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See Reports

SUBJECT: Forwards responses to RAI re review of 980706 application for renewal of licenses DPR-38,DPR-47 & DPR-55.Response to potential open item 3.4.5-9 contains license renewal applicant items listed in SER submitted on 990426.

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M. S. Tuckman
Executive Vice President
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May 10, 1999

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U. S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: License Renewal
Response to Requests for Additional Information
Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

By letter dated July 6, 1998, Duke Energy Corporation submitted an Application for Renewed Operating Licenses for Oconee Nuclear Station, Units 1, 2, and 3 (Application). Exhibit A of the Application contains the technical information required by 10 CFR Part 54.
The NRC staff is reviewing the information provided by Duke Energy in the Application and by several letters and telephone requests identified areas where additional information is needed to complete its review. Duke Energy provided responses to these requests for additional information by letters dated December 14, 1998, January 25, 1999, February 8, 1999, February 17, 1999, March 18, 1999, March 29, 1999, and April 6, 1999.

By letter dated April 8, 1999, the staff provided 62 potential open items that had been identified during the process of preparing its Safety Evaluation Report regarding the Oconee Application. Duke Energy responses to these potential open items are provided in Attachment 1. By letter dated April 16, 1999, the staff provided 17 new potential open items and revised the text of two potential open items that had been provided in the April 8 letter (Potential Open Items 2.5.8-1(c) and 2.7-12). Duke Energy responses to the 17 new potential items are provided in Attachment 2 while the responses to the two revised potential open items are contained in Attachment 1 where the questions were initially conveyed.

By letter dated April 26, 1999, the staff provided its Safety Evaluation Report for BAW-2251, "Demonstration of the Management of Aging Effects for the Reactor Vessel," June 1996. The response to Potential Open Item 3.4.5-9 (New) provided in Attachment 1 contains the Oconee-specific responses to the license renewal applicant items listed in the above Safety Evaluation Report.

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Responses to 13 items (or portions of items) are being deferred. Duke has decided to defer addressing Potential Open Item 3.4.6-3 (Items A and C), 3.4.6-4, and 3.4.6-5 because the final Safety Evaluation Report for BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals," July 1997, is not available. In addition, Duke has decided to defer the follow-up responses to RAI 3.4.10-1, 4.25-5, 4.25-6, 4.25-7, 4.25-8, 5.3.1-1 5.4.1-6 (portions) and the SSF HVAC portions of responses to RAIs 4.17-4, 4.17-6 and 4.17-7 until after the staff issues the SER for Oconee License Renewal because the responsible Duke engineers were not available to assist in the preparation of a response. Finally, a meeting to discuss Potential Open Item 2-2 (New) is currently scheduled for May 11, 1999. Duke will provide additional information after this meeting, as required.

Attachment 1 contains the following statements that revise an existing commitment:

- The description of the Oconee Master Integrated Reactor Vessel Surveillance Program will be revised per our response to Potential Open Item 3.4.5-9 (New), Renewal Applicant Action Item #13;
- The description of the Reactor Building Spray System Inspection will be revised as described in our responses to RAIs 4.3.9-7 and 4.3.9-8; and
- As discussed further in our response to RAI 4.3.8-8 (New), the original commitment to perform a one-time assessment of ten specific preventive maintenance activities is being withdrawn.

If there are any questions, please contact Bob Gill at 704-382-3339.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

M. S. Tuckman, being duly sworn, states that he is Executive Vice President, Nuclear Generation Department, Duke Energy Corporation, that he is authorized on the part of said Company to sign and file with the U. S. Nuclear Regulatory Commission these responses to NRC requests for additional information concerning the Application to Renew the Facility Operating Licenses of Oconee Nuclear Station submitted by letter dated July 6, 1998; and that all statements and matters set forth herein are true and correct to the best of his knowledge and belief. To the extent that these statements are not based on his personal knowledge, they are based on information provided by Duke employees and/or consultants. Such information has been reviewed in accordance with Duke Energy Corporation practice and is believed to be reliable.

M. S. Tuck

M. S. Tuckman, Executive Vice president
Duke Energy Corporation

Subscribed and sworn to before me this 10TH day of MAY 1999.

Mary P. Nelms
Notary Public

My Commission Expires:

JAN 22, 2001

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Attachment 1

Oconee Nuclear Station
Application for Renewed Operating Licenses
Responses to NRC Identified Potential Open Items

(provided by letter dated April 8, 1999)

May 10, 1999

Attachment 1

Oconee Nuclear Station
Application for Renewed Operating Licenses
Responses to Additional NRC Identified Potential Open Items
(provided by letter dated April 8, 1999)

May 10, 1999

Potential Open Item related to RAI G-1 and 4.13-1

RAI G-1 requested clarification regarding Duke's commitment to extend 10 CFR Part 50, Appendix B requirements for corrective actions, confirmation process, and administrative controls to cover non-safety related structures and components subject to an aging management review (AMR) program. Similarly, RAI 4.13-1 requested a description of the methodology and processes that will be used by Duke to address corrective actions, confirmation process, and administrative controls for non-safety related SSCs subject to an AMR program at Oconee in a manner consistent with the guidance in the SRP.

As described in Duke's RAI response letter to the NRC dated February 17, 1999, in accordance with the provisions of the SRP they have elected to include the non-safety related structures and components in a separate renewal program that is summarized in the FSAR Supplement. Specifically, Duke stated that the program elements for corrective action, including confirmation, and administrative controls will be clarified for each program that addresses the aging effects on non-safety related structures and components within the scope of license renewal to assure these elements are properly addressed. These updated program elements will be summarized for the applicable programs in the Oconee FSAR.

Therefore, pending the development and implementation of this separate renewal program to address corrective action, confirmation processes, and administrative controls for non-safety related structures and components that are subject to an AMR program, this item is identified as an open item.

Response

Duke's response to RAI G-1 was misleading with respect to a "separate" renewal program for non-safety related structures and components. Duke does not plan to implement a new, separate program specifically for license renewal to address corrective actions, confirmation process, and administrative controls for non-safety related structures and components subject to aging management review at Oconee. Rather, plant processes currently in place, particularly the plant Problem Investigation Process (PIP) which forms the basis of the aging management program corrective action element, address these aspects of non-safety related structures and components.

Oconee Nuclear Station
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Responses to Additional NRC Identified Potential Open Items
(provided by letter dated April 8, 1999)

May 10, 1999

The attributes of each program listed in response to RAI G-1 that contain non-safety related structures and components apply to both the non-safety as well as the safety related structures and components. For example, if during the implementation of a program, the acceptance criteria are not met, the corrective action process must be entered, regardless of safety classification of the component. Nuclear station directives govern the corrective action process. The corrective action process is implemented in part by the PIP process. Details about the PIP process are described in response to RAI 4.3.9-4.

Oconee Nuclear Station
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Responses to Additional NRC Identified Potential Open Items
(provided by letter dated April 8, 1999)

May 10, 1999

RAI 1.5.5-1

This RAI involves the resolution of GSI-190. Duke has referenced the industry studies performed by EPRI to resolve this issue. We have provided NRC staff comments concerning the EPRI reports in a November 2, 1998, letter, to NEI. We had additional discussions with EPRI and NEI regarding the industry response to the staff comments on March 23, 1999. NEI plans to provide a response to the staff comments the week of April 14. This item remains open until either GSI-190 is resolved and the resolution implemented at Oconee, or Duke provides a basis which demonstrates the current licensing basis will be adequately managed through the extended period of operation.

Status: Waiting for NEI submittal

Response

RAI 1.5.5-1 involves issues associated with GSI-190. As noted in the staff's letter dated April 8, 1999 related to the Oconee Potential Safety Evaluation Report (SER) Open Items, the issues surrounding the GSI-190 topic are larger than Duke and involve NEI feedback to the NRC. Duke notes that the NEI feedback was provided to the NRC in a April 8, 1999 NEI letter from D.J. Walters, NEI, to C.I.Grimes, NRC.

Duke has raised the significance of the staff's dealings with GSI-190 in relation to the Oconee renewal application in several of the monthly NRC/Duke license renewal management meetings. This issue remains a topic for management attention. In the Duke response to RAI 1.5.5-1 (contained in Duke letter dated February 17, 1999, Attachment 1), Duke offered a basis to demonstrate that the current licensing basis of Oconee will be adequately managed through the extended period of operation. The staff has not yet offered their feedback on this response, but has provided this potential open item. Duke seeks to understand the NRC's concerns surrounding GSI-190 and will wait until the Oconee License Renewal Safety Evaluation Report is available in order to better understand the staff's thinking on this issue.

Oconee Nuclear Station
Application for Renewed Operating Licenses
Responses to Additional NRC Identified Potential Open Items
(provided by letter dated April 8, 1999)

May 10, 1999

Potential Open Item 2-2 (New)

A 10 CFR Part 54 scoping issue was raised during a March 11, 1999, meeting with the staff. In that meeting Duke reiterated its belief that the set of design basis events contained in Chapter 15 of the Oconee updated final safety analysis report (UFSAR) complies with the requirements of 10 CFR 54.4(a)(1) and meets the definition of 10 CFR 50.49(b)(1). The staff does not agree with this position and intends to raise this issue to the license renewal steering committee for resolution. Pending satisfactory resolution of this issue this item will remain an open item.

Response

A meeting to discuss this item is scheduled for May 11, 1999. Duke will provide additional information after this meeting, as required.

RAI 2.2-7

In response to RAI Question 2.2-7 on control room radiation monitors, Duke stated that the radiation monitors shown on the OLRP series of drawings (OLRFD-116C-1.1, OLRFD-124B-1.5, OLRFD-133A-1.5) do not support any system intended functions as defined in §54.4(a)(1), (2), (3), or (b). Therefore, the radiation monitors are excluded from the scope of license renewal. The staff does not agree with Duke's assessment based on the reasons stated below. Additionally, Duke needs to address other OLRP series drawings (OLRFD-116J-1.2 for Units 1 and 2 control room and OLRFD-116J-3.2 for Unit 3 control room) showing radiation monitors (1,2RIA-39 and 3RIA-39).

The design basis function of radiation monitors (1,2RIA-39 and 3RIA-39) is stated in FSAR Section 9.4.1.1, Design Bases, which states that "The radiation monitor, RIA-39, has a continuous sample of control room air pumped through the detector. High radiation level and loss of sample flow are annunciated at which time the operator energizes the outside air filter trains. The outside air filter trains act to filter particulate matter from the outside air to minimize uncontrolled infiltration into the Control Room." The continuous radiation monitoring is a safety related function which cautions the control room operators to manually activate the filtration train of the Control Room Pressurization and Filtration System for Units 1, 2, and 3 control rooms in a given accident conditions to filter outside air for control room intake in order to pressurize them and thus meeting TMI Action Plan Item III.D.3.4, Control Room Habitability, requirements through dose analysis. The continuous radiation monitoring is supported by UFSAR Section 9.4.1.3 which states that "Return air from the Control Room is continuously monitored by a radiation monitor before recirculating back to the Control Room. A high radiation level will alert the operators to energize the outside air filter trains". TMI Action Plan Item III.D.3.4 retrofits the requirements of General Design Criterion (GDC) 19, Control Room, during bounding accident conditions and may be required during the severe accidents, transients, or station blackout conditions.

Response

The return air from the control room is continuously monitored by radiation monitors RIA-39 (Units 1 and 2 control room) and 3RIA-39 (Unit 3 control room). This continuous radiation monitoring is a nonsafety related function. Radiation monitors RIA-39 and 3RIA-39 are not safety related. Oconee dose analyses do not rely upon operation of the radiation monitors to prompt operator actions required for design basis event mitigation. It is true that operators will energize the outside filter trains when the monitors alarm. Operation of the monitors is not relied upon for the successful mitigation of any design basis event. Likewise, failure of the monitors will not prevent the successful mitigation of any design basis event. The monitors are also not relied upon to meet the requirements of the regulated events of §54.4(a)(3). Therefore, the radiation monitors are not within the scope of license renewal.

Radiation monitors shown on the OLRP series of drawings OLRFD-116C-1.1, OLRFD-124B-1.5, OLRFD-133A-1.5 are also all nonsafety related. Their function is not required for the successful mitigation of any design basis event. Likewise, failure of the monitors will not prevent the successful mitigation of any design basis event. The monitors are also not relied upon to meet the requirements of the regulated events of §54.4(a)(3). Therefore, the radiation monitors are not within the scope of license renewal.

RAI 2.5.8-1

(a) With respect to the component level scoping of the HVAC systems, Duke stated that (1) no heating coils, cooling coils, compressors, valves and air dryers are found within the license renewal portions of the Auxiliary Building Ventilation System and (2) chilled water system components including compressor, valves, and air dryer are not a part of the Control Room Pressurization and Filtration System on Figure 9-24 that is within the scope of license renewal. Duke needs to provide the bases corresponding to 10 CFR 54 requirements that justify the exclusions of these components from the scope of the license renewal.

(b) With respect to the control room pressurization and filtration system cooling function, Duke stated that the chilled water system is not required to support the control room pressurization and filtration system function. Duke needs to provide justification of why the cooling function is not safety-related, and thus, within the scope of license renewal. Typically, these components are required to conform with TMI Action Plan Item III.D.3.4, Control Room Habitability which retrofits the General Design Criterion (GDC) 19, Control Room, requirements

during bounding accident conditions and maybe required during severe accidents, transients, or station blackout conditions.

By letter dated April 16, 1999, the staff withdrew the version of question (c) that was previously provided by letter dated April 8, 1999 and replaced it with the following question:

(c) With respect to the component level scoping of the HVAC systems concerning the heating coils and valves, the applicant stated that (1) the heating coils are subject to an aging management review and are identified in Table 2.5-13 for the CRPFS and (2) valves for the PRVS are subject to an aging management review and are listed in Table 2.5-13. Provide justifications for why these valves are excluded from AMR for the auxiliary building ventilation system and Pressurization and Filtration System.

Response

The Chilled Water System is a nonsafety-related system that provides chilled water to the Control Room Pressurization and Filtration System cooling coils for air conditioning of the control room area, cable rooms, and equipment rooms. For certain design basis events, the Control Room Pressurization and Filtration System must maintain a positive pressure in the control room for accident conditions. Maintaining a positive pressure in the control room does not require air conditioning, and therefore, the Chilled Water System is not required for design basis event mitigation and does not meet the §54.4(a)(1) scoping criteria.

A failure of the nonsafety-related Chilled Water System does not affect the Control Room Pressurization and Filtration System function of maintaining a positive pressure in the control room for accident conditions. In addition, no portion of the Chilled Water System is classified Oconee Pipe Class D for seismic II/I concerns. Therefore, the Chilled Water System does not meet the §54.4(a)(2) scoping criteria.

The Control Room Pressurization and Filtration System is credited with maintaining a suitable environment in the control room during a fire event and providing smoke removal for a fire in the control room. Neither of these functions requires the air conditioning function supported by the Chilled Water System. In addition, the Control Room Pressurization and Filtration System and the supporting Chilled Water System do not perform a function in support of pressurized thermal shock, ATWS, equipment qualification, or station blackout. Therefore, the Chilled Water System does not meet the §54.4(a)(3) scoping criteria.

Since the Chilled Water System does not perform a function that meets of the scoping criteria of §54.4(a)(1), (a)(2), or (a)(3), the Chilled Water System is not within the scope of license renewal.

The Penetration Room Ventilation System is constructed of pipe and valves. The Auxiliary Building Ventilation System and the Control Room Pressurization and Filtration System are constructed of ductwork and dampers. A review of the license renewal portions on flow diagram series OLRFD-116G for the Auxiliary Building Ventilation System and OLRFD-116J for the Control Room Pressurization and Filtration System did not identify any valves within scope. Dampers, a ventilation valve, are shown within the license renewal portions of the Auxiliary Building Ventilation System and Control Room Pressurization and Filtration System. Per §54.21(a)(1)(i), dampers are within the scope of license renewal but are not subject to an aging management review. Since there are no valves within the license renewal evaluation boundaries of the Auxiliary Building Ventilation System and Control Room Pressurization and Filtration System and dampers are not subject to an aging management review, valves and dampers are not listed in Table 2.5-13 of Exhibit A of the Application.

RAI 2.6-4

Section 2.6.6.1.2 of the application identified insulated cables and connections used for fire detectors as part of the fire detection system and excluded them from an aging management review because they are replaced based on a performance or condition program. In response to RAI 2.6-4, Duke referenced SOC Section III.f.(i)(b) and 10 CFR 54.21(a)(1)(ii) as the basis for excluding fire detector cables from an aging management review. However, Duke also stated that fire detector cables are not physically different than other insulated cables. 10 CFR 54.21(a)(1) requires that cables and connections are subject to an aging management review. Since fire detector insulated cables are not physically different than other insulated cables and performance or condition programs have not been proven to adequately predict cable insulation degradation due to aging, the staff does not agree that these cables can be excluded from an AMR.

Response

Duke agrees that cable condition programs have not been proven to adequately predict cable insulation degradation due to aging. The most prominent aging effect for cable is reduced insulation resistance caused by heat or radiation, which can eventually result in cracks in the cable insulation. The major concern is that failures of deteriorated cables might be induced during accident conditions due to moisture intrusion into cracks in the

cable jacket and cable insulation. The failures caused by such deterioration and moisture intrusion of moisture would be shorting between phases or shorting to ground.

Cables installed to provide the function of connecting the fire detector to the rest of the fire detection system are performance tested monthly and the wiring is supervised to provide a trouble alarm for any open circuit, short circuit, or ground condition. Any degradation of the fire detector cables that affects the cable function will always cause an alarm – even if no fire is present. Upon receipt of an alarm, the cause of the alarm would be investigated and any degraded or failed cable would be replaced. Fire detector cables are fail-safe and, therefore, adequately monitoring insulation degradation or predicting failures of fire detector cables is not an issue.

10 CFR 54.21(a)(1) does not require that cables and connections are subject to an aging management review. 10 CFR 54.21(a)(1) indicates that cables and connections meet the criteria of §54.21(a)(1)<u>(i)</u> (i.e., are passive) but indicates no determination as to their meeting the criteria of §54.21(a)(1)<u>(ii)</u>. It is Duke's position that fire detector cables meet the exclusion criteria provided in the SOC guidance regarding the application of §54.21(a)(1)(ii) for performance tested components.

Should further clarification of Duke's position be required regarding screening fire detector cables via the criteria of §54.21(a)(1)(ii), the following information will be important:

(1) Can the replacement based on performance testing guidance and criteria for exclusion in the SOC (page 22478) be used in a license renewal application? If not, the criteria for use of SOC guidance and associated criteria needs to be clarified for future reference.
(2) If so, are the SOC guidance and criteria applied correctly to fire detector cables? If the SOC guidance and criteria is not applied correctly, details of their misuse in this example need to be clarified for future reference and application.

<u>RAIs 2.6-6 & 2.6-7</u>

Sections 2.6.6.5 and 2.6.6.6 of the application conclude that resistance temperature detectors (RTDs) and thermocouples do not perform their function without moving parts or without a change in configuration or properties and thus are not subject to an aging management review. However, the industry guidance for license renewal in NEI 95-10, Revision O, Appendix A, recommends that RTDs and thermocouples should be subject to an aging management review. During a clarification conference call with Duke on January 7, 1999, the staff stated that the

pressure boundary intended function for RTDs and thermocouples is subject to an aging management review. Therefore, Duke should supplement the February 17, 1999, response (RAIs 2.6-6 and 2.6-7) by including RTDs and thermocouples (pressure boundary) as being subject to an aging management review and reference the sections of the LRA that address the mechanical review (pressure boundary) for these components. This is a confirmatory item.

Response
The pressure boundary intended function and applicable aging effects associated with RTD and thermocouples pressure boundaries are addressed in either the Reactor Coolant System components aging management review or the mechanical system components aging management review. These aging management reviews include all applicable aging effects. See Sections 2.4 and 3.4 of Exhibit A of the Application for the Reactor Coolant System components review and Sections 2.5.12 and 3.5.12 of Exhibit A of the Application for the mechanical system components review. For Oconee License Renewal, the commodity group "PIPE" includes thermowells.

Potential Open Item 2.7-11 (New)
Section 2.7.1 of report OLRP-1001 indicates that the conceptual boundary of auxiliary buildings includes the hot machine shop and spent fuel pools for Units 1, 2, and 3. Section 2.7.3 of report OLRP-1001 describes the auxiliary buildings and the hot machine shop but not the spent fuel pool. Explain why the spent fuel pool is not described and where in report OLRP-1001 you address the spent fuel pool?

Response
The spent fuel pools are included within the boundary of the Auxiliary Buildings as noted in Section 2.7.1 and Table 2.7-1 of Exhibit A of the Application. Specific aspects of structures, like the spent fuel pool within the Auxiliary Building, are not described in the application. The spent fuel pools are described in Oconee UFSAR Section 9.1.2. The aging management review of the spent fuel pools are addressed with Auxiliary Building concrete and steel components in a fluid environment (including stainless) in Section 3.7.3 and Table 3.7-1 of Exhibit A of the Application.

Potential Open Item 2.7-12 (New)
In an April 8, 1999 letter, the staff sent the following question:

The earthen embankments, Keowee structures, and yard structures are not described clearly in report OLRP-1001 and very little information on these structures can be found

in the UFSAR. Provide information on (1) configuration, (2) location, and (3) structural classification on each of these structural components. A drawing of Oconee and Keowee sites is helpful to locate these structures.

By letter dated April 16, 1999, Duke was requested to amend its response to this question as noted below.

The staff has reviewed Duke's response to RAI 2.7-10, which provided a drawing of the Oconee site with the boundaries identified for the structures subject to an aging management review. The staff has reviewed this drawing and has no more questions regarding earthen embankments at this time. Regarding Keowee structures, the staff will review Duke's response to RAI 2.5.13-2 and will inform Duke if additional information is needed in this area. At this time Duke does not need to supply any additional information regarding Keowee for potential open item 2.7-12.

Regarding the yard structures the staff has the following specific questions:

(a) Are the three lattice towers for the 230 kV transmission line within the yard structures Class 2 or QA-class structures?
(b) Are all the cable trenches or pipe trenches addressed in Section 2.7.10 for the yard structures covered with reinforced concrete blocks that are designed for missile protection?

<u>Response</u>
(a) The three lattice towers for the 230kV transmission line from Keowee to Oconee are addressed with Yard Structures in Section 2.7.10 of Exhibit A of the Application. The three lattice towers for the 230kV transmission line are Class 2 structures.
(b) Cable trenches and pipe trenches are addressed in Section 2.7.10 with the Yard Structures. All of the trenches are not covered with reinforced concrete blocks designed to provide missile protection. Those that are required to provide missile protection, such as the Standby Shutdown Facility trench, are designed to provide missile protection. Other trenches, such as the Intake trench, are covered with checkered plates that are not designed to provide missile protection.

<u>Potential Open Item 2.7-13 (New)</u>
Section 2.7.3 of report OLRP-1001 states that the hot machine shop extension is a QA 4 structure. Give the definition of QA 4 structure and why this QA 4 structure is within the scope of license renewal.

<u>Response</u>

The hot machine shop is identified as a QA 4 structure. The definition of QA 4 is provided in the Duke Quality Assurance Topical Report. The Duke Quality Assurance Topical Report is discussed in Section 4.13 of Exhibit A of the Application with the *Duke Quality Assurance Program.* In keeping with the classification of the other structures in the Application, the hot machine shop should have also been identified as a Class 2 structure.

Potential Open Item 2.7-14 (New)

Section 2.7.9.1 states that the Units 1 & 2 transformer and switchgear enclosures and Unit 3 switchgear enclosure are Class 1 structures. Clarify if these enclosures are for transformer CT4 and 4kV switchgear.

Response

The Units 1 & 2 transformer and switchgear enclosure and the Unit 3 switchgear enclosure are identified as Class 1 structures in Section 2.7.9.1 of Exhibit A of the Application. The Units 1 & 2 enclosure contains transformer CT4 and 4kV switchgear. The Unit 3 enclosure contains 4 kV switchgear.

RAI 3.2-3

As the high flux neutron fluence is known to have adverse effects on structures, Table 3.2-2 should contain the 40-year and 60-year estimates of high flux neutron fluence to ensure that they are within the threshold limits stated in the response to this RAI for all structures in the Table.

Response

The bounding high flux neutron fluence for 40-year and 60-year estimates were provided in response to RAI 3.2-2. The following table identifies the high flux neutron fluence that is estimated for 60 years for each structure. As stated in Duke's original response to RAI 3.2-3, the fluence is below the limit where degradation could occur.

Structure	60-year Estimate of High Flux Neutron Fluence
Auxiliary Buildings	< 9.8×10^{17} neutrons/cm^2
Earthen Embankments	Negligible
Intake Structure	Negligible
Keowee Structures	Negligible
Reactor Buildings	9.8×10^{17} neutrons/cm^2
Standby Shutdown Facility	Negligible
Turbine Buildings	< 9.8×10^{17} neutrons/cm^2
Yard Structures	Negligible

RAI 3.3-19

In accordance with 10 CFR 54.21(a)(1)(ii), components not subjected to a qualified life or a specified time period are subjected to aging management review. Based on operating experience, if the applicant has a specified replacement schedule for moisture barriers, etc., these components would not be subjected to AMR under the license renewal rule. However, an ad-hoc replacement based on the component's condition would not justify this exclusion. The licensee is requested to justify the replacement interval for these passive components in order to meet the exclusion criteria specified in the license renewal rule.

Response

The Commission has provided the staff the flexibility to approve on a case by case basis the replacement of components and materials based on condition monitoring or performance testing. This is in addition to replacement based on qualified life or specified time interval. The replacement of caulking and sealants within the Oconee

Reactor Building will be in accordance with ASME Section XI, Subsection IWE, Examination Category E-D, which has been endorsed by the NRC in 10 CFR Part 50, §50.55a.

Potential Open Item 3.3.3-1 (New) - Applicable Aging Effects for Steel Components

Under the Coating Program in Section 4.7 of the LRA, there is no mention of training and qualification of applicators. Please provide this information.

Response

The Oconee *Coatings Program* is discussed in Section 4.7 of Exhibit A of the Application. An effective coatings program includes the proper selection of coating, the proper preparation of the surface, the proper application of the coating and quality control measures during each phase of the coating process. Duke specifications, which are based on high quality industry standards, define the requirements for surface preparation, application of the coating, and quality control during the coating process. The training and qualification of applicators are included in these Duke specifications. The oversight of the training and qualification of the applicators is governed by the Duke *Quality Assurance Topical Report*. The Duke *Quality Assurance Topical Report* is discussed with the Duke *Quality Assurance Program* in Section 4.13 of Exhibit A of the Application.

Potential Open Item 3.4.3-2 (New) - Reactor Coolant System

Cast austenitic stainless steel (CASS) materials were reviewed in the Staff's SER on Topical Report BAW-2243. It does not contain our newest position on thermal embrittlement of CASS. Our position on thermal embrittlement of CASS has been modified as a result of our review of Topical Report EPRI TR-106092. To be consistent with our position on thermal embrittlement of CASS, the aging management program for CASS piping and valve bodies must meet the criteria discussed for RCP pump casings.

Response

The Class 1 pressure retaining items fabricated from cast asutenitic stainless steel at Oconee are limited to bodies and/or bonnets of the following valves: core flood check valves, decay heat injection check valves, decay heat dropline isolation valves, PORV, pressurizer safety valves, pressurizer spray line valves, auxiliary spray line isolation and check valves, and selected valves within instrumentation, vent, and drain lines. Oconee has no Class 1 piping fabricated from cast austenitic stainless steel.

In reviewing the NRC's SER of BAW-2243A, the B&WOG did not use the ferrite screening criteria prescribed in EPRI-106092 to eliminate reduction of fracture toughness as an applicable aging effect. Reduction of fracture toughness was identified as an applicable aging effect for all RCS valve bodies within the scope of BAW-2243A. The B&WOG proposed a new program for managing reduction of fracture toughness of Class 1 CASS valve items that makes use of the favorable comparison between the fracture toughness of aged austenitic stainless steel and SAW weldments. This program was accepted by the NRC in their SER of BAW-2243A. The new program allows use of existing ASME Section XI inspections for valve bodies (i.e., B-M-1 and B-M-2) as supplemented by the flaw evaluation procedure prescribed in IWB-3640 if a flaw is detected. Duke credited the flaw evaluation procedure described in BAW-2243A to manage reduction of fracture toughness for valve bodies as described in Section 4.18.2 of Exhibit A of the Oconee License Renewal Application.

In accordance with ASME Section XI, Examination Category B-M-1, welded joints in selected Class 1 valve bodies that equal or exceed 4-inch NPS are volumetrically inspected and welded joints in Class 1 valve bodies less than 4-inch NPS receive a surface examination. In addition, Examination Category B-M-2 requires that selected Class 1 valves exceeding 4-inch NPS receive a visual inspection of the internal surfaces of the valve bodies. The favorable comparison of fracture toughness between aged cast austenitic stainless steel and SAW weldments precludes the need for supplemental volumetric or visual examinations of CASS valves bodies that do not meet the NRC's proposed ferrite screening criteria. Existing inspections of CASS valve bodies in accordance with ASME Section XI, as supplemented by the proposed flaw evaluation procedure described in IWB-3640, are sufficient to manage reduction of fracture toughness of CASS valve bodies for the period of extended operation.

Potential Open Item 3.4.5-9 (New) - Reactor Vessel

The staff intends to issue the final safety evaluation shortly on the Babcock and Wilcox Topical Report BAW-2251 "Demonstration of the Management of Aging Effects for the Reactor Vessel." The final safety evaluation will contain plant-specific items that an applicant who references the report needs to address in its license renewal application. After Duke receives the final safety evaluation, it must demonstrate to the staff that these plant-specific items are already contained in its license renewal application, or it must provide the plant-specific information to the staff.

Attachment 1

Oconee Nuclear Station
Application for Renewed Operating Licenses
Responses to Additional NRC Identified Potential Open Items
(provided by letter dated April 8, 1999)

May 10, 1999

Response

The following table provides the Oconee-specific responses to the license renewal applicant action items contained in the staff's Safety Evaluation Report that was transmitted by NRC letter dated April 26, 1999, Project No. 683. The left-hand column contains the license renewal applicant action item contained in the safety evaluation report. The right-hand column contains the Oconee-specific response.

Renewal Applicant Action Items from BAW-2251 Safety Evaluation Dated April 26, 1999	Oconee – Specific Response
The following are license renewal applicant action items to be addressed in the plant-specific license renewal application when incorporating the B&WOG topical report in a renewal application:	
(1) The license renewal applicant is to verify that its plant is bounded by the topical report. Further, the renewal applicant is to commit to programs described as necessary in the topical report to manage the effects of aging during the period of extended operation on the functionality of the reactor vessel components. Applicants for license renewal will be responsible for describing any such commitments and identifying how such commitments will be controlled. Any deviations from the aging management programs within this topical report described as necessary to manage the effects of aging during the period of extended operation and to maintain the functionality of the reactor vessel components or other information presented in the report, such as materials of construction, will have to be identified by the renewal applicant and evaluated on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).	Exhibit A of the Application, Section 2.4.5, contains a description of the Oconee reactor vessels and the process used by Duke to ensure that BAW-2251 bounds the Oconee reactor vessels. Exhibit A of the Application, Section 3.4.5, contains a discussion of applicable aging effects and a listing of the programs credited for managing the applicable aging effects. Detailed descriptions of the aging management programs for the reactor vessel are provided in Chapter 4.0 of the Application. Exhibit A of the Application, Section 5.4.2, contains a summary of the TLAA applicable to the reactor vessel.
(2) A summary description of the programs and evaluation of TLAAs is to be provided in the license renewal FSAR supplement in accordance with 10 CFR 54.21(d).	Exhibit B of the Application contains a summary description of the programs and evaluation of the TLAA associated with the Oconee reactor vessels.

Renewal Applicant Action Items from BAW-2251 Safety Evaluation Dated April 26, 1999	**Oconee – Specific Response**
(3) Since the staff has not made any finding on whether the B&WOG topical report provides the complete list of reactor vessel components subject to an aging management review or whether the scoping methodology is adequate, individual plant applicants will need to provide a comprehensive list of structures and components subject to an aging management review and the methodology for developing this list as part of their license renewal applications. Any components determined by the applicant to be subject to an aging management review for license renewal but not within the scope of the topical report are required to be addressed in the license renewal application.	Section 2.4.5 of Exhibit A of the Application contains a description of the reactor vessel and a list of the items subject to aging management review. Section 2.4.5 discusses the extent to which BAW-2251 bounds the Oconee reactor vessels.
(4) The B&WOG has determined that the lower CRDM service support structure, including the weld that connects the lower CRDM service support skirt to the reactor vessel closure head, and the reactor vessel support skirt, including the weld that connects the reactor vessel support skirt to the transition forging, are subject to an aging management review for license renewal. However, the B&WOG has decided to exclude them from the scope of the topical report. Thus, a renewal applicant needs to address them in its license renewal application.	The reactor vessel support skirt (including the attachment weld to the transition forging) and control rod drive service structure (including the attachment weld to the reactor vessel head) are described in Sections 2.4.11.3 and 2.4.11.4, respectively, of Exhibit A of the Application. The applicable aging effects for the reactor vessel support skirt and control rod drive service structure are discussed in Section 3.4.11.2 of Exhibit A of the Application. Aging management programs that manage the applicable aging effects are listed in Section 3.4.11.5 and are described in Sections 4.5, 4.18 and 4.19 of Exhibit A of the Application.
(5) The license renewal application for Oconee needs to address the fatigue evaluation of the reactor vessel studs on a plant-specific basis.	Section 5.4.1.1 of Exhibit A of the Application addresses the fatigue of the reactor vessel studs.

Renewal Applicant Action Items from BAW-2251 Safety Evaluation Dated April 26, 1999	Oconee – Specific Response
(6) A license renewal applicant needs to discuss the plant-specific methodology and instrumentation used to assess the number of operational transients in its renewal application for staff review. The staff review will also include the number of operating cycles applicable to the reactor vessel studs.	Section 5.4.1.3 of Exhibit A of the Application addresses the Oconee Thermal Fatigue Management Program.
(7) The B&WOG identifies flaw growth acceptance in accordance with the ASME Section XI ISI program as a TLAA, but indicates that flaw growth acceptance evaluation is plant-specific, is not within the scope of the report, and will be resolved on a plant-specific basis. Thus, a license renewal applicant needs to address it in the renewal application.	Section 5.4.1.2 of Exhibit A of the Application addresses the flaw growth acceptance under ASME Section XI.
(8) Alloy 600 components in the reactor vessel such as CRDM housings and other penetrations may be subject to crack initiation and growth. The B&WOG originally proposed to use the ASME Section XI program, supplemented by leak detection and surveillance of boric acid, to manage cracking of Alloy 600 components. In an April 1, 1997, response to the staff's request for additional information concerning Generic Letter 97-01, "Stress Corrosion Cracking of Control Rod Drive Mechanisms and Other Vessel Head Penetrations," the B&WOG stated: "Each participating plant will address additional requirements for RV head penetrations, including closure head penetrations less than 2 inch N.S. (i.e., thermocouple nozzles at TMI-1 and ONS-2)." Thus, a license renewal applicant referencing the topical report will need to submit its plant-specific program to manage cracking of Alloy 600 components in the reactor vessel in its renewal application for staff review.	Section 4.3.1 of Exhibit A of the Application addresses the Alloy 600 Aging Management Program. Also see the responses to RAI 4.3.1-1 through 4.3.1-6.

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Renewal Applicant Action Items from BAW-2251 Safety Evaluation Dated April 26, 1999	Oconee – Specific Response
(9) During the review of the topical report, the staff had a question regarding the need to update the reactor vessel fracture toughness estimates with new data as it become available. In its August 11, 1997, RAI response, the B&WOG states: "Each license renewal applicant will define a process to ensure that the time-dependent parameters used in the TLAA evaluations reported in BAW-2251 are tracked such that the TLAA remains valid through the period of extended operation. The process will be defined on a plant-specific basis at the time of the licensee renewal application." Thus, a license renewal applicant needs to describe such a process in its application for staff review. If new information affects the conclusions of the topical report for the applicant's plant, the applicant needs to update its TLAA evaluations as appropriate and provide the updated evaluations in its renewal application for staff review.	Exhibit B of the Application contains a summary description of the programs and evaluation of the TLAA associated with the Oconee reactor vessels. The execution of the *Reactor Vessel Integrity Program* and the *Oconee Thermal Fatigue Management Program* will assure that the TLAA contained in the Oconee UFSAR remain valid for the period of extended operation.
(10) In its August 11, 1997, RAI response, the B&WOG indicated that Oconee Unit 2 and TMI Unit 1 will provide updated predictions of RT_{PTS} for welds WF-25 and SA-1526, respectively, when the plant-specific application for license renewal is submitted. For plants with an RT_{PTS} value for 48 EFPY exceeding the corresponding PTS screening criterion, a license renewal applicant must address the requirements in 10 CFR 50.61(b)(3) by developing, and requesting staff approval for reasonably practicable flux reduction programs to avoid exceeding the PTS criterion.	Section 5.4.2.1 of Exhibit A of the Application and the response provided to RAI 5.4.2-1 (Duke letter dated February 17, 1999, Attachment 1, page 77) provide updated predictions of RT_{PTS} for Oconee.

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Renewal Applicant Action Items from BAW-2251 Safety Evaluation Dated April 26, 1999	**Oconee – Specific Response**
(11) If an applicant has installed flow stabilizers using Alloy 600 and/or Alloy 82/182 weld material, the applicant must include the flow stabilizers in its Alloy 600 aging management program. Alloy 600 and Alloy 82/182 weld materials are susceptible to cracking in primary water environments.	The Oconee flow stabilizers are made of stainless steel and are attached to the stainless steel cladding with stainless steel weld material.
(12) Embrittlement of the reactor vessel will be managed to ensure intended functions of the reactor vessel for 60 years. For the staff to determine if the plant could be operated for 60 years, an applicant must show that an operating window will be available between the pressure-temperature limits and the net positive suction curves for the RC pumps for 60 years. Otherwise, the applicant will propose aging management activities to minimize the extent of embrittlement, or other alternatives, to permit safe plant operation for 60 years. Should the applicant show that the reactor could only be operated for a time period less than 60 years, the duration of the renewed license, if granted, would be limited to that time period.	Refer to Duke response to RAI 3.4.5-8 (Duke letter dated February 17, 1999, Attachment 1, page 32) for further details on this topic.

Renewal Applicant Action Items from BAW-2251 Safety Evaluation Dated April 26, 1999	Oconee – Specific Response
(13) The neutron fluence must be experimentally monitored by ex-vessel or in-vessel dosimetry, and if modifications to the design and operation of the plant changes either the neutron energy spectrum, gamma heating or the reactor inlet temperature, as discussed in section 3.3.4.1 of this safety evaluation, the licensee must notify the NRC and propose a program to determine the impact of the modifications.	1. Descriptions of the Oconee Cavity Dosimetry Program and the Oconee Fluence and Uncertainty Calculations are provided in Sections 4.24.2, *Cavity Dosimetry Program* and 4.24.3, *Fluence and Uncertainty Calculation,* respectively, of Exhibit A of the Application. The *Cavity Dosimetry Program* includes ex-vessel cavity dosimetry for Unit 2. In addition, the MIRVSP includes in-vessel dosimetry. 2. A description of *Master Integrated Reactor Vessel Surveillance Program* is provided in Section 4.24.1 of Exhibit A of the Application. If modifications to design and operation result in changes to neutron energy spectrum, gamma heating, or the reactor inlet temperature relative to that discussed in BAW-1543, Revision 4, then NRC will be notified and a program to determine impact will be proposed. This commitment will be added to Section 4.24.1 of Exhibit A of the Application.

Potential Open Item 3.4.6-3 (New) - Reactor Vessel Internals

Below are plant-specific items from the staff Draft SER on Topical Report BAW-2248 (the Draft SER for Topical Report BAW-2248 is under preparation).

(A) According to B&WOG, one of its objectives in BAW-2248 states, "It is intended that NRC review and approval of this report will allow that no further review of the matters described herein will be needed when the report is incorporated by reference in a plant specific renewal license application." The license renewal applicant must address matters not described in the report, such as the issues addressed in Section 3.3.1 of the SE pertaining to the following letters: (1) Letter from Raj F. Anand, (NRC), to David J. Firth, December 2, 1998, Request for Additional Information Regarding the Babcock & Wilcox Owners Group Generic License Renewal Program Topical Report, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals, BAW-2248, July 1997," and (2) Letter from William R. Gray to David

B. Mathews, (NRC), dated February 18, 1999, B&WOG Generic License Renewal Program Topical Report BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (RAIs 1 through 14 from December 4, 1998).

(B) License renewal applicants must identify whether the intended function of the RVI is to provide shielding for the RPV. If not an intended function, the license renewal applicant should provide justification for that conclusion. Should a license renewal applicant determine that the RVI's intended function is to provide shielding for the RPV, then the items that support this intended function, such as, the thermal shield and the thermal shield upper restraint assemblies, must be identified and reviewed in accordance with 10 CFR 54.21(a)(3).

(C) Plant-specific analysis is required to demonstrate that under loss-of-coolant-accident (LOCA) and seismic loading the RVI's have adequate ductility to absorb strains at the regions of maximum stress intensity and that irradiation accumulated at the expiration of the renewal license will not adversely affect deformation limits. The Reactor Vessel Internals Aging Management Program (RVIAMP) must develop data to demonstrate the RVI's will meet the deformation limits at the expiration of the renewal license.

Response
Duke has decided to defer the response to Potential Open Item 3.4.6-3 (New), Items (A) and (C) because the final Safety Evaluation Report (SER) for BAW-2248 was not available for review. With regard to Item (B) above, Duke included the intended function "Provide Gamma and Neutron Shielding," in Exhibit A of the Oconee License Renewal Application (Table 2.4-4). Consequently, an aging management review of the thermal shield and thermal shield upper restraint were included in the Oconee Application (Section 3.4.6) since these items were omitted from the B&WOG report.

Potential Open Item 3.4.6-4 (New)
Below are Open Items from staff Draft SER on Topical Report BAW-2248. These open items need to be addressed by the B&WOG. If these issues are not resolved prior to the issuance of the Oconee SER they will be carried as open items in the Oconee SER.

(A) The B&WOG must modify their aging management program to include the program proposed by the staff in Section 3.3 of the Draft SER for BAW-2248 for managing the effects of SCC, IASCC, thermal embrittlement and neutron embrittlement.

(B) To determine whether CASS components are above or below the threshold value of 1 x 10^{17}n/cm^2 discussed in Section 3.3.3 of the Draft SER for BAW-2248, the B&WOG must

provide estimates of the neutron fluence of each CASS component at the expiration of the license renewal term, identify the method of determining the neutron fluence and provide justification for why the method is applicable for components above or below the core.

Response
Duke has decided to defer providing a response to this potential open item until after the staff issues the final Safety Evaluation Report (SER) for BAW-2248, because the final SER was not available for review.

Potential Open Item 3.4.6-5 (New) (The item below refers to RAIs 12 & 13 on BAW-2248)
The staff's review of BAW-2248, "Demonstration of the Management of Aging Effects for the Reactor Internals" resulted in RAIs 12 and 13 which were transmitted to the B&W Owners Group (B&WOG) in a letter from Anand to Firth dated December 2, 1998. In RAI 12, the staff requested the B&WOG to describe the baffle bolt inspections that will be conducted prior to the start of the extended license renewal period and indicate how these actions provide the basis for assuring that the monitoring and inspection techniques that are planned for implementation during the period of extended operation are appropriate. In RAI 13, the staff requested that the B&WOG describe the program that will be implemented as outlined in Section 4.6 of BAW-2248 with regard to the aging management of the RVI baffle bolts and include a description of the overall inspection program, the inspection techniques, intervals and monitoring.

The B&WOG responded to these RAIs, by stating that an Issues Task Group (ITG) on reactor vessel internals is currently addressing the issues related to baffle bolts, and the data and information acquired from these various program activities will be used to determine the necessary steps in managing the issues of baffle bolt age-related degradation, including future inspection plans. The B&WOG will provide support to the ITG by providing plant-and design-specific data. The licensees will be responsible for using the tools provided by both the ITG and the Owners Group to determine the necessary steps to manage the applicable aging effects. These plans are expected to be outlined on a plant specific basis, therefore, the information requested in RAI 12 and 13 remain as an open item pending the licensee's submittal of plant specific information and plans.

Response

Duke is working closely with the B&WOG and the ITG to determine when inspection of the baffle bolts at a B&W operating plant may be appropriate. Duke has decided to defer providing a response to this potential open item until after the staff issues the final Safety Evaluation Report (SER) for BAW-2248, because the final SER was not available for review.

RAI 3.4.7-1

It is the staff's understanding that the tubes in the Oconee Unit 3 steam generators (SG) were subjected to stresses slightly beyond the allowable values during an event in August 1994 involving the injection of cold feedwater into a hot, dry SG. In RAI 3.4.7-1, the staff requested the applicant to discuss whether or not this event contributed to the aging of the SG tubes. In addition, the applicant was requested to describe the procedures that are used to evaluate the impact of such events on the adequacy of the aging management programs (AMPs).

In its response, the applicant stated that the Unit 3 event that occurred on August 10,1994, was a reactor trip from full power that resulted in the dry-out of the B SG. The overcooling that occurred as a result of the inadvertent opening of a turbine bypass valve resulted in tube-to-shell differentials in excess of established tube-to-shell limits. According to the applicant, reactor trip is an upset event (i.e., level B) and is not an emergency or faulted event (level C or D). A subsequent evaluation by B&W using actual transient data indicated that the axial tube loads (both compressive and tensile) were within the limits of the allowable tube loads. Since the allowable tube loads were not exceeded, the event of August 10,1994, did not impact the integrity of the SG tubes.

It is the staff's understanding that there was uncertainty associated with the actual transient data, and therefore potential for damage to tubes cannot be ruled out. The B&WOG recognize the significance of this event and agreed to incorporate changes in procedures to manage such events more effectively in the future. The applicant's response did not address this aspect of the staff's concern. Therefore, this issue remains an open item.

Response

The uncertainty in the transient data regarding the August 10, 1994 event at Oconee Unit 3 refers to a shell thermocouple attached to the affected steam generator that was providing false readings prior to and during the event. Specifically, FTI evaluated steady state and transient shell thermocouple data and determined that thermocouple A0977 was either not calibrated properly or was not in full contact with the shell since the reading was significantly lower than the two shell thermocouples directly above it during full

power operation and throughout the transient—the readings should be approximately the same. Since the shell temperature used for the tube-to-shell differential is an average of five shell thermocouples, an erroneous low reading would result in an abnormally high tube-to-shell differential since the average shell temperature would be too low.

The August 10, 1994, Oconee Unit 3 transient data was evaluated and the time that the maximum tube-to-shell differential occurred was identified (i.e., 12:05 p.m.--max tube-to-shell differential of +82 °F). For the data at 12:05 p.m., the erroneous thermocouple reading A0977 (378 °F) was replaced with the lowest value of the two thermocouples directly above the affected thermocouple (i.e., A0976--454 °F), and the average shell temperature and tube-to-shell differential were recalculated. The maximum tube-to-shell differential, which occurred at 12:05 p.m., was reduced from 82 °F to 66 °F. The compressive limit of +60 °F was exceeded by 6 °F, and a structural evaluation was performed prior to the restart of the unit. The maximum tube-to-shell differential of 66 °F was used in the structural evaluation and the resulting maximum axial tube load was lower than the maximum compressive load for a design basis heatup (i.e., 100 °F/hr). Based on this result it was concluded that the tube compressive load associated with August 10, 1994 event did not jeopardize the structural integrity of the tubes.

Oconee EOP or Operator Action Revisions Following the 8/10/94 Event

Subsequent to the August 10, 1994 event the Oconee Emergency Operating Procedures (EOPs) were reviewed and the following changes were made. A caution was added to Section 502 (Loss of Heat Transfer) and Section 503 (Excessive Heat Transfer) that will alert Operators to the possibility of exceeding the compressive tube-to-shell limit of 60 °F should a significant delay occur in reestablishing feed to a dry and intact OTSG. In addition, a training package was issued to inform all licensed personnel of the change to the EOPs.

Potential Open Item 3.4.8-6 (New) - Reactor Coolant Pumps

RCP Casings which are fabricated from Cast Austenitic Stainless Steel (CASS) should have the following aging management program:

The CASS components should be evaluated to the criteria in EPRI TR-106092 with the following additional criteria:

(a) Statically cast components with a molybdenum content meeting the requirements of SA-351 Grades CF3 and CF8 and with a delta ferrite content less than 10 percent will not need supplemental examination.

(b) Ferrite levels will be calculated using Hull's equivalent factors or a method producing an equivalent level of accuracy (±6 percent deviation between measured and calculated values).

(c) Cast stainless components with Niobium are subject to supplemental examination.

(d) Flaws in CASS with ferrite levels less than 25 percent and no niobium may be evaluated using ASME Code IWB-3640 procedures.

(e) Flaws in CASS with ferrite levels exceeding 25 percent or niobium will be evaluated using ASME Code IWB-3640 procedures. If this occurs, fracture toughness data will be provided on a case-by-case basis.

Components that have delta ferrite levels below the screening criteria have adequate fracture toughness and do not require supplemental inspection. Components that have delta ferrite levels exceeding the screening criteria may not have adequate fracture toughness, as a result of thermal embrittlement, and do require supplemental volumetric inspection with techniques qualified to Appendix VIII of Section XI of the ASME Code, provided inspection techniques can be developed. The licensee must acknowledge that they will implement the above criteria.

Response

Duke is not clear how the additional criteria provided in Potential Open Item 3.4.8-6 are to be applied at Oconee. In order to assess how these criteria may relate to the process Duke has used, please identify the reference for these additional criteria.

The process Duke used to consider the applicability of loss of fracture toughness for CASS components does not involve screening criteria. The screening portion of EPRI TR-106092 is not being applied at Oconee; rather, all Class 1 CASS components are considered subject to loss of fracture toughness and managed via ASME Section XI examination and the modified analysis methods described in EPRI TR-106092. The following additional information is provided to relate the process Duke used to the criteria provided in this potential open item.

The Westinghouse RCP casings at Oconee Unit 1 are fabricated from statically cast ASTM A-351 grade CF-8 with no niobium. Metallurgical laboratory reports for the pump casings were located and the amount of ferrite in each casting was calculated using

Hull's equivalent factors. All Oconee Unit 1 pump casings are below the 10% ferrite screening criterion with the exception of the lower casing of pump 1RC-P1B2, which has a calculated ferrite content of 11.2%. When performing inservice inspection of the reactor coolant pumps in accordance with ASME Section XI, Examination Categories B-L-1 and B-L-2, and flaws are detected in the lower casing of 1RC-P1B2, the flaw evaluation procedure specified in IWB-3640 may be used as discussed in Section 4.18.2.1 of Exhibit A of the Application.

The Bingham RCP casings at Oconee Units 2 and 3 are fabricated from statically cast ASTM A-351 grade CF-8M with no niobium. Metallurgical laboratory reports for the pump casings were located and the amount of ferrite in each casting was calculated using Hull's equivalent factors. The maximum calculated ferrite content is 20.1% for the lower casing of one pump and all remaining casing halves are below 20%. Reduction of fracture toughness by thermal embrittlement is an applicable aging effect for the Oconee Unit 2 and 3 pump casings. When performing inservice inspections in accordance with ASME Section XI, Examination Categories B-L-1 and B-L-2, and flaws are detected in the pump casings, the flaw evaluation procedure specified in IWB-3640 may be used as discussed in Section 4.18.2.1 of Exhibit A of the Oconee License Renewal Application.

RAI 3.4.10-1 - Letdown Coolers

Oconee operating experience was reviewed to validate the identified applicable aging effects. The review of Oconee operating experience identified that the letdown cooler heat exchanger tubes did experience cracking in the past as a result of improper operation of the coolers. Two of the six letdown coolers have been replaced; the other four have been repaired and operating procedures have been changed to eliminate improper operation.

In RAI 3.4.10-1, the staff requested the applicant to describe the repairs which were performed on the damaged letdown coolers and the specific analyses which were performed to assure that thermal and vibrational stresses during normal and off-normal operation will not cause fatigue failure during the period of extended operation.

In its response, the applicant stated that the letdown coolers are of the shell and spiral tube design. A review of operational history identified some events where the tubes cracked due to thermal and vibrational stresses caused by improper operation of the coolers.

The applicant is requested to provide its evaluation of the damage to the various components of the letdown coolers or the specific analyses performed to assure that all the components of the damaged letdown coolers have experienced no degradation as a result of improper operation.

Further, the applicant is requested to provide an analytical assessment to assure that these letdown coolers are operating in a condition that precludes potential failure due to thermal fatigue during the extended period of operation.

Response
Duke has decided to defer the response to RAI 3.4.10-1 until after the staff issues the SER for Oconee License Renewal because the responsible Duke engineers were not available to assist in the preparation of a response.

Potential Open Item 3.5.7-3 (New)
The LR application does not specify what components in the Chemical Addition System (CAS) will be included in the Treated Water Systems Stainless Steel Inspection program. Also, the applicant does not specify what corrective actions will be taken if this one-time inspection finds that corrosive action of caustic solution is sufficiently significant to cause future problems.

Response
Table 3.5-5 on page 3.5-126 of Exhibit A of the Application lists those components in the Chemical Addition System included within the scope of the Treated Water Systems Stainless Steel Inspection. These components are shown on license renewal flow diagrams OLRFD-110A-1.8 and 3.8 upstream (toward the caustic pump) of valves 1CA-62, 2CA-63, and 3CA-62.

Table 3.5-5 notes "caustic area" for the internal environment that would infer that these components are exposed to a caustic environment. Actually, these components are exposed to demineralized water used to test the pump and have never been exposed to a caustic environment. A caustic environment would only be present following certain design basis events that have never occurred.

Any indication of loss of material or cracking identified during the Treated Water Systems Stainless Steel Inspection will initiate the Problem Investigation Process (PIP) that could ultimately lead to a formalized program. The PIP is a process governed by a nuclear generation department directive that will initiate an engineering evaluation of the identified condition. The PIP evaluation is designed to consider the need to conduct a root cause analysis and determine corrective actions, which may include actions to prevent recurrence. For this example, the PIP evaluation will determine the need for a root cause analysis, including the establishment of methods to determine the amount of material lost. The piping evaluation will also determine corrective actions that assure a

degraded component condition has not or will not exceed the limit required for qualifying the component for further service.

RAI 3.5.8-3 (11/20/98B) - Air Conditioning, Heating, Cooling and Ventilation Systems

In order to minimize vibration and subsequent dynamic loads, isolators are commonly installed to the attached devices (such as fans) in nuclear power plants. As described in the RAI, the operating experience of other nuclear power plants indicates that cracking of ductwork due to vibration-induced fatigue and loosening of fasteners due to dynamic loading cannot be avoided by the installation of isolators. The response to this RAI did not address this concern beyond stating that isolators are expected to completely eliminate the operational vibration and dynamic loads. The applicant should either address these types of aging effects in the application or provide justification for not including them in the AMR program.

Response

Since the Penetration Room Ventilation System is constructed of pipe instead of ductwork, the response to this potential open item pertains only to the Auxiliary Building Ventilation System and Control Room Pressurization and Filtration System. Oconee has had good operating experience with respect to isolators in these two ventilation systems in preventing the transmission of vibration and dynamic loads to surrounding equipment to preclude ductwork cracking and loosening of fasteners. A review of the Oconee Problem Investigation Process (PIP) database and Oconee specific LERs did not identify any instances of cracking of ductwork or loosening of fasteners in the Control Room Pressurization and Filtration System and the Auxiliary Building Ventilation System. A review of industry events in the Nuclear Plant Reliability Data System (NPRDS) cited design deficiency as the cause for cracking of ductwork at other plants.

In addition, the Auxiliary Building Ventilation System and the Control Room Pressurization and Filtration System are normally inservice. These two systems have been inservice for over twenty-five years and cracking of ductwork and loosening of fasteners would have revealed itself as a concern by now. As a result, Duke concluded that cracking of ductwork and loosening of fasteners in the Auxiliary Building Ventilation System and Control Room Pressurization and Filtration System are not applicable aging effects for these systems.

Potential Open Item 3.5.10-1 (New)

The LR application does not include evaluation of the aging effects which may exist in the hydrogen recombiner. Although this piece of equipment is constructed from stainless steel

which, as demonstrated by the applicant, resists the environment to which the hydrogen recombiner is exposed, there are some ancillary components such as gaskets and cable insulation which may be prone to aging effects.

Response

The hydrogen recombiner is a complex assembly of components that was purchased as a unit and is mounted on an equipment skid. Other examples of complex assemblies are diesel generators and HVAC refrigeration units. The intended function of the hydrogen recombiner is to recombine (back to water vapor) the hydrogen and oxygen generated when a loss-of-coolant accident (LOCA) occurs. The hydrogen recombiner performs this intended function by heating the two gases to the point of thermal recombination, cooling the gases, and returning the effluent to containment. The hydrogen recombiner performs its intended function with moving parts (the blower) and with a change in configuration or properties (the heater element) and is therefore active and not subject to an aging management review.

Connections (piping and electrical) to the hydrogen recombiner on the equipment skid form the evaluation boundary. That is, the hydrogen recombiner includes all components mounted on the equipment skid. Connections to the skid equipment are not part of the hydrogen recombiner.

Plant insulated cables connecting to the hydrogen recombiner are evaluated along with other insulated cables in Sections 2.6.3 and 3.6.3 of Exhibit A of the Application.

The hydrogen recombiner serves as part of the containment boundary when it is placed in service. Because of the importance of this function, the passive mechanical components that form the containment boundary were evaluated in the mechanical aging management review. Consistent with positions stated in the April 20, 1999, letter from Christopher I. Grimes of NRC to Douglas J. Walters of NEI, "License Renewal Issue No. 98-0012, Consumables", gaskets on the hydrogen recombiner are not subject to aging management review.

RAI 3.5.14-4 - Standby Shutdown Facility

It is stated in Section 3.4.14.8 of the LRA, that no applicable aging effects have been identified for the components of the starting air system. The emergency diesel generator (EDG) starting air system at several other facilities has experienced degradation due to excessive vibration in the piping and starting air valves which in some cases rendered the air receivers incapable of delivering starting air to the diesel engines at the design pressures. In RAI 3.5.14-4, the staff

requested the applicant to discuss the upgrades, if any, and/or surveillance requirements for the starting air system at Oconee to assure operability of this system during the period of extended operation beyond 40 years.

In its response, the applicant stated that cracking due to vibrational (mechanical or hydrodynamic) loads was a potential aging effect that was determined not to be applicable to the starting air system components subject to an AMR. The applicant further stated that cracking due to vibration can be attributed to design deficiencies. Vibration characteristically leads to cracking in a short period of time, on the order of hours to days of operation. For example, a component with a 1 Hz vibratory load will be subjected to 10^7 cycles in four months of service. Therefore, the applicant contends that failure due to vibratory stresses above the endurance limit is likely to occur early in life. Because this time period is short when compared to the overall plant operational life, the applicant argues that any cracking will be identified and corrected long before the period of extended operation. Therefore, according to the applicant, cracking due to vibrational loads, both mechanical and hydrodynamic, is not an applicable aging effect for the starting air system components subject to an AMR.

The staff does not agree with this assessment because the starting air system is used very infrequently and there is no assurance that vibratory stress cycles necessary for causing fatigue failures will occur early in plant life. In view of the operational failures of this system at other facilities, adequate assurance of the operability of this system during the period of extended operation is needed. Therefore, this remains an open item.

Response

Duke assumes that the Staff is referring to RAI 3.5.14-3. In general, vibration leads to cracking in a relatively short period of time. It is true that the SSF diesel generator operates infrequently. However, Oconee operating experience has revealed that design deficiencies leading to vibration-induced failures have manifested themselves in the components of the diesel generator skid. Cracks due to vibration were observed in fuel oil piping. The piping design was modified to preclude the effects of the vibration on the fuel oil piping by the installation of flexible hoses. The components of the Starting Air system that are subject to aging management review have always been physically separated from the diesel generator, and thus its vibratory loads, by flexible hose. Thus, by design, these components are not subjected to the vibratory loads experienced by the other diesel components mounted on the diesel skid.

Additionally, the starting air operability is verified every 31 days when the diesel generator is tested. Surveillance requirements for the diesel generator and starting air are dictated by ITS Sections 3.10.1.5 and 3.10.1.6.

Oconee Nuclear Station
Application for Renewed Operating Licenses
Responses to Additional NRC Identified Potential Open Items
(provided by letter dated April 8, 1999)

May 10, 1999

RAI 3.5.14-5 - Standby Shutdown Facility
The applicant's response to this RAI is that the Preventative Maintenance Activities aging management program will be used to control corrosion of underground carbon steel components. The applicant does not discuss cathodic protection of underground or on-ground carbon steel components. NACE provides considerable guidance (RP-0169, RP-0193-93) on programs for protection of underground or on-ground carbon steel components. The applicant has not indicated that it is aware of this guidance.

Response
Cathodic protection is not discussed because it is not credited as managing the aging of any underground or on-ground components.

RAI 3.5.14-6 - Standby Shutdown Facility
The applicant's response states that portion of the auxiliary service water system exposed to raw water and subject to fouling is the limited portion of the system that includes the pump, the pump discharging piping up to the pump discharge isolation valve, and the minimum recirculation piping. The remainder of the piping in the system is drained and is not subject to fouling. The System Performance Testing Activities will be used to control fouling.

Industry experience has shown that after fouling has taken place, flushing may not be an efficient way to remove fouling. Also, the licensee has not stated how it responded to Generic Letter 89-13, which states that licensees should have a program in place to control fouling.

Response
The System Performance Testing that is credited for managing fouling in the system components is used to ensure that the aging effect is detected and corrected before it progresses to a loss of system function. It is not intended to be a flush of the system to remove fouling deposits. The program has proven to be an effective method of managing fouling in service water systems, including the SSF ASW System. A decreasing trend in flowrates during performance testing has resulted in piping replacement in the SSF ASW pump suction piping. Response to Generic Letter 89-13 included the reliance upon this system performance testing to manage fouling.

As an additional point of correction, the response to RAI 3.5.14-6 mistakenly provided details about the wrong system. Two auxiliary service water systems exist at Oconee: the SSF Auxiliary Service Water (SSF ASW) System and the Auxiliary Service Water (ASW) System, commonly referred to as the "station" Auxiliary Service Water System.

The RAI number referenced the SSF ASW Application section, but asked about the "auxiliary service water" system. The details in the response provided information associated with the station Auxiliary Service Water System and not for the SSF Auxiliary Service Water System. The answer to the RAI is generally the same for both systems whose wetted portions rely on the System Performance Testing Activities to manage fouling.

Potential Open Item 3.5.14.7.1-1 (New)

This section pertains to the Standby Shutdown Facility Auxiliary Service Water System, and references Section 3.5.2.4 which discusses aging effects for cast iron in a raw water environment. Section 3.5.14.7.1 notes that there are no applicable aging effects for cast iron submersible pump casings. Please resolve this discrepancy and discuss applicable aging management programs.

Response

As noted in Section 3.5.14.7.1 the SSF Submersible Pump is not a permanently installed piece of equipment. The SSF Submersible Pump is stored in the SSF and exposed internally and externally to the ambient air within the Standby Shutdown Facility. Every two years the submersible pump is placed in the intake canal (Lake Keowee) for testing and then returned to the SSF for storage. Since exposure to raw water is infrequent, the ambient air of the SSF is the environment for identifying the applicable aging effects for the internal and external surfaces of the submersible pump. No applicable aging effects were identified for the cast iron pump expose to the ambient air of the SSF that could result in a loss of the component intended function of the submersible pump. Since the SSF ambient air is the environment to which the SSF Submersible Pump is exposed, Table 3.5-12 should have not contain the entry of the cast iron pump casing exposed to raw water.

RAI 3.7.3-2 (11-30-98A)

The applicant stated that the ingress of ground water is not possible through an internal concrete slab which was found with cracks determined to be the result of slab shrinkage during initial concrete placement. The Oconee response to the RAI also stated that cracks of foundation slabs have not been validated by Oconee operating experience. Oconee further stated that the cover provided in the Oconee structures meets or exceeds the applicable requirements of ACI 318-63, which has over the years proved to be effective for preventing chemical corrosion of the concrete reinforcement that exists in an environment not subject to special chemical exposures. For the above stated reasons, Oconee determined that cracks of foundation slabs need not be identified as applicable aging effects. Shrinkage cracks or cracks in concrete that are smaller than a few mills, may not lead to moisture ingress to rebars in concrete. Under purely shrinkage cracks, degradation of rebars is not likely; however, that is not the case with larger cracks that can be

found in most nuclear power plant structures. For cracks in concrete surfaces above ground or in those surfaces that are embedded underground where ground water is contaminated with corrosive agents, it is necessary to establish an aging management program (AMP). The AMP can be as simple as a periodic observation of evidence of rebar degradation and taking appropriate corrective action when significant rebar degradation is detected. Oconee is requested to provide its most recent chemical analysis data of the ground water surrounding Oconee safety-related structures and to address the effects of ground water on rebar. In addition, the broader aspect of managing rebar degradation needs to be addressed by the licensee. Therefore, this issue remains open.

Response

Degradation due to aggressive chemical attack on concrete and loss of material due to corrosion of rebar were identified as potential aging effects in Section 3.7.2.1 of Exhibit A of the Application. Aggressive chemical attack could result in degradation of concrete and steel in locations where the components are exposed to aggressive groundwater. Aggressive groundwater contains chlorides and/or sulfates that exceed the threshold limits. The chemical analysis of the Oconee groundwater is provided in Section 3.2.2 of Exhibit A of the Application and maintained on site for review. The chemical analysis of the groundwater shows that the levels of chlorides and sulfates are well below the threshold levels where degradation could occur.

In addition, the Oconee concrete was designed to ensure low permeability and to minimize the likelihood of concrete cracking that would allow penetration. The concrete was designed to ACI Standard 318 and its relevant ACI standards and ASTM specifications. These standards provide requirements such as the physical property requirements of aggregate and air-entraining admixtures, chemical and physical requirements of air-entraining cements, and proportioning of concrete to water. Therefore, effects due to aggressive chemical attack are not applicable for Oconee concrete and rebar because of the protection offered by the concrete and the non-aggressive environment of the groundwater.

RAI 3.7.7-1 (11-18-98A)

In order to make a realistic evaluation of the tendon system performance in the secondary shield wall (SSW), the staff needs information regarding (1) the minimum required prestress in each group of tendons, (2) results of the tendon forces as found during the periodic lift-off testing, and (3) information regarding retensioning of tendons, if performed. This information is needed for each Unit at Oconee to assess the time-dependent behavior of the SSW tendon system and their impact on license renewal related AMP and TLAA.

Response

Information concerning the performance of the secondary shield wall (SSW) tendon system is provided in Section 4.28 of Exhibit A of the Application. Section 4.28 contains the aging management program, *Tendon – Secondary Shield Wall – Surveillance Program*. In addition, RAIs 3.7.7-1, 4.28-1 and 4.28-2 provide additional details on the SSW tendon program. The results of the program (minimum required prestress, lift-off testing results, retensioning, etc.) are maintained on site and are available for staff review.

The loss of prestress of the SSW tendon does not meet the definition of a TLAA as described in the response to RAI 2.7-8. Loss of prestress of the SSW tendon system is managed by the *Tendon – Secondary Shield Wall – Surveillance Program* that is discussed in Section 4.28.

RAI 3.7.7-2 (11-30-98A)

The staff does not agree with the applicant's hypothetical logic in demonstrating that the concrete cracking has not occurred and will not occur in the internal concrete structures of the reactor building. The threshold temperature limits established in ACI 349 and in NUREG-1557 only assure that the concrete properties (compressive strength, modulus of elasticity, etc.) do not significantly change if a concrete structure experiences temperatures within those temperatures limits. They do not guard the structures against cracking. If the applicant wants to exclude the concrete cracking and potential corrosion of the embedded rebars in slabs and walls of the internal structures from Table 3.7-5 (Applicable Aging Effects), it should demonstrate that there are no existing cracks in those components (based on focused inspections), and justify why there will not be any future cracking.

Response

The logic that is provided to disposition concrete cracking is well documented in industry and NRC documents. The Nuclear Utility Management and Resources Council (NUMARC), now part of the Nuclear Energy Institute (NEI), *Containment* and *Class I Industry Reports* uses this logic. In addition, this logic is described in the NRC's working draft of *Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants* [See Chapter 3, Section III.B].

Section 3.2 of Exhibit A of the Application describes the process used by Duke to identify the *applicable* aging effects for the structures and components subject to an aging management review. First, a list of *potential* aging effects to consider for Oconee structures and components subject to aging management review were identified by

reviewing available industry literature. As noted in NRC Inspection Procedure 62002, *Inspection of Structures, Passive Components, and Civil Engineering Features at Nuclear Power Plants*, thermal effects may cause widespread cracking [NRC IP 62002, Section 62002-03.01(j)]. Therefore, the material and service environment required for the onset and propagation for each potential aging effect, such as cracking, were identified. Next, for those Oconee structures and components subject to an aging management review, the plant-specific materials of construction and service environment were identified. The potential aging effects became applicable aging effects for the Oconee structures and components both when the plant-specific materials of construction and service environment match the materials and service environment necessary for the aging effect to occur and when the component-aging effect combination could result in a loss of the component intended function during the extended period of operation if left unmanaged. To provide reasonable assurance that all applicable aging effects had been identified for the structures and components subject to an aging management review, NRC generic communications, industry experience, and relevant Oconee experience were also reviewed. The logic was not used to demonstrate that concrete cracking has not occurred. Oconee operating experience was reviewed and documented to validate the absence or occurrence of concrete cracking.

Based on the logical disposition of the aging effects, cracking of the Reactor Building internal structure's concrete is not an applicable aging effect. The Oconee Reactor Building concrete components have been designed and fabricated in accordance with well established industry standards and construction practices. By considering these features along with the plant operating environments to which the concrete components are exposed, concrete cracking is not an applicable aging effect.

RAI 3.7.7-3 (11-30-98A)

The applicant cites Section III.f.(i)(b) of the Statement of Consideration (SOC) to 10 CFR Part 54 to justify exclusion of certain structural components (i.e., caulking, sealant) from an AMR. As a matter of record, in response to a relevant question in the cited Section, the Commission states, "Absent the specific nature of the performance or condition replacement criteria (e.g., routine testing program) it is not appropriate for the Commission to generically exclude all such replacement programs for passive structures and components." The applicant should identify the specified replacement schedule for these components in order to justify their exclusion from the AMR program. See also staff position - RAI 3.3-19.

Response

The Commission has provided the staff the flexibility to approve on a case by case basis the replacement of components and materials based on condition monitoring or performance testing. This is in addition to replacement based on qualified life or specified time interval. The replacement of some caulking and sealants within the Oconee Reactor Building will be in accordance with the Inspection Program for Civil Engineering Structures and Components which is described in Section 4.19 of Exhibit A of the Application. Acceptance criteria for this program are described on page 4.19-2 of the Application.

The replacement of other caulking and sealants within the Oconee Reactor Building will be in accordance with ASME Section XI, Subsection IWE, Examination Category E-D, which has been endorsed by the NRC in 10 CFR Part 50, §50.55a.

RAI 3.7.7-6 (11/30/98A)

The Duke Power response to this RAI stated that the Inspection Program for Civil Engineering Structures and Components is relied upon by the licensee as the aging management program for unique structural items, such as sump screen and the Unit Vent Stacks. The program will be enhanced to include items that are currently not covered specifically in the program (e.g., sump screens). The program consists of visual inspection of structures and components for aging effects. The sump screens and the Unit Vent Stacks are inspected for loss of material. The licensee stated that the acceptance criteria are provided in Section 4.19 of the Oconee LRA and the acceptance criteria are defined as "No unacceptable visual indication of loss of material, cracking or change of material properties for concrete, and loss of material for steel," as identified by the accountable engineer. The licensee in Section 4.19 of the LRA also stated that inspected structures and components classified as acceptable are those structures and components that are capable of performing their intended function. The staff finds that the above definition of the acceptance criteria for the broadly used inspection program rather vague and non-specific. The staff is also concerned that the application of the acceptance criteria relies heavily on the judgment of a so-called "accountable engineer" whose minimum qualification, if any required, is not clearly defined. The licensee should provide additional information as well as guidance including some examples of past inspection to better define the acceptance criteria. The licensee also need to discuss what qualifications or conditions an engineer must meet in order to be qualified as an "accountable engineer." The licensee is also asked to list key examples of past inspection findings which led to classification of some inspected structures as "not acceptable." Pending satisfactory resolution of the above items by the licensee, this RAI remains open.

Response

The *Inspection Program for Civil Engineering Structures and Components* is described in Section 4.19 of Exhibit A of the Application. As noted in Section 4.19, Oconee management assigns the inspector or "accountable engineer". The individual is chosen based on education, work experience, etc. to ensure that the "accountable engineer" is well qualified. The qualifications of the "accountable engineer" are documented in Oconee site documents. The oversight of the training and qualification of the "accountable engineer" is governed by the Duke *Quality Assurance Topical Report.* The Duke *Quality Assurance Topical Report* is discussed with the Duke *Quality Assurance Program* in Section 4.13 of Exhibit A of the Application.

The acceptance criteria and the aging effects that are monitored are addressed in Section 4.19. Examination and assessment of the condition of a structure is performed by the "accountable engineer" using guidance provided in codes and standards such as:

- NEI 96-03, *Industry Guideline for Monitoring Structures*
- NRC Regulatory Guide 1.127, *Inspection of Water-Control Structures Associated with Nuclear Power Plants*
- ACI 349.3, *Evaluation of Existing Nuclear Safety-Related Concrete Structures*

The acceptability of a structure is based on whether the "accountable engineer" determines the structure is capable of performing its intended function(s). Acceptability based on condition monitoring and the capability to perform the intended function is described in NRC Regulatory Guide 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*. The condition of the structure is determined by visual inspection for degradation such as:

- Spalling, cracking, delaminations, honeycombs, water in-leakage, and chemical leaching of concrete
- Cracks in joints of masonry walls
- Corrosion, peeling paint, cracked welds, and loose or missing anchors of structural steel
- Settlement and cracked concrete for equipment foundations

Visual inspections for these types of degradation have been addressed in NRC Inspection Procedure 62002, *Inspection of Structure, Passive Components, and Civil Engineering Features at Nuclear Power Plants*, and NEI 96-03, *Industry Guide for Monitoring Structures*.

Previous inspections of structures and structural components have identified some components as "not acceptable." The findings of these inspections are documented in

Section 4.19 of Exhibit A of the Application. Items that were found to be "not acceptable" were evaluated for continued service, monitored, or corrected. Specific corrective actions were implemented in accordance with Duke's *Quality Assurance Program.*

As discussed in the previous response to this RAI, inspections of the Unit Vent have identified loss of material due to minor corrosion (rust) of the Unit Vent Stack support steel. As part of the corrective action, the support steel was cleaned and recoated. The program was effective in identifying the aging effect in a timely manner and correcting the degradation prior to loss of the intended function.

Potential Open Item 4.3.2-2 (New)

In the LR application the acceptance criterion for evaluating degradation of the component due to selective leaching corrosion is based on the acceptability of wall thickness which will be judged in accordance with the Oconee component design code on record. However, loss of the ferrite phase from cast iron by selective leaching will affect not only its dimensions but also its mechanical properties, ductility being the major affected property. Basing acceptability of a component only on its measured thickness may not give a true picture and may significantly underestimate the degree of degradation which has occurred. The applicant should address this issue.

Response

The purpose of the Cast Iron Selective Leaching Inspection is to determine the existence of selective leaching of cast iron components. As noted in Section 4.3.2 of Exhibit A of the Application, a sample of cast iron components will be inspected performing a Brinnell Hardness check to determine the existence of selective leaching. A Brinnell Hardness check will detect a change in the ductility of the cast iron brought about by the selective leaching. Any indication of selective leaching of cast iron identified during the Cast Iron Selective Leaching Inspection will initiate the Problem Investigation Process (PIP) that could ultimately lead to a formalized program. The PIP is a process governed by a nuclear generation department directive that will initiate an engineering evaluation of the identified condition. The PIP evaluation is designed to consider the need to conduct a root cause analysis and determine corrective actions, which may include actions to prevent recurrence. For this example, the PIP evaluation will determine the need for a root cause analysis, including the establishment of methods to determine the amount of material lost. The evaluation will also determine corrective actions that assure a degraded component condition has not or will not exceed the limit required for qualifying the component for further service.

Potential Open Item 4.3.8-7 (New)

For the Component Cooler Tubing Examination, the Condensate Cooler Tubing Examination, the Decay Heat Cooler Tubing Examination, and the Main Condenser Tubing Examination, the applicant applies an acceptance criteria of 60% throughwall. The staff requests the applicant provide the basis for this value.

Response

An Oconee analysis has been performed that calculates the minimum wall thickness for the decay heat cooler tubes to be used for eddy current acceptance criteria. This cooler is safety related and an important barrier to the release of radioactive fluid to the environment. The decay heat removal cooler tubes are designed to ASME Section III, 1968 Edition. Subsection C of this Code states that the requirements of ASME Section VIII of the Code shall apply to the materials, design, fabrication, inspection, and testing. Therefore, the calculation is performed using ASME Section VIII, 1968 Edition, Winter 1969 Addenda.

Three possible failure modes could be experienced by the tubes while in service. The analysis calculates the wall thickness required for each of the three cases and uses the most restrictive case to determine the wall thickness acceptance criteria. The first case is the minimum wall thickness required for internal design pressure limited by circumferential (hoop) pressure stress. The thickness is determined using the equation from paragraph UG-27 of ASME Section VIII. The second case is minimum wall thickness required to withstand internal design pressure (longitudinal stress) plus bending moment stress due to flow-induced force and tube and fluid dead weight. The equation from paragraph UG-27 of ASME Section VIII was used to calculate the minimum thickness needed to withstand longitudinal stress. Actual physical parameters of the tubes were used to calculate the wall thickness required due to bending stresses. The third case is minimum wall thickness required to withstand external pressure. The procedure in part UG-28 of ASME Section VIII was used to calculate the minimum required wall thickness. This case was determined to be the most limiting case. Based on the resultant minimum wall required, a 60% wall loss or through wall indication criteria was conservatively chosen as the acceptance criteria for the eddy current testing.

The component coolers, condensate coolers and main condenser are nonsafety components and do not possess a rigorous calculation similar to that described above. For these heat exchangers, Duke follows accepted industry practice that considers a 60% wall loss or through wall indication criteria a conservative standard for eddy current

testing acceptance criteria. For these nonsafety heat exchangers, the evaluation of every tube identified to contain a 60% wall loss or through wall indication is considered proactive and conservative.

Potential Open Item 4.3.8-8 (New)
The applicant should provide operating experience that demonstrates the effectiveness of the PM activities. The applicant plans to perform a formal documentation of this as part of the Preventive Maintenance Activity Assessment. The applicant plans to complete this assessment by February 2013. The staff requests the applicant provide this assessment to the staff upon completion. This is a Confirmatory Item.

Response
The original commitment to perform a one-time assessment of ten specific Preventive Maintenance Activities was made to ensure that these activities would be effective in managing aging during the period of extended operation. Subsequently, Duke has determined that the ongoing confirmation process that is performed as part of these activities will ensure that the Preventive Maintenance Activities are effective in managing aging. The performance of both an ongoing confirmation process and a one-time assessment is not necessary to demonstrate the effectiveness these activities. Accordingly, the commitment to perform the one-time assessment of the ten Preventive Maintenance Activities by February 2013 (Section 4.3.8 of the Application) is withdrawn.

Potential Open Item 4.3.8-9 (New)
The applicant committed to performing these PM activities consistent with the descriptions provided in the December 14, 1998, submittal. The applicant will revise the UFSAR Supplement (Exhibit B of the application) by November 30, 1999, to reflect this commitment. This is a Confirmatory Item.

Response
Duke agrees that this potential open item is a Confirmatory Item.

Potential Open Item 4.3.9-6 (New)
For the Reactor Building Spray System Inspection, the applicant stated it will inspect the most bounding of six susceptible locations. Confirm that these six locations consist of the entire susceptible population for this system. If not, provide the basis for selecting these six locations. Provide the parameters you will evaluate to select the most bounding or representative inspection location.

In the response to RAI 4.17-1, the applicant provided information related to operating experience related to Heat Exchanger Performance Testing Activities. Confirm that no heat exchangers in this program have been found to not meet minimum required cooling capacity. If heat exchangers have been found to not meet minimum acceptance criteria, describe the ensuing corrective actions.

Response
Both statements concerning reactor building cooling unit test frequency are correct. Oconee Technical Specifications ITS 3.6.5.4 require the units to be tested on a refueling frequency. Additionally, Selected Licensing Commitment SLC 16.6.3 require the units to be tested as needed, based on projected fouling rates. When testing is performed, a conservative daily fouling rate is calculated and projected to determine when testing should be conducted again. Generally, this "as needed" testing has occurred on a quarterly basis.

As discussed in response to Potential Open Item 4.17-4, operability of these coolers is dependent on the heat removal capacities of the other containment heat removal systems. Each time test results are collected, applicable calculations are performed to determine total heat removal capacities. When heat exchangers have been found to not meet minimum acceptance criteria, ITS 3.6.5 provides corrective action, including out-of-service durations for these units.

Potential Open Item 4.17-4 (New) - "Heat Exchanger Performance Testing Activities"
How do you simulate or extrapolate to normal operating and accident conditions when performing the heat exchanger performance tests?

Response
Heat removal capacities of the Reactor Building Cooling Units and Decay Heat Coolers are interdependent. The heat removal capacities of those two coolers in conjunction with the Reactor Building Spray System determine total Containment Heat Removal capability. When heat exchanger performance test data are collected for the Reactor Building Cooling Units and Decay Heat Coolers, the data is used to effectively calculate a heat transfer coefficient. This heat transfer coefficient is then used to calculate heat removal capacity in normal and accident conditions using variables that apply to normal and accident conditions, such as lake temperature and Reactor Building atmospheric temperature. These calculations use the test data to prove that total required containment heat removal capacity of the Reactor Building Cooling Units, Decay Heat Coolers, and Reactor Building Spray System during normal and accident conditions exists. Details on

performance relative to conditions of the test setup and the models used to analyze test data can be found in Section 6.2.1.1.3, Subsection "Containment Heat Removal Systems", of the Oconee UFSAR.

Duke has decided to defer the portion of the response to Potential Open Item 4.17-4 (New) relative to the SSF HVAC coolers until after the staff issues the SER for Oconee License Renewal because Duke engineers responsible for the SSF HVAC coolers were not available to assist in the preparation of the response.

Potential Open Item 4.17-5 (New)

In Section 3.5.2, page 3.5-3 of the LRA, the applicant lists systems with raw water that may be susceptible to fouling. The staff noted this list appears to be incorrect in that neither 3.5.11 nor 3.5.12 mention raw water. Also, there appears to the staff that there are more systems with raw water and potential fouling issues than those listed (for example, 3.5.3, 3.5.5 and 3.5.14 have raw water and fouling issues but are not included on the list on page 3.5-3). Please explain or correct these apparent discrepancies.

Response

The list of sections in Section 3.5.2 of Exhibit A of the Application that contain raw water systems is incorrect. The list of sections should be 3.5.6, 3.5.13, and 3.5.14. Sections 3.5.11 and 3.5.12 should not have been in that list. Sections 3.5.3 and 3.5.5 referenced in the potential open item do not contain raw water systems and therefore should not have been in the list. Fouling is mentioned in these sections because the process systems "own" the heat exchangers and there is raw water on the cooling side of the heat exchangers.

Potential Open Item 4.17-6 (New)

In Section 3.5.14.4, the applicant stated that flow rate of the cooling water through the SSF HVAC cooling units is measured, but not the delta-temperature across this heat exchanger. Without additional information, the staff considers this inadequate because it appears that not all aging effects are managed. For example, one of the aging affects identified by the applicant is loss of material for the aluminum fins of the cooling coils. If one assumed some or all of these fins were broken such that cooling capacity is degraded, this condition will not be identified by the applicant because the flow rate through the condenser tubes, which is the only parameter the applicant is measuring, will remain the same. Thus, the staff concludes measuring just flow rate is not enough to verify the cooling units are maintaining their heat transfer capacity.

Response

Duke has decided to defer the portion of the response to Potential Open Item 4.17-4 (New) relative to the SSF HVAC coolers until after the staff issues the SER for Oconee License Renewal because Duke engineers responsible for the SSF HVAC coolers were not available to assist in the preparation of the response.

Potential Open Item 4.17-7 (New)

For the decay heat removal coolers and the reactor building cooling units, the applicant determines heat removal capacity and compares the test results to the acceptance criteria as well as to previous test results. For the SSF heat exchangers, the applicant verifies acceptable cooling water flow rates through these heat exchangers. Please specify what are the acceptance criteria and what they are based on.

Response

Please refer to the response to Potential Open Item 4.17-4 for information concerning acceptance criteria for the Reactor Building Cooling Units and the Decay Heat Coolers.

Duke has decided to defer the portion of the response to Potential Open Item 4.17-4 (New) relative to the SSF HVAC coolers until after the staff issues the SER for Oconee License Renewal because Duke engineers responsible for the SSF HVAC coolers were not available to assist in the preparation of the response.

Potential Open Item 4.17-8 (New) - Heat Exchanger Performance Testing Activities

The heat exchanger performance testing activities described in Section 4.17 of the LRA discuss, in general terms, the testing and maintenance activities related to the standby shutdown facility heat exchangers, but there is not specific reference to NRC Information Notice (IN) 97-41 "Potentially Undersized Emergency Diesel Generator Oil Coolers." It is not clear that the applicant has reviewed the applicability of this IN to the Oconee heat exchangers. Since fouling of heat exchanger tubing has been identified as an applicable aging effect, appropriate actions to avoid problems similar to those discussed in the IN may be necessary for Oconee heat exchangers.

Response

Information Notice 97-41, "Potentially Undersized Emergency Diesel Generator Oil Coolers," notes that EDG oil coolers designed and constructed prior to 1985 would be considered undersized when inputting the design data into the new industry design equation developed in 1985. The design equation for cooling *viscous* shell side fluids was modified after extensive testing of industrial-sized heat exchangers. The Information

Noticed noted that this does not necessarily mean that they are now inadequate for meeting the design requirements, but they do have a lower fouling margin that would require more frequent cleaning and testing. This Information Notice was reviewed within Duke for applicability to Oconee EDG oil coolers. The results of this review determined that the oil coolers were adequately sized and no corrective actions were needed. Section 4.17 of Exhibit A of the Application presents Heat Exchanger Performance Testing Activities for managing fouling of the reactor building cooling units, decay heat removal coolers, and the SSF HVAC heat exchangers in the SSF Auxiliary Service Water System. None of these heat exchangers are oil coolers. Therefore, the Information Notice was considered not relevant for these heat exchangers.

Potential Open Item 4.23-2 (New)
Discuss why the RCS Operational Leakage Monitoring program is not credited for managing aging effects associated with the reactor coolant piping casing.

Response
RCS Operational Leakage Monitoring is credited for managing cracking, loss of preload, and loss of material for RCP bolting (see Exhibit A of the Application, Table 3.4-1, Pages 3.4-33 and 3.4-34). Duke acknowledged that the RCS Operational Leakage Monitoring program was inadvertently omitted from the Application in the response to RAI 3.4.8-4 dated February 17, 1999. In addition, Duke does not believe that RCS Operational leakage Monitoring is an appropriate aging management program for cracking at the welded joints that join the pump casing halves.

Potential Open Item 4.23-3 (New)
Confirm Table 3.5-3 contains appropriate references to the RCS Operational Leakage Monitoring for the HPIS. It appears from this table that this program is credited with managing aging effects only for the RCP coolers and seal return coolers and not the Class 1 pressure boundary portion of the HPIS as stated in the applicant's response to RAI 4.23-1, dated February 8, 1999.

Response
Table 3.5-3 contains the appropriate references to the RCS Operational Leakage Monitoring for the non-Class 1 portions of the High Pressure Injection System. The Class 1 portions of the High Pressure Injection System are presented with the Reactor Coolant System portion in Exhibit A of the Application. In addition, the Class 1 portions are highlighted in yellow on the license renewal flow diagrams OLRFD-101A-1.1, 1.4, 2.1, 2.4, 3.1, and 3.4.

Table 3.4-1 on pages 3.4-34, 35, and 36 contains entries that include the Class 1 portions of the High Pressure Injection System. The component entries either specifically identify the High Pressure Injection System component or refers to them as branch lines. The Class 1 portions of the High Pressure Injection System range in size from ¾-inch to 4-inches.

The third table entry on page 3.4-34 includes the 4-inches piping. Note that RCS Operational Leakage Monitoring is not credited with managing aging of these components. Our philosophy is not to credit RCS Operational Leakage for managing aging of components 4-inches and larger. The Inservice Inspection Plant manages these components.

Potential Open Item 4.23-4 (New)
To provide for early warning of leakage, the applicant relies on its containment air monitoring of radioactivity and the containment sump level monitoring. To monitor its primary-to-secondary leakage through steam generator tubes, the applicant relies on effluent monitoring in the secondary systems or by comparison of primary and secondary radioisotope concentrations. How often are these parameters monitored? If not continuously, provide monitoring frequency.

Response
An installed radiation monitor continuously monitors containment airborne radioactivity. Containment sump level is monitored continuously by an installed level instrument. An installed radiation monitor on the air ejectors of each main condenser continuously monitors for radioactivity resulting from primary-to-secondary leakage through steam generator tubes.

Potential Open Item 4.25-3 (New)
The staff identified the following discrepancies within the applicant's LRA. The staff considers these minor, however, Duke should address these discrepancies.

(a) Section 4.25 includes within its scope the Low Pressure Injection System. This system is not listed in Section 3.5.2.4.

(b) Section 3.5.2.4 includes the SSF Sanitary Lift System. This is not consistent with Section 3.5.14.6 and Table 3.5-12 because no raw water environment appears to be in this system.

(c) The description of component materials in 2.5.6.3 does not match the description in 3.5.6.3.

(d) The description of component materials in 2.5.6.4 does not match the description in 3.5.6.4.

(e) The description of component materials in 2.5.6.5 does not match the description in 3.5.6.5.

(f) The description of component materials in 2.5.13.6 does not match the description in 3.5.13.6.

Response

(a) The Service Water Piping Corrosion Program presented in Section 4.25 of Exhibit A of the Application is credited with managing loss of material of the carbon steel shell of the decay heat removal coolers exposed to the raw water supplied by the Low Pressure Injection System. In Section 3.5.2, mechanical systems within the scope of license renewal are grouped and listed according to their primary internal environment. Systems are not listed in Section 3.5.2 for their secondary environments. The primary internal environment of the Low Pressure Injection System is borated water and is listed in Section 3.5.2.2. As noted in Section 3.5.5.3.1, Section 3.5.2.4 identifies the applicable aging effects for the Low Pressure Injection System carbon steel components exposed to raw water.

(b) The primary internal environment of the SSF Sanitary Lift System is air, and the system should have been listed in the Section 3.5.2.1 instead of Section 3.5.2.4. The SSF Sanitary Lift System receives wastewater from the SSF drinking water fountain and toilets. Waste collection systems are designed for gravity to move wastewater along, leaving an air environment. Since the SSF is not normally manned, the SSF Sanitary Lift System is infrequently used. Therefore, the primary environment of the system is air.

(c) The materials of construction of the Condenser Circulating Water System components are carbon steel, stainless steel, brass, bronze, and cast iron. Table 2.5-9 lists the correct materials of construction. Section 2.5.6.3 should have listed the same materials of construction listed in Table 2.5-9. Section 3.5.6.3 lists only those materials of components that require an aging management program to manage the applicable aging effects that would result in a loss of the component intended function if left unmanaged for the period of extended operation.

(d) The materials of construction of the High Pressure Service Water System components are brass, bronze, carbon steel, cast iron, copper, and stainless steel. Section 3.5.6.4 lists only those materials of components that require an aging management program to manage

the applicable aging effects that would result in a loss of the component intended function if left unmanaged for the period of extended operation.

(e) Section 3.5.6.5 should not have listed cast iron as a material of construction. No components in the Low Pressure Service Water System are constructed of cast iron. Section 2.5.6.5 correctly lists the materials of construction of the Low Pressure Service Water System.

(f) The materials of construction of the Service Water System components are bronze, carbon steel, cast iron, ductile cast iron, and stainless steel. Tables 2.5-23 and 3.5-11list tubing constructed of brass, bronze, copper, and carbon steel. No brass, bronze, copper, and carbon steel tubing exists in the Service Water System. Tubing should not have been listed in Tables 2.5-23 and 3.5-11. As a result, brass and copper are not materials of construction for this system. Other components constructed of bronze do exist in the Service Water System as shown in Tables 2.5-23 and 3.5-11.

Potential Open Item 4.25-4 (New)
The staff identified the following discrepancies within the applicant's LRA. The staff requests the applicant clarify these discrepancies so that the staff can ensure all aging effects, systems and components are completely and correctly identified for aging management:

(a) On page 3.5-118 and 3.5-121 in Table 3.5-4, page 3.5-140 in Table 3.5-11, and page 3.5-149 in Table 3.5-12, the applicant stated that several components in the Condenser Circulating Water System, High Pressure Service Water System, Keowee Service Water System, and the SSF Auxiliary Service Water System are not subject to aging effects. Without additional clarification from the applicant, the staff believes this finding contradicts the applicant's position on aging effects stated in Section 3.5.2.4. Explain why no aging management programs are identified for the various materials exposed to a raw water environment.

Response
The section of Table 3.5-4 of Exhibit A of the Application in question is related to the Condenser Circulating Water System. The statement that there are no aging effects is somewhat misleading. Actually, the components in question have no component intended function and therefore do not require aging management review. The explanation for each component in question will be handled separately.

Recirculating Cooling Water Heat Exchanger: Heat transfer is not a required function of these coolers. The coolers do form part of the Condenser Circulating Water pressure

boundary. However, the Recirculating Cooling Water System operates at a higher pressure than the Condenser Circulating Water System. If a tube were to leak, water from the Recirculating Cooling Water System would leak into the Condenser Circulating Water System. The pressure boundary function is therefore not required for these coolers. If left unmanaged, loss of material in the Recirculating Cooling Water Heat Exchangers will not result in a loss of the Condenser Circulating Water System pressure boundary.

Screens: The screens in the Condenser Circulating Water System are collector screens that remove condenser tube cleaning balls from the Condenser Circulating Water System flowpath after the balls travel through the condenser tubes. They are in-line components and have no pressure retaining parts. Also, the filtration function of the collection screens is not required to support the system intended function. Therefore, since they perform no component intended function, the collection screens in this system are excluded from aging management review.

Tubing: There are no instruments with associated tubing in the Condenser Circulating Water System that are required to function to support any system intended function. Additionally, if pressure boundary integrity of this small diameter tubing is lost, Condenser Circulating Water System intended function would not be lost. Therefore, since pressure boundary is not required of these components, they are excluded from aging management review.

The section of Table 3.5-11 of Exhibit A of the Application in question is related to the Keowee Service Water System. Tubing should not have been included in this system at all. There is no tubing in the Keowee Service Water System.

The inconsistency associated with the cast iron pump casing on page 3.5-149 in Table 3.5-12 is addressed in the response to Potential Open Item 3.5.14.7.1-1.

Potential Open Item 4.25-5 (New)

As stated on page 4.25-1, under Section 4.25.1, the scope of the Service Water Piping Corrosion program includes all bronze, carbon steel, cast iron and stainless steel components exposed to raw water and included within the scope of license renewal. (The staff notes that this statement is not consistent with the introductory paragraph on the same page.) How is loss of material managed for the other material types exposed to raw water; e.g., copper, brass, and ductile iron?

Response
Duke has decided to defer the response to Potential Open Item 4.25-5 (New) until after the staff issues the SER for Oconee License Renewal because the Duke engineers responsible for the Service Water Piping Corrosion Program were not available to assist in the preparation of a response.

Potential Open Item 4.25-6 (New)
The applicant stated that the focus of the Service Water Piping Corrosion Program to date is on the carbon steel piping components exposed to raw water because they are the most susceptible to general corrosion and can serve as a leading indicator of the general material condition of the system components (page 4.25-1). Thus, the staff assumes that the applicant has not performed and has no plans to perform, at this time, inspections of components fabricated from materials other than carbon steel. The staff is unaware of any relationship between the course of general corrosion of carbon steel components and pitting or MIC attack of stainless steel components. Provide the technical basis for relying on inspections of carbon steel component for general corrosion to "serve as a leading indicator" of the condition of other components fabricated from other materials and susceptible to other corrosive mechanisms such as pitting or MIC. Without additional information, the staff finds the program, as described in the LRA, insufficient to manage aging effects for components fabricated from materials other than carbon steel.

Response
Duke has decided to defer the response to Potential Open Item 4.25-6 (New) until after the staff issues the SER for Oconee License Renewal because the Duke engineers responsible for the Service Water Piping Corrosion Program were not available to assist in the preparation of a response.

Potential Open Item 4.25-7 (New)
The applicant stated that the program does not currently include inspections of the Keowee systems because the components in that system remain bounded by the overall program results. State specifically how the Keowee system is bounded.

Response
Duke has decided to defer the response to Potential Open Item 4.25-7 (New) until after the staff issues the SER for Oconee License Renewal because the Duke engineers responsible for the Service Water Piping Corrosion Program were not available to assist in the preparation of a response.

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Potential Open Item 4.25-8 (New)

The applicant inspects the bounding locations in the various system within the scope of the Service Water Piping Corrosion Program using ultrasonic test techniques, supplemented by visual inspections if access to the interior surfaces is allowed such as during plant modifications. The staff finds this technique acceptable for general corrosion, but questions the validity of this technique for detecting localized degradation such as pitting or MIC. Describe more fully your inspection technique to justify the use of UT for localized degradation.

Response

Duke has decided to defer the response to Potential Open Item 4.25-7 (New) until after the staff issues the SER for Oconee License Renewal because the Duke engineers responsible for the Service Water Piping Corrosion Program were not available to assist in the preparation of a response.

RAI 5.3.1-1 (11-19-98)

The response to RAI 5.3.1-1, dated February 8, 1999, refers to Table 5.2 of the Oconee UFSAR. Table 5.2 of the Oconee UFSAR shows 360 design cycles for heat up from 70 F to 8% power and cooldown back to 70 F. However, the table also shows other normal operating design transients, such as 1440 cycles of power change from 0% to 15% and back to 0%, 18000 cycles of power loading from 8% to 100% and unloading back to 8%, and 8000 cycles of 10% load increase and decrease. Provide a justification as to why the thermal expansion of the RCS under these additional cycling conditions, and its effect on the steam and feedwater lines, should not be included in the fatigue assessment of the containment liner penetrations for the period of extended operation of these components that are shown in Figure 3.20 of the UFSAR.

Response

Duke has decided to defer addressing the follow-up question to RAI 5.3.1-1 until after the staff issues the SER for license renewal because the Duke engineers responsible for this issue were not available to assist in the preparation of a response.

RAI 5.3.2-2 (11-19-98)

In responses to RAIs 4.8-1, 4.8-2, and 4.8-3, the applicant provided information regarding the aging management programs related to the post-tensioning tendon system of the containments of the three Units at Oconee. To establish a reasonable assurance regarding the quantitative aspect of the tendon force TLAA, the applicant should provide information regarding the trending of the measured prestressing forces in the Oconee containments. If an adequate sampling is not available to perform a reliable regression analysis to construct trend lines, this information should be collected in the future and provided by the applicant to confirm the accuracy of the tendon monitoring activity. This remains an open item.

Response

Programmatic oversight of the containment post-tensioning system is discussed in Section 4.8 of Exhibit A of the Application. The programmatic oversight includes testing, evaluation, and reporting. The documentation of the program and trending of the loss of prestress are maintained on site and available for staff review. Duke responses to RAIs 4.8-3, 5.3.2-1, and 5.3.2-2 contains additional information on the containment post-tensioning system program.

Duke has evaluated the loss of prestress in the post-tensioning system for 60 years of plant operation. Discussion of this evaluation was provided in a Duke Power letter dated March 2, 1998. The letter describes the extrapolation of the prescribed lower limit line to 60 years of operation. The prescribed lower limits for all three tendon groups remain

above the minimum required values. The graphs of the prescribed lower limit and minimum required value for each tendon group were provided to the NRC in a letter dated September 23, 1997 and included as Figures 1, 2, and 3 of Appendix 16.6-2 of Exhibit B of the Application.

The Nuclear Regulatory Commission staff visited the Oconee site April 27 – 30, 1998. The public announcement of the visit and topics for the visit were addressed in an NRC April 15, 1998 letter. As part of this visit, the NRC reviewed the following information related to tendon surveillance:

- Results of the latest tendon surveillance test.
- Maintenance and inspection records and site issued reports related to aging degradation for the last 5 years for the containments.
- Discussion with plant personnel about selection of random tendons to pursue the trend analysis for the 60 year renewal period, frequency of readings, etc.

Oconee Improved Technical Specification SR 3.6.1.3 requires that containment structural integrity be verified in accordance with the Containment Tendon Surveillance Program. Oconee Improved Technical Specification (ITS) 5.5.7 describes the *Pre-Stressed Concrete Containment Tendon Surveillance Program.* Oconee Improved Technical Specification 5.6.7 requires that a report be sent to the NRC when abnormal degradation is identified during an inspection.

<u>RAIs 5.4.1-2, 5.4.1-3 and 5.4.1-4</u>

The RAIs involve TLAAs that are not completed. Duke indicates that these items will be completed prior to the extended period of operation. Duke further indicated that after the evaluations are completed, these items would be managed by the Thermal Fatigue Management Program. Duke indicated that this complies with 54.21(c)(1)(iii). Completion of the additional evaluations prior to the extended period of operation are confirmatory items.

Status: Duke provided supplemental response on 3/29/99.

<u>Response</u>

Duke agrees that this potential open item is a Confirmatory Item. The re-analysis of the Unit 3 EFW nozzle is expected to be complete by August 1, 1999 (See response to RAI 5.4.1-2). The re-analysis of attached piping that had been originally analyzed to USAS B31.7 Class II standards is scheduled for completion by August 31, 1999 (See response to RAI 5.4.1-3). The supplemental response to NRC Bulletin 88-08 will be provided by July 1, 2000 (See page 5.4-7 of Exhibit A of the Application).

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Potential Open Item 5.7.1-2 (New) - TLAA for fatigue of Polar Cranes

The fatigue due to lift cycles of the polar cranes (PCs) at/or near-rated loads is considered by the applicant to be a time-limited aging analysis for Oconee because the analyses meet all of the criteria contained in §54.3. The analyses addressing heavy load lifts for the polar and spent fuel cranes are summarized in Section 5.7.1 of the LRA. According to the applicant, these analyses demonstrate that the fatigue requirements are satisfied for the period of extended operation. Typically, some of the components of the PC system, such as PC rails, are constructed of carbon steel, which has a lower allowable stress range. The applicant's analyses do not distinguish between components which have different allowable stress ranges. The applicant should provide a justification that the lower limit of the stress range will not be exceeded during service life.

Response

The Oconee polar crane rails and girders were evaluated for stress cycles which may cause fatigue. Fatigue of the crane rails and girders is discussed in Section 5.7.1 of Exhibit A of the Application. The crane rails and girders are constructed of A36 steel and A7 steel, respectively. For 60 years of operation with the crane lifting at or near its rated capacity, the crane will be subjected to approximately 243 cycles. The number of projected cycles is much less than the minimum number of allowable cycles for any steel, 20,000 cycles. Since the crane is not expected to exceed the originally assumed number of design loading cycles, the original design remains bounding, and fatigue is not considered an applicable aging effect for the polar crane rails and girders.

Attachment 2

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Duke's Response to RAI 2.3-8

As stated in Section 2.3.2 of Exhibit A of the Application, the lower tendon access gallery does not support the intended functions of the Containment and is therefore not within the scope of license renewal. The scoping requirements of 10 CFR 54.4(a)(2) include all non-safety related systems, structures, and components whose failure could prevent satisfactory accomplishment of any of the functions identified in paragraphs 10 CFR 54.4 (a)(1)(i), (ii), or (iii). The function of the tendon access gallery is to provide access to the bottom of the vertical tendons so that they can be tested. Loss of function of the tendon access gallery is highly unlikely and to consider such is hypothetical. In order for the lower tendon access gallery to interact with the Containment structure, a hypothetical failure of the gallery would need to be assumed. As per NEI 95-10 Rev. 0, consideration of hypothetical failures that could result from system interdependencies that are not part of the CLB and that have not been previously experienced is not required. The discussion of whether the tendon access gallery performs an intended function has previously been discussed in Duke's January 14, 1998 response to RAI 2.3-3.

The aging effects of water infiltration into the tendon gallery on the tendon anchorage system are discussed in Sections 3.3.4.2 of Exhibit A of the Application and were previously provided in Response to RAI 2.3-3 which was clarified in the NRC letter to Duke on May 26, 1998.

Staff Concern - RAI 2.3-8

As stated in the applicant's response, this issue has been well discussed. The staff believes that the tendon galleries provide a significant function of protecting the anchorages of vertical tendons, and they should be within the scope of license renewal in accordance with 10 CFR 54.4(a)(2). Otherwise, Duke must provide assurance through the aging management program that the tendon anchorage function is not jeopardized by the environment in the tendon gallery.

Duke's Supplemental Response

As discussed in Section 3.3.4.2 of Exhibit A of the Application, the tendon anchorage located in the tendon gallery are exposed to a humid air environment. The tendon anchorage on the external surface of the dome are also exposed to a humid air environment. Therefore, the tendon anchorage located in both the tendon gallery and the dome are exposed to a moist environment. The program that has been credited to manage the aging effects is the same for the tendons located in the tendon gallery and the dome. The program, *Containment Inservice Inspection Plan*, is discussed in Section 4.8.2 in Exhibit A of the Application. The *Containment Inservice Inspection Plan* includes inspection of the tendon anchorage hardware. The inspection is conducted to manage loss of material and cracking of the tendon anchorage hardware. Based on the discussion

in Section 4.8.2, the implementation of the *Containment Inservice Inspection Plan* provides reasonable assurance that the aging effects of the tendon anchorage, including those located in the gallery, will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

Duke's Response to RAI 2.3-9

Sections 2.3.3.2 and 2.7.7 of Exhibit A of the Application include discussions of the miscellaneous attachment welds to the liner. Miscellaneous attachment welds to the liner are within the scope of license renewal and subject to aging management review. Section 2.3.3.2 states that the welds are not considered to be within the evaluation boundary of the Reactor Building (Containment). These welds are not considered part of the pressure retaining boundary of Containment. This position is consistent with the jurisdictional boundary for these welds as defined by ASME Section XI Subsection IWE. These welds are considered to be within the evaluation boundary of the Reactor Building internal structures that are addressed in Section 2.7.7. Section 2.7.7 discusses the steel components subject to an air environment. These components are identified in Table 2.7-5 of Exhibit A of the Application. The table includes cable tray supports, equipment component supports, stair supports, platform supports, etc. which may be welded to the liner.

As identified in Duke's August 12, 1998 letter to the NRC in Response to RAI 2.3-4, these welds will be addressed with the *Inspection Program for Civil Engineering Structures and Components* (or by ASME Section XI rules if ASME clarifies that these welds are in the scope of Subsection IWE). At this time, the ASME has not clarified that these welds are within the scope of Subsection IWE. Therefore, Duke has determined that the welds are not within the Containment boundary. The welds have been determined to be included within the boundary of the attachment. The Inspection Program for Civil Engineering Structures and Components is discussed in Section 4.19 of Exhibit A of the Application.

Staff Concern - RAI 2.3-9

The staff's concern is the effects of aging degradation of the attachment welds on the liner integrity. The staff believes these welds should be identified in Section 2.3.3.4 of the license renewal application and the aging management program for these welds should be within scope of ASME Section XI Subsection IWE.

Duke's Supplemental Response

As identified in Duke's August 12, 1998 letter to the NRC in Response to RAI 2.3-4, these welds will be addressed with the *Inspection Program for Civil Engineering Structures and Components* (or by ASME Section XI rules if ASME clarifies that these

welds are in the scope of Subection IWE). At this time, the ASME has not clarified that these welds are within the scope of Subsection IWE. Therefore, Duke has determined that the welds are not within the Containment boundary. The aging degradation of the welds will be adequately managed by the *Inspection Program for Civil Engineering Structures and Components.*

Potential Open Item 2.5-1 (New)
Rules for highlighting of OLRFD drawing in the front of each OLRP-1002 volume contains the statement, "All instrumentation lines normally open to the process system through, but not including the instrument, are included in License Renewal. These lines are not highlighted except for containment penetrations."

Section 2.5 of the application lists the mechanical systems within the scope of license renewal and provides a table at the end of Section 2.5 for each system identifying the components that are subject to an aging management review. Several of these systems do not include "tubing" as a component subject to an AMR. Below is a list of systems within the scope of license renewal (as indicated by the application), with instrumentation open to portions of those systems that are within scope, but do not list tubing as a component subject to an aging management review in the table at the rear of the section. For each system listed below, determine whether tubing should be listed on the system's table of components subject to an AMR, or provide a justification for its omission.

- Reactor Building Cooling
- Reactor Building Spray
- Component Cooling
- Condenser Circulating Water
- Auxiliary Building HVAC
- Feedwater

In addition, the SSF HVAC system diagram was not available for review by the staff. Tubing was not listed with the components subject to an AMR for this system. Please review the portions of this system that are within scope of license renewal and state whether tubing should be listed with the components subject to an AMR or provide a justification for its omission.

Response
The results of a review of the appropriate OLRFD series for the following systems yielded the following determinations about the inclusion of tubing within the systems' table of components subject to aging management review:

- Reactor Building Cooling System - No tubing exists within the license renewal evaluation boundaries of this system.
- Reactor Building Spray System - Stainless steel tubing is included in the scope of license renewal and was inadvertently omitted from tables 2.5-3 and 3.5-7 of Exhibit A of the Application for this system.
- Component Cooling System – Stainless steel tubing is included in the scope of license renewal and was inadvertently omitted from Tables 2.5-5 and 3.5-2 of the Application for this system.
- Condenser Circulating Water - The license renewal portion of this system does contain tubing. The tubing serves no component intended function and, therefore, is not subject to aging management review. Since the tubing does not require an aging management review, tubing is not listed in the Application for this system.
- Auxiliary Building Ventilation System - No tubing exists within the license renewal evaluation boundaries of this system.
- Feedwater System - Stainless steel tubing is included in the scope of license renewal and was inadvertently omitted from tables 2.5-15 and 3.5-7 of the Application for this system.
- SSF HVAC System - No tubing exists within the license renewal evaluation boundaries of this system.

Potential Open Item 2.5.2-1 (New)

Section 2.5.2 describes the process used by the applicant to scope and screen the systems, structures and components subject to an aging management review. However, details regarding this methodology that would give the staff an understanding about how the requirements of 10 CFR 54.21 are being met are not provided. Please provide a brief narrative that explains how the screening of SSCs within the scope of license renewal is actually performed.

Response

The mechanical component screening is consistent with the guidance provided in NEI 95-10, Rev. 0. Components subject to aging management review are those that are "passive" and "long-lived." A menu of every mechanical component type installed at Oconee was developed, going beyond the list of components in NEI 95-10. Using the "passive" and "long-lived" guidance, a determination was made for each of those mechanical component types. The components within the evaluation boundaries shown on the license renewal flow diagrams were "driven" through the menu to determine if the are subject to aging management review. From this exercise, a list of components subject to aging management review was developed.

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Duke's Response to RAI 2.7-7

The primary and secondary shield walls are included with the reinforced concrete beams, etc. in Table 2.7-5 of Exhibit A of the Application. The purpose of the primary and secondary shield walls are to provide biological and missile shielding [Reference Oconee UFSAR Section 3.5.1.1]. Therefore, intended function 1 (i.e., provides pressure boundary and/or fission product barrier) is not applicable to these components. The intended function that is associated with the biological shielding is intended function 3, "provides shelter/protection to safety-related equipment (including radiation shielding)." Intended function 3 is identified for these components in Table 2.7-5.

Staff Concern - RAI 2.7-7

The staff believes the primary and secondary shield walls are required to withstand certain differential pressures due to loss of coolant accidents as part of the subcompartment analysis. The implication in context of the license renewal is that the aging management programs for these walls should assure that the walls will perform this intended function during the license renewal period.

Duke's Supplemental Response

The purpose of the primary and secondary shield walls are to provide biological and missile shielding [Reference Oconee UFSAR Section 3.5.1.1]. The primary and secondary shield walls are also designed for differential pressure [Reference Oconee UFSAR Section 3.8.3.4]. Differential pressure may build up within the cavities created by the primary and secondary shield walls. The cavities do not contain the pressure. The pressure is released through the openings at the top and bottom of the shield walls. Therefore, the shield walls do not provide a pressure boundary. The function of the shield walls is "to provide shelter/protection to safety-related equipment." The aging management programs for the shield walls are discussed in Sections 4.19 and 4.28 of Exhibit A of the Application. The *Inspection Program for Civil Engineering Structures and Components* is discussed in Section 4.19. The *Tendon – Secondary Shield Wall – Surveillance Program* is discussed in Section 4.28. The implementation of these programs provides reasonable assurance that the shield walls will continue to perform their intended function in accordance with the current licensing basis for the period of extended operation.

Duke's Response to RAI 2.7-8

The prestressing forces and prestressing losses of the tendons in the secondary shield wall are not identified as a time-limited aging analysis (TLAA). The definition of TLAA is provided in

§54.3. To meet the definition of a TLAA, the analysis must meet six criteria. The analysis of the prestressing forces and losses of the secondary shield wall (SSW) tendons does not meet the sixth criteria, "contained or incorporated by reference in the CLB." The aging management program that addresses the prestressing forces and losses is discussed in Section 4.28, *Tendon – Secondary Shield Wall – Surveillance Program.*

Staff Concern - RAI 2.7-8

Oconee UFSAR (December 1997) Section 3.8.3.3 (related to the internal structures of the containment) states that the loads and load combinations considered for the design of the interior structures are described in UFSAR Section 3.8.1.3. Section 3.8.1.3 discusses the "calculated prestressing force," (after consideration of appropriate losses) as a load to be considered in load combinations tabulated in Table 3-14. Thus, the requirements for prestressing force set by the UFSAR for the Oconee prestressed concrete containments are applicable to the SSWs. Thus, maintaining the specified amounts of prestressing forces in the SSW tendons is part of the CLB. Unless the applicant demonstrates that the SSW integrity can be maintained without the precompression from the prestressing tendons, the prestressing tendon forces in the SSWs must be subjected to TLAA as required by 10 CFR 54.3.

Duke's Supplemental Response

Information concerning the performance of the secondary shield wall (SSW) tendon system is provided in Section 4.28 of Exhibit A of the Application. Section 4.28 contains the aging management program, *Tendon – Secondary Shield Wall –Surveillance Program.* In addition, response to RAIs 3.7.7-1, 4.28-1 and 4.28-2 provide additional details on the SSW tendon program.

The loads and load combinations for the design of the Reactor Building internal structures are described in Oconee UFSAR Section 3.8.1.3. Section 3.8.1.3 discusses the consideration of prestressing forces. This section addresses the Reactor Building Containment prestressing forces, not those of the secondary shield wall. Therefore, the requirements for the prestressing force set by the UFSAR for the containment tendon system are not applicable to the secondary shield wall tendon system. Loss of prestress of the secondary shield wall (SSW) tendons does not meet the definition of a TLAA as described in Duke's previous response to this RAI. While not a TLAA, loss of prestress of the SSW tendons is managed by the *Tendon – Secondary Shield Wall – Surveillance Program.* The results of the program (minimum required prestress, lift-off testing results, retensioning, etc.) are maintained on site and are available for staff review.

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Response to RAI 2.7-9

Cracking due to settlement (and differential settlement) of in-scope structures was identified as a potential aging effect in Section 3.7.2 of Exhibit A of the Application. The amount of settlement of a structure depends on the physical properties of the foundation material. These properties range from rock (with little or no settlement likely) to compacted soil (with some settlement expected). Cracking due to settlement was evaluated for each of the in-scope structures and was determined not to be an applicable aging effect based on the associated foundation material. Also, see response to RAIs 3.7.3-1, 3.7.6-1, and 3.7.9-1 for additional discussion of cracking due to settlement.

Staff Concern - RAI 2.7-9

For concrete structures, the uniform settlement of the structures affect the alignment of piping and electrical/instrumentation lines passing through these structures, and the differential settlement affects the stresses in the foundation slabs, and at the superstructure discontinuity areas. Eventually, these affects result in cracking of the structures. For steel structures (including cranes and monorails), the differential settlement results in higher stresses at connections, and misalignment of the interconnected components. Thus, the AMP related to settlement effects should incorporate more than managing the settlement related cracking in the concrete structures. On this basis, the staff believes that the settlement of structures (in general) should be considered in the aging management review.

Duke's Supplemental Response

The amount of settlement of a structure depends on the physical properties of the foundation material. These properties range from rock (with little or no settlement expected) to compacted soil (with some settlement expected). The aging effects due to settlement were evaluated for each of the in-scope structures. Based on the associated foundation material for each structure, no applicable aging effects due to settlement were identified for any of the in-scope structures. Also, see response to RAIs 3.7.3-1, 3.7.6-1, and 3.7.9-1 for additional discussion of settlement.

Potential Open Item 2.7-15 (New)

The staff has the following questions regarding the intake structure (Section 2.7.5 of OLRP-1001):

(a) The applicant did not address the function of the utility trench which is attached to the back of the intake structure. What is the function of this trench and does Duke consider the trench to be within scope of license renewal?

(b) Does the intake structure have internal components that serve as storm flood or rated fire barriers?

(c) Does the intake structure have a steel superstructure, lifting crane or rails on top?

Response
(a) The function of the utility trench attached to the back of the Intake Structure is to protect cables that run to the Intake Structure. The Intake Structure, along with the attached trench, is a Class 2 structure as defined by the Oconee UFSAR. Class 2 structures have been determined to be within the scope of license renewal.
(b) The Intake Structure does not contain any internal components that serve as storm flood or fire rated barriers.
(c) The Intake Structure does not have a steel superstructure, lifting crane or rails on top.

Potential Open Item 2.7-16 (New)
Section 2.7.7 of OLRP-1001 states that the pressurizer and quench tank are in one compartment. Are they in the steam generator compartment or in a separate compartment? A drawing to show the reactor building internals would be helpful.

Response
The pressurizer and quench tank are located within one of the steam generator compartments. Figures 1-3 and 1-5 in the Oconee UFSAR provide general arrangements of the Reactor Building with the layout of the NSSS equipment.

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Potential Open Item 3.7.1-3 (New)

In the discussion of the environment around the steel components in a fluid environment, Duke states a temperature limit of 183°F for the spent fuel pools at Oconee. This temperature has no effects on the steel components. However, it could have effects on the concrete of the spent fuel pool walls and slabs. The applicable Code (i.e. ACI 349) limits the concrete temperature to 150°F. This limit does not guard against additional cracking. However, it assures that the concrete properties, such as compressive strength and modulus of elasticity, would not be significantly affected. The applicant should discuss the aging effects of the temperature (183°F) on the concrete cracking and concrete properties.

Response

The temperature limits for the Oconee spent fuel pools are discussed in Section 3.7.1 of Exhibit A of the Application. Normal operating temperature for the spent fuel pool is below 150°F [Reference Oconee UFSAR Section 9.1.3.1]. Spent fuel pool normal operating temperature ranges from approximately 90°F to 120°F. This temperature is well below the threshold where degradation would occur to concrete. The temperature limit of 183°F in Section 3.7.1 of Exhibit A of the Application is incorrect. Temperature limits for the spent fuel pools are addressed in Sections 9.1.3.1 and 3.8.4.4 of the Oconee UFSAR. As discussed in Section 3.8.4.4 of the Oconee UFSAR, the spent fuel pool walls were analyzed for thermal loads.

Potential Open Item 3.7.1-4 (New)

The discussion of industry and Oconee specific experience data base in Sections 3.7.1 and 3.7.2 of OLRP-1001 does not capture (1) the essence of the results of the Oconee baseline inspections that would have been performed during the implementation of the Maintenance Rule, and (2) the instances of the reported unusual events, such as, the water leakages from the spent fuel pool liners. The conclusions drawn regarding the applicable aging effects would be affected by such additional database. Discuss such database in the applicable Sections.

Response

Industry and Oconee specific operating experience are included in the discussion in Sections 3.7.1 and 3.7.2 of Exhibit A of the Application. More detailed information concerning the findings during the implementation of the Maintenance Rule are included in Section 4.19. Also see response to RAI 3.7.2-1 included in Duke's February 8, 1999 letter.

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Potential Open Item 4.3.7-3 (New)

If the Oconee pressurizer spray heads do not satisfy the thermal embrittlement criteria specified in the RAI for cast stainless steel pump casings and valve bodies, they could be subject to significant thermal embrittlement and the proposed examination may require an enhanced VT-1 examination. Until the licensee determines whether the spray heads satisfy the thermal embrittlement criteria, this will be an open item.

Response

The following details provide the information available to Duke associated with the criteria provided for the reactor coolant pumps in Potential Open Item 3.4.8-6. The pressurizer spray heads are fabricated from ASTM A-351 grade CF-8M with no niobium; the casting method could not be determined. Metallurgical laboratory reports for the spray heads were located and the amount of ferrite is less than 21%. Since the spray nozzle is not a pressure-retaining item, reduction of fracture toughness would not compromise the pressure boundary and would only affect crack propagation. Therefore, the proposed visual examination described in section 4.3.7 of Exhibit A of the Application should be sufficient to detect cracking of CASS spray nozzle. For Duke to have a better understanding of what the NRC defines to be "significant thermal embrittlement," please identify the reference source for the associated criteria.

Potential Open Item 4.3.7-4 (New)

The applicant must identify when the surface examination of the removed pressurizer bundle will be performed.

Response

As described in Section 4.3.7.2 of Exhibit A of the Application, the surface examination of a sample population of peripheral pressurizer heater bundle penetration welds on one heater bundle will be performed when a heater bundle is replaced—replacement of the bundle would probably occur as a result of the inability to meet operational requirements for heater operation owing to inoperable heater elements. Heater bundle replacement may occur either prior to the period of extended operation or during the period of extended operation. Duke believes that this inspection may be deferred until the period of extended operation since the failure of a structural weld that attaches a heater sheath to the diaphragm plate would result in leakage within the make-up system capacity and the integrity of the heater bundle bolted closure would not be compromised.

Structural loads from the heater sheaths are taken by the diaphragm plate through the partial penetration welds that connect the heater sheaths to the diaphragm plate. These

loads are then taken out through the cover plate, which is bolted to the shell of the pressurizer, assuming no credit for the diaphragm plate-to-heater belt forging clad seal weld. Should any cracking of the heater sheaths, end plugs, heater sleeves, or the partial penetration weld connecting the heater sheaths to the diaphragm plate occur, the resultant loading on the diaphragm plate is the same. Should the partial penetration weld crack, a 360 degree through-wall crack would be required, due to the tight tolerances between the diaphragm plate and the heater sheath, in order for the full load to be transmitted to the castellated nut. The probability of this event occurring is considered to be extremely low. Please see the B&WOG response to RAI#25 regarding BAW-2244A for additional details concerning the heater sheath to diaphragm plate partial penetration welds.

Cracking of the partial penetration welds and heater sleeves (Oconee Unit 1) would result in minor leakage, limited by the tortuous path past the heater sheath collar and the castellated nut-to-diaphragm plate threads. In all cases, the structural integrity of the bolted closure would not be compromised, leakage would be detected, and appropriate action taken. Therefore, examination of the partial penetration welds after a bundle is removed is appropriate.

Potential Open Item 4.3.7-5 (New)

Since the Oconee-1 heater bundles are susceptible to PWSCC and the Oconee units 2 and 3 heater bundles are not susceptible to PWSCC, the heater bundle from Oconee-1 should be removed for surface examination. In addition, both the heater sheath-to-sleeve and heater sleeve-to-bundle diaphragm plate need to be inspected to determine whether the Alloy 600 materials in the heater bundle is susceptible to PWSCC.

Response

The Alloy 600 pressurizer heater bundle items (i.e., diaphragm plate, sleeves and associated welds) at Oconee Unit 1 are all within the scope of the Oconee Alloy 600 Aging Management Program—see Section 4.3.1 of Exhibit A of the Application. However, as discussed in the response to Open Item 4.3.7-4, Duke will only remove a heater bundle and perform an inspection of selected welds when a heater bundle is replaced as a result of inoperable heater elements. Cracking of the partial penetration welds and heater sleeves (Oconee Unit 1) by PWSCC would result in minor leakage, limited by the tortuous path past the heater sheath collar and the castellated nut-to-diaphragm plate threads. In all cases, the structural integrity of the bolted closure would not be compromised, leakage would be detected, and appropriate action taken.

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Potential Open Item 4.3.7-6 (New)

Since the license renewal term includes the fifth and sixth inspection interval, the inservice examination of the pressurizer heater bundle welds to Examination Category B-E must be performed during the fifth and sixth inspection intervals. This appears to be contrary to note 4 on page 4.3-19 of the Oconee license renewal application.

Response

The pressurizer heater bundle penetration welds will be included within Examination Category B-E or equivalent at each Oconee unit for the 5th and 6th inspection intervals. Please see the description of the Oconee Inspection Plan provided in Section 4.18 of Exhibit A of the Application (page 4.18-1).

Potential Open Item 4.3.7-7 (New)

The applicant is to confirm that the aging effects for the spray head are cracking and reduction in fracture toughness due to thermal aging of the cast stainless steel.

Response

As stated in Section 4.3.7 of Exhibit A of the Application, the applicable aging effect for the spray head is cracking due to reduction of fracture toughness. Since the spray head is not a pressure-retaining item, reduction of fracture toughness would not affect spray head function and would only affect crack propagation. Cracking of the spray head is the applicable aging effect. The proposed one-time inspection described in Section 4.3.7 will serve to verify this understanding.

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Potential Open Item 5.4.1-6 (New)

Section 5.4.1.2 of the license renewal application (LRA) describes time-limited-aging-analyses (TLAA) related to flaw growth acceptance for the reactor coolant system and Class 1 components at Oconee. As described in the LRA, inservice inspection (ISI) at Oconee, in accordance with ASME Section XI ISI requirements, has lead to the identification of crack-like indications, primarily in welds. The LRA states that fracture mechanics analyses used for flaw acceptance through the current license period have been reviewed for acceptability for the period of extended operation. This review has identified several general flaw locations that could not be demonstrated to be acceptable for the number of controlling design basis transients:

Oconee Unit 1:

- Pressurizer near heater bundle
- Pressurizer support lugs
- Steam generator at the upper head to tubesheet region
- Reactor vessel at the reactor vessel flange to shell region
- Control rod drive motor tube housings

Oconee Unit 2:

- Core flood tank dump valve to nozzle
- Pressurizer upper head to shell region
- Control rod drive motor tube housings

Oconee Unit 3:

- None

Management of these locations is through the Oconee "Thermal Fatigue Management Program."

Regarding these locations:

(a) Characterize the indications identified by the ISI for each of the locations listed (i.e., nature, length, through-wall extent and through-wall location).

(b) From the results of successive ISI of the same flaw locations, characterize the extent of growth of the indication(s) as indicated by the successive examinations.

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(c) For each of the fracture mechanics analyses, identify the transient and number of cycles assumed in the analyses, and the ASME Code Section XI, IWB-3600 criteria that was not satisfied at the end of the license renewal period.

(d) As of January 1, 1999, what is the status of the actual number of transient cycles for each location, the plant status regarding effective-full-power-years (EFPY), and the estimated EFPY at the end of the license renewal period?

(e) If the transient cycle count approaches or exceeds the allowable design limit, identify the corrective action steps that could be taken.

Response to Items (a), (b) and (c):
Duke has decided to defer the response to items (a), (b), and (c) Potential Open Item 5.4.1-6 (New) until after the staff issues the SER for Oconee License Renewal because the Duke engineers responsible for the Oconee Thermal Fatigue Monitoring Program were not available to assist in the preparation of a response.

The responses to items (d) and (e) follow:

Response to Item (d):

	Unit 1	**Unit 2**	**Unit 3**
Heatups as of 1/1/99	101	111	85
Cooldowns as of 1/1/99	102	117	83
EFPY as of 1/1/99	18.55	18.32	17.92
Estimated EFPY at the end of the license renewal period	48 EFPY	48 EFPY	48 EFPY

Response to Item (e):
The controlling transients identified in the fracture mechanics calculations, and summarized in Tables 1 and 2, are all being monitored by the Oconee *Thermal Fatigue Monitoring Program.* See the supplemental Duke response to RAI 5.4.1-5 provided by letter dated March 29, 1999.