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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 307 TO FACILITY OPERATING LICENSE DPR-38

AMENDMENT NO. 307 TO FACILITY OPERATING LICENSE DPR-47

AND AMENDMENT NO. 307 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 May 11, 1999, (Reference 1), as supplemented by letter dated July 13, 1999, Duke Energy Corporation (the licensee) submitted a request for changes to the Oconee Nuclear Station, Units 1, 2, and 3 Technical Specifications (TS). The requested changes would incorporate revisions to the pressure-temperature (P/T) limits; the heatup, cooldown, and inservice test (IST) limits for the reactor coolant system (RCS) to a maximum of 33 Effective Full Power Years (EFPY); the low temperature overpressure protection (LTOP) system setpoints; and operational requirements for the reactor coolant pumps (RCPs). The supplement dated July 13, 1999, provided clarifying information that did not change the scope of the May 11, 1999, application and the initial proposed no significant hazards consideration determination.

A proposed change would extend the current P/T curves (Figures 3.4.3-1 through 3.4.3-9) to 33 EFPY, which is beyond the current Oconee license limit of 26 EFPY. A proposed change to TS 3.4.12, Low Temperature Overpressure Protection, would increase the power operated relief valve (PORV) setpoint from 460 pounds per square inch gage (psig) to 535 psig. The LTOP pressures, temperatures, and setpoints developed in this application are the same for all three units.

Table 3.4.3-1, "Operational Requirements for Unit Heatup," and Table 3.4.3-2, "Operational Requirements for Unit Cooldown," would be revised to allow two RCP operation in a single loop, rather than the present limit of one pump per loop during heatup and cooldown evolutions. Operation of two pumps will reduce the required net positive suction head for each pump, thereby reducing pump impeller wear due to cavitation, which has resulted in excessive impeller wear in the past.

The proposed changes include a new fluence determination based on the topical report BAW-2241P and the use of the American Society of Mechanical Engineers (ASME) Code Cases N-514, N-626, and N-588. An exemption for the application of these Code Cases was processed separately and issued by letter dated July 29, 1999 (Reference 7).

Associated Bases changes were also submitted.

2.0 BACKGROUND

2.1 Pressure -Temperature Limit Curves

The NRC has established requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the 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 (Rev.) 1; GL 92-01, Rev. 1, Supplement 1; Regulatory Guide (RG) 1.99, Rev. 2; and Standard Review Plan (SRP) Section 5.3.2. GL 88-11 advised licensees 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 (USE) resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff for inclusion in the staff's Reactor Vessel Information Database (RVID) 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 (PTS) assessments (10 CFR Part 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 Code.

The licensee's P/T limit curves and LTOP analysis satisfy the requirements of 10 CFR Part 50.60(a) with the additional provisions allowed by the following NRC-approved ASME code cases.

1. ASME Code Case N-626 (Now designated as Code Case N-640):

   Revised P/T limits have been developed using the K_{Ic} fracture toughness curve of ASME Section XI, Appendix A instead of the K_{Ia} curve of Appendix G as authorized and explained in Reference 7.

2. ASME Code Case N-588:

   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 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. However, if the flaw is postulated to be in a circumferential weld, it is physically unrealistic for the postulated flaw to be 1.5 times the vessel thickness, which is much longer than the width of the reactor vessel girth weld. It is unlikely that an axial flaw will extend from the circumferential weld into an adjacent plate or forging. In addition, due to the orientation of weld beads in a circumferential weld, the most likely orientation of the flaw is circumferential. Thus, for a circumferential weld, the postulated flaw should have a circumferential orientation. The approval and use of this case is further discussed in Reference 7.

3. Code Case N-514:

   The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary (RCPB) in nuclear power plants. As part of these requirements, 10 CFR Part 50, Appendix G, requires that P/T limits be established for RPVs during normal operating and hydrostatic or leak rate testing conditions. Specifically, 10 CFR Part 50, Appendix G, states: "The appropriate requirements on...the pressure-temperature limits and minimum permissible temperature must be met for all conditions." PWR licensees have installed cold overpressure mitigation systems (COMS)/LTOP in order to protect the RCPBs from being operated outside of the boundaries established by the P/T limit curves and to provide pressure relief of the RCPBs during low temperature overpressurization events. The staff has determined that the 110 percent of the Appendix G stress limit provisions of Code Case N-514 and the use of the K_{1c} fracture toughness curve permitted by Code Case N-640 (N-626) may not be applied simultaneously. In this submittal, Code Case N–514 was used for the determination of the LTOP enable temperature rather than the stress level. The LTOP enable temperature has been accepted in a staff position and is thus acceptable for use in conjunction with Code Case N-626 (N-640). This applies to the LTOP determination only and does not affect the determination of the licensee's P/T limit curves discussed in this safety evaluation, but is noted for the sake of completeness, since the licensee's submittal references the code case. The LTOP limits and the approval and use of Code Case N-514 are discussed in detail in Reference 7.

SRP 5.3.2 provides an acceptable method of determining the P/T Limits for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics (LEFM) methodology of Appendix G to Section XI of the ASME Code. 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 ASME Code Appendix G methodology requires that licensees determine the adjusted reference temperature (ART or adjusted 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 (M) 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 (CF) 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 the initial RT_{NDT}, the copper and nickel contents, the fluence, and the calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

2.2 LTOP Changes

The LTOP system is designed to protect the pressure vessel boundary from low temperature over-pressurization by designating P/T limits that satisfy the requirements of the ASME Code, Section XI, Division 1, Code Case N-514. Code case N-514 specifies that "LTOP systems shall be effective at coolant temperatures less than 200 °F or at coolant temperatures corresponding to a reactor vessel metal temperature less than RT_{NDT} + 50 °F, whichever is greater. LTOP systems shall limit the maximum pressure in the vessel to 110 percent of the pressure determined to satisfy Appendix G, paragraph G-2215 of Section XI, Division 1." Code Case N-514 has been approved for many plants including the Oconee Units. Code Case N-588 provides procedures for determining P/T limits derived from postulating a circumferential weld flaw rather than an axial flaw in the computation of the circumferential welds. In addition, a new computational procedure is incorporated. Code Case N-626 provides an alternate method for the computation of the fracture toughness of reactor vessel materials in determining the P/T limits. Code Case N-626 has been approved for use by ASME Section XI on September 1998. (As has been noted above, it has been renumbered and is now referred to as Code Case N-640).

2.3 Regulatory Requirements

For the protection of the RCS boundary, General Design Criteria (GDC) 14 and 31 are applicable. GDC 14 requires that the RCS boundary be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture. GDC 31 requires that sufficient margin be provided to assure that the reactor coolant pressure boundary behaves in a non-brittle manner under the stresses of normal operation, maintenance, test, and accident conditions, with a very low probability of rapidly propagating fracture.

Section 50.60 and Section 50.61 requires that licensees demonstrate that the effects of progressive embrittlement by neutron irradiation do not compromise the integrity of the reactor pressure vessel. To this end, two analyses are required: one to determine the P/T limits for normal heatup, criticality, cooldown, and inservice test operations; and another to assess the ability of the reactor vessel to maintain its integrity during an emergency shutdown with cold water injection (i.e., pressurized thermal shock (PTS)). 10 CFR 50.60 invokes Appendices G and H to 10 CFR Part 50, while 10 CFR 50.61 is the PTS rule, which requires a PTS assessment. PTS is not addressed in this evaluation, but has been reviewed by the staff.

Appendix G to 10 CFR Part 50 specifies fracture toughness requirements for ferritic materials within the reactor coolant boundary. It requires that the P/T limits for the RCS be at least as conservative as those obtained by the methodology specified in the 1989 edition of Appendix G to Section XI of the ASME Code. Alternatives to Appendix G may be used via an exemption, granted by the NRC. In this submittal, Code Cases N-514, N-626 and N-588 are used. Appendix H to 10 CFR Part 50 requires a reactor vessel materials surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region. These changes result from exposure of these materials to neutron irradiation and changes of the thermal environment. Material specimens exposed in the surveillance capsules are removed and tested at specified time intervals to monitor changes in the fracture toughness of the material.

2.4 RCP Operating Combination Change

During relatively low temperature RCS operation, the present TS require that no more than one RCP be operated per loop (the Oconee RCS design includes two RCS loops with two RCPs per loop). RCP operation at low pressure with either one pump in one RCS loop, or with one RCP in each RCS loop, results in gradual RCP impeller wear from cavitation. The degraded net positive suction head (NPSH) conditions are caused by the restricted P/T operating envelope at low pressures and temperatures.

3.0 <u>EVALUATION</u>

3.1 Heatup, Criticality, Cooldown and Inservice Test Limits P/T Curves - Licensee Evaluation

<u>Oconee, Unit 1</u>

According to the licensee, the projected 33 EFPY ART values at the 1/4T and 3/4T locations for the beltline regions were calculated by the licensee in accordance with RG 1.99, Rev. 2, and the guidelines presented by the NRC in the November 12, 1997, briefing concerning the review of responses to GL 92-01, Rev. 1, Supplement 1. The RG credibility criteria were applied to determine the appropriate margin, M, term. The licensee calculations determined the ART using RG 1.99, Rev. 2, Regulatory Positions 1.1 and 2.1. The licensee stated that the selected controlling values were those RV locations with the highest ART for 1/4T and 3/4T locations. The ART can be determined using RG 1.99, Rev. 2, Regulatory Position 1.1 by calculations using the Tables 1 and 2 values for CF or using Position 2.1 by calculations using surveillance data for the CF.

According to the tables titled, "Data for Preparation of Pressure-Temperature Limit Curves" in the submittal for Oconee, the licensee determined that the highest ART for the Unit 1 reactor vessel at the ¼T location is the circumferential weld (SA-1229) of the intermediate shell plate to the upper shell plate which was fabricated using weld wire heat 71249. The licensee calculated an ART of 203.1 °F, based on a neutron fluence of 5.22×10^{18} n/cm^2. The chemistry factor was 167.6 °F, which was determined using Table 1 (RG 1.99, Rev. 2, Position 1.1). The initial RT_{NDT} for the controlling weld material (SA-1229) was +10 °F. The margin term used in calculating the ART for the limiting weld was 56 °F.

The licensee's P/T data table indicated that for the ¾T location in Unit 1, the controlling ART is the circumferential weld of the intermediate shell plate to the upper shell plate (WF-25) which was fabricated using weld wire heat 299L44. The licensee calculated an ART of 188.0 °F at the ¾T location at 33 EFPY, and the chemistry factor used by the licensee was 223.7 °F, both of which were determined using Position 2.1 of RG 1.99, Rev. 2. The neutron fluence used in the ART calculation was 1.90×10^{18} n/cm^2. The licensee's initial RT_{NDT} for the limiting weld was the B&W generic value of -5 °F. The margin term used in calculating the ART for the limiting weld was 68.5 °F, consistent with RVID and RG 1.99, Rev. 2.

Oconee, Unit 2

For the 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 (WF-25) to the lower shell forging. This weld was fabricated using weld wire heat 299L44. The licensee calculated an ART of 248.4 °F at the ¼T location and 189.6 °F at the ¾T location at 33 EFPY. The neutron fluence used in the ART calculation was 5.38 X 10^{18} n/cm^2 at the ¼T location and 1.95 X 10^{18} n/cm^2 at the ¾T location. The chemistry factor used by the licensee was 223.7 °F, which was determined using Position 2.1 of RG 1.99, Rev. 2. The initial RT_{NDT} for the limiting weld used was the B&W generic value of -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 calculated using RG 1.99, Rev. 2, Position 2.1, consistent with the RVID.

Oconee, Unit 3

For the 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 (WF-67). This weld was fabricated using weld wire heat 72442. The licensee calculated an ART of 211.7 °F at the ¼T location and 164.5 °F at the ¾T location at 33 EFPY. The neutron fluence used in the ART calculation was 5.32 X 10^{18} n/cm^2 at the ¼T location and 1.93 X 10^{18} n/cm^2 at the ¾T location at 33 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 taken as the B&W generic value of -5 °F. The margin term used in calculating the ART for the limiting weld was 68.5 °F at the ¼T and ¾T locations, in accordance with RG 1.99, Rev. 2. All these values are consistent with RVID data.

3.2 Heatup, Criticality, Cooldown and Inservice Test Limits P/T Curves - Staff Evaluation

As stated above, the licensee submitted ART calculations and P/T limit curves, for Oconee, Units 1, 2, and 3, valid for 33 EFPY. The staff independently calculated the ARTs using the staff-reviewed and approved data and calculations found in the publicly available NRC data base, RVID. In addition, the staff independently generated P/T curves for normal operations and inservice hydrostatic testing conditions effective to 33 EFPY for each of the three Oconee units. Although the staff's calculations using the NRC-approved data and methodology differed in some instances from the licensee's, the licensee's curves were found to be conservative with respect to the staff determinations and are, therefore, acceptable. The details of this evaluation are provided below.

The ART is determined using the chemistry values of percent copper and percent nickel for each beltline material of Oconee, Units 1, 2, and 3. RVID contains chemistry values for each beltline material for all light water reactors in the U.S. The licensee's and the vendor's data were verified by the staff before incorporation in the RVID data base. Chemical composition, fluence, and initial RT_{NDT} values in RVID were updated to the data provided for the beltline materials of Oconee, Units 1, 2, and 3, in the letter dated February 2, 1999, from the B&WOG to the NRC that submitted report BAW-2325, Rev. 1, dated January 1999. It should be noted that the 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 in the licensee's submittal and found that the chemistry data in the licensee's May 11,1999, 33 EFPY, P/T submittal for the beltline materials of Oconee, Units 1, 2, and 3, were the same as those indicated in the BAW-2325, Rev. 1, report. The staff also found that the May 11, 1999, calculations proposed by the licensee in their submittal were at least as conservative as those values derived by the staff.

The NRC-verified data for the chemical compositions, initial RT_{NDT}, and margin values are available in RVID on the NRC INTERNET site (http:/www.nrc.gov) or by request from the NRC. The appropriate methodology and equations are in SRP 3.5.2, RG 1.99, Rev. 2, (also available from NRC). The code cases have the requisite instructions, equations, and curves to perform the calculations and are available from ASME. Therefore, the numerical values used by the staff in its calculations will not be repeated here. The fluences used were those reviewed and verified by the NRC (Reference 8) and also stated above in Section 3.1 of this evaluation.

Oconee, Unit 1, unlike Units 2 and 3, is not fabricated from ring forgings but has two longitudinal welds joining the beltline forgings. Therefore, the staff calculated the P/T curves based on longitudinal flaws in the longitudinal welds after checking to be sure the circumferential weld flaws would not be controlling. It was determined that the licensee's P/T curves were conservative, for both longitudinal (axial) and circumferential weld flaws with respect to the staff's calculations.

With respect to the ART calculations used for weld wire heat 299L44 for Units 1 and 2 (but not Unit 3, which does not contain this heat) the staff has determined that the surveillance data does not meet the staff's credibility criteria and was, therefore, not used in the RVID or in the staff's P/T calculations. Nevertheless, the licensee did use surveillance data and the methodology of RG 1.99, Rev. 2, Position 2.1, in its calculations for Units 1 and 2. The staff's calculated ART values for weld wire heat 299L44 used the CF values in Table 1 of RG 1.99, Rev. 2, Position 1.1 for Units 1 and 2. The licensee's calculated ARTs were as, or more, conservative than the staff's calculations and were, therefore, acceptable. For Unit 3, the licensee and staff used RG 1.99, Rev.1, Position 1.1 (Tables) for its calculations. Since both the staff's and licensee's ARTs were in agreement, the licensee's Unit 3 ART and curves are acceptable.

In summary, the staff evaluated each of the licensee's P/T limit curves for acceptability by performing check calculations using the methodology referenced in the Code (as indicated by SRP 5.2.3) and verified that the licensee's proposed P/T limits satisfy the requirements in Paragraph IV.2.b. of 10 CFR Part 50, Appendix G. The staff independently generated P/T curves for normal operations and hydrostatic test pressures effective to 33 EFPY for each of the three Oconee units. In comparing the curves generated by the staff to those generated by the licensee, the staff determined that the licensee's proposed P/T curves for Units 1, 2, and 3, meet the requirements of Appendix G of Section XI of the ASME Code as modified by the referenced code cases. Therefore, the licensee's curves meet the requirements of 10 CFR 50.60 and Appendix G.

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. Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange region that is 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 as stated in RVID and also confirmed by the licensee, 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.

3.3 LTOP Changes

In the Oconee plants, over-pressure mitigation is accomplished using a combination of a pressurizer PORV and steam volume in the pressurizer (by limiting the pressurizer water level) and/or a RCS vent to depressurize the reactor. The system is manually enabled by the operator and uses a single setpoint as the lift pressure for the PORV. The design basis for the Oconee LTOP system considers the Adjusted-RT_{NDT} by estimating the pressure vessel fluence at the end of 33 EFPY. This allows the determination of the material properties, which in turn determines the pressure vs temperature behavior of the material. The maximum pressure is determined from a number of assumed transients, including mass and heat addition. The result of the transient analyses indicate that the mass addition transient is limiting, while the heat addition is self limiting below the P/T limits.

The staff has previously determined that the 110 percent of the Appendix G stress limit provision of Code Case N-514 and the K_{1c} feature of Code Case N-626 cannot be applied simultaneously. However, in the Oconee submittal Code Case N-514 is used for the determination of the enable temperature, rather than the stress level. This use has been accepted by the staff to determine the enable temperature and is, therefore, acceptable for use in conjunction with Code Case N-626.

3.4 Pressure Vessel Fluence

The Oconee units are part of a utility group with Babcock and Wilcox (B&W) designed and fabricated reactor vessels. This group applied for and was granted an exemption from the provisions of Appendix H to 10 CFR Part 50 to use an integrated surveillance program. This program was documented in the topical report BAW-1543A (Reference 3) and, among others, provided for a reactor cavity surveillance program to replace the in-vessel surveillance capsules that were relocated to two host plants (Crystal River and Davis Besse). This program was described in the topical report BAW-1875A (Reference 4) and received staff approval in June 1986. The dosimetry information collected from the integrated surveillance program (along with in-vessel dosimetry) was utilized in the validation of the methodology described in BAW-2241P, which has been reviewed by the staff and approved for use at Oconee. Issuance of the approved version of BAW-2214P is pending.

The projected 33 EFPY Adjusted-RT_{NDT} at the 1/4 T and 3/4T locations for the beltline regions were calculated. The Regulatory Positions 1.1 and 2.1 of Regulatory Guide 1.99, Rev. 2, were observed. From all of the beltline materials, the highest values were selected as the controlling material at each location. The fluence estimates in this submittal (to 33 EFPY) were calculated using the approved methodology in BAW-2241P and Regulatory Guide 1.99, Rev. 2 and, therefore, are acceptable.

3.5 Pressure -Temperature Limits

According to the licensee, the proposed P/T limits were developed using the computer code PTPC-3.3 (Reference 5) as modified for the application of Code Case N-518 for circumferential flaws in welds and by Code Case N-626 for use of the K_{1c} fracture toughness curve. The criteria employed to establish operating pressure and temperature limits are described in the staff approved Topical Report, BAW-10046A (Reference 6). The method used in determining the P/T limits includes the beltline region, the closure head, and the nozzle region for normal heat-up, cool-down, and in-service leak and hydrostatic tests.

Justification for the use of the above code cases and granting of the exemptions were approved by letter dated July 29, 1999 (Reference 7).

The design basis events are the following:

- Erroneous actuation of the high pressure injection (HPI) system. The Oconee TS currently require that both trains of the HPI be deactivated during LTOP. Analysis of this event was not performed because it is not considered credible.

- Erroneous opening of the core flood tank discharge valve. The current TS require that the core flood tanks be deactivated during LTOP conditions. Therefore, analysis of this event was not performed because it is not considered credible.

- Erroneous addition of nitrogen to the pressurizer. The high pressure nitrogen system is administratively controlled. Therefore, analysis of this event was not performed because is not considered credible.

- Makeup control valve failing full open. The maximum makeup flow is limited in this event to ensure that 10 minutes are available for operator action. The analysis distinguishes three regions with respect to pressure and temperature: (1) $T < 220$ °F and $P < 100$ psig, (2) $T < 220$ °F and 100 psig $< P < 375$ psig and (3) 220 °F $< T < 325$ °F. And $P < 525$ psig. In all three regions the PORV is assumed to be inoperable. With appropriate initial conditions for each region, the pressurizer level is determined so as to assure a 10-minute time window for operator mitigative action.

- Pressurizer heaters erroneously energized. The acceptance criterion is that 10 minutes be available for operator action before the pressure reaches 535 psig. The pressurizer PORV is assumed to be inoperable. Steam or a nitrogen bubble is assumed in the pressurizer. The pressurizer level is 80 inches, which becomes 100 inches assuming a 20-inch measurement uncertainty.

- Loss of decay heat removal system. Three cases are analyzed. The first case assumes a rapid cooldown and end-of-cycle decay heat, followed by failure of the decay heat removal system. The second case assumes that a pressurizer cooldown is in progress with pressure at or below 100 psig, pressurizer level at or below of 310 inches, and a HPI pump in operation. With these initial conditions, loss of the heat decay system is assumed to assure that 10 minutes are available for operator mitigative action. Finally, the third case evaluates a scenario where RCS fill/drain activities are under way with pressure at or below

100 psig pressurizer level at or below 380 inches. The analysis aims to verify that 10 minutes is available for mitigative operator action.

- RCP start induced transient. Two types of RCP-induced transients are evaluated. The first is filling of the once-through steam generator (OTSG) on the secondary side with hot water, followed by starting of the RCPs. The second transient is the restart of an RCP during heatup following a period of stagnant conditions. The results of the first transient indicate that the peak pressure is 505 psig, which is below the allowable reactor vessel pressure. For the second transient, the initial P = 450 psig and T = 275 degrees F are assumed. The resulting peak pressure is 600 psig, but the limiting pressure (for the assumed temperature) is 1050 psig.

The results of the above analysis indicate that there is a minimum of 10 minutes for operator action or that the maximum pressure does not reach the allowable limits.

Because the licensee is using Code Case N-626 in conjunction with ASME XI Appendix G, the P/T limits are based on 100 percent of the steady state (Appendix G) limits. The enable temperature is the greater of RT_{NDT} +50 °F or 200 °F. Unit 2 is the most limiting (has the highest enable temperature) and bounds Units 1 and 3. The most limiting Adjusted-RT_{NDT} is 248.4 °F. With the additional allowance of 50 °F plus 11.6 °F for instrument error plus 15 °F for margin, the LTOP enable temperature becomes 325 °F. (The corresponding values are 279.7 °F for Unit 1 and 288.3 °F for Unit 3). The 15 °F margin was conservatively added because the above limits were based on 100 percent of the Appendix G limits; therefore, there is no allowance for thermal gradient through the vessel thickness (although none is required). Because a steam or a nitrogen bubble is in the pressurizer whenever the vessel is pressurized, an LTOP transient is slowly changing temperatures and pressures, justifying the 100 percent limit.

Differential pressure corrections were applied to the P/T limits to account for the pressure differential between the analyzed regions and the system pressure sensor locations in the reactor vessel. These corrections are based on the RCP constrains as follows:

- Coolant Temperature T < 250 °F two pumps in one loop or one pump on both loops

- Coolant Temperature T > 250 °F two pumps on both loops.

The RCS pump operational constraints are considered in conjunction with the following linear heatup and cooldown rates:

<u>Heatup</u>

Between 60 °F and 280 °F, at 50 °F/hr

Between 280 °F and 570 °F, at 100 °F/hr

Cooldown

Between 570 °F and 280 °F at 50 °F steps with 30 minute hold periods or equivalent.

Between 280 °F and 150 °F at 25 °F steps with 30 minute hold periods or equivalent.

At 240 °F: Starting of the decay heat removal system is modeled as a step change from 240 °F to 207 °F and held for one minute at 207 °F, followed by a step increase to 227 °F. It is assumed that two RCPs in one loop are operating.

Between 150 °F and 60 °F at 10 °F steps with 60 minute hold periods or equivalent.

The maximum allowable pressure is taken to be the lowest of the calculated allowable pressures under transient and steady-state conditions. The collection (loci) of these points form the P/T limits. Pressure and temperature instrument errors are added in the operating procedures. Instrument uncertainty is calculated based on a licensee directive that complies with the intent of (the Instrument Society of America) ISA-67.04, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." The LTOP operating limits included in TS 3.4.12 and TS Bases 3.4.12 include an allowance for instrument error.

The pressurizer PORV (single) setpoint is set a 535 psig. This maximum allowable pressure bounds all other pressures over the LTOP temperature range. It includes a 20 psig difference from the 555 psig calculated for the lowest temperature of 60 °F. (This difference accounts for 12.9 psig instrument error and 7.1 psig margin.)

The PORV is a fast opening valve with approximately 0.2 second stroke time (after a 2.1 second time delay) and the transient pressure overshoot is negligible during LTOP events. Therefore, no special allowance is provided.

3.6 Administrative Controls

In the proceeding analysis some LTOP initiators were assumed as not credible; for others, a 10 minute time interval is available for the operator to take the appropriate mitigative action. To ensure that these assumptions are correct the following administrative controls were proposed by the licensee:

A. RCS Pressure:

P < 375 psig when T < 220 °F
P < 525 psig when 220 °F < T < 325 °F

B. Pressurizer Level:

P > 100 psig and T < 325 °F, then L < 220 inches
P < 100 psig and T < 220 °F, then L < 310 inches
P < 100 psig and T < 160 °F, then L < 380 inches while filling or draining the RCS and the HPI pumps running.

C. Make-up flow is restricted with the make-up control valve (HP-120) to < 98 gallons per minute (gpm) for all three units.

D. The high pressure nitrogen system is administratively controlled to prevent inadvertent pressurization of the RCS.

E. Three audible pressurizer level alarms are set at 225, 260, and 315 inches.

F. Two audible RCS pressure alarms are set at 375 and 525 psig.

G. The core flood tanks must be deactivated.

H. The HPI safety injection flowpaths must be deactivated.

I. The pressurizer heater banks 3 and 4 must be deactivated.

The above limitations correspond to the analysis assumptions.

3.7 RCP Operating Combination Change

The limits on allowable operating RCP combinations control the pressure differential between the reactor vessel wall and the pressure measurement point and are used as inputs for calculating the heatup, cooldown, and the leak rate and hydro test limit curves. For example, with one RCP operating in a loop, the pressure differential between the low range pressure transmitter tap and the actual pressure at the vessel beltline is approximately 20 psi. With two RCPs in the same loop operating this differential pressure is approximately 50 psi. The differential pressure created by the operation of two RCPs must be accounted for in the development of the P/T limit curves so the upper pressure limit is not allowed to be exceeded.

Limits on the number of allowable operating RCP(s) at low temperature became necessary as the pressure limits at low temperatures decreased. The pressure limits deceased due to both the effects of ongoing neutron exposure to reactor vessel materials and the conservative methodology then needed to assure that the P/T limit curves provided adequate protection from reactor vessel brittle fracture. As the pressure limits decreased, the higher pressure differential of the two operating RCPs in a loop resulted in an ever shrinking operating P/T window. The number of operating RCPs is currently limited to increase the size of the operating P/T window.

As a result of the change in the P/T limits described in this submittal and safety evaluation, the licensee has proposed increased pressure limits at low temperatures. The licensee has determined that this increase in the pressure limit restores sufficient pressure margin to accommodate operation of two RCPs in a loop or one RCP in each loop at low temperatures as shown in the proposed changes to TS Tables 3.4.3-1 and 3.4.3-2. The staff has reviewed this information and found it to be acceptable.

4.0 SUMMARY

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 33 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, TS as proposed by the licensee.

The staff has reviewed the proposed LTOP changes to TS 3.4.3 and the associated bases and determined that the proposed revisions satisfy the Appendix G (to 10 CFR Part 50) requirements as modified by the ASME Code Cases N-514, N-588 and N-626, for which exemption requests have been approved. The proposed modification extends the period of the LTOP applicability to 33 EFPY. These changes are acceptable.

The LTOP enable temperature and the P/T curves, are the same for all three units and are based on the Unit 2 circumferential weld WE-25, which is the critical weld for each of the units. We find that the estimation of the LTOP enable temperature, the PORV actuation pressure, the P/T curves, and the associated pressurizer level were performed in a manner consistent with the approved methodologies. Therefore, the results are acceptable, and we conclude that the proposed modification of TS 3.4.3 and the associated bases are acceptable.

Based on the acceptability of these changes, the staff has found the changes to the number of operating RCPs per loop to be acceptable.

5.0 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.

6.0 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. 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 (64 FR 32289). 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.

7.0 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.

8.0 REFERENCES

1. Letter from W. R. McCollum, Jr., Duke Energy Corporation to USNRC "Proposed Revision to Technical Specifications Pressure-Temperature Operating Curves Technical Specification Change No. 99-02," dated May 11, 1999.

2. BAW-2241P, "Fluence and Uncertainties Methodologies," B&W Owners Group, dated May 14, 1997.

3. BAW-1543A, Rev. 2, "Integrated Reactor Vessel Material Surveillance Program," A.L. Lowe, Jr., et al, B&W Nuclear Division, May 1985.

4. BAW-1875A, "The B&WOG Cavity Dosimetry Program," S. Q. King, B&W Nuclear Division, August 1985.

5. FTI Document 32-1171775-05, "Verification of PTPC & User's Manual," by J. W. Moore the III, March, 1994.

6. BAW-10046A, Rev. 2, "Methods of Copliance with Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G," by H.W. Behnke, et. al. B&W Nuclear Technologies, Lynchburg, VA, June, 1976.

7. Letter from D. E. LaBarge, NRC, to W. R. McCollum, Jr., Duke Energy Corporation, "Specific Exemptions to Section 50.12, Part 50 of Title 10, Code of Federal Regulations Concerning ASME Code Cases N-588 and N-626," dated July 29,1999.

8. Memorandum to D. E. LaBarge, Project Manager, NRC, from Lambros Lois, Sr. Nuclear Engineer, Reactor Systems Branch, NRC, Subject: Oconee Units 1, 2, 3 License Amendment to Change the Heatup, Cooldown and LTOP Limits - Code Case N-514, "Low Temperature Overpressure Protection," Section XI of the ASME Boiler and Pressure Vessel Code, dated July 1, 1999.

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