April 25, 2000

Mr. Craig G. Anderson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 SR 333
Russellville, Arkansas 72801

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE ARKANSAS NUCLEAR ONE, UNIT 1, LICENSE RENEWAL APPLICATION

Dear Mr. Anderson:

By letter dated January 31, 2000, Entergy Operations, Inc. (Entergy), submitted for the Nuclear Regulatory Commission's (NRC's) review an application pursuant to 10 CFR Part 54, to renew the operating license for Arkansas Nuclear One, Unit 1, (ANO-1). The NRC staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete its safety review. Specifically, the enclosed questions are from the Electrical and Instrumentation and Controls Branch and the Mechanical Engineering Branch regarding Sections 3.3.2.4, 3.3.2.5, 3.3.2.6, 3.7, 4.1, 4.2, 4.3, 4.4, 4.8.1, 4.8.2, and 4.8.3.

Please provide a schedule by letter, electronic mail, or telephonically for the submittal of your responses within 30 days of the receipt of this letter. Additionally, the staff would be willing to meet with Entergy prior to the submittal of the responses to provide clarifications of the staff's requests for additional information.

Sincerely,

/***RA***/

Robert J. Prato, Project Manager
License Renewal Project Directorate
Division of Regulatory Improvement Program
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosure: Request for Additional Information

cc w/encl: See next page

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| DATE | 4/25 /00 | 4/25 /00 | 4/25/00 |

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REQUEST FOR ADDITIONAL INFORMATION
ARKANSAS NUCLEAR ONE, UNIT 1, LICENSE RENEWAL APPLICATION,
SECTIONS 3.3.2.4, 3.3.2.5, 3.3.2.6, 3.7, 4.1, 4.2, 4.3, 4.4, 4.8.1, 4.8.2, and 4.8.3

3.3.2.4 REACTOR VESSEL

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

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

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

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

3.3.2.5 REACTOR VESSEL INTERNALS

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

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

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

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

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

3.3.2.6 ONCE-THROUGH STEAM GENERATORS

**3.3.2.6.1-1** Section 2.3.1.7 of the LRA states that "Secondary piping attached to the once-through steam generator nozzles, including the main and auxiliary feedwater headers and riser piping, is addressed in Section 2.3.4.2." However, Section 2.3.4.2 does not address main and auxiliary feedwater headers and riser piping. The lack of AFW header and raiser discussion in Section 2.3.4.2 will be addressed in the RAIs associated with that section. However, a description of the AMRs performed for the main and auxiliary feedwater headers and riser piping is needed for the staff's review of the OTSG.

**3.3.2.6.2.1-1** Section 3.7 of Appendix B of the LRA describes the wall thinning inspection program. This description of the wall thinning inspection program does not address OTSGs. Identify where in the LRA is wall thinning of the OTSG addressed, or provide a justification for excluding wall thinning as an aging effect for the OTSG.

**3.3.2.6.2.1-2** Flow of secondary fluid can cause high-frequency vibration and/or fluid elastic instability conditions of tubes and interaction with the tube support structures. Specifically, where the structural integrity of tube support plates and stabilizers are weakened due to loss of material, it may lead to tube failures. Past operating experience has indicated that this kind of fatigue failure was noted at Oconee leading to forced outages in 1994. Also, although outside-diameter stress-corrosion cracking (ODSCC) had not been identified as an active degradation mechanism in OTSGs, this should be considered as a potential effect of aging. Finally, a recent B&W owners group report, prepared by Framatome Technologies (Report No. 77-5003013-00, 2/99) on the OTSG internals, has indicated that flow-accelerated corrosion (FAC) can occur if there is a significant blockage of flow due to fouling. Identify where ODSCC and FAC related to the OTSG is discussed in the LRA, or provide a justification as to why fatigue, ODSCC, and FAC are not considered as applicable aging effects for the OTSG components.

**3.3.2.6.2.1-3** NRC IN 94-05, "Potential Failure of Steam Generator Tubes with Kinetically Welded Sleeves" discussed cracking that occurred in steam generator tubes sleeved with kinetically (explosively) welded sleeves supplied by B&W. Identify where cracking of kinetically welded sleeves is discussed in the LRA, or provide a justification as to why cracking of these sleeves are not considered as an applicable aging effect for the OTSG.

**3.3.2.6.2.2-1** Table 3.2-1 (pages 3-34 to 3-36) of the LRA lists aging management programs (AMPs) for components of an OTSG.

(a) Explain the type of inspection associated with the examination category B-Q of the ASME Section XI ISI-IWB applicable to tubes, plugs and sleeves.

(b) Explain why the Alloy 600 AMP is not included for managing aging of secondary side nozzles made of Alloy 600, (e.g., temperature sensing nozzles/connections).

**3.3.2.6.2.2-2** The ANO-1 steam generator integrity program is structured to meet the NEI Steam Generator (SG) Program Guidelines (NEI-97-06) and the plant's technical specification 4.18. According to Table 3.2-1 of the LRA, this program mitigates the aging effects in tubes, plugs, and sleeves only. No other SG internal components whose aging effects are managed by this AMP have been identified, although the applicant has clearly indicated in the scope that this AMP applies to the SG internals in addition to SG tubes, plugs, and sleeves. Clarifications are needed in the following areas:

(a) Scope: The program includes SG internals in the AMP. Identify the SG internal components that are included in the program.

(b) Aging Effects: The program includes aging effects for loss of material, cracking, and fouling. Confirm whether or not these aging effects include FAC, ODSCC, and fatigue.

(c) Method: Eddy current testing of tubes is mentioned. No discussion is provided on type of probes used for detecting different kinds of tube degradation. Also eddy current testing (ECT) has been used to detect degradation of other internal components such as tube support plates (TSPs) made of carbon steel. Clarify the inspection scope and expansion criteria for the ECT used at the site. Also, indicate if these techniques are industry-qualified and are performed by qualified personnel.

**3.3.2.6.2.2-3** Describe the applicable aging management activities in response to the recommended seven action items of GL 85-02 for addressing aging effects regarding SG tube integrity (e.g., FOSAR of loose parts, ISI of tubes, and water chemistry of both primary and secondary systems).

(a) Clarify how the ANO-1 SG integrity program includes recommended action items of GL 85-02.

(b) Review your steam generator tube failure history and identify each applicable aging effect over the life of the plant. For each applicable aging effect, identify the AMP that will be used to manage that aging.

In addition, as described in NRC IN 97-49, "B&W Once-through Steam Generator Tube Inspection Findings," degradation has been observed in OTSGs (e.g.,degradations at dented locations, the expansion transition region, freespan locations, sleeved regions, and sludge pile region). Specialized probes such as rotating probes may be required to reliably detect these indications. NRC IN 97-88, "Experiences During Recent Steam Generator Inspections," discusses the potential difficulties experienced by the applicant in qualifying and applying eddy current depth-sizing techniques. Describe the changes in the SG tube integrity program at ANO-1 to address tube degradation identified in NRC IN 97-49 and 97-88.

3.3.7: ELECTRICAL INSTRUMENTATION AND CONTROLS

**3.3.7.3.1-1** Section 3.7.1, Table 3.7-1, and Section 3.2 of Appendix B of the ANO-1 LRA describes the aging management review activities including the identification of splices, connectors, terminal blocks, and cables as passive electrical components, the associated materials and environments of these components that may lead to accelerated aging, the applicable aging effects, and the proposed aging management programs for these components. Environmental conditions that should be evaluated include temperature, radiation, humidity, water, electrical, mechanical, vibration, chemical, electrochemical, and contaminants. The staff was unable to identify where the applicant addresses vibration and contaminant environmental conditions associated with electrical splices, connectors, terminal blocks and cables in Section 3.7.1, Table 3.7-1, or Section 3.2 of Appendix B, of the ANO-1 LRA. Identify specifically where the vibration and contaminant environmental conditions are addressed, and, if not, provide a justification for not considering these environments in determining the applicable aging effects and for excluding these aging effects from an aging management review.

**3.3.7.3.1-2** Identify where in the LRA is the AMR for the following electrical components:

Electrical components whose bodies perform a pressure boundary function such as the following:

- instrumentation racks, frames, panels and enclosures
- electrical panels, racks, cabinets and other enclosures
- selected cables
- electrical busses
- insulators

The staff was unable to identify where the applicant specifically addresses these components in Section 3.7, Table 3.7-1, or Section 3.2 of Appendix B. Identify specifically how these electrical components will be addressed in Section 3.7 and Section 3.2 of Appendix B.

A similar question was asked during a conference call on March 30, 2000, relating to the review of Section 2.5 of the LRA. It was noted that some of these components may have been addressed in the structural portion of the application, but we were unable to definitively identify before the end of that telecommunication where in the LRA was the AMR for any of theses components. Identify where the AMR for these components can be found in the LRA, submit an AMR for these components, or provide a justification for excluding these components from an AMR.

**3.3.7.3.1-3** Section 3.7.1, Table 3.7-1, and Section 3.2 of Appendix B of the ANO-1 LRA describes the aging management review activities including the identification of splices, connectors, terminal blocks, and cables as passive electrical components, the associated materials and environments of these components that may lead to accelerated aging, the applicable aging effects, and proposed aging management programs (visual inspection) for these components.

(a) Identify the means for monitoring the applicable aging effects for any inaccessible in-scope cables, splices, connectors, and terminal blocks.

(b) Provide a discussion of the acceptance criteria and corrective actions when the acceptable criteria are not met for visual inspection of in-scope cables, splices, connectors, and terminal blocks.

## 4.1 IDENTIFICATION OF TIME-LIMITED AGING ANALYSES

**4.1-1** Table 4.1-1 of the LRA provides a list of time-limited aging analyses (TLAAs) that are applicable to RVI. The table indicates that a TLAA related to the acceptability of the RVI under LOCA and seismic loading is addressed in Section 3.5, Appendix B of the LRA which references topical report BAW-2248. The topical report indicated that these TLAAs will be resolved on a plant-specific basis per 10 CFR 54.21(c)(1)(iii) based on the results and conclusion of the RVI AMP. The RVI AMP will provide the necessary data on neutron embrittlement for fracture resistance of the RVI components. This was identified as Renewal Applicant Action Item 12 in the staff safety evaluation concerning BAW-2248. Table 2.3-5 of the LRA contains the response to Renewal Action Item 12 for the RVI. The response indicates that a plant-specific analysis will be performed to demonstrate that under LOCA and seismic loading, the internals have adequate ductility to absorb the local strain intensity during the extended period of operation. However, the LRA does not address how and when this TLAA will be evaluated. Provide a discussion of the program plan for evaluating the TLAA including the schedule for completion of the evaluation.

## 4.2 TLAA ON REACTOR VESSEL NEUTRON EMBRITTLEMENT

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

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

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

## 4.3 TLAA ON METAL FATIGUE

**4.3.3-1** Section 4, Table 4.1-1, of the LRA lists the components that have TLAAs. The table indicates that the TLAAs associated with fatigue and flaw growth are addressed in Section 4.3 of the LRA. However, Section 4.3 of the LRA does not contain a specific reference to the steam generators. Indicate how each TLAA listed in Table 4.1-1 for the steam generators is addressed.

**4.3.3-2** Section 4.3.3.4 of the LRA contains a discussion of environmentally assisted fatigue. The discussion indicates that an effective approach to manage this issue is to identify the locations that are most susceptible to failure from thermal fatigue and include these locations in the augmented inservice inspection program. The application references the six locations listed in NUREG/CR-6260, “Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plants.” The application further indicates that three locations have been evaluated in BAW-2251, “Demonstration of the Management of Aging Effects for the Reactor Vessel.” Provide the following information for the remaining three locations; the pressurizer surge line, the makeup/HPI nozzles, and the decay heat removal Class 1 piping:

(a) An assessment of the potential for fatigue cracking of these locations considering the assessment of the environmental fatigue data presented in NUREG/CR-5704, “Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels.”

(b) A discussion of the augmented inservice inspection planned for these locations. This discussion should address the specific areas of the components that will be inspected, the method used to qualify the inspections for areas not adjacent to weld joints, and the frequency of the inspections given the assessment performed for item a.

(c) Describe how the augmented inspections satisfy the applicable requirements of 10 CFR Part 54.21 with regard to demonstrating that the effects of thermal fatigue for these three locations will be adequately managed so that the intended function(s) will be maintained for the period of extended operation.

**4.3.3-3** Section 4.3.4.4 of the LRA describes actions taken in response to NRC Bulletin 88-08, “Thermal Stresses in Piping Connected to Reactor Coolant Systems.” The application indicates that stratified flows were identified in lines that were monitored at ANO-2. The application indicates that, because of the ANO-2 experience, further monitoring and evaluation of four ANO-1 lines was performed. The application further indicates that temperature monitoring and evaluations have demonstrated that the ANO-1 lines are qualified for their service conditions. Describe in more detail the measurements, calculations, and criteria that led to the conclusion that the four ANO-1 lines are qualified for their service conditions.

**4.3.3-4** As discussed in Section 4.3.4.4 of the LRA, the applicant committed to perform enhanced ultrasonic examination of 17 high-pressure injection (HPI) welds and two segments of HPI piping as part of the ANO-1 ten-year interval ISI plan in response to Bulletin 88-08, “Thermal Stresses in Piping Connected to Reactor Coolant Systems.” The LRA further indicates that the scope of the HPI ISI inspections was subsequently modified as a result of the implementation of ASME Code Case N-560, “Alternative Examination Requirements in Class 1, Category B-J Piping Welds Section XI, Division 1.” Describe the modifications to the scope of the enhanced ultrasonic ISI examination of 17 HPI welds and two sections of HPI piping that were made as a result of the implementation of Code case N-560 at ANO-1.

The LRA discussion regarding NRC Bulletin 88-08 concludes: “This issue has therefore been resolved for the period of extended operation”. The intent of this statement in the LRA is not clear to the staff. Specify precisely what issue was resolved for the period of extended operation. Also, describe how the ANO-1 inspections of the HPI piping and welds meet the applicable requirements of 10 CFR Part 54.21 with regard to demonstrating that the effects of thermal fatigue of the HPI piping will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

**4.3.3-5** As discussed in Section 4.3.4.4 of the LRA, the applicant originally committed to performing enhanced ultrasonic examination of two elbows of the surge line as part of an ANO-1 ten-year interval ISI plan in response to Bulletin 88-11, “Pressurizer Surge Line Thermal Stratification.” Subsequently, the scope of the ISI was changed based on an ANO-1 risk analysis performed consistent with the requirements of ASME Code Case N-560, “Alternative Examination Requirements for Class 1, Category B-J Piping Welds, Section XI, Division 1". Describe the modifications to the scope of the enhanced ISI examinations of the two surge line elbows that were made as a result of the implementation of Code Case N-560 at ANO-1.

The LRA discussion regarding NRC Bulletin 88-11 concludes: “This issue has therefore been resolved for the period of extended operation”. The intent of this statement in the LRA is not clear to the staff. Specify precisely what issue was resolved for the period of extended operation. Also, describe how the ANO-1 inspections of the surge line meet the applicable requirements of 10 CFR Part 54.21 with regard to demonstrating that the effects of thermal fatigue of the surge line piping will be adequately managed so that the intended function(s) will be maintained consistent with the CLB for the period of extended operation.

**4.3.3-6** Section 4.3.3.4 of the LRA contains a discussion of HPI/MU Nozzle cracking at B&W plants. The LRA indicates that, in order to manage cracking effects in the HPI/MU nozzle, ultrasonic testing of the knuckle region of the HPI nozzles will be performed every fifth refueling cycle, and that radiography of the thermal sleeves will continue through the period of extended operation. The scope and frequency of radiographic testing  was not specified in the application.  Provide the scope and frequency of the thermal sleeve radiography proposed for the period of extended operation.

4.4 ENVIRONMENTAL QUALIFICATION

4.4-4 Section 4.4.18 of the application on General Atomic Radiation Detectors is based on option (i) of 10 CFR Part 54.21 (c)(1) to demonstrate that the analyses remain valid for the period of extended operation.

Please provide a summary of the thermal and radiation analyses used to illustrate the basis upon which the qualified life remains valid for the period of extended operation. Include a description of the materials of construction, and a comparison of design specifications with actual service conditions.

4.4-5 The following sections of the application are based on option (ii) of 10 CFR  art 54.21(c)(1) to demonstrate that the analyses have been projected to the end of the period of extended operation.

- **4.4.1 Allis Chalmers Motors** ★
- 4.4.2 Anaconda Instrumentation Cable, FR-EP Insulation
- 4.4.3 Anaconda Control and Power Cable, EP Insulation
- **4.4.4 Anaconda EPR Insulated Instrumentation, Control/Power Cable** ★
- 4.4.8 Buchanan Terminal Blocks, Outside Reactor Building
- **4.4.9 Buchanan Terminal Blocks, Inside Reactor Building** ★
- **4.4.10 Conax Thermocouples** ★
- **4.4.11 Conax Resistance Temperature Detectors** ★
- 4.4.12 Conax Multipin Connector
- **4.4.13 Conax Electrical Penetration Assemblies** ★
- 4.4.14 Conax Electrical Connection Seal Assembly
- 4.4.15 Conax Electrical Feedthrough Adapters
- 4.4.16 Eaton Flame Retardant Ethylene Propylene Diene Monomer Insulated Cable
- 4.4.17 Gems De Laval Level Sensors
- 4.4.19 General Electric Terminal Blocks
- **4.4.21 Limitorque Motor Operated Valve Actuators; Alternating Current/Inside Reactor Building (most applications)** ★
- 4.4.22 Limitorque Motor Operated Valve Actuators; Alternating Current/Outside Reactor Building
- 4.4.23 Limitorque Motor Operated Valve Actuators; Direct Current/Outside Reactor Building
- 4.4.26 NAMCO EA-740 Limit Switches with NAMCO Connectors
- 4.4.27 NAMCO EA-740 Limit Switches
- 4.4.28 NAMCO Quick Connectors
- 4.4.29 Okonite 5 kV Power Cable with EPR Insulation and an Okolon Jacket

- 4.4.30 Okonite 2 kV Power and Control Cable with Okonite or Okoguard Insulation and Okoprene or Okolon Jackets
- **4.4.31 Okonite 600V Power Cable with Okonite Insulation and an Okolon Jacket (most applications)** ★
- 4.4.32 Okonite 600V Power Cable with FMR Insulation (most applications)
- 4.4.33 Okonite T-95 and No. 35 Splicing Tapes (most applications)
- 4.4.34 Raychem 600V Flamtrol XLPE Cable
- **4.4.35 Raychem Cable Splice and Jacket Repair Tape (type NJRT)** ★
- 4.4.36 Raychem Cable Splices (types WCSF-N, NPK, NMCK, ANK, etc.)
- 4.4.37 Reliance Electric, Electric Motors
- **4.4.38 Rockbestos Coaxial Cable (most applications)** ★
- 4.4.39 Rockbestos Firewall III Irradiation Cross-Linked Polyethylene Cable
- **4.4.40 Rockbestos Firezone R Silicone Rubber High Temperature Cable (some applications)** ★
- 4.4.41 Rockbestos Firewall III Chemically Cross-Linked Polyethylene Cable
- 4.4.44 Rotork Motor Operated Valve Actuators, Model NA1
- 4.4.45 Target Rock Solenoid Operated Valves (Report 2375)
- 4.4.47 Target Rock Modulating Solenoid Operated Valves (Report 3414)
- **4.4.48 Target Rock Solenoid Operated Valves (Reports 2375 and 1827)** ★
- 4.4.49 TEC Valve Flow Monitoring System (some subcomponents)
- 4.4.50 TEC Reactor Vessel Level Monitoring System (some subcomponents)
- 4.4.51 Weed Resistance Temperature Detectors
- 4.4.52 Dow-Corning 3145 Silicone Sealant
- **4.4.55 Westinghouse Motors, Models TBFC and SBDP** ★
- 4.4.56 Babcock & Wilcox Core Exit Thermocouples (pin half connector with mineral- insulated cable)
- 4.4.57 Gamma Metrics Neutron Detectors and Cable Assemblies (organic cable)
- **4.4.58 Brand Rex Cross-Linked Polyethylene Coaxial Cable** ★
- 4.4.59 Brand Rex Cross-Linked Polyethylene Power and Control Cable
- 4.4.60 NDT International Acoustic Sensor, Connector and cable
- 4.4.61 American Insulated Wire 600V Instrumentation Cable
- 4.4.62 American Insulated Wire 600V Power and Control Cable
- 4.4.63 AMP Pre-insulated Butt Splices
- 4.4.64 EGS Quick Disconnect Electrical Connectors (except connector o-rings)
- 4.4.65 EGS Grayboot Electrical Connectors
- 4.4.67 Valcor Model V526-5961-1 Solenoid Operated Valve
- 4.4.68 General Cable Corporation 5 kV Power Cable

Provide analyses for the items marked with an asterisk (★) to illustrate the basis upon which the analyses were projected to the end of the period of extended operation. Each analysis should include a discussion on the analytical methods used and why they are applicable, the data collection and reduction methods used, the underlying assumptions, the acceptance criteria, and the corrective actions if the acceptance criteria are not met. In addition, please provide a list of all equipment for which an activation energy different from that in the current qualification basis documents was used in the reanalysis, along with the justification for using a different value.

4.4-6 The following sections of the application are based on option (iii) of 10 CFR Part 54.21(c)(1) to demonstrate that the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

- 4.4.5 ASCO Solenoid Valves, Outside Reactor Building (some applications)
- 4.4.6 ASCO Solenoid Valves, Inside Reactor Building
- 4.4.7 Boston Insulated Wire, Instrumentation, Control, and Power Cable
- 4.4.20 ITT/General Controls Electro-hydraulic Actuators
- 4.4.21 Limitorque Motor Operated Valve Actuators; Alternating Current/Inside Reactor Building (some applications)
- 4.4.24 NAMCO EA-170 Limit Switches
- 4.4.25 NAMCO EA-180 Limit Switches
- 4.4.26 NAMCO EA-740 Limit Switches with NAMCO Connectors
- 4.4.31 Okonite 600V Power Cable with Okonite Insulation and an Okolon Jacket (some applications)
- 4.4.32 Okonite 600V Power Cable with FMR Insulation (some applications)
- 4.4.33 Okonite T-95 and No. 35 Splicing Tapes (some applications)
- 4.4.38 Rockbestos Coaxial Cable (some applications)
- 4.4.40 Rockbestos Firezone R Silicone Rubber High Temperature Cable (some applications)
- 4.4.42 Rosemount Model 1153 Series D Pressure Transmitters
- 4.4.43 Rosemount Model 1154 Pressure Transmitters
- 4.4.46 Target Rock Solenoid Operated Valves (Reports 2375 and 3996)
- 4.4.49 TEC Valve Flow Monitoring System (some subcomponents)
- 4.4.50 TEC Reactor Vessel Level Monitoring System (some subcomponents)
- 4.4.53 Westinghouse Hydrogen Recombiners
- 4.4.54 Westinghouse Motors, Model ABDP
- 4.4.56 Babcock & Wilcox Core Exit Thermocouple (except the pin half connector with the mineral-insulated cable)
- 4.4.57 Gamma Metrics Neutron Detectors and Cable Assemblies (except organic cable)
- 4.4.64 EGS Quick Disconnect Electrical Connectors (connector o-rings)
- 4.4.66 Valvor Model V526-5683 Solenoid Operated Valve

For one representative sample of each type of component (i.e., cable, connector, instrument transmitter, etc.), please provide the following information for the option chosen:

Replacement - Describe the activities for replacement of equipment qualified to 10 CFR 50.49 and any sound reasons to the contrary (Regulatory Guide 1.89, Rev. 1) that will be used for replacement equipment.

Refurbishment - Describe the activities that will result in the equipment being returned to its original (like new) qualified condition.

Reanalysis - Provide the analytical methods, data collection and reduction methods, underlying assumptions, acceptance criteria, corrective actions if the acceptance criteria are not met, and the period of time prior to the end of qualified life when reanalysis will be completed.

4.8.2 TLAA ON METAL FATIGUE REACTOR VESSEL INCORE INSTRUMENTATION NOZZLE - FLOW INDUCED VIBRATION ENDURANCE LIMIT

**4.8.2.3-1** Section 4.8.2 of the LRA indicates that flow induced vibrations of the reactor vessel incore instrumentation nozzles was identified as an additional TLAA for ANO-1 that had not been identified in BAW-2251. The application indicates that BAW-10051, "Flow Induced Vibration Endurance Limit Assumptions," contains a comparison of the calculated stress values for the incore instrumentation nozzles to the endurance limit (stress values). The endurance limit values for the current licensing basis of 40 years used an assumption of $10^{12}$ cycles. The application indicates that, after the number of cycles was increased to that expected after 60 years of operation, and the component stress levels were compared to the recalculated endurance limit, the stress levels are acceptable. Provide the basis for the new fatigue endurance limit. Also, provide the comparison of component stress level and the recalculated endurance limit for the expected number of fatigue cycles for 60 years of operation.

4.8.3 TLAA ON LEAK-BEFORE-BREAK

**4.8.3.3-1** In Section 4.8.3, NUREG/CR-6177 is cited as providing information relevant to the leak-before-break (LBB) reassessment of the **CASS** reactor coolant pump (RCP) inlet and exit nozzles in the area of the welded joint to the austenitic stainless steel 28-inch transition piece. Confirm that the δ-ferrite level of the CASS RCP (made from statically-cast CF8M) nozzles is within the bounds of applicability for the NUREG/CR-6177 correlations. If not, explain why the information in NUREG/CR-6177 applies to your material, or provide other information which supports your analysis.

cc:

Executive Vice President
& Chief Operating Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Director, Division of Radiation
Control and Emergency Management
Arkansas Department of Health
4815 West Markham Street, Slot 30
Little Rock, AR 72205-3867

Winston & Strawn
1400 L Street, N.W.
Washington, DC 20005-3502

Manager, Rockville Nuclear Licensing
Framatone Technologies
1700 Rockville Pike, Suite 525
Rockville, MD 20852

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 310
London, AR 72847

Mr. Doug Levanway
600 Heritage Blvd.
Jackson, MS 39201

Vice President, Operations Support
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Wise, Carter, Child & Caraway
P. O. Box 651
Jackson, MS 39205

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

County Judge of Pope County
Pope County Courthouse
Russellville, AR 72801

Mr. Douglas J. Walters
Nuclear Energy Institute
1776 I Street, N.W., Suite 400
Washington, DC 20006-3708

Mr. Garry Young
Entergy Operations, Inc.
Arkansas Nuclear One
1448 SR 333 GSB-2E
Russellville, Arkansas 72802