Docket No. 50-400 May 31, 1990

Mr. Lynn W. Eury
Executive Vice President
Power Supply
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Eury:

SUBJECT: ISSUANCE OF AMENDMENT NO. 19 TO FACILITY OPERATING LICENSE NO. NPF-63 - SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1, REGARDING PRESSURE TEMPERATURE LIMITS RELATING TO GENERIC LETTER 88-11 (TAC NO. 71500)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 19 to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1. This amendment consists of changes to the Technical Specifications (TS) in response to your request dated June 30, 1989, as supplemented November 27, 1989, February 1, 1990 and April 20, 1990.

This amendment incorporates changes to the methodology for predicting reactor vessel material embrittlement for the Shearon Harris Nuclear Power Plant, Unit 1, (Harris) TS. The revisions affect the pressure-temperature limitations on the Reactor Coolant System (RCS), the heatup and cooldown rates for the RCS, and the associated Low Temperature Overpressure Protection System (LTOP) set points. In addition, the new methodology requires related changes in: (1) recalculated limiting material reference temperature (RT sub-NDT), (2) modified LTOP enable temperature, (3) the selection of instrumentation for monitoring RCS average temperature and (4) revisions to the TS Bases. Some administrative changes are made such as: (1) rewording to clarify certain specifications, (2) deleting redundant surveillances, and (3) removing the reference to criticality limits in TS 3.4.9.1, 3.4.9.2 and Figure 3.4-3.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's regular Bi-weekly Federal Register notice.

Sincerely,
Orignal signed by:
Stephen T. Hoffman
Richard A. Becker, Project Manager
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 19 to NPF-63
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	:LA:PD21:DRPR	PM:PD21:DRPR	D:PD21:DRPR			
NAME	:PAnderson	:RBecker:dt	:EAdensam			
DATE	:5/10/90	:5/10/90	:5/15/90			

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AMENDMENT NO. 19 TO FACILITY OPERATING LICENSE NO. NPF-63 - HARRIS, UNIT 1

Docket File
NRC PDR
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cc: Licensee Service List

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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 19
License No. NPF-63

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

   A. The application for amendment by Carolina Power & Light Company, (the licensee), dated June 30, 1989, as supplemented November 27, 1989, February 1, 1990, and April 20, 1990, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations 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;

   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 amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-63 is hereby amended to read as follows:

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid tank via either a boric acid transfer pump or a gravity feed connection and a charging/safety injection pump to the Reactor Coolant System if the boric acid tank in Specification 3.1.2.5a. or 3.1.2.6a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging/safety injection pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.5b. or 3.1.2.6b. is OPERABLE.

APPLICABILITY: MODES 4*, 5*, and 6*.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the flow path between the boric acid tank and the charging/safety injection pump suction header is greater than or equal to 65°F when a flow path from the boric acid tank is used, and

b. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

*A maximum of one charging/safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 325°F and the reactor vessel head is in place.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 One charging/safety injection pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency power source.

APPLICABILITY: MODES 4*, 5*#, and 6*#.

ACTION:

With no charging/safety injection pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The above required charging/safety injection pump shall be demonstrated OPERABLE by verifying, on recirculation flow or in service supplying flow to the reactor coolant system and reactor coolant pump seals, that a differential pressure across the pump of greater than or equal to 2446 psid is developed when tested pursuant to Specification 4.0.5.

4.1.2.3.2 All charging/safety injection pumps, excluding the above required OPERABLE pump, shall be demonstrated inoperable** by verifying that each pump's motor circuit breaker is secured in the open position prior to the temperature of one or more of the RCS cold legs decreasing below 325°F and at least once per 31 days thereafter, except when the reactor vessel head is removed.

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*A maximum of one charging/safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 325°F and the reactor vessel head is in place.

**An inoperable pump may be energized for testing provided the discharge of the pump has been isolated from the RCS by a closed isolation valve with power removed from the valve operator or by a manual isolation valve secured in the closed position.

#For periods of no more than 1 hour, when swapping pumps, it is permitted that there be no OPERABLE charging/safety injection pump. No CORE ALTERATIONS or positive reactivity changes are permitted during this time.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

a. Reactor Coolant Loop A and its associated steam generator and reactor coolant pump,**

b. Reactor Coolant Loop B and its associated steam generator and reactor coolant pump,**

c. Reactor Coolant Loop C and its associated steam generator and reactor coolant pump,**

d. RHR Loop A, or

e. RHR Loop B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is an RHR loop, be in COLD SHUTDOWN within 24 hours.

b. With no loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

*All reactor coolant pumps and RHR pumps may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 325°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE**, or

b. The narrow range secondary side water level of at least two steam generators shall be greater than 10%.

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The narrow range secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

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*The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

***A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 325°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4*

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one PORV inoperable as a result of causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve.

c. With two PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With all three PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close their associated block valve(s) and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

e. With one or more block valve(s) inoperable, within 1 hour: (1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and (2) apply the ACTION b., c. or d. above, as appropriate, for the isolated PORV(s).

f. The provisions of Specification 3.0.4 are not applicable.

* MODE 4 when the temperature of all RCS cold legs is greater than 325°F.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; if the pressure and temperature limit lines shown on Figures 3.4-2 and 3.4-3 were exceeded, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.2 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate as shown on Table 4.4-6.

b. A maximum cooldown rate as shown on Table 4.4-6.

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 4, 5, and 6 with reactor vessel head on.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; if the pressure and temperature limit lines shown on Figures 3.4-2 and 3.4-3 were exceeded, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or maintain the RCS T_{avg} and pressure at less than 200°F and 500 psig, respectively.

SURVEILLANCE REQUIREMENTS

4.4.9.2.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.2.2 Deleted from Technical Specifications. Refer to Plant Procedure PLP-106, Technical Specification Equipment List Program.

TABLE 4.4-6

MAXIMUM COOLDOWN AND HEATUP RATES
FOR MODES 4, 5, AND 6 (WITH REACTOR VESSEL HEAD ON)

COOLDOWN RATES

TEMPERATURE*	COOLDOWN IN ANY 1 HOUR PERIOD**
350-200°F	50°F
200-140°F	20°F
140-110°F	10°F
< 110°F	5°F

HEATUP RATES

TEMPERATURE*	HEATUP IN ANY 1 HOUR PERIOD**
< 135°F	10°F
135-155°F	20°F
155-200°F	30°F
200-350°F	50°F

*Temperature range used should be based on the lowest RCS cold leg value.

**Temperature used should be based on lowest RCS cold leg value except when no RCP is in operation; then use an operating RHR heat exchanger outlet temperature.

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.4 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with setpoints which do not exceed the limits established in Figure 3.4-4, or

b. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.9 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 325°F, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.9 square inch vent within the next 8 hours.

b. With both PORVs inoperable, depressurize and vent the RCS through at least a 2.9 square inch vent within 8 hours.

c. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.4.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE;

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE charging/safety injection pump,*

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the charging/safety injection pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 24 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or RHR pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*A maximum of one charging/safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 325°F.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

4.5.3.2 (Deleted).

BORATION SYSTEMS (Continued)

condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single boron injection flow path becomes inoperable.

The limitation for a maximum of one charging/safety injection pump (CSIP) to be OPERABLE and the Surveillance Requirement to verify all CSIPs except the required OPERABLE pump to be inoperable below 325°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The boron capability required below 200°F is sufficient to provide the required SHUTDOWN MARGIN as defined by Specification 3/4.1.1.2 after xenon decay and cooldown from 200°F to 140°F. This condition requires either 7100 gallons of 7000 ppm borated water be maintained in the boric acid storage tanks or 106,000 gallons of 2000-2200 ppm borated water be maintained in the RWST.

The gallons given above are the amounts that need to be maintained in the tank in the various circumstances. To get the specified value, each value had added to it an allowance for the unusable volume of water in the tank, allowances for other identified needs, and an allowance for possible instrument error. In addition, for human factors purposes, the percent indicated levels were then raised to either the next whole percent or the next even percent and the gallon figures rounded off. This makes the LCO values conservative to the analyzed values. The specified percent level and gallons differ by less than 0.3%.

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

The BAT minimum temperature of 65°F ensures that boron solubility is maintained for concentrations of at least the 7750 ppm limit. The RWST minimum temperature is consistent with the STS value and is based upon other considerations since solubility is not an issue at the specified concentration levels. The RWST high temperature was selected to be consistent with analytical assumptions for containment heat load.

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the design DNBR value during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e.; by opening the Reactor Trip System breakers. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 325°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 380,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

SPECIFIC ACTIVITY (Continued)

distinction between the radionuclides above and below a half-life of 15 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture occur, since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The Temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, and 10 CFR 50 Appendix G and H. 10 CFR 50, Appendix G also addresses the metal temperature of the closure head flange and vessel flange regions. The minimum metal temperature of the closure flange region should be at least 120°F higher than the limiting RT NDT for these regions when the pressure exceeds 20% (621 psig for Westinghouse plants) of the preservice hydrostatic test pressure. For Shearon Harris Unit 1, the minimum temperature of the closure flange and vessel flange regions is 120°F because the limiting RT NDT is 0°F (see Table B 3/4 4-1). The Shearon Harris Unit 1 cooldown and heatup limitations shown in Figures 3.4-2 and 3.4-3 and Table 4.4-6 are not impacted by the 120°F limit.

1. The reactor coolant temperature and pressure and system cooldown and heatup rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 and Table 4.4-6 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

PRESSURE/TEMPERATURE LIMITS (Continued)

b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 625°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness testing of the ferritic materials in the reactor vessel was performed in accordance with the 1971 Winter Addenda to Section III of the ASME Boiler and Pressure Vessel Code. These properties are then evaluated in accordance with the NRC Standard Review Plan.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 3 effective full power years (EFPY) of service life. The service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, copper content, and nickel content of the material in question, can be predicted using Figure B 3/4.4-1 and the largest value of ΔRT_{NDT} computed by Regulatory Guide 1.99, Revision 2, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials."

FIGURE B 3/4.4-2 DELETED

PRESSURE/TEMPERATURE LIMITS (Continued)

The cooldown and heatup limits of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} as well as adjustments for possible errors in the pressure and temperature sensing instruments.

Values of ΔRT_{NDT} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185, are available. Capsules will be removed and evaluated in accordance with the requirements of ASTM E185-82 and 10 CFR Part 50, Appendix H. The results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The cooldown and heatup curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule exceeds the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various cooldown and heatup rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which cooldown and heatup curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the

PRESSURE/TEMPERATURE LIMITS (Continued)

heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

LOW TEMPERATURE OVERPRESSURE PROTECTION

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.9 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 325°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than 50°F above the RCS cold leg temperatures, or (2) the start of a charging/safety injection pump and its injection into a water-solid RCS.

The maximum allowed PORV setpoint for the Low Temperature Overpressure Protection System (LTOPS) is derived by analysis which models the performance of the LTOPS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those

LOW TEMPERATURE OVERPRESSURE PROTECTION (Continued)

assumed cannot occur, Technical Specifications require lockout of all but one charging/safety injection pump while in MODES 4 (below 325°F), 5, and 6 with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature.

The maximum allowed PORV setpoint for the LTOPS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and in accordance with the schedule in Table 4.4-5.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1977 Edition and Addenda through Summer 1978.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures that the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plant Requirements," November 1980.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met. The value of 66% indicated level ensures that a minimum of 7440 gallons is maintained in the accumulators. The maximum indicated level of 96% ensures that an adequate volume exists for nitrogen pressurization.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 AND 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The limitation for a maximum of one charging/safety injection pump to be OPERABLE and the Surveillance Requirement to verify one charging/safety injection pump OPERABLE below 325°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

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 beltline 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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