ACRSR-1876

March 13, 2000

The Honorable Richard A. Meserve
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Dear Chairman Meserve:

SUBJECT: REPORT ON THE SAFETY ASPECTS OF THE LICENSE RENEWAL APPLICATION FOR THE OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3

During the 470th meeting of the Advisory Committee on Reactor Safeguards, March 1-4, 2000, we completed our review of Duke Energy Corporation's application for license renewal of the Oconee Nuclear Station, Units 1, 2 and 3 and the related Final Safety Evaluation Report (FSER). Our review included a plant visit and four meetings, one of which was conducted in Clemson, South Carolina. We had the benefit of insights gained from two meetings concerning generic license renewal issues and the review of another license renewal application. We provided an interim letter dated September 13, 1999, concerning the Oconee license renewal application. During these reviews, we had the benefit of the documents referenced.

Conclusion

On the basis of our review of Duke's application, the staff's FSER, and the resolution of the open and confirmatory items identified in the June 1999 Safety Evaluation Report (SER), we conclude that:

- Duke has properly identified the structures, systems, and components (SSCs) that are subject to aging management programs according to the requirements of 10 CFR Part 54.
- Possible aging mechanisms associated with passive, long-lived SSCs have been appropriately identified.
- The programs instituted to manage aging-related degradation of the identified SSCs are appropriate and provide reasonable assurance that Oconee Units 1, 2 and 3 can be operated in accordance with their current licensing basis for the period of the extended license without undue risk to the health and safety of the public.

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Background and Discussion

This report is intended to fulfill the requirement of 10 CFR 54.25 that each license renewal application be referred to the ACRS for a review and report. Duke requested renewal of the operating licenses for the Oconee Units 1, 2 and 3 for a period of 20 years beyond the current license term. The FSER documents the results of the staff's review of information submitted by Duke, including those commitments that were necessary to resolve open and confirmatory items identified by the staff in its SER. The staff's review included the verification of the completeness of the identification and categorization of the SSCs considered in the application; the validation of the integrated plant assessment process; the identification of the possible aging mechanisms associated with each passive long-lived component; and the adequacy of the aging management programs.

In the SER, the staff identified a number of open and confirmatory items. The staff and Duke have now resolved all open items and addressed all confirmatory items, in part through additional commitments made by Duke. The Duke commitments will become a part of the plant's licensing basis and will be added to the Oconee Final Safety Analysis Report (FSAR). This will make the commitments enforceable.

Several of the open items, such as the completeness of the methodology used to identify SSCs that are within the scope of Part 54 and the consideration of the effects of the reactor coolant environment on fatigue life, may have generic implications for future license renewal applications.

Because Oconee was licensed before NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," was issued in September 1975, the safety-related SSCs at Oconee do not completely bound the set of SSCs that are relied upon to be functional during and following design basis events. Consequently, nonsafety-related components that are relied upon to perform safety-related functions are within the scope of Part 54. As noted in our interim letter, this is a generic issue for older plants. The process of identifying these additional SSCs without expanding the current licensing basis of Oconee required significant interaction between the staff and the licensee.

In accordance with the license renewal scoping criteria specified in 10 CFR 54.4 (a), the staff identified a set of additional events that had not been considered in Duke's license renewal application. Although these events were not part of the original FSAR accident analysis, Duke was asked to perform a plant-specific evaluation. We agree with the staff determination that these events should be considered in the analysis of scope. Duke evaluated these events to identify additional SSCs that should be included within the scope of license renewal. This evaluation did not identify any additional SSCs and provides further evidence that SSCs within the scope of 10 CFR Part 54 have been appropriately identified.

Insulated cables in localized areas in the Oconee containment have been identified in station problem reports as exhibiting accelerated thermal and radiation-induced aging effects due to adverse environments. Where the design and installation conditions responsible for the accelerated aging have not been corrected, the staff requested that an aging management program be instituted as part of the license renewal application. The staff also requested that an aging management program be instituted for medium-voltage cables located in trenches or buried in the ground, where the cables are exposed to moisture.

Duke responded by instituting an Insulated Cables Aging Management Program that includes cables within the scope of license renewal that are installed in locations with adverse environments and could be subject to aging effects from radiation, heat, or moisture. The only insulated cables excluded from this program are those covered by the Environmental Qualification Program. The Insulated Cables Aging Management Program identifies inspections, parameters to be monitored, and corrective actions to be taken in accordance with the requirements of 10 CFR Part 50, Appendix B. We concur with the staff's conclusion that this comprehensive program resolves this open item.

A number of SER open items involved reactor vessel internal components. Aging effects to be addressed included changes in dimensions due to void swelling, cracking in reactor vessel internal noncast austenitic stainless steel components, cracking of baffle-former bolts, embrittlement of cast austenitic stainless steel components, thermal embrittlement of vent valves, and reduction in fracture toughness. Duke has addressed these open items in the Oconee Reactor Vessel Internals Aging Management Program (RVIAMP). This program includes participation in industry initiatives to investigate these aging effects, inspections, and reports to be provided to the NRC on a periodic basis. A final report will be submitted by Duke to the NRC near the end of the initial license period for Unit 1. The final report will contain the test results from the Babcock & Wilcox Owners Group's RVIAMP and the recommended inspection program for Oconee. On the basis of this information, Duke will implement an aging management program for the reactor vessel internals. We find the proposed program comprehensive and adequate for resolving the reactor vessel internals open items.

Duke committed to implementing a plant-specific fatigue monitoring program in which it will use correlations published in NUREG/CR-5704 to calculate environmental penalties at the high fatigue-usage locations identified in NUREG/CR-6260 to assess the effects of the reactor coolant environment on the fatigue life of components and piping. The correlations reflect data developed to resolve Generic Safety Issue (GSI) 190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life." We concur with the staff's conclusion that Duke's proposed program is an acceptable plant-specific approach for resolving GSI-190 concerns.

The Oconee license renewal application described the process and the results of a time-limited aging analysis to demonstrate the adequacy of prestressing forces in the containment post-tensioning tendons during the period of extended operation. The staff requested additional information needed to support this demonstration. Duke has responded by proposing a Post-Tensioning System Loss of Prestress Aging Management Program to identify and correct degradation of the post-tensioning system prior to an unacceptable loss of prestress. This program implements the requirements of the American Society of Mechanical Engineers (ASME) Code Section XI, Subsection IWL, for in-service inspection, trending, and repair or replacement activities of the post-tensioning systems of concrete containments. We concur with the staff's assessment that the implementation of this program adequately resolves this open item.

As Oconee Units 1, 2 and 3 age, inspection and operating experience may prompt significant adjustments to their aging management programs. Duke has committed to document in the

FSAR Supplement that all components subject to an aging management program fall under the requirements of its Problem Investigation Process corrective action program. Furthermore, the staff has required that Duke include in the Oconee FSAR the license renewal application commitments that the staff relied upon to conclude that aging effects will be adequately managed for the period of extended operation. These steps ensure that future changes to the aging management programs can be controlled under the 10 CFR 50.59 process.

The staff has performed a comprehensive and thorough review of Duke's application. The additional programs required by the staff are appropriate and sufficient. Current regulatory requirements and existing Duke programs provide adequate management of aging-induced degradation for those SSCs within the scope of the license renewal rule.

Mr. John D. Sieber did not participate in the Committee's deliberations regarding this matter.

Dr. William J. Shack did not participate in the Committee's deliberations regarding aging-induced degradation.

Sincerely,

/s/

Dana A. Powers
Chairman

References:

1. Letter dated February 3, 2000, from David B. Matthews, Office of Nuclear Reactor Regulation, to William R. McCollum, Jr., Duke Energy Corporation, Subject: Final Safety Evaluation Report.
2. ACRS letter dated September 13, 1999, from Dana A. Powers, Chairman, ACRS, to William D. Travers, Executive Director for Operations, NRC, Subject: Interim Letter Related to the License Renewal of Oconee Nuclear Station.
3. Letter dated June 16, 1999, from David B. Mathews, Office of Nuclear Reactor Regulation, to William R. McCollum, Jr., Duke Energy Corporation, Subject: Oconee Nuclear Station, Units 1, 2 and 3, License Renewal Safety Evaluation Report.
4. Letter dated April 26, 1999, from Christopher I. Grimes, Office of Nuclear Reactor Regulation, to David J. Firth, B&W Owners Group, Subject: Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled, "Demonstration of the Management of Aging Effects for the Reactor Vessel," BAW-2251, June 1996.
5. Letter dated June 27, 1996, from D. K. Croneberger, B&W Owners Group, to Document Control Desk, NRC, Subject: Submittal of BAW-2251, "Demonstration of the Management of Aging Effects for the Reactor Vessel," June 1996.
6. U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation Office Letter Transmittal No. 805, "License Renewal Application Review Process," June 19, 1998.
7. U. S. Nuclear Regulatory Commission Safety Evaluation Report (SER) related to the Babcock & Wilcox (BAW) Topical Report 2251, "Demonstration of the Management of Aging Effects for the Reactor Vessel," April 26, 1999.

8. U. S. Nuclear Regulatory Commission, NUREG/CR-5704, "Effects of LWR Coolant Environments on Fatigue Design Curves of Austenitic Stainless Steels," April 1999.

9. U. S. Nuclear Regulatory Commission, NUREG/CR-6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," March 1995.