November 20, 1998

Mr. William R. McCollum, Jr.
Vice President, Oconee Nuclear Site
Duke Energy Corporation
P. O. Box 1439
Seneca, SC 29679

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3, LICENSE RENEWAL APPLICATION

Dear Mr. McCollum:

By letter dated July 6, 1998, Duke Energy Corporation (Duke) submitted for the U.S. Nuclear Regulatory Commission's (NRC's) review an application pursuant to 10 CFR Part 54, to renew the operating licenses for the Oconee Nuclear Station (Oconee), Units 1, 2, and 3. Exhibit A to the application is the Oconee Nuclear Station License Renewal Technical Information Report (OLRP-1001), which contains the technical information required by 10 CFR Part 54. The NRC staff is reviewing the information contained in OLRP-1001 and has identified, in the enclosure, areas where additional information is needed to complete its review. Specifically, the enclosed questions are from the Materials and Chemical Engineering Branch regarding Sections 3.4.5, 4.10, 4.3.1, 4.24, and 5.4.2 of OLRP-1001.

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

Sincerely,

original signed by:
Joseph M. Sebrosky, Project Manager
License Renewal Project Directorate
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270,
and 50-287

Enclosure: Request for Additional Information

cc w/encl: See next page

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REQUEST FOR ADDITIONAL INFORMATION
OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3
LICENSE RENEWAL APPLICATION, EXHIBIT A

OLRP-1001 Section No.

3.4.5 Reactor Vessel

3.4.5-1 The following general issue with respect to plant aging needs to be addressed:

Based on its evaluation of operating experience, the NRC has determined that potential aging effect mechanisms in components of pressurized water reactor vessels are as indicated in the Table 3.1-3 of the Draft Standard Review Plan for License Renewal. Table 3.1-3 identifies components that are considered part of the reactor pressure vessel (RPV) and identifies the associated aging effects for the components. Identify the equivalent components in the Oconee reactor pressure vessels and identify the aging effects (identified as significant or unresolved in Table 3.1-3) applicable to these components and where they are addressed in the application. For those aging effects that are not addressed explain why they are not applicable.

Note: Questions 3.4.5-2 through 3.4.5-8 discuss how the Oconee license renewal application relates to BAW-2251. There are aspects of the questions that involve sections 3.4.5, 4.24, and 5.4.2 of Oconee's license renewal application. The questions have all been placed in this section for convenience.

3.4.5-2 The following are action items to be addressed by a plant-specific license renewal application when incorporating by reference the Babcock & Wilcox Owners Group (B&WOG) topical report, BAW-2251. Provide the following:

a) 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. Duke Energy, the applicant for license renewal will be responsible for verifying that any such commitments are subject to appropriate regulatory control. As such, identify any deviations from the aging management programs described in Topical Report BAW-2251. Evaluate any deviations on a plant-specific basis in accordance with 10 CFR 54.21(a)(3) and (c)(1).

b) B&WOG has determined that the lower control rod drive mechanism (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 BAW-2251. Identify which aging effects are applicable to these components and describe your aging management program for these components in the license renewal application.

Note: Additional plant-specific open Items that need to be addressed relative to the contents of the license renewal application and Topical Report BAW-2251 are discussed in question 3.4.5-3 through 3.4.5-8.

3.4.5-3 Intended Function of Reactor Vessel Components

Identify whether the intended function of the reactor vessel internals is to maintain the capability to shut down the reactor and maintain it in a safe-shutdown condition.

3.4.5-4 Flow Stabilizers Subject to Aging Management Review

The staff has concerns about whether the flow stabilizers should be excluded from an aging management review for license renewal. Although the flow stabilizers themselves do not have safety-related functions, they were installed to address flow-induced vibration (FIV) problems experienced during hot functional testing. Thus, cracking of the attachment weld may cause the reactor vessel shell to crack thereby affecting its intended functions. Indicate if an aging management program is provided to manage the aging effects on the flow stabilizers. If so, provide the details of such a program; if not justify why such a program is not needed to ensure the integrity of these stabilizers over the extended life for the units.

3.4.5-5 Wear of Core Guide Lugs

The staff considers loss of material due to mechanical wear of the core guide lugs a potential applicable aging effect that should be managed for license renewal. This potential aging effect is discussed in Section 3.1 of the working draft standard review plan for license renewal. Indicate if an aging management program is provided to manage the aging effects on the lugs. If so, provide the details of such a program; if not justify why such a program is not needed to ensure the integrity of the lugs over the extended life for the units.

3.4.5-6 Underclad Cracking

Cracking has been detected under the austenitic stainless steel weld cladding in reactor vessel forgings. When cracks are detected, the licensee performs a time-limited aging analysis (TLAA) to evaluate the integrity of the reactor vessel. However, the staff considers the potential for underclad cracks to grow during plant operation an applicable aging effect to be managed for license renewal. Indicate if an aging management program is provided to manage the aging effects on the stainless steel cladding in the forgings. If so, provide the details of such a program. If not, justify why such a program is not needed to ensure the integrity of reactor vessel forgings.

3.4.5-7 Reactor Vessel Materials Surveillance Program

To ensure that the results of fracture toughness tests remain valid during the extended license period, describe the operating limitations necessary for ensuring that each plants' operating conditions (temperature and neutron fluence) do not invalidate the results of fracture toughness tests conducted on surveillance capsules removed from the Oconee reactor pressure vessels during the original 40-year license periods for the plants.

3.4.5-8 Additional Limitations on Pressure-temperature (P-T) Limits and Reactor Coolant Pump Seal Limits

Based on the projected P-T limits at the end of the extended license period and other plant operating limits (e.g., limits of pump seal pressure), identify whether the operating windows for the Oconee units will be sufficient to start up and shut down the units at the end of the extended license period. If the operating windows are insufficient, provide aging management programs to increase the operating windows or reduce the amount of neutron embrittlement to the Oconee RPVs.

4.3.1 Alloy 600 Aging Management Program

Note: Question 4.3.1-1 and 4.10-1 were originally grouped together.

4.3.1-1 In regard the content of Section 4.3.1, "Alloy 600 Aging Management Program" (henceforth the Alloy 600 AMP) to the License Renewal Application:

a. The section states that the Alloy 600 AMP will be used to identify and inspect the four most susceptible locations within the Oconee reactor coolant systems (RCS). Clarify whether the scope of the proposed inspections of the four most susceptible locations will be on different components within the RCS or on redundant ("sister") components in the RCS.

b. Clarify whether the aging management program (Section 4.10 of the License Renewal Application) for the Oconee Alloy 600 vessel head penetration (VHP) nozzles and associated Alloy 82/182 partial penetration welds is a separate program from the Alloy 600 AMP and if it will be implemented in addition to the Alloy 600 AMP.

4.10 Control Rod Drive Mechanism Nozzle and Other Vessel Closure Penetrations Inspection Program

4.10-1 Regarding the content of Section 4.10, "Control Rod Drive Mechanism Nozzle and Other Vessel Closure Penetrations Inspection Program:"

In Section 4.10 to the License Renewal Application Duke indicated, in part, that the existing regulatory basis for the aging management program for Alloy 600 VHP nozzles is provided in the Duke response to Generic Letter 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations." In its response to GL 97-01, Duke indicated that it was a participant in the joint Babcock and Wilcox Owners Group (BWOG)/Nuclear Energy Institute (NEI) integrated program for assessing primary water stress corrosion cracking (PWSCC) in VHP nozzles to B&W designed VHP nozzles, and that this program was contained in BWOG Topical Report BAW-2301. On May 14, 1998, the NEI submitted an integrated "industry Histogram for Reactor Vessel Head Penetration" on behalf of PWR licensees participating in NEI's integrated assessment program for control rod drive mechanism (CRDM) penetration nozzles and other VHP nozzles in domestic PWR designs. The histogram ranked the CRDM penetration nozzles in "less than 5 year," "5 to 15 year," and "beyond 15 year" probabilities of failure categories. The CRDM penetration nozzles of Oconee have been designated as falling into the "less than 5 year" category and inspections of the Oconee Unit 2 CRDM penetration nozzles have been scheduled to be reinspected for a second time in the year 1999. However, the current integrated program and susceptibility assessment for the PWR industry is based on a 40-year (normal life) time frame. Provide the following information with respect to how the license renewal term for the Oconee units relates to the industry's integrated program for assessing domestic PWR VHPs:

a. Indicate whether Duke is committed to extending its participation in the BWOG integrated aging management program for VHP nozzles during the license renewal term for the Oconee units.

   i. If Duke is committed to extending its participation in the integrated program to the license renewal term, indicate how the integrated program will be used as the basis for proposing any further inspections of the VHP nozzles at Oconee Units 1, 2, and 3 during the extended license terms for the facilities.

   ii. If Duke is not committed to extending its participation in the integrated program to the license renewal term, describe what the basis (in addition to the inspections of the Oconee Unit 2 VHP nozzles in 1999) will be for assessing the potential for primary water stress corrosion cracking to exist in the Oconee VHP nozzles and for proposing any further inspections of the Oconee VHP nozzles during the extended license terms for the facilities.