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Stephenie L. Pyle
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1CAN021503

February 10, 2015

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: Responses to Request for Additional Information
Reactor Vessel Internals Aging Management Program Plan
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

REFERENCES:

1. Entergy letter to NRC, "Reactor Vessel Internals Aging Management Program Plan," dated May 20, 2014 (1CAN051403) (ML14141A554)

2. NRC email to Entergy, dated December 2, 2014, "Requests for Additional Information – Reactor Vessel Internals Aging Management Plan – ANO-1 – TAC No. MF4201"

Dear Sir or Madam:

Entergy Operations, Inc. (Entergy) submitted the Arkansas Nuclear One, Unit 1 (ANO-1) Reactor Vessel Internals Aging Management Program Plan (Reference 1) to fulfill a commitment made as part of the ANO-1 License Renewal Application. The plan identified the internals components that must be included for aging management review and identifies the augmented inspection plan for the ANO-1 reactor vessel internals.

The NRC Staff has reviewed the submittal and developed a request for additional information (RAI). This request was provided via Reference 2. The purpose of this submittal is to provide the requested information.

The responses to the RAIs were provided by Structural Integrity Associates, Inc. (SI) (author of the ANO-1 Aging Management Program Plan) and AREVA (manufacturer of the ANO-1 reactor vessel internals).

Attachment 2 to this letter contains proprietary information – Attachment 2 is withheld from public disclosure per 10 CFR 2.390.

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SI developed the responses to RAIs 6(a, c, d, and e), 9, 10, 14, 15, and 16. These responses are provided in Attachment 1 to this submittal.

AREVA document ANP-3383P, Revision 0, "Response to Request for Additional Information for the Reactor Pressure Vessel Internals Aging Management Program Plan for Arkansas Nuclear One Unit 1," has been prepared with the responses to the remaining requests. The information contained in the AREVA document is considered proprietary to AREVA. Attachment 2 is the proprietary version of the document. A non-proprietary version of the AREVA document is included in Attachment 3. AREVA requests that the proprietary information be withheld from public disclosure in accordance with 10 CFR 2.390. AREVA has provided Entergy with authorization to provide the proprietary information. An affidavit by the information owner, AREVA, supporting the request for non-disclosure is provided in Attachment 4. Therefore, Entergy requests that Attachment 2 of this submittal be withheld from public disclosure in accordance with 10 CFR 2.390. It should be noted that the information contained in Attachment 1 is not considered proprietary.

Several new regulatory commitments have been identified in this letter. These commitments are listed in Attachment 5.

If you have any questions or require additional information, please contact me.

Sincerely,

SLP/rwc

Attachments

1. Structural Integrity Associates, Inc. document 1401460.402.R0, "Responses to Requests for Additional Information on the Reactor Vessel Internals Aging Management Program for Arkansas Nuclear One, Unit 1"
2. AREVA document ANP-3383P, "Response to Request for Additional Information for the Reactor Pressure Vessel Internals Aging Management Program Plan for Arkansas Nuclear One Unit 1," PROPRIETARY
3. 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
4. Affidavit
5. List of Commitments

Attachment 2 to this letter contains proprietary information – Attachment 2 is withheld from public disclosure per 10 CFR 2.390.

cc: Mr. Marc L. Dapas
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Attachment 1

1CAN021503

Structural Integrity Associates, Inc. document 1401460.402.R0

Responses to Requests for Additional Information on the Reactor Vessel Internals Aging Management Program for Arkansas Nuclear One, Unit 1

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February 5, 2015
Report No. 1401460.402.R0
Quality Program: ☒ Nuclear ☐ Commercial

Christopher Walker
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Subject: Responses to Requests for Additional Information on the Reactor Vessel Internals Aging Management Program for Arkansas Nuclear One, Unit 1

Dear Chris:

Here are the responses to the NRC requests for additional information (RAIs) on the reactor vessel internals (RVI) aging management program (AMP) for RAIs 6 (a, c, d, e), 9, 10, 14, 15, and 16 for Arkansas Nuclear One, Unit 1 (ANO-1).

1.0 INTRODUCTION

SI previously developed the RVI AMP for ANO-1 in May of 2014, which was submitted to the NRC as part of ANO-1's license renewal commitment. SI has provided responses to RAIs 6 (a, c, d, e), 9, 10, 14, 15, and 16 which are contained in the following sections.

2.0 RAI RESPONSES

2.1 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.

Responses to EVIB-RAI-6

6(a)

The screening and categorization is expected to be completed in 2015.

6(c)

The Reactor Vessel Level Monitoring System (RVLMS) probe supports and subcomponents (i.e., brazement guide assembly j-bolt and nut) were identified in the aging management review as being potentially susceptible to cracking and stress relaxation.

Part of the ANO-1 surveillance specimen holder tube (SSHT) was removed in 1976. The upper SSHT assembly and some of the brackets and their bolts are still installed in the ANO-1 reactor vessel. Although the specimens have been removed and these components are not required to maintain their integrity for the specimens, these components must not become loose parts. Therefore, cracking and stress relaxation are applicable aging effects for the ANO-1 SSHT bolting.

The thermal shield and thermal shield upper restraint assemblies were not evaluated previously (in BAW-2248) but are fabricated from the same materials and exposed to the same environment as other core barrel assembly items. The applicable aging effects include cracking and reduction of fracture toughness.

In prior Section XI, 10-year ISI examinations for B-N-3 components in ANO-1, visual (VT-3) examination methods were performed of the remaining portions of the SSHT assembly still in place as well as the thermal shield and restraints. No relevant conditions were detected. This technique is adequate for detection of broken or loose parts from the bolted structures, and, therefore, VT-3 is the appropriate inspection method.

The other orphan components (i.e., brackets, bolts and nuts) were not included in the Section XI, 10-year ISI examinations since they are not considered to be B-N-3 (Removal Core Support Structures) and had not previously been identified as having potential aging concerns.

Using a VT-3 technique is adequate for detection of broken or loose parts from the bolted structures, and, therefore, VT-3 is the appropriate inspection method.

<u>6(d)</u>

Yes, these upcoming VT-3 examinations are intended to be an interim measure until screening and categorization of the orphan components have been completed. As indicated above in the response to EVIB-RAI-6(a), the screening and categorization process for the orphan components is expected to be completed in 2015.

<u>6(e)</u>

The orphan components were identified for ANO-1 as part of the aging management review for license renewal. These components include:

- Reactor vessel level monitoring system (RVLMS) components internal to the reactor vessel

  The ANO-1 reactor vessel internals have an additional component function, which is to support the RVLMS probes. Cracking and stress relaxation were identified as aging effects for the RVLMS brazement guide assembly j-bolt and nut.

- Remaining portions of surveillance specimen holder tubes (SSHT)

  Although all the specimens have been removed, portions of the shroud tube and the supports that are bolted to the core support shield remain in place. These components have the function of remaining secured to prevent loose parts in the reactor coolant system. Cracking and stress relaxation were identified as aging effects for the SSHT bolting.

- Thermal shield and thermal shield upper restraint

  Support the intended function of providing gamma and neutron shielding. Cracking and reduction of fracture toughness were identified as aging effects for the thermal shield and thermal shield upper restraint assemblies.

These orphan components have been put into a special category separate from the Existing Program Components. See Table 5-6 of the AMP. These orphan components have been placed in a separate category until the screening and categorization process is completed. At that time, the components will be placed in the appropriate category (primary, expansion, or no additional measures) based on the results of the screening and categorization. As indicated above in the response to EVIB-RAI-6(a), the screening and categorization process for the orphan components is expected to be completed in 2015.

2.2 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.

Response to EVIB-RAI-9

The previous 10-year ISI exams were reviewed. The vent valves were examined as part of the VT-3 exam performed in the vessel. The examination in 2008 noted raised metal in two locations. These indications were already known and previously dispositioned during the visual exams that are performed on all vent valves during each refueling outage. See response to EVIB-RAI-1(a).

2.3 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.

Response to EVIB-RAI-10

The inservice testing program specifies that the vent valves shall be stroke tested in the open and closed directions. There are three requirements that are to be performed every 18 months (every refueling outage as ANO-1 is on an 18 month cycle) for all of the vent valves. These requirements for testing of the valves are contained in ANO-1's technical requirements manual.

1. Conduct a remote visual inspection of visually accessible surfaces of each reactor internals vent valve body and disc surfaces and evaluate any observed surface irregularities.
2. Verify each reactor internals vent valve is not stuck in an open position.
3. Verify through manual activation that each reactor internals vent valve is fully open with a force $\leq$ 400 lbs (applied vertically upward).

2.4 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.*

Responses to EVIB-RAI-14

14(a)

This action will require further assessment of the flaw tolerance or functionality of the following components in the degraded condition:

- 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)

It is well known that the bolted configurations in the RV internals are highly redundant structures and that the structural integrity and functionality can be maintained with a fraction of the bolts remaining intact. Also, the core barrel cylinder and welds and the former plates must be capable of maintaining core support and alignment in the event of an accident condition so that the reactor may be shut down safely.

With regard to the acceptability of the inaccessible components by analyses, work is ongoing to evaluate these components to demonstrate no loss of functionality. Specifically, for the core barrel cylinder, this inaccessible component is an expansion item linked to cracking in the baffle plate. The goal of this evaluation is to show that the core barrel cylinder (and vertical and circumferential seam welds) will maintain functionality during the period of extended operation, through analyses/evaluations that confirm 1) the likelihood of a manufacturing defect is low, 2) susceptibility to SCC and IASCC is unlikely, 3) failure due to irradiation embrittlement is unlikely, and 4) even in the unlikely event of a failure of the core barrel cylinder, the reactor will still be able to shut down safely. For the barrel-to-former and baffle-to-baffle bolts, these inaccessible components are linked to baffle-to-former bolt cracking. The goal of this functionality evaluation is to show that the core barrel assembly bolting will maintain its function for a given period of operation, as limited by degraded core barrel-to-former and baffle-to-baffle bolts. For the former plates, this inaccessible component is an expansion item linked to cracking in the baffle plate. The goal of the evaluation is to show that the former plates will maintain functionality during the period of extended operation. These evaluations will be submitted by the end of 2016.

14(b)

There is no proposed alternative at this time. However, depending on the outcome of the inspections of the primary components and the evaluations of functionality of the inaccessible components noted above, ANO-1 will make a determination about the need for additional actions related to these components.

2.5 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.*

Responses to EVIB-RAI-15

15(a)

A revised screening and evaluation method for combined thermal and irradiation embrittlement of CASS has been proposed by the industry for NRC consideration and review. The continuing dialogue between the industry and NRC regarding these synergistic effects for CASS items may result in a different method to screen the CASS components. Pending the outcome of these discussions, this revised approach would be applicable for the ANO-1 CASS internals components. This two-phase screening approach will be applied to the ANO-1 CASS RV internals in order to identify and then prioritize which components may require further attention.

In the interim, ANO will perform additional studies of the CASS and PH stainless steel materials for the components identified here.

In order to confirm that the CASS and PH materials in the RV internals for ANO-1 can maintain functionality in the degraded condition, a multi-phase effort will be performed. The first task will be to identify susceptible RVI components using the screening criteria. Additional tasks are as follows:

- CRGT Spacer Castings – analyses will consider potential of failure based on the degree of susceptibility to embrittlement and magnitude of stresses present and plant-specific functionality.
- Core Support Shield Vent Valve Bodies – ferrite content has been determined to be below the screening criteria.
- IMI Spider Castings – an assessment will be performed of the potential of failure based on the degree of susceptibility to embrittlement and magnitude of stresses present for the IMI guide tube spiders.
- Core Support Shield Vent Valve Retaining Rings – an assessment will be performed of the potential of failure based on the degree of susceptibility to embrittlement and magnitude of stresses present for the vent valve retaining rings and plant-specific functionality.

These evaluations will be submitted by September 2015.

15(b)

There is no proposed alternative at this time. However, if functionality cannot be assured by these analyses, additional options will be identified to ensure functionality of these components after having completed the tasks described above in the response to EVIB-RAI-15(a).

2.6 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.

Response to EVIB-RAI-16

ANO-1 is on an 18 month refueling cycle. Refueling outage 1R26 will occur in 2016 and refueling outage 1R33 is projected to occur in 2027.

Prepared by:		Reviewed by:	
	2-5-15		2-5-15
Chris Lohse, P.E. Senior Consultant	Date	Timothy Griesbach Senior Associate	Date

Approved by:

	2-5-15
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