April 16, 1999

Mr. William R. McCollum, Jr.
Vice President, Oconee Nuclear Site
Duke Energy Corporation
P. O. Box 1439
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SUBJECT: POTENTIAL SAFETY EVALUATION REPORT (SER) OPEN ITEMS REGARDING THE REVIEW OF THE OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3, LICENSE RENEWAL APPLICATION

Dear Mr. McCollum:

As you are aware, the Nuclear Regulatory Commission (NRC) staff is in the process of writing the SER for the Oconee license renewal application. The staff expects to issue this SER by June 17, 1999. At this time the staff has identified potential open items for the SER. In an effort to minimize the amount of SER open items it has been decided that the potential open items list will be made available to you (See Enclosure). The staff will support phone calls and meetings in an attempt to resolve as many of the open items as possible prior to issuing the SER. The aim of this process is to minimize the amount of SER open items without impacting the schedule for issuing the SER. It should be noted that the enclosed list represents the staff's potential open items as of this date. During the preparation of the SER, additional open items may be identified. To the extent it is practical, the staff will notify Duke of these additional open items and attempt to close the items prior to issuing the SER.

The potential SER open items in the Enclosure have the original request for additional information (RAI) number for reference. There are cases where an open item has been identified that is not associated with a previous RAI. For tracking purposes, new numbers have been assigned to these open items. In addition, the word "new" appears in parentheses after the number. Unlike the first round of RAIs, Duke is not expected to provide answers to all the open items in the Enclosure. Rather, to the extent that the staff and Duke resolve issues, the issue resolution will be documented in a phone call summary, meeting summary, or in a letter from Duke. If an issue is not resolved it will be carried forward in the SER as an open item.

Sincerely,
original signed by:
Joseph M. Sebrosky, Project Manager
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Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287
Enclosure: Potential SER Open Items
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Potential Safety Evaluation Report Open Items Regarding
the License Renewal Application for Oconee Units 1, 2, and 3

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.

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.

Revision to Potential Open Item 2.5.8-1(c)

In an April 8, 1999, letter the staff sent Duke the following question:

(c) With respect to the component level scoping of the HVAC systems concerning the heating coils and valves, Duke stated that (1) the heating coils are subject to an aging management review and are identified in

> Table 2.5-13 for the control room pressurization system and (2) valves for the Penetration Room Ventilation System are subject to an aging management review and are listed in Table 2.5-13. Provide clarification of why the above items are not listed in Table 2.5-13 of OLRP -1001 and provide justifications of why the valves are excluded from AMR for the Auxiliary Building Ventilation System and Pressurization and Filtration System.

Duke is requested to amend its response to this question. Specifically, the staff is withdrawing this question and replacing it with the following question:

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.

Revision to Potential Open Item 2.7-12

In an April 8, 1999, letter the staff sent Duke 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.

Duke is 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?

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?

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.

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 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 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 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.

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.

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.

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.

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.

Potential Open Item 4.3.7-4 (New)

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

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.

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.

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.

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.

(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.