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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 19 TO FACILITY OPERATING

LICENSE NO. NPF-63

CAROLINA POWER & LIGHT COMPANY, et al.

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

## 1.0 INTRODUCTION

By letter dated June 30, 1989, as supplemented November 27, 1989, February 1, 1990, and April 20, 1990, in response to Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Effect on Plant Operations," the Carolina Power & Light Company (the licensee) requested revisions to the pressure/temperature (P/T) limits in the Shearon Harris Nuclear Power Plant, Unit 1, (Harris) Technical Specifications (TS), Section 3.4. This revision would also change the effectiveness of the P/T limits from 4 to 3 effective full power years (EFPY). The proposed P/T limits were developed based on Section 1 of Regulatory Guide (RG) 1.99, Revision 2. The proposed revision provides up-to-date P/T limits for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest. The November 27, 1989, February 1, 1990, and April 20, 1990, letters provided clarifying information that did not change the proposed determination of no significant hazards consideration as published in the Federal Register (54 FR 40924) dated October 4, 1989.

To evaluate the P/T limits, the staff used the following NRC regulations and guidance: Appendices G and H to 10 CFR Part 50; the American Society of Testing Materials (ASTM) Standards and the American Society of Mechanical Engineers (ASME) Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Revision 2; Standard Review Plan (SRP), Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide TS for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the TS. The P/T limits are among the limiting conditions of operation in the TS for all commercial nuclear plants in the U.S. Appendices G and H to 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G to 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME

Code and the testing requirements for the beltline materials in the surveillance capsules be tested in accordance with Appendix H to 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees use the methods in RG 1.99, Revision 2, to predict the effect of neutron irradiation on reactor vessel materials. This RG defines the ART as the sum of the unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H to 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

## 2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Harris reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Revision 2. The staff has determined that the material with the highest ART at 3 EFPY was the intermediate shell plate (B4197-2) with 0.10% copper (Cu), 0.50% nickel (Ni), and an initial $RT_{NDT}$ of 86°F.

The licensee has not removed any surveillance capsules from Harris. According to the licensee's Final Safety Analysis Report, the first surveillance capsule will be removed after 3 EFPY. All surveillance capsules contained Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, plate B4197-2, the staff calculated the ART to be 165°F at 1/4T (T = reactor vessel beltling thickness) for 3 EFPY. The staff used a neutron fluence of 0.33E19 n/cm$^2$ at 1/4T. The ART was determined using Section 1 of RG 1.99, Revision 2, because no surveillance capsules have been withdrawn from the reactor vessel.

The licensee used the method in RG 1.99, Revision 2, to calculate an ART of 167°F at 3 EFPY at 1/4T for the same limiting plate material. Substituting the ART of 167°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G to 10 CFR Part 50.

In addition to beltline materials, Appendix G to 10 CFR Part 50 also imposed P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% of the preservice system hydrostatic test pressure, the temperature of the closure flange regions highly stressed by the bolt preload must exceed the 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 flange reference temperature of 0°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. Based on data from the licensee's submittal, the lowest measured Charpy USE is 74 ft-lb for the intermediate shell plate metal B4197-2. Using the method in RG 1.99, Revision 2, the predicted Charpy USE of the plate material at the end of life (5.7E19 $n/cm^2$) will be above 50 ft-lb and, therefore, is acceptable.

The staff agrees that the proposed P/T limits for the reactor coolant system for heatup, cooldown and leak test are valid through 3 EFPY because the limits conform to the requirements of Appendices G and H to 10 CFR Part 50. The licensee's submittal also satisfies Generic Letter 88-11 because the licensee used the method in RG 1.99, Revision 2, to calculate the ART. Hence, the proposed changes to the TS P/T limits for the reactor are acceptable.

As predicted in Generic Letter 88-11, the new curves shift down and to the right, i.e., to lower pressures and higher temperatures, respectively. The new 3 EFPY curves impose more restrictive limits on plant operations than do the existing 4 EFPY curves developed from Revision 1 of the Regulatory Guide. The primary cause of the more restrictive operating curves is the new weighting factor in RG 1.99, Revision 2, assigned to nickel. The more restrictive curves have been offset, in part, by determining with greater accuracy the initial $RT_{NDT}$ for the limiting reactor vessel material. This was accomplished by applying the method described in ASME Boiler and Pressure Vessel (B&PV) Code, Section III NB-2331(a)(4) for calculating $RT_{NDT}$. Recalculation accounts for a 4°F reduction in the initial reference temperature.

Due to the more restrictive pressure-temperature curves, the Low Temperature Over-pressure (LTOP) setpoints and the heatup/cooldown rates are also revised. Revised LTOP setpoints and the heatup/cooldown rates were chosen to: (1) ensure that given a limiting mass or heat input to the RCS during normal operations, including anticipated occurrences and system hydrostatic testing, the Appendix G pressure-temperature curves are not challenged, and (2) ensure that operational flexibility is maintained. In order to accomplish this, both the LTOP low and high power-operated relief valve (PORV) sliding scale setpoints between 100°F and 120°F were lowered and, in general, heating up below 200°F and cooling down below 140°F were slowed, i.e., the rates were reduced. The staff finds the new LTOP setpoints and heatup/cooldown rates are acceptable because they have been determined consistent with RG 1.99, Revision 2, methodology.

In addition, the LTOP enable/disable temperature is lowered from 335°F to 325°F. This provides a 25°F buffer between the Mode 3, Hot Shutdown and Mode 4, Hot Standby break at 350°F and the LTOP enable/disable setpoint. The lowered arming setpoint is well within the guidance for automatic overpressure protection at low temperatures provided in the Regulatory Analysis developed for RG 1.99, Revision 2. The Regulatory Analysis states: "The low temperature overpressure protection system should be operable during startup and shutdown conditions below the enable temperature, defined as the water temperature corresponding to a metal temperature of at least $RT_{NDT}$ + 90°F at the beltline location (1/4t or 3/4t) that is controlling in the Appendix G limit calculations." For Harris, the LTOP enable temperature is conservatively calculated to be 296°F. The LTOP enable temperature is acceptable because it is calculated using the RG 1.99, Revision 2, methodology.

Technical Specification 3.4.9.2, "REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS," and associated Table 4.4-6, "Maximum Cooldown and Heatup Rates For Modes 4, 5, and 6," provide guidance for acceptable heatup and cooldown rates based on the lowest RCS cold leg temperature. The specification ensures the plant is in compliance with Appendix G requirements, which protect the reactor vessel from operational occurrences that could cause brittle fracture. It is the temperature of the reactor vessel metal which is of concern, and RCS temperatures are used as an estimate of the metal temperature.

When no reactor coolant pumps are operating, the wide range temperature instruments are not an accurate indication of the metal temperature. The temperature of the water leaving the RHR Heat Exchanger, which flows to the RCS cold legs and into the vessel, would be more accurate in determining this temperature. Therefore, in order to provide a more accurate RCS temperature while an RHR loop is in operation, the footnote to Table 4.4-6 is being amended to use the RHR Heat Exchanger outlet temperature when no reactor coolant pumps are running. The staff agrees that the RHR heat exchanger outlet temperature is more representative of the system temperature when no reactor coolant pumps are running and, therefore, finds this change acceptable.

Technical Specifications 3.4.9.1 and 3.4.9.2, "REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS," and Figure 3.4-3, "Reactor Coolant System Heatup Limitations - Applicable to 4 EFPY," provide criticality limits for the RCS at various heatup rates and hydrostatic test conditions. These criticality limits are similar to the vessel pressure-temperature limits in that they separate the region of normal operation from that where brittle fracture is a potential concern; the only difference being the mechanism deals with temperature/neutronics versus temperature/pressure. However, these limitations serve no operating purpose since Technical Specification 3.1.1.4 requires the RCS to be at a minimum of 551°F prior to achieving criticality. Technical Specification 3.10.3 provides an exception to that requirement but only allows a 10°F reduction to 541°F. Since the criticality limits of Specifications 3.4.9.1 and 3.4.9.2 are bounded by Specification 3.1.1.4 and do not provide any other operational purpose the staff finds this change is considered administrative in nature. It is acceptable to remove the criticality limits imposed by TS 3.4.9.1 and 3.4.9.2.

The Action Statements of Specifications 3.4.9.1 and 3.4.9.2, "REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS," are changed to provide clear direction of when an engineering evaluation is needed. These specifications provide RCS pressure-temperature limits, maximum operating heatup and cooldown rates and a maximum temperature rate of change during hydrostatic tests of the RCS. The existing Action statement specifies that if any of the limits are exceeded, restore the desired RCS conditions and perform an engineering evaluation to determine the effects of the out-of-limit condition. The Appendix G pressure-temperature limits were developed to protect the reactor pressure vessel from brittle fracture by clearly separating the region of normal operations, including operational transients, from the region where the vessel is subject to brittle fracture. The heatup and cooldown rates and LTOP setpoints are designed to ensure that the Appendix G RCS pressure-temperature limits are not challenged. Exceeding the heatup or cooldown rates by themselves will not result in exceeding the Appendix G curves. Therefore, an engineering evaluation to determine continued operability of the reactor vessel is not necessary. The revised Action statement takes this into account by requiring an engineering evaluation only if the Appendix G pressure-temperature limits are exceeded. The staff agrees that the current Action statement is overly restrictive in that it requires an engineering evaluation anytime a heatup or cooldown rate is exceeded and, therefore, the proposed change is acceptable.

Technical Specification 3.1.2.3, "REACTOR COOLANT SYSTEMS/CHARGING PUMP - SHUTDOWN," Surveillance Requirement 4.1.2.3.2 concerns the verification of all but one charging/safety injection pump as inoperable while in Modes 4, 5 and 6 and while the temperature in one or more of the Reactor Coolant System (RCS) Cold legs is less than the LTOP enable temperature setpoint. This surveillance has been modified to appropriately reference one breaker per pump, include all relevant requirements and provide a more concise description. The staff agrees that this administrative change will avoid the possibility of operator confusion with regard to the applicability and conditions of this surveillance requirement and, therefore, is acceptable.

Technical Specification 3.5.3, "ECCS SUBSYSTEMS - $T_{avg}$ LESS THAN 350°F," requires one charging/safety injection pump, one RHR pump and heat exchanger, and an injection flow path capable of taking suction from either of two specified sources to be operable while in Mode 4 - Hot Shutdown. Surveillance Requirement 4.5.3.2 requires that the remaining charging/safety injection pumps are verified inoperable when the temperature in one or more of the RCS cold legs is below the LTOP enable temperature. This surveillance requirement is redundant to and bounded by 4.1.2.3.2. Technical Specification 3.1.2.3 "CHARGING PUMP - SHUTDOWN" requires one charging/safety injection pump operable in Mode 4 - Hot Shutdown, Mode 5 - Cold Shutdown and Mode 6 - Refueling. Associated Surveillance Requirement 4.1.2.3.2 requires all other charging/safety injection pumps to be verified as inoperable whenever RCS temperature is below the LTOP enable setpoint in Modes 4, 5 or 6. The Surveillance 4.1.2.3.2 bounds Surveillance Requirement 4.5.3.2 and 4.5.3.2 serves no other purpose. Therefore, deleting Surveillance Requirement 4.5.3.2 is acceptable.

## 3.0 ENVIRONMENTAL CONSIDERATION

This amendment changes requirements with respect to installation or use of a facility component located within the restricted areas as defined in 10 CFR Part 20 and changes the surveillance requirements. The staff has determined that the amendment involves 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 this amendment involves no significant hazards consideration, and there has been no public comment on such finding. Accordingly, this 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 this amendment.

## 4.0 CONCLUSION

The Commission made a proposed determination that this amendment involves no significant hazards consideration which was published in the Federal Register (54 FR 40924) on October 4, 1989, and consulted with the State of North Carolina. No public comments or requests for hearing were received, and the State of North Carolina did not have any comments.

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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