February 18, 1994

Docket No. 50-400

Mr. W. R. Robinson
Vice President - Harris Plant
Nuclear Generation
Carolina Power & Light Company
Post Office Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

Dear Mr. Robinson:

SUBJECT: ISSUANCE OF AMENDMENT NO. 43 TO FACILITY OPERATING LICENSE NO. NPF-63 REGARDING RTD BYPASS MANIFOLD SYSTEM FOR SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 (TAC NO. M87299)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1 (SHNPP). This amendment changes the Technical Specifications (TS) in response to your request dated August 27, 1993, as supplemented November 10, 1993, and February 1, 1994.

The amendment revises the SHNPP TS to allow elimination of the existing reactor coolant's resistance temperature detector (RTD) bypass manifold system and the substitution of a new design with RTDs mounted in thermowells that extend directly into the flow stream of the reactor coolant system.

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

Sincerely,

Original Signed by:
Ngoc B. Le, Project Manager
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 43 to NPF-63
2. Safety Evaluation

cc w/enclosures:
See next page

with changes so noted

OFFICE	LA:PD21:DRPE	PM:PD21:DRPE	D:PD21:DRPE	OGC
NAME	PAnderson	NLe	SBajwa SSB	CPW
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AMENDMENT NO. 43 TO FACILITY OPERATING LICENSE NO. NPF-63 - HARRIS, UNIT 1

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

February 18, 1994

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cc w/enclosures:
See next page

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Shearon Harris Nuclear Power Plant
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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. 43
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 August 27, 1993, as supplemented November 10, 1992, and February 1, 1994, 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:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 43 , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Bajwa

S. Singh Bajwa, Acting Director
Project Directorate II-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 18, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 43

FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Remove Pages	Insert Pages
2-7	2-7
2-8	2-8
B 2-4	B 2-4
B 2-5	B 2-5
3/4 3-14a	3/4 3-14a

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left[\frac{1}{1 + \tau_3 S}\right] \leq \Delta T_o \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left[\frac{1}{1 + \tau_6 S}\right] - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3 s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constants utilized in the lag compensator for ΔT, τ_3 = 0 s;
ΔT_o	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.17;
K_2	=	0.0224/°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 20 s, τ_5 = 4 s;

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: (Continued)

T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;
T'	$\leq$	588.8°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.001072/psig;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between -21.6% and +6.0%, f_1 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds -21.6%, the ΔT Trip Setpoint shall be automatically reduced by 2.36% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + 6.0%, the ΔT Trip Setpoint shall be automatically reduced by 1.57% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.1% ΔT span.

Power Range, Neutron Flux (Continued)

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range Positive Rate trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing. Specifically, this trip complements the Power Range Neutron Flux High and Low trips to ensure that the criteria are met for rod ejection from mid-power.

The Power Range Negative Rate trip provides protection for control rod drop accidents. At high power a single or multiple rod drop accident could cause local flux peaking which could cause an unconservative local DNBR to exist. The Power Range Negative Rate trip will prevent this from occurring by tripping the reactor. No credit is taken for operation of the Power Range Negative Rate trip for those control rod drop accidents for which DNBRs will be greater than the design DNBR value.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to transport to and response time of the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for transport to and response time of the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for transport to and response time of the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases."

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power the Low Setpoint trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power the Pressurizer High Water Level trip is automatically blocked by the loss of P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Reactor Coolant Low Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90.5% of nominal full loop flow. Above P-8

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(11) CHANNEL CALIBRATION shall include the RTD response time.

(12) Verify that appropriate signals reach the undervoltage and shunt trip relays, for both the main and bypass breakers, from the manual reactor trip switch.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 43 TO FACILITY OPERATING LICENSE NO. NPF-63

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By letter dated August 27, 1993, as supplemented November 10, 1993, and February 1, 1994, the Carolina Power & Light Company (CP&L or the licensee) submitted a request for changes to the Shearon Harris Nuclear Power Plant, Unit 1, (SHNPP) Technical Specifications (TS). The requested change would revise the SHNPP TS to allow the elimination of the existing reactor coolant's resistance temperature detector (RTD) bypass manifold system and the substitution of a new design with RTDs mounted in thermowells that extend directly into the flow stream of the reactor coolant system (RCS). The November 10, 1993, and February 1, 1994, letters provided clarifying information within the scope of the original application and did not change the initial proposed no significant hazards consideration determination.

The licensee stated in its submittal that the existing RTD bypass manifold system is used to provide the RCS hot leg and cold leg temperature input parameters to the reactor control and protection functions. The new thermowell/RTD temperature measurement system, when installed, will provide the same RCS temperature inputs with the same overall response time as that of the current system, while at the same time providing additional protection to the control system from the effects of a failed RTD.

The licensee provided the following design information regarding the existing RTD bypass manifold and the new thermowell-mounted RTD temperature measurement system:

Existing RTD Bypass Manifold Design

The existing temperature measurement system utilizes a bypass manifold to collect a representative sample of RCS flow for determining RCS hot leg and cold leg temperatures. Each of the three RCS coolant loops has a hot leg and cold leg bypass manifold. Each loop hot leg bypass manifold is fed by three flow scoops that extend into the RCS hot leg flow stream. The flow from the three scoops is then mixed and sent through the bypass manifold where the system coolant temperature is measured by direct immersion RTDs. The three RCS cold leg manifolds are similarly configured, except there is a single tap that feeds each cold leg bypass manifold. The cold leg manifolds are not equipped with flow scoops or multiple taps because cold leg temperatures are more uniform due to the mixing action of the RCS pumps.

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New Thermowell-Mounted RTD Design

The new design will remove the RTD bypass piping and valves from all three RCS loops, weld thermowells (three in each hot leg and one in each cold leg) into the main RCS pipe and weld caps on the RTD bypass return lines. The hot leg thermowells will be mounted in the same scoops that feed the manifold piping, then a flow hole will be cut in each flow scoop, so that the flow from the inlet holes will pass by the tip of all new thermowell-mounted RTDs. One thermowell/RTD installation will be used on each cold leg in the same piping nozzle, where the manifold's return pipe is connected. New fast response RTDs will be placed in the thermowells to provide the same temperature measurement and time response as the previous RTDs did in the bypass manifolds of hot and cold legs. Since the new design will use three RTDs in the hot leg of each RCS loop, an average temperature from the three RTDs will be used to generate the T-hot signal for each loop. If one hot leg fails in a loop, the second element of the dual element RTD may be used if available, or a bias mode (i.e., the channel is to be placed in trip position) will be used to substitute for the failed RTD. The signal inputs from the new thermowell/RTD system will replace the existing hot leg and cold leg control system RTDs and, therefore, the reactor control system will use the same temperature signals with a median signal selector (MSS) circuit and appropriate isolation, currently used by the reactor protection system (RPS).

2.0 EVALUATION

The proposed change to the RTD bypass system will affect the SHNPP TS in two locations:

(1) Limiting Safety System Settings: Specification 2.2, Reactor Trip System Instrumentation Setpoints, Table 2.2-1 and the BASES for that section, and

(2) Instrumentation: Surveillance 4.3.1, Reactor Trip system Instrumentation, Table 4.3-1.

In considering the licensee's proposed change request, the staff reviewed information provided in the CP&L submittals to determine if the new thermowell-mounted RTD system will provide the same RCS flow temperature inputs and time responses to the plant protection functions and the related control functions to ensure that design margin would exist as provided by the existing RTD bypass manifold system. Consequently, the staff evaluated the following related parameters for the new thermowell/RTD system (1) the response time of the new RTD measurement system, (2) the accuracy of the new method for measuring the RCS flow temperature, and (3) the reliability of the failure detection for the new RTD system. The following is the staff's evaluation:

2.1 Response Time of the New RTD Measurement System

As shown in the tabulation below, the over-temperature delta-T response time for the proposed new system has some gains and losses when compared to the existing RTD bypass system. The licensee stated that the total loop response time for the reactor protection system for the new RTD temperature sensing system will be the same as that of the existing system, i.e., 6.0 seconds. The total response time is the elapsed time from the time the temperature difference in the RCS loops exceeds the trip setpoint until the rods are free to fall.

From the Table below, the actual measured value of the electronics delays for the trip circuit is about 0.5 seconds; and in its revised safety analysis, the licensee assumed 1.25 seconds. This delay time provides an approximate 0.75 second margin that would sufficiently compensate for the possibility of any degradation in the thermowell/RTD system response time during a fuel cycle. To detect any greater degree of this probable degradation, the licensee stated that the RTD response times will be checked as part of the reactor trip system instrumentation (Technical Specification Item 7, Table 3.3-2) during each refueling outage using the loop current step response (LCSR) methodology.

RESPONSE TIME BREAKDOWN FOR RCS TEMPERATURE MEASUREMENT

	Current RTD Bypass	New (Fast) Thermowell RTD
RTD Bypass and loop fluid transport delay and piping thermal capacity.....	2.0	N/A
Hot leg scoop thermal lag and transport delay (sec).................	N/A	0.25
RTD Response Time (sec)...............	2.0	4.5
Electronics Delay (sec)...............	2.0	1.25
Total Response Time (sec)............	6.0	6.0

The staff also reviewed information provided in NUREG-0809, "Safety Evaluation Report - Review of Resistance Temperature Detector Time Response Characteristics," dated August 1981, and NUREG/CR-5560, "Aging of Nuclear Plant Resistance Temperature Detectors," dated June 1990, and finds that the LCSR methodology is a recommended onsite method for checking RTD response times. The NUREG/CR-5560 further notes that the LCSR method provides results that are within 10 percent accuracy.

In order for the total loop response time for the TS surveillance test to be within an acceptable margin, an RTD response time of 4.0 seconds for the new temperature measurement system is required and this is well within margin for the assumed safety analysis value of 4.5 seconds, as shown in the above table. The staff finds that the margin adequacy of the new RTD response time has been addressed in an acceptable manner.

2.2 Accuracy of the New RTD System

The licensee stated that the new method of measuring each hot leg temperature with three thermowell RTDs (one in each scoop) has been evaluated to be at least as accurate as the existing bypass system with three scoops in each hot leg and one RTD measurement. The staff considered the uncertainties associated with the system flow and calibration drift when evaluating the new temperature measurement systems accuracy . The staff has a concern that the new thermowell-mounted RTD that measures the RCS flow temperature at the midpoint of each scoop may have a small streaming error relative to the former scoop flow measurement because of a temperature gradient over the 5-inch scoop span. In response to this concern, the licensee stated that this gradient has been calculated to have a small effect on the over-all loop accuracy. Additionally, the new design uses three RTDs for each hot leg temperature measurement; it is statistically a more accurate temperature measurement than the former method that used only one RTD on for each hot leg temperature measurement.

With respect to the uncertainty associated with the RCS flow with the RTD bypass eliminated, the licensee stated that this flow uncertainty (with allowances for temperature streaming) was previously calculated in WCAP-12340 and a reanalysis of this calculation shows that it remains conservative (i.e., bounding) for the new thermowell-mounted RTD system. This flow measurement uncertainty, including the elbow tap, has been found to be plus or minus 2.1 percent for the indicated flow. Additionally, the total flow measurement uncertainty (FMU), including the required feedwater fouling allowance of 0.1 percent, has been found to be 2.2 percent, and this RCS FMU of 2.2 percent will continue to be used for the Harris plant with the RTD bypass elimination. Because this FMU value agrees with the reanalysis value, the staff finds this to be acceptable.

With respect to the uncertainty associated with the calibration drift on the new system, the licensee stated that a Westinghouse-recommended RTD cross-calibration method of calibrating the RTDs at each refueling prior to startup (the spare RTD elements and wide range RTDs are included) will be used; and during this time an evaluation of RTDs for drift will also be conducted, and if the drift of any narrow range RTD exceeds 0.5 °F, it will be replaced. Since about 30 RTD sensors are included in the SHNPP plant procedure, and they are not all from the same manufacturing lot, the licensee stated that it is reasonable to assume that not all RTDs will drift in the same direction at the same rate. The licensee stated that they use low leakage core loading. They have no data on the effects on the hot leg temperature layers. However, they have not experienced any operational problems. There is an allowance for hot leg temperature streaming included in the uncertainty analysis for the RTDs. This also carries over into the uncertainty analysis for the RCS flow as measured by a primary heat balance.

To ascertain that the new method of recording the hot leg temperature is reasonably accurate in comparison to the old way of measuring hot leg temperature, the licensee will compare the delta-T in each loop (normalized to full power) with the delta-T reading taken before the plant was shut down. There are measurement uncertainties with both the bypass manifold system and the thermowell system, and there may be some differences in temperature layers in the hot leg as a result of the fuel changes. Therefore, it is unlikely that the before and after readings will be exactly the same. However, any differences will be evaluated to determine if there is an error in the new thermowell RTD system. Any unexpected deviation or anomaly would be investigated and addressed.

The staff finds that all uncertainties associated with the calibration drift for the new system have been adequately addressed and that the temperature measuring channels for the RCS flow will not make any significant change in the overall system accuracy as a result of the proposed modification.

2.3 RTD Failure Detection

At SHNPP, the RTD input signals for T-ave and Delta-T measurements derived from the RCS RTDs are required for plant control-related systems like rod control and steam dump control systems. These RTD input signals are processed by the Westinghouse 7300 process protection system. The cold leg temperature T-cold and the three hot leg temperatures are processed to produce the average hot leg temperature T-have. The T-have is then combined with T-cold to produce the loop average temperature (T-avg) and the loop difference temperature (Delta-T). Consequently, the reliability of these RTD input signals is important to other plant control systems. The licensee stated that there are alarms for RTD failure detection for Delta-T (±2 °F) and T-have (± 3°F) channels and that the Delta-T indicators are channel checked on a shift basis (every 12 hours) as required by the plant TS.

An RTD is considered to be inoperable if the deviation alarm for that channel is received. Operability can be verified by performing a quality check of the individual RTD signal using the plant computer or by taking its voltage readings at the process instrumentation cabinets. If an RTD is determined inoperable, its associated channel is placed in a tripped condition under plant procedure. Then the spare element of the RTD is connected in place of the failed element, and the channel is tested and returned to service. If both elements in the thermowell/RTD have failed, then a bias option may be used for one of the three hot leg RTDs in each loop. The licensee also has a plant procedure for recording the past history of the RTD readings. Based upon this history, the bias setting can be calculated. If two RTDs have failed in one hot leg, and no spare elements are available, the channel will remain in the tripped condition and the TS action statements will be followed. The bias setting will not be used on any two RTDs in the same hot leg.

As previously stated, control signals for T-ave and Delta-T measurements are required for the control systems, e.g., as rod control and steam dump control. These signals will now be derived from the protection T-avg and Delta-T signals from all three loops by using a MSS circuit with appropriate isolation. The MSS circuit is designed to select the median of the three temperature signals and reject any signal that is faulty. By selecting the median signal, the system can reject a faulty signal and still provide appropriate isolation between protection and control circuit(s). Therefore, a single random failure will not cause a control system action which could result in a condition requiring a protective action. This meets the requirements of Institute of Electrical and Electronics Engineers (IEEE) Standard 279-1971. The MSS circuit that is provided by the Westinghouse Company has been previously evaluated and approved by the staff for applications at other plants. In addition, the licensee stated that the new thermowell/RTD system will comply with the environmental qualification requirements of 10 CFR 50.49, and the new RTDs have been tested by the manufacturer to qualify under IEEE Standards 344-1975 and 323-1983. The thermowells will be welded and tested to conform to the requirements of American Society of Mechanical Engineers (ASME) Code, Section III, Class I, 1974 (1976 Winter Addendum); and ASME Code, Section XI (1983 Summer Addendum) and the new Class 1E qualified wiring will be installed between the RTDs and the process cabinets. The staff finds that the adequacy in detection of RTD failure has been addressed in an acceptable manner and that control signals derived from the new temperature measurement system are reliable as a result of the proposed modification.

Based on the above review, the staff finds the proposed change to allow the removal of the SHNPP RTD bypass system and the replacement with RTDs mounted in thermowells that extend directly into the RCS to be acceptable.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement 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 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 48379). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

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

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