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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 May 5, 2000 (1CNA050002), June 1, 2000 (1CNA060002), and June 23, 2000 (1CNA060006), 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 structural Sections 3.1.3, 3.6, 4.5, and 4.6 of the ANO-1 LRA. Should you have any further questions, please contact me.

Very truly yours,

Jimmy D. Vandergrift
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Attachment

A082

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Request for Additional Information Regarding ANO-1 LRA Structural Sections 3.1.3, 3.6, 4.5, and 4.6 dated May 5, 2000 (1CNA050002), June 1, 2000 (1CNA060002), and June 23, 2000 (1CNA060006),

3.3.1.3-1 **Provide additional description of the criteria for assessing or categorizing the overall condition of the structures and components that are monitored as part of the Maintenance Rule aging management program (AMP) (Appendix B, Section 4.13). Include specific examples such as indications of spalling or cracking on concrete surfaces; corrosion, peeling paint, or excessive deflection of structural steel components; and changes in material properties of teflon.**

The Maintenance Rule Program inspection criteria used to assess and categorize the condition of structural components include the following:

Concrete is inspected for spalling (>1"in depth), cracking (>1/16" in width), exposed rebar, water in-leakage, chemical leaching, peeling paint, and discoloration.

Masonry walls are inspected for cracks, deteriorated penetrations and missing or broken blocks.

Structural steel is inspected for flaking rust, widespread corrosion (>1/32" in depth), deteriorated coatings, beam/column deflection, loose or missing fasteners or support items, support misalignment, degradation of close tolerance machined or sliding surfaces (i.e., Teflon), missing grout beneath base plates, and pitting (>1/32" in depth).

3.3.1.3-2 **Proactive monitoring and understanding of trending behavior is needed to monitor structural aging to allow corrective actions to be taken prior to exceeding acceptance criteria. Describe the monitoring and trending activities that are used by the Maintenance Rule AMP (Appendix B, Section 4.13) to track the extent and rate of degradation and their relationship to the acceptance criteria.**

The Maintenance Rule Program uses standardized monitoring and trending activities to track degradation. Technical requirements for performing structural condition monitoring, evaluating against acceptance criteria, and performing corrective actions in accordance with 10CFR50.65 are described in an ANO engineering standard. Deficiencies are documented so that results can be trended. In addition to preparing a written description and noting the location, this may also include collecting measurements to determine the severity of deterioration, taking photographs, or drawing sketches. When

structures or components do not meet the acceptance criteria, corrective actions (i.e., condition report, repair/replacement) are initiated and tracked in accordance with the site 10CFR50 Appendix B Corrective Action Program. Acceptance criteria are typically established such that corrective actions are initiated prior to loss of function.

3.3.1.3-3 The description of the frequency of structural and component walk-downs for the Maintenance Rule AMP (Appendix B, Section 4.13) states only that, "Structural and component walk-downs are performed periodically, and the frequency varies depending on the structure or component being inspected." Describe the method(s) used to determine the frequency of walk-downs as well as the minimum walk-down frequency for the different applications of the structural and component walk-downs.

Methods used to determine the frequency of structural walk-downs for the Maintenance Rule Program were derived from reviewing general plant monitoring activities and procedures, pre-existing structural condition assessments, ANO and industry wide operating experience (i.e., NUREG-1526 "Lessons Learned from Early Implementation of the Maintenance Rule at Nine Nuclear Power Plants"), and pre-defined inspection frequencies for structures under existing programs.

The frequency of structural walk-downs is in accordance with the ANO-1 current licensing basis (CLB) and will be adjusted as necessary, based on information gained from on-going, site-specific or industry experience. The current periodic walk-down frequency for in-scope structures is once per five years.

3.3.1.3-4 The description of the acceptance criteria for the Maintenance Rule AMP Appendix B, Section 4.13) states only that, "No unacceptable visual indications of cracking, loss of material, or change in material properties of structures or components." For each commodity group (e.g., concrete, steel, coatings) provide a description of the criteria that are used to (1) assess the severity of the observed degradations and (2) determine whether corrective action is appropriate. Discuss any reliance on reference documents such as ANSI/ASCE 11-90, "Guidelines for Structural Condition Assessment of Existing Buildings", and ACI-349.3R-96, "Evaluation of Existing Nuclear Safety-Related Concrete Structures."

Three guidelines are used to evaluate the acceptability of structural components and commodities. Those that have no deficiencies or deficiencies

that would not lead to loss of function are classified as "acceptable". Those components and commodities which are capable of performing their intended function, but that are degraded or have deficiencies which could deteriorate to an unacceptable condition if not corrected, are classified as "acceptable with deficiencies". The third classification of "unacceptable" applies to structural components and commodities that are damaged or degraded and not capable of performing their intended function.

The guidelines used to determine whether corrective action is appropriate for the different material groups is also the same. Potentially degraded conditions are reported to the civil engineer responsible for the Maintenance Rule. The responsible engineer performs an assessment of the degradation in accordance with 10CFR50.65 and implements appropriate actions as necessary for resolution

NUREG-1522 "Assessment of Inservice Conditions of Safety Related Nuclear Plant Structures", NUREG-1526 "Lessons Learned from Early Implementation of the Maintenance Rule at Nine Nuclear Power Plants", and Nuclear Energy Institute (NEI) 96-03 "Guideline for Monitoring the Condition of Structures at Nuclear Power Plants" were used in developing the Maintenance Rule Program. ACI 349.3 is a reference document for both NUREG-1522 and NEI 96-03. In addition, the engineering standard for performing Maintenance Rule structural monitoring relies on general practices and criteria established in the ACI Manual of Concrete Practice and the AISC Manual of Steel Construction.

3.3.1.3-5 The description in Appendix B, Section 4.13, of the Maintenance Rule AMP does not provide a description of the findings of the Maintenance Rule baseline inspection and subsequent Maintenance Rule inspection activities. Identify any aging identified prior to loss of intended function or failures not detected prior to loss of intended function. In addition, indicate whether these findings have been used to enhance or improve the Maintenance Rule AMP in order to show that this AMP will adequately manage the effects of aging during the period of extended operation.

Indications of aging identified during the structural baseline inspection included some areas with exposed rebar which was slightly rusted, minor water inleakage, and numerous concrete surface cracks, which did not exceed the acceptance criteria. Indications of degradation were assessed as requiring no action or a request for repair was initiated. None of the findings resulted in a loss of a component's intended function. No loss of intended function has occurred for a component from degradation not detected during the baseline inspection.

The program will continue to focus on detection and repair prior to loss of component function. Program adjustments will be made based on information gained from ongoing, site-specific and industry experience.

3.3.1.3-6 Provide a description of the training and qualifications of the personnel that (1) perform the Maintenance Rule AMP (Appendix B, Section 4.13) structure and component walk-downs and (2) evaluate the adequacy of the walk-down procedures and findings.

Structural monitoring conducted under the Maintenance Rule Program requires that structure and structural component walk-downs be performed by qualified engineers. ANO personnel are qualified in accordance with the ANO Engineering Support Personnel (ESP) Program that provides assurance of an appropriate level of knowledge and experience prior to performing engineering activities. The ANO ESP Program is an accredited INPO training program.

The engineer responsible for Maintenance Rule assessment of structures and supports evaluates the identified findings. This person is also trained in accordance with the ESP Program. As such, he/she is qualified to perform a preliminary assessment of degradation in accordance with 10CFR50.65. The Maintenance Rule Program walk-down guidelines have been approved in accordance with the site 10CFR50 Appendix B Program.

3.3.1.3-7 Describe the provisions of the Maintenance Rule AMP (Appendix B, Section 4.13) for inspecting normally inaccessible structures and components.

The ANO-1 Maintenance Rule Program provides for walk-downs of accessible areas and normally inaccessible areas that become accessible with changing plant conditions (i.e., high radiation areas). If findings on accessible structures or components indicate that potential degradation may be occurring in an inaccessible area, an evaluation will be performed. The aging management review did not identify unique aging effects for inaccessible structures and components. Thus, inspection of the accessible structures and components is a representative sample of both accessible and inaccessible structures and components.

3.3.1.3-8 Describe the walk-down procedures, checklists, or inspection forms, if any, that are provided to the personnel that perform the structure and component walk-downs as part of the Maintenance Rule AMP (Appendix B, Section 4.13).

Checklists are provided to qualified personnel performing the structural walk-downs. Various indications of potential degradation are listed on the checklists. Refer to the RAI response 3.3.1.3-1. Deficiencies are documented on program specific report forms.

3.3.6-1 ANO-1 LRA Section 2.4 – Structures and Structural Components does not specify expansion joint sealants, structural sealants or caulking as components in the license renewal scope. The only reference to expansion joint sealant is in Section 3.6.1.1, where degradation of the joint sealant is identified as a pre-condition for loss of material of the containment liner plate below the floor. In the staff guidance regarding consumables (see license renewal Issue No. 98-0012, "Consumables," dated 4/20/99), when these non-metallic components are part of the structures and structural components within the scope of license renewal, they are also considered to be within the scope of license renewal and subject to an aging management review. In addition, IWE Examination Category E-D specifically addresses inspection of these non-metallic components for containment. Therefore, provide the following information:

(a) Is IWE Examination Category E-D credited for license renewal to manage aging of expansion joint sealants, structural sealants, and caulking for the reactor building?

Yes, the IWE Examination Category E-D (Seals, Gaskets, and Moisture Barriers), in addition to the Maintenance Rule Program, manages the aging of in-scope expansion joint sealant, structural sealant and caulking for the reactor building. Similar to the discussion in NUREG-1723, expansion joint sealant, structural sealant and caulking are considered to be parts of structural components or commodities within the scope of license renewal, important in maintaining the integrity of the component or commodity to which they are connected.

(b) If not, identify the alternate AMP, and where in the LRA is this program specifically discussed with respect to expansion joint sealants, structural sealants, and caulking for the reactor building.

Referring to the RAI response 3.3.6-1(a), aging of the reactor building expansion joint sealant, structural sealant and caulking within the scope of license renewal is managed by IWE Examination Category E-D and the

ANO-1 Maintenance Rule Program. ANO-1 LRA Appendix B, Sections 4.3.4 and 4.13, did not specifically discuss expansion joint sealant, structural sealant or caulking since they are considered parts of components or commodities that are managed by these programs.

(c) **For structures and structural components other than the reactor building, describe the AMP which is credited to manage aging of these non-metallic components consistent with the 10 elements in the Standard Review Plan in sufficient detail to allow the staff to assess the adequacy of this program to manage the applicable aging effects.**

Non-metallic items such as expansion joint sealant, structural sealant and caulking associated with in-scope structural components and commodities other than the reactor building are managed by the same aging management programs that manage the aging effects on components or commodities to which expansion joint sealant, structural sealant and caulking are attached. The primary aging management program is the Maintenance Rule Program. Elements of this program are described in ANO-1 LRA Appendix B, Section 4.13.

(d) **If no AMP is credited, provide a justification for excluding the structural sealants within the scope of license renewal from being subject to an aging management review.**

Refer to the RAI response 3.3.6-1(c).

3.3.6-2 Section 2.4.1.1 of the ANO-1 LRA indicates that attachment welds to the liner plate are included in Reactor Building Internal Structural Components. However, welds of integral attachments to the liner plate are included within the scope of American Society of Mechanical Engineers (ASME) Section XI, Subsection IWE. On this basis, provide the following information:

(a) **The liner plate welds have a pressure boundary intended function as well as a structural support intended function. Discuss why the liner plate welds were grouped with the Reactor Building Internal Structural Components instead of the reactor building, and discuss how the AMPs selected for the reactor building internal structural components will manage the aging effects that may effect the pressure boundary intended function.**

Liner plate attachment welds are not considered part of the pressure-retaining boundary of the reactor building; they are considered to be surface welds. This position is consistent with the jurisdictional

boundary for these welds defined by ASME Section XI Subsection IWE. As stated in Section 2.4.1.1 of the ANO-1 LRA, attachment welds are within the evaluation boundary of the reactor building internals since the evaluation boundary of the reactor building is at the inside surface of the liner plate. However, since the liner plate is a reactor building component and ASME Section XI ISI-IWE manages aging of the liner plate and liner plate attachment welds, for program grouping purposes, the attachments were listed in both Table 3.6-2 (reactor building) and Table 3.6-3 (reactor building internals).

(b) What is the AMP that manages aging of attachment welds to the liner plate?

The aging management programs that manage aging of attachment welds to the liner plate are ASME Section XI, IWE Inspections and the Maintenance Rule Program.

(c) Describe the AMP program for the attachment welds to liner plate consistent with the 10 elements in the Standard Review Plan in sufficient detail to allow the staff to assess the adequacy of this program to manage the applicable aging effects and discuss if the inspection requirements are equivalent to or more stringent than the requirements of IWE?

If inspection activities are less stringent, provide a technical justification for relaxation of the IWE requirements.

For descriptions of the aging management programs for the attachment welds to the liner plate, refer to the ANO-1 LRA, Appendix B, Section 4.3.4 and Section 4.13. Inspection requirements of the ASME Section XI, IWE Inspections are equivalent to the requirements of ASME Section XI IWE.

3.3.6-3 **Section 2.4.1.1 of the ANO-1 LRA states that surveillance requirements for gears, latches, linkages, etc. of both the larger personnel hatch and the smaller emergency hatch are included in the ANO-1 Technical Specifications. Identify where in the LRA is fretting and lockup of hinges, locks and closure mechanisms for personnel hatches discussed, or provide a technical justification for not considering fretting and lockup as applicable aging effects for these components. If these aging effects are determined to applicable, identify where in the LRA or provide a description of the AMP for the personnel hatches consistent with the 10 elements in the Standard Review Plan in sufficient detail to allow the staff to assess the adequacy of this program to manage the applicable aging effects.**

As also stated in Section 2.4.1.1, since the hatch operating mechanisms, which include gears, latches, hinges, and linkages, perform their intended function with moving parts and with a change of configuration, they are not subject to an aging management review. Therefore, aging effects for hatch operating mechanisms, such as fretting and lockup, were not considered.

3.3.6-4 **Section 2.4.1.1 of the ANO-1 LRA states that the seals of each personnel hatch and the equipment hatch in the reactor building are not long-lived, passive components and do not require an aging management review because they are replaced when warranted by their condition. 10CFR54.21(a)(1)(ii) states that structures and components that are not subject to replacement based on a qualified life or specified time period are subject to an aging management review.**

(a) **Provide a technical justification consistent with the requirements of 10CFR54.21(a)(1)(ii) to support the determination that seals for personnel and equipment hatches need not be subject to an aging management review based on the seals not being long-lived.**

Although within the scope of license renewal, the personnel and equipment hatch seals are considered short-lived. As stated in NEI's March 26, 1999 letter to Mr. Christopher Grimes of the NRC, "Industry Paper on Structural Monitoring – License Renewal Generic Topic 98-057", non-metallic items (i.e., seals) have "a useful life-time of five to ten years" and are "routinely inspected and replaced in accordance with the manufactures recommendations". Because the personnel and equipment hatch seals are inspected and replaced during preventive maintenance activities, the seals are not subject to an aging management review.

(b) **If the personnel and equipment hatch seals are determined to be subject to an aging management review identify where in the LRA is**

the aging management review, or provide an aging management review (AMR) and a description of the AMPs consistent with the 10 elements in the Standard Review Plan in sufficient detail to allow the staff to assess the adequacy of this program to manage the applicable aging effects.

As noted in the RAI response.3.6-4(a), the personnel and equipment hatch seals were not subject to an aging management review. However, since the seals are piece parts of the personnel and equipment hatches, their inspection and replacement is performed during preventive maintenance activities conducted under the aging management programs that are credited for managing aging effects for the hatches. Referring to Table 3.6-2 of the ANO-1 LRA, the aging management programs that manage the personnel and equipment hatches are ASME Section XI IWE Inspections, the Maintenance Rule Program, and Reactor Building Leak Rate Testing. These programs govern the replacement of the personnel and equipment hatch seals. Elements of these programs are discussed in Appendix B of the ANO-1 LRA, Sections 4.3.4, 4.13 and 4.16, respectively.

3.3.6-5 In Section 2.4.1.2 of the ANO-1 LRA, the Elastomeric Silicone Rubber Coating of the reactor building dome is identified as providing protection for the dome from weathering conditions. In the past, when protective coatings have been credited for eliminating plausible aging effects, the coating itself would be included within the scope of license renewal and subject to an AMR. The application does not discuss whether the reactor building dome coating is "credited" for protecting the dome or if this coating is included within the license renewal scope.

(a) If damage to the coating has been identified, what AMP will be used to address the loss of material and resulting defects to the reactor building dome?

The coating is not within the scope of license renewal. It has no intended function. If damage to the reactor building dome's elastomeric roofing is identified, loss of the roofing membrane and potential resulting defects to the reactor building dome will be addressed by ASME Section XI IWL Inspections and the Maintenance Rule Program. As indicated in Table 3.6-2, these programs are credited with managing the aging of the reactor building dome.

(b) Based on this information, identify where the AMR of the reactor building dome coating can be found in the LRA, provide a technical

justification for not managing the aging of the reactor building dome coating, or perform an AMR of the reactor building dome.

The protective coating (elastomeric roof membrane) is not a system, structure, or component within the scope of license renewal since it has no intended function. Results of the aging management review of the reactor building dome are provided in Table 3.6-2 of the LRA.

3.3.6-6 **In Section 2.4.1.2 of the ANO-1 LRA, the Tendon Access Gallery is identified as a separate structure from the reactor building that does not perform an intended function, and consequently is outside the scope of license renewal. However, NUREG-1522 states that an adverse environmental condition in the tendon access gallery can have a deleterious effect on the lower tendon anchor components and surrounding concrete. On the basis of this information, provide the following information:**

(a) Describe the history of contaminants, humidity, and water infiltration into the gallery?

During the most recent structural walk down performed to comply with 10CFR50.65 and NUREG-1522, evidence of inleakage through cracks was observed in the tendon access gallery. Inleakage was minimal and no corrective action was required.

Puddling of water has been observed on the floor of the access gallery due to water infiltration. However, since the tendon anchorages are in the overhead of the gallery, this water inleakage has had no effect on the lower tendon anchor components and surrounding concrete. There have been no observations of abnormal levels of contaminants or humidity in the tendon access gallery.

(b) Have any remedial steps been taken in the past to alleviate any existing adverse conditions in the gallery such as water infiltration or deterioration of concrete and embedded rebar?

The most recent structural walk-down did not identify degradation in the tendon access gallery that required correction.

(c) Is there an existing inspection/maintenance program which addresses deterioration of the gallery and its environment?

The gallery is open to the auxiliary building. Tendon access gallery ventilation fans operate in conjunction with the auxiliary building ventilation system so the environment is essentially the same as the auxiliary building environment. No aging management program is necessary to address deterioration of the gallery since the gallery is not within the scope of license renewal.

(d) Identify any previous or existing condition of the lower tendon anchor components and surrounding concrete, with respect to corrosion of steel and loss of material, cracking, and change in material properties of concrete.

During the ANO-1 10-year inservice tendon inspection, some corrosion was present on anchorage components, but it was within acceptable limits. Likewise, corrosion levels at the ends of anchorage components were acceptable during the 15-year inservice inspection. During the 20-year inservice inspection, corrosion levels were also found to be acceptable except for the level on a shim at the shop end (i.e., top end) of one tendon. The shim was subsequently replaced. Corrosion was within acceptable levels during the 25-year tendon inspection and cracks surrounding the bearing plates were also within the allowable tolerance except for the shop end of one, horizontal tendon. A condition report was issued to address the unacceptable crack.

The unacceptable findings noted above for the 20-year and 25-year inservice tendon inspections were not within the tendon access gallery.

(e) On the basis of NUREG-1522 and your responses to the above questions, provide a technical justification for excluding the tendon gallery from an AMR.

The tendon access gallery provides access to the bottom of the vertical tendons so that they can be tested. The tendon access gallery is non-seismic and provides no structural support to the reactor building. Degradation of the tendon access gallery's concrete has no impact on the integrity of the reactor building. Because the gallery has no intended function, it is not within the scope of license renewal.

3.3.6-7 Section 3.6.7 of the LRA evaluates the following two aging effects/mechanisms for elastomers:

- **cracking due to ultraviolet radiation, thermal exposure, and ionizing radiation**
- **change in material properties due to ultraviolet radiation, thermal exposure, and ionizing radiation**

In order to complete the evaluation of the effects of aging for elastomers, the staff requests Entergy Operations to provide the following information:

(a) Subsection 3.6.7.1 of the ANO-1 LRA concludes that cracking of elastomers (waterstops) due to thermal exposure is not an applicable aging effect because the temperatures will be less than 95°F. Cracking due to ionizing radiation is not an applicable aging effect for waterstops since the radiation levels will be less than 10^6 rads. Provide the technical bases (e.g., technical references) for the stated threshold values for temperature (less than 95°F) and radiation levels (less than 10^6 rads).

The technical bases for not considering thermal aging significant if the ambient temperature is less than 95°F is from the following reference:

- W. M. Denny et al., "Aging Management Guideline for Electrical and Mechanical Penetrations," Draft report, Sandia National Laboratories., Contract AI-7321, March 1, 1996.

The technical references for the stated radiation threshold level are the following:

- "Radiation Data for Design and Qualification of Nuclear Plant Equipment," EPRI NP-4172M, Electric Power Research Institute, Palo Alto, California, August 1985;
- W. M. Denny et al., "Aging Management Guideline for Electrical and Mechanical Penetrations," Draft report, Sandia National Laboratories., Contract AI-7321, March 1, 1996, and;
- M. H. Van de Voorde and C. Restat, "Selection Guide to Organic Materials for Nuclear Engineering", CERN 72-7, European Organization for Nuclear Research, Geneva, 1972.

(b) Provide a description of the applicable site-specific operating history and include occurrences of observable seepage or leaching through concrete walls below grade, which would be indicative of degradation of waterstops, waterproofing membranes, caulking, and/or sealants.

During structural walk-downs, initiated under the Maintenance Rule Program, indications were observed of in-leakage of water through cracks in the tendon access gallery concrete. The in-leakage was minor and required no corrective action. As noted for RAI response 3.3.6-6(e), the tendon access gallery is not subject to an aging management review. Except for cracks in concrete components in other below grade locations (none of which exceeded the acceptance criteria), no other indications of in-leakage have been observed to date.

(c) Because seepage through these materials has been previously identified in other nuclear power plant structures, which is indicative of elastomer aging, provide a technical justification for not identifying aging that is applicable to elastomers.

Potential aging effects for elastomers are discussed in the ANO-1 LRA Section 3.6.7. Waterstops are indicated to be a type of elastomer that is subject to an aging management review. However, no aging effects requiring management for waterstops were identified. As noted for RAI response 3.3.6-1(a) through 3.3.6-1(d), expansion joint sealant, structural sealant, and caulking are considered part of the in-scope components and commodities to which they are attached and are managed accordingly. As noted for RAI responses 3.3.6-4(a) and 3.3.6–4(b), seals are routinely examined by inspections performed in accordance with preventive maintenance activities and replaced when their condition indicates they are no longer acceptable for service. Referring to RAI response 3.3.6-5(b), elastomeric roofing is considered a protective coating; similarly, waterproofing membranes are also considered protective coatings.

(d) If such conditions exist at ANO-1, provide an aging management review for the affected items or explain why such a review is not required.

Aging effects for elastomers that could lead to seepage were considered during the ANO-1 aging management review. There was one known instance of apparent in-leakage that occurred within the tendon access gallery. As indicated in RAI response 3.3.6-6(e), the tendon access gallery is not subject to an aging management review.

3.3.6-8 In Section 2.4.4 of the ANO-1 LRA, Category 2 building areas of the Intake Structure appear to be excluded for the license renewal scope. An earlier statement in Section 2.2.2 of the LRA indicates that some Category 2 structures are included in the license renewal scope. To clarify which Category 2 structures are included in the license renewal scope, Entergy Operations is requested to provide the following information:

(a) What Category 2 structures are included in the license renewal scope and what Category 2 structures are excluded?

Category 2 components and structures included in the license renewal scope are those that could possibly affect the function of a safety-related structure, system, or component or those that provide support or protection to equipment or components associated with in-scope systems. In addition, category 2 structures are included in the scope of license renewal if they are relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with NRC regulations for fire protection, EQ, PTS, ATWS, or station blackout.

Referring to ANO-1 LRA Section 2.4.3, category 2 components and structures within the auxiliary building which are within the scope of license renewal include the liner plate within the spent fuel pool area and the small pipe chase at elevation 341'. As also noted in Section 2.4.3, 10CFR50.48-required fire barriers within the auxiliary building are in-scope, which includes category 2 masonry block walls required by 10CFR50.48. Similarly, as noted in Sections 2.4.3 and 2.4.6.2, 10CFR50.48-required fire barriers within the category 2 turbine building are in-scope. As described in the response to RAI 2.4-5, the turbine building is within the scope of license renewal because it contains the above noted fire barriers. Referring to the ANO-1 LRA Section 2.4.6.1, category 2 yard structures within the scope of license renewal include the bulk fuel oil storage tank foundation and the alternate AC diesel generator building foundation.

(b) If any Category 2 structures are excluded from the license renewal Scope, provide the technical justification for each.

As stated in Section 5.1.2.2 of the ANO-1 SAR, seismic category 2 structures are those whose failure would not result in the uncontrolled release of radioactivity and would not prevent a safe reactor shutdown or the immediate and long-term operation following a loss of coolant accident. Category 2 structures excluded from the scope of license renewal do not meet the criteria of 10CFR54.4(a)(1), 10CFR54.4(a)(2), or 10CFR54.4(a)(3). Examples of category 2 structures not within the scope

of license renewal are the administration building, the emergency response facility and the transformer yard.

(c) For each Category 2 structure included in the license renewal scope, specify the LRA section which addresses the aging management review.

As stated in Section 3.6.1, the aging management review of the auxiliary building's category 2 spent fuel pool liner plate is addressed in Sections 2.3.3 and 3.4. As noted in Section 2.4.3, the category 2 small pipe chase at elevation 341' within the auxiliary building, which is constructed of concrete, is addressed in Section 3.6.2. Results of the aging management reviews are summarized in Table 3.6-4. Section 3.6.2 and Table 3.6-4 also summarize the aging management review of category 2 masonry block walls within the auxiliary building that are 10CFR50.48-required fire barriers. 10CFR50.48-required fire doors made of steel are addressed in Section 3.6.1 and 10CFR50.48-required concrete floors and concrete or masonry walls within the category 2 turbine building are addressed in Section 3.6.2; their aging management reviews are summarized in Table 3.6-4. As noted in Section 2.4.6.2, the aging effects of 10CFR50.48-required fire barrier commodities within the category 2 turbine building are addressed in Section 3.6.5 and summarized in Tables 3.6-8. As described in the response to RAI 2.4-5, the turbine building is within the scope of license renewal because it contains the above noted fire barriers. Referring to the ANO-1 LRA Section 2.4.6.1, category 2 yard structures within the scope of license renewal include the bulk fuel oil storage tank foundation and the alternate AC diesel generator building foundation. Aging effects for these structures, which are constructed of concrete, are discussed in Section 3.6.2 and summarized in Table 3.6-7.

3.3.6-9 Section 2.4.5 of the ANO-1 LRA discusses earthen embankments included in the license renewal scope. The intake and discharge canals to Lake Dardanelle and the emergency cooling pond are listed. The earthen embankments provide a heat sink during DBA or station blackout conditions, according to the LRA. From the information provided in Section 2.4.5 of the LRA, it is not clear whether Lake Dardanelle is in the scope of the rule, and if the lake is in the scope of the rule, is the water control structures and earth embankments, such as dams, within the scope of the rule. On the basis of this discussion, provide the following information:

(a) Is Lake Dardanelle in the scope of license renewal, and, if so, what is the intended function of Lake Dardanelle?

The portions of Lake Dardanelle controlled by Entergy Operations are considered within the scope of license renewal. The intended function of the Lake Dardanelle intake and discharge canals is to provide a heat sink during design basis accidents or station blackout as indicated in ANO-1 LRA Section 2.4.5 and Table 3.6-6.

(b) If the lake was determined not to be within the scope of license renewal, provide a technical justification as how the ultimate heat sink function is satisfied without relying on the lake.

As stated in correspondence dated November 19, 1999 (1CNA119903), the ultimate heat sink complex for ANO-1 consists of both the emergency cooling pond and Lake Dardanelle.

(c) If the lake does perform an intended function, identify any water control structures which are relied upon to maintain the water inventory of the lake.

Water control structures relied upon to maintain the water inventory of the lake include the Dardanelle Dam and the intake and discharge canals. Refer to RAI response 3.3.6-9(a) and RAI response 2.2-1 (in correspondence dated August 30, 2000 (1CAN080007)).

(d) For any water control structures identified in (c) provide an aging management review, identify applicable aging effects, and describe the AMPs which are relied on to manage aging.

A discussion of the aging management review for the intake and discharge canals is provided in ANO-1 LRA Section 3.6-6. Since no aging effects requiring management were identified for the intake and discharge canals, no AMP is credited for their management. For a discussion of the aging management of the Dardanelle Dam, refer to RAI response 3.3.6-9(a) and RAI response 2.2-1 (in correspondence dated August 30, 2000 (1CAN080007)).

3.3.6-10 Sections 3.6.1, 3.6.2, and 3.6.3 of the ANO-1 LRA do not address how aging effects of structures and structural components in inaccessible areas are managed. In the LRA the only mention of inaccessible areas is in Appendix B, Section 4.3.6 – IWL Inspections, which states that "items exempt from the examination requirements include inaccessible end anchors and concrete surfaces." Managing aging degradation of

inaccessible areas needs to be addressed for all structures and structural components within the scope of license renewal.

ASME Section XI exempts inaccessible areas in containment. However, 10CFR50.55a provides additional requirements that the licensee evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. In accordance with NUREG-1611 for license renewal, the staff has determined that applicants also need to evaluate, on a case-by-case basis, the acceptability of inaccessible areas when conditions in accessible areas may not indicate the presence of or result in degradation to such inaccessible areas, in order to ensure that the intended functions of the structures and components will be maintained consistent with the CLB during the period of extended operation. This additional evaluation needs to be performed for four aging mechanisms presented in NUREG-1611.

In accordance with the conclusion from NUREG-1557, aggressive chemical attack and corrosion of reinforcing steel may cause potentially significant degradation of below-grade portions of Class 1 concrete structures. Also, corrosion is potentially significant for inaccessible structural steel. To address these issues, applicants for license renewal are expected to describe a plant-specific aging management program that could include monitoring of groundwater chemistry, inspection and testing.

On the basis of this discussion, provide the following information:

(a) For the reactor building (containment), describe how aging effects for inaccessible areas will be addressed when conditions in accessible areas may not indicate the presence of or result in degradation to such inaccessible areas. The aging management review for this condition(s) needs to consider the aging effects discussed in NUREG-1557 such as loss of material, increased porosity, cracking, etc. (caused by leaching of calcium hydroxide from concrete, aggressive chemical attack on concrete, corrosion of structural steel and liners, and corrosion of embedded reinforcing steel). If any of these aging effects do not apply, provide technical justification in accordance with the guidance presented in NUREG-1611.

The aging effects discussed in NUREG-1557 and NUREG-1611 for which conditions in accessible areas may not indicate the presence of or result in degradation to such inaccessible areas include the following: 1) increase of porosity and permeability of concrete due to leaching of calcium hydroxide and aggressive chemical attack , 2) cracking and spalling of concrete due to

aggressive chemical attack, 3) loss of material due to corrosion of steel and corrosion of concrete embedded steel, and 4) loss of bond due to corrosion of concrete embedded steel.

With regard to leaching, Section 3.6.3.3 of the ANO-1 LRA states that a change in material properties, which is manifested in concrete as increased permeability and increased porosity, is an aging effect requiring management for the dome and cylinder wall. The leaching will not challenge the intended functions of the ANO-1 concrete components. Because the ANO-1 reactor building is constructed of dense, well-cured high strength concrete with low permeability, inaccessible areas are no more susceptible to leaching than accessible areas. A change in material properties (increase of porosity and permeability) due to aggressive chemical attack is not applicable to accessible or inaccessible concrete. ANO-1 was designed and constructed in accordance with ACI and ASTM. Its concrete has a high cement content and low water-cement ratio resulting in concrete which has a low permeability and a high resistance to aggressive chemical solutions. Furthermore, the acidic, chloride, and sulfate concentrations in the ANO-1 raw water (i.e., ground water and lake water) are below the aggressive chemical threshold limits which cause degradation in concrete.

In regard to the second aging effect, Section 3.6.3.2 of the ANO-1 LRA indicates that cracking is an aging effect requiring management for the dome and cylinder wall. Similar to leaching, it was determined that cracking of exposed surfaces will not challenge the intended functions of the concrete. Cracking due to reactions with aggregates is not applicable since non-reactive aggregates were used. For reasons stated above for the first aging effect, loss of material due to aggressive chemical attack, which is manifested in concrete as spalling, is not applicable. Therefore, inaccessible areas are no more susceptible to spalling than accessible areas.

In regard to the third aging effect and referring to Section 3.6.1.1 of the ANO-1 LRA, loss of material due to corrosion is an aging effect requiring management for accessible steel components and commodities. Protective coatings help to prevent the onset of this aging effect. However, loss of material due to corrosion is not an applicable aging effect for inaccessible steel (i.e., embedded steel, reinforcing steel). In the absence of concrete degradation by other mechanisms, the adequate concrete cover over embedded steel, the lack of exposure to aggressive groundwater, and the high alkaline environment of concrete preclude corrosion of embedded steel as an aging mechanism.

In regard to the fourth aging effect noted above and referring to Section 3.6.2.3 of the ANO-1 LRA, a change in material properties, which is

manifested in concrete as a reduction or loss of bond strength is not an aging effect for accessible or inaccessible concrete. Corrosion of embedded steel is an aging mechanism associated with the aging effect, loss of material. However, as discussed in the preceding paragraph, corrosion is not an applicable aging mechanism for embedded steel.

In summary, aging effects requiring management for inaccessible areas have been addressed in the aging management review process which considered construction materials of components and commodities in both accessible and inaccessible areas and their exposure to the various, applicable environments. Based on the review process, no unique aging effects were identified for inaccessible areas. Furthermore, by considering design and construction practices, and operating environments, inaccessible areas were determined to be no more susceptible to aging than accessible areas. Therefore, no unique aging management programs are required for inaccessible areas.

(b) For other structures within the scope of license renewal, describe how aging effects for inaccessible areas will be addressed. If the aging effects discussed in NUREG-1557 such as loss of material, increased porosity and permeability, spalling, scaling, expansion, loss of bond, distortion, reduction in foundation strength, and cracking are determined to be not applicable, provide your technical justification in accordance with the guidance presented in NUREG-1557.

For other structures within the scope of license renewal, the determination of aging effects for components and commodities in areas that are inaccessible for inspection has been addressed in the same manner as the determination of aging effects for components and commodities within accessible areas. Aging effects were determined for both inaccessible and accessible components and commodities based on materials of construction and operating environments. Applicability of aging effects discussed in NUREG-1557 to ANO-1 is provided in the response above for RAI response 3.3.6-10(a).

(c) Chemicals in groundwater and chemicals in raw water have been identified as environments applicable to ANO-1 concrete. Provide a description of the AMP consistent with the ten elements of an AMP identified in the Standard Review Plan in sufficient detail for the staff to adequate evaluate the program.

Although some below grade concrete components are exposed to chemicals in raw water (i.e., ground water and lake water), the concentrations of aggressive chemicals in the ANO-1 raw water are below aggressive chemical threshold limits that cause degradation. Since no

aging effects requiring management were identified for concrete exposed to raw water, an aging management program is not required.

3.3.6-11 Sections 3.6.2 and 3.6.3 of the LRA discuss potential aging effects of concrete in accessible areas. A statement is made that "other potential aging effects and aging mechanisms do not apply to ANO-1 concrete components and commodities due to the absence of susceptible material and environmental conditions." A number of potential aging effects identified in NUREG-1557 have not been addressed in the LRA. Provide a technical justification for determining that the aging effects identified in NUREG-1557 need not be managed during the period of extended operation including the loss of material, increased porosity and permeability, spalling, scaling, expansion, loss of bond, distortion, reduction in foundation strength, and cracking.

Loss of material and cracking due to various aging mechanisms are both addressed as aging effects in Sections 3.6.2 and 3.6.3 of the ANO-1 LRA. As noted in Section 3.6.2.1, concrete spalling and scaling may occur as a result of one or more aging mechanisms associated with a loss of material. Similarly, referring to Section 3.6.2.3, increased permeability, increased porosity, and reduction in bond strength are results of the aging effect, change in material properties.

Distortion, expansion and reduction in foundation strength fall under the statement in Section 3.6 of the ANO-1 LRA that other potential aging effects and aging mechanisms do not apply to ANO-1 components and commodities due to the absence of susceptible material and environment combinations. Distortion due to component support movement from unanticipated thermal expansion may occur as a result of improper design; as such, this is not considered an applicable aging effect for steel components and commodities. However, the ANO-1 aging management review considered mechanical distortion due to creep and fatigue. Mechanical distortion due to creep is not applicable to steel components and commodities since operating temperatures are below the threshold for creep damage. In general, mechanical distortion due to fatigue is not applicable since loads are applied gradually and remain constant or are accounted for by design codes. Where appropriate, fatigue has been addressed as a time-limited aging analysis (TLAA) in the ANO-1 LRA.

Likewise, the ANO-1 aging management review also considered a loss of material and a change in material properties, both due to aggressive chemicals, as potential aging effects since acid attack may reduce strength and groundwater chemicals may damage foundation concrete. However, neither aging effect due to aggressive chemicals is applicable to concrete components and commodities. Since ANO-1 was designed in accordance with ACI and

ASTM, its concrete has a high cement content and low water-cement ratio resulting in concrete which has a low permeability and a high resistance to aggressive chemical solutions. Furthermore, the acidic, chloride, and sulfate concentrations in the ANO-1 raw water (i.e., ground water and lake water) are below the aggressive chemical threshold limits which cause degradation in concrete.

3.3.6-12 It is noted in Section 3.6.1 of the ANO-1 LRA that the spent fuel pool steel liner is addressed in Sections 2.3.3 and 3.4 of the ANO-1 LRA. However, the staff's aging management review of the spent fuel pool liner is included with the steel structures. Section 3.4.2 of the ANO-1 LRA, states that cracking is a potential aging effect for the spent fuel pool liner. Section 3.4.3 lists the Spent Fuel Pool Monitoring Program (a new program described in App. B, Section 3.6) and the Spent Fuel Pool Level Monitoring Program (described in App. B, Section 4.21.8) as applicable to the spent fuel pool liner plate. The Spent Fuel Pool Monitoring Program is used to monitor the spent fuel pool monitoring trench drains. The Spent Fuel Pool Level Monitoring Program is used to monitor the water level in the spent fuel pool. The staff has previously concluded that stress corrosion cracking and crevice corrosion of fuel storage facility stainless steel liners are adequately managed by periodic monitoring of the leak chase system drain lines and the leak detection sump. On the basis of the above discussion, provide the following information:

a. A description of the spent fuel pool monitoring trench system with identification of any differences from a typical spent fuel pool leak chase system.

The spent fuel pool is a reinforced concrete pool lined with welded stainless steel and fitted with tell-tale drains (i.e., monitoring trench drains) to indicate any leak in the liner. These drains are nonsafety-related and are not required for pool integrity since they are located behind the liner. The drains collect leakage into the area between the spent fuel pool liner and the concrete wall and are connected to the auxiliary building equipment drain tank. A connection is provided to allow leakage from the drains to be sampled and measured. The ANO-1 spent fuel pool monitoring trench system is typical of other spent fuel pool leak chase systems.

b. **Will the monitoring of trench system drains, coupled with the monitoring of the fuel pool water level, provide for fuel pool leakage detection at least equivalent to that which would be achieved with the monitoring of a spent fuel pool leak chase system? Describe the basis for this conclusion.**

Yes, the monitoring of trench system drains, coupled with the monitoring of the spent fuel pool water level will provide for fuel pool leakage detection at least equivalent to that which would be achieved with the monitoring of a spent fuel pool leak chase system. Since the ANO-1 spent fuel pool monitoring trench system is typical of other spent fuel pool leak chase systems, the Spent Fuel Pool Monitoring Program alone provides equivalent leak detection to a spent fuel pool leak chase system. Implementation of the Spent Fuel Pool Level Monitoring Program provides an additional means of leak detection.

c. **If not, provide a technical justification for the specified monitoring programs.**

Based on the response to part (b) of this RAI, this part of the RAI is not applicable.

3.3.6-13 Provide a description of the findings from the operating experience review for steel components, IWE, IWF, Boric Acid Corrosion Prevention and Maintenance Rule AMPs.

Operating experience pertaining to steel components was reviewed during the structural aging management reviews. Information reviewed included NRC generic communications such as information notices, IE bulletins, and generic letters dating back to 1973. The results of this review validated the aging effects discussed in the LRA. Because IWE Inspections are a recent addition to the ISI Program, site specific operating experience is not available for this program. IWF Inspections have been conducted as required by 10CFR50.55a. Deficiencies have been found under this program and corrected in accordance with the site corrective action program. Discrepancies are typically identified and corrected prior to loss of function. Since implementation of the Boric Acid Corrosion Prevention Program, over 60 leaks have been identified and evaluated. These leaks have been reported on both reactor coolant system components and on other systems that contain borated water. Following engineering evaluations of these leaks, component repairs and replacements were performed as appropriate. The Maintenance Rule Program uses visual inspections to identify aging effects. While actual experience with the Maintenance Rule walk-downs is limited, visual inspections have proven

effective throughout the industry in managing aging effects on plant equipment and have been accepted under the current licensing basis as appropriate for implementation of the maintenance rule.

3.3.6-14 In Section 3.6.1 and Table 3.6-8 of the ANO-1 LRA, Thermashield [associated with pipe supports] is listed as a submaterial under steel components. No further reference to this material is provided in the ANO-1 LRA. Provide the following information:

(a) A description of this material, its purpose and its use in/on ANO-1 structures and structural components.

Thermashield is a patented material used at pipe supports to insulate and structurally support high temperature piping.

(b) Intended function(s) associated with Thermashield and the technical basis for its inclusion/exclusion in the scope of license renewal.

As a submaterial of steel associated with pipe supports, the intended functions of Thermashield are the same as those for pipe supports. In conjunction with pipe supports, Thermashield provides structural support or functional support to piping associated with safety-related equipment and provides structural or functional support to piping associated with nonsafety-related equipment, failure of which could directly prevent satisfactory accomplishment of required safety-related functions. Although Thermashield by itself is not considered a structure, component, or commodity, the technical basis for including it within the scope of license renewal is because of its use with in-scope pipe supports.

(c) Discuss if the aging effects and the AMPs associated with the steel components are applicable to Thermashield. If so, identify the attributes monitored to detect the aging associated with Thermashield and the attributes monitored in the managing the aging.

Yes, the aging effect, loss of material, which requires management for pipe supports also requires management for Thermashield. Referring to Table 3.6-8, the Maintenance Rule Program and ASME Section XI, Subsection IWF are credited for managing Thermashield. Since Thermashield is associated with high temperature piping, the Service Water Chemical Control Program, which applies to the intake structure's pipe supports submerged in water, is not an applicable aging management program for Thermashield. The attribute monitored to detect aging of Thermashield is the same as that for pipe supports (i.e., visual inspection for general condition).

(d) **Discuss potential aging of steel components due to contact with Thermashield.**

Thermashield is an insulator, which also provides structural support. As an inert material, it does not affect aging of steel components.

(e) **Can application of Thermashield reduce or compromise the effectiveness of AMPs credited with managing the aging of the Thermashield-treated structural components (e.g., render component inaccessible for inspection)? If so, how does the credited aging management program compensate for this potential concern.**

No, Thermashield does not compromise the effectiveness of aging management programs credited for managing aging effects for pipe supports. It is removable and reuseable and does not render associated pipe or support steel inaccessible.

3.3.6-15 Section 3.6.2.4 identifies the review of ANO-1 operating experience as one basis for concluding "no additional aging effects beyond those discussed in this section have been identified" for concrete. Discuss the results from any concrete structure and structural components, concrete commodity groups, and concrete submaterials inspections that have been performed at ANO-1, and describe any follow-up corrective actions as a result of these inspections.

An initial structural walk-down was recently performed under the ANO-1 Maintenance Rule Program. Concrete cracking was observed, but no cracks exceeded the acceptance criteria. Water in-leakage was indicated in the tendon access gallery, but was below the threshold requiring corrective action. As noted in Sections 3.6.3.2 and 3.6.3.3 of the ANO-1 LRA, minor cracking and leaching observed on the reactor building concrete surfaces were evaluated and determined to not challenge the intended function of the concrete.

3.3.6-16 Section 3.6.2.4 and Table 3.6-5 refers to the Maintenance Rule Program (Section 4.13 of Appendix B) as the AMP for managing the effects of loss of material for the intake structure exterior concrete wall at the normal lake level. In accordance with NUREG-1557 conclusions Regulatory Guide 1.127, "Inspection of Water-Control Structures Associated with Nuclear Power Plants" describes an acceptable basis to the staff for developing an appropriate in-service inspection and surveillance program for water control structures, such as the intake structure. On the basis of this information, provide the following:

(a) Describe the inspection requirements for Maintenance Rule AMP with respect to (equivalent to or more stringent) the aging management activities describe in NUREG-1557.

Section 3.10.3.2.10 of NUREG-1557 indicates that aging management activities for the Calvert Cliffs Nuclear Power Plant intake structure include 1) periodic draining of the intake structure cavities, 2) performing visual inspections to assess the condition of the structure for degradation, specifically related to steel corrosion, and 3) periodic inspections of the sluice gates for corrosion.

Similar to the management activities described in NUREG-1557, the ANO-1 Maintenance Rule Program, which has been reviewed by the NRC, requires that periodic inspections of the intake structure be performed to identify potential areas of structural degradation. Like NUREG-1557, the ANO-1 Maintenance Rule Program includes checklists that not only target structural steel, but that also assess the condition of the intake structure's concrete and roofing components and commodities. Indications of potential degradation for steel under the ANO-1 Maintenance Rule Program include flaking rust, widespread corrosion, pitting, loose or missing fasteners, and deteriorated coatings. As denoted in Tables 3.6-5 and 3.6-8 of the ANO-1 LRA, visual inspection of submerged steel components and commodities under the Maintenance Rule Program is coordinated with the inspection of the service water bays.

In accordance with the ANO-1 Maintenance Rule Program, the current plans are to perform a structural evaluation of the intake structure every five years, compared to the 6-year evaluation frequency in NUREG-1557. The 5-year evaluation frequency corresponds to that suggested by Regulatory Guide 1.127. Unlike the example cited in NUREG-1557, the ANO-1 intake structure is not exposed to salt water. Inspection requirements for cleaning and inspecting the service water bays and sluice gates fall under the ANO-1 Service Water Integrity Program, rather than the Maintenance Rule Program. Attributes of the Service Water Integrity Program are described in Appendix B, Section 4.19 of the ANO-1 LRA.

(b) If not, provide a technical justification for relaxation of the aging management activities described in NUREG-1557.

The ANO-1 Maintenance Rule Program in conjunction with the Service Water Integrity Program is equivalent to the management activities described in NUREG-1557.

3.3.6-17 Section 3.6.2.4 and Table 3.6-4 refers to the Fire Protection Program (Fire Barrier Inspections) and the Maintenance Rule Program (Sections 4.8.1 and 4.13 of Appendix B) as the Aging Management Program for managing the effects of cracking for masonry block walls in the Auxiliary Building. In accordance with NUREG-1557, inspection requirements imposed in I&E Bulletin No. 80-11, "Masonry Wall Design" and plant-specific monitoring proposed by NRC Information Notice (IN) No. 87-65, "Lessons Learned from Regional Inspection of Licensee Actions in Response to IE Bulletin 80-11" are effective programs for managing aging effects of masonry block walls. On the basis of this information, provide the following:

(a) Identify the masonry walls and the applicable intended functions that are included within the scope of license renewal and subject to an AMR.

Referring to the RAI response 3.3.6-8(a) and RAI responses 2.3.3.2-1(c) and 2.3.3.2–1(d) in correspondence dated August 30, 2000 (1CAN080007), the ANO-1 safety-related masonry block walls that fall within the scope of IE Bulletin 80-11 and masonry block walls that provide 10CFR50.48-required fire protection are within the scope of license renewal and subject to an aging management review. For the intended functions of in-scope masonry block walls, refer to Table 3.6-4, Page 3-118.

(b) Identify any masonry wall included in the scope of IE Bulletin 80-11 and Unresolved Safety Issue (USI) A-46 for ANO-1 that is not within the scope of license renewal and subject to an aging management review. Provide a justification for excluding any of these walls from an aging management review.

There are no masonry walls within the scope of IE Bulletin 80-11 and USI A-46, which are not within the scope of license renewal. Referring to RAI response 2.3.3.2-1(e) in correspondence dated August 30, 2000 (1CAN080007), no new walls were identified by the ANO-1 USI A-46 program.

(c) Describe how the ANO-1 AMP for periodic inspection and surveillance of masonry walls incorporates the insights provided in IN No. 87-67.

With regard to inspections of masonry walls, IN 87-67 highlighted the need for careful field verification of critical parameters used in the qualification of the walls. The ANO-1 Maintenance Rule Program and Fire Protection Program (Fire Barrier Inspections) incorporate insights provided by IN

87-67 through the performance of periodic inspections. Periodic inspections of masonry walls provide reasonable assurance that the physical conditions of the walls remain as analyzed. Both programs have procedural controls that require corrective actions if deficiencies are observed so that the continuing structural integrity of the block walls is assessed.

3.3.6-18 LRA Appendix B identifies and briefly describes the AMPs which are credited for ANO-1. Appendix B Sections 4.3.4 (IWE Inspections), 4.3.5 (IWF Inspections), and 4.3.6 (IWL Inspections) identify three inspection programs which are credited by 10CFR50.55a for the current licensing term. However, the description of these AMPs do not reference the additions and modifications imposed by 10CFR50.55a, nor the specific editions/addenda of ASME Section XI that are specified in 10CFR50.55a. For IWE Inspections, page B-34 of the LRA specifies the ASME B&PV Code Section XI, 1992 Edition, 1993 Addenda for Pressure Testing; 10CFR50.55a specifies the ASME Code Section XI, 1992 Edition, including the 1992 Addenda. For IWF Inspections, page B-36 indicates that the ASME Section XI, 1992 Edition, 1993 Addenda for Pressure Testing will be used to develop the aging program; 10 CFR 50.55a specifies the use of IWF, ASME Code Section XI, 1989 Edition for support inspection. For IWL Inspections, page B-37 specifies the ASME Code Section XI, Subsection IWL, with no indication of the edition. 10CFR50.55a specifies the ASME Code Section XI, 1992 Edition, including the 1992 Addenda.

On the basis of the above discussion, provide the following additional information:

a. For IWE Inspections, describe the differences between the Code editions, especially any relaxation afforded by the 1993 Addenda for Pressure Testing. Also, clearly state whether the additions and modifications specified in 10CFR50.55a are part of the AMP; justify any exclusion.

For IWE inspections, page B-36 contains an administrative error. The correct Code edition is the 1992 Edition, including the 1992 Addenda. Page B-36 incorrectly listed the 1993 Addenda. Additions and modifications specified in 10CFR50.55a are part of the ANO-1 aging management program. Entergy Operations will continue to use the IWE Inspection program that is approved by the NRC for the ANO-1 current licensing basis.

b. **For IWF Inspections, describe the differences between the Code editions, especially any relaxation afforded by the 1993 Addenda for Pressure Testing.**

For IWF Inspections, the ASME Section XI, 1992 Edition, 1992 Addenda will be used and the alternatives approved by the NRC in letter 0CNA129613, dated December 12, 1996, will also be used. We will continue to use the IWF Inspection program that is approved by the NRC for the ANO-1 current licensing basis.

c. **For IWL Inspections, clarify if "Alternative Examination 99-0-002" for Arkansas Nuclear One, Units 1 accepted by NRC on June 2, 1999 will constitute your AMP.**

For IWL Inspections, the ASME Section XI, 1992 Edition, 1992 Addenda will be used and the "Alternative Examination 99-0-002" will be used as approved by the NRC in letter 0CNA069902, dated June 2, 1999. We will continue to use the IWL Inspection program that is approved by the NRC for the ANO-1 current licensing basis.

3.3.6-19 Appendix B, Section 4.3.4 of the LRA indicates that the scope of the ASME Section XI, Subsection IWE Inspections, credited for license renewal, includes inspections of the reactor building liner plate. No mention is made of the other components (Examination Categories E-A, E-B, E-C, E-D, E-F, E-G, E-P) included in the scope of IWE Inspections. In addition it is not clear whether all requirements contained in Subsection IWE of the Code are included in the aging management program. Therefore, provide the following information:

(a) **Whether the aging management program (IWE Inspections) for license renewal commits to the entire scope and all requirements stated in ASME Section XI, Subsections IWE and the additions and modifications specified in 10CFR50.55a.**

IWE Inspections conducted under the ANO-1 Inservice Inspection Program meet the scope and requirements of 10CFR50.55(a), as they relate to ASME Section XI, Subsection IWE and NRC approved alternatives to ASME Section XI, Subsection IWE.

(b) **If not, identify and provide a justification where the scope and requirements differ.**

Not applicable. See response to part (a).

3.3.6-20 **Appendix B, Section 4.3.6 of the LRA indicates that IWL inspections are performed on the reactor building's post-tensioning systems and concrete components that are subject to an aging management review as identified in Sections 2.4 and 3.6 of the LRA. Please verify that all reactor building post-tensioning components within the scope of ASME, Section XI, Subsection IWL are included in the scope of components requiring aging management. Identify any additional reactor building post-tensioning components requiring aging management. In addition, please verify that all the requirements contained under Subsection IWL of the Code, including limitations under 10CFR50.55(a)(b)(2)(ix), are included in the AMP. If not, provide a description and justification where the scope and/or aging management requirements differ.**

Yes, the reactor building post-tensioning components within the scope of ASME, Section XI, Subsection IWL are within the scope of components requiring aging management. No post-tensioning components outside the scope of Subsection IWL require aging management. As indicated in Table 3.6-2 of the ANO-1 LRA, steel components of the post tensioning system include tendon wires and tendon anchorages. Prestressed concrete components consist of the dome and cylinder wall.

IWL Inspections conducted under the ANO-1 Inservice Inspection Program meet the requirements under Subsection IWL of the Code, including limitations under 10CFR50.55(a)(b)(2)(ix).

3.3.6-21 **Section 4.5 of Appendix B of the ANO-1 LRA states that Boric Acid Corrosion Prevention program is credited with monitoring the boric acid corrosion of carbon steel surfaces exposed to leakage from borated water. It is stated that, in this program, ANO-1 completes visual inspections to identify pressure boundary leaks and gives consideration to the possibility of flow paths from the leak to carbon steel components, or the accumulation of boric acid in insulation.**

(a) Is a periodic visual examination of structures, components and supports adjacent to (in proximity and below) pressure boundary components included in the visual inspections performed in this program?

Yes, the Boric Acid Corrosion Prevention Program in conjunction with the reactor building walk-down procedure includes visual inspections of components and supports adjacent to pressure boundary components. Under the Boric Acid Corrosion Prevention Program, inspections during plant shutdown and plant heatup are performed to identify boric acid

leakage. If leakage or corrosion is identified, visual examination of components and supports in the proximity of pressure boundary components may be required to determine flow path. The Boric Acid Corrosion Prevention Program was developed in accordance with recommendations in Generic Letter 88-05 which indicates that the source of leakage and the path taken by leaking fluid are to be evaluated.

(b) Based on staff experience, we have required inspection of adjacent structures and components (including supports) to manage the effects of boric acid corrosion. If your program does not include inspection of adjacent structures and components, provide a technical justification for excluding these inspection activities or include them as part of the proposed AMP.

The Boric Acid Corrosion Prevention Program includes inspection of adjacent structures and components. Refer to the above RAI response 3.3.6-21(a).

3.3.6-22 It is stated in Section 3.6.1 of the ANO-1 LRA that the types of steel components and commodities subject to an AMR for ANO-1 are summarized in Tables 3.6-1 through 3.6-8. An aging effect not listed in the tables is the loss of component support for Class 1, 2, 3, or MC piping. On the basis of the staff's experience from the review of the first two LRAs, the loss of component support for Class 1,2,3, or MC piping due to corrosion, distortion, dirt, overload, vibratory, cyclic thermal loads and elastomer hardening is typically within the scope of license renewal and subject to an AMR. Identify where in the LRA is this AMR described, or provide a justification for not considering loss of component support as an applicable aging effect for Class 1,2,3, or MC piping.

Loss of component support pertains to the intended function of piping supports rather than an aging effect. Referring to ANO-1 LRA Table 3.6-8, piping supports provide structural support or functional support (i.e., component support) to piping associated with safety-related equipment or to piping associated with nonsafety-related equipment, failure of which could directly prevent satisfactory accomplishment of required safety-related functions. The aging effect requiring management for piping supports and their threaded fasteners is a loss of material. As discussed in Section 3.6.1, corrosion is an aging mechanism that may lead to a loss of material for steel pipe supports and threaded fasteners. Distortion that may occur as a result of improper design (i.e., component support movement owing to water and steam hammer or unanticipated thermal expansion) is not considered an applicable aging effect. The ANO-1 aging management review did not identify unique aging effects due to dirt or overload for steel or threaded fasteners. Overload

would be associated with improper design. As stated in Section 3.6.4.4, self-loosening by vibration is not an aging effect for ANO-1 since it is accounted for by adequate design and installation. Cyclic load embedment is an aging mechanism associated with loss of preload. Threaded fasteners for structural components and commodities are not subjected to cyclic loads. Likewise, thermal effects which may lead to loss of preload for threaded fasteners is not applicable since embedment is minimal and the high temperatures required for creep are not observed in structural bolting applications. Lastly, based on the ANO-1 aging management review, elastomer hardening was not identified as an aging mechanism for steel or threaded fasteners. However, aging effects for passive, long-lived elastomers are discussed in Section 3.6.7.

3.3.6-23 Subsection 3.6.4.4 of the ANO-1 LRA states that the same AMPs that are credited for the management of aging effects for steel components and commodities are credited for the management of aging of threaded fasteners. A review of Tables 3.6-2 through 3.6-8 of the LRA indicates that the table entries omit reference to various aging effects for threaded fasteners. Examples are loss of material from boric acid wastage for threaded fasteners in structural connections in the vicinity of the spent fuel pool, and stress corrosion cracking and intergranular attack of stainless steel threaded fasteners in raw water. It is not clear what AMPs correspond to which threaded fasteners and whether every identified aging effect for threaded fasteners at ANO-1 is being managed.

In addition, subsection 3.6.4.4 contains a statement that self-loosening of bolted connections due to vibration is not an aging effect because of adequate preload during installation. However, expansion and undercut anchors in concrete may become loose due to local degradation of the surrounding concrete as a result of vibratory loads. On the basis of the above discussion, provide the following information:

(a) Identify the specific AMP(s) which is credited for managing each applicable aging effect for threaded fasteners;

The last paragraph of Section 3.6.4.4 of the ANO-1 LRA is a general statement regarding the aging management programs for threaded fasteners. The specific aging management programs managing aging effects for threaded fasteners are identified in Tables 3.6-2 through 3.6-8. For example, refer to heading "Threaded Fasteners" for Table 3.6-4 on Page 3-117 and Table 3.6-5 on Page 3-120. Note that the tables only indicate the aging effects requiring management for threaded fasteners. Aging mechanisms (i.e., boric acid wastage), that are not required to be identified in an LRA, are not listed in the tables.

For structural connections near the spent fuel pool, refer to the discussion in Section 3.6.4.1. For cracking of stainless steel threaded fasteners in raw water due to stress corrosion and intergranular attack, refer to the discussion in Section 3.6.4.2. Similarly, other aging mechanisms associated with aging effects are discussed for threaded fasteners in Section 3.6.4.

(b) Provide a technical justification for not identifying loss of preload due to the effects of vibration on concrete surrounding expansion and undercut anchors.

Loss of preload due to vibration on concrete surrounding expansion and undercut anchors is considered to be a minor contributor to the loosening of structural steel threaded fasteners. Loss of preload is eliminated by initial preload bolt torquing.

3.3.6-24 Section 3.6.3.3 indicates that "prestressed concrete components are not exposed to temperatures that exceed the threshold for degradation identified in ACI 318-63." It should be noted that ACI 318-63 (as well as other Editions of ACI 318) only refers to various types of losses in prestressing forces similar to those described by the applicant in its SAR. It does not address the effects of temperature and radiation on the material properties of concrete or prestressing steel. Information Notice (IN) 99-10 describes operating experience at various plants where it was discovered that the normal sustained temperature (> 90/F) effects on the relaxation of prestressing steel is significant in the prestressed concrete containments. In light of the discussion in Attachment 2 of the IN, provide a summary of how the prestressing forces at ANO-1 are affected by such sustained temperatures.

The ANO-1 twenty-fifth–year tendon surveillance performed in accordance with ASME Section XI, Subsection IWL was recently completed. ANO-1 has not experienced more than projected loss of prestressing forces.

3.3.6-25 During review of the LRA, a number of potential editorial errors were identified. Verify if the following were editorials.

(a) Section 3.1.1 (p. 3-2) - verify that "4.7" should be "4.6"

The reference to the description of the chemistry monitoring programs should have been to Section 4.6 of Appendix B of the ANO-1 LRA. This was previously corrected in correspondence dated April 11, 2000 (1CAN040001).

(b) Section 3.1.3 (p. 3-2) - specify the proper reference to Appendix B for "Structures and Systems Walk-downs." It appears that "4.16" should be "4.13"

The reference to the description of structure and system walk-downs should have been to Section 4.13 of Appendix B of the ANO-1 LRA. This was previously corrected in correspondence dated April 11, 2000 (1CAN040001).

(c) Table 3.6-2 (p. 3-113) – verify that "Fuel Transfer Tube" should be "Fuel Transfer Tube Penetration."

"Fuel Transfer Tube" is correct as listed in Table 3.6-2 of the ANO-1 LRA. Although it is a penetration, Entergy Operations refers to it as the fuel transfer tube, similar to the personnel and equipment hatches. For further description of the fuel transfer tube, refer to Section 2.4.1.1.

(d) Table 3.6-5 (p. 3-121) – verify that Footnote F should be specified one row above where it is currently specified. If this is not the case, explain the apparent discrepancy between the wall elevation and the normal lake level elevation.

Referring to Table 3.6-5 (page 3-121) of the ANO-1 LRA, footnote 'F' appears in the row for exterior concrete walls above grade (or concrete above lake level) since the aging effect (loss of material) does not apply to exterior concrete walls below grade (or concrete below lake level). Even though the normal lake level (El. 338') and grade elevation (El. 354') are not the same, it was deemed more appropriate to group the wall at the normal lake level with the intake structure's above grade exterior walls.

(e) Appendix B, Section 4.16 (p. B-79, fifth paragraph) - verify that "Type C" should be "Type B". Type B tests are described as an AMP for the reactor building in App. B, Section 4.16.2.

The fifth paragraph of Section 4.16 of Appendix B of the ANO-1 LRA should indicate that Type B tests, along with Type A and Type C tests, are considered for aging management for license renewal. Section 4.16.1 of Appendix B describes how Type A testing is used for aging management. Section 4.16.2 of Appendix B describes how Type B and C testing is used for aging management.

(f) **Appendix B, Section 4.21.8 (p B-103) - in the fourth line from the bottom of the page, verify that "3.7" should be "3.6."**

In Section 4.21.8 of Appendix B, the reference to the Spent Fuel Pool Monitoring Program should be to Section 3.6 of Appendix B of the ANO-1 LRA.

(g) **Table 3.4-1 (p. 3-58) - Verify that the Spent Fuel Pool Monitoring Program should also be listed as a program for liner plate cracking along with the Spent Fuel Pool Level Monitoring Program. If not, provide a justification.**

In Table 3.4-1, along with the Spent Fuel Pool Level Monitoring Program, the Spent Fuel Pool Monitoring Program should also be listed as an aging management program for cracking of the external surfaces of the spent fuel pool liner plate.

4.5-1 **Loss of reactor building prestress has been identified in the ANO-1 LRA as a TLAA. It is evaluated in Section 4.5 of the LRA. Section 4.5 of the LRA states that "ANO-1 is completing a calculation of the final effective tendon prestress based on additional information on concrete creep from existing creep tests and results of the tendon surveillance testing." This calculation is expected to confirm projections on the relaxation of the tendons and will show that the tendons will be acceptable for the period of extended operation. This type of analysis would be consistent with a TLAA performed in accordance with 10CFR54.21(c)(1)(ii). The application also that the ASME Section XI Inservice Inspection Program, IWL Inspections will be adequate to manage the effects of aging on the intended function for the period of extended operation. This is followed by a statement in LRA Section 4.5 that the "implementation of this program dispositions this time-limited aging analysis in accordance with 10CFR54.21(c)(1)(iii)." Therefore, it is not clear which approach is being taken to address the TLAA for loss of tendon prestress.**

If the TLAA is performed in accordance with 10CFR54.21(c)(1)(ii), provide the following:

a. **the minimum required prestressing force value (MRV) for each group,**
b. **the predicted lower limit (PLL) prestressing force for each group of tendons (per NRC R.G. 1.35.1),**
c. **a plot comparing the measured prestressing forces obtained from each inspection and the PLL, trend lines of the measured prestressing forces for each group of tendons (per IN 99-10, the trend lines will be**

developed using a regression analysis considering individual tendon lift off forces rather than the average lift off forces for each group of tendons),

d. trend lines of the measured prestressing forces for each group of tendons (per IN 99-10, the trend lines will be developed using a regression analysis considering individual tendon lift off forces rather than the average lift off forces for each group of tendons),

e. extension of the PLL and trend lines for 60 years, and

f. description of corrective actions if item e above is unsuccessful.

If the TLAA is performed in accordance with 10CFR54.21(c)(1)(iii), the staff considers certain attributes to be significant for adequate management of the aging effects. These include:

a. Identification of parameters monitored.

b. Documentation of prestressing monitoring activities and trending of results.

c. Definition of acceptance criteria.

d. Identification of corrective actions when the acceptance criteria are not met.

e. Inclusion of plant specific and applicable industry operating experience.

In addition, provide a summary of the documentation which forms the basis for the SAR section addressing the tendon prestress calculations corresponding to the end of the 40-year service life.

The TLAA for loss of tendon prestress has been addressed in accordance with 10CFR54.21(c)(1)(iii). With regard to the five attributes considered by the NRC Staff to be significant in demonstrating that aging effects are adequately managed, refer to Appendix B, Section 4.3.6 which addresses these and other attributes of the ASME Section XI Inservice Inspection Program, IWL Inspections. Identification of parameters monitored is discussed under the heading 'Scope'. For documentation of monitoring activities and trending results, refer to the heading 'Method'. As stated in the program description for IWL Inspections, acceptance criteria are specified in IWL-3000. Corrective actions are performed in accordance with ASME Code Section XI, Subsection IWL, as indicated under the 'Industry Codes or Standards' heading. For operating experience, refer to the heading entitled the same.

A summary of the analysis for tendon prestress corresponding to the end of the 40-year service life for the CLB is provided in the ANO-1 SAR Section 5.2.4.2.1. A structural proof test was performed to verify the adequacy of the reactor building design. Since the overall structural integrity of the reactor building depends on the tensile strength of the tendons, the reactor building

was subjected to a pressure which created a stress in the tendons equivalent to the stress at design conditions due to dead load, pressure and temperature at the end of 40-years. The reactor building was proof tested at 115% of design pressure.

4.6-1 **LRA Section 4.6 describes the TLAA for fatigue of the reactor building liner plate and penetrations for ANO-1. The applicant concludes that the design-basis fatigue analysis remains valid for the extended period of operation, and that it meets the criteria of 10CFR54.21(c)(1)(i):**

The staff has determined that additional technical information is needed in the LRA, Section 4.6, to substantiate the conclusion. Therefore, the staff requests the following additional information:

(a) For the liner plate, describe how pressure cycling due to integrated leak rate tests has been included in the calculation of cumulative fatigue usage, and define the number of cycles assumed in the design-basis calculation and the projected number of cycles through the extended period of operation. Describe the basis for the projection through the extended period of operation.

In regard to thermal cycling, pursuant to ANO-1 SAR Section 5.2.1.4.7.3 and Section 4.6 of the ANO-1 LRA, the fatigue conditions considered in the design of the liner plate are 1) thermal cycling due to annual outdoor temperature variations, 2) thermal cycling due to interior temperature varying during heatup and cooldown of the reactor system, and 3) one thermal cycle due to design basis accident conditions. Loads associated with the integrated leak rate tests are not included in the fatigue evaluation of the liner plate under the ANO-1 CLB. Rather, the loads associated with the leak rate testing were implicitly accounted for by selecting a bounding number of thermal cycles (i.e., 500) for the fatigue evaluation.

Pressure cycling due to integrated leak rate tests is not applicable to cumulative fatigue. Section 5.2.1.4.7.1 of the ANO-1 SAR states that "all components of the liner which must resist the full design pressure . . . are selected to meet the requirements of Paragraph N-1211 of Section III, Nuclear Vessels, of the ASME Code". According to the ANO-1 SAR Section 5.2.4.4, following completion of the reactor building, the liner and its penetrations were tested at 115 percent of the design pressure to establish structural integrity. The initial leak rate test was conducted at 100% of the design pressure and at successively lower pressures to demonstrate leak tightness and establish a reference for future, periodic leak testing. This section of the ANO-1 SAR further states that there is no

reason to anticipate that the liner's effectiveness as a vapor barrier will progressively deteriorate during the life of the plant.

(b) Describe the basis for concluding that the number of heatup-cooldown cycles assumed in the design-basis (500) envelopes the number of such cycles projected through the extended period of operation. What is the projection based on plant operating experience to date?

Existing ANO-1 site outage reports reflect the number of heatup-cooldown cycles to date and support the assumptions presented in the ANO-1 SAR. The number of cycles to date is less than 115. Conservatively assuming three heatup-cooldown cycles per year, at the end of the period of extended operation, the projected number of cycles is less than 240, which is less than half of the design allowable cycles.

(c) For each liner plate penetration within the scope of this TLAA, define all transient pressure and temperature events considered in the design-basis calculation, define the number of occurrences assumed for design, and define the projected number of occurrences through the extended period of operation. Describe the basis for the projection through the extended period of operation.

Considering the full set of containment mechanical penetrations, time limited aging analyses (i.e., fatigue evaluations) are limited to the main steam and main feedwater mechanical penetrations. The discussion that follows is based on the information contained in Section 5.2.2.2.2 of the ANO-1 SAR. The loading conditions for the main feedwater and main steam mechanical penetrations are the same as those defined in Section 5.2.1.4.7.3 of the ANO-1 SAR for the liner plate; that is, 500 thermal cycles of reactor coolant system (RCS) startup and shutdown. The design number of thermal load cycles in the main feedwater and main steam systems (500) bound the number of design heatup and cooldown cycles of the reactor coolant system (240). The projected number of heatup and cooldown cycles for the ANO-1 RCS through 60 years of operation has been determined to be less than these original design limits, as discussed in Section 4.3 of the ANO-1 LRA. Therefore, the original fatigue evaluation of the main steam and main feedwater mechanical penetrations is valid for the period of extended operation.

The approach by ANO-1 to address fatigue of the main steam and main feed mechanical penetrations is consistent with the approach taken for Oconee, as documented in Section 4.2.1.3 of the NRC Safety Evaluation Report (NUREG-1723).

(d) **For the feedwater and main steam line penetrations, identify the evaluation boundary between the liner plate and the piping. Also describe the TLAA that addresses any part of the penetration not included in the liner plate TLAA.**

As stated in Section 2.4.1.1 of the ANO-1 LRA, the evaluation boundary for mechanical penetrations, which includes the feedwater and main steam line penetrations, consists of the penetration assembly and the weld to the process piping, but does not include the process piping within the penetration.

Fatigue of the main steam and main feed mechanical penetrations, including the weld that connects the penetration to the process piping, is addressed in the RAI response 4.6-1(c). Fatigue of the main steam line and main feed line piping is addressed through the aging management review of the main steam and main feed systems reported in Section 3.5 of the ANO-1 LRA. The main steam and main feed piping were designed to ANSI B31.1, which does not require an explicit fatigue analysis but does specify allowable stress levels based on the number of anticipated thermal cycles. Specifically, a stress reduction is not required in the design of piping that is not expected to experience more than 7,000 cycles. The main feed and main steam piping will not exceed 7,000 cycles over 60 years of operation and the initial fatigue evaluation of the main steam and feed lines is valid for the period of extended operation.

(e) **Provide the basis for the statement that the design of the reactor building penetrations meets the general requirements of ASME Section III for thermal cycling. Show that this statement is applicable to the feedwater and steam line penetrations.**

As described in the ANO-1 SAR, Section 5.2.2.1, "all (reactor building) penetrations are ... designed ... in accordance with the ASME Nuclear Vessel Code, Section III, Class B vessels." Accordingly, ANO-1 SAR Section 5.2.2.1.2, penetration closure anchorage connecting piping to the reactor building wall is designed to resist postulated pipe rupture, seismic and thermal loads. Section 5.2.2.2.1 of the ANO-1 SAR states that "penetrations are, in general, designed as pressure vessels" and that "thermal stresses are considered to be secondary stresses". Section 5.2.2.2.2 of the ANO-1 SAR further indicates that the feedwater and main steam lines are thermally insulated, restricting the temperature rise in concrete. Therefore, the design of reactor building penetrations meets the general requirements of ASME Section III for thermal cycling.