January 9, 1989

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Docket No. 50-261

Mr. E. E. Utley
Senior Executive Vice President
Power Supply and Engineering & Construction
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

SUBJECT: ISSUANCE OF AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. DPR-23 - H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2, REGARDING RESISTANCE TEMPERATURE DETECTORS SYSTEM PARAMETERS (TAC NO. 68988)

The Nuclear Regulatory Commission has issued the enclosed Amendment No.121 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant, Unit No. 2. This amendment consists of changes to the Technical Specifications in response to your request dated July 26, 1988, and supplemented by letters August 26, and November 1, 1988.

The amendment changes certain reactor parameters to reflect the correct reactor coolant loop resistance temperature detector (RTD) system response time and to support the elimination of the RTD bypass system. The amendment also reduces the range of reactor operation over which the allowable Moderator Temperature Coefficient could be positive.

A copy of the related Safety Evaluation is enclosed. The Notice of Issuance has been forwarded to the Federal Register for publication.

Sincerely,

Ronnie H. Lo, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects I/II

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PDR ADOCK 05000261
P PDC

Enclosures:
1. Amendment No.121 to DPR-23
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page
ROB AMEND 68988

CP&L

DF01

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	:D:PD21:DRPR	:	:	:
NAME	:P. Anderson	:R. Lo:jfw	:E. Adensam	:	:	:
DATE	:12/28/88	:12/28/88	:01/09/89	:	:	:

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AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. DPR-23 - ROBINSON, UNIT 2

Docket File
NRC PDR
Local PDR
PDII-1 Reading
S. Varga (14E4)
G. Lainas
E. Adensam
P. Anderson
R. Lo
H. Balukjian (8E23)
OGC
D. Hagan (MNBB 3302)
E. Jordan (MNBB 3302)
B. Grimes (9A2)
T. Meeks (4) (P1-137)
W. Jones (P-130A)
E. Butcher (11F23)
ACRS (10)
GPA/PA
ARM/LFMB

cc: Licensee/Applicant Service List

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

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 121
License No. DPR-23

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

   A. The application for amendment by Carolina Power & Light Company (the licensee), dated July 26, 1988, as supplemented August 26, and November 1, 1988, 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 3.B of Facility Operating License No. DPR-23 is hereby amended to read as follows:

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/S/ for

Elinor G. Adensam, 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: January 9, 1989

OFC	LA:PD21:DRPR	PM:PD21:DRPR	CSB	OGC	D:PD21:DRPR		
NAME	PAnderson	RLo:jfw	S. Newberry	EChan	E. Adensam		
DATE	12/28/88	12/28/88	12/29/88	12/30/88	01/09/89	~~12/ /88~~	

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ATTACHMENT TO LICENSE AMENDMENT NO. 121

FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

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
2.1-3	2.1-3
2.3-2	2.3-2
2.3-4	2.3-4
2.3-5	2.3-5
2.3-6	2.3-6
	2.3-6a
3.1-11	3.1-11
4.1-3	4.1-3
4.1-5	4.1-5

are set to preclude bulk boiling at the vessel exit. An arbitrary upper safety limit of 118% thermal power is shown. This limit is based on the high flux trip including all uncertainties.

Radial power peaking factors consistent with the limit on $F_{\Delta H}$ given in Specification 3.10.2.1 have been employed in the generation of the curves in Figure 2.1-1. An additional heat flux factor of 1.03 has been included to account for fuel manufacturing tolerances and in-reactor densification of the fuel.

The safety limit curves given in Figure 2.1-1 are based on a minimum RCS flow of 97.3×10^6 lbm/hr. These curves would not be applicable in the case of a loss of flow transient. The evaluation of such an event would be based upon the analysis presented in Section 15.3 of the FSAR. The minimum RCS flow is 99.8×10^6 lbm/hr, which is the minimum thermal design flow of 97.3×10^6 lbm/hr with a 2.6% allowance added for instrument uncertainty associated with the precision calorimetric flow measurement.[3]

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure, and thermal power level that would result in a DNB ratio of less than 1.17[2] based on steady state nominal operating power levels less than or equal to 100%, steady state moninal operating Reactor Coolant System average temperatures less than or equal to 575.4°F, and a steady state nominal operating pressure of 2235 psig. Allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature, and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions.

Reference

(1) XN-NF-711(P) Rev. 0, "XNB Addendum for 26 Inch Spacer."
(2) FSAR Section 15.
(3) WCAP-11889, "RTD Bypass Elimination Licensing Report for H. B. Robinson, Unit 2"

where:

ΔT_o = Indicated ΔT at rated thermal power, °F;
T = Average temperature, °F;
P = Pressurizer pressure, psig;
K_1 < 1.1365;
K_2 = 0.01228;
K_3 = 0.00089;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = The function generated by the lead-lag controller for T_{avg} dynamic compensation;

τ_1 & τ_2 = Time constants utilized in the lead-lag controller for T_{avg}, τ_1 = 20 seconds, τ_2 = 3 seconds;

T' = 575.4°F Reference T_{avg} at rated thermal power;

P' = 2235 psig (Nominal RCS Operating Pressure);

S = Laplace transform operator, sec^{-1};

and $f(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant start-up tests such that:

1) For $(q_t - q_b)$ within +12% and -17%, where q_t and q_b are percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ is total core power in percent of rated power (2300 Mwt), $f(\Delta I) = 0$. For every 2.4% below rated power (2300 Mwt) level, permissible positive flux difference range is extended by +1 percent. For every 2.4% below rated power (2300 Mwt) level, the permissible negative flux difference range is extended by -1 percent.

2) For each percent that the magnitude of $(q_t - q_b)$ exceeds +12% in a positive direction, the ΔT trip setpoint shall be automatically reduced by 2.4% of the value of ΔT at rated power (2300 Mwt).

2.3.2 Protective instrumentation settings for reactor trip interlocks shall be as follows:

2.3.2.1 The low pressurizer pressure trip, high pressurizer level trip, and the low reactor coolant flow trip (for two or more loops) may be bypassed below 10% of rated power.

2.3.2.2 The single-loop-loss-of-flow trip may be bypassed below 45% of rated power.

2.3.3 The RCS narrow range temperature sensors response time shall be less than or equal to a 4.0 second lag time constant.

Basis

The power range reactor trip low setpoint provides protection in the power range for a power excursion beginning from lower power. This trip value was used in the safety analysis.(1)

In the power range of operation, the overpower nuclear flux reactor trip protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure-protective circuitry. The prescribed set point, with allowance for errors, is consistent with the trip point assumed in the accident analysis.(2)

The source and intermediate range reactor trips do not appear in the specification, as these settings are not used in the transient and accident analysis (FSAR Section 15). Both trips provide protection during reactor startup. The former is set at about 10^{+5} counts/sec and the latter at a current proportional to approximately 25% of full power.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip is also a backup to the pressurizer code safety valves for overpressure protection, and is therefore set lower than the set pressure for these valves (2485 psig). The low pressurizer pressure reactor trip also trips the reactor in the unlikely event of a loss-of-coolant accident.(3)

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution provided only that (1) the transient is slow with respect to transport to and response time of the temperature detectors (about 4 seconds), and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors,[2] 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 Specification 2.3.1.2.d.

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 118% of design power density as discussed in Section 7.2.2 of the FSAR and includes corrections for axial power distribution, change in density, and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.[2]

The setpoints in the Technical Specifications ensure the combination of power, temperature, and pressure will not exceed the core safety limits as shown in Figure 2.1-1.

The low flow reactor trip protects the core against DNB in the event of a sudden loss of power to one or more reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.[5] The undervoltage and underfrequency reactor trips protect against a decrease in flow caused by low electrical voltage or frequency. The specified setpoints assure a reactor trip signal before the low flow trip point is reached.

The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. Approximately 1150 ft^3 of water corresponds to 92% of span. The specified setpoint allows margin for instrument error[2] and transient level overshoot beyond this trip setting so that the trip function prevents the water level from reaching the safety valves.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified set point assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system.[6]

The specified reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed set point above which these trips are unblocked assures their reliability in the power range where needed.

Above 10% power, an automatic reactor trip will occur if two reactor coolant pumps are lost during operation. Above 45% power, an automatic reactor trip will occur if any pump is lost. This latter trip will prevent the minimum value of the DNB ratio, DNBR, from going below 1.17 during normal operational transients and anticipated transients when only two loops are in operation and the overtemperature ΔT trip setpoint is adjusted to the value specified for three loop operation.

The turbine and steam-feedwater flow mismatch trips do not appear in the specification, as these settings are not used in the transient and accident analysis. (FSAR Section 15)

The RCS temperature measurement response time parameters define the time delay between when the OTΔT reactor trip conditions are reached and when the control rods are released and free to fall and is based on a sensor lag of 4.0 seconds for the narrow range temperature measurement with a 0.75 second electromechanical delay.[7][8][9]

<u>References</u>

(1) FSAR Section 15.4
(2) FSAR Section 15.0
(3) FSAR Section 15.6
(4) Deleted
(5) FSAR Section 15.3
(6) FSAR Section 15.2
(7) FSAR Section 7.2.2.2.2

(8) WCAP-11889, "RTD Bypass Elimination Licensing Report for H. B. Robinson, Unit 2"

(9) ANF-88-094, "H. B. Robinson, Unit 2, Chapter 15, OTΔT Trip Event Analysis for Elimination of RTD Bypass Piping"

3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at any temperature, above which the moderator temperature coefficient is greater than:

a) +5.0 pcm/°F at less than 50% of rated power, or

b) 0 pcm/°F at 50% of rated power and above.

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit shown on Figure 3.1-1a or 3.1-1b (as appropriate per 3.1.2.1).

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is greater than as specified in 3.1.3.1 above, the reactor shall be subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

Basis

During the early part of fuel cycle, the moderator temperature coefficient may be slightly positive at low power levels. The moderator coefficient at low temperatures or powers will be most positive at the beginning of the fuel cycle, when the boron concentration in the coolant is the greatest. At all times, the moderator coefficient is calculated to be negative in the high power operating range, and after a very brief period of power operation, the coefficient will be negative in all circumstances due to the reduced boron concentration as Xenon and fission products build into the core. The requirement that the reactor is not to be made critical when the moderator coefficient is more positive than as specified in 3.1.3.1 above has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase of moderator temperature or decrease of coolant

Calibration

Calibration is performed to ensure the presentation and acquisition of accurate information.

The nuclear flux (linear level) channels daily calibration against a thermal power calculation will account for errors induced by changing rod patterns and core physics parameters.

For RCS narrow range temperature sensors, verification of response time will be a part of calibration. Cross calibration of RCS narrow range temperature sensors will be performed on a refueling interval.

Other channels are subject only to the "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibration. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at intervals of each refueling shutdown.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, the minimum calibration frequencies set forth are considered acceptable.

Testing[1]

Minimum testing frequency is based on evaluation of unsafe failure rate data and reliability analysis. This is based on operating experience at conventional and nuclear plants. An "unsafe failure" is defined as one which negates channel operability and which, due to its nature, is revealed only when the channel is tested or attempts to respond to a bona fide signal. The minimum testing frequency for those instrument channels connected to the safety system is based on an average unsafe failure rate of 2.5×10^{-6} failure/hr per channel.

TABLE 4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Test	Remarks
1.	Nuclear Power Range	S	D (1) M* (3) R* (3)	B/W (2)	(1) Thermal Power calculations during power operations (2) Signal to ΔT; bistable action (permissive, rod stop, trips) (3) Upper and lower chambers for symmetric offset: monthly during power operations. When periods of reactor shutdown extend this interval beyond one month, the calibration shall be performed immediately following return to power.
2.	Nuclear Intermediate Range	S (1)	N.A.	S/U (2)	(1) Once/shift when in service (2) Log level; bistable action (permissive, rod stop, trip)
3.	Nuclear Source Range	S (1)	N.A.	S/U (2)	(1) Once/shift when in service (2) Bistable action (alarm, trip)
4.	Reactor Coolant Temperature	S	R (4)	B/W (1)(2) R (3)	(1) Overtemperature - ΔT (2) Overpower - ΔT (3) Narrow range RTD response time (4) To include narrow range RTD cross calibration
5.	Reactor Coolant Flow	S	R	M	
6.	Pressurizer Water Level	S	R	M	
7.	Pressurizer Pressure	S	R	M	
8.	4 Kv Voltage	N.A	R	M	Reactor Protection circuits only

*By means of the movable in-core detector system

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. DPR-23

RELATING TO TECHNICAL SPECIFICATION FOR RTD BYPASS SYSTEM REMOVAL

CAROLINA POWER & LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

DOCKET NO. 50-261

1.0 INTRODUCTION

By letter dated July 26, 1988 (Ref. 1), Carolina Power & Light Company (the licensee) indicated that the reactor coolant temperature measurement system for the hot and cold legs for the H. B. Robinson Steam Electric Plant, Unit No. 2 (Robinson), will be modified and requested changes to the plant's Technical Specifications. This modification eliminates the Resistance Temperature Detector (RTD) bypass manifold system and replaces it with dual element RTDs located directly in the reactor coolant system hot and cold leg piping. This will improve availability of the RTD, reduce radiation exposure, and reduce maintenance. The licensee also requested a reduction in the moderator temperature coefficient (MTC) when operating at or above 50% of rated power. In letters dated August 26 and November 1, 1988, the licensee provided additional information related to the proposed changes. The November 1, 1988 submittal also included additional restrictions in the TS to account for instrument response uncertainty and to require certain surveillance. This submittal was made at the request of the NRC staff.

2.0 BACKGROUND

2.1 Current Method

The current method of measuring the hot and cold leg reactor coolant temperatures uses the RTD bypass system. This system was designed to address temperature streaming in the hot legs and, by use of shutoff valves, to allow replacement of the direct immersion narrow-range RTDs without draindown of the Reactor Coolant System (RCS). For increased accuracy in measuring the hot leg temperatures, sampling scoops are located in each hot leg at three locations of a cross section, 120 degrees apart. Each scoop has five orifices which sample the hot leg flow temperature. The flow from the scoops is piped to a manifold where a direct immersion RTD measures the average hot leg loop flow downstream of the steam generator. The cold leg temperature is measured in a similar manner with piping to the bypass manifold, except that no scoops are used, as temperature streaming is not a problem due to the mixing action of the RCS pump.

2.2 Proposed Method

The new method proposed for measuring the hot and cold leg temperatures uses narrow-range, dual element, fast response RTDs manufactured by Weed Instruments, Inc. One of each of the RTD dual elements is used while the other is installed as a spare. The RTDs are placed in thermowells to allow replacement without draindown. The thermowells, however, increase the response time.

The three RTDs in the hot legs of loops B and C are to be placed within the existing scoops for the Robinson plant. A hole will be drilled through the end of each scoop so that water will flow through the existing holes in the leading edge of the scoop, past the RTD, and out through the end hole. With the new method, the RTD measures the temperature at one point. This is in contrast to the temperature measurement of the average of the flow from the five sample holes from the hot leg scoops used in the RTD bypass flow method. However, with the new method, each RTD measurement location is at the same radius as the center hole of the scoop. Therefore, the licensee states, it is the equivalent of the average scoop sample even if a linear temperature gradient exists in the pipe. The three RTDs in each hot leg and the single RTD used in each cold leg are used to calculate the reactor coolant loop differential temperature (delta-T) and average temperature (T_{avg}).

The RTDs in the hot legs for loop A will not have the three RTDs mounted in the existing scoops as described above due to structural interference. Instead, these thermowells are to be located approximately 24 inches downstream of the existing scoop in the same circumferential spacing. The thermowells will be in independent bosses with one thermowell at the center and the other two 120 degrees to either side. Although these RTDs are not in scoops, the sensor will be at the same radial location as the other RTDs which are mounted inside the existing scoops. The licensee stated that the response time of the RTD in the hot leg free stream should, theoretically, be slightly faster than those located in the scoops. However, data from another plant indicates that this slight advantage is not reflected in a measured response time with respect to the scoop mounted RTD/thermowells. Since all three RTDs are being moved downstream together, the accuracy of their combined measurement will be the same as if they were all still in the scoops. The flow perturbation caused by the scoop upstream of the new RTD location will have dissipated prior to reaching the RTD. It is expected that this small perturbation will help mix the hot leg fluid and somewhat reduce the magnitude of the temperature streaming.

The design for the measurement of the cold leg temperature has also been modified. A single dual-element thermowell RTD will be mounted in each cold leg with one element as a spare. This is in place of the original method in which the measurement was by an external RTD in the cold leg bypass manifold. If the active cold leg RTD fails, then it is disconnected and the installed spare RTD lead is connected in the electronic cabinet in the failed RTD's place.

The hot and cold leg RTD temperature measurements are entered into the calculations for the reactor coolant system delta-T and T_{avg} temperatures. Because of the variation in temperature in the cross section of the hot legs due to streaming, the three RTD measurement locations in each hot leg are used to get an average value of the variation. An electronic-based system will be used to perform the averaging of the reactor coolant hot leg signals from the three RTDs in each hot leg and transmit the average hot leg temperature signal to protection and control systems. There is a procedure for performing a quality check of the three temperature signals for each hot leg. Capability exists to add a positive (or zero) bias to the averaging calculation, if needed, in order to compensate for the loss of one of the three hot leg RTD sensor inputs. The bias considers the past history of the previous readings. If two or more of the three hot leg dual elements RTDs or both cold leg RTD elements fail in the same protection channel, then that channel, is considered inoperable and placed in trip.

3.0 ANALYSIS

The licensee responded to the NRC staff's questions regarding the response time and uncertainty effects of the new measurement system in a letter dated November 1, 1988 (Ref. 2). The increased response time has the primary impact on the results of the accident analysis. The uncertainty of the hot leg temperature measurement also affects the accident analysis and is the principal contributor in the analysis for calculating the RCS flow measurement uncertainty.

3.1 RTD Response Time

The overall response time of the new thermowell RTD hot leg temperature system as presented in the licensee's July 26, 1988 submittal (Ref. 1) is of the same order as the existing RTD bypass system overall response time of 5.0 seconds. While performing analysis in support of a planned modification to eliminate the RTD bypass system the licensee had found an error in the original RTD response time used in the accident analysis (LER 88-002). The correct response time of 5.0 seconds was greater than that originally assumed (Table 15.0.6-1 of the UFSAR shows a value of 2.30 seconds for the delay time of the overtemperature delta-T trip), necessitating a revision to the accident analysis. The revised analysis provided was for the new thermowell RTD with an increased overall response time of 4.75 seconds. Because of the increased response time, there are longer delays from the time when fluid conditions in the RCS require an overtemperature delta-T (OTDT) or overpower delta-T (OPDT) reactor trip until a trip is actually generated. The licensee does not take credit for the OPDT trip in the accident analysis. The licensee presented information in Reference 3 concerning the FSAR Chapter 15 non-LOCA accidents that rely on the OTDT reactor trip and which were evaluated for the longer response time.

In the submittal of November 1, 1988 (Ref. 2), the licensee proposed to revise the affected Technical Specifications for the new RTD response time. The licensee stated that the 5.0 second overall response time for the existing design consisted of 0.5 seconds for the RTDs and thermowells and 4.5 seconds for RTD bypass transport, thermal lag in piping, filter and electronic delays. The revised safety analysis assumed a 4.75 second overall response time for the new thermowell RTD. This is composed of 4.0 seconds for RTD/thermowell lag time and 0.75 seconds for electronics delay. The licensee stated that this reponse time is achievable with some margin; as the slowest RTD/thermowell response time measured in factory tests ranged from 3.7 to 2.7 seconds.

As noted in NUREG-0809 (Reference 5), extensive RTD testing has revealed RTD time response degradation with aging. In view of this, surveillance tests are needed. The licensee provided a revision to Technical Specification Table 4.1-1 which includes the surveillance test schedule for RTD response time. The required RTD response time is given in Technical Specification Section 2.3.3. The NRC-approved in-situ method for measuring RTD response time is the Loop Current Step Response (LCSR) method, which is the method to be used at Robinson.

3.2 RTD Uncertainty

The new method of measuring each hot leg temperature with three thermowell RTDs manufactured by Weed Instruments, Inc., in place of the RTD bypass system with three scoops, has been analyzed by the licensee. The new method measures at one point for each scoop center hole location, compared to the former method in which there were five sample points in a 5-inch span of the scoop measurements. This may result in a small streaming error relative to the former scoop flow measurement because of a temperature gradient over the 5-inch scoop span. However, this gradient has been calculated to have only a small effect. Also, since possible temperature uncertainties from imbalanced scoop flows are eliminated, the overall result is more accurate and is within the bias shift allowance in the analysis. In addition, since the new method uses three RTDs for each hot leg temperature measurement, it is statistically a more accurate temperature measurement than the former method which used only one RTD for each hot leg temperature measurement. Additional uncertainties are introduced when the signals are processed for averaging before being sent to the processing system. The licensee stated that sufficient allowance has been made in the reactor protection system setpoints. Therefore, the current values of nominal setpoints in the Technical Specifications are still valid.

In the proposed new configuration, T_{hot} is first calculated by averaging three (3) independent RTD measurements. Since the contribution of each RTD is divided by three in this averaging process, a single T_{hot} RTD will have to shift three times the original four degrees (12 degrees F) to trigger an alarm. This T_{hot} signal is then used to calculate T_{avg} in the same manner as before. Even though a single RTD must shift further to trigger an alarm, the controlling function (T_{avg}) has still only shifted 2 degrees F.

The same logic is applied to the delta-T deviation alarm. Again the influence of a single T_{hot} RTD is only one third of the total. Therefore, a single T_{hot} RTD must shift 6 degrees before the 2 degree deviation alarm is triggered. A channel check is performed on a shift basis. The temperatures are recorded on a monthly basis and the records are available to supply a bias to compensate for the loss of one of the hot leg RTD sensor inputs as discussed in Section 2.0. Upon loss of one hot leg RTD, the preferred method for correction is to substitute the spare sensor of each dual element RTD by switching leads in the electronic cabinet.

The licensee states that for re-calibrating the RTDs at each refueling prior to startup, they use the Westinghouse recommended RTD cross-calibration method. Also, there is a small allowance in the OTDT reactor trip analysis to account for RTD delta-T drift. The platinum resistance temperature sensors are believed to be very stable and to have relatively small calibration drifts. However, according to several sources (Refs. 5, 6, 7, 8), RTDs have been known to shift in calibration with time and corrections were necessary. The licensee addressed the ability to check the accuracy of the new thermowell RTD measuring method, with the RTD bypass system removed, as compared to the previous method. The licensee stated that they will perform a comparison of the delta-T temperature indications after the modification with measurements prior to the modification. The NRC will be notified of the results of this comparison.

3.3 Flow Measurement Uncertainty

The thermal design flow (TDF) for the Robinson plant is 97.29×10^6 lb/hr or 258,900 gpm based on an inlet temperature of 550.2° F. From the results of the latest precision calorimetric RCS measurement, the indicated flow is 103.8×10^6 lb/hr or 273,400 gpm. The current flow measurement uncertainty (FMU) is 1.87%. Therefore, the coresponding minimum flow, including flow measurement uncertainty, is 101.9×10^6 lb/hr or 270,250 gpm. These values are well above the TDF. The licensee provided a flow measurement uncertainty analysis in References 2 and 3 that accounts for changes due to the RTD bypass system removal. The licensee stated that the methodology used was the same as that used for the Shearon Harris Unit 1 plant (Ref. 13) and is consistent with NUREG/CR-3659 (Ref. 14). There is a statistical advantage to using three RTDs for the hot leg temperature measurement in the new method rather than the one RTD in the former method. This analysis used the plant-specific instrumentation for the Robinson plant. The results of the analysis indicated that the flow measurement uncertainty value is 2.6% (including the cold leg elbow taps and 0.1% for feedwater venturi fouling).

With the modifications due to the RTD bypass system removal, the new flow measurement uncertainty value of 2.6% will be used in place of the former value of 1.87%. This FMU is used for calculating the RCS flow from the precision calorimetric measurement for comparison against the TDF. The minimum allowable flow (MAF) is obtained from the following equation:

MAF = TDF * (1+FMU)

For the Robinson plant, MAF= 99.8 $x10^6$ lb/hr.

The licensee has committed to have a procedure to require cleaning the feedwater venturi meters at each refueling before the precision calorimetric is made. We have found the flow measurement uncertainty analysis to be acceptable.

3.4 Positive Moderator Temperature Coefficient

The licensee requested a change in the moderator temperature coefficient (MTC) as reflected in the accident analysis. Originally, Technical Specification 3.1.3, Mininmum Conditions for Criticality, stated that "the reactor shall not be made critical at any temperature above which the MTC is greater than + 5.0 pcm/°F at 50% of rated power and linearly decreasing to 0 pcm/°F at rated power." This has been changed to state that "The reactor shall not be made critical at any temperature above which the MTC is greater than 0 pcm/°F at 50% of rated power and above." This is acceptable as it is a change in the conservative direction.

3.5 Non-LOCA Accidents Reanalyzed

The licensee stated that the primary impact of the RTD bypass system elimination is the different response time characteristics of fast response thermowell RTDs. Thus, only those events which rely on the OTDT and OPDT reactor trips are impacted. The Robinson plant does not take credit for the OPDT trip in the accident analysis.

The Chapter 15 accidents in the FSAR were examined by the licensee and the following non-LOCA accidents affected by the longer response time were reanalyzed: (1) Loss of External Electrical Load/Turbine Trip; (2) the Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal at Power; and (3) Control Rod Misoperation. The licensee stated that the approved plant transient computer code PTSPWR2 (Ref. 9) was used for the analysis of these events. Its output is used as input to the approved XCOBRA-IIIC methodology (Ref. 10) to predict the minimum departure from nucleate boiling (MDNBR) for the event. The MDNBRs were calculated with the approved XNB critical heat flux correlation (Ref. 11) for which the critical heat flux limit (CHFL) is 1.17. The analyses were structured to support a Technical Specification F delta-H limit of 1.65.

The initial operating conditions included a power level of 102% of full power or 2346 MWt, an RCS flow rate set at the thermal design flow rate of 97.29 x 10^6 lbm/hr, pressurizer pressure of 2220 psia and core inlet temperature of 550.2°F. Basic assumptions used in the analyses included: (a) precluding the withdrawal function of automatic rod control, (b) non-positive MTC above 50% power for the rod drop transient, and (c) a k1 value in the OTDT trip function of 1.24 including uncertainties (this is for the RTD overall response time value of 4.75 seconds). The discussion of the three reanalyzed accidents follows.

3.5.1 Loss of External Load

This accident is described in Section 15.2.2 of the FSAR and the reference analysis is presented in Reference 12. The licensee's analysis addresses the DNBR part of the Loss of External Load event. The other part relates to a challenge of the vessel pressurization criteria and was not considered. This is because the mitigating features of the pressurizer spray and pressurizer relief valves were assumed to function which results in a conservative evaluation of the MDNBR for this event. The analysis included the effect of the RTD response time on the OTDT trip function. The event initiates with closure of the turbine control valves and after several pressurizer PORV and pressurizer and steamline valve openings, the reactor scram occurs on OTDT, with rod insertion at 15.5 seconds. The DNBR challenge results from core power and primary coolant temperature increase. A plot of DNBR versus time was provided. The minimum DNBR occurred at 16.3 seconds and was computed with a correlated CHFL of 1.19. Since this is above the limit of 1.17, we find that the acceptance criterion has been met.

3.5.2 Uncontrolled Control Rod Assembly at Power

This accident is described in Section 15.4.2 of the FSAR. The licensee's analysis addresses the limiting uncontrolled rod withdrawal transient resulting in reactor trip on the OTDT turbine trip. The reference analysis is presented in Reference 12. The reactivity insertion rate was that resulting in minimum DNBR and in simultaneous OTDT and power range high flux trips. The limiting event was a reactivity insertion ramp of 2 pcm/sec from beginning of cycle (BOC) full power initial conditions with positive reactivity feedback. The pressure increase due to coolant expansion and insurge flow to the pressurizer was limited to a maximum of 2274 psia by the PORVs.

Increasing core power and temperature resulted in a reactor trip on the OTDT reactor trip at 27.2 seconds. The minimum DNBR of 1.19 occurred shortly after at 27.4 seconds. Since the minimum DNBR was greater than the correlated CHFL of 1.17, we find that the acceptance criterion has been met.

3.5.3 Control Rod Misoperation (System Malfunction or Operator Error)

This accident is described in Section 15.4.3 of the FSAR. The licensee's analysis addresses the limiting rod drop transient resulting in a reactor trip on the OTDT reactor trip. The event is initiated by a dropped rod cluster control assembly. This promptly inserts negative reactivity which reduces reactor power and turbine runback begins. The analysis indicated that turbine load reaches its programmed value at 9 seconds. The average coolant temperature first decreased in response to the power reduction and then increased due to the reduced secondary load demand. The temperature increase caused an insurge to the pressurizer, resulting in an opening of the PORVs at 16.1 seconds. In addition, there was a reactor trip due to OTDT at 61.1 seconds. The minimum DNBR occurred at 61.2 seconds with a value of 1.23. Since the minimum DNBR was greater than the correlated CHFL of 1.17, we find that the acceptance criterion has been met.

3.6 LOCA Evaluation

The elimination of the RTD bypass system impacts the uncertainties associated with RCS temperature and flow measurement. The licensee stated in Reference 2 that the magnitude of the uncertainties are such that RCS inlet T in and outlet T out temperatures, thermal design flow rate and the steam generator performance data used in the LOCA analyses will not be affected. Past sensitivity studies concluded that the inlet temperature effect on peak clad temperature is dependent on break size. As a result of these studies, the LOCA analyses are performed at a nominal value of T_{in} without consideration of small uncertainties. The RCS flow rate and steam generator secondary side temperature and pressure are also determined using the loop average temperature (T_{avg}) output. These nominal values used as inputs to the analyses are not affected due to the RTD bypass elimination. It is concluded that the elimination of the RTD bypass piping will not affect the LOCA analyses input and hence, the results of the analyses remain unaffected. Therefore, the plant design changes due to the RTD bypass elimination are acceptable from a LOCA analysis standpoint without requiring reanalysis.

4.0 EVALUATION OF TECHNICAL SPECIFICATIONS

Based on the evaluation stated above, the following addresses each of the specific proposed Technical Specification (TS) changes.

Technical Specification 2.1

A new statement was put in the TS relating to the minimum allowable flow and the flow measurement uncertainty. This states - "The minimum RCS flow is 99.8×10^6 lbm/hr, which is the minimum thermal design flow of 97.3×10^6 lbm/hr with a 2.6% allowance added for instrument uncertainty associated with the precision calorimetric flow measurement." This is acceptable as discussed in Section 3.3 of this Safety Evaluation (SE) . The 2.6% flow measurement uncertainty includes 0.1% for feedwater venturi meter fouling. A reference to the uncertainty calculation has been added to this TS section.

Technical Specification 2.3.1.2

The licensee performed an anlaysis and demonstrated that in the formula to determine the maximum value of OTDT (item d of TS 2.3.1.2), a calculated value for the constant k1 of 1.24 is adequate to conservatively prevent DNB. To allow for the estimated uncertainty, the licensee proposed the k1 value of 1.1365 to be used in the TS. The staff finds this TS k1 value to be acceptable since it incorporates the appropriate uncertainty into the calculated value.

Technical Specification 2.3.3

This is a new TS item that states that "The RCS narrow range temperature sensors response time shall be less than or equal to a 4.0 second lag time constant." This is acceptable as discussed in Section 3.1 of this SE.

Technical Specification Bases 2.3

The reference statement on the RTD bypass piping as it affects transport time has been removed and replaced by a statement on the transport and response time of the RTDs. This is an editorial change to reflect the new design and is acceptable.

In addition, reference to FSAR Section 15.4.2 on the RTD bypass system piping is deleted. Since this piping will be removed, it is no longer pertinent. Also, a new paragraph is inserted which states - "The RCS temperature measurement response time parameters define the time delay between when the OTDT reactor trip conditions are reached and when the control rods are released and free to fall and is based on a sensor lag of 4.0 seconds for the narrow range temperature measurement with a 0.75 second electromechanical delay (Refs. 7, 8, 9)." New References 7, 8, and 9 were also added. These changes reflect the modification and are acceptable.

Technical Specification 3.1.3.1

This change reduces MTC by a specification that states that the MTC shall be 0 pcm/°F at 50% of rated power or above. Formerly, the specification stated that the MTC shall be +5.0 pcm/°F at 50% of rated power and linearly decreased to 0 pcm/°F at rated power. This change is acceptable as discussed in Section 3.4 of this SE.

Technical Specification Bases 4.1

This is a new addition to the Technical Specifications which states - "For RCS narrow range temperature sensors, verification of response time will be a part of calibration. Cross calibration of RCS narrow range temperature sensors will be performed on a refueling interval." This is acceptable as discussed in Section 3.1 of this SE.

Technical Specification Table 4.1-1 -Minimum Frequencies for Checks, Calibrations and Test of Instrument Channels

Item 4 of this table, Reactor Coolant Temperature, has several additions. The narrow range RTD response time will be tested and cross calibrated at least once every 18 months. These changes are acceptable as discussed in Sections 3.1 and 3.2 of this SE.

5.0 SUMMARY

The impact of the RTD bypass elimination for H. B. Robinson, Unit No. 2 on the FSAR Chapter 15 non-LOCA accident analyses has been evaluated and found to be acceptable. For the events impacted by the increase in the channel response time, it has been demonstrated that the conclusions presented in the FSAR remain valid. For the remaining Chapter 15 non-LOCA events, the effect of the increased initial RCS average temperature error allowance has been ascertained by separate evaluations. In all instances, the conclusions presented in the FSAR remain valid under this error allowance assumption and the DNBR limit value is met. The reduction of the MTC when operating at or above 50% rated power was found to be acceptable as it is in the conservative direction. The licensee's analysis to support an RCS flow measurement uncertainty value, which includes the new hot leg RTD temperature accuracy, were found to be acceptable and pertinent Technical Specification changes were proposed.

The Technical Specification changes were found to be acceptable. The licensee has committed to obtain data for comparison of the temperature indications after modification with measurements prior to modification. NRC will be notified of the results of the comparison. The licensee has also committed to implement new procedures to clean the feedwater venturi meters at refueling before the precision calorimetric is made at each refueling.

6.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on December 19, 1988 (53 FR 51021). Accordingly, based on the environmental assessment, the Commission has determined that the issuance of these amendments will have no significant effect on the quality of the human environment.

7.0 CONCLUSION

The Commission has issued a Notice Of Consideration of Issuance of Amendment of Facility Operating License and Opportunity for Hearing which was published in the Federal Register (53 FR 30879) on August 16, 1988. No petition to intervene or request for hearing has been filed on this action.

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 opertation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendments will not be inimical to the common defence and security or to the health and safety of the public.

Prinicpal Contributor: Harry Balukjian

Dated: January 9, 1989

8.0 REFERENCES

1. Letter from L. W. Eury, Carolina Power and Light Company, to USNRC, dated July 26, 1988.

2. Letter from L. W. Eury, Carolina Power & Light Company, to USNRC, dated November 1, 1988.

3. WCAP-11889, RTD Bypass Elinination Report for H. B. Robinson, Unit 1, June 1988 (Proprietary), WCAP-11890 (Non-Proprietary with addendum 1) dated October 1988.

4. ANF-88-094, H. B. Robinson, Unit 2 Chapter 15 Overtemperature Delta-T Trip Event Analysis for Elimination of RTD Bypass Piping, July 1988. November 1, 1988.

5. NUREG-0809, Safety Evaluation Report, Review of Resistance Temperature Detector Time Response Characteristics, August 1981.

6. NUREG/CR-4928, Degradation of Nuclear Plant Temperature Sensors, June 1987.

7. K. R. Carr, An Evaluation of Industrial Platinum Resistance Thermometer Temperature - Its Measurement and Control in Science and Industry, ISA publication, Vol. 4, Part 2, 1972, pages 971-982.

8. B. W. Mangum, The Stability of Small Industrial Platinum Resistance Thermometers, Journal of Research of the NBS, Vol. 89, No. 4, July-August 1984, pages 305-350.

9. XN-NF74-5(A) and Sups. 1-6, Rev. 2, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR)," Exxon Nuclear Company, Richland, WA, October 1986.

10. XN-NF-82-21(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company, Richland, WA, September 1983.

11. XN-NF-621(A), Rev. 1, "Exxon Nuclear DNB Correlation for PWR Fuel Designs," Exxon Nuclear Company, Richland, WA, September 1983.

12. XN-NF-84-74, Rev. 1, "Plant Transient Analysis for H. B. Robinson Unit 2 at 2300 MWt With Increased FN_H" Exxon Nuclear Company, Richland, WA, April 1986.

13. WCAP-11168 Rev. 1, "RCS Flow Uncertainty for Shearon Harris Unit 1," October 1986.

14. NUREG/CR-3659, "A Mathematical Model for Assessing the Uncertainties of Instrumentation Measurements for Power and Flow of PWR Reactors," February 1985.

UNITED STATES NUCLEAR REGULATORY COMMISSION

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No.121 to Facility Operating License No. DPR-23 to the Carolina Power & Light Company (the licensee), which revised the Technical Specifications for operations of the H. B. Robinson Steam Electric Plant, Unit No. 2, located in Darlington County, South Carolina. The amendment is effective as of the date of its issuance.

The amendment changes certain reator parameters to reflect the correct reacto coolant loop resistance temperature detector (RTD) system response time and to support the elimination of the RTD bypass system. The amendment also reduces the range of reactor operation over which the allowable Moderator Temperature Coefficient could be positive.

The application for amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on August 16, 1988 (53 FR 30879).

Also in connection with this action, the Commission prepared an Environmental Assessment and Finding of No Significant Impact, which was published in the FEDERAL REGISTER on December 19, 1988 (53 FR 51021).

For further details with respect to the action, see (1) the application for amendment dated July 26, 1988, as supplemented August 26, and November 1, 1988, (2) Amendment No. 121 to Facility Operating License No. DPR-23, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 2120 L Street, N.W., Washington, D.C., 20555 and at the Hartsville Memorial Library, Nuclear Information Depository, 220 N. Fifth Street, Hartsville, South Carolina 29550.

Dated at Rockville, Maryland this 9th day of January 1989.

FOR THE NUCLEAR REGULATORY COMMISISON

Edward A. Reeves

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