February 28, 2002

Mr. William R. McCollum, Jr.
Vice President, Oconee Site
Duke Energy Corporation
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION, UNIT 3 ISSUANCE OF AMENDMENT RE: ONE-TIME SURVEILLANCE INTERVAL EXEMPTION OF CONTAINMENT INTEGRATED LEAK RATE TEST (TAC NO. MB1377)

Dear Mr. McCollum:

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 321, to Renewed Facility Operating License DPR-55 for the Oconee Nuclear Station, Unit 3. The amendment consists of changes to the Technical Specifications in response to your application dated March 5, 2001, as supplemented by letter dated September 4, 2001.

The TS change allows a one-time extension to the interval for conducting the 10 CFR Part 50, Appendix J containment integrated leak rate test.

A copy of the related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Leonard N. Olshan, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-287

Enclosures:
1. Amendment No. 321 to DPR-55
2. Safety Evaluation

cc w/encls: See next page

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DUKE ENERGY CORPORATION

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 321
Renewed License No. DPR-
55

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Renewed Facility Operating License No. DPR-55 filed by the Duke Energy Corporation (the licensee) dated March 5, 2001, as supplemented by letter dated September 4, 2001, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Renewed Facility Operating License No. DPR-55 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 321, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard J. Laufer, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Technical Specification Changes

Date of Issuance: February 28, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 321

RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
5.0-7	5.0-7

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 321 TO RENEWED FACILITY OPERATING LICENSE DPR-55

DUKE ENERGY CORPORATION

OCONEE NUCLEAR STATION, UNIT 3

DOCKET NO. 50-287

1.0 INTRODUCTION

By letter dated March 5, 2001, as supplemented by letter dated September 4, 2001, Duke Energy Corporation (the licensee) submitted a request for changes to the Oconee Nuclear Station, Unit 3, Technical Specifications. The requested change would allow a one-time extension to the interval for conducting the 10 CFR Part 50, Appendix J containment integrated leak rate test (ILRT).

2.0 BACKGROUND

Option B of Appendix J of 10 CFR Part 50 requires that a Type A test (containment ILRT) be conducted at a periodic interval based on historical performance of the overall containment system. Oconee Unit 3 is currently required to conduct the Type A test every 10 years. Since the last Type A test for Oconee Unit 3 was conducted on September 11, 1992, the 10-year interval for the next test ends on September 11, 2002. With the 15-month extension permitted by Nuclear Energy Institute (NEI) 94-01 and Regulatory Guide (RG) 1.163, the next Type A test may be deferred to December 11, 2003.

The steam generators (SG) for Oconee Unit 3 are scheduled to be replaced during the refueling outage currently scheduled to begin in October 2004. Following SG replacement, a Type A test is required. Since the October 2004, SG replacement is approximately 10 months past the end of the 15-month grace period of December 11, 2003, Oconee Unit 3 would be required to perform two consecutive Type A tests. Thus, the licensee has proposed this one-time extension, which extends the ILRT approximately 16 months to the outage when the SG replacement occurs.

3.0 EVALUATION

The licensee has performed a risk impact assessment of extending the Type A test interval to 151 months. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," and RG 1.174, "An Approach

For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The basis for the current 10-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," September 1995, provided the technical basis to support rulemaking to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement the NRC's rulemaking basis, NEI undertook a similar study. The results of that study are documented in EPRI TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The EPRI study estimated that relaxing the test frequency from 3 per 10 years to 1 per 10 years will increase the average time that a leak detectable only by a Type A test goes undetected from 18 to 60 months. Since Type A tests only detect about 3 percent of leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a 10 percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage, in percent of person-rem/year, for the pressurized water reactor representative plant was estimated to increase from .032 percent to .035 percent. This confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3 per 10 years to 1 per 10 years leads to an "imperceptible" increase in risk.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem/year frequency. The licensee quantified the leakage from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the Option B rulemaking in 1995, the staff has issued RG 1.174 on the use of probabilistic risk assessment in risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per reactor year and increases in large early release frequency (LERF) less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original 3 per a 10-year interval. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the change in the conditional containment failure probability as an element demonstrating that the defense-in-depth philosophy is met.

The licensee provided a sensitivity analysis which estimated all of these risk metrics and has a methodology that is consistent with previously approved submittals. The following conclusions can be drawn from the sensitivity analysis associated with extending the Type A test frequency:

1. A slight increase in risk is predicted when compared to that estimated from current requirements. Given the change from a 10-year test interval to a 15-year test interval, the increase in the total integrated plant risk is estimated to be 0.04 percent. The increase in the total integrated plant risk, given the change from a 3 per 10-year test interval to a 15-year test interval, was found to be 0.11 percent. This is reasonable when compared to the range of risk increase, 0.02 to 0.14 percent, estimated in NUREG-1493 when going from a 3 per 10 year test interval to a 10-year interval. NUREG-1493 concluded that a reduction in the frequency of tests from 3 per 10 years to 1 per 10 years leads to an “imperceptible” increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. RG 1.174 provides guidance for determining the risk impact of plant-specific changes to the licensing basis. RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in CDF less than 10^{-6} per reactor year and increases in LERF less than 10^{-7} per reactor year. Since the Type A test does not impact CDF, the relevant criterion is LERF. The increase in LERF resulting from a change in the Type A test interval from 1 per 10 years to 1 per 15 years is estimated to be 2.7×10^{-8}/year. The increase in LERF resulting from a change in the Type A test interval from the original 3 per 10 years to 1 per 15 years is estimated to be 8.2×10^{-8}/year. Increasing the Type A interval to 15 years is considered to be a very small change in LERF.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. The change in the conditional containment failure probability was estimated to increase by 0.001 for the proposed change and 0.0031 for the cumulative change of going from a test interval of 3 per 10 years to 1 per 15 years. The staff finds that the defense-in-depth philosophy is maintained based on the very small change in the conditional containment failure probability for the proposed change.

   The staff recognizes the limitations of a conditional containment failure probability approach. For plants, such as Oconee Unit 3, with core damage frequency estimates well below 10^{-4}, the ability of the containment to withstand events of even lower probability becomes less clear. Therefore, it is important to consider other risk metrics in conjunction with the conditional containment failure probability, such as total LERF. The licensee has sufficiently demonstrated that the total LERF is less than 10^{-5} for the purpose of this evaluation.

The licensee is using the 1992 Edition and the 1992 Agenda of Subsections IWE and IWL of Section XI of the ASME Boiler and Pressure Vessel Code (the Code) for conducting its inservice inspection of the Oconee Unit 3 containment with approved relief from certain Code requirements. The first containment ISI interval began on September 9, 1998, and will end on September 8, 2008. The licensee's letter dated September 4, 2001, provided responses to five questions from the staff. In responses to Questions 1 and 2, the licensee described its containment ISI, including areas of augmented inspections, and discussed how it assured that the containment structural and leak-tight integrity will be maintained until verification can be obtained during the next ILRT.

In response to Question 3 regarding the ISI of seals, gaskets and bolts, the licensee stated that at Oconee Unit 3, Type B and Type C tests are performed at the intervals required by Option A of Appendix J (i.e., the time interval between the Type B and C tests does not exceed 24 months). The licensee stated that this frequency of testing provides reasonable assurance that the integrity of the containment pressure boundary is maintained during the period of the ILRT extension.

Question 5 was related to the effects of degradations in uninspectable areas of the steel liner (i.e., areas that cannot be visually examined). Because, ILRTs help to identify major areas of through-wall degradations when the containment is pressurized, the staff questioned how the potential leakages due to age-related degradation are considered in the risk assessment of the extended ILRT. In response to Question 5, the licensee explained that the potential for containment leakage is explicitly included in the risk assessment. By definition, the intact containment cases (EPRI Containment Failure Class 1) include a leakage term, which is independent of the source of the leak.

Moreover, the licensee emphasizes that the Oconee probability risk assessment (PRA) is a full scope, level 3 PRA. Revision 2 of the PRA uses the containment capacity analysis developed for the Oconee Individual Plant Examination (IPE). Appendix G of the Oconee PRA report, Revision 1, is a detailed description of the containment capacity assessment. The analysis identified expected failure locations, which would result in a large leak area, and quantified the expected failure pressure for use in the IPE. The analysis was then utilized in the development of the IPE source terms as well as the PRA (Revision 2) source terms. The external consequences (e.g., dose) were also analyzed in Revision 2 of the Oconee PRA. The dose results were than used to estimate the impact of extending the Oconee ILRT interval.

The staff's review of Appendix G of the Oconee PRA found that the licensee calculated the mean containment capacity for one failure mode, i.e. excessive hoop strains as the post-tensioning tendons yield. The median accident pressure associated with this failure mode is estimated as 144 pounds per square inch-gage (psig). No other failures modes, effects of accident temperatures or the structural degradations were considered in the estimate. At the conditional containment failure probability of 10 percent (signifying a large leakage), the non-degraded containment (i.e., no corrosion on the liner) can withstand an internal pressure of 120 psig. It should be recognized that the Code allows liner corrosion up to 10 percent of the liner thickness with a requirement for monitoring the degradation during subsequent inspections. The ILRT pressure for Oconee Unit 3 is 59 psig, about half the pressure that might result in a large leakage. If sufficient time is allowed for corrosion on the uninspectable side of the liner plate to continue, the ILRT is likely to result in appreciable leakage.

Based on these responses, the staff finds that (1) the containment structural integrity is verified through periodic ISIs conducted as required by Subsections IWE and IWL of the

ASME Code, Section XI, (2) the potential for large leakage from the areas that cannot be examined by the ISI has been explicitly modeled in performing the risk assessment, and (3) the integrity of the penetrations and containment isolation valves are periodically verified through Type B and Type C tests as required by the Oconee Unit 3 TS. Moreover, the system pressure tests for containment pressure boundary (i.e. Appendix J tests, as applicable) are required to be performed following repair and replacement activities in accordance with Article IWE-5000 of Section XI of the AME Code. Serious degradation of the primary containment pressure boundary is required to be reported under 10 CFR 50.72 and 10 CFR 50.73.

Based on these conclusions, the staff finds that the increase in predicted risk due to the proposed change is within the acceptance criteria while maintaining the defense-in-depth philosophy of RG 1.174. Therefore, the amendment is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (66 FR 50466). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: M. Snodderly
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Date: February 28, 2002

Oconee Nuclear Station

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