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August 24, 2000

1CAN080003

U. S. Nuclear Regulatory Commission
Document Control Desk
Mail Station OP1-17
Washington, DC 20555

Subject: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
License Renewal Application RAIs (TAC No. MA8054)

Gentlemen:

By letters dated April 25, 2000 (1CNA040006), and June 1, 2000 (1CNA060002), the NRC requested additional information concerning the Arkansas Nuclear One, Unit 1 (ANO-1) License Renewal Application (LRA). Attached are the responses to the requests for additional information (RAIs) pertaining to the reactor vessel and reactor vessel internals, sections 2.3.1.5, 2.3.1.6, 3.2.4, 3.2.5, and 4.2 of the ANO-1 LRA. Should you have any further questions, please contact me.

Very truly yours,

Jimmy D. Vandergrift
Director, Nuclear Safety Assurance

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Attachment

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Request for Additional Information Regarding ANO-1 LRA Reactor Vessel and Reactor Vessel Internals Sections 2.3.1.5, 2.3.1.6, 3.2.4, 3.2.5, and 4.2, dated April 25, 2000 (1CNA040006) and June 1, 2000 (1CNA060002)

2.3.1-1 **The reactor vessel leakage monitoring piping from the closure head is typically subject to an aging management review (AMR) (see letter from C.I. Grimes to D.J. Firth, dated October 27, 1999). This conclusion is consistent with the staff's guidance documented in license renewal issue number 98-0012, "Consumables" that was issued to the NEI on April 20, 1999. In this guidance, the staff stated that packing, gaskets, seals, and O-rings are not typically required by the current licensing basis to fulfill the functions of 10CFR Part 54.4(a)(1)(i) in accordance with American Society of Mechanical Engineers (ASME), Section III, NB2121, NC2121, and ND2121 because (by design) they are not relied upon for a pressure retaining function in components for which these Code design practices apply. In addition, the staff stated that "applicants can exclude packing, gaskets, seals, and O-rings where there is a clear basis for concluding that such components are not relied upon for a system, structure, or component to perform its intended function (s) under Part 54 . . . ."**

Inasmuch as these Code design practices do not apply to the O-ring in the closure head, the sealing surface of the vessel flange does provide the pressure boundary intended function for the closure head. Because the leakage monitoring piping penetrates the sealing surfaces of the vessel flanges, it should be treated as part of the reactor coolant system pressure boundary, and therefore, are within the scope of Part 54. Furthermore, by letter dated November 19, 1999, the Babcock and Wilcox Owners Group (BWOG) stated that the reactor vessel leakage monitoring pipes would be removed from the scope of BAW-2251A and should be addressed on a plant-specific basis by the applicant.

On page 2-25 of the LRA, the applicant states that "[t]he lines do not support the RCS pressure boundary..... If the reactor vessel closure O-rings fail and RCS fluid is introduced into the monitoring piping, leak flow would be limited since the ½-inch diameter hole in the vessel flange, which connects the region between the O-rings to the monitoring pipe, is less than the ID of the monitoring pipe. Therefore, the reactor vessel leakage monitoring piping is not subject to AMR since the piping does not directly support the RCS pressure boundary." These statements are not consistent with the guidance discussed above with respect to the leakage monitoring piping being part of the reactor coolant system pressure boundary. In addition,

more information is needed to justify the exclusion of the monitoring piping from AMR.

Include the leakage monitoring piping within the scope of components requiring aging management or provide additional technical justification for exclusion of the leakage monitoring piping consistent with the rule and staff guidance in sufficient detail to allow the staff to adequately assess its possible exclusion. The additional information should explain whether the ½-inch diameter hole in the vessel flange would limit leakage to less than normal reactor coolant system (RCS) makeup capacity thereby minimizing the consequences of failure of the reactor vessel monitoring pipes.

Please see RAI response 3.3.2.4.2-1.

2.3.1-4 **Please clarify whether the reactor vessel level monitoring system probe itself is subject to an AMR. If not, provide a justification for excluding the level probe from an AMR.**

The reactor vessel level monitoring system probe was installed to monitor fluid level in the upper plenum and head of the reactor vessel as part of the post-TMI modifications. It does not support an intended function and is not required to satisfy the criteria in 10CFR54.4 (a)(1), (2), or (a)(3).

3.3.2.4.2-1 **The staff considers that the monitoring pipes from the closure head should be subject to an aging management review (AMR) (see letter from C. I. Grimes to D.J. Firth, dated October 27, 1999). The October 27, 1999 letter describes the staff's reassessment of the conclusion that the innermost O-ring seal is the first pressure boundary. This conclusion is consistent with the staff's guidance for license renewal issue number 98-0012, "Consumables" that was issued to the NEI on April 20, 1999. In this position the staff stated that packing, gaskets, seals, and O-rings are not typically required by the current licensing basis to fulfill the functions of 10CFR Part 54.4(a)(1)(i) in accordance with ASME, Section III, NB2121, and ND2121 because (by design) they are not relied upon for a pressure retaining function in components for which these Code design practices apply. In addition, the staff stated that "applicants can exclude packing, gaskets, seals, and O-rings where there is a clear basis for concluding that such components are not relied upon for a system, structure, or component to perform its intended function (s) under Part 54....."**

Inasmuch as these Code design practices do not apply to the O-ring in the closure head, the sealing surface of the vessel flange does provide the pressure boundary intended function for the closure head. Because the leakage monitoring pipes penetrate the sealing surfaces of the vessel flanges, they should be treated as part of the reactor coolant system pressure boundary, and therefore, are within the scope of Part 54. To resolve this issue, the staff requests that the applicant correct the inconsistency with the Code regarding the pressure retaining function of the O-ring and provide an AMR of the monitoring pipes.

The reactor vessel leakage monitoring piping is discussed in Section 2.3.1.3 of the ANO-1 LRA. It has been determined that the piping used in this application does not perform a pressure boundary intended function and is not subject to aging management review. The following explanation is provided in ANO-1 LRA Section 2.3.1.3 and is consistent with the position taken by Duke Power for the Oconee Units, which has been reviewed and approved by the NRC (NUREG-1723, page 3-104).

"The 1-inch, Schedule 160, reactor vessel leakage monitoring system piping attached to the reactor vessel is Class 3 at ANO-1. The lines do not support the RCS pressure boundary and were not in the scope of BAW-2243A (RCS Piping Report) or BAW-2251A (Reactor Vessel Report). If the reactor vessel closure flange O-rings fail and RCS fluid is introduced into the monitoring piping, leak flow would be limited since the ½-inch diameter hole in the vessel flange, which connects the region between the O-rings to the monitoring pipe, is less than the inside diameter of the monitoring pipe. Therefore, the reactor vessel leakage monitoring piping is not subject to aging management review since the piping does not directly support the RCS pressure boundary."

In the event of leakage past the inner O-ring (i.e., does not seat properly), the leak flowrate through the downstream ½-inch diameter penetration in the reactor vessel flange during normal operation (i.e., heatup, cooldown, and power operation) is estimated to be within the makeup system capacity at ANO-1. The leakage monitoring piping does not support the reactor vessel pressure boundary intended function and is not subject to aging management review. The O-rings are not subject to aging management review in accordance with 10CFR54.21(a)(1)(ii) since they are replaced each refueling outage at ANO-1 as discussed in BAW-2251A, Section 2.5.1, page 2-9.

3.3.2.4.2-2 The description of the Alloy 600 aging management program (AMP) indicates that, in addition to the augmented inspections of the top three location groupings susceptible to primary water stress corrosion cracking (PWSCC), additional inspection locations may be added based on visual examination and qualitative assessment of risk. Explain what is meant by "qualitative assessment of risk" as it relates to potential additional inspections.

As discussed in Section 4.1 of Appendix B of the ANO-1 LRA, the augmented visual inspections of the three most susceptible Alloy 600 and Alloy 82/182 groupings will be carried forward to the period of extended operation. At present, the selection of inspection locations is based entirely on the susceptibility of Alloy 600 and Alloy 82/182 to PWSCC as determined by physical characteristics and operating experience to date. As additional ANO and industry operating experience is obtained, Entergy Operations will review the applicability of the PWSCC ranking model used to identify the top three locations, incorporate applicable operating experience and reassess the rankings. In addition, Entergy Operations may select a risk-based approach to identify additional locations for inspection.

With regard to risk-based inspection of Alloy 600 items, it is anticipated that a process would be selected that is similar to the method used to select RCS piping welds for inspection at ANO-1. That is, Entergy Operations uses a risk-informed approach to select locations for inspection in Class 1 piping based on ASME Code Case N-560. Entergy Operations considers the requirements for ISI selection prescribed by Code Case N-560 to be a method that uses a "qualitative assessment of risk."

3.3.2.4.2.2-1 As described in Section 4.18.1 of the LRA, the applicant is relying on the Master Integrated Reactor Vessel Surveillance Program (MIRVSP) to provide materials irradiation data on reactor vessel embrittlement to cover the period of extended operation. However, the LRA does not justify this approach by comparing the target fluence values for the capsules from the MIRVSP with the expected peak ANO-1 48 effective full power year (EFPY) fluence estimate at the inside surface of the reactor vessel. In addition to providing the target capsule fluence to reactor vessel comparison, provide a discussion of how neutron energy spectrum, gamma heating, and reactor inlet temperature will be monitored in order to ensure that the MIRVSP is applicable to the reactor vessel for the period of extended operation. .

A comparison of the 48 EFPY target fluence at ANO-1 to the capsules from the MIRVSP is provided in Section 4.1 of the BWOG Topical Report

entitled "Demonstration of the Management of Aging Effects for the Reactor Vessel", BAW-2251A, which was approved by the NRC in April 1999. As discussed in Section 4 of BAW-2251A (pages 4-6 and 4-7), the limiting weld at ANO-1 is WF-112 (Heat 406L44). The target 48 EFPY peak fluence at the inside surface of the reactor vessel at ANO-1 is less than 1.5×10^{19} n/cm^2 and in accordance with Table 4-1, page 4-26, Capsule AN1-C (WF-112, Heat 406L44) received a fluence of 1.5×10^{19} n/cm^2 and has been tested. Capsule DB1-LG2 (Weld WF-112, Heat 406L44) was irradiated to 1.6×10^{19} n/cm^2 and has been tested. In addition, WF-193, a surrogate of WF-112, is a material in the Point Beach Unit 2 reactor vessel surveillance program. A capsule containing this material was irradiated to 2.2×10^{19} n/cm^2 and has been tested. Therefore, the comparison of the target fluence to capsule data for the period of extended operation has been completed and is reported in BAW-2251A.

The ANO-1 Reactor Vessel Integrity Program described in Section 4.18 of Appendix B of the ANO-1 LRA will ensure that the neutron energy spectrum, gamma heating, and reactor inlet temperature are monitored such that the MIRVSP capsule data continues to be applicable to ANO-1 during the current term of operation as well as the period of extended operation. The requirements specified in Section 4 of BAW-2251A regarding fluence, neutron energy spectrum, gamma heating, and vessel inlet temperature will be included in the ANO-1 Reactor Vessel Integrity Program.

3.3.2.5.1-1 **On page 3-13 of the LRA, in the paragraph describing the industry experience, it is stated that "Subsequent to the issuance of BAW-2248A, the NRC issued Information Notice 98-11"... The BAW-2248A was published in 1999 while the IN 98-11 was issued in 1998. The earlier version of the BAW-2248A was published in 1997 as BAW-2248. Clarify this discrepancy or update the information as necessary.**

The text should read "Subsequent to the issuance of BAW-2248, the NRC issued Information Notice 98-11..." The BWOG Topical Report BAW-2248 was submitted to the NRC for review and approval in July 1997. The industry operating experience review reported in the original submittal of BAW-2248 pre-dated Information Notice 98-11. The BWOG topical report identified cracking of baffle bolting by irradiation-assisted stress corrosion cracking as an applicable aging effect and an update of the report to include a reference to Information Notice 98-11 would not have introduced any new aging effects.

3.3.2.5.2.2-1 An important issue for reactor vessel internal (RVI) components is characterizing the material behavior under high temperature and neutron irradiation conditions. The industry and BWOG have formed the Materials Reliability Project (MRP) to characterize loss of fracture toughness in materials of those components subject to these environmental conditions. An issue task group (ITG) has been formed to address the baffle-former bolt cracking issue. Also, the applicant has the responsibility to demonstrate that critical RVI components will have sufficient ductility to absorb local strain under loss-of-coolant accident (LOCA) and safe shutdown earthquake (SSE) loadings. The licensee has referenced the RVI AMP as part of the solution to Action Items 1, 3, 4, 5, 6, 7, 8, 9, and 12 of the NRC's safety evaluation report (SER) on the BWOG Topical Report BAW-2248. However, the content of the RVI AMP does not provide a demonstration that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation.

(a) Explain how the RVI AMP described in the LRA, Appendix B, will address the concerns raised in each of the Action Items noted above.

The responses for ANO-1 to the applicant action items listed in BAW-2248A are provided in Table 2.3-5 of the ANO-1 LRA. Specifically, the RVI Aging Management Program is referenced in the responses to action items 1, 4, 5, 6, 7, 8, 9, and 12. Applicant action item 3 is a scoping issue regarding the thermal shield that is addressed in Sections 2.3.1.6 and 3.2.5 of the ANO-1 LRA.

The RVI Aging Management Program addresses three groups of internals items: (1) plates, forgings, welds, core barrel bolts, and thermal shield bolts, (2) baffle bolts, and (3) cast austenitic stainless steel (CASS) and martensitic stainless steel items. Inspection of limiting items within each of the three groups will be conducted at ANO-1 in the fifth inspection interval unless research data indicate inspection of limiting items is not required. The specific method of inspection for limiting items within the three groups listed above is discussed in Section 3.5 of Appendix B to the ANO-1 LRA. Please see RAI response 4.1-1 for the response to applicant action item 12.

The RVIAMP program elements that address specific applicant action items are addressed below.

Reactor Vessel Internals Applicant Action Items from Section 4.1 of BAW-2248A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
(1) The license renewal applicant is to verify that the critical parameters for the plant are bounded by the topical report. Further, the renewal applicant is to commit to programs described as necessary in the topical report to manage the effects of aging during the period of extended operation on the functionality of the reactor vessel internals components. The applicant for license renewal will be responsible for describing any such commitments and proposing the appropriate regulatory controls. Any deviations from the aging management programs within this topical report described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor vessel internal components or other information presented in the report, such as materials of construction, will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10CFR54.21(a)(3) and (c)(1).	Entergy Operations participated in the development of BAW-2248A by providing ANO-1 specific design and operational information. Entergy Operations has reviewed the current design and operation of the reactor vessel internals using the process described in Section 2.3.1.2 and confirms that ANO-1 is bounded by the description contained in BAW-2248A with regard to critical plant parameters. Reactor vessel internals items that ANO-1 identified as subject to aging management review that were not within the scope of BAW-2248A are described in Section 2.3.1.6. Section 3.2.5 of the ANO-1 LRA contains a discussion of applicable aging effects and a list of programs credited for managing the applicable aging effects. Specifically, the ANO-1 aging management programs for the reactor vessel internals include ASME Section XI, Subsection IWB, Primary Chemistry Monitoring Program, and the Reactor Vessel Internals Aging Management Program. These programs are described in Appendix B of the ANO-1 LRA.
(4) Applicants must commit to participation in the BWOG RVIAMP, and any other industry programs as appropriate, to continue the investigation of potential aging effects for RVI components and to establish monitoring and inspection programs for RVI components. The applicant shall provide the NRC with either annual or periodic updates (after completion of significant milestones) on the status of the RVIAMP, commencing within one year of the issuance of the renewed license.	As described in Section 3.5 of Appendix B of the ANO-1 LRA, ANO-1 will participate in appropriate BWOG and industry level programs, as appropriate, to ensure that the reactor vessel internals intended functions are maintained consistent with the current licensing basis during the period of extended operation. Also as described in Section 3.5 of Appendix B of the ANO-1 LRA, Entergy Operations will provide written reports to the staff upon completion of significant RVIAMP milestones commencing within one year of the issuance of the renewed license.

Reactor Vessel Internals Applicant Action Items from Section 4.1 of BAW-2248A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
(5) The applicant must describe plans for augmented inspection of RVI components for management of SCC/IASCC and loss of fracture toughness (neutron embrittlement) of the RVI components. This description should specify the sample size, the examination method, acceptance criteria and timing of the inspection, or the process to be used to specify these items.	As described in Appendix B of the ANO-1 LRA, items that may be susceptible to cracking by SCC/IASCC and reduction of fracture toughness by irradiation embrittlement include plates, forgings, welds, core barrel bolts, and thermal shield bolts. The ANO-1 inspection is planned for the 2nd or 3rd period of the fifth inspection interval. Sample size, appropriate inspection methods (e.g., visual or volumetric), and acceptance criteria will be determined based on the results of the BWOG Reactor Vessel Internals Aging Management Program and other industry programs investigating the aging effects for the reactor vessel internals.
(6) According to the BWOG, one of its objectives in BAW-2248 states, "It is intended that NRC review and approval of this report will allow that no further review of the matters described herein will be needed when the report is incorporated by reference in a plant specific renewal license application." The license renewal applicant must address the baffle-former bolt cracking issues addressed in Section 3.3.2 of this SE pertaining to Refs. 4 and 5, with regard to the ITG project, initiated after April 23, 1998, to address generic RVI materials issues. The BWOG indicates this industry effort resulted in subsequent changes in the BWOG RVI aging management program. The ITG is currently addressing the issues of cracking of baffle bolts. The BWOG indicates that the changes in the aging management program now requires the applicants to be responsible for using the industry ITG project developed information to determine the necessary steps (e.g., inspection, operability determinations, and replacements) for the management of the applicable baffle bolt aging effects.	The RPV Internals Issue Task Group (ITG) has developed generic programs that address applicable aging effects for baffle bolting such as cracking, reduction of fracture toughness, void swelling, and stress relaxation. In addition, the EPRI Joint Baffle Bolting Program has been incorporated into the RPV internals ITG to provide plant inspection and research data. The BWOG will utilize data and information from the ITG programs to perform design-specific assessments and analyses. The BWOG will provide support to the ITG by providing plant- and design-specific data, such as fluence and stress profiles. ANO-1 will be responsible for using the tools provided by the ITG and the BWOG to determine the necessary steps (e.g., inspections, operability determinations, and replacements) to manage the applicable aging effects at ANO-1. The ANO-1 RVIAMP includes commitments to participate in BWOG and applicable ITG programs.

Reactor Vessel Internals Applicant Action Items from Section 4.1 of BAW-2248A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
(7) The applicant must describe plans for augmented inspection of RVI components for management of loss of fracture toughness due to thermal aging embrittlement of the RVI components. This description should specify the sample size, the examination method, acceptance criteria and timing of the inspection, or the process to be used to specify these items.	In accordance with Appendix B of the ANO-1 LRA, all items fabricated from CASS and martensitic stainless steel may be susceptible to reduction of fracture toughness by thermal embrittlement. ANO-1 will rely on the EPRI ITG and BWOG programs to quantify the susceptibility of the CASS and martensitic parts to reduction of fracture toughness by thermal and irradiation embrittlement. Where reduction of fracture toughness is identified as a concern for the period of extended operation, the sample size for augmented inspection will be determined by considering both the susceptibility and consequences of failure of the affected parts. Once the sample size is determined, analytical evaluations will be performed to determine critical flaw sizes for the affected parts. ANO-1 will use fracture toughness data obtained through the ITG and BWOG programs. The inspection methods and acceptance criteria for the inspections will be based on the calculation of critical flaw sizes. At present, it is anticipated that enhanced visual examinations will be sufficient to detect flaws in CASS and martensitic steel parts; however, the analytical evaluations must be completed before the inspection method can be selected. The inspection is planned for the 2nd or 3rd period of the fifth inspection interval at ANO-1.
(8) The applicant must describe plans for management of stress relaxation for bolted closures of the RVI. This description should specify the critical locations, and monitoring and inspection techniques, and timing of the inspection, or the process to be used to specify these items.	Activities are in progress in industry (ITG) and the BWOG (RVIAMP) to further define the applicability of stress relaxation to the B&W-designed reactor vessel internals for the period of extended operation. In the RVIAMP described in Section 3.5 of the ANO-1 LRA, Entergy

Reactor Vessel Internals Applicant Action Items from Section 4.1 of BAW-2248A SER	
Renewal Applicant Action Item	**ANO-1 Specific Response**
	Operations committed to participate in these industry programs. In addition, the inspection at ANO-1, planned for the end of the fifth interval, will focus on the effect of stress relaxation for critical bolted connections at ANO-1.
(9) The applicant must address aging management of void swelling. An adequate aging management program (AMP) would include participation in industry program(s) to address the significance of void swelling (either individually or through an owners or industry group), a commitment to develop a sufficient inspection program (including the basis, methods, locations to be examined, timing, frequency and acceptance criteria) for management of the issue based upon the results of the industry programs, and a commitment to implement the inspection program prior to the end of the current license period.	As described in the response to applicant action item 4 above, Entergy Operations is participating in industry and BWOG programs that will assess the applicability of void swelling to the B&W-designed internals. In addition, the inspections at Oconee and ANO-1 will further assess the applicability of void swelling to the ANO-1 reactor vessel internals during the period of extended operation. The inspection program will be developed prior to the end of the current license period. However, actual inspections, if required, would be performed in the 2nd or 3rd period of the fifth inspection interval at ANO-1.
(12) Plant-specific analysis is required to demonstrate that, under loss-of-coolant-accident (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 adversely affect deformation limits. The RVIAMP must develop data to demonstrate that the internals will meet the deformation limits at the expiration of the renewal license.	The AMP requires that a plant-specific analysis 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 through the ITG and BWOG to demonstrate that the internals will meet the deformation limits at the expiration of the renewed license. Please see the ANO-1 response to RAI 4.1-1 for further clarification regarding this applicant action item.

(b) Section 4.6 of BAW-2248 does not address program elements for void swelling. In its final safety evaluation report for the topical report, the staff concluded that the aging effect of void swelling in RVI components will not be managed by the current ASME Section XI inspection requirements, since the dimensional changes due to void swelling may not be large enough to be detected by VT-3 visual inspection. Provide program elements to manage the effects of void swelling on RVI internals.

Void swelling is addressed by the ANO-1 RVI Aging Management Program and is discussed in Appendix B, Section 3.5, of the ANO-1 LRA. The approach used by ANO-1 to manage dimensional changes caused by void swelling and other applicable aging effects delineated in the applicant action items during the period of extended operation is consistent with the approach taken by Duke Power for the Oconee Units, which was reviewed and approved by the NRC [References: Duke Power letter from M. Tuckman to the NRC, dated December 17, 1999, Docket Nos. 50-269,-270,-287, entitled License Renewal Response to NRC Letter dated November 18, 1999; and, NUREG-1723, Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3, pages 3-120 through 3-122].

3.3.2.5.2.2-2 In Section 3.5 of Appendix B of the LRA, the applicant stated that the reactor vessel internals aging management inspection will begin during the period of extended operation. The scope and frequency of the inspection was not specified in the application. Provide the scope, the timing of the first inspection, and frequency of the RVI aging management inspection proposed for the period of extended operation.

The scope of the inspection is provided in Appendix B, Section 3.5, and is consistent with the scope identified by Duke Power for the Oconee Units as approved by the NRC in NUREG-1723, pages 3-120 through 3-122. The text from Appendix B, Section 3.5, regarding Scope is repeated below. Please see the ANO response to RAI 3.3.2.5.2.2-1 for additional details regarding the inspection at ANO-1.

"The inspection applies to the reactor vessel internals stainless steel items for ANO-1. These items can be separated into three groups: (1) items comprised of plates, forgings, welds, core barrel bolts, and thermal shield bolts; (2) baffle bolts and; (3) items fabricated from CASS and martensitic steel. More specifically, the items fabricated from CASS and martensitic steel include control rod guide tube spacers, vent valve bodies, and incore

guide tube assembly spiders. The vent valve retaining rings, fabricated from martensitic stainless steel, are also included in this inspection."

Entergy Operations plans to perform the inspection of the ANO-1 internals in the second or third period of the fifth inspection interval in accordance with the summary description in Appendix B, Section 3.5. At present, the inspection at ANO-1 is planned after the first inspection is completed at Oconee. The need for subsequent inspections at ANO-1 will be based on the inspection results at Oconee and ANO-1.

The Oconee and ANO-1 reactor vessel internals are similar in design, materials of construction, and operating environment as described in BAW-2248A. This has been confirmed through plant-specific references to the BWOG topical report in the ANO-1 and Oconee Applications. The first inspection at Oconee, which is scheduled early in the period of extended operation, will be useful in assessing the applicability of the aging effects described in BAW-2248A to the B&W-designed reactor vessel internals prior to the inspection at ANO-1. The inspections in the period of extended operation, if necessary, at Oconee and at ANO-1 will be coordinated through the BWOG RVIAMP. Should data or evaluations indicate that the inspection at ANO-1 can be modified or eliminated, Entergy Operations will provide plant-specific justification to demonstrate the basis for modification or elimination of the inspection.

4.1-1 **Table 4.1-1 of the LRA provides a list of time-limited aging analyses (TLAAs) that are applicable to RVI. The table indicates that a TLAA related to the acceptability of the RVI under LOCA and seismic loading is addressed in Section 3.5, Appendix B of the LRA which references topical report BAW-2248. The topical report indicated that these TLAAs will be resolved on a plant-specific basis per 10CFR54.21(c)(1)(iii) based on the results and conclusion of the RVI AMP. The RVI AMP will provide the necessary data on neutron embrittlement for fracture resistance of the RVI components. This was identified as Renewal Applicant Action Item 12 in the staff safety evaluation concerning BAW-2248. Table 2.3-5 of the LRA contains the response to Renewal Action Item 12 for the RVI. The response indicates that a plant-specific analysis will be performed to demonstrate that under LOCA and seismic loading, the internals have adequate ductility to absorb the local strain intensity during the extended period of operation. However, the LRA does not address**

how and when this TLAA will be evaluated. Provide a discussion of the program plan for evaluating the TLAA including the schedule for completion of the evaluation.

The TLAA reported in BAW-2248A regarding ductility of stainless steel and deformation limits will be evaluated by Entergy Operations when sufficient embrittlement data is collected through the BWOG and EPRI MRP programs. Once the embrittlement data is available, the TLAA reported in BAW-2248A will be updated. The TLAA will be resolved using 10CFR54.21(c)(1)(iii), which is consistent with the approach taken by Duke Power for the Oconee Units that was approved by the NRC (NUREG-1723, page 4-24). At present, ANO plans to complete the evaluation prior to the end of the current term of operation.

4.2.3-1 **The aging management program for reactor pressure vessel (RPV) neutron embrittlement relies on the acquisition of data from RPV surveillance capsules. The LRA indicated that this data may be applied to assess RPV material embrittlement through the use of RG 1.99, Revision 2, Position 2. For the Position 2 methodology described in Section 4.2.1 of the LRA, address how the ratio procedure (discussed in the first paragraph of RG 1.99, Revision 2, Section 2.1) will be incorporated into the determination of RPV material chemistry factors based on credible surveillance data.**

As documented in the NRC's SER of BAW-2251 [NRC Letter from C.I. Grimes to D.J. Firth, "Acceptance for Referencing of Generic License Renewal Program Topical Report entitled, "Demonstration of the Management of Aging Effects for the Reactor Vessel, June 1996," dated April 26, 1999], Section 3.4.2.2, 48 EFPY RT_{PTS} values for the ANO-1 beltline materials are below the pressurized thermal shock (PTS) screening criteria using Position 1.1 of Regulatory Guide 1.99, Revision 2. Should Position 2 be used by ANO-1 at some time in the future, the method that would be used to apply the ratio procedure for the calculation of RT_{PTS} would be consistent with the method reported in BAW-2325, Revision 1.

4.2.3-2 **The staff notes that in Section 4.2.1 of the LRA, reference is made to, "[s]ince BAW-2251A was completed prior to the staff's approval of BAW-2325, Revision 1..." It should be recognized that the staff has not reviewed and approved any revision of BAW-2325. This statement in the application should therefore be amended.**

The text should have stated that "Since BAW-2251A was completed prior to the staff's review of BAW-2325, Revision 1....."

With regard to NRC review of BAW-2325, Entergy Operations provides the following summary:

- In letter dated July 1, 1998 (1CAN079801), Entergy Operations informed the staff that responses to RAIs regarding Generic Letter 92-01 are included in BAW-2325, Revision 0.

- In letter dated October 13, 1998, the BWOG received a letter from the NRC with RAIs regarding BAW-2325, Revision 0.

- Supplement 1 to BAW-2325, issued to the NRC on December 16, 1998 by the BWOG (OG-1732), included responses to the RAIs.

- During the development of Supplement 1, the BWOG found errors in the BAW-2325, Revision 0, submittal. The errors were corrected and discussed in Supplement 1. The corrections discussed in Supplement 1 resulted in Revision 1 to BAW-2325, which was submitted by the BWOG to the NRC on February 2, 1999 (OG-1739).

- Entergy Operations submitted a letter on March 11, 1999 (1CAN039902), informing the NRC that Revision 1 to BAW-2325 impacted the lower nozzle belt to upper shell circumferential weld only. It was not the limiting weld and the pressure-temperature, low temperature overpressure protection, and PTS evaluations remain bounding.

In letter dated July 15, 1999 (1CNA079902), the NRC stated that Entergy Operations had provided responses to the NRC's RAIs on July 1, 1998, with corrections submitted on March 11, 1999. It appears, therefore, that the NRC has reviewed BAW-2325, Revision 1, to complete closeout of Generic Letter 92-01 for ANO-1.

4.2.3-3 **In Section 4.2.2 of the LRA, reference is made to the determination of the T/4 fluence values for the purpose of evaluating upper shelf energy drop for RPV materials. It is not clear how the T/4 fluence value is calculated. Confirm that, in accordance with RG 1.99, Rev. 2, Section 1.1, the attenuation of the corresponding clad-to-base metal fluence value was performed by using either: (1) equation 3 from the RG 1.99, Rev. 2, or (2) is based on the calculation of plant-specific displacement per atom (DPA) attenuation. This use of DPA to establish the plant-specific attenuation function is also addressed in Section 1.1 of RG 1.99, Rev. 2.**

The method used by the BWOG to calculate the T/4 fluence reported in BAW-2251A was the subject of RAI #6. Please see the response to RAI #6 in Appendix E to BAW-2251A for a detailed discussion of the calculation of T/4 fluence for the purpose of calculating reduction in upper shelf energy. In addition, please see Appendix D to BAW-2251A and BAW-2241PA for discussions of fluence monitoring for the B&W-designed rector vessels. The T/4 fluence values were approved by the NRC in accordance with the SER of BAW-2251A [Letter from C.I. Grimes of the NRC to David J. Firth, April 26, 1999, Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled "Demonstration of the Management of Aging Effects for the Reactor Vessel," BAW-2251, June 1996].