ANO-2 Type 403 stainless steel holddown ring experienced stress relaxation during the early fueling cycles. The holddown ring was inspected during refueling outages for approximately 20 years, and the deflection resulting from the stress relaxation was monitored. The deflection in the holddown ring stabilized at an acceptable level, and no further action was necessary.

The licensee's responses to RAI-3 and RAI-4 (Reference 2) discuss the ANO-2 OE related to the ICI thimble tubes. The licensee stated that the ANO-2 Zircalloy-4 ICI thimble tubes experienced greater than expected irradiation induced growth, similar to other CE plants. ANO-2 replaced the majority of its ICI thimble tubes in its refueling outage in the fall of 2006. The response also states that periodic measurements will continue to monitor the growth and inform the replacement of ICI thimble tubes. Additionally, the licensee provided a summary description of the plant-specific aging management activities associated with the growth of the ICI thimble tubes and stated that the aging management of the ICI thimble tubes at ANO-2 is performed in accordance with the guidance provided by Westinghouse.

The NRC staff finds the licensee's response to RAI-3 and RAI-4 acceptable because the ANO-2 AMP bounds the applicable OE available to date. Also, the results of the periodic inspections of the ICI thimble tube are used to analyze and project the growth, which ensures they are replaced in an appropriate timeframe. The NRC staff finds the licensee's discussion of the ANO-2 plant-specific OE acceptable.

3.5 Examination, Acceptance, and Expansion Criteria

3.5.1 Licensee Evaluation of ANO-2 Examination, Acceptance, and Expansion Criteria

Section 4.0, "Examination and Acceptance and Expansion Criteria," of the ANO-2 RVI Plan discusses the examination methods, acceptance criteria, expansion criteria, corrective actions, and reporting of findings relevant to the age related degradation of the ANO-2 RVI. Section 4.1, "Examination Acceptance Criteria," of the ANO-2 RVI Plan describes the examination methods applicable to the RVI and their associated acceptance criteria. Section 4.1.2, "Visual (VT-1) Examination," states that visual examination, per ASME Code Section XI (VT-1), is used to detect gaps between the upper-to-lower mating surfaces of vertical sections of the core shroud. Section 4.1.6, "Physical Measurements Examination," states that if the VT-1 examination detects a gap between the mating surfaces of vertical sections of the core shroud, then physical measurements must be performed. ANO-2 RVI Plan Section 5.5, "SE Section 4.2.5, Applicant/Licensee Action Item 5," is referenced within Section 4.1.6 for more details on the physical measurements examinations.

The criteria for scope expansion is provided in the ANO-2 RVI Plan Table 5-4, "CE Plants Examination Acceptance and Expansion Criteria from Table 5-2 of MRP-227-A." The licensee states in ANO-2 RVI Plan Section 4.3, "Evaluation, Repair, and Replacement Strategy," that NRC-approved methodologies will be used to determine acceptability of detected conditions. The licensee further states that reporting and documentation of relevant conditions that do not meet the examination acceptance criteria will be performed in accordance with MRP-227-A and the ANO-2 Corrective Action Program.

3.5.2 NRC Staff Assessment of ANO-2 Examination, Acceptance, and Expansion Criteria

The NRC staff reviewed the licensee's response to Action Item 5 in ANO-2 RVI Plan Section 5.5. Action Item 5 states, in part, that licensees shall identify plant-specific acceptance criteria to be applied when performing physical measurements required by MRP-227-A for a gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections. Action Item 5 also requires that the licensee include its proposed acceptance criteria as part of its submittal to the NRC. ANO-2 RVI Plan, Section 5.5, states that if a gap between the mating surfaces of vertical sections of the core shroud is identified, then an evaluation is performed. However, acceptance criteria is not provided for the examination.

By letter dated May 9, 2017 (Reference 21), the NRC staff issued RAI-5 requesting that the licensee provide acceptance criteria for the examination used for the physical measurement of the gap between the top and bottom core shroud segments in ANO-2. The RAI also requested that the licensee justify that the acceptance criteria is consistent with the licensing basis to ensure that the core shroud remains capable of performing its required functions.

By letter dated September 6, 2017 (Reference 2), the licensee responded to RAI-5 by providing an acceptance criteria of 0.125 inch for the gap between the top and bottom core shroud segments in ANO-2. The licensee stated that it determined the maximum expected gap during shutdown by combining the maximum allowable as-fabricated gap and maximum gap expected due to void swelling. The potential adverse effects of the resulting maximum gap were determined to be acceptable. This maximum gap of 0.125 inch is set as the acceptance criteria.

The NRC staff finds the licensee's response to RAI-5 acceptable because the acceptance criteria for the gap between the top and bottom core shroud segments is a bounding value for a postulated gap between the top and bottom core shroud segments in ANO-2. The NRC staff finds the licensee's description of the ANO-2 RVI examination methods, acceptance criteria, expansion criteria, corrective actions, and reporting criteria acceptable.

3.6 Licensee Action Items of Safety Evaluation for MRP-227-A

3.6.1 Action Item 1 - Applicability of Failure Modes, Effects, and Criticality Analysis and Functionality Analysis Assumptions

Section 4.2.1, "Applicability of FMECA [Failure Modes, Effects, and Criticality Analysis] and Functionality Analysis Assumptions," of the SE for MRP-227-A states, in part, the following:

> [E]ach applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W [Babcock and Wilcox]) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227. This is Applicant/Licensee Action Item 1.

3.6.1.1 Licensee Evaluation of Action Item 1

Section 5.1 "SE Section 4.2.1, Applicant/Licensee Action Item 1 (Applicability of FMECA and Functionality Analysis Assumptions)," of the ANO-2 RVI Plan states, in part, that "[t]he assumptions regarding plant design and operating history made in MRP-191 ["Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191)"] (Reference 22)] are appropriate for ANO-2. Section 1.8.4.1, "MRP-227-A Applicability to ANO-2," of the ANO-2 RVI Plan states that ANO-2 conforms to the following assumptions in Section 2.4, "Guidelines Applicability," of MRP-227-A:

- ANO-2 historic core management practices are bounded by the general assumption of 30 years of operation with high leakage core loading patterns followed by implementation of a low-leakage fuel management strategy for the remaining 30 years of operation;

- ANO-2 operates as a base load unit;
- ANO-2 has not made modification of the RVI components beyond those identified in general industry guidance or recommended by CE since May 2007.

The licensee also stated that it is participating in a PWROG program aimed at addressing the 20 percent cold work issue for austenitic stainless steel components on a generic basis rather than plant-specific basis. The final report for this effort was issued in April 2016 (PWROG-15105-NP (Reference 23)).

3.6.1.2 NRC Staff Assessment of Action Item 1

In order to resolve the generic issue of the information needed from licensees to resolve Action Item 1, a series of proprietary and public meetings were conducted, at which the NRC, Westinghouse, EPRI, and utility representatives discussed regulatory concerns and determined a path for a comprehensive and consistent utility response to demonstrate applicability of MRP-227-A, specifically for Westinghouse and CE-design PWR RVI. A summary of the proprietary meeting presentations and supporting proprietary generic design bases information are contained in Westinghouse proprietary report WCAP-17780-P, "Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs" (Reference 24). This report provides background on the proprietary design information regarding variances in stress, fluence, and temperature in the RVI components that were designed by Westinghouse and CE to support NRC reviews of utility submittals to demonstrate plant-specific applicability of MRP-227-A.

As a result of the technical discussions with the NRC staff, the basis for a plant to respond to the NRC staff RAI questions regarding Action Item 1, to demonstrate compliance with MRP-227-A for originally licensed and uprated conditions, was determined to be satisfied with plant-specific responses to the following two questions:

Question 1: Does the plant have non-weld or bolting austenitic stainless steel (SS) components with 20 percent cold work or greater, and, if so, do the affected components have operating stresses greater than 30 ksi [kilo-pounds per square inch]? (If both conditions are true, additional components may need to be screened in for stress corrosion cracking, SCC.)

Question 2: Does the plant have atypical fuel design or fuel management that could render the assumptions of MRP-227-A, regarding core loading/core design, non-representative for that plant?

MRP Letter 2013-025, "MRP-227-A Applicability Template Guideline," dated October 14, 2013 (Reference 25), provides guidance for responding to the two questions above. An NRC SE (Reference 26), assessed MRP Letter 2013-025 and the technical basis contained in WCAP-17780-P, and concluded that if a licensee demonstrates that its plant complies with the guidance in MRP Letter 2013-025, there is reasonable assurance that the guidance of MRP-227-A will be applicable to the specific plant. The NRC staff evaluation also concluded that the guidance in MRP Letter 2013-025 provides an acceptable basis for licensees to respond to the generic Questions 1 and 2 addressed above.

The NRC staff reviewed the information provided in PWROG-15105-NP to determine if the generic evaluation of the 20 percent cold work issue for austenitic stainless steel components is consistent with the guidance in MRP Letter 2013-025 and adequately addresses plant-specific Question 1 for ANO-2. The NRC staff noted that the information provided in PWROG-15105-NP was partially reviewed in Section 3.3, "Applicant/Licensee Action Item 1 of SE for MRP-227-A," of NRC staff SE dated October 24, 2016 (Reference 27) and in the NRC staff assessment dated April 21, 2017 (Reference 28). Therefore, the NRC staff will not reiterate the generically applicable details of PWROG-15105-NP in this document. The NRC staff has determined that the information provided in PWROG-15105-NP is applicable to ANO-2, adequately addresses Question 1, and provides reasonable assurance that the ANO-2 RVI does not contain austenitic stainless steel with 20 percent cold work or greater. This determination is based on the PWROG's evaluation of: component drawings; material specification requirements; heat treatment; cold work induced by fabrication; and limitations imposed on material strength and hardness.

By letter dated May 9, 2017 (Reference 21), the NRC staff issued RAI-1 requesting that the licensee provide a plant-specific response to Question 2. By letter dated September 6, 2017 (Reference 2), the licensee responded to RAI-1 by stating that ANO-2 has not utilized atypical fuel design or fuel management that could render the assumptions of MRP-227-A non-representative. The response also provides the applicable plant-specific information for the MRP Letter 2013-025 screening criteria.

The NRC staff finds the licensee's response to RAI-1 acceptable because the licensee followed the guidance in MRP Letter 2013-025 (Reference 25), including Appendix B, "MRP-227-A Applicability Guideline for CE and Westinghouse Pressurized Water Reactor Designs, Question 2 Response Template." The NRC SE (Reference 26) for MRP Letter 2013-025 and WCAP-17780-P concluded that if an applicant or licensee demonstrates that its plant complies with the guidance in MRP Letter 2013-025, there is reasonable assurance that the inspection and evaluation (I&E) guidance of MRP-227-A will be applicable to the specific plant. The NRC staff finds the licensee's evaluation of Action Item 1 acceptable.

3.6.2 Action Item 2 - PWR Vessel Internal Components Within the Scope of License Renewal

Section 4.2.2, "PWR Vessel Internal Components Within the Scope of License Renewal," of the SE for MRP-227-A states, in part, the following:

> [E]ach applicant/licensee is responsible for identifying which RVI components are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RVI components that are within

> the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by this SE, when submitting its plant-specific AMP. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the period of extended operation. This issue is Applicant/Licensee Action Item 2.

3.6.2.1 Licensee Evaluation of Action Item 2

Section 5.2 "SE Section 4.2.2, Applicant/License Action Item 2 (PWR Vessel Internal Components Within the Scope of License Renewal)," of the ANO-2 RVI Plan states, in part, that the "information contained in Table 4-5 ["Components and Materials for CE-Designed Plants,"] of MRP-191 [Reference 22] was reviewed and that review determined that this table contained all of the RVI components that are within the scope of license renewal for ANO-2." ANO-2 RVI Plan, Section 1.7.1, describes the aging management review and LRA Table 3.1.2-2 (Reference 5) summarizes the results.

3.6.2.2 NRC Staff Assessment of Action Item 2

The NRC staff reviewed ANO-2 RVI Plan, Table 5-1, "CE Plants Primary Category Components from Table 4-2 of MRP-227-A," and Table 5-2, "CE Plants Expansion Category Components from Table 4-5 of MRP-227-A," as revised by letter dated September 6, 2017 (Reference 2). The NRC staff confirmed that these tables contain all of the RVI components that are within the scope of LR for ANO-2. The NRC staff's assessment of the ANO-2 AMR line items for the RVI is documented in NUREG-1828 (Reference 6), Section 3.1.2.3.2. The NRC staff finds the licensee's evaluation of Action Item 2 acceptable.

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

Section 4.2.3, "Evaluation of the Adequacy of Plant-Specific Existing Programs," of the SE for MRP-227-A states, in part, the following:

> [A]applicants/licensees of CE and Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to the programs that should be implemented to manage the aging of these components for the period of extended operation. The results of this plant-specific analyses and a description of the plant-specific programs being relied on to manage aging of these components shall be submitted as part of the applicant's/licensee's AMP application. The CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (Section 4.3.2 in MRP-227), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227). This is Applicant/Licensee Action Item 3.

3.6.3.1 Licensee Evaluation of Action Item 3

Section 5.3, "SE Section 4.2.3, Applicant/Licensee Action Item 3 (Evaluation of the Adequacy of Plant-Specific Existing Programs)," of the ANO-2 RVI Plan states that ANO-2 complies with Action Item 3 through management and replacement of in-core instrumentation thimble tubes.

The licensee also states that thermal shields are not present in the ANO-2 reactor vessel internals.

3.6.3.2 NRC Staff Assessment of Action Item 3

The NRC staff reviewed ANO-2 SAR Table 3.9-5 (Reference 3), "Comparison of Structural and Hydraulic Design Parameters for Reactor Internals," and confirmed that thermal shields are not present in the design of the ANO-2 reactor vessel internals. The NRC staff RAI-3 and RAI-4, which are related to the ANO-2 OE and aging management of the ICI thimble tubes are evaluated in Section 3.4.2 of this document and found acceptable. The NRC staff finds the licensee's evaluation of Action Item 3 acceptable.

3.6.4 Action Item 4 - B&W Core Support Structure Upper Flange Stress Relief

Section 4.2.4, "B&W Core Support Structure Upper Flange Stress Relief," of the SE for MRP-227-A states, in part, the following:

> [T]he B&W applicants/licensees shall confirm that the core support structure upper flange weld was stress relieved during the original fabrication of the Reactor Pressure Vessel in order to confirm the applicability of MRP-227, as approved by the NRC, to their facility. If the upper flange weld has not been stress relieved, then this component shall be inspected as a "Primary" inspection category component. If necessary, the examination methods and frequency for non-stress relieved B&W core support structure upper flange welds shall be consistent with the recommendations in MRP-227, as approved by the NRC, for the Westinghouse and CE upper core support barrel welds. The examination coverage for this B&W flange weld shall conform to the [NRC] staff's imposed criteria as described in Sections 3.3.1 and 4.3.1 of this SE. The applicant's/licensee's resolution of this plant-specific action item shall be submitted to the NRC for review and approval. This is Applicant/Licensee Action Item 4.

3.6.4.1 Licensee Evaluation of Action Item 4

Section 5.4, "SE Section 4.2.4, Applicant/Licensee Action Item 4 (B&W Core Support Structure Upper Flange Stress Relief)," of the ANO-2 RVI Plan states that Action Item 4 is not applicable to ANO-2 because it is a CE plant.

3.6.4.2 NRC Staff Assessment of Action Item 4

The NRC staff acknowledges that Action Item 4 addresses the aging management of B&W components and is not applicable to ANO-2, which is a CE designed unit.

3.6.5 Action Item 5 - Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components

Section 4.2.5, "Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components," of the SE for MRP-227-A states, in part, the following:

> [A]pplicants/licensees shall identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the

> NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections. The applicant/licensee shall include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP-227. This is Applicant/Licensee Action Item 5.

3.6.5.1 Licensee Evaluation of Action Item 5

Section 5.5, "SE Section 4.2.5, Applicant/Licensee Action Item 5 (Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components)," of the ANO-2 RVI Plan states that if a gap between the two core barrel shroud vertical sections is identified during the required VT-1 examination, measurements of the gap opening is required. It is also stated that an evaluation of the gap shall be performed to determine the frequency and method for additional examinations. The licensee further states that criteria consistent with the licensing basis will be developed to ensure that the core shroud remains capable of performing its required functions.

3.6.5.2 NRC Staff Assessment of Action Item 5

The NRC staff RAI-5, which is related to the acceptance criteria for the physical measurement of the gap between the top and bottom core shroud segments in ANO-2, is discussed in Section 3.5.2 of this staff assessment and found acceptable. The NRC staff finds the licensee's evaluation of Action Item 5 acceptable.

3.6.6 Action Item 6 - Evaluation of Inaccessible B&W Components

Section 4.2.6, "Evaluation of Inaccessible B&W Components," of the SE for MRP-227-A states, in part, the following:

> MRP-227 does not propose to inspect the following inaccessible components: the B&W core barrel cylinders (including vertical and circumferential seam welds), B&W former plates, B&W external baffle-to-baffle bolts and their locking devices, B&W core barrel-to-former bolts and their locking devices, and B&W core barrel assembly internal baffle-to-baffle bolts. The MRP also identified that although the B&W core barrel assembly internal baffle-to-baffle bolts are accessible, the bolts are non-inspectable using currently available examination techniques.
>
> Applicants/licensees shall justify the acceptability of these components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components. As part of their application to implement the approved version of MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible components and, if necessary, provide their plan for the replacement of the components for NRC review and approval. This is Applicant/Licensee Action Item 6.

3.6.6.1 Licensee Evaluation of Action Item 6

Section 5.6, "SE Section 4.2.6, Applicant/Licensee Action Item 6 (Evaluation of Inaccessible B&W Components)," of the ANO-2 RVI Plan states that ANO-2 Action Item 6 is not applicable to ANO-2 because it is a CE plant.

3.6.6.2 NRC Staff Assessment of Action Item 6

The NRC staff acknowledges that Action Item 6 addresses the aging management of B&W components and is not applicable to ANO-2, which is a CE designed unit.

3.6.7 Action Item 7 - Plant-Specific Evaluation of CASS Materials

Section 4.2.7, "Plant-Specific Evaluation of CASS Materials," of the SE for MRP-227-A states, in part, the following:

> [A]pplicants/licensees of B&W, CE, and Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI [Incore Monitoring Instrumentation] guide tube assembly spiders and CRGT [Control Rod Guide Tube] spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS, martensitic stainless steel or precipitation hardened stainless steel materials. These analyses shall also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The requirement may not apply to components that were previously evaluated as not requiring aging management during development of MRP-227. That is, the requirement would apply to components fabricated from susceptible materials for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/Licensee Action Item 2. The plant-specific analysis shall be consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. The applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227. This is Applicant/Licensee Action Item 7.

3.6.7.1 Licensee Evaluation of Action Item 7

Section 5.7, "SE Section 4.2.7, Applicant/Licensee Action Item 7 (Plant-Specific Evaluation of CASS Materials)," of the ANO-2 RVI Plan states that ANO-2 does not have lower support columns fabricated from CASS, martensitic stainless steel, or precipitation hardened stainless steel as part of the reactor vessel internals. The licensee states that the ANO-2 lower support columns are fabricated from Type 304 stainless steel. Additionally, in the licensee's response (Reference 2) to RAI-2 it confirmed that the ANO-2 RVI does not contain components fabricated from Type 347 stainless steel, Type 341 stainless steel, 17-4 PH (precipitation hardened) stainless steel, or 15-5 PH.

The licensee states that the control element assemblies (CEAs) shroud tube is the only component fabricated from CASS material listed in the LRA, Table 3.1.2-2. The licensee also stated that the CEA shroud tubes were initially screened in for SCC and thermal embrittlement but the FMECA determined that these age-related degradation mechanisms have minimal likelihood to cause failure. Therefore, the CEA shroud tubes were designated as "Category A" components in accordance with MRP-191.

3.6.7.2 NRC Staff Assessment of Action Item 7

The NRC staff reviewed ANO-2 SAR (Reference 3) Section 4.2.2.2.1.1, "Core Support Structure," and confirmed that the lower support structure, which consists in part of the core support columns, is fabricated from Type 304 stainless steel. Since the CE lower support columns are not fabricated from a material susceptible to loss of fracture toughness due to thermal and irradiation embrittlement, the NRC staff finds the portion of Action Item 7 related to development of a plant-specific analyses demonstrating functionality of the CE lower support columns during the period of extended operation, is not applicable to ANO-2.

Action Item 7 does not apply to components evaluated as not requiring aging management during development of MRP-227. The CEA shroud tubes were designated as "Category A" components. SE Section 2.2 of MRP-227-A, summarizes "Category A" as either: (a) those components that have been judged to be not susceptible to any of the eight degradation mechanisms or (b) those components that have been judged to be somewhat susceptible to one or more aging degradation mechanisms but are not expected to lose functionality. Therefore, Action Item 7 is not applicable to the CEA shroud tubes. The CEA shroud tubes will be managed by the ANO-2 inservice inspection program and are not applicable to Action Item 7. The NRC staff finds the licensee's evaluation of Action Item 7 acceptable.

3.6.8 Action Item 8 - Submittal of Information for Staff Review and Approval

Section 4.2.8, "Submittal of Information for Staff Review and Approval," of the SE for MRP-227-A states the following:

> [A]pplicants/licensees shall make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE, as an AMP for the RVI components at their facility. This submittal shall include the information identified in Section 3.5.1 of this SE. This is Applicant/Licensee Action Item 8.

Section 3.5.1, "Submittal of Information for Staff Review and Approval," of the SE for MRP-227-A states, in part, the following:

> In addition to the implementation of MRP-227 in accordance with NEI 03-08, applicants/licensees whose licensing basis contains a commitment to submit a PWR RVI AMP and/or inspection program shall also make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE. An applicant's/licensee's application to implement MRP-227, as amended by this SE shall include the following items (1) and (2). Applicants who submit applications for LR after the issuance of this SE shall, in accordance with the NUREG-1801, Revision 2, submit the information provided in the following items (1) through (5) for staff review and approval.

1. An AMP for the facility that addresses the 10 program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A.

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

3.6.8.1 Licensee Evaluation of Action Item 8

Section 5.8, "SE Section 4.2.8, Applicant/Licensee Action Item 8 (Submittal of Information for Staff Review and Approval)," of the ANO-2 RVI Plan states that the attributes of the ANO-2 RVI AMP and their compliance with the 10 program elements of NUREG-1801, Revision 2, AMP XI.M16A, that are essential for successful management of component aging are described in Section 2.3 of the licensee's submittal. Section 2.3 of the ANO-2 RVI Plan discusses each AMP element and individually concludes that each program element is consistent with NUREG-1801, Revision 2, AMP XI.M16A (Reference 14). The "conclusions" subsection for each individual ANO-2 AMP element also states that it is consistent with applicable LR commitments in Appendix A of the LR safety evaluation report (Reference 6) and ANO-2 SAR (Reference 3). Furthermore, the inspection of "Primary," "Expansion," and components credited as part of plant specific existing programs identified in Table 5-1 through Table 5-4 of the ANO-2 AMP Plan, as amended by letter dated September 6, 2017, will be performed in accordance with MRP-227-A.

3.6.8.2 NRC Staff Assessment of Action Item 8

The attributes of the ANO-2 AMP elements are reviewed in Section 3.2 of this staff assessment and found to be acceptable. The NRC staff also confirmed that the "Primary," "Expansion," and credited components identified in Table 5-1 through Table 5-4 of the ANO-2 AMP Plan, as amended by letter dated September 6, 2017, are consistent with MRP-227-A. Therefore, the NRC staff finds the licensee's evaluation of Action Item 8 acceptable.

4.0 CONCLUSION

The NRC staff has reviewed the ANO-2 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.

The NRC staff assessment of the ANO-2 RVI AMP does not reduce, alter, or otherwise affect the current ASME Code, Section XI ISI requirements, or any other ANO-2 licensing basis requirements related to ISI of structures, systems, and components. The NRC 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

1. Pyle, S. L., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Reactor Vessel Internals Aging Management Program Plan, Arkansas Nuclear One – Unit 2," dated July 18, 2016 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML16202A168).

2. Pyle, S. L., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Response to Request for Additional Information Reactor Vessel Internals Aging Management Program Plan Arkansas Nuclear One, Unit 2," dated September 6, 2017 (ADAMS Accession No. ML17251A758).

3. Browning, J. G., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "Amendment 26 to the ANO Unit 2 Safety Analysis Report Arkansas Nuclear One, Unit 2, Docket Number 50-368, License No. NPF-6, dated April 28, 2016 (not publicly available).

4. James, D. E., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "License Renewal Application Clarifications TAC No. MB8402, Arkansas Nuclear One – Unit 2, Docket No. 50-368, License No. NPR-6," dated September 10, 2004 (ADAMS Accession No. ML042660110).

5. Anderson C., Entergy Operations, Inc., letter to U.S. Nuclear Regulatory Commission, "License Renewal Application, Arkansas Nuclear One – Unit 2, Docket No. 50-368, License No. NPF-6," dated October 14, 2003 (ADAMS Package Accession No. ML032890483).

6. U.S. Nuclear Regulatory Commission, NUREG-1828, "Safety Evaluation Report Related to the License Renewal of the Arkansas Nuclear One, Unit 2," June 2005 (ADAMS Accession No. ML051730233).

7. Electric Power Research Institute, Material Reliability Program, letter to U.S. Nuclear Regulatory Commission, "Transmittal: PWR Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," dated January 9, 2012 (ADAMS Package Accession No. ML120170453).

8. Nelson, R. A., U.S. Nuclear Regulatory Commission, letter to Neil Wilmshurst, Electric Power Research Institute, "Final Safety Evaluation of EPRI Report, Materials Reliability Program Report 1016595 (MRP-227), Revision 0, 'Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines,' (TAC No. ME0680)" dated June 22, 2011 (ADAMS Accession No. ML111600498).

9. Larsen, C. B., 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, Revision 0)," dated January 12, 2009 (ADAMS Package Accession No. ML090160212).

10. 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).

11. Nelson, R. A., U.S. Nuclear Regulatory Commission, letter to Neil Wilmshurst, Electric Power Research Institute, "Revision 1 to the Final Safety Evaluation of Electric Power Research Institute (EPRI) Report, Materials Reliability [Program] (MRP) Report 1016596 (MRP-227), Revision 0, 'Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines," dated December 16, 2011 (ADAMS Accession No. ML11308A770).

12. Arkansas Nuclear One, Unit 2, Renewed Facility Operating License No. NPF-6, Entergy Operations, Inc., June 30, 2005 (ADAMS Accession No. ML053130317).

13. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components."

14. U.S. Nuclear Regulatory Commission, NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," Final Report, December 2010 (ADAMS Accession No. ML103490041).

15. Lubinski, J. W., U.S. Nuclear Regulatory Commission, letter to Ms. Jean Smith, Electric Power Research Institute, "License Renewal Interim Staff Guidance LR-ISG-2011-04: "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors," dated May 28, 2013 (ADAMS Accession No. ML12270A251).

16. Entergy Procedure EN-OE-100, Revision 24, "Operating Experience Program," August 11, 2015.

17. Westinghouse, WCAP-17435-NP, Revision 0, "Results of the Reactor Internals Operating Experience Survey Conducted under PWROG Project Authorization PA-MSC-0568," September 2011.

18. Westinghouse, WCAP-17435-NP, Revision 1, "Results of the Reactor Internals Operating Experience Survey Conducted under PWROG Project Authorization PA-MSC-0568," October 2012 (ADAMS Accession No. ML16327A136; not publicly available).

19. Rudell, B. C. and Demma, A., Electric Power Research Institute, letter to U.S. Nuclear Regulatory Commission, "Biennial Report of Recent MRP-227-A Reactor Internals Inspection Results (Project 694)," MRP Technical Report No. 2014-009, dated May 12, 2011 (ADAMS Accession Nos. ML14135A383, ML14135A384, and ML14135A385).

20. Rudell, B. C. and Demma, A., Electric Power Research Institute, letter to U.S. Nuclear Regulatory Commission, "Biennial Report of MRP-227-A Reactor Internals Inspection Results," MRP 2016-008, dated May 18, 2016 (ADAMS Accession No. ML16144A789).

21. Wengert, T. J., U.S. Nuclear Regulatory Commission, letter to Entergy Operations, Inc., "Arkansas Nuclear One, Unit 2 - Request for Additional Information Regarding Reactor Vessel Internals Aging Management Plan Review (CAC No. MF8155)," dated May 9, 2017 (ADAMS Accession No. ML17123A297).

22. Electric Power Research Institute, "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191)," Technical Report No. 1013234, November 2006. (ADAMS Accession No. ML091910130).

23. Pressurized Water Reactor Owners Group Report, PWROG-15105-NP, Revision 0, "PWR RV Internals Cold-Work Assessment," Materials Committee PA-MSC-1288, April 2016. (ADAMS Accession No. ML16222A300).

24. Gresham, J. A., Westinghouse Electric Company, LLC, letter to U.S. Nuclear Regulatory Commission, "Submittal of WCAP-17780-P, 'Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs,'" June 28, 2013 (ADAMS Accession No. ML13183A372).

25. Demma, A. and Wells, T., Electric Power Research Institute, letter to U.S. Nuclear Regulatory Commission, "MRP-227-A Applicability Template Guideline," MRP Letter 2013-025, dated October 14, 2013 (ADAMS Accession No. ML13322A454).

26. U.S. Nuclear Regulatory Commission, "Office of Nuclear Reactor Regulation Evaluation of WCAP-17780-P, 'Reactor Internals Aging Management MRP-227-A Applicability for Combustion Engineering [(CE)] and Westinghouse [Electric Company (Westinghouse)] Pressurized Water Reactor Designs,' and MRP-227-A Applicability Guidelines for CE and Westinghouse Pressurized Water Reactor Designs (ADAMS Accession. No. ML14309A484).

27. Mozafari, B., U.S. Nuclear Regulatory Commission, letter to Mr. Bryan C. Hanson, Exelon Generation Company, LLC, "R.E. Ginna Nuclear Power Plant: Safety Evaluation Related to Reactor Vessel Internals Inspection Plan Based on MRP-227-A (TAC No. MF6713)," dated October 24, 2016 (ADAMS Accession No. ML16271A088).

28. U.S. Nuclear Regulatory Commission, "Office of Nuclear Reactor Regulation Summary Assessment of Pressurized-Water Reactor Owners Group – 15105-NP, Revision 0, 'PA-MSC-1288 Pressurized-Water Reactor Vessel Internals Cold Work Assessment,'" dated April 21, 2017 (ADAMS Accession No. ML17081A010).

Principal Contributor: C. Hovanec

Date: October 26, 2017

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 2 – STAFF ASSESSMENT REGARDING PROGRAM PLAN FOR AGING MANAGEMENT FOR REACTOR VESSEL INTERNALS (CAC NO. MF8155; EPID L-2016-LRO-0001)
DATED: OCTOBER 26, 2017

DISTRIBUTION:
PUBLIC
LPL4-2 R/F
RidsACRS_MailCTR Resource
RidsNrrDorlLpl4 Resource
CHovanec, NRR
SRuffin, NRR
RidsNrrLAPBlechman Resource
RidsNrrPMANO Resource
RidsRgn4MailCenter Resource

ADAMS Accession No. ML17291A009 ***via memorandum**

OFFICE	NRR/DORL/LPL4/PM	NRR/DORL/LPL4/LA	NRR/DMLR/MVIB/(A)BC*	NRR/DORL/LPL4/BC
NAME	TWengert	PBlechman	SRuffin	RPascarelli
DATE	10/25/17	10/24/17	10/13/2017	10/26/17

OFFICIAL RECORD COPY