Attachment 3 to

1CAN021503

AREVA document ANP-3383NP,

"Response to Request for Additional Information for the Reactor Pressure Vessel Internals Aging Management Program Plan for Arkansas Nuclear One Unit 1"

NON-PROPRIETARY

AREVA

Response to Request for Additional Information for the Reactor Pressure Vessel Internals Aging Management Program Plan for Arkansas Nuclear One Unit 1

ANP-3383NP
Revision 0

February 2015

AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
Revision 0	All	Original Issue

Contents

List of Tables

Nomenclature

Acronym	Definition
ANO-1/ANO1	Arkansas Nuclear One Unit 1
AMP	Aging Management Program
ASME	American Society of Mechanical Engineers
ASTM	ASTM International (formerly, American Society for Testing and Materials)
B&W	Babcock & Wilcox
B&WOG	Babcock & Wilcox Owner's Group
CASS	Cast Austenitic Stainless Steel
CFR	Code of Federal Regulations
Entergy	Entergy Operations, Inc.
EOC	End of Cycle
FIV	Flow-Induced Vibration
FD	Flow Distributor
FMECA	Failure, Modes, Effects, and Criticality Analysis
GALL	Generic Aging Lessons Learned
IBSP	Internals Bolting Surveillance Program
IGSCC	Intergranular Stress Corrosion Cracking
ISI	In-Service Inspection
LCB	Lower Core Barrel
LER	Licensee Event Report
LR	License Renewal
LRA	License Renewal Application
LTS	Lower Thermal Shield
MRP	Materials Reliability Program
NDE	Nondestructive Examination
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Support System
OE	Operating Experience
ONS-1	Oconee Nuclear Station Unit 1
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owner's Group
RAI	Request for Additional Information
RPV	Reactor Pressure Vessel
RV	Reactor Vessel

Acronym	Definition
RVI[1]	Reactor Vessel Internals
RVLMS	Reactor Vessel Level Monitoring System
SCC	Stress Corrosion Cracking
SE	Safety Evaluation
SI	Structural Integrity Associates
SSHT	Surveillance Specimen Holder Tube
TLAA	Time-Limited Aging Analysis
TS	Technical Specification
UCB	Upper Core Barrel
VT-3	Visual Testing (Level 3)
US	United States
UT	Ultrasonic Testing
UTS	Upper Thermal Shield

[1] RVI is used in the direct quotations from the NRC to indicate reactor vessel internals. In the AREVA responses, other forms such as reactor vessel internals or RV internals are used to avoid confusion with reactor vessel integrity (RVI), as it commonly used in other AREVA reports.

ABSTRACT

By letter dated May 20, 2014, Entergy Operations, Inc. (hereafter Entergy) submitted a Pressurized Water Reactor (PWR) Aging Management Program Plan for Arkansas Nuclear One Unit 1 to the United Stated Nuclear Regulatory Commission (US NRC). The US NRC has issued Requests for Additional Information (RAIs) on this submittal. This report provides the responses for RAIs 1, 2, 3, 4, 5, 6 (b), 7, 8, 11, 12, and 13.

1.0 INTRODUCTION AND SUMMARY

By letter dated May 20, 2014, Entergy Operations, Inc. (hereafter Entergy) submitted a Pressurized Water Reactor (PWR) Aging Management Program Plan for Arkansas Nuclear One Unit 1 (ANO-1) to the United Stated Nuclear Regulatory Commission (US NRC) (1). The US NRC has issued Requests for Additional Information (RAIs) on this submittal and this report provides the responses to those RAIs assigned to AREVA.

Upon receipt of the draft RAIs, Entergy and AREVA reviewed the RAIs and determined who would respond to each RAI. The responses for RAIs 1, 2, 3, 4, 5, 6 (b), 7, 8, 11, 12, and 13 were assigned to AREVA. The responses to RAIs 6 (a, c, d, and e), 9, 10, 14, 15, and 16 were assigned to Structural Integrity Associates (SI) by Entergy. For completeness, this document lists all sixteen RAIs; however, the responses for those RAIs that were assigned to SI only say "Assigned to SI by Entergy by the Division of Responsibility."

Information considered proprietary to AREVA is enclosed in the following brackets: **[ ]**

2.0 REQUESTS FOR ADDITIONAL INFORMATION

The NRC RAIs are reproduced in Sections 2.1.1 through 2.16.1. The AREVA/SI responses are in Sections 2.1.2 through 2.16.2.

2.1 EVIB-RAI-1

2.1.1 *Statement of EVIB-RAI-1*

Topical Report Condition 7 of the NRC staff's SE, on MRP-227-A (2) states: "MRP-227, Appendix A shall be updated to include...the Operating Experience Summary." Appendix A of MRP-227-A discussed operating experience under various degradation mechanisms for the RPV internal components of all nuclear steam supply system (NSSS) designs, including Babcock and Wilcox (B&W) plants.

(a) Identify the MRP-227-A, Appendix A operating experience that was contributed to by ANO, Unit 1.

(b) Provide any ANO1 plant-specific operating experience relevant to age-related degradation of RPV internal components that was not discussed in Appendix A of MRP-227-A.

2.1.2 *Response to EVIB-RAI-1*

2.1.2.1 *Response to Part (a)*

The following operating experience (OE) identified in Appendix A of MRP-227-A was contributed by ANO-1. The wording in *italic* text is taken from Appendix A of MRP-227-A and the text beneath it is an explanation of how the OE is attributed to ANO-1.

Multiple PWR internals bolt failures of the lower thermal shield bolts were discovered during the 1981 and 1982 in-service inspections performed at three B&W-design PWRs. The thermal shield bolt locking clips at these three plants were visually observed to be missing or loose.

Subsequent examinations during 1982, 1983, and 1984 revealed bolt failures at four additional units. These failures included upper core barrel, lower core barrel, upper thermal shield, and surveillance specimen holder tube bolts.

Table 2-1 summarizes the high-strength bolting inspections and replacements for ANO-1 that contribute to this OE description in MRP-227-A.

Table 2-1. High-Strength Bolting Inspection and Replacement Summary for ANO-1

	Originally Installed Bolt		Bolt Inspection and Replacement				
High Strength Bolt Location	Number of Bolts	Material & Condition	Inspection Date	Inspection Method	Number Inspected	Number Failed	Number Replaced
Upper Core Barrel (UCB)	120	Alloy A-286 Condition A	[				
Lower Core Barrel (LCB)	108	Alloy A-286 Condition A					
Upper Thermal Shield (UTS)	60	Alloy A-286 Condition A					
Lower Thermal Shield (LTS)	96	Alloy A-286 Condition A					
Flow Distributor (FD)	96	Alloy A-286 Condition A					]

Note 1 to Table 2-1: [

]

Problems were noted involving the original locking devices for the B&W-design vent valve jackscrews in the late 1970s and early 1980s. The jackscrew locking mechanism was vibrating and wearing through the locking cup. A new locking mechanism was designed and supplied to most B&W units. At least four of the eight vent valves were modified with the redesigned locking devices. The four vent valves next to the two outlet nozzles were replaced. In the late 1970s and early 1980s, problems were also noted involving the original jackscrew guide bushing, which was found to be improperly secured on some valves. Procedures were developed to install the modified locking device on the jackscrew and to secure the bushing when necessary.

The original locking devices for the four vent valves near the outlet nozzles at ANO-1 were replaced with the modified locking devices to correct the problems with vibration. As of February 2015, the four vent valves near the outlet nozzles have modified locking devices.

B&W-design Vent Valves *— Vent valve jackscrew locking cup damage has also been observed at some units, which was due to an interaction with the plenum assembly during insertion and removal activities. Vent valves are replaceable items and as noted above, have been replaced as necessary.*

Damage to the vent valve locking device parts due to apparent impact from the plenum, such as deformation and scratches, has been observed at ANO-1 as recently as the June 2013 refueling outage. Some of the damage was preexisting from a previous outage. Some damage has been deemed as acceptable for future use and some damage has required vent valve replacement.

2.1.2.2 *Response to Part (b)*

Operating experience (OE) for the RV internals at ANO-1 is discussed in Section 2.3.10.1 of the ANO-1 aging management program plan (1). Industry and ANO-1 specific information relevant to aging was compiled into the ANO-1 OE program. Industry OE sources in this program include applicable NRC Generic Publications (including Information Notices, Circulars, Bulletins and Generic Letters), NRC Generic Aging Lessons Learned (GALL) Report, etc. Plant specific OE sources in the database include applicable maintenance work history, licensee event reports (LERs), corrective action process documents, etc. Other than the bolting and vent valve issues discussed in the response to EVIB-RAI-1(a), the review of operating experiences did not identify any new aging issues related to the ANO-1 RV internals. Note that the bolting issues identified in the review of OE in Section 2.3.10.1 of Reference (1) are those relevant to the 1980s inspections included in Table 2-1. The 2008 inspections of the UCB bolts and LCB bolts are not included in Section 2.3.10.1 of Reference (1). The 2008 inspection of the UCB bolts showed [ ] bolts had failed and the 2008 inspection of the LCB bolts showed [ ] bolts had failed, as shown in Table 2-1.

In addition to the ANO-1 OE contained in MRP-227-A (see response to EVIB-RAI-1(a)), the following ANO-1 OE is not included in Appendix A of MRP-227-A. These OE instances are not due to aging (1):

- One of the ANO-1 control rod drive mechanisms was removed and the control rod guide assembly in the plenum was modified to accept the reactor vessel level monitoring probe.
- Portions of the surveillance specimen holder tubes (SSHT) at ANO-1 are attached to the internals. Although all the specimens have been removed, portions of the shroud tube and the supports that are bolted to the core support shield remain. These items only have the function of preventing loose parts in the reactor coolant system. Removal of the SSHT was due to a design flaw and was considered not to be caused by aging related degradation.

Additionally, in 2010, a fuel assembly was found damaged during 1R21 and Cycle 22 by interaction between the reactor vessel level monitoring probe and a burnable poison rod assembly. The reactor vessel level monitoring probe guide tube assembly was planned to be replaced during 1R23, but the replacement was aborted when an associated bolt could not be removed. An engineering review determined the guide tube assembly was acceptable for use as-is. A root cause evaluation[2] was performed for this event.

2.2 EVIB-RAI-2

2.2.1 *Statement of EVIB-RAI-2*

Section 2.3.10.1 of the ANO1 RPV internals AMP summarizes industry and plant-specific operating experience relevant to aging management of the RPV internals at ANO, Unit 1. Regarding plant-specific operating experience, the licensee states the following:

"Based on the discussions in BAW-2248A and the review of ANO1 operating data using the station information management system, condition reporting system, and licensee event database, cracking of the thermal shield bolting and core barrel bolting fabricated from Alloy A-286 was identified as an issue. These failures were attributed to intergranular stress corrosion cracking (IGSCC), and were not detected by visual examinations."

(a) Please discuss whether any other instances of age-related degradation have been identified for other RPV internals components at ANO, Unit 1.

(b) For the IGSCC of the thermal shield bolting and core barrel bolting fabricated from Alloy-286, please provide the details regarding the inspections which discovered the indications or failures. Specifically, state the examination method, the number of bolts covered in the inspection, and the calendar year when the inspections occurred.

(c) State the aging management program(s) under which these inspections were conducted (ASME Code, Section XI inservice inspection (ISI) Program, RPV Internals MRP-227-A initial primary components inspections, or other program), or whether the findings were just incidental to other outage/maintenance activities.

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[2] This event and subsequent corrective actions are addressed through Entergy's corrective action program.

(d) Please discuss any operability assessments that were performed for the cracked bolting, including the consideration of inspection results for the operability analysis, assumptions made for future bolt failures, or minimum bolting pattern analyses. Provide references for these operability reports.

(e) Please document any component repair/replacement activities for the cracked Alloy A-286 bolting, noting in particular any changes in bolting material specification and/or heat treatment for the repaired configuration.

(f) Please discuss any subsequent inspections that have been performed (after the discovery of cracking in the original Alloy A-286 thermal shield bolting and core barrel bolting) for the original and/or replacement core barrel and thermal shield bolting during the 40-year license term, including the examination method, frequency and coverage.

(g) Discuss the subsequent inspection results for the thermal shield and core barrel bolting, noting whether additional degradation was found for these bolts during the 40-year license term. Please discuss whether this additional operating experience supports the MRP-227-A-specified examination method, frequency and coverage for these thermal shield bolting and core barrel bolting components.

2.2.2 *Response to EVIB-RAI-2*

2.2.2.1 *Response to Part (a)*

The information requested in EVIB-RAI-2(a) can be found in the response to EVIB-RAI-1(a) and EVIB-RAI-1(b).

2.2.2.2 *Response to Part (b)*

The information requested in EVIB-RAI-2(b) can be found in the response to EVIB-RAI-1(a) and Table 2-1.

2.2.2.3 *Response to Part (c)*

In-service inspections using an ultrasonic testing (UT) technique at three Babcock & Wilcox (B&W) units showed that the majority of the lower thermal shield (LTS) bolts were failed; these examinations were performed in 1981 and 1982. A sampling of other high-strength bolts at these units was also examined using UT techniques. As a result of these findings, the B&W Owner's Group (B&WOG), now a part of the Pressurized Water Reactor Owner's Group, or PWROG, instituted a repair and inspection program for the LTS bolts along with inspections of other high-strength bolts at the B&W units. The ultrasonic examinations performed in 1983 and 1984 at ANO-1 on high-strength bolting, described in Table 2-1, were performed in response to these issues under guidance from this program (3, 9).

The examinations of the UCB and LCB bolts at ANO-1 in 2008 were performed in response to guidance from NEI 03-08 (4) "Needed" category recommendations from the B&WOG. These recommendations were considered during the development of MRP-227, Rev. 0 (5).

2.2.2.4 *Response to Part (d)*

As previously stated in the response to EVIB-RAI-1(a) and Table 2-1, during examinations performed in the 1980s, portions of the populations of the upper core barrel (UCB) bolts and LTS bolts were found to be failed. However, after discovery of these findings in the 1980s, these bolts were also replaced and have been reinspected since the bolting replacement with [ ] failed bolts identified. A report regarding the initial findings was prepared and submitted to the NRC (6); NRC review indicated that the information provided in the report is consistent with the bases upon which the NRC agreed with the licensees' positions and the actions by the NRC were considered complete (7). Therefore, a discussion of operability assessments for these findings is unnecessary for this response. However, operability assessments for continued operation after the 2008 LCB bolt examinations at ANO-1 are discussed below.

As previously stated in the response to EVIB-RAI-1(a) and Table 2-1, [ ] lower core barrel (LCB) bolts were examined using ultrasonic methods in the fall of 1984 with [ ] bolts failed. [ ] LCB bolts at ANO-1 were examined using UT techniques in the fall of 2008 with [ ] of the LCB bolts considered failed. No replacement of the LCB bolts has taken place at ANO-1 to date.

An assessment was performed to address the results of the fall 2008 UT of the LCB bolts at ANO-1 in early 2010. The following interdisciplinary technical assessments were performed for possible extension of the inspection interval for the LCB bolts at ANO-1:

- Statistical (Weibull analysis) assessment of a future LCB bolt stress corrosion cracking (SCC) failure rate based on the past UT results and bolt condition
- Structural analysis assessment of potential additional LCB bolt failures on the acceptability criteria
- Nondestructive examination (NDE) assessment of the LCB UT report and results
- Safety assessment of the consequences of LCB bolt failure on safe shutdown of the unit

Review of the EOC-21 LCB bolt NDE results concludes that [ ] However, it must be noted that to confirm this conclusion, [ ]

A conservative extrapolation predicts [ ] additional failed LCB bolts by EOC-24, for a total of [ ] failed LCB bolts. Only if the [ ] LCB bolts are in the worst possible configuration [ ] However, such an occurrence of a failed LCB bolt pattern is considered highly unlikely considering the [ ]

Ultrasonic reinspection of the LCB bolts was determined to be unnecessary for the next three refueling outages (EOC-22 in spring 2010, EOC-23 in fall 2011, and EOC-24 in spring 2013)[3]. Considering the assumptions used in the 2010 analyses and the limited recent UT inspection data that existed for LCB bolts in the operating B&W units as of 2010, it was recommended that a LCB UT reinspection be made at EOC-25 at ANO-1.

However, in early 2015, following a review of the most recent inspection results from other B&W units in 2011-2014, a second assessment of the ANO-1 LCB bolts was prepared. The purpose was to evaluate the possible extension of the currently recommended UT reexamination during the EOC-25 outage for one additional fuel cycle (i.e., to the EOC-26) at ANO-1. The future LCB bolt failure rate was assessed using a Weibull-based analysis, which showed that an estimated total number of failed bolts at ANO-1 EOC-26 could conservatively be taken as [ ]. The [ ] failed bolts consist of the [ ] failed bolts identified in 2008 [ ] This conservative estimate was then used as input for a [ ]

[3] Note that since this assessment was prepared in early 2010, the refueling outage schedule for ANO-1 has changed due to an extended period of downtime in early 2013. The EOC-24 outage at ANO-1 was completed in the summer of 2013.

To determine if the estimated likelihood is acceptably low, [

]

However, if all the conservatisms that are used in each of the assessments are considered, the likelihood of ANO-1 having [ ] during fuel cycle 26 is still considered acceptably low; [

]

Therefore, it is concluded that operation of ANO-1 to the EOC-26 outage before UT reexamination of the LCB bolts is performed is acceptable. As described in Table 5-1 of (1), the LCB bolts will be inspected using a UT examination technique in 1R26.

The information discussed in this response to part (d) regarding LCB bolts is based on two operability assessments that are available for review on site.

2.2.2.5 *Response to Part (e)*

As previously stated in the response to EVIB-RAI-1(a) and Table 2-1, LTS bolts and UCB bolts at ANO-1 have been replaced; the LTS bolts in 1983 and the UCB bolts in 1984.

The Alloy A-286 material used for the original LTS and UCB bolts at ANO-1 met the properties and requirements of the ASTM International (ASTM) A-453 condition A material, fabricated by hot forging (hot heading) the bolt head and cold rolling the threads. The only distinctive feature noted during a review of the operating plant bolt material test reports was that one material heat was heavily cold worked (40 to 50%) prior to the condition A treatment and hot heading. This material was used exclusively for the LTS bolts (8). The heat treatment for condition A used for the original LTS and UCB bolts is 1650°F (900°C) for 2 hours and the aging treatment is 1325°F (718°C) for 16 hours.

The [ ] replacement LTS bolts at ANO-1 are fabricated from Alloy A-286 Condition A material. [ ] Of the 120 UCB bolt locations at ANO-1, [ ] replacement Alloy A-286 UCB bolts were installed. The [ ] remaining UCB bolt locations were plugged. The [ ] replacement UCB bolts at ANO-1 are fabricated from Alloy A-286 Condition A material. [ ]

In addition, as summarized in Reference (8), the design and fabrication process for the replacement LTS bolts and UCB bolts were optimized to reduce susceptibility to stress corrosion cracking (e.g., elimination of the hot heading operation).

2.2.2.6 *Response to Part (f)*

Information requested in EVIB-RAI-2(f) can be found in the response to EVIB-RAI-1(a) and Table 2-1.

2.2.2.7 *Response to Part (g)*

As described in the response to EVIB-RAI-1(a) and Table 2-1, failures of portions of the UCB and LTS bolting were identified during inspections performed in 1983 and those UCB and LTS bolts were replaced in 1984 and 1983, respectively. Subsequent inspection of the replacement LTS bolts was performed in 1984 with [ ] identified. Subsequent inspection of the replacement UCB bolts was performed in 2008 with [ ] identified. It is also noted (see Table 2-1) that [ ] failures of the original LCB, UTS, or FD bolts were identified during inspections performed in 1984.

As described in Reference (9), a program (the internals bolting surveillance program [IBSP]) was established to evaluate SCC of the high-strength bolting used in these and other locations in the B&W-design RV internals. This program resulted in establishing, in 2005, a set of recommendations (method, frequency, and coverage) for UT inspections of the high-strength bolts in the B&W units, which ultimately were considered in the industry guidelines (MRP-227, Rev. 0). As noted in Table 2-1 and the response to EVIB-RAI-2(c), the UCB and LCB bolts were inspected in 2008 using UT techniques, which was in accordance with these recommendations.

The examination methods used previously for the high-strength (UCB, LCB, UTS, LTS, and FD) bolts at ANO-1 was UT examination, which is the same examination method specified in Table 4-1 and Table 4-4 of MRP-227-A for these bolts for the B&W units. Additionally, previous review of the EOC-21 LCB bolt NDE results at ANO-1 also concluded that [

] Therefore, UT examination is considered an appropriate examination method for the high-strength bolts at ANO-1.

The 2008 inspections resulted in [ ] failures being identified in the replacement UCB bolts and [ ] potential failures of the original LCB bolts at ANO-1 (see responses to EVIB-RAI-11 and EVIB-RAI-12 for further detail). [ ] two assessments were performed to address these results (see response to EVIB-RAI-2(d)). These assessments resulted in a recommendation to reexamine the LCB bolts during the ANO-1 EOC-26 outage, which is after approximately eight years (2008-2016) of operation and within the MRP-227-A recommended frequency (i.e., "subsequent examination(s) on the 10-year ISI interval..."). Therefore, the current operating experience and recommended reinspection period for the UCB and LCB bolts at ANO-1 supports the MRP-227-A specified frequency for high-strength bolts at ANO-1.

The examination coverage specified in Table 4-1 of MRP-227-A for the primary high-strength bolts and in Table 4-4 of MRP-227-A for the expansion high-strength bolts applicable to ANO-1 is "100% of accessible bolts [and their locking devices]" with a note that a minimum 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria (Table 5-1 of MRP-227-A) for the primary high-strength bolts, must be examined for inspection credit. As noted above, [ ] original LCB bolts were UT examined in [ ] Therefore, it is expected that the 108 LCB bolts, whether original or ever replaced, should be accessible for UT examination. Therefore, the examination coverage specified in Table 4-1 and Table 4-4 of MRP-227-A for the high-strength bolts is appropriate for ANO-1.

Therefore this additional OE supports the MRP-227-A-specified examination method, frequency, and coverage for the UTS, LTS, FD, UCB, and LCB bolts at ANO-1.

2.3 EVIB-RAI-3

2.3.1 *Statement of EVIB-RAI-3*

Historically, the following materials used in the pressurized water reactor (PWR) RPV internal components were known to be susceptible to some of the aging degradation mechanisms that are identified in MRP-227-A. In this context, the NRC staff requests that the licensee confirm that these materials are not currently used in the RVI components at ANO, Unit 1.

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

(2) Stainless steel type 347 material (excluding baffle-former bolts);

(3) Precipitation hardened (PH) stainless steel materials - 17-4 and 15-5;

(4) Type 431 stainless steel materials.

If one or more of these materials were used in the RVI components at ANO, Unit 1, provide information regarding their proposed inspections and basis to demonstrate that the proposed inspections are consistent with the intent of MRP-227-A.

2.3.2 *Response to EVIB-RAI-3*

A response to this request for additional information will be provided by December 31, 2015.

2.4 EVIB-RAI-4

2.4.1 *Statement of EVIB-RAI-4*

Identify all time-limited aging analyses TLAAs of the RPV internal components that were evaluated for the ANO, Unit 1, License Renewal pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 54.21(c)(1). This would include, for example, cumulative fatigue usage analyses of the internals based on transient cycle count inputs, environmental fatigue, and other plant-specific TLAAs, such as loss of fracture toughness due to neutron irradiation embrittlement. Please indicate whether the MRP-227-A-specified inspection provisions (examination method, frequency, and coverage) for these RPV internal components will coexist with the management of the TLAAs, or if the TLAAs for these components would result in any deviation from the MRP-227-A inspection guidelines. Provide justification if deviation from MRP-227-A is identified for any of the RPV components.

2.4.2 *Response to EVIB-RAI-4*

The TLAA that are applicable to ANO-1 are summarized in Table 4.1-1, Page 4-3, of the ANO-1 License Renewal Application (LRA), Reference (10), and are summarized by the NRC in Section 4.1 of NUREG-1743 Reference (11), including those of the reactor pressure vessel (RPV) internals items. TLAA of the ANO-1 RPV internals that were evaluated pursuant to 10 CFR 54.21(c)(1) reported in References (10) and (11) are based on BAW-2248-A, Reference (12), and are as follows:

- Metal fatigue
- Analytical evaluation of flaws
- Flow-induced vibration (FIV) (BAW-10051)
- Stress and Deflection Analyses (BAW-10008 / neutron embrittlement)

Metal fatigue is addressed, for all components including the RPV internals, in Section 4.3 of the ANO-1 LRA (10) and in (12), Section 4.5.1. As described in Section 4.5.1 of BAW-2248A (12) the reactor vessel internals were designed and constructed prior to the development of ASME Code requirements for core support structures. As such, existing industry structural practice was used in the design of the internals structural members and the only specific fatigue analyses performed in the original design were those that addressed high cycle fatigue as described in BAW-10051. However, in modifications following original design, plant-specific fatigue analyses were performed for the reactor vessel internals replacement bolts as presented in BAW-1843PA and BAW-1789P. These topical reports summarize fatigue analyses performed to the ASME code (Section III, Subsection NG) including both high cycle fatigue (FIV) and low cycle fatigue (design transients) for replacement bolting.

The fatigue evaluation of replacement bolting that covers the period of extended operation for ANO-1 is addressed in Section 4.5.1 of BAW-2248A, Page 4-11. BAW-2248A, Section 4.5.1 includes the following:

- The total usage factor was recalculated for the replacement bolting items by considering both the existing design transient usage factors shown acceptable for 60 years, and by the recalculation of fatigue usage factors due to flow-induced vibration. Therefore, the total usage factor for the replacement bolting items are shown acceptable for the period of extended operation in accordance with a combination of 10 CFR 54.21(c)(1)(i) and (ii).

- Environmentally assisted fatigue is addressed in Section 4.3.4.4 of the ANO-1 LRA (10). The environmentally assisted fatigue evaluation for ANO-1 was based on the limiting NUREG 6260 locations, which include RCS pressure boundary items only. Therefore, there are no RV internals items that are within the scope of the environmentally assisted fatigue assessment for ANO-1 as confirmed by the NRC SE (11), Section 4.3.2 for the ANO-1 LRA.

Internals analytical evaluation of flaws is addressed in Section 4.3 of the ANO-1 LRA (10). Reference 10, Table 2.3-5, reports that no flaws requiring analytical evaluation have been discovered in the inspections of the reactor vessel internals at ANO-1. As such, no aging management program is required or credited for flaw evaluation of ANO-1 RPV internals.

BAW-10051 (FIV Analysis) is addressed in Section 4.8.2 of the ANO-1 LRA (10). Report BAW-10051, "Flow Induced Vibration Endurance Limit Assumptions" is an analysis that calculated stress values for the reactor vessel incore nozzles and compared them to endurance limit (stress) values. These endurance limit values were based on an assumed value of 10^{12} cycles for 40 years of operation. In support of the ANO-1 LRA, the number of fatigue cycles was extended for 60 years, and the component item stress values were compared to the recalculated endurance limit values and shown to be acceptable. Therefore, this time limited aging analysis is acceptable per 10 CFR 54.21(c)(1)(ii).

BAW-10008 (Stress and Deflection Analyses) neutron embrittlement, ductility TLAA is addressed as follows:

- BAW-10008 is addressed in Table 2.3-5 of the ANO-1 LRA (10), Item 12.

- The ANO-1 commitment to address this TLAA is "A plant-specific analysis will be performed to demonstrate that under LOCA and seismic loading, the internals have adequate ductility to absorb local strain at the regions of maximum stress intensity and that irradiation accumulated at the expiration of the renewal license will not affect deformation limits. Data will be developed to demonstrate that the internals will meet the deformation limits at the expiration of the renewed license.
- The ANO-1 plant-specific reactor vessel internals aging management program plan (1), Section D.2, describes the evaluations which address this TLAA for ANO-1 and indicates the results of these evaluations have been submitted to the NRC under letter 1CAN051401, and that submittal of this letter fulfills the Item 12 action.
- This plant-specific analysis is contained in ANP-3281P (13) which is Attachment 1 of 1CAN051401 and is subject to a separate NRC review and SE.

All TLAA for the ANO-1 RV internals have been adequately addressed by Entergy in the ANO-1 LRA and the ANO-1 reactor vessel internals aging management program plan, which addresses the TLAA for RPV internals component ductility in Section D.2. As such, MRP-227-A-specified inspection provisions (i.e., examination method, frequency, and coverage) are not credited for management of RPV internal items TLAA in accordance with 10 CFR 54.21(c)(1)(iii) at ANO-1 and will not result in any deviation from the MRP-227-A inspection guidelines.

2.5 EVIB-RAI-5

2.5.1 *Statement of EVIB-RAI-5*

In the response to Applicant/Licensee Action Item 2 from the NRC staff's SE for MRP-227-A, Section 5.2 of the ANO1 RPV internals AMP states that the ANO, Unit 1, RPV internals have three additional components that were not listed in BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," March 2000 (ADAMS Accession No. ML003708443), but were included in the ANO, Unit 1 ASME Code, Section XI ISI Program. These components, which are categorized as "ANO1 Orphan Components" in AMP Table 5-6, "ANO1 Program Enhancement and Implementation Schedule," are:

- RPV level monitoring system (RVLMS) probe supports
- Remaining portions of the surveillance specimen holder tubes (SSHT)

- Thermal shield and thermal shield upper restraint.

Confirm whether these are the only RPV internal components that are applicable to your response to Applicant/Licensee Action Item 2 of the SE for MRP-227-A, specifically in regard to whether Tables 4-1 and 4-2 in MRP-189, Rev. 1 (14) have missed any RPV internal components that should be within the scope of LR in accordance with 10 CFR 54.4.

2.5.2 *Response to EVIB-RAI-5*

A response to this request for additional information will be provided by December 31, 2015.

2.6 EVIB-RAI-6

2.6.1 *Statement of EVIB-RAI-6*

Section 5.2 of the ANO1 RPV internals AMP states that the above orphan components will undergo a future screening and characterization, and based on the screening results, they will be removed from future inspections if they screen out, or added to the primary or expansion categories if they screen in. Further, it is stated that, until that time when the above components are screened and characterized, these components will be inspected during the 10-year ISI intervals based on their aging effects.

(a) Please identify the calendar year when these orphan components will be screened and characterized.

(b) Please provide additional details regarding the screening and characterization process for the above orphan components, including whether this process will be consistent with that performed for MRP-189, Rev. 1.

Section 5.2 of the ANO1 RPV internals AMP lists the aging effects for the orphan components and states that, based on a review of these aging affects, the orphan components will be visually inspected (VT-3) during the 10-year ISI inspections.

(c) Please state whether previous VT-3 or other inspections have been performed for these components, and discuss the results of these inspections if any. Based on the aging effects for the components described in Section 5.2, please justify the adequacy of performing VT-3 visual examinations on a 10-year ISI interval.

(d) Please state whether the VT-3 examinations are intended as an interim measure for aging management until the screening and characterization is complete.

(e) Since these components are currently designated to receive a VT-3 examination every 10-year ISI interval and were included in the ANO, Unit 1 ASME Code, Section XI ISI Program, please explain why they are not currently included in Table 5-3, "B&W Plants Existing Program Components from AREVA Guidance," of the RPV internals AMP.

2.6.2 *Response to EVIB-RAI-6*

2.6.2.1 *Response to Part (a)*

Assigned to SI by Entergy by the Division of Responsibility.

2.6.2.2 *Response to Part (b)*

The screening and initial steps of the categorization process will be consistent with the work performed in MRP-189, Revision 1 (14). The following sequence steps will be used for screening and categorization that are included in MRP-189, Revision 1:

1. Identify PWR internals components, materials, and environments
2. Characterize components and screen for degradation (A, non-A) using screening criteria from MRP-175 (15)
3. Failure, Modes, Effects, and Criticality Analysis (FMECA) review
4. Severity categorization (A, B, C)

The final two steps of the categorization process will be consistent with the work performed in MRP-231 (16):

5. Engineering evaluation and assessment
6. Categorize for inspection (primary, expansion, existing, no additional measures) and aging management strategy

2.6.2.3 *Response to Part (c)*

Assigned to SI by Entergy by the Division of Responsibility.

2.6.2.4 *Response to Part (d)*

Assigned to SI by Entergy by the Division of Responsibility.

2.6.2.5 *Response to Part (e)*

Assigned to SI by Entergy by the Division of Responsibility.

2.7 EVIB-RAI-7

2.7.1 *Statement of EVIB-RAI-7*

Table 5-2, "B&W Plants Expansion Category Components from Table 4-4 of MRP-227-A," of the ANO1 RPV internals AMP currently lists the surveillance specimen holder tube (SSHT) studs/nuts or bolts and their locking devices, under Core Barrel Assembly, as "not applicable to ANO1." Bolt or stud/nut cracking is listed as the aging effect for the SSHTs in this table and in Table 4-4 of MRP-227-A. However, Section 5.2 of the AMP states that for the remaining portions of the SSHTs, cracking and stress relaxation were identified as aging effects for the SSHT bolting. Please discuss whether this discrepancy should be resolved through a modification to Table 5-2.

2.7.2 *Response to EVIB-RAI-7*

A response to this request for additional information will be provided by December 31, 2015.

2.8 EVIB-RAI-8

2.8.1 *Statement of EVIB-RAI-8*

Section 5.2 (last paragraph) of the ANO1 RPV internals AMP states that:

"In addition to these components [the orphan components], AREVA has also advised B&W utilities that the vent valve locking devices for the original and modified designs be included as an existing program under MRP-227-A based on recent OE from ONS-1. Based on this recommendation, Table 5-1 [B&W Plants Primary Components from Table 4-1 of MRP-227-A] was modified and Table 5-3 was added to the list of the existing programs under the current [ASME Code] Section XI [ISI] program."

Please identify the modification to Table 5-1 of the AMP which addresses the vent valve locking devices, or confirm whether this modification refers to the revision and relocation of "Note 1" from Table 5-1 to Table 5-3 of the AMP.

2.8.2 *Response to EVIB-RAI-8*

AREVA interim guidance provided to the B&W utilities in May 2013 included the recommendation to remove Note 1 of Table 4-1 of MRP-227-A (regarding verification of operation of the vent valves), which is Table 5-1 of Reference (1), and the addition of Note 1 (regarding verification of operation of the vent valves) to a new Existing Programs table for utility license renewal implementation, which is Table 5-3 of Reference (1). These table revisions reflect the modifications recommended in the AREVA interim guidance for the Primary and Existing Program tables.

2.9 EVIB-RAI-9

2.9.1 *Statement of EVIB-RAI-9*

Table 5-3 of the ANO1 RPV internals AMP provided the B&W existing program inspection criteria for the core support shield vent valve miscellaneous locking devices for the original and modified design – VT-3 visual examination of the locking devices on the 10-year ISI interval per the requirements of the ASME Code, Section XI. Please state whether these VT-3 examinations were performed as part of the ASME Code, Section XI, 10-year interval inservice inspections during the original 40-year licensed operating term. Please identify and discuss any relevant indications for the ASME Code, Section VT-3 examinations of these items.

2.9.2 *Response to EVIB-RAI-9*

Assigned to SI by Entergy by the Division of Responsibility.

2.10 EVIB-RAI-10

2.10.1 *Statement of EVIB-RAI-10*

Table 5-3 of the ANO1 RPV internals AMP, "Note 1" describes additional vent valve testing and examinations for leakage and degradation in the valve components, which are to be performed in accordance with the plant's TS or the inservice testing programs. Please identify the applicable TS requirements and/or inservice testing program requirements for the additional testing and examination of the vent valve components.

2.10.2 *Response to EVIB-RAI-10*

Assigned to SI by Entergy by the Division of Responsibility.

2.11 EVIB-RAI-11

2.11.1 *Statement of EVIB-RAI-11*

For the core support shield assembly, upper core barrel bolts and their locking devices, Table 4-1 of MRP-227-A and Table 5-1 of the ANO1 AMP specify that the initial volumetric examination (UT) of the bolts must be performed within two refueling outages from January 1, 2006 or the next 10-year ISI interval, whichever is first. Table 5-1 of the ANO1 AMP states that UT examination of the upper core barrel bolts is to be performed during Refueling Outage 26 (1R26).

a) Please clarify whether the UT examination of the upper core barrel bolts scheduled for 1R26 is an initial examination or a subsequent examination.

b) If the initial UT examination of the upper core barrel bolts has already been performed, then please discuss the examination results for these items.

c) If the initial UT examination of the upper core barrel bolts has not yet been performed, then please justify the performance of this initial examination in 1R26 (which would occur several years after the start of period of extended operation on May 20, 2014) given that MRP-227-A calls for performance of this examination within two refueling outages from January 1, 2006 or the next 10-year ISI interval, whichever is first.

2.11.2 *Response to EVIB-RAI-11*

2.11.2.1 *Response to Part (a)*

The UT examination of the UCB bolts scheduled for 1R26 (1) is a subsequent examination.

2.11.2.2 *Response to Part (b)*

The UCB bolt examinations conducted during the Fall 2008 refueling outage constitute the initial inspections as required by MRP-227-A within two refueling outages of January 1, 2006. MRP-227-A (2) and MRP-228 (17) were not completed in time to be fully implemented at ANO-1 prior to this inspection. The examination technique used in 2008 for the UCB and LCB bolts was demonstrated to EPRI on June 10, 2008 at the WesDyne Waltz Mill, Pennsylvania facility. The examination technique demonstrated the capability to detect degradation greater than 25% cross sectional area with no false or indeterminent calls. The original Alloy A-286 UCB bolts at ANO-1 were replaced in 1984 with [ ] replacement bolts of an improved design and fabrication process. The 2008 UT examination revealed [ ] rejectable indications in the [ ] replacement UCB bolts at ANO-1 (see Table 2-1 in response to EVIB-RAI-1(a)).

2.11.2.3 *Response to Part (c)*

No response is required to part (c), based on the response to part (b).

2.12 EVIB-RAI-12

2.12.1 *Statement of EVIB-RAI-12*

Table 4-1 of MRP-227-A and Table 5-1 of the ANO1 RPV internals AMP list the following initial inspection requirements for the B&W primary components, which may require completion prior to entering the period of extended operation:

- Core Barrel Assembly, Lower core barrel bolts and their locking devices – Volumetric (UT) of the bolts during the next 10-year ISI from 1/1/2006.
- Flow Distributor Assembly, Flow distributor bolts and their locking devices – Volumetric examination (UT) of the bolts during the next 10-year ISI from 1/1/2006.

a) Please state whether these initial RPV internal component examinations have been completed prior to the end of the current 40-year license per the requirements for primary components in Table 4-1 of MRP-227-A. Briefly discuss the results of these initial examinations if applicable.

b) If these examinations have not yet been completed please explain how performance of these initial examinations during 1R26 satisfies the examination criteria for B&W primary components in Table 4-1 of MRP-227-A.

2.12.2 *Response to EVIB-RAI-12*

2.12.2.1 *Response to Part (a)*

The UT examination of the LCB bolts scheduled for 1R26 (1) is a subsequent examination.

The LCB bolt UT examinations conducted during the fall 2008 refueling outage constitute the initial inspections as required by MRP-227-A during the next 10-year ISI interval from 1/1/2006. MRP-227-A and MRP-228 were not completed in time to be fully implemented at ANO-1 prior to this inspection. The examination technique used in 2008 for the UCB and LCB bolts was demonstrated to EPRI on June 10, 2008 at the WesDyne Waltz Mill, Pennsylvania facility. The examination technique demonstrated the capability to detect degradation greater than 25% cross sectional area with no false or indeterminent calls. Rejectable UT indications were identified in [ ] of the 108 original Alloy A-286 LCB bolts at ANO-1 (see Table 2-1 in response to EVIB-RAI-1(a)).

The FD bolts have not yet been examined per the requirements for primary components in Table 4-1 of MRP-227-A.

2.12.2.2 *Response to Part (b)*

The UT examination of the FD bolts scheduled for 1R26 (1) in Fall 2016 is an initial examination and is required to be performed during the next 10-year ISI interval from 1/1/2006. The next 10-year ISI interval from 1/1/2006 began on May 31, 2008 and includes refueling outage 1R26 in Fall 2016. Therefore the UT examination of the FD bolts scheduled for 1R26 in Fall 2016 is during the next 10-year ISI interval from 1/1/2006 and meets the requirements of Table 4-1 of MRP-227-A.

The FD bolts were an expansion item in MRP-227, Revision 0 (5), which was released in December 2008. The requirement to inspect the FD bolts during the next 10 year ISI interval from 1/1/2006 was imposed in 2011 as part of the NRC's review and approval of MRP-227-A. Therefore, the FD bolts were not inspected in 2008 along with the inspections of the UCB and LCB bolts due to the timing of the publication of the requirement to inspect the FD bolts.

2.13 EVIB-RAI-13

2.13.1 *Statement of EVIB-RAI-13*

Applicant/Licensee Action Item 5 of the NRC staff SE for MRP-227-A requires the licensee to include, as part of their AMP submittal, an explanation of how the proposed acceptance criteria for the physical measurements are consistent with the plants' licensing basis and the need to *maintain the functionality of the component under all* licensing basis conditions of operation during the period of extended operation.

In the response to Applicant/Licensee Action Item 5, Section 5.5 of the ANO1 RPV internals AMP states that the acceptance criteria for this one time physical measurement shall be an average measured differential height from the top of the plenum rib pads to the RPV seating surface of 0.004 inches relative to the as-built condition in accordance with Table 5-4 of the AMP.

Please elaborate on how the acceptance criterion of 0.004 inches for the average measured differential height from the top of the plenum rib pads to the RPV seating surface, relative to the as-built condition, is consistent with the ANO, Unit 1, licensing basis and the need to maintain the functionality of the component under all licensing basis conditions of operation during the period of extended operation.

2.13.2 *Response to EVIB-RAI-13*

Background information regarding the acceptance criteria of 0.004 inches for core clamping items at the B&W units can be found in the response to RAI 5 (18) in the first set of RAIs (19) to WCAP-17096 (20). Applicant/Licensee Action Item 5 is only applicable to Westinghouse and Combustion Engineering nuclear steam supply systems (NSSSs). ANO-1 is a B&W NSSS.

2.14 EVIB-RAI-14

2.14.1 *Statement of EVIB-RAI-14*

Applicant/Licensee Action Item 6 of the NRC staff SE for MRP-227-A requires the licensee to justify the acceptability of each of the inaccessible B&W expansion components for continued operation by performing an evaluation, or propose a schedule for replacement of the components. The inaccessible B&W components are:

- the core barrel cylinder and welds,
- the former plates, and
- the bolting (core barrel-to-former bolts, internal and external baffle-to-baffle bolts, and associated locking devices),

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 one 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.

As stated in Section 4.2.6 of the MRP-227-A NRC staff SE, the justification for the continued operability of the above inaccessible components for the period of extended operation and, if necessary, schedule for replacement of these components must be provided for NRC review and approval as part of the licensee's application to implement MRP-227-A.

a) Therefore, in order to complete its evaluation of the ANO1 RPV internals AMP, the NRC staff requests that the licensee provide the information required by Action Item 6 of the MRP-227-A NRC staff SE. Specifically, the staff requests that the licensee justify the acceptability of the inaccessible components for continued operation through the period of extended operation by performing an evaluation and, if necessary, provide a schedule for replacement of the components for staff review and approval.

b) If the licensee cannot justify the acceptability of the inaccessible components for continued operation through the period of extended operation by performing an evaluation and, if necessary, providing a schedule for replacement of the components, for staff review and approval as part its current application, the staff requests that the licensee propose an alternative process for ensuring the operability of the inaccessible components during the period of extended operation.

2.14.2 *Response to EVIB-RAI-14*

2.14.2.1 *Response to Part (a)*

Assigned to SI by Entergy by the Division of Responsibility.

2.14.2.2 *Response to Part (b)*

Assigned to SI by Entergy by the Division of Responsibility.

2.15 EVIB-RAI-15

2.15.1 *Statement of EVIB-RAI-15*

Applicant/Licensee Action Item 7 of the NRC staff SE for MRP-227-A requires the licensee to develop plant-specific analyses to be applied for their facilities to demonstrate that RPV internals components that may be fabricated from cast austenitic stainless steel (CASS), martensitic stainless steel or precipitation hardened stainless steel, will maintain their functionality during the period of extended operation, considering possible loss of fracture toughness in these components due to thermal and irradiation embrittlement and limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The action item states that the licensee shall include the plant-specific analysis as part of their submittal to apply MRP-227-A.

In the response to Applicant/Licensee Action Item 7, Section 5.7 of the ANO1 RPV internals AMP provides information indicating that future analytical evaluations will be performed for assessing the effects of reduction in fracture toughness, due to thermal and irradiation embrittlement, on the CASS and precipitation hardened stainless steel RPV internals components at ANO, Unit 1. The ANO, Unit 1, CASS and precipitation hardened stainless steel components requiring these analytical evaluations for demonstrating functionality during the period of extended operation are:

- CASS Components:
  - Control Rod Guide Tube Assembly Spacer Castings
  - Core Support Shield Assembly Vent Valve Bodies
  - Incore Monitoring Instrumentation Guide Tube Assembly Spider Castings
- Precipitation Hardened Stainless Steel Components:
  - Core Support Shield Assembly Vent Valve Retaining Rings.

The licensee provided regulatory commitments to complete the analytical evaluations of these components 12 months prior to the second refueling outage after entering the period of extended operation.

As stated in Section 4.2.7 of the MRP-227-A NRC Staff SE, the plant-specific analysis of these components required by this action item shall be included as part of licensees' submittals to implement MRP-227-A.

a) Therefore, in order to complete its evaluation of the ANO1 RPV internals AMP, the NRC staff requests that the licensee provide the information required by Action Item 7 of the MRP-227-A NRC staff SE. Specifically, the staff requests that the licensee provide plant-specific analyses to demonstrate that the above CASS and precipitation hardened stainless steel RPV internal components will maintain their functionality during the period of extended operation, considering possible loss of fracture toughness in these components due to thermal and irradiation embrittlement and limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques.

b) If the licensee cannot provide the plant-specific analysis of the CASS and precipitation hardened stainless steel RPV internal components required by Action Item 7 for staff review and approval as part its current application, the staff requests that the licensee propose an alternative process for ensuring that the functionality of these components will be maintained during the period of extended operation, considering possible loss of fracture toughness in these components due to thermal and irradiation embrittlement and limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques.

2.15.2 *Response to EVIB-RAI-15*

2.15.2.1 *Response to Part (a)*

Assigned to SI by Entergy by the Division of Responsibility.

2.15.2.2 *Response to Part (b)*

Assigned to SI by Entergy by the Division of Responsibility.

2.16 EVIB-RAI-16

2.16.1 *Statement of EVIB-RAI-16*

In order for the NRC staff to verify the adequacy of the program implementation schedule provided in Table 5-6 of the ANO1 RPV internals AMP, please confirm that the length of the ANO, Unit 1, refueling cycle corresponds to 18 months. In addition please specify the projected calendar years corresponding to ANO, Unit 1, refueling outages 1R26 and 1R33.

2.16.2 *Response to EVIB-RAI-16*

Assigned to SI by Entergy by the Division of Responsibility.

3.0 REFERENCES

1. Entergy Operations, Inc. Letter, 1CAN051403 to the NRC dated May 20, 2014 (ADAMS Accession Number ML14141A554), Attachment 1, "PWR Internals Aging Management Program Plan for Arkansas Nuclear One Unit 1," Structural Integrity Report Number 1200459.401, Revision 1, May 2014 (ADAMS Accession Number ML14141A555).
2. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A). EPRI, Palo Alto, CA: 2011. 1022863.
3. Hall, J. B., Fyfitch, S., and Moore, K. E., "Laboratory and Operating Experience with Alloy A-286 and Alloy X-750 RV Internals Bolting Stress Corrosion Cracking," Proceedings of the Eleventh International Conference on Environmental Degradation of Materials in Nuclear Power Systems – Water Reactors, ANS, 2003.
4. Nuclear Energy Institute, "Guideline for the Management of Materials Issues," NEI 03-08, Revision 0, May 2003. NRC Accession Number ML032190048.
5. Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0). EPRI, Palo Alto, CA: 2008. 1016596.
6. Forward of Proprietary BAW-1843P and BAW-1842 "Evaluation of Internals Bolting Concerns in 177FA Plants," August 17, 1984, NRC Accession Number 8409050338.
7. NRC letter from Gus C. Lainas to Mr. H. B. Tucker, Babcock & Wilcox Owners Group, Forwards Comments on BAW-1843P, "B&W Owner's Group Evaluation of Internal Bolting Concerns in 177FA Plants," June 11, 1985, NRC Accession Number 8506200042.
8. Piascik, R. S. and Moore, K. E., "Behavior of Alloy A-286 Reactor Vessel Internals Bolting Material," Nuclear Technology, Volume 75(3), pp 370-377, December 1986.
9. Davidsaver, S. B., Fyfitch, S., and Xu, H., "Developing PWR Aging-Management Strategies for Reactor Vessel Internals," Proceedings of the Fifteenth International Conference on Environmental Degradation, TMS, 2011.
10. Arkansas Nuclear One-Unit 1, License Renewal Application, NRC Accession Number ML003679667, January 31, 2000
11. Safety Evaluation Report Related to the License Renewal of Arkansas Nuclear One, Unit 1 (NUREG-1743), NRC Accession Number ML011640099, ML011640177, ML011640217, May 31, 2001.
12. BAW-2248A (43-2248A), Demonstration of the Management of Aging Effects for the Reactor Vessel Internals, NRC Accession Number ML003708443, April 21, 2000.
13. Letter from Stephenie L. Pyle, Entergy to NRC, Entergy Document 1CAN051401, "Time-Limited Aging Analysis Regarding Reactor Vessel Internals Loss of Ductility for Arkansas Nuclear One, Unit 1 at 60 Years," NRC Accession Number ML14126A816, May 6, 2014.
14. Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals Component Items (MRP-189-Rev. 1). EPRI, Palo Alto, CA: 2009. 1018292.

15. Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175). EPRI, Palo Alto, CA: 2005. 1012081.

16. Materials Reliability Program: Aging Management Strategies for B&W Pressurized Water Reactor Internals (MRP-231-Rev. 2). EPRI, Palo Alto, CA: 2010. 1021028.

17. Non-Proprietary Version of Materials Reliability Program: Inspection Standard for PWR Internals (MRP-228). EPRI, Palo Alto, CA: 2009. 1016609.

18. EPRI, Revised Responses to the NRC Request for Additional Information on WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Date Requirements" (TAC No. ME4200) PA-MSC-0473, June 14, 2012, ADAMS Accession Number ML12171A374.

19. NRC letter from Sheldon D. Stuchell to Mr. Neil Wilmshurst, EPRI, "Request for Additional Information on WCAP-17096-NP, Revision 2, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements' (TAC No. ME4200)," May 19, 2011, NRC Accession Number ML111160066.

20. Letter from Terry McAlister to NRC (MRP 2010-034), "Report Transmittal, Westinghouse Non-Proprietary Class 3 Report, 'Reactor Internals Acceptance Criteria Methodology and Data Requirements, WCAP-17096-NP, Revision 2'," December 2009,"May 19, 2010, NRC Accession Number ML101460156.

Attachment 4 to

1CAN021503

Affidavit

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle Elliott. I am Manager, Product Licensing, for AREVA Inc. (AREVA) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA to determine whether certain AREVA information is proprietary. I am familiar with the policies established by AREVA to ensure the proper application of these criteria.

3. I am familiar with the AREVA information contained in ANP-3383P, Revision 0, entitled, "Response to Request for Additional Information for the Reactor Pressure Vessel Internals Aging Management Program Plan for Arkansas Nuclear One Unit 1," dated February 2015 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA Inc. for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA, would be helpful to competitors to AREVA, and would likely cause substantial harm to the competitive position of AREVA.

The information in this Document is considered proprietary for the reasons set forth in paragraphs 6(c), 6(d) and 6(e) above.

7. In accordance with AREVA's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside AREVA only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 9th

day of February, 2015.

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/18
Reg. # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2018

Attachment 5 to

1CAN021503

List of Commitments

List of Commitments

This table identifies actions discussed in this letter for which Entergy commits to perform. Any other actions discussed in this submittal are described for the NRC's information and are not commitments.

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Entergy will provide the response to RAI 3.	✓		December 31, 2015
Entergy will provide the response to RAI 5.	✓		December 31, 2015
Entergy will complete the screening and categorization described in the response to RAI 6(a).	✓		December 31, 2015
Entergy will provide the response to RAI 7.	✓		December 31, 2015
Entergy will provide the evaluations described in the response to RAI 14(a).	✓		December 31, 2016
Entergy will provide the evaluations described in the response to RAI 15(a).	✓		September 30, 2015