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2CAN070405

July 22, 2004

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Request for Additional Information Responses for License Renewal Application TAC No. MB8402
Arkansas Nuclear One – Unit 2
Docket No. 50-368
License No. NPF-6

Dear Sir or Madam:

By letter dated June 11, 2004 (2CNA060401), the NRC requested additional information on the Arkansas Nuclear One, Unit 2 (ANO-2) License Renewal Application (LRA) within 45 days. The requests for additional information (RAIs) are from the LRA Section 3.1, Reactor Vessel Internals and Reactor Coolant System, Section 4.2, Other Plant-Specific Time-Limited Aging Analyses, and Appendix B. The responses to the RAIs are contained in the Attachment 1.

New commitments contained in this submittal are summarized in Attachment 2. Should you have any questions concerning this submittal, please contact Ms. Natalie Mosher at (479) 858-4635.

I declare under penalty of perjury that the foregoing is true and correct. Executed on July 22, 2004.

Sincerely,

Timothy G. Mitchell
Director, Nuclear Safety Assurance

TGM/nbm

Attachments

A100

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Attachment 1

2CAN070405

RAI Responses

Section 3.1, 4.2, and Appendix B RAI Responses

RAI 3.1.2.1-1: The Staff requests additional information on the applicant's aging management reviews for managing cracking in low-alloy steel components that are exposed to an external air environment. Aging management strategies for license renewal are somewhat dependent on the specific types of aging mechanisms that can induce age-related degradation, and not specifically on the general classification of the aging effect. For the low-alloy steel components in the reactor coolant system (RCS), confirm that cracking is an applicable aging effect requiring aging management. Specifically, define which aging mechanism or mechanisms are known to induce cracking in low-alloy steel components that are exposed to an external air environment.

Response: As specified in the Statement of Considerations (SOC) for the license renewal rule, "the intent of the license renewal review has been clarified to focus on the adverse effects of aging rather than identification of all aging mechanisms. The final rule is intended to ensure that important systems, structures, and components will continue to perform their intended function during the period of extended operation. Identification of individual aging mechanisms is not required as part of the license renewal review. The definitions of age-related degradation, age-related degradation unique to license renewal, aging mechanisms, renewal term, and effective program have been deleted." The license renewal rule focuses on maintaining the intended functions of components subject to aging management review and does not focus on identification of aging mechanisms.

Low-alloy steel items exposed to an external air environment that are susceptible to cracking are limited to fasteners (for example, reactor vessel closure studs) and exterior attachments to vessels. Fasteners are not intentionally exposed to water or steam, but exposure may result from gasket leaks. If leakage is combined with contaminant species, such as sulfides or chlorides, an aggressive environment that can promote stress corrosion cracking (SCC) may result.

For the RCS components fabricated from low-alloy steel, including exterior attachments to vessels, cracking (initiation by fatigue and growth of pre-service flaws at welded joints due to service loadings) at welded joints is considered an aging effect requiring management for the period of extended operation. The most susceptible locations from a structural standpoint for flaw initiation and growth are at the welded joints. Growth of fabrication flaws due to service loads is the basis for the American Society of Mechanical Engineers (ASME) Section XI inspections.

RAI 3.1.2.1-2: The Staff requests additional information on the applicant's aging management reviews for managing loss of material in nickel-based alloy components that are exposed to an internal environment of treated borated water. For the nickel-based alloy components in the RCS, define which aging mechanism or mechanisms are known to induce loss of material in nickel-based alloy components that are exposed to an internal environment of treated borated water and explain how this aging effect will be addressed by the aging management strategy for license renewal.

Response: Loss of material can be induced by crevice and pitting corrosion in nickel-based alloy steels. If RCS fluid chemistry is not rigorously controlled, the concentration of system fluid contaminants could lead to loss of material due to pitting or crevice corrosion of the nickel based material. This aging effect is addressed by maintaining rigorous control of RCS chemistry under the Water Chemistry Control Program.

Loss of material due to wear has the potential to occur between the nickel-based alloy core stabilizing lugs and the core barrel. While there has been no operating experience at ANO-2 showing that wear has occurred in this location, relative motion due to handling of the vessel internals or thermal expansion during heatup and cooldown could lead to loss of material due to wear. Loss of material due to wear is managed by the Inservice Inspection (ISI) Program.

The nickel-based alloy steam generator U-tubes are subject to loss of material by sliding wear at tube support locations. Loss of material by sliding wear occurs when forces imposed on the tubes by the secondary fluid cause high frequency vibration of the tubes and tube support structures. Loss of tube material due to wear is managed by the Steam Generator Integrity Program.

RAI 3.1.2.1-3: The Staff requests additional information on the applicant's aging management reviews for managing loss of material in stainless steel components that are exposed to an internal environment of treated borated water. For the stainless steel components in the RCS, define which aging mechanism or mechanisms are known to induce loss of material in stainless steel components that are exposed to an internal environment of treated borated water and explain how this aging effect will be addressed by the aging management strategy for license renewal.

Response: Loss of material can be induced by crevice and pitting corrosion of stainless steel in treated borated water if the RCS fluid chemistry is not rigorously controlled. This aging effect is addressed by maintaining rigorous control of RCS chemistry under the Water Chemistry Control Program.

Various stainless steel components at ANO-2 (such as the reactor vessel internals) are subject to flow induced vibration during plant operation and differential thermal expansion and contraction movement during plant heat-up, cool-down, and changes in power operating cycles. Flow induced vibration and thermal expansion can cause repetitive relative movement between stainless steel interfacing and mating surfaces. This relative movement between the interfacing and mating surfaces may result in surface wear. Loss of material due to wear of these interfacing and mating surfaces is managed by the ISI Program.

RAI 3.1.2.1-4: Clarify where the boric acid corrosion aging mechanism is considered in Section 3.1 of the License Renewal Application (LRA), and in Tables 3.1.2-1 and 3.1.2-3. Specify which component types, materials, environments, aging effects requiring management, and aging management programs are associated with this aging mechanism and explain how this aging effect will be addressed by the aging management strategy for license renewal.

Response: In Section 3.1 and in Tables 3.1.2-1 and 3.1.2-3, boric acid corrosion is an applicable mechanism for loss of material for carbon steel and low-alloy steel components with an external air environment.

Carbon and low-alloy steel components (including all bolting materials, piping and fittings, reactor coolant pump (RCP) driver mounts, and vessels and support skirts) of the RCS exposed to an external air environment are susceptible to loss of material by boric acid corrosion. This aging effect is managed by the Boric Acid Corrosion Prevention Program discussed in Section B.1.3 of the LRA.

RAI 3.1.2.2-1: The Staff requests additional information on the applicant's aging management reviews for managing loss of material and cracking in cast austenitic stainless steel (CASS) components that are exposed to an internal environment of treated borated water. For the CASS components in the RCS, define which aging mechanism or mechanisms are known to induce loss of material and cracking in CASS components that are exposed to an internal environment of treated borated water and explain how this aging effect will be addressed by the aging management strategy for license renewal.

Response: If RCS fluid chemistry is not rigorously controlled, the concentration of system fluid contaminants could lead to loss of material from pitting and crevice corrosion of CASS material.

The CASS material may be susceptible to cracking by SCC/intergranular attack (IGA) if exposed to high concentrations of contaminants in the treated borated water. In addition, irradiation-assisted stress corrosion cracking (IASCC) is a degradation mechanism for CASS reactor internals items where materials become more susceptible to stress corrosion cracking with increasing exposure to neutron irradiation. The relatively benign environment of the RCS fluid, which incorporates hydrogen overpressure to reduce oxygen levels, reduces the potential for IASCC degradation of ANO-2 reactor internals CASS items.

Loss of material and cracking of CASS RCS items are managed by the combination of the ISI Program and the Water Chemistry Control Program.

Loss of material due to pitting or crevice corrosion and cracking by SCC/IGA or IASCC are potential aging mechanisms for CASS reactor internals items in treated borated water. The only CASS item in the reactor vessel internals is the control element assembly shroud tube. Loss of material of this item is managed by the combination of the ISI Program and the Water Chemistry Control Program. In addition to the ISI Program and the Water Chemistry Control Programs, cracking of this item is managed by the Reactor Vessel Internals - CASS Components Program described in LRA Section B.1.22.

RAI 3.1.2.3-1: In Table 3.1.2-3, on page 3.1-79, the applicant identifies treated water as the external environment for the RCP thermal barrier heat exchanger inner coil. In addition, on page 3.1-80, the applicant identifies treated water as the internal environment for the RCP thermal barrier heat exchanger outer coil and bored-hole heat exchanger. Loss of material, cracking, and fatigue are defined as the aging effects requiring management.

The aging management programs identified to manage these aging effects are "Inservice Inspection" and "Time-Limited Aging Analysis (TLAA) - Metal Fatigue." The applicant's Auxiliary Systems Water Chemistry Control Program, described in Section B.1.30.1, identifies its purpose as managing loss of material, cracking, and fouling of components exposed to treated water systems. The applicant has identified similar components of the same material which are exposed to the same environment as being managed by a Water Chemistry Program and referenced concurrence with NUREG-1801, VII.C2.2-a. Provide justification for excluding an aging management program to manage the water chemistry of the treated water environment as applicable to these components.

Response: The treated water identified in the ANO-2 LRA which supplies cooling to the RCP thermal barrier heat exchangers is part of the ANO-2 component cooling water system (CCW). The chemistry controls for this system are not sufficiently rigorous to control the contaminants which could potentially lead to loss of material and cracking in the RCP thermal barriers. Therefore, the Component Cooling Water Chemistry Control Program is not credited as managing these aging effects. The ISI Program will manage these aging effects such that corrective action may be taken prior to loss of the intended function.

RAI 3.1.2.4-1: The Staff requests additional information on the applicant's aging management reviews for managing cracking in carbon steel components that are exposed to an external air environment (i.e., support skirt). For the carbon steel components in the RCS, define which aging mechanism or mechanisms are known to induce cracking in carbon steel components that are exposed to an external air environment and explain how this aging effect will be addressed by the aging management strategy for license renewal.

Response: Refer to response to RAI 3.1.2-1.1 regarding aging mechanisms associated with cracking of low-alloy steel exterior attachments to vessels.

RAI 3.1.2.4-2: The Staff requests additional information on the applicant's aging management reviews for managing cracking in stainless steel components that are exposed to an external air environment (i.e., mechanical nozzle seal assembly clamp bolting (studs, nuts, washers)). For the stainless steel components in the RCS, define which aging mechanism or mechanisms are known to induce cracking in stainless steel components that are exposed to an external air environment and explain how this aging effect will be addressed by the aging management strategy for license renewal.

Response: Generally, stainless steel exposed to an external air environment is not susceptible to aging effects requiring management. Insulation material used for RCS components has low soluble chloride and other halide content to minimize the possibility of SCC of stainless steel components. However, stainless steel items such as flange and valve bolting in air are subject to cracking as indicated in Table 3.1.2-3 on page 3.1-68 of the LRA. Stainless steel fasteners are not intentionally exposed to water or steam, but exposure may result from gasket leaks. If leakage is combined with contaminant species, such as sulfides or chlorides, an aggressive environment that can promote SCC may result. Therefore, cracking of stainless steel flange and valve bolting is considered an aging effect requiring management for the period of extended operation. Even though cracking is not expected, the ISI Program is credited to confirm the absence of cracking due to SCC.

RAI 3.1.2.4-3: The Staff requests additional information on the applicant's aging management reviews for managing loss of material and cracking in low-alloy steel-clad with stainless steel and nickel-based alloy components that are exposed to an internal environment of treated borated water. For the low-alloy steel clad with stainless steel and nickel-based alloy components in the RCS, define which aging mechanism or mechanisms are known to induce loss of material and cracking in low-alloy steel clad with stainless steel and nickel-based alloy components that are exposed to an internal environment of treated borated water and explain how this aging effect will be addressed by the aging management strategy for license renewal.

Response: The stainless steel cladding and nickel-based alloy cladding are susceptible to cracking by SCC and primary water stress corrosion cracking (PWSCC), respectively. Both the stainless steel cladding and nickel-based alloy cladding are susceptible to loss of material by crevice or pitting corrosion as discussed in the response to RAI 3.1.2-2 and 3.1.2-3. The aging effect and associated mechanisms applicable to the underlying ferritic steel are discussed below.

For the underlying ferritic steel, service loadings may result in the growth of pre-service flaws or initiation and growth of service-induced flaws. Cracking at the welded low-alloy steel joints is considered an aging effect requiring management for the period of extended operation. Growth of fabrication flaws due to service loads is the bases for the ASME Section XI inspections as documented in EPRI NP-1406-SR, Nondestructive Examination Acceptance Standards techncial basis and development of Boiler and Pressure Vessel Code, ASME section XI, Division 1.

RAI 3.1.2.4-4: Table 3.1.2-4, Page 3.1-84 identifies the pressurizer lower head, lower shell, upper shell, and upper head as component types. The applicant identified the aging effect of loss of material, and specified that it is applicable to the unclad low-alloy steel of the lower head only. Provide justification for limiting the aging effect to only the lower head since many components of the pressurizer are susceptible to boric acid corrosion in a treated borated water environment, and would require that the aging effect of loss of material is managed.

Response: Table 3.1.2-4, on page 3.1-84 identifies loss of material as an aging effect requiring management for unclad lower vessel head low-alloy steel exposed to treated borated water. The applicable locations for this table entry are heater nozzle penetrations that have been repaired. The Alloy-600 nozzle may contain a through-wall flaw that exposes the underlying ferritic steel to treated borated water. These locations are susceptible to loss of material due to exposure to treated borated water. See the response to RAI 4.7.5-1 for further discussion regarding expected corrosion rates at these locations. In addition, loss of material is identified in Table 3.1.2-4 for low-alloy steel pressurizer items exposed to air with the potential for leaking borated water. In Table 3.1.2-4 on page 3.1-83 of the LRA, the ANO-2 pressurizer upper head, upper shell, lower head, and lower shell are identified as susceptible to loss of material due to boric acid corrosion in an external air environment.

RAI 3.1.2.4-5: Recent operational experience at both domestic and foreign facilities (Palo Verde Units 2 and 3, Millstone Unit 2, Waterford Unit 3, and Tsuruga Unit 2 in Japan) has shown that leakage of pressurizer penetrations due to PWSCC is an aging effect that requires management. Since the aging management program B.1.19 Pressurizer

Examinations is limited only to managing cracking of the stainless steel and nickel-based alloy cladding and attachment welds by examination of the adjacent base metal, discuss how the aging effect of PWSCC will be managed for the pressurizer penetrations for the period of extended operation at ANO-2. Include scope, frequency, technique, acceptance criteria, and the technical basis for future examinations.

Response: Nickel-based alloy penetrations associated with the ANO-2 pressurizer include pressure measurement, vent, level, and temperature nozzles, heater penetration nozzles and plugs, and Alloy-82/182 welds. All of these nickel-based alloy items are exposed to treated borated water and are susceptible to PWSCC. This aging effect is managed by a combination of the ISI Program, the Water Chemistry Control Program, and the Alloy-600 Program. In the ANO-2 LRA, details of these programs, including scope, frequency, technique, acceptance criteria, and the technical basis for future examinations, are discussed in Sections B.1.14, B.1.30, and B.1.1, respectively.

RAI 4.2-1: The applicant assumes a capacity factor of 80% for TLAAs associated with reactor vessel neutron embrittlement that are described in Section 4.2 of the LRA. These evaluations are based on end-of-license fluences corresponding to 48 effective power years (EFPY). Staff reviews of current and future trends for plant operations in the nuclear power industry indicate capacity factors of 90% or greater for many plants. Provide justification for the estimated 48 EFPY fluence for ANO-2. If the estimated 48 EFPY fluence cannot be justified, provide results of revised evaluations for reactor vessel neutron embrittlement at higher levels of fluence projected to the end of the period of extended operation.

Response: The ANO-2 end-of-life fluence estimate for the period of extended operation is based on 48 EFPY, which assumes a plant capacity factor of 80% over 60 years. This is consistent with the method used to calculate 40-year fluence estimates reported in the ANO-2 response to Generic Letter 92-01. At present, the lifetime capacity factor for ANO-2 through 26 years of operation is approximately 80%. Therefore, it is reasonable to assume a lifetime capacity factor of 80% when evaluating 60 years of operation.

The impact on fracture toughness of operation at capacity factors in excess of 80% is addressed by the Reactor Vessel Integrity Program. As described in Section 4.2 of the ANO-2 LRA, the ANO-2 Reactor Vessel Integrity Program described in Appendix B will ensure that the time-dependent parameters (e.g., end-of-life fluence) used in the TLAA remain valid through the period of extended operation. As capsules are pulled and tested, fluence updates and end-of-life vessel fluence extrapolations will be performed. The updated fluence projections will be compared to the 48 EFPY fluence estimates reported in the ANO-2 LRA. If the revised end-of-life fluence extrapolations exceed the values provided in the ANO-2 LRA then the corresponding fracture toughness parameters (adjusted reference temperature (RT), upper shelf energy (USE), and RT_{PTS}) will be updated accordingly.

RAI 4.2-2: Pursuant to 10CFR54.21(d), the Safety Analysis Report (SAR) Supplement for a facility license renewal application (LRA) must contain a summary description for each aging management program and TLAA proposed for management of the effects of aging. The Staff has determined that Appendix A of the LRA (SAR Supplement) did not include a corresponding SAR Supplement summary description for Table 4.2-2 in TLAA 4.2, "Reactor Vessel Neutron Embrittlement" of the LRA. Table 4.2-2 contains an evaluation of reactor vessel extended life for pressurized thermal shock (PTS). The Staff notes that the

corresponding table for the upper-shelf energy extended life evaluation (Table 4.2-1) was included in the SAR Supplement. Pursuant to 10CFR54.21(d), the Staff requests that a corresponding SAR Supplement summary description for LRA Table 4.2-2 be included in the SAR Supplement.

Response: 10CFR54.21(d) requires a SAR Supplement that contains a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs for the period of extended operation. Table 4.2-2 of the LRA provides data for and tabular results of the evaluation of the reactor vessel PTS TLAA for the period of extended operation. Table 4.2-2 is neither the evaluation of the TLAA nor a summary description of the evaluation. Section A.2.2.1.2 of the LRA is provided pursuant to 10CFR54.21(d) as the SAR Supplement summary description of the evaluation of the TLAA for PTS, the results of which are shown in LRA Table 4.2-2.

The ANO-2 SAR does not contain a table equivalent to Table 4.2-2 of the LRA. The limiting beltline material with respect to PTS is identified within the text of SAR Section 5.2.4.3.3, as well as in the proposed SAR Supplement, Appendix A of the LRA. The ANO-2 SAR does contain a table summarizing results of the reactor vessel USE evaluation. Correspondingly, the proposed SAR Supplement includes an update to this table to account for the period of extended operation. However, no equivalent PTS table is required to maintain the current licensing basis as defined in the ANO-2 SAR for the period of extended operation.

RAI 4.7.1-1: In Section 4.7.1 of the LRA, the applicant addresses the RCS piping leak-before-break (LBB) TLAA and concludes that the LBB evaluation for fatigue crack growth remains valid for the period of extended plant operation. How much additional crack growth was predicted by the updated calculations for the end of 60 years compared to that originally predicted for 40 years? What were the criteria or basis for concluding that this amount of additional crack growth was insufficient to exclude fatigue as a damage mechanism that would limit the application of LBB to ANO-2 RCS piping in accordance with the NRC guidance for LBB?

Response: The LBB fatigue crack growth analysis reported in CEN-367-A is based on 40-year design limits for RCS fatigue transient cycles. In CEN-367-A, fatigue crack growth was performed to show that fatigue will not cause degradation of the pressure boundary integrity. In the fatigue crack growth analysis, the normal and upset cyclical loadings cause postulated flaws to grow. These cyclical loadings are based on reactor coolant design transient cycles. As described in Section 4.3.1 of the LRA, the number of transient cycles assumed in the original design for 40 years was found acceptable for 60 years of operation. Therefore, the postulated flaw growth in CEN-367-A (based on the RCS original design transient cycles) is unchanged for 60 years of operation.

RAI 4.7.2-1: In Section 4.7.2 of the LRA, the applicant addresses the RCP Code Case N-481 TLAA. The applicant used fully aged (saturated) properties in the analysis, and concluded that effects of thermal aging on material properties of CASS are addressed for the period of extended operation. Discuss whether these properties are the same bounding properties that were used for embrittled cast stainless materials assumed in the Combustion Engineering (CE) report, CEN-367-A which is an analysis for LBB. If other material properties were used, provide justification for the properties that were used for the Code Case N-481 analysis.

Response: Due to the variety of materials used at the different plants, bounding values from participating plants were used in CEN-367-A for the material properties for stainless steel safe ends. In contrast, the Code Case N-481 evaluation was completed specifically for ANO-2 and thus, used ANO-2 specific material properties for the RCP casings.

RAI 4.7.2-2: In Section 4.7.2 of the LRA, the applicant addresses the RCP Code Case N-481 TLAA and concludes that the evaluation for fatigue crack growth remains valid for the period of extended plant operation. Discuss the additional crack growth that was predicted by the updated calculations at the end of 60 years, and compare the crack growth to that originally predicted for 40 years. Provide the criteria or basis for concluding that this amount of additional crack growth is sufficiently small to justify continued application of Code Case N-481.

Response: The Code Case N-481 calculation was not updated for 60 years. As described in Section 4.7.2 of the ANO-2 LRA, the Code Case N-481 analysis remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i). In the Code Case N-481 evaluation, fatigue crack growth was determined to assure that fatigue will not cause degradation of pressure boundary integrity. In the fatigue crack growth analysis, the normal and upset cyclical loadings cause postulated flaws to grow. These cyclical loadings are based on RCS design transient cycles. As described in Section 4.3.1 of the LRA, the number of transient cycles assumed in the original design for 40 years was found acceptable for 60 years of operation. Therefore, there is no additional crack growth in the postulated flaw when extending the period of operation to 60 years.

RAI 4.7.3-1: In 4.7.3 (Page 4.7-2) of the LRA, the applicant concluded that the RCP flywheel is not a TLAA. The basis for this conclusion is a 1997 safety evaluation of a fatigue crack growth analysis that was presented in a CE Owners Group (CEOG) topical report. This safety evaluation allowed the licensee to lengthen the RCP flywheel inspection period for ANO-1, ANO-2, and five other units. The fatigue crack growth analysis for ANO-1 and ANO-2 is based on 4,000 RCP startup and shutdown cycles. The RCP flywheel was identified as a TLAA in the LRA for ANO-1, and two other units that are identified in the topical report and that have been granted renewed licenses.

Please provide justification why the RCP flywheel is not a TLAA for ANO-2. If the RCP flywheel is a TLAA, provide the TLAA for the RCP flywheel for ANO-2, and include the justification for why 4,000 RCP startup and shutdown cycles remain bounding through the end of the extended period of operation for ANO-2. In addition, the applicant must include a SAR Supplement summary description, in Appendix A, of the LRA for the TLAA on fatigue-induced crack growth of the ANO-2 RCP flywheel. The summary description should include a discussion on the safety margin for the acceptable flaw size, and the justification for why 4,000 RCP startup and shutdown cycles remain bounding through the end of the extended period of operation for ANO.

Response: As defined in 10CFR54.3, TLAAs are those licensee calculations and analyses that involve time-limited assumptions defined by the current operating term, for example, 40 years. The RCP flywheel analysis was based on an assumption of 4,000 startup and shutdown cycles. These 4,000 cycles are not a time-limited assumption defined by the current operating term. Therefore, this analysis does not meet the 10CFR54.3 definition of a TLAA.

RCP startup and shutdown cycles typically occur only in conjunction with RCS heatup or cooldown cycles. As indicated in LRA Table 4.3-1, the allowable number of heatup and cooldown cycles for 60 years of operation is 500. The number of RCP startup and shutdown cycles assumed in the flywheel fatigue crack growth analysis is eight times the number of RCS heatup and cooldown cycles allowed through the period of extended operation.

RAI 4.7.5-1: Demonstrate that the designs of repaired nozzles will have sufficient structural integrity against loss of material by corrosion and will meet their minimum wall thickness requirements through the expiration of the extended period of operation.

Response: Analyses have been completed to estimate the corrosion rate assuming the crevice between the repaired nozzle and underlying ferritic steel will be exposed to aerated borated water. The service lifetimes for repairs to the hot leg pipe nozzles, pressurizer nozzles, and pressurizer heater sleeves are 76, 56, and 196 years, respectively, before ASME Code limits would be exceeded. The most limiting is the service lifetime of 56 years for the pressurizer nozzle repair. The 56 years from the date of the nozzle repair extends the service life beyond the period of extended operation. Therefore, loss of material by corrosion will not impair the ability of the repaired nozzles to maintain sufficient structural integrity for the period of extended operation.

RAI 4.7.5-2: Justify and validate the CEOG's conclusion that growth of the existing flaw in the original Alloy-600 J-groove weld material by stress corrosion cracking is not a plausible effect during the period of extended operation.

Response: The repaired nozzles will have cracks in the Alloy-600 nozzles or the partial penetration attachment welds remaining after completion of the repair. Since residual stresses from the welding will remain, these cracks may continue to propagate through the nozzle/weld metal by SCC to the carbon or low-alloy steel base metal. Further growth into the carbon or low-alloy steel base metal is unlikely since low oxygen levels due to pressurized water reactor (PWR) water chemistry will result in corrosion potentials below critical cracking potential in a high temperature environment. As described in Section B.1.30.3 of the LRA, the ANO-2 Primary Water Chemistry Control Program is based on the Electric Power Research Institute (EPRI) TR-105714 which requires stringent oxygen controls. This program will continue into the period of extended operation. Therefore, the ANO-2 Primary Water Chemistry Program will provide an environment which limits the corrosion potential of the applicable material below the critical cracking potentials and SCC will not cause growth of the existing flaw.

RAI B.1.1-1: Confirm that all of the components listed in the Alloy-600 Program are covered under the ISI requirements of Section XI of the ASME Code, and for any components not covered by Section XI ISI requirements, provide a complete description of the proposed inspections including a technical justification for the inspection method and frequency.

Response: All nickel-based alloy items listed in Section B.1.1 of the LRA are included in the ANO-2 ISI Program with the exception of the thermal sleeves, the cladding on pressurizer lower head, reactor vessel lower shell and head, and steam generator tubesheet, the steam generator channel head divider plate and primary nozzle rings, and the pressurizer heater support plates and heater support plate brackets.

The items that are inspected as part of the ANO-2 ISI Program have a greater susceptibility to PWSCC due to physical configuration or operational conditions (e.g., temperature) than those listed above. Therefore, the Alloy-600 or Alloy-690 and Alloy-82/182 or Alloy-52/152 items listed above that are not volumetrically or visually inspected are bounded by the items that receive examinations in accordance with ASME Section XI. In addition, the EPRI Materials Reliability Program (MRP) in conjunction with the PWR owners groups is developing a strategic plan to manage PWSCC of nickel-based alloy components. Guidance developed by the MRP and the owners groups will be used to identify the need for augmenting existing ISI inspections at ANO-2 where appropriate.

RAI B.1.1-2: The applicant stated that no preventative actions will be taken as part of the Alloy-600 Program to prevent aging effects or mitigate aging degradation. The NRC Staff notes that several preventive actions and common industry practices have been used to mitigate PWSCC associated with Alloy-600. Examples of these include: nickel plating of the surfaces of Alloy-600 components that are exposed to treated water, replacement of leaking Alloy-600 instrument nozzles with Alloy-690 material, preventive replacement of selected pressurizer and RCS piping instrument nozzles with Alloy-690 material, monitoring the electrochemical potential, and water chemistry control. Provide a description of any preventive actions and/or water chemistry monitoring programs ANO-2 is currently implementing that may be used to address the Alloy-600 cracking issue.

Response: ANO-2 has taken preventive actions to address the Alloy-600 cracking issue, however, these actions are not part of the Alloy-600 Program. Various Alloy-600 pressurizer heater sleeves, instrument nozzles, and hot leg instrument nozzles have been repaired due to PWSCC. The repairs involve one of two methods, both of which remove Alloy-600 material from a pressure boundary function. One repair method replaces the Alloy-600 nozzles with Alloy-690 nozzles while the other utilizes no nickel-based alloy material in a pressure boundary role (mechanical nozzle seal assemblies). Alloy-690 demonstrates strong resistance to PWSCC and is an industry standard for replacement of Alloy-600 components. In addition, the ANO-2 Water Chemistry Control Program rigorously controls contaminants known to contribute to PWSCC. As described in Section B.1.30.3 of the LRA, the ANO-2 Primary Water Chemistry Control Program is based on EPRI TR-105714.

Therefore, ANO-2 does take preventive actions to mitigate degradation of Alloy-600 components. These actions are consistent with industry practice and include maintenance of stringent water chemistry controls in accordance with industry accepted guidelines and replacement of faulty Alloy-600 components with materials significantly less susceptible to PWSCC.

RAI B.1.1-3: In the Alloy-600 Program under the program attribute, "Detection of Aging Effects," the applicant states that the measurement, vent, upper level, and temperature nozzles, and heater sheath, heater sleeve, and end plug received visual examination (VT-2) from the exterior of the vessel in accordance with ASME Section XI, Examination Category B-P. For many of these components, the Alloy-600 pressure boundary welds are covered by insulation. Service experience has shown that, early indications of through-wall leakage (e.g., boric acid on the component surface) are very difficult to detect when VT-2 examinations are performed with the insulation in place. Provide justification for not removing insulation when performing VT-2 examinations on the components mentioned

above. In addition, provide the frequency of inspection, and the results of any volumetric non-destructive examination that has been performed.

Response: As described in Section B1.1.4 on page B-12 of the LRA, the ANO-2 pressurizer heater and small-bore nozzles are visually inspected in addition to the ASME Section XI Examination Category B-P inspections. Insulation is removed if required to allow for bare metal examination of an area 360° around the small nozzles and penetrations for evidence of boric acid residue. The inspections are performed each refueling outage.

RAI B.1.1-4:

A. In the Alloy-600 Program under the program attribute, "Detection of Aging Effects," the applicant states that "guidance from the MRP in conjunction with the PWR owners groups will be used to identify critical locations for inspection and augmentation of existing ISI inspections at ANO-2 where appropriate." The Staff believes that the strategic plan developed by the industry will be comprehensive and recommendations may be applicable to all 10 elements of the Alloy-600 Program. Identify the date that ANO-2 commits to submit, for review and approval, an augmented aging management program that includes all recommendations from the industry's strategic plan, and meets the 10 elements in accordance with the guidance in NUREG-1800, Appendix A.1, "Aging Management Review - Generic," Table A.1-1, "Elements of an Aging Management Program for License Renewal." The date must be prior to the period of extended operation.

Response: PWSCC of nickel-based alloys is a current license term issue. As such, interaction between Entergy and the NRC Staff is ongoing to develop a program to manage the effects of aging due to this mechanism. In accordance with the SOC, issues that are relevant to current plant operation will be addressed by the existing regulatory process within the present license term rather than deferred until the time of license renewal. Consequently, the existing regulatory process provides assurance that aging effects caused by PWSCC of nickel-based alloys will be adequately managed during the period of extended operation. Consistent with all programs credited for license renewal at ANO-2, the Alloy-600 Program will be available on-site for NRC review.

RAI B.1.1-4:

B. The SAR for ANO-2 does not contain a commitment to use guidance developed by the EPRI MRP, and to submit the inspection plan for review and approval. Confirm that the SAR will be revised to reflect the above mentioned commitment for management of Alloy-600 components.

Response: The following commitment will be added to the SAR Supplement: During development of the ANO-2 Alloy-600 Program, guidance developed by the EPRI MRP for the selection, inspection, and evaluation of nickel-based alloy items will be considered.

RAI B.1.1-5: In the Alloy 600 Program under the program attribute, "Operating Experience," the applicant states that the Alloy-600 Program is a new program for which there is no specific operating experience for ANO-2. The Staff is aware of several reported failures related to Alloy-600 welded components in other PWRs including several failures in other CE NSSS design. Specifically, PWSCC has been reported in Alloy-82/182 J-groove welds that are used to join Alloy-600 small bore nozzles to CE-designed pressurizers, steam generators, and/or hot legs. The Staff believes it important for the applicant to

review relevant industry service experience and incorporate lessons learned into the Alloy-600 Program. Therefore, the applicant should discuss what industry initiatives it plans to follow in order to incorporate experience related to Alloy-600 into the ANO-2 Alloy-600 Program.

Response: As defined in the Standard Review Plan (NUREG-1800), the Operating Experience program element describes the operating experience of the aging management program, including past corrective actions resulting in program enhancements or additional programs. As a new program, the ANO-2 Alloy-600 Program has no relevant operating experience as defined in NUREG-1800.

In Section B.1.1.4 of the ANO-2 LRA it states: "The EPRI MRP in conjunction with the PWR owners groups is developing a strategic plan to manage PWSCC of nickel-based alloy components. Guidance developed by the MRP and the owners groups will be used to identify critical locations for inspection and augmentation of existing ISI inspections at ANO-2 where appropriate." In addition, the existing regulatory process provides assurance that the ANO-2 Alloy-600 Program will incorporate industry guidance and relevant experience.

RAI B.1.3-1: Provide the basis for the proposed acceptance criteria that will be developed as part of the following enhancement to the Boric Acid Corrosion Prevention Program:

> The program acceptance criteria will be revised to address electrical components in addition to ferritic steel.

Response: NUREG-1801 will be the basis for the acceptance criteria for electrical components exposed to boric acid. In accordance with "Acceptance Criteria" of NUREG-1801, Section XI.M10, acceptance criteria will be the absence of any detected leakage or crystal buildup. If identified during inspections, evidence of leakage or crystal buildup will be evaluated to determine the need for corrective actions prior to continued service. The acceptance criteria will apply to electrical components as well as ferritic steel components.

RAI B.1.3-2: In the Operating Experience Section of B.1.3, Boric Acid Corrosion Prevention, the applicant states that recent industry events regarding reactor vessel head degradation required assessments at each site to ensure that boric acid corrosion prevention programs are adequate and functioning effectively. The applicant also states that a self assessment was performed in February 2003, and no significant findings were identified during this assessment. Discuss how program revisions have incorporated lessons learned from the Davis Besse vessel head degradation and the control rod drive mechanism penetration cracking discussed in NRC Bulletins 2001-01, 2002-01, 2002-02, and NRC Order EA-03-009. Also, provide a discussion on implementation of corrective actions in the program to prevent reoccurrence of degradation caused by boric acid leakage, as required by Generic Letter 88-05.

Response: The Boric Acid Corrosion Prevention Program solely addresses the loss of material of carbon and low-alloy steel components exposed to a treated (borated) water environment. The assessment performed in 2003 concluded that the ANO-2 Boric Acid Corrosion Prevention Program was sufficient to detect loss of material by boric acid wastage of the reactor vessel head in the event of leaking control element drive mechanism

(CEDM) penetrations. However, ANO-2 does not rely on leak detection through the Boric Acid Corrosion Prevention Program to manage cracking of CEDM penetrations. The ANO-2 Reactor Vessel Head Penetration Program described in Section B.1.20 of the LRA addresses reactor vessel head degradation and CEDM penetration cracking as discussed in the referenced NRC bulletins and NRC Order EA-03-009. Measures taken in response to NRC Order EA-03-009 and its successors carry forward into the period of extended operation.

As described in Section B.1.3 of the LRA, the ANO-2 Boric Acid Corrosion Prevention Program is consistent with NUREG-1801, Section XI.M.10. As such, the ANO-2 corrective actions for this program are consistent with NUREG-1801. The program is consistent with the ANO-2 commitments in response to NRC Generic Letter 88-05. ANO-2 applies the requirements of 10CFR50, Appendix B to the Boric Acid Corrosion Prevention Program through the ANO-2 Corrective Action Program.

Boric acid corrosion is a current license term issue. As such, interaction between Entergy and the NRC Staff is ongoing to develop a program to manage the effects of aging due to this mechanism. In accordance with the SOC, issues that are relevant to current plant operation will be addressed by the existing regulatory process within the present license term rather than deferred until the time of license renewal. Consequently, the existing regulatory process provides assurance that ongoing operating experience will be incorporated into the Boric Acid Corrosion Prevention Program as appropriate.

RAI B.1.20-1: The applicant states that the Corrective Action Program was used to incorporate industry operating experience into the Reactor Vessel Head Penetration Program, and to develop inspection requirements that are specific to ANO-2. The applicant also states that recent reactor vessel head penetration nozzle inspections were performed in accordance with the commitments in the ANO-2 response to NRC Bulletin 2001-01. The NRC Staff notes that in February 2003, NRC Order EA-03-009 was issued. This order supercedes NRC Bulletins 2001-01 and 2002-01, and requires that licensees assess the susceptibility of the reactor vessel head to PWSCC-related degradation. The Order also requires the licensee to commit to an Augmented Inspection Program for the reactor pressure vessel head based upon the susceptibility to PWSCC. Provide a commitment for implementation of the ANO-2 augmented inspection plan in accordance with NRC Order EA-03-009 for the period of extended operation and confirm that the SAR will be revised to reflect the above mentioned commitment for inspection of the reactor pressure vessel head.

Response: Order EA-03-009 required ANO-2 to implement an augmented inspection program for the reactor pressure vessel head based upon its susceptibility to PWSCC. This confirms that as noted in the SOC, issues that are relevant to current plant operation are being addressed by the existing regulatory process within the present license term rather than deferred until the time of license renewal. Consequently, the existing regulatory process provides assurance that ongoing interaction between Entergy and the NRC Staff is occurring to ensure appropriate measures are included in the Reactor Vessel Head Penetration Program in response to NRC Order EA-03-009 and subsequent relevant industry experience and regulatory requirements.

As stated in Section A.2.1.21 of the LRA SAR Supplement, the ANO-2 Reactor Vessel Head Penetration Program identifies both visual and volumetric examination in accordance with the requirements of NRC Order EA-03-009 and will be modified as appropriate to

implement evolving commitments in response to industry experience and regulatory requirements.

RAI B.1.21-1: The description of this aging management program does not include a specific reactor vessel specimen capsule withdrawal schedule for the period of extended operation. Please provide a specific schedule through the end of the period of extended operation for Staff review. In addition, please revise the SAR Table 5.2-12 accordingly.

Response: The capsule withdrawal schedule is included as part of changes to the existing SAR in Table 5.2-12 of Section A.1, page A-8 of the ANO-2 LRA. Capsule 3 is scheduled to be removed at 30 EFPY. It is estimated that this capsule will receive approximately 4.9E19 n/cm^2 at 30 EFPY, which is slightly less than the expected 48 EFPY fluence of 5.0E19 n/cm^2 discussed in the ANO-2 LRA, Section 4.2.2. As discussed in Section B.1.21 of the ANO-2 LRA, the ANO-2 specimen capsule withdrawal schedule will be revised to withdraw and test a standby capsule to cover the peak fluence expected through the end of the period of extended operation. As specified in Note (a) to Table 5.2-12 in the ANO-2 LRA, if required, Capsules 4, 5, or 6 will be repositioned to ensure that peak fluence is obtained prior to 60 years. Alternatively, Entergy may decide to delay the withdrawal of Capsule 3 to cover the period of extended operation and would at that time notify the NRC of the change to the withdrawal schedule as required by 10CFR50 Appendix H.

RAI B.1.22-1: The Reactor Vessel Internals CASS Components Program is currently not in place. The applicant states in LRA Section B.1.22, that the aging management program will be consistent with NUREG-1801 (Generic Aging Lessons Learned (GALL)), and that it will initiate the program prior to the period of extended operation. Management of the aging effects associated with void swelling of PWR reactor vessel internals is not included in the GALL report. The Staff requests that the applicant formally make a commitment to participate in industry initiatives, and to implement industry recommendations regarding void swelling when they become available. The Staff also requests that the applicant submit the inspection program to manage the aging effects associated with void swelling to the NRC for review and approval no later than three years prior to the period of extended operation.

Response: The Reactor Vessel Internals CASS Program will manage the effects of distortion due to void swelling. This program will provide visual inspections and non-destructive examinations of the reactor vessel internals during the period of extended operation. In addition, the investigation of the internals aging effects through the activities of EPRI and other industry groups focused on reactor vessel internals will ensure a better understanding of void swelling and other aging effects. The results of these investigations will be considered when developing the ANO-2 Reactor Vessel Internals CASS Program.

As described in the LRA, the Reactor Vessel Internals Program is a new program to be developed prior to entering the period of extended operation. Consistent with all programs credited for license renewal at ANO-2, the Reactor Vessel Internals CASS Program, once developed, is available on site for NRC review.

RAI B.1.22-2: The Staff found some differences between Table 3.1-2 in NUREG-1800 (Standard Review Plan for License Renewal) and the ANO-2 LRA SAR Section A.2.1.23, Reactor Vessel Internals CASS Program. Table 3.1-2 in the Standard Review Plan provides a description of what should be included in the SAR Supplement for aging

management of reactor vessel internals, and the RCS for license renewal reviews. The SAR should state that the ISI Program will be augmented to include enhanced examinations of non-bolted components, and other demonstrated acceptable methods for bolted components for certain susceptible or limiting components or locations. Clarify why the enhanced examination and/or component-specific flaw evaluation for the CASS component, which are specified in NUREG-1800, are not included in ANO-2 LRA SAR Section A.2.1.23.

Response: The Reactor Vessel Internals CASS Program will be consistent with NUREG-1801 XI.M13, Thermal Aging and Neutron Irradiation Embrittlement of CASS. In Table 3.1-2 of NUREG-1800, the description of this program states, "The program consists of (1) determination of the susceptibility of CASS components to thermal aging embrittlement, (2) accounting for the synergistic effects of thermal aging and neutron irradiation, and (3) implementing a supplemental examination program, as necessary." As described in Section A.2.1.23 of the LRA, the ANO-2 Reactor Vessel Internals CASS Program will: "...manage aging effects of CASS reactor vessel internals components. This program will supplement the reactor vessel internals inspections required by the ASME Section XI ISI Program. The program will manage cracking, reduction of fracture toughness, and dimensional changes using inspections of applicable components which will be determined based on the neutron fluence and thermal embrittlement susceptibility of the component." Bolting is not addressed by the program summarized in A.2.1.23 since ANO-2 has no CASS reactor vessel internals bolting. Therefore, the description of the ANO-2 Reactor Vessel Internals CASS Program in LRA Section A.2.1.23 is consistent with the program description in NUREG-1800.

RAI B.1.23-1: This Reactor Vessel Internals' Stainless Steel Plates, Forgings, Welds, and Bolting Program is currently not in place. The applicant states in LRA Section B.1.23, that the aging management program will be consistent with NUREG-1801, and that it will initiate the program prior to the period of extended operation. Management of the aging effects associated with void swelling of PWR reactor vessel internals is not included in the GALL report. The Staff requests that the applicant formally make a commitment to participate in industry initiatives, and to implement industry recommendations regarding void swelling when they become available. The Staff also requests that the applicant submit the inspection program to manage the aging effects associated with void swelling to the NRC for review and approval no later than three years prior to the period of extended operation.

Response: The Reactor Vessel Internals' Stainless Steel Plates, Forgings, Welds, and Bolting Program will manage the effects of distortion due to void swelling. This program will provide for visual inspections and non-destructive examinations of the reactor vessel internals during the period of extended operation. In addition, investigation of the internals aging effects through the activities of EPRI and other industry groups will ensure a better understanding of void swelling and other aging effects. The results of these investigations will be considered during development of the ANO-2 Reactor Vessel Internals' Stainless Steel Plates, Forgings, Welds, and Bolting Program.

As described in the ANO-2 LRA, the Reactor Vessel Internals Program is a new program to be developed prior to entering the period of extended operation. Consistent with all programs credited for license renewal at ANO-2, the Reactor Vessel Internals Program, once developed, is available on site for NRC review.

RAI B.1.23-2: The Staff noted that the water chemistry system and the enhanced examination of non-bolted components are not discussed in LRA SAR Supplement, Section A.2.1.24 as is stated in NUREG-1800, Table 3.1-2, page 3.1-27. Clarify why the water chemistry system and the enhanced examination of non-bolted components, which are specified in NUREG-1800, are not included in ANO-2 LRA SAR Section A.2.1.24.

Response: Control of ANO-2 primary water chemistry in accordance with the appropriate EPRI guidelines is discussed in the Water Chemistry Control Program section of the SAR Supplement, Section A.2.1.33. As indicated in LRA Table 3.1.2-2, the Water Chemistry Control Program applies to reactor vessel internals items.

As stated in LRA Section B.1.23, the ANO-2 program will be consistent with NUREG-1801 XI.M16, *PWR Vessel Internals.* As such, the program will include enhanced VT-1 examinations unless component-specific evaluation determines the loading is compressive or is low enough to preclude fracture. Enhanced VT-1 is defined in NUREG-1801, XI.M16, as the ability to achieve a 0.0005-inch resolution. However, this visual acuity requirement is based on boiling water reactor research, which may not be applicable to pressurized water reactor (PWR) items. As such, ANO-2 will utilize enhanced VT-1 for additional inspection of RVI items unless additional research indicates that a different visual acuity requirement is appropriate.

Attachment 2

2CAN070405

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (Check One)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
In addition, the EPRI Materials Reliability Program (MRP) in conjunction with the PWR owners groups is developing a strategic plan to manage PWSCC of nickel-based alloy components. Guidance developed by the MRP and the owners groups will be used to identify and the need for augmenting existing ISI inspections at ANO-2 where appropriate.		X	July 17, 2018
The following commitment will be added to the SAR Supplement: During development of the ANO-2 Alloy-600 Program, guidance developed by the EPRI MRP for the selection, inspection, and evaluation of nickel-based alloy items will be considered.	X		Upon issuance of renewed license