UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 302 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 302 TO FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 302 TO FACILITY OPERATING LICENSE DPR-55

DUKE ENERGY CORPORATION

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS: 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By letter dated October 15, 1998, as supplemented December 17, 1998, and January 11 and January 21, 1999, Duke Energy Corporation (the licensee) submitted a request for changes to the Oconee Nuclear Station, Units 1, 2, and 3 (Oconee), Technical Specifications (TSs). The requested changes would revise the heatup, cooldown, and inservice test limitations for the reactor coolant system of each unit for a maximum of 26 effective full-power years (EFPY). The letter dated October 15, 1998, also included a request for an exemption from certain requirements of Title 10 of the Code of Federal Regulations (10 CFR) Part 50, Section 50.60 and Appendix G that would allow the use of the alternate methodology of the American Society of Mechanical Engineers (ASME) Code Case N-514 in determining the low temperature overpressure protection (LTOP) system setpoints. This was processed separately, but must be implemented in conjunction with these amendments.

The supplements dated December 17, 1998, and January 11 and 21, 1999, provided clarifying information that did not change the scope of the original Federal Register notice and the initial proposed no significant hazards consideration determination.

1.1 Description of Proposed TS Changes

The specific TS changes proposed by the licensee are as follows:

- Figure 3.4.3-1, RCS [Reactor Coolant System] Normal Operational Heatup Limitations, Unit 1, from 21 EFPY to 26 EFPY.

- Figure 3.4.3-2, RCS Normal Operational Cooldown Limitations, Unit 1, from 21 EFPY to 26 EFPY.

- Figure 3.4.3-3, RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations, Unit 1, from 21 EFPY to 26 EFPY.

- Figure 3.4.3-4, RCS Normal Operational Heatup Limitations, Unit 2, from 19 EFPY to 26 EFPY.

- Figure 3.4.3-5, RCS Normal Operational Cooldown Limitations, Unit 2, from 19 EFPY to 26 EFPY.

- Figure 3.4.3-6, RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations, Unit 2, from 19 EFPY to 26 EFPY.

- Figure 3.4.3-7, RCS Normal Operational Heatup Limitations, Unit 3, from 21 EFPY to 26 EFPY.

- Figure 3.4.3-8, RCS Normal Operational Cooldown Limitations, Unit 3, from 21 EFPY to 26 EFPY.

- Figure 3.4.3-9, RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations, Unit 3, from 21 EFPY to 26 EFPY.

- TS 3.4.12.a, PORV lift setpoint would be changed from ≤ 480 psig to ≤ 460 psig.

- Surveillance Requirement 3.4.12.6.a, RCS temperature specified would be changed from 220 °F to 325 °F.

- Add Surveillance Requirement 3.4.12.6.d and .e, verify that pressurizer heater bank 3 or 4 is deactivated every 12 hours when TS 3.4.12 is applicable.

- Related Bases page changes as appropriate.

2.0 <u>EVALUATION</u>

2.1 Pressure-Temperature Limit Curves

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the pressure-temperature (P-T) limits based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Revision 1 (Rev. 1); GL 92-01, Revision 1 (Rev. 1), Supplement 1; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2); and Standard Review Plan (SRP) Section 5.3.2. GL 88-11 advises the licensee that the staff would use RG 1.99, Rev. 2, to review P-T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation. GL 92-01, Rev. 1, requests that licensees submit their reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1, requests that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P-T limit curves, and as the basis for the staff's review of pressurized thermal shock assessments (10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code (ASME Code).

SRP 5.3.2 provides an acceptable method of calculating the P-T limits for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions and requires a safety factor of 1.5 during hydrostatic testing.

The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to one-fourth of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P-T limit curves are the ¼ thickness (¼T) and ¾ thickness (¾T) locations, which correspond to the depth of the maximum postulated flaw, if initiated and grown from the inside and outside surfaces of the RPV, respectively.

The Appendix G methodology of the ASME Code requires that licensees determine the adjusted reference temperature (ART or RT_{NDT}). The ART is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin term.

The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence and calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

For the Oconee Unit 1 reactor vessel, the licensee determined that the most limiting material at the ¼T location is the circumferential weld of the intermediate shell plate to the upper shell plate, which was fabricated using weld wire heat 71249. The licensee calculated an ART of 194 °F at the ¼T location for 26 EFPY. The neutron fluence used in the ART calculation was 4.3×10^{18} n/cm^2 at the ¼T location. The chemistry factor was 167.6 °F, which was determined using Table 1 of RG 1.99, Rev. 2. The initial RT_{NDT} for the limiting weld was +10 °F. The margin term used in calculating the ART for the limiting weld was 56 °F at the ¼T, as permitted by RG 1.99, Rev. 2.

For the Oconee Unit 1 reactor vessel, the licensee determined that the most limiting material at the ¾T location is the circumferential weld of the intermediate shell plate to the upper shell plate, which was fabricated using weld wire heat 299L44. The licensee calculated an ART of 178 °F at the ¾T location at 26 EFPY. The neutron fluence used in the ART calculation was 1.56×10^{18} n/cm^2 at the ¾T location. The chemistry factor was 220.6 °F, which was determined using Table 1 of RG 1.99, Rev. 2. The initial RT_{NDT} for the limiting weld was -5 °F. The margin term used in calculating the ART for the limiting weld was 68.5 °F at the ¾T location, as permitted by RG 1.99, Rev. 2.

For the Oconee Unit 2 reactor vessel, the licensee determined that the most limiting material at the ¼T and ¾T locations is the circumferential weld of the reactor vessel upper shell forging to the lower shell forging. This weld was fabricated using weld wire heat 299L44. The licensee calculated an ART of 234 °F at the ¼T location and 179.4 °F at the ¾T location at 26 EFPY. The neutron fluence used in the ART calculation was 4.42 X 10^{18} n/cm^2 at the ¼T location and 1.61 X 10^{18} n/cm^2 at the ¾T location. The chemistry factor was 220.6 °F, which was determined using Table 1 of RG 1.99, Rev. 2. The initial RT_{NDT} for the limiting weld was -5 °F. The margin term used in calculating the ART for the limiting weld was 68.5 °F at the ¼T and ¾T locations, as permitted by RG 1.99, Rev. 2.

For the Oconee Unit 3 reactor vessel, the licensee determined that the most limiting material at the ¼T and ¾T locations is the circumferential weld of the reactor vessel upper shell forging to the lower shell forging. This weld was fabricated using weld wire heat 72442. The licensee calculated an ART of 202.1 °F at the ¼T location and 156.3 °F at the ¾T location at 26 EFPY. The neutron fluence used in the ART calculation was 4.38 X 10^{18} n/cm^2 at the ¼T location and 1.59 X 10^{18} n/cm^2 at the ¾T location at 26 EFPY. The chemistry factor was 180 °F, which was determined using Table 1 of RG 1.99, Rev. 2. The initial RT_{NDT} for the limiting weld was -5 °F. The margin term used in calculating the ART for the limiting weld was 68.5 °F at the ¼T and ¾T locations, as permitted by RG 1.99, Rev. 2.

2.2 Staff Evaluation of the Proposed Changes to the Oconee Units 1, 2, and 3 P-T Limit Curves

As previously stated, the licensee submitted ART calculations and P-T limit curves, for Oconee Units 1, 2, and 3, valid for 26 EFPY. The staff independently verified the accuracy of the licensee's ART calculations. In addition, the staff independently generated P-T curves for normal operations and hydrostatic test pressures effective to 26 EFPY for each of the three Oconee units. The details of this evaluation are provided below.

The ART is determined using the chemistry values percent copper and percent nickel for each beltline material of Oconee Units 1, 2, and 3. The Reactor Vessel Integrity database contains chemistry values for each beltline material for all light water reactors in the U.S. The licensee provided updated chemistry data for the beltline materials of Oconee Units 1, 2, and 3 by letter dated June 25, 1998. It should be noted that the licensee and staff used the most recent updated chemistry data for the beltline materials in the Oconee Units 1, 2, and 3 P-T limit evaluations. The staff compared the chemistry data and the initial RT_{NDT} values of the licensee's submittal of October 15, 1998, to the data in the BAW-2325, Revision 1, Report. The staff found that the chemistry data in the licensee's submittal, for the beltline materials of Oconee Units 1, 2, and 3, were the same as those indicated in the BAW-2325, Revision 1, Report. The staff also found that the initial RT_{NDT} values used by the licensee (in submittal dated October 15, 1998) were either the same or more conservative than those values specified in the BAW-2325, Revision 1, Report.

The staff performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Rev. 2. Based on these calculations, the staff verified that the licensee's limiting materials, for the Oconee Units 1, 2, and 3 reactor vessels, are those mentioned in Section 2.1 of this Safety Evaluation (SE). It should be noted that the staff verified the credibility of the surveillance data for the limiting material (weld wire heat 299L44) of Oconee Unit 2. The staff's calculated ART values for the limiting material agreed with the licensee's calculated ART values at 26 EFPY for Oconee Units 1, 2, and 3. Substituting the ART values for the Oconee Units 1, 2, and 3 limiting welds into the equations in SRP 5.3.2, the staff verified that the proposed P-T limits satisfy the requirements in Paragraph IV.A.2 of Appendix G of 10 CFR Part 50. The staff independently generated P-T curves for normal operations and hydrostatic test pressures effective to 26 EFPY for each of the three Oconee units. In comparing the staff's generated curves to the licensee's generated curves, the staff determined that the P-T curves for Oconee Units 1, 2, and 3 meet the requirements of Appendix G of Section XI of the ASME Code. Therefore, the staff also determined that these curves meet the requirements of Appendix G of Section XI of the ASME Code.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the RPV based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the adjusted reference temperature of the material in those regions by at least 120 °F for normal operation and by 90 °F for hydrostatic pressure tests and leak tests. Based on the RT_{NDT} of 60 °F for the limiting flange and upper shell materials, the staff has determined that the proposed P-T limits satisfy the requirement for the closure flange region during normal operation and hydrostatic pressure test and leak test for Oconee Units 1, 2, and 3.

The staff concludes that the proposed P-T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality for Oconee Units 1, 2, and 3 satisfy the requirements in Appendix G to Section XI of the ASME Code and Appendix G of 10 CFR Part 50 for 26 EFPY. The proposed P-T limits also satisfy GL 88-11 because the method in RG 1.99, Rev. 2, was used to calculate the ART. Hence, the proposed P-T limits may be incorporated into the Oconee Units 1, 2, and 3 TSs.

2.3 Low Temperature Overpressure Protection System

The LTOP system mitigates overpressure transients at low temperatures so that the integrity of the reactor coolant pressure boundary is not compromised by violating the 10 CFR Part 50, Appendix G P/T limits. The LTOP system for Oconee, Units 1, 2, and 3 uses the pressurizer power-operated relief valve (PORV) to accomplish this function. The system is manually enabled by operators and uses a single lifting setpoint for the PORV. Since there is only one PORV in each unit, administrative controls are in place to assure that greater than 10 minutes are available for operator actions to manually mitigate an LTOP event. The design basis of the LTOP considers both mass-addition and heat-addition transients. The mass-addition analyses account for the injection from the makeup pumps with the makeup control valve failing open to a restricted fully open position. The heat-addition analyses accounts for an event scenario involving pressurizer heaters erroneously energized, loss of the decay heat removal system, and reactor coolant pump start-induced transients. With administrative controls in place, the results of the licensee's analyses show that more than 10 minutes are available for operator actions to mitigate these events assuming the PORV is inoperable. The proposed TSs with their associated Bases provided administrative restrictions in plant operation within the configuration assumed in the analysis for LTOP design.

The proposed LTOP enable temperature and the PORV actuation setpoint were established using methodology consistent with the guidelines contained in the NRC Branch Technical Position RSB 5-2 and ASME Code Case N-514.

The proposed limiting conditions for operation (LCO) in TS 3.4.12 requires that an LTOP system shall be operable with high pressure injection deactivated, the core flood tanks isolated, an operable PORV with a lift setpoint of less than or equal to 460 pounds per square inch gauge (psig); and administrative controls implemented with greater than or equal to 10 minutes available for operator action to mitigate an LTOP event. The detailed administrative controls necessary are specified in the Bases of TS 3.4.12 including the maximum RCS pressures in various temperature ranges, the maximum pressurizer levels in various plant operating conditions, and the maximum makeup flow rate when LTOP is needed.

The licensee has proposed modifications to TS 3.4.12 and associated Bases to reflect the proposed LTOP system and to restrict the plant operational configuration consistent with the design of the LTOP system. The licensee also proposed changes to TS 3.4.3 to incorporate the P-T limits effective up to 26 EFPY. The staff has reviewed the proposed changes regarding the design of the LTOP system and finds that they are acceptable. The staff evaluation of the LTOP setpoints is presented below.

Enable Temperature

The LTOP enable temperature is the temperature below which the LTOP system is required to be operable. The licensee proposed to establish an LTOP enable temperature methodology to: (1) account for instrument uncertainties associated with the instrumentation used to enable the LTOP system; and (2) implement the methodology provided in ASME Code Case N-514 using an enable RCS water temperature corresponding to a metal temperature of at least RT_{NDT} + 50 °F at the belt line location (¼T or ¾T). Therefore, the licensee proposed to calculate the enable temperature as RT_{NDT} + 50 °F plus the temperature difference between RCS and metal plus instrument uncertainties. Using the above equation, the calculated limiting enable temperature is 324.4 °F based on the ¼T location. The licensee proposed an enable temperature of 325 °F that includes an additional margin of 0.6 °F.

The staff finds that this proposed LTOP enable temperature is conservative with respect to the enable temperature allowed by ASME Code Case N-514 and, therefore, is acceptable.

LTOP Actuation Setpoint

The LTOP at Oconee is conservatively designed to mitigate overpressure transients at low temperatures to prevent violating 10 CFR Part 50, Appendix G P-T limits under normal heatup, and cooldown operating conditions. The LTOP actuation setpoint is the pressure at which the PORV will lift to limit the peak RCS pressure during a pressurization transient.

Using the methodology provided by ASME Code Case N-514, the maximum allowed RCS pressure, corresponding to the minimum RCS temperature of 60 °F, is 473 psig that bounds all other maximum allowable pressures over the LTOP temperature range. Considering an instrument uncertainty of 13 pounds per square inch, the licensee proposed a single PORV setpoint of 460 psig for entire temperature range.

The licensee has concluded that the transient pressure response during an LTOP event is relatively slow since a steam or nitrogen bubble is maintained in the pressurizer during low temperature operations. Also, the PORVs are fast opening valves with approximately 0.2 second stroke time. Therefore, the peak pressure during an LTOP transient will be limited to the PORV setpoint. The staff has evaluated the licensee's design of the LTOP system and finds the licensee's proposed PORV setpoint acceptable.

RCS Vent Size

In TS 3.4.12, it is stated that the LCO is applicable during Mode 3 when any RCS cold leg temperature is less than or equal to 325 °F, and during Modes 4, 5, and 6 when an RCS vent path capable of mitigating the most limiting LTOP event is not open. The licensee has specified a vent size with a minimum equivalent diameter of 1-3/32-inch. The basis of this sizing is that the 1-3/32 diameter is equal to the inner throat diameter of the PORVs. The staff finds that the basis for determining the vent size is conservative and the proposed vent size is acceptable.

The staff has reviewed the licensee's proposed TS 3.4.12 for the LTOP enable temperature and actuation setpoint. The staff also reviewed the licensee's assessment related to the proposed enable temperature of 325 °F and PORV actuation setpoint as discussed in Sections 2.1 and 2.2 of this SE. The licensee has considered instrument uncertainties in its setpoint calculation using ISA S67.04-1994. The staff finds that the licensee's analyses were performed in a manner consistent with the approved methodology and that the results of the analyses conservatively demonstrated that the 10 CFR Part 50, Appendix G P-T limits will be adequately protected with these setpoints and, therefore, the staff finds the proposed TS 3.4.12 with its associated Bases regarding LTOP acceptable.

2.4 Reactor Pressure Vessel Fluences

The licensee participates in the integrated surveillance program to satisfy the requirements of 10 CFR Part 50, Appendix H. In addition, in-cavity surveillance was used to lower the uncertainty of the available data. It was stated in the staff's evaluation of the integrated surveillance program that cavity dosimetry measurements are allowed for the integrated program participants. Cavity dosimetry results along with the available surveillance data formed the basis of the benchmark used for the calculational methodology, which was used in the evaluation of the 26 EFPY Oconee fluence. This methodology is the same as that described in a topical report which has been reviewed by the staff for applications related to the Babcock and Wilcox Owner's Group (B&WOG). The staff finds this methodology acceptable because it complies with the recommendations of Draft Regulatory Guide DG-1053, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," issued June 1996, which describes methods and assumptions that are acceptable to the NRC staff for determining the pressure vessel fluence. Thus, the staff finds the proposed fluence values acceptable.

2.5 Summary

The staff has reviewed the changes to the Oconee Nuclear Station Units 1, 2, and 3 TS proposed by the licensee in its submittal dated October 15, 1998, as supplemented by letters dated December 17, 1998, and January 11 and January 21, 1999, and found them to be acceptable based on the analysis previously described.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding (63 FR 66592, dated December 2, 1998). Accordingly, the amendments meet 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 amendments.

5.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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